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Nuclear Waste Policy Act
(Section 113)

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Consultation Draft

Site Characterization Plan

Yucca Mountain Site, Nevada Research and Development Area, Nevada

Volume III

January 1988

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
Washington, DC 20585

Prepared Under Contract No. DE-AC08-87NV 10576

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CONCEPTUAL DESIGN OF A REPOSITORY

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Chapter 6

CONCEPTUAL DESIGN OF A REPOSITORY

INTRODUCTION

The purpose of this chapter is to describe the basis for facility design, the completed facility conceptual design, the completed analytical work relating to the resolution of design issues, and a brief description of future design-related work. The basis for design and the conceptual design information presented in this chapter meet the requirements of the Nuclear Waste Policy Act of 1982, Section 113(b)(1)(C)(NWPA, 1983) and 10 CFR 60.17(c), for a conceptual repository design that takes into account site-specific requirements. The description of completed analytical work allows the reader to become familiar with the analytical methods and data used in the design of repository facilities and that form the basis for related site characterization activities. This information is presented to permit a critical evaluation of planned site characterization activities.

The presentation of material in this chapter is grouped to provide insight into several different aspects of the conceptual design activity. Section 6.1 presents a summary overview of the various legislation, regulatory requirements, U.S Department of Energy (DOE) orders, and State requirements that were considered in developing the repository conceptual design. Section 6.2 presents a summary of the repository conceptual design. The purpose of the conceptual design was to establish project feasibility, identify information on site characteristics that would be needed for future design efforts, and to obtain a preliminary cost estimate for facility construction and operation. The summary is brief, and focuses on the site related aspects of the design. The complete conceptual design is contained in the Site Characterization Plan-Conceptual Design Report (SNL, 1987). This document is referred to as the SCP-CDR throughout Chapter 6. The conceptual design is a preliminary step in the overall repository design process that helps to guide the gathering of information for later design phases. Design concepts may be refined and design detail will be provided in the later phases of design. Section 6.4 presents a summary of work completed to date relevant to answering questions about repository facility performance. The section is arranged topically according to licensing-related questions, called issues (Section 8.2 is an introduction to the issues). Plans for future activities to obtain additional information relative to these issues are contained in Section 8.3.2. Section 6.3 provides a cross reference for the reader between the information identified in U.S Nuclear Regulatory Commission (NRC) Regulatory Guide 4.17 (NRC, 1987), and Sections 6.1, 6.4, and 8.3.2 of this document.

The Site Characterization Plan-Conceptual Design (SCP-CD) completes the first of four repository design phases described in the Mission Plan (DOE, 1985a). The phases are as follows:

1. Site Characterization Plan-Conceptual Design.

2. Advanced conceptual design.

3. License application design.

4. Final procurement and construction design.

The results of preliminary site investigations are presented in Chapters 1 through 5 of this document; site data that had the principal effect on facility design are contained in Chapters 1, 2, 3, and 4 (Geology, Geoengineering, Hydrology, and Geochemistry). Some information contained in Chapter 5 (Climatology and Meteorology) was used in the siting of surface facilities. Waste package design is discussed in Chapter 7, and preclosure and postclosure performance assessment plans are presented in Chapter 8. Descriptions of the information that will be obtained during site characterization and the plans for obtaining this information are also presented in Chapter 8.

If the Yucca Mountain site is selected as the site for the first repository, the mined geologic disposal system (MGDS) surface and underground facilities will be constructed on federally owned land on and adjacent to the Nevada Test Site (NTS) in southern Nevada. Yucca Mountain is a north-trending fault-block ridge. Its crest is more than 370 m above the western edge of Jackass Flat (to the east) and 300 m above the eastern edge of Crater Flat (to the west). The location for MGDS surface facilities is on the gently sloping alluvial fan that forms the western edge of Jackass Flats, and the location for the MGDS underground facilities is beneath Yucca Mountain.

The design requirements, which have had the greatest impact on facility design, are as follows:

1. The underground facilities shall contribute to the containment and isolation of radionuclides (10 CFR 60.133(a)(1)).

2. The underground facilities shall be designed to permit retrieval of waste (10 CFR 60.133(c)).

3. All MGDS facilities shall be constructed, operated, decommissioned, and closed using reasonably available technology (10 CFR 960.5-1(a)(3)).

4. The MGDS facilities shall be capable of receiving, preparing, and emplacing waste equivalent to 70,000 metric tons of uranium (MTU) in a period of 25 yr (Table 2-2 in DOE, 1985a).

Consistent with the requirements for emplacing nuclear waste in a mined geologic disposal setting, the underground facility has been designed to limit disruption to the natural environment and thereby contribute to the containment and isolation of the waste. To inhibit subsidence, a low extraction ratio (the ratio of the excavated area to the total underground area) has been maintained. To limit the possible tendency for the extension of fractures from the surface into the underground facilities, the thermally induced loading (loading due to the decay of the radioactive materials in the nuclear waste) has been distributed over a large area. Further, the underground facilities have been located above the water table. Therefore, water that could contact the waste containers and transport waste to the accessible environment has been limited. Also, selected repository operations are planned to limit the quantity of water used (i.e., emplacement borehole drilling) and therefore the potential for change in the waste container environment.

The underground facility design provides for retrieval of waste by providing stable underground openings for a time period sufficient to allow waste emplacement and retrieval operations in accordance with the DOE Mission Plan (DOE, 1985a) and 10 CFR Part 60. Further, design constraints have been imposed such that the temperature rise in the underground facilities will not hamper retrieval of emplaced waste during the time period specified in 10 CFR Part 60. Currently available technology is used in all operational phases of the Yucca Mountain MGDS although some demonstration of the use of the technology in MGDS applications is needed. This use of currently available technology is reflected in the conceptual design summarized in this chapter and presented in detail in the SCP-CDR. The Yucca Mountain MGDS is designed to receive and emplace waste equivalent to 400 MTU for the first three years of operation, 900 MTU in the fourth year, 1,800 MTU in the fifth year, and 3,400 MTU in each succeeding year until the full 70,000 MTU have been emplaced. Under this schedule, it will take just under 25 yr to emplace waste equivalent to 70,000 MTU.

Data obtained during preliminary site investigations indicate that Yucca Mountain and the area of Jackass Flats immediately to the east of Yucca Mountain are suitable for construction and operation of MGDS facilities (Section 2.2.4 in DOE, 1986b). Yucca Mountain consists of a layered sequence of welded, nonwelded, and bedded tuff in which conventional hard-rock mining practice will yield stable underground openings. The waste emplacement horizon, located within the welded ash-flow portion of the Topopah Spring Member of the Paintbrush Tuff and designated as unit TSw2, is in the unsaturated zone, 200 to 400 m above the water table. The surface facilities for waste receiving, unloading, preparation, and storage are founded on alluvial material on the western edge of Jackass Flats.

The location of all MGDS underground facilities in the unsaturated zone will reduce and possibly eliminate the problems associated with control of ground water during underground development. Because ground-water inflow is not expected to be an operational problem, installation of seals in the shafts and ramps is not planned before decommissioning of the MGDS. However, seals will be installed as part of MGDS facility closure operations.

The surface and underground facilities at the Yucca Mountain site are planned so that either shaft or ramp access from the surface to the underground could be used. Four shafts and two ramps are incorporated in the conceptual design. Two of the four shafts are planned to be constructed as part of the exploratory shaft facility (ESF). These two exploratory shafts (ES) would provide underground access and ventilation for the ESF before construction of the MGDS underground facilities and serve as ventilation intake shafts during repository operations. The remaining shafts (the men-and-materials shaft and a ventilation exhaust shaft) and ramps (the surface-to-underground waste transfer ramp and the mined-materials removal ramp) would be constructed as part of the initial development activities for the Yucca Mountain MGDS. A distinctive feature of the Yucca Mountain repository conceptual design is the use of a ramp access for spent-fuel and high-level waste transfer from the surface storage facilities to the underground facilities. The ramp access permits the use of a vehicle to remove the waste container from surface storage, convey the container to the waste emplacement borehole, and emplace the container in the waste emplacement borehole. The use of a ramp and transporter eliminates the need for intermediate handling equipment and the associated facilities and operating personnel.

Site-specific geologic, rock characteristics, and hydrologic information is required for MGDS facility design. However, sufficient latitude exists in the conceptual design of MGDS facilities to accept substantial variations in the site properties without major redesign of either the surface or the underground facilities.

The design process is an ongoing one that requires periodic documentation of the status of the design. For example, the status of the Yucca Mountain MGDS design studies has been reported previously in the environmental assessment (EA) for the Yucca Mountain Site (DOE, 1986b). The current design, documented in this chapter and in the SCP-CDR (SNL, 1987), reflects both maturing design concepts and more recent guidance. Specifically, the current design reflects the use of the DOE Mission Plan (DOE, 1985a) and the Generic Requirements for a Mined Geologic Disposal System (GR) (DOE, 1984b; Appendix D of DOE, 1986d).

To document this design in a timely manner, it was necessary to stop making changes in the guidance and the design in about mid-1986. It is recognized that several changes have occurred since that time and that additional design studies have been initiated (or completed) since that time. These changes will be reflected in future designs for the Yucca Mountain repository. Some of the changes and studies that may impact the design are briefly mentioned below.

Changes in the future designs are expected to occur as a result of more recent guidance presented in the Draft Mission Plan Amendment (DOE, 1987b) and in the revised Generic Requirements (GR) document (DOE, 1986d). The principal impact of these changes is likely to be on the schedule for waste acceptance, the evaluation of the feasibility of using shorter horizontal boreholes, and variations in the second exploratory shaft.

The reference emplacement orientation is vertical emplacement with a single waste container in a vertical borehole. The alternative orientation is horizontal emplacement with as many as 18 waste containers in a horizontal borehole. A study is currently being conducted that will assess the feasibility of variations in the horizontal emplacement concept. This study will compare short horizontal borehole options, with one to three containers in a borehole, to the present long, horizontal borehole concept. Items for comparison include reliability, retrievability, thermomechanical effects, cost, licensability, and other relevant factors.

The exploratory shafts have also been the subject of continuing design studies. These studies address concerns that were identified during public and NRC reviews of the EA as well as more recent reviews. The most significant items include the location of the shaft and related surface facilities, shaft diameter and construction method changes for the second exploratory shaft, investigation of structural features by drifting, and the rearrangement of the exploratory shaft underground facilities.

An issues hierarchy approach to MGDS design and performance activities has been adopted by the DOE. The issues hierarchy identifies the design and performance issues that the DOE feels must be resolved before MGDS license application. For each issue in this hierarchy, an issue resolution strategy (IRS) is developed and implemented. A characterization program has also been identified that will provide the site data needed to support issue resolution. The methodology adopted by the DOE for the development of an IRS for site characterization activities at the Yucca Mountain site is presented in Section 8.1.2 of this document. Issue resolution strategies are presented in Section 8.3 for those issues requiring information that will be obtained during site characterization. Readers should familiarize themselves with the material contained in Sections 8.1, 8.2, and in the following subsections of 8.3 and 8.4 before reading this chapter:

Chapter 8 section	Issue	Subject
8.3.2.2	1.11	Configuration of underground facilities (postclosure)
8.3.2.3	2.7	Repository design criteria for radiological safety
8.3.2.4	4.2	Nonradiological health and safety
8.3.2.5	4.4	Preclosure design and technical feasibility
8.3.3.2	1.12	Seal characteristics
8.3.5.2	2.4	Waste retrievability
8.3.5.3	2.1	Public radiological exposures--normal conditions
8.3.5.4	2.2	Worker radiological safety--normal conditions
8.3.5.5	2.3	Accidental radiological releases
8.4.2		Underground test facility

This chapter is divided into four sections. Section 6.1 contains a summary of the design basis used for the conceptual design of the Yucca Mountain repository facilities. The summary is based on requirements contained in the SCP-CDR, Sections 2.4 and 2.6 and the geologic and hydrologic site properties determined during preliminary site investigations (reported in SCP Chapters 1, 2, and 3). Section 6.2, contains a summary description of the conceptual design of the Yucca Mountain MGDS surface and subsurface facilities. The summary description is based on the SCP-CDR. Section 6.3 provides a relationship among the topics identified in NRC Regulatory Guide 4.17 (NRC, 1987); the facility design requirements and elements in Sections 6.1 and 6.2, respectively. The status of completed work and plans for future work are presented in Section 6.4 and the IRS and identification of data needs are presented in Section 8.3 (Stein, 1986). Section 6.4 is organized around the seven issues that require repository design information for their resolution. For each issue, the status of completed work, the plans for future work, and the design data needs are discussed.

6.1 DESIGN BASIS

6.1.1 REPOSITORY DESIGN REQUIREMENTS

This section briefly discusses the repository functions and associated requirements that are the basis for the development of the repository design. These requirements for the design, construction, licensing, and operation of a repository for the disposal of spent fuel, commercial, and defense high-level waste (DHLW) are derived from legislation and implementing regulations directly addressing radioactive waste disposal (DOE, 1985a). Additional Federal, State, and local regulations, DOE Orders, and DOE guidance are included.

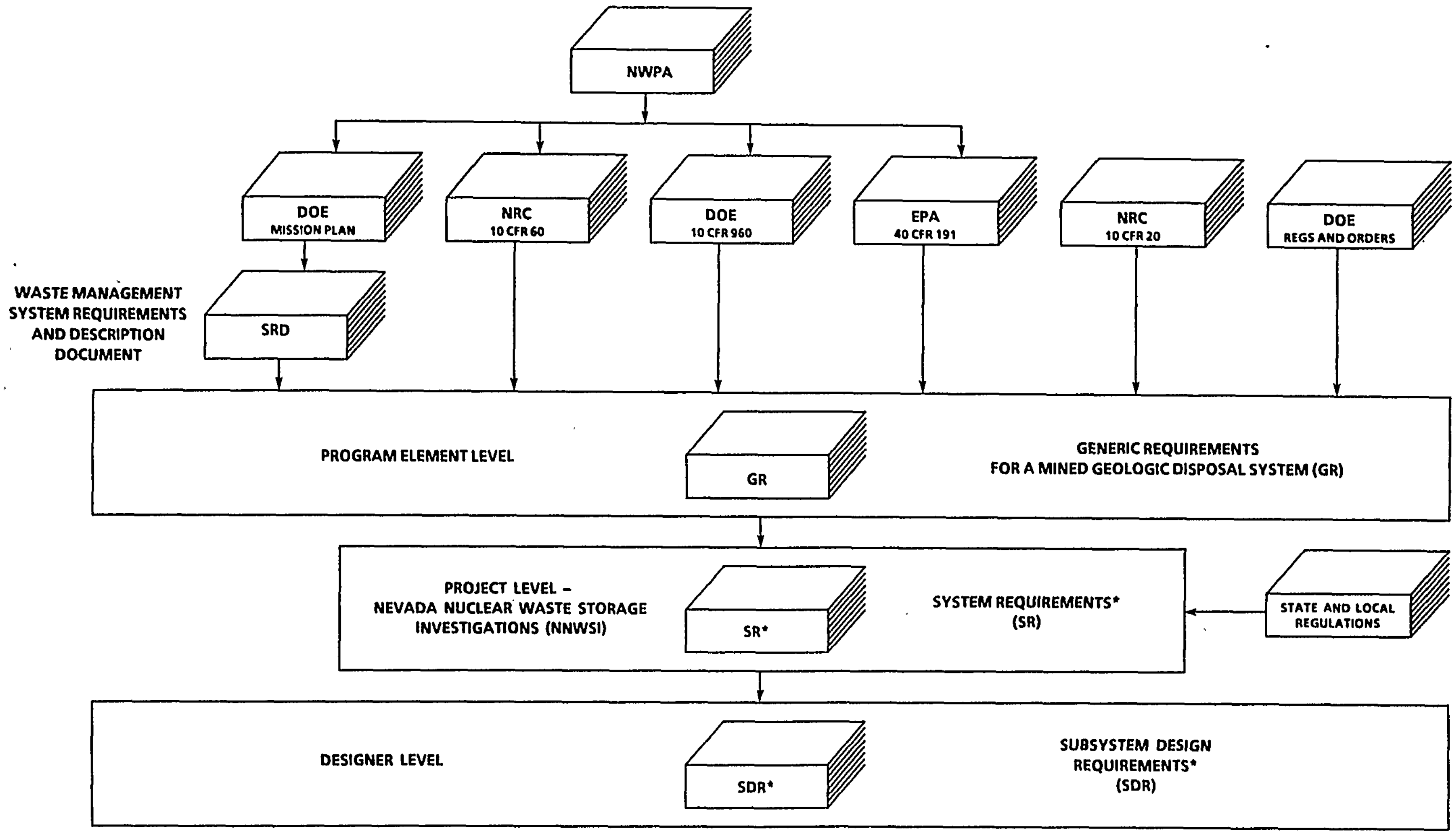
A description of the design requirements for the repository is contained in the Site Characterization Plan-Conceptual Design Report (SNL, 1987). (This report is referred to as the SCP-CDR throughout this section. The conceptual design is referred to as the SCP-CD.) The design criteria contained in the SCP-CDR include definitions of project scope, legal and functional requirements, design parameters, applicable codes, standards, regulations, and other criteria.

Section 6.1.1 summarizes information contained in Sections 2.4 and 2.6 of the SCP-CDR. The development of these design requirements from original source requirements is depicted in Figure 6-1.

The current conceptual design for the Yucca Mountain repository is found in the SCP-CDR. The design addresses the design constraints that have been imposed by site conditions at Yucca Mountain as well as by legal requirements. The site conditions are those that are understood before formal site characterization. The Federal legal constraints are those imposed by the Nuclear Waste Policy Act of 1982 (NWPA, 1983), the NRC, the DOE, and the U.S. Environmental Protection Agency (EPA) regulations.

6.1.1.1 Legal requirements

The legal requirements for the disposal of high-level radioactive waste in geologic repositories begin with the Nuclear Waste Policy Act of 1982



* THESE DOCUMENTS ARE CURRENTLY PUBLISHED AS APPENDICES TO THE SCP-CDR.
[SDR IS CURRENTLY BEING REVISED AND WILL BE TITLED "REPOSITORY DESIGN REQUIREMENTS" (RDR)].

DOE - DEPARTMENT OF ENERGY
NRC - U.S. NUCLEAR REGULATORY COMMISSION
EPA - U.S. ENVIRONMENTAL PROTECTION AGENCY

Figure 6-1. Relationship of subsystem design requirements to primary requirements.

(NWPA, 1983). The purpose of this act is "to provide for the development of repositories for the disposal of high-level radioactive waste and spent nuclear fuel, to establish a program of research, development, and demonstration regarding the disposal of high-level waste and spent nuclear fuel, and for other purposes." The NWPA does the following:

1. Assigns the DOE the responsibility for siting, constructing, and operating repositories.

2. Directs the NRC to develop technical requirements for licensing the disposal of radioactive waste in mined geologic repositories.

3. Directs the EPA to establish environmental standards for the disposal of radioactive waste.

As required by the NWPA, new parts have been added to Titles 10 and 40 of the Code of Federal Regulations (CFR) to specifically address nuclear waste repositories. Applicable CFR parts that were considered during conceptual design activities are summarized in Table 6-1.

Because the repository will be licensed by the NRC to receive and possess high-level radioactive waste, the NRC requirements cited directly or by reference in 10 CFR Part 60 will form the basis for repository design. Appropriate DOE Orders will be used to direct the design in areas not specifically addressed by 10 CFR Part 60 or requirements included in 10 CFR Part 60 by reference.

In addition to the recent CFR parts, for those requirements for which Federal sovereign immunity has not been waived, DOE intends to comply with the applicable State of Nevada regulations to the extent that they are not inconsistent with the DOE responsibilities under NWPA. The regulations considered for the SCP-CD are identified in Table 6-2.

DOE also has specified that the California Administrative Codes for mines be used in the conceptual design process (DOE, 1984). The portions of the codes considered in the SCP-CD are summarized in Table 6-3.

6.1.1.2 Department of Energy functional requirements

In addition to the legal requirements described in Section 6.1.1.1, the DOE has established directives that apply to a repository. These directives consist of (1) guidance and policy issued by the Office of Geologic Repositories for the geologic disposal of radioactive waste and (2) requirements in the form of DOE Orders that apply generally to all DOE projects and are written to establish policy and procedures for DOE activities. These directives are summarized in Table 6-4.

From these DOE directives, the functional requirements and specific guidance for the repository are developed. Functional requirements are primary statements of purpose and definitions of what repository subsystems must accomplish. The basic function of the Yucca Mountain nuclear waste repository is to receive, prepare, and dispose of spent nuclear fuel and

Table 6-1. Parts of the Code of Federal Regulations considered in the conceptual design[a] (page 1 of 7)

Part of CFR	Description	Requirements
10 CFR Part 20, "Standards for Protection Against Radiation"	Prescribes allowable exposure levels for personnel in restricted areas and for members of the public in unrestricted areas due to normal operational releases.	Annual dose limits for • personnel in restricted area: 5 rem whole body or individual organs, 75 rem to hands and feet, and 30 rem to skin. • members of the public in unrestricted areas: 0.5 rem whole body. Allowable concentration limits (CL) for individual radionuclides in air and water are specified.
10 CFR Part 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories"	Subpart B - Licenses: Prescribes rules governing the licensing of the DOE to receive and possess source, special nuclear, and byproduct material at a geologic repository operations area.	For licensing the NRC requires • an application consisting of general information, a safety analysis report, and an environmental report. • in order to authorize construction, there must be reasonable assurance that - the radioactive materials described in the application can be received, possessed, and disposed of without unreasonable risk to the health and safety of the public; - the activities proposed in the application will not be inimical to the common defense and security; and - environmental qualities are protected. • in order to issue a license to receive and possess source, special nuclear, or byproduct material at a geologic repository operations area, - construction of the geologic repository operations area has been substantially completed in conformity

Table 6-1. Parts of the Code of Federal Regulations considered in the conceptual design[a] (page 2 of 7)

Part of CFR	Description	Requirements
10 CFR Part 60 (continued)		with the application as amended, the provisions of the Atomic Energy Act, and rules and regulations of the commission. The activities to be conducted at the geologic repository operations area will be in conformity with the application as amended, the provisions of the Atomic Energy Act, and the Energy Reorganization Act, and the rules and regulations of the Commission; and the issuance of the license will not be inimical to the common defense and security and will not constitute an unreasonable risk to the health and safety of the public. • in a license amendment for permanent closure - the DOE shall submit an application to amend the license before decommissioning and shall update its environmental report.
	<u>Subparts D, E, and F</u>: Include general record keeping and reporting requirements, technical criteria, repository and waste package performance requirements, and performance confirmation requirements.	NRC requires that DOE records, reports, tests, and inspections include • the records and reports required for the licensed activity; • the construction record of the geologic repository operations area; • a written report on each significant deficiency found in the characteristics of the site, and design and construction of the geologic repository operations area; tests the NRC deems appropriate or necessary for the administration of 10 CFR Part 60;

Table 6-1. Parts of the Code of Federal Regulations considered in the conceptual design[a] (page 3 of 7)

Part of CFR	Description	Requirements
10 CFR Part 60 (continued)		• a performance confirmation program in accordance with 10 CFR 60, Subpart F; • NRC inspections of the premises of the geologic repository operations area and adjacent DOE access areas; NRC inspection of DOE activity records; and • provision of office space for NRC inspectors. NRC technical criteria requirements include • the performance objectives for protection against radiation exposures and releases of radioactive material, as well as for retrievability of waste in the geologic repository operations area through permanent closure must be met and assurance that releases of radioactive materials to the accessible environment following permanent closure will conform to EPA standards must be provided; • the land on which the geologic repository operations area and the controlled area are located must be under the jurisdiction and control of DOE or be withdrawn and reserved for this use and be free and clear of all significant encumbrances; • the geologic setting and engineered barriers system must have sufficient favorable conditions present to provide reasonable assurance that the performance objectives relating to waste isolation will be met;

Table 6-1. Parts of the Code of Federal Regulations considered in the conceptual design[a] (page 4 of 7)

Part of CFR	Description	Requirements
10 CFR Part 60 (continued)		• the minimum design criteria specified for the geologic repository operations area and the additional design criteria for the surface facilities, the underground facility, and the design of seals for shafts and boreholes must be met; • the design criteria for the waste package and its components must be met; and • the geologic repository operations area must be designed to permit implementation of a performance confirmation program meeting the requirements of 10 CFR Part 60, Subpart F.
		NRC requires a performance confirmation program that • confirms geotechnical and design parameters and • monitors and tests waste packages.
	<u>Subpart G - Quality Assurance</u>: Identifies quality assurance requirements for the geologic repository and its subsystems or components. (Provisions to ensure compliance with long-term containment requirements, similar to EPA's 40 CFR 191.13, will be added to this regulation.)	NRC requires that the quality assurance program • applies to all systems, structures, and components important to safety; • applies to design and characterization of barriers important to waste isolation and related activities; and • is implemented by DOE based on 10 CFR 50 Appendix B.

Table 6-1. Parts of the Code of Federal Regulations considered in the conceptual design[a] (page 5 of 7)

Part of CFR	Description	Requirements
10 CFR Part 960, "General Guidelines for the Recommendation of Sites for the Nuclear Waste Repositories"	Contains guidelines that specify the factors considered in evaluating and comparing sites on the basis of expected repository performance before and after closure.	DOE preclosure system guidelines include[b] • Any projected radiological exposures of the general public and any projected releases of radioactive materials to restricted and unrestricted areas during repository operation and closure shall meet the applicable safety requirements set forth in 10 CFR Part 20, 10 CFR Part 60, and 40 CFR 191, Subpart A. • during repository siting, construction, operation, and decommissioning, the public and the environment shall be adequately protected from the hazards posed by the disposal of radioactive wastes • repository siting, construction, operation, and closure shall be demonstrated to be technically feasible on the basis of reasonably available technology, and the associated costs shall be demonstrated to be reasonable relative to other available and comparable siting options. DOE postclosure system guideline requires that[b] • the geologic setting at the site shall allow for the physical separation of radioactive waste from the accessible environment after closure in accordance with the requirements of 40 CFR Part 191, Subpart B, as implemented by the provisions of 10 CFR Part 60. The geologic setting at the site will allow for the use of engineered barriers to ensure compliance with the requirements of 40 CFR Part 191 and 10 CFR Part 60.

Table 6-1. Parts of the Code of Federal Regulations considered in the conceptual design[a] (page 6 of 7)

Part of CFR	Description	Requirements
40 CFR Part 191, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level, and Transuranic Radioactive Wastes"	Sets standards for radiation doses and the release of radioactive materials. <u>Subpart A: Management and Storage</u> <u>Subpart B: Disposal</u> Appendix A is Table 1, the allowable release quantities. Appendix B contains guidance for implementation of Subpart B, the environmental standards for disposal. Final Rule issued August 15, 1985. Published September 19, 1985 <u>Federal Register</u> Vol. 50, No. 182, p. 3806ff	EPA requires that • the management and storage of spent nuclear fuel, high-level, or transuranic radioactive wastes shall be conducted in a manner to provide reasonable assurance that the combined annual radiation dose to any member of the public from these operations shall not exceed 25 mrem to the whole body, 75 mrem to the thyroid, or 25 mrem to any other organ (40 CFR 191.03) for 1,000 yr (40 CFR 191.15); • disposal systems for high-level or transuranic wastes shall be designed to provide a reasonable expectation based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 yr after disposal shall (1) have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 of 40 CFR 191, Appendix A and (2) have a likelihood of less than one chance in 1,000 of exceeding 10 times the quantities calculated according to Table 1 of 40 CFR 191, Appendix A. • assurance requirements applicable to non-NRC regulated facilities include reliance on active institutional controls for 100 yr only, post-emplacement site monitoring to verify performance, placement of passive institutional controls, avoidance of areas where subsurface resources are likely to be extracted, and design for waste recovery (40 CFR 191.14[c]); and

Table 6-1. Parts of the Code of Federal Regulations considered in the conceptual design[a] (page 7 of 7)

Part of CFR	Description	Requirements
40 CFR Part 191 (continued)		• special sources of groundwater not to be degraded below EPA's drinking water standards for 1,000 yr after disposal (40 CFR 191.16).

[a] 49 CFR Parts 171-178, covering transport of radioactive material, is considered to cover an offsite activity and is therefore not addressed in the conceptual design. DOE = Department of Energy, NRC = U.S Nuclear Regulatory Commission, EPA = U.S. Environmental Protection Agency.

[b] The statements of the preclosure and postclosure system guidelines are intended to convey that all disqualifying, qualifying, favorable, and potentially adverse conditions in 10 CFR Part 960 were considered in the conceptual design, as appropriate for this stage of design.

[c] Alternative dose standards for waste management and storage activities at facilities not regulated by the NRC or Agreement States do not apply (40 CFR 191.04). Provisions to ensure compliance with the long-term performance requirements, 40 CFR 191.14, apply only to facilities not regulated by the NRC.

Table 6-2. State of Nevada regulations considered in the conceptual design

Regulations	Description
Nevada Revised Statutes (NRS, Title 40, Chapter 444), Public Health and Safety, Sanitation, 1986	Solid waste disposal standards
Nevada Revised Statutes (NRS, Title 40, Chapter 445), Public Health and Safety, Water Controls; Air Pollution, 1986	Water and air quality standards
Nevada Revised Statutes (NRS, Title 40, Chapter 459), Public Health and Safety, Hazardous Materials, 1986	Hazardous and radioactive waste management standards
Nevada Revised Statutes (NRS, Title 45, Chapter 501), Wildlife, Administration and Enforcement, 1986	Wildlife protection standards
Nevada Revised Statutes (NRS, Title 46, Chapter 512), Mines and Minerals, Inspection and Safety of Mines, 1986	Health and safety standards applicable to mining activities
Nevada Revised Statutes (NRS, Title 48, Chapter 533), Water, Adjudication of Vested Water Rights; Appropriation of Public Waters, 1986	Water resources standards
Nevada Revised Statutes (NRS, Title 48, Chapter 533), Water, Underground Water and Wells, 1986	Water resources standards

[a]NRS = Nevada Revised Statutes.

Table 6-3. State of California administrative codes considered in the conceptual design

Regulation	Description
Tunnel Safety Orders, Administrative Code, Title 8, Chapter 4, Subchapter 20, State of California	Tunnel safety standards
Mine Safety Orders, Administrative Code, Title 8, Chapter 4, Subchapter 17, State of California	Mine safety standards including items such as design and selection of conveyors and components[c]

[a]Mandatory compliance to these codes required by DOE Order 5480.4 (DOE, 1984a), Attachment 2. Exemptions to compliance with the State of California Codes must be approved by the DOE Assistant Secretary, Policy, Safety, and Environment.

[b]California Administrative Code (1981a).

[c]California Administrative Code (1981b).

other high-level nuclear wastes. The DOE and functional requirements for the repository are summarized in Table 6-5. The performance confirmation and closure phase functions and specific requirements are summarized in Table 6-6.

6.1.1.3 Mined geologic disposal system for waste

A system requiring licensing by the NRC, which is used for the disposal of high-level radioactive waste in excavated geologic media, is referred to as a mined geologic disposal system (MGDS). The MGDS acts to isolate the disposed radioactive waste from the accessible environment and to ensure the protection of the public health and safety and the quality of the environment.

The environmental standards for the disposal of high-level waste are provided by 40 CFR Part 191. These standards are supplemented by the NRC regulations under which the MGDS is designed, constructed, and operated. The MGDS is composed of the site, the repository, and the waste package (DOE, 1984b). It is recognized that the entire system will act to provide containment and isolation of the waste and that the system will be composed of multiple natural, engineered, and institutional barrier components. These multiple barriers act to reinforce each other to provide containment or isolation capability. The advantage of this approach is that reliance placed on any one barrier will not be so great as to jeopardize the successful functioning of the overall system. In designing each subsystem to perform

Table 6-4. Department of Energy directives considered in the conceptual design (page 1 of 2)

Directive	Description	Requirements
Mission Plan for the Civilian Radioactive Waste Management Program (DOE, 1985a)	Using governing regulations, explicitly defines the mission of the repository and the requirements for its performance. States the broad basis for anticipating the kinds of scientific, engineering, and environmental information required for a repository. Identifies a hierarchy of unresolved questions for which information can be collected to provide answers.	Requires a very thorough subsurface exploration program to ensure that the natural barriers of the site provide waste isolation.
Generic Requirements for a Mined Geologic Disposal System (GR) (DOE, 1984b)	Contains a functional description of the generic structure of a mined geologic disposal system and explicitly prescribes the minimum set of functional requirements and performance criteria to be be satisfied. Organized by systems, with the mined geologic disposal system subdivided into the waste package, repository, and site subsystems during preclosure and engineered, natural, and institutional barrier subsystems after closure.	Reconfirms the Federal regulatory requirements including the requirement that any or all of the emplaced waste must be retrievable in about the same length of time as that devoted to construction and emplacement, starting at any time up to 50 yr after emplacement operations are initiated. Compiles DOE orders applicable to specific subsystems and components.
DOE Order 6410.1, "Management of Construction Projects" (DOE, 1983b)	Establishes policies and procedures to be followed during the planning and execution of DOE construction programs and projects. Includes an outline of the fundamental objectives of conceptual design and describes the content of a conceptual design report.	Establishes a framework to ensure planning, design, and construction of DOE facilities are properly managed.

Table 6-4. Department of Energy directives considered in the conceptual design (page 2 of 2)

Directive	Description	Requirements
DOE Order 6430.1, "General Design Criteria Manual" (DOE, 1983a)	Provides the general criteria for the conceptual design of DOE projects. Includes references to the codes, standards, guides, DOE orders, and other directives that are to be followed.	Provides general design criteria that ensures implementation of DOE policy covering • the basic architectural and engineering disciplines, • certain types of known facility requirements of the DOE, and • specialized requirements based on programmatic and operating experience.
DOE Order 4320.1A, "Site Development and Facility Utilization Planning" (DOE, 1983c)	Establishes policies and procedures for the site development and facility utilization planning of real property at sites owned, leased, or controlled by DOE for production, separation, research, development, or demonstration.	Requires production of a "Site Development and Facility Utilization Plan" to encompass projected programmatic needs, or other activity projections within financial, physical, or other imposed constraints.
DOE Order 5440.1B, "Implementation of the National Environmental Policy Act;" (DOE, 1982) DOE Order 5480.1B, "Environmental Protection, Safety, and Health Protection Program for DOE Operations;" (DOE, 1986b) DOE Order 5480.3, "Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes;" (DOE, 1985b) and DOE Order 5480.4, "Environmental Protection, Safety, and Health Protection Standards" (DOE, 1984a)	Establishes policies, procedures, and the general framework for the environmental protection, safety, and health protection programs.	Specifies and provides requirements for the application of mandatory environmental protection, safety, and health standards.

Table 6-5. Functional requirements of repository facilities (page 1 of 4)

Repository facilities and components	Functional requirements
Site preparation for facilities Clearing Communications Drainage control Explosive distribution system Fencing Grading Landscaping Layout Railroad Roads Utilities (i.e., water, sewage, electrical, fuel)	Provide auxiliary facilities and general services during construction, operation, closure, and decommissioning of the repository (Section 1.2.4 of the Generic Requirements document (GR) (DOE, 1984b)).
Surface facilities Waste-handling facilities Building and structures Handling and packaging equipment Hot cells Heating, ventilation, and air conditioning Support facilities Surface storage Utilities	Receive, prepare, and store radioactive waste (GR 1.2.2). The surface facilities must be able to Accept spent fuel from pressurized water reactors with a nominal burnup of 32.5 GWd/MTU[a] and from boiling water reactors with a nominal burnup of 27.5 GWd/MTU; Accept during Stage 1, beginning in 1998, 400 MTU/yr of spent fuel waste with the possibility that some waste will have been consolidated at reactors; Accept during Stage 2, 900 MTU of waste in 2001; 1,800 MTU in 2002 and 3,400 MTU/yr thereafter, with the possibility of consolidating spent fuel waste at the repository; Receive truck and rail deliveries of incoming waste for up to 80% by rail and up to 70% by truck; Be capable of converting to 100% waste receipt by rail or truck;

Table 6-5. Functional requirements of repository facilities (page 2 of 4)

Repository facilities and components	Functional requirements
Waste-handling facilities (continued)	Accept canisters containing 650 MTU of West Valley high-level waste if Yucca Mountain is the first repository built and operated; Accept up to 15,000 canisters of solidified defense high-level waste at the rate of its generation by defense facilities; and Provide surface storage of 100 MTU of spent fuel during Stage 1 and 750 MTU during Stage 2.
Balance of the plant Administrative offices Air and steam facility Backfill and packing Backup generators Change room Chemical storage facility Control facility Cooling and chilled water facility Explosives storage area Fire stations Fuel storage facility Laboratory facilities Maintenance yard Medical building Potable water system Security stations Sewage systems Tuff pile Visitor's center Warehousing areas	Provide auxiliary facilities and general services during the construction, operation, closure, and decommissioning of the repository.

Table 6-5. Functional requirements of repository facilities (page 3 of 4)

Repository facilities and components	Functional requirements
Waste-handling facilities (continued)	
Exhaust shaft filter building Building and filters Equipment Utilities and support	Provide a ventilation exhaust system that is capable of controlling the discharge of airborne radioactive materials to the environment within the release limits established under 10 CFR 20, Appendix B, Table II, and the dose equivalent limits in 40 CFR 191.03 (GR 1.2.2.5).
Shafts and Ramps Type Emplacement exhaust shaft Exploratory shafts Men-and-materials shaft Tuff ramp	Provide surface openings necessary for transporting radioactive waste from the surface to the emplacement areas and to provide necessary subsurface facilities required for handling excavated tuff, equipment and supplies, and adequate quantities of supply and exhaust air (GR 1.2.1). The requirements are
Components Equipment Excavation Fixtures Hoists and headframes Lining Seals Utilities Ventilation ducting	Establish necessary subsurface openings for disposal operations and Provide subsurface facilities for handling excavated tuff and materials, utilities, ground-water control, and ventilation.
Underground facilities Emplacement operations Waste transporters and emplacement equipment	Move radioactive waste from the surface storage vault to the underground emplacement horizon and to emplace waste underground in prepared locations (GR 1.2.2).

Table 6-5. Functional requirements of repository facilities (page 4 of 4)

Repository facilities and components	Functional requirements
Underground facilities (continued)	
Underground support Contamination control Control room Emergency medical and rescue Fire protection Maintenance Materials and hardware Mine water control Operations support Ventilation	Provide auxiliary facilities and general services during the construction operation, closure, and decommissioning of the repository (GR 1.2.4).
Underground utilities Chilled water Communications Compressed air Electrical Potable water Waste disposal (sanitary and solid)	Provide water, fuel, sanitary and solid waste disposal, electric power, communications, and other utility services to meet the construction, operation, and closure needs of the repository (GR 1.2.5).

[a]GWd/MTU = gigawatt days per metric ton of uranium.

Table 6-6. Performance confirmation and closure phases

Name	Description	Function	Specific requirement
Performance confirmation phase	The geologic repository operations area shall be designed to preserve the option of waste retrieval throughout the period of waste emplacement and until the performance confirmation program has been completed and reviewed by the U.S. Nuclear Regulatory Commission (NRC). Performance confirmation refers to a program of tests, experiments, and analyses conducted to evaluate the accuracy and adequacy of the information used to determine with reasonable assurance that the performance objectives for the period after permanent closure will be met.	Provide data through the performance confirmation program that indicates - actual subsurface conditions during construction and waste emplacement operations are within the limits assumed in the licensing review; and - natural and engineered systems and components required for repository operation, which are designed or assumed to operate as barriers after permanent closure, are functioning as intended and anticipated (10 CFR 60.140(a)).	Preserve the option of waste retrieval throughout the waste emplacement period and until the performance confirmation program has been completed and reviewed by NRC and a license amendment for permanent closure has been issued. Begin performance confirmation during site characterization and continue until permanent closure (10 CFR 60.140).
Closure and decommissioning phase	Begins after successful completion of the performance confirmation program. Consists of those activities required to shut down and permanently prevent access to the nuclear wastes stored at the repository.	Close and seal underground and repository access systems. Decommission and decontaminate surface facilities. Implement the procedures, records assembly, and physical barriers of the institutional barrier system.	

its function, however, care will be taken to ensure that interactions between the subsystems are considered and that no component will unacceptably affect the planned performance of another component.

Additional information on barriers important to waste isolation can be found in Section 6.1.5. The components of the MGDS are discussed in Section 6.1.1.3.1 through 6.1.1.3.4.

6.1.1.3.1 Site

The Yucca Mountain site must provide natural barriers for waste containment and isolation. These barriers must keep radionuclides from reaching man in unacceptable quantities by (1) maintaining the waste in its emplaced location for a given period of time (providing waste containment), (2) limiting radionuclide mobility through the geohydrologic environment to the biosphere (providing isolation), and (3) making human interference difficult, principally by locating the repository deep in a host rock. The functional requirement of natural barriers is to minimize or substantially delay the movement of radionuclides to the accessible environment (Section 2.1, DOE, 1984b). An overburden of at least 200 m is defined as a minimum requirement in 10 CFR 960.4-2-5(d). The selected site must contain a host rock considered suitable for constructing the repository and containing the waste, as well as surrounding rock formations that provide adequate isolation. Desirable geohydrologic features include low ground-water flux and velocity, long radionuclide transport paths to the biosphere, and long-term geologic stability. Design values considered in design activities to date for seismic activity and other natural phenomena (SNL, 1987) are presented in Table 6-7.

6.1.1.3.2 Repository

In designing and operating the repository, there are three overall capabilities that must be considered. The repository must be designed to safely emplace waste, retain the option to retrieve waste, and provide the long-term containment and isolation of the waste. The safety and retrieval aspects of design involve considerations of worker radiological and nonradiological health and safety as well as considerations of excavation stability both from the perspective of worker safety and maintenance of access to emplacement boreholes, should retrieval be initiated. The long-term containment and isolation of the waste involve both using engineered barriers that maintain the capabilities of the system and limiting the negative effects of the engineered portions of the repository on the surrounding site. The limitation of negative or adverse effects of the repository on the surrounding rock mass must be a principal consideration during repository design and construction. The adverse impacts that must be limited include the thermal and radiation effects of the waste on the host rock and hydrology, the effects of excavation on the surrounding rock, and the impacts of penetrations, such as boreholes, on ground-water flow paths. Far-field thermal constraints can be satisfied by distributing spent fuel in such a way that the initial maximum areal power density is 57 kW/acre (Johnstone et al., 1984).

Table 6-7. Design values considered for natural phenomena for design activities to date

Natural phenomena	Requirement	Design value
Earthquake	Analyze risks due to seismic activity.	0.40g horizontal
	Determine the peak acceleration, and use for conceptual repository design.	0.27g vertical
Wind	Determine wind velocities to be used in the design of Class II and Class III elements using the 100-yr-mean recurrence level in accordance with DOE Order 6430.1 (DOE,1983a) and ANSI A58.1 (1982).	80 mph
Tornado	Designs for structures, systems, and components requiring tornado protection shall be designed for tornado loads based on the tornado characteristics specified in ANSI/ANS-2.3 (1983).	180 mph
Flood	Surface facilities shall be protected against the probable maximum flood as defined by ANSI/ANS 2.8 (1981) by channel lining and diversionary structures.	Refer to Sections 6.1.2.5.1 and 6.2.4.2 for additional discussion on flooding.

The repository also must contribute to the assurance of long-term containment and isolation by providing for monitoring of system characteristics that are indicative of repository performance. Data from the system monitoring will be analyzed to evaluate the performance of the repository and to verify compliance with the performance objectives established by the NRC.

6.1.1.3.3 Waste package

The waste package is a system of engineered components that may include waste form, stabilizer, canister, container, and packing material designed to contain nuclear waste for an extended period of time. It contributes to waste retrievability through the required periods and acts as a barrier to waste migration and release into the geologic system (DOE, 1985a). The waste package is described in Chapter 7 and the issue resolution strategies related to the waste package are addressed in Sections 8.3.4, 8.3.5.9, and 8.3.5.10.

6.1.1.4.1 Radiological protection design requirements

As specified in 10 CFR 60.131(a), the geologic repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits specified in 10 CFR Part 20. The design shall include the following:

1. Means to limit concentrations of radioactive material in air.

2. Means to limit the time required to perform work in the vicinity of radioactive materials including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation.

3. Suitable shielding.

4. Means to monitor and control the dispersal of radioactive contamination.

5. Means to control access to high-radiation areas or airborne radioactivity areas.

6. A radiation alarm system to warn of significant increases in radiation levels, of concentrations of radioactive materials in air, and of increased radioactivity released in effluents. The alarm system shall be designed with provisions for calibration and for testing its operability (10 CFR 60.131(a)).

6.1.1.4.2 Design classifications

The requirement to provide confinement for radioactive materials necessitates a more rigorous design treatment for some design elements than for others. The design classifications are presented in Sections 6.1.4 and 6.1.5.

6.1.1.4.3 Safety design considerations

Structures, systems, and components are identified as being important to safety if, due to natural phenomena or anticipated environmental conditions, they fail to perform their intended function, and an accident could result that causes a dose greater than 500 mrem to the whole body or any organ of an individual in an unrestricted area (10 CFR 60.2).

As required in 10 CFR 60.131(b), the structures, systems, and components important to safety shall be designed so that

1. Natural phenomena and environmental conditions will not interfere with safety functions.

6.1.1.4 Public safety considerations

To ensure that public safety will be considered during design, specific design requirements for safety have been identified. These requirements are identified as radiological protection, design classifications, and safety design considerations.

A number of stages are involved in the process leading from site characterization through repository construction and operation. At each stage, steps will be taken to protect the quality of the environment and to mitigate any significant environmental impacts. The operations stage is the portion of the process during which offsite radiological safety will be of greatest concern. This stage includes the activities of waste emplacement, possible waste retrieval, and permanent closure. During the operations stage, instantaneous doses must be in compliance with 10 CFR Part 20 and the combined annual dose equivalent to any member of the public may not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any critical organ (40 CFR 191.03). In addition to the annual dose specified in 40 CFR 191.03, there is a dose limit of 100 mrem in any seven consecutive days for an individual continuously present in the unrestricted area (10 CFR 20.105). Following closure, the MGDS shall be capable of isolating the nuclear waste from the accessible environment for 10,000 yr so that the release of radionuclides to the accessible environment will be no greater than the limits specified in 40 CFR 191.13. The accessible environment includes (1) the atmosphere, (2) land surface, (3) surface water, (4) oceans, and (5) all the lithosphere that is outside the controlled area. The controlled area includes both the surface location, identified by passive institutional controls that encompass no more than 100 km^2 and extend horizontally no more than 5 km in any direction from the outer boundary of the original location of the radioactive wastes in the MGDS, and the subsurface underlying such a surface location (40 CFR 191.12).

Following emplacement the annual release rate of any radionuclide shall not exceed the limits established in 10 CFR 60.113(b). As defined in 10 CFR Part 60, favorable pre-waste emplacement ground-water travel times along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment are those that substantially exceed 1,000 yr (10 CFR 60.122). The ultimate releases of radioactivity to the environment over a 1,000-yr period following emplacement must not exceed the release limits for ground-water protection and individual protection requirements identified in 40 CFR Part 191. To accomplish this goal, the repository performance must preclude any individual from receiving more than 25 mrem to the whole body or 75 mrem to any critical organ for 1,000 yr following disposal (40 CFR 191.15). No offsite source of ground water designated as "special" or Class I by the EPA may be contaminated in excess of the Primary Interim Drinking Water Standards (40 CFR Part 141) for 1,000 yr following disposal (40 CFR 191.16).

2. Dynamic effects, such as earthquakes, will not interfere with safety functions.

3. During and after fires or explosions, safety functions will be performed.

4. Control of radioactive wastes and radioactive effluents can be maintained with safe and timely response to emergency conditions.

5. Utility services important to safety can perform under both normal and accident conditions. Redundant systems shall be included to the extent necessary.

6. Inspection testing and maintenance can be performed to ensure that readiness and continued functioning are not impaired.

7. A criticality accident is not credible.

8. Instrumentation and control systems are functional over anticipated ranges for normal and accident conditions.

9. Compliance with mining regulations will be ensured.

10. Shaft conveyances will fail safely upon malfunction.

Additional information can be found in Section 6.1.4 on items important to safety.

6.1.1.5 Site constraints

The repository will be designed and constructed to prevent the failure of safety systems due to the effect of natural phenomena and environmental conditions and will be in accordance with 10 CFR 60.131(b). Design requirements used in the SCP-CD for site constraints known to have significant impacts are summarized in Table 6-7. Site properties to be used for design values are provided in Section 6.1.2. For information on structures, systems, and components important to safety, refer to Section 6.1.4.

6.1.1.6 Operations scheduling

The SCP-CD was based on the waste acceptance schedule, presented in Table 6-8, provided in the DOE Mission Plan (DOE, 1985a). A new acceptance schedule has been presented in the Draft Mission Plan Amendment (DOE, 1987b). This schedule will be used as a basis for the advanced conceptual design (ACD). The new schedule is not anticipated to have any significant impact on the plans for site characterization (Stein, 1987b).

The schedule of operations presented here is based on a combination of the reference schedule and possible schedule durations provided by the Mission Plan (DOE, 1985a), the Generic Requirements document (DOE, 1984b) and

Table 6-8. Waste acceptance schedule[a]

Year	Facilities: Waste-handling building 1	Facilities: Waste-handling building 2	Waste type: Spent fuel	Waste type: High-level waste[b,c]	Total	Cumulative total
1998	400[d]		400		400	400
1999	400		400		400	800
2000	400		400		400	1,200
2001	400	500	900		900	2,100
2002	400	1,400	1,800		1,800	3,900
2003	400	3,000	3,000	400	3,400	7,300
2004	400	3,000	3,000	400	3,400	10,700
2005	400	3,000	3,000	400	3,400	14,100
2006	400	3,000	3,000	400	3,400	17,500
2007	400	3,000	3,000	400	3,400	20,900
2008	400	3,000	3,000	400	3,400	24,300
2009	400	3,000	3,000	400	3,400	27,700
2010	400	3,000	3,000	400	3,400	31,100
2011	400	3,000	3,000	400	3,400	34,500
2012	400	3,000	3,000	400	3,400	37,900
2013	400	3,000	3,000	400	3,400	41,300
2014	400	3,000	3,000	400	3,400	44,700
2015	400	3,000	3,000	400	3,400	48,100
2016	400	3,000	3,000	400	3,400	51,500
2017	400	3,000	3,000	400	3,400	54,900
2018	400	3,000	3,000	400	3,400	58,300
2019	400	3,000	3,000	400	3,400	61,700
2020	400	3,000	3,000	400	3,400	65,100
2021	400	3,000	3,000	400	3,400	68,500
2022	400	1,100	1,100	400	1,500	70,000
Total	10,000	60,000	62,000	8,000		70,000

[a]Mission Plan (DOE, 1985a).

[b]Approximate waste acceptance for high-level waste from atomic energy defense activities and commercial high-level waste from the West Valley Demonstration Project Quantities have been normalized to MTU on a curie-equivalent basis. Direct comparison with spent fuel is not appropriate because defense high-level waste (DHLW) and commercial high-level waste (CHLW) result from the reprocessing of spent fuel. Actual acceptance rates are to be negotiated between Defense Programs and the DOE.

[c]The first repository currently is designed to begin operating in two stages. This example shows the acceptance of DHLW and CHLW in waste-handling building I after waste-handling building 2 reaches its maximum receipt rate in the year 2003.

[d]Units are metric tons of uranium per year.

a document describing retrievability strategy (Flores, 1986). A reconciliation of these schedules is provided in Section 6.2.9 (retrieval). Figure 6-2 illustrates the preliminary schedule of repository operations used for the conceptual design.

6.1.1.6.1 Construction schedule

To receive spent fuel by 1998 as required by the Nuclear Waste Policy Act of 1982, the DOE has elected to construct and operate the repository in two stages (SNL, 1987). The rate of waste receipt is planned to increase until full processing capacity is reached. The construction and testing of the first waste-handling building (WHB-1) and the construction of the emplacement area, Stage 1, is scheduled to be completed in 53 months. Construction of the second waste-handling building (WHB-2), Stage 2, is scheduled to be completed in 90 months.

6.1.1.6.2 Waste handling and disposal schedule

WHB-1 is scheduled to be ready to receive unconsolidated commercial spent fuel in January 1998 and to continue for 3 yr at a rate of 400 MTU/yr (Table 6-8). During the planned transition from Stage 1 to Stage 2, spent fuel would be handled in both WHB-1 and WHB-2.

After the transition period, WHB-1 could be used for preparing defense high-level waste (DHLW) for disposal. West Valley high-level waste (WVHLW) could be prepared for disposal at the same time. The combined total of DHLW and WVHLW that could be handled at WHB-1 is 8,000 MTU.

During this time, WHB-2 is scheduled to reach its full capacity for receiving, consolidating, and packaging spent fuel. The planned disposal rate for waste packaged at both WHB-1 and WHB-2 is 3,400 MTU/yr. The total capacity of the repository for design purposes is 70,000 MTU.

6.1.1.6.3 Caretaker and closure schedule

The caretaker period begins with the emplacement of the last waste package and continues through the NRC review of and concurrence with the performance confirmation program. It is assumed for design purposes that the caretaker period will extend for 25 yr as shown in Figure 6-2. During this period, repository personnel would be reduced to the number necessary for maintenance activities. Permanent closure will occur for design purposes either after the caretaker period or after waste retrieval. Closure operations would require approximately 4 to 10 yr during this period depending on whether the horizontal or vertical emplacement orientation was used.

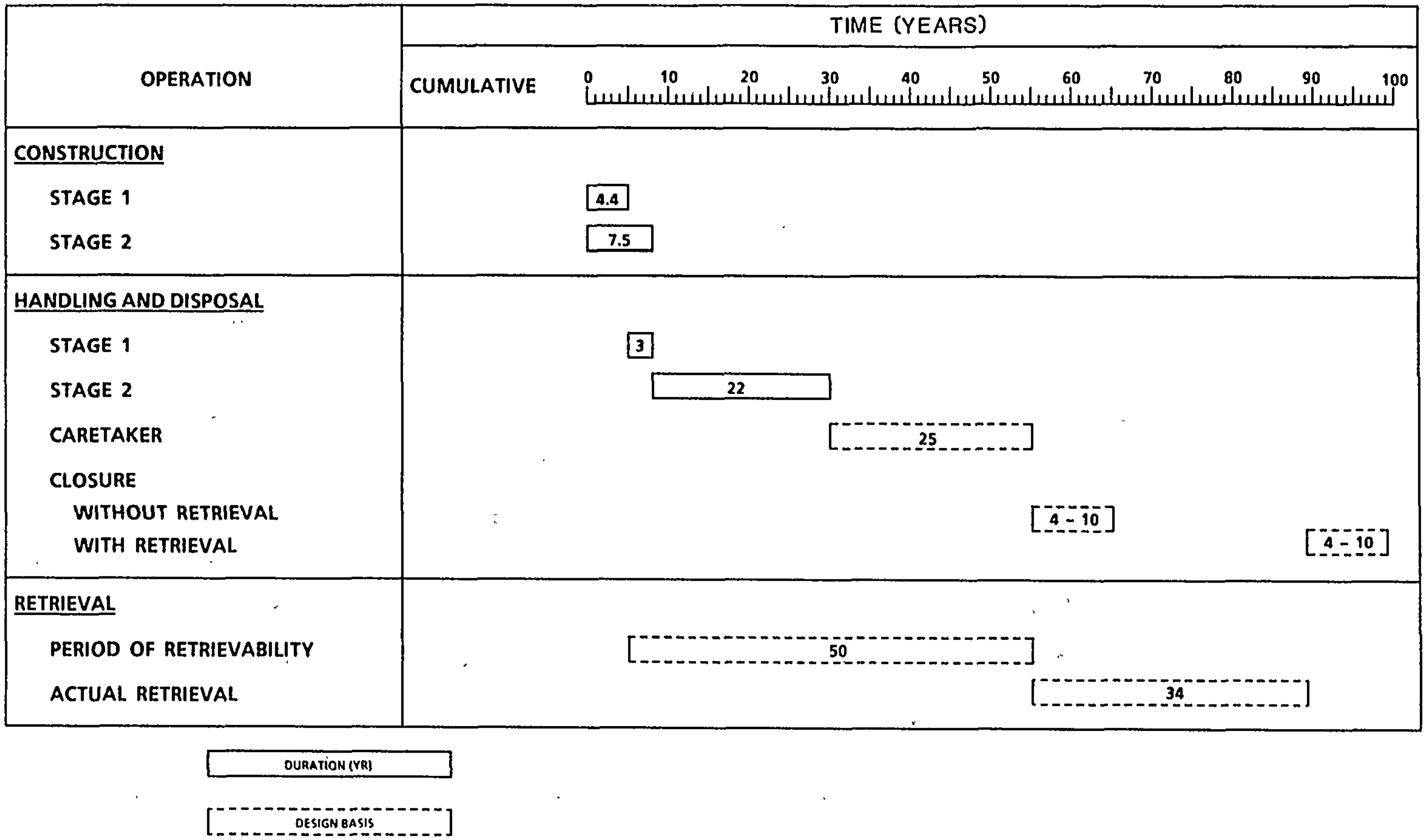


Figure 6-2. Schedule of repository construction and operation.

6.1.1.6.4 Waste retrieval schedule

It is assumed for design purposes that the period of retrievability is a maximum of 50 yr. This period begins when the first waste is emplaced and ends when the performance confirmation program is completed. It is assumed that the time period required to retrieve all waste packages, actual retrieval, is approximately equal to the construction period (6 yr) plus the emplacement period (28 yr), 34 yr, using the schedule from the Generic Requirements document (DOE, 1984b). Additional information concerning the retrieval process and equipment is presented in Section 6.2.9.

6.1.1.7 Retrievability-related design criteria

The conceptual design incorporated many retrievability-related design criteria. To understand the criteria, it is important to briefly synopsize the philosophy used to date relative to retrieval. The current conceptual design reflects a retrieval philosophy that is based on regulatory requirements and is consistent with DOE guidance (DOE, 1984b; Appendix D, DOE, 1986a; DOE, 1985a). This philosophy is summarized in the following:

1. The design of the repository at Yucca Mountain will incorporate the option to retrieve the emplaced waste as a planned contingency operation. Therefore, the equipment and facilities necessary to carry out full repository retrieval need not be constructed at the time of repository construction.

2. The inclusion of the retrieval option will not compromise the safety of the repository, nor will it compromise the ability of the repository to isolate the emplaced waste.

3. The method of retrieval will anticipate off-normal conditions and will be designed to operate under expected off-normal conditions. (In this chapter the term off-normal is used to identify conditions expected to occur infrequently. In future documents the term off-normal will be replaced with the term abnormal.)

4. The design of facilities and equipment for retrieval will be based upon technology that is reasonably available at the time of license application. In addition, the design of retrieval methods and proof-of-principle demonstrations must be completed at the time of license application.

The current list of design criteria was developed using the retrieval philosophy indicated above, the retrieval strategy report (Flores, 1986), and the performance allocation process described for retrieval in Section 8.3.5.2. The current design criteria for retrieval are as follows:

1. The access and drifts will remain usable for at least 84 yr.

2. The borehole liner lifetime will be at least 84 yr.

3. The design basis for the actual retrieval period is 34 yr.

4. The time required for the removal of a waste package will not exceed twice the amount of time required for emplacement of the waste package.

5. For the vertical emplacement concept, the temperature in the access drifts will not exceed 50°C for 50 yr after waste emplacement.

6. For the horizontal emplacement concept, the temperature in the emplacement drifts will not exceed 50°C for 50 yr after waste emplacement.

7. The time required to modify the environment in closed drifts for unprotected workers will not exceed 8 weeks.

8. The worker dose rate during removal operations will not exceed the allowable limit for emplacement operations.

9. For operations areas, all applicable air quality standards will be met.

10. The ability to remove the waste under normal and selected off-normal conditions will be demonstrated.

11. The maximum liner deflection is 5 cm for the vertical emplacement concept and 8 cm for the horizontal concept.

12. For the horizontal emplacement concept, the minimum radius of curvature for the liner is 34 m (110 ft) over the length of a waste package.

13. The ability to perform the retrieval operations using reasonably available technology is required.

This list will be revised as the design and the performance allocation work are refined. In addition, as more definition of off-normal conditions is obtained, additional design criteria will be developed and included in the design basis documents (SNL, 1987; Section 2.4).

6.1.1.8 Waste containment and isolation-related design criteria

Criteria related to waste isolation and containment are important in the design of a repository. The postclosure design criteria identified in 10 CFR 60.133 have been directly considered in establishing four principal functions related to containment and isolation that the postclosure waste disposal system must perform; these functions are as follows:

1. Provide orientation, geometry, layout, and depth of the underground facility such that the facility contributes to containment and isolation taking into account flexibility to accommodate site-specific conditions (10 CFR 60.133(a)(1) and 10 CFR 60.133(b)).

2. Limit water usage and potential chemical effects, thereby contributing to containment and isolation of radionuclides and assisting engineered barriers in meeting performance objectives (10 CFR 60.133(a)(1) and 10 CFR 60.133(h)).

3. Limit potential for excavation-induced changes in rock mass permeability (10 CFR 60.133(f)).

4. Provide thermal loading taking into account performance objectives and thermomechanical response of the host rock (10 CFR 60.133(i), 10 CFR 60.133(e)(2), and 10 CFR 60.133(h)).

To aid in ensuring that these functions can be performed, design criteria have been established for use in the designs prepared to date.

The current design criteria related to waste containment and isolation are as follows:

1. Ensure the usable area for the repository will have greater than 200 m overburden, be within the TSw2 portion of the Topopah Spring Member, be more than 70 m above the water table, and be in the primary area.

2. Design accesses, drifts, and boreholes so that drainage is away from containers.

3. Limit quantity of cement, shotcrete, and grout used in borehole and drift construction.

4. Limit quantity of organics introduced during underground construction.

5. Limit underground water usage during underground development to that required for dust control and proper equipment function; remove all excess water.

6. Limit repository extraction ratio to less than 30 percent for vertical emplacement (<10 percent for horizontal) and limit drift spans to less than 10 m (35 ft).

7. Limit potential for subsidence by backfilling underground openings during decommissioning.

8. Limit impact on surface environment by limiting surface temperature rise to less than 6 Celsius degrees.

9. Establish borehole spacing to assure that areal power density of 57 Kw/acre is not exceeded, borehole wall temperatures remain below 275°C, and rock mass temperature at 1 m into rock is below 200°C. This spacing must consider the 50°C at 50-yr criteria identified in retrieval-related criteria (Section 6.1.1.7).

Additional criteria planned for use during the advanced conceptual design (ACD) and the licensing application design (LAD) are provided in Section 8.3.2.2 where performance measures and goals are documented.

6.1.2 REFERENCE DESIGN DATA BASE

In this section a summary of the geologic and geotechnical data used for the repository design is presented. The reference design data base is a subset of the Reference Information Base (RIB) (Appendix Q of SNL, 1987) which is currently being developed. The RIB will be revised periodically and will contain all reference technical information for the NNWSI Project that support analyses for site characterization, design, and performance assessment. The RIB contains a summary of all physical, thermal, and mechanical data including recommended values and ranges that pertain to site characteristics. The recommended values and ranges will change as more site specific data become available.

The SCP-CDR (SNL, 1987, Appendix O) is the primary source of geotechnical information for thermal and mechanical analyses. This appendix of the SCP-CDR includes a complete set of design data including the required ranges for parametric and sensitivity analyses. Also, more sophisticated material models evolved that required data that were previously uncompiled and not analyzed. The more recent data supplement the older data and, in general, do not replace it. This section contains the data that were used in the conceptual design.

The sources of the data are discussed, and appropriate sections of Chapters 1 through 6 are referenced as needed. A description of the principal site characteristics that form the basis for the design required to perform the design analyses is provided.

Specific consideration is given to site geology (stratigraphy and structure), in situ stress and thermal conditions, hydrologic regime, rock strength, rock discontinuities, thermal properties, and seismic-tectonic conditions in the context of the design. A reasonably expected range for each characteristic is established, determined through quantitative analysis or engineering judgment, as appropriate. Discussions of the methods used to establish these ranges follows.

The mean values and standard deviations for site parameters have been determined using data obtained during preliminary site investigations (Chapters 1-5). The value of the standard deviation relative to the mean value for a given parameter is currently being used as a measure of the uncertainty associated with that parameter. The data currently available from preliminary testing do not permit any better quantification of uncertainties than those obtained by this method and do not permit the identification of probability density functions for the site characterization parameters. Essentially all data presented contain the uncertainty related to limited samples from the few boreholes that exist at the site. The number of tests needed to quantify the uncertainties in the context of addressing design and performance issues are discussed in Chapter 8.

6.1.2.1 Site geology

Information regarding site geology is important to site selection and location (Figure 6-3), surface facilities location, and underground boundaries constraints. The site geology also provides the framework for all geotechnical and hydrological data collected. The data provided in Chapter 2, Geoengineering, combined with the three-dimensional geologic perspective of the site presented in Chapter 1, Geology, permit the development of a conceptual geologic model for design analyses. A detailed description of the regional and site geology is given in Chapter 1 and provides essential background information for the design data base. The information presented in Chapter 1 results from extensive field mapping of the site and adjacent areas, study of core taken from drillholes, and geophysical methods applied both at the surface and in the drillholes. Specifically, Section 1.2 (stratigraphy and lithology), characterizes the lithostratigraphic sequences and their major unconformities, ages, ranges and thickness, spatial extent, major rock units, and vertical and lateral variation. Also, Section 1.3 (structural geology and tectonics of the candidate area and site) contains a characterization of the structural elements, for example faults, fractures, and joints. Specific aspects of the site topography, stratigraphy, and structure most important to design and performance are given here.

6.1.2.1.1 Topography and terrain

An understanding of the topography and terrain of the site and vicinity is important to the design for a number of reasons. For example, locations of surface facilities and accesses to them are influenced by terrain (Section 6.2.2, Current repository design description). Local topographic variations are important in assessing flood potential (Section 6.1.2.6). Topographic variations influence the underground design in terms of determination of the usable area while maintaining the required amount of overburden (the overlying material above the horizon of interest). Also, in situ stresses are influenced, in part, by local topography.

The Yucca Mountain site lies within the Basin and Range physiographic province, a broad region of generally linear mountain ranges and intervening valleys. The site is in the southern part of the Great Basin, a subdivision of the Basin and Range Province. Figure 6-4 shows the physiographic features in the region. The elevation of northern Yucca Mountain is approximately 1,500 m. This is more than 370 m above the western edge of Jackass Flats to the east and more than 300 m higher than the eastern edge of Crater Flat to the west of Yucca Mountain.

Yucca Mountain is a prominent group of north-trending fault-block ridges that extend southward from Beatty Wash on the northwest to U.S. Highway 95 in the Amargosa Desert. The terrain at the site is controlled by high angle normal faults and volcanic rocks that tilt eastward. The terrain is locally steep (15 to 30°) on the west-facing side of Yucca Mountain and along some of the valleys that cut into the more gently sloping (5 to 10°) east side of Yucca Mountain. The valley floors are covered by alluvium. Alluvial and colluvial fans extend down from the lower slopes of the ridges. Fortymile Wash is cut from 13 to 26 m into the surface of Jackass Flats. North of

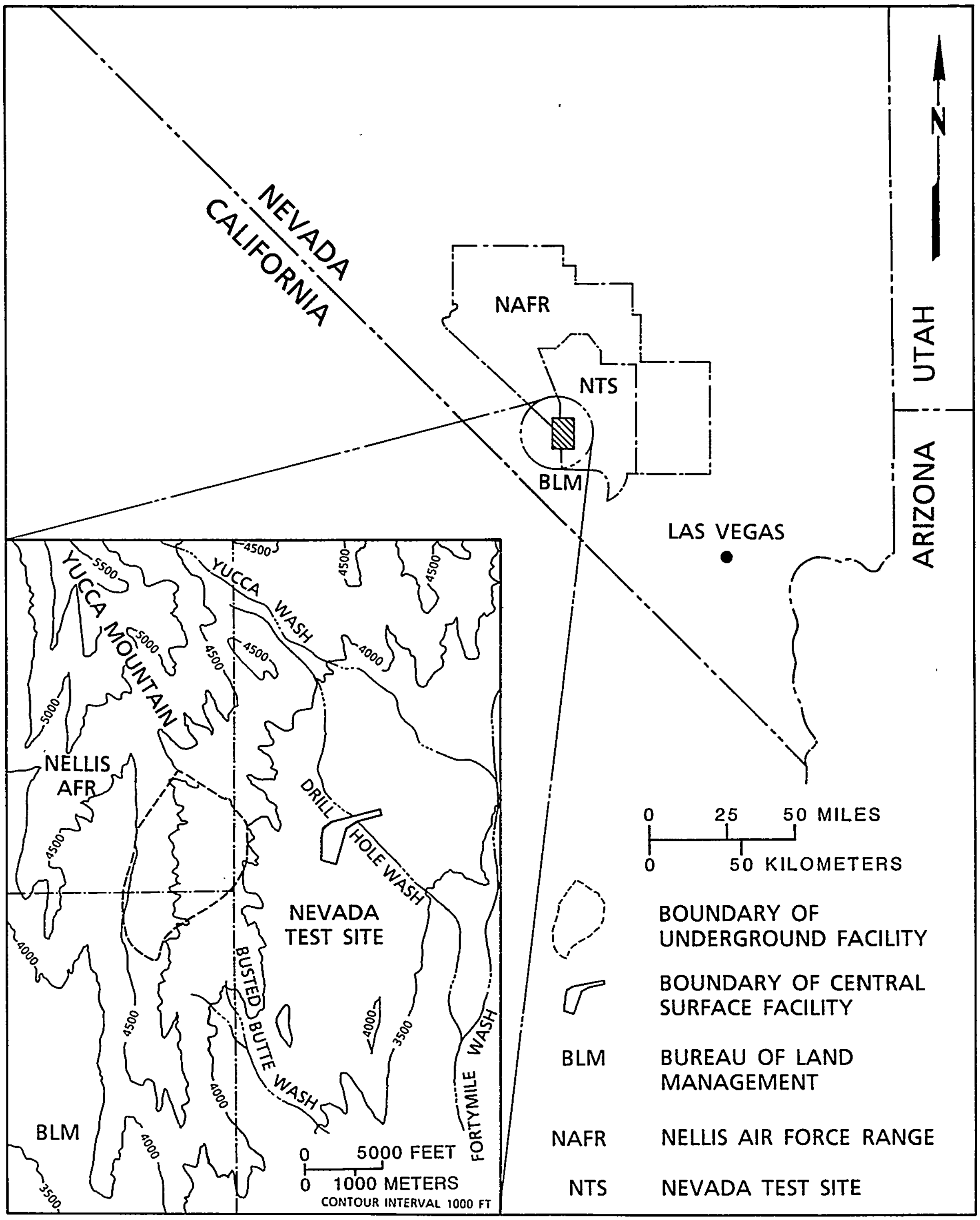


Figure 6-3. Location of Yucca Mountain site in southern Nevada. Modified from SNL (1987).

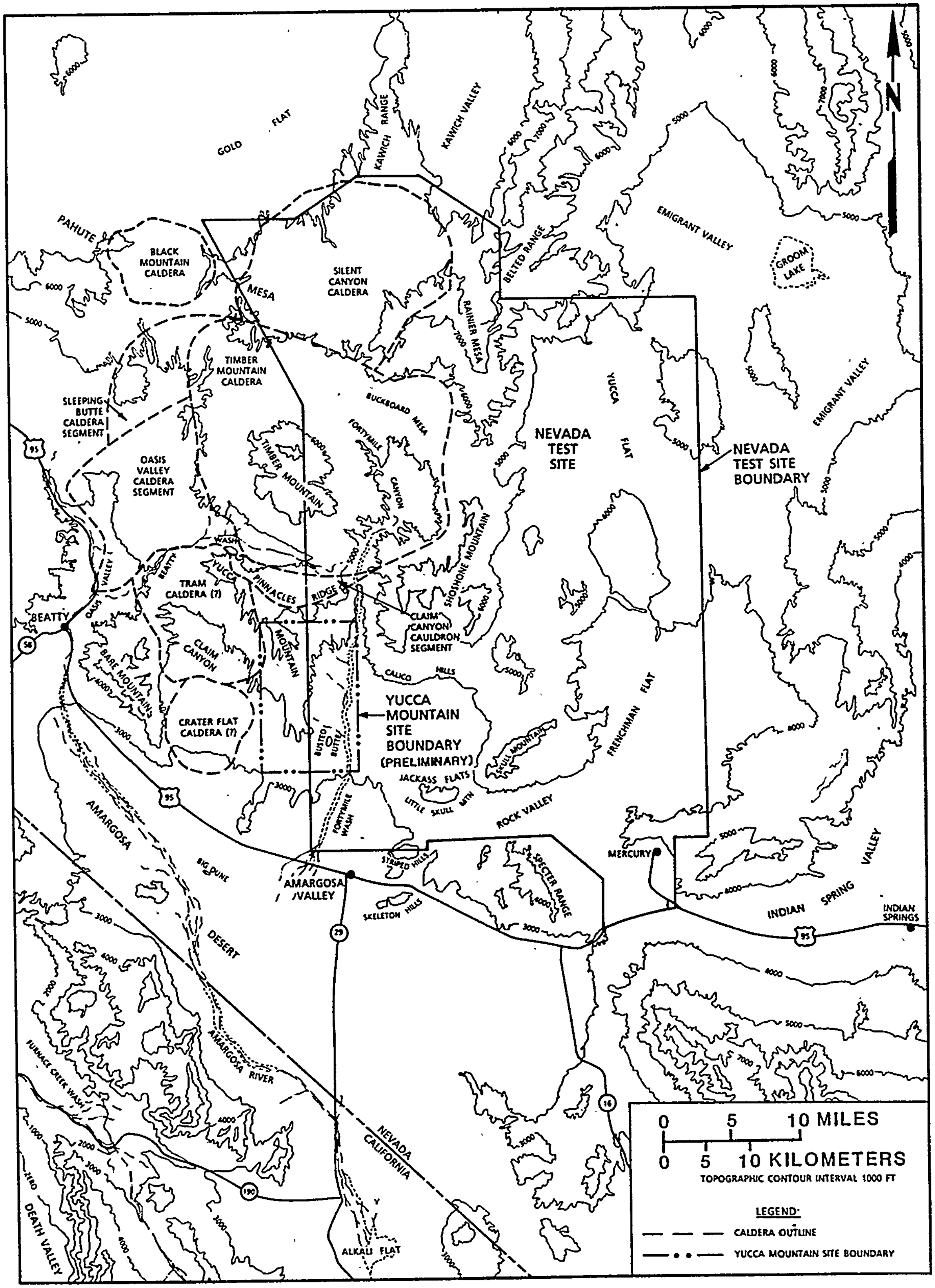


Figure 6-4. Physiographic features of Yucca Mountain and surrounding region. Modified from SNL (1987).

Yucca Mountain is the high, rugged volcanic terrain of Pinnacles Ridge. To the west of Yucca Mountain, along the west side of Crater Flat, steep alluvial fans extend from deep valleys that have been cut into Bare Mountain. Basalt cones and small lava flows are present on the surface of the southern half of Crater Flat. The topographic expression of these physiographic features is shown in Figure 6-5.

6.1.2.1.2 Near-surface soil and rock

The physical and engineering properties of the near-surface soil and rock are required for design because they influence the selection of specific locations for surface facilities and associated structural foundations. Limited preliminary investigations, consisting of four test pits and eight exploratory borings, contributed to the selection of a reference site for the central surface facilities for the purpose of the conceptual design (Neal, 1985). In general, the top 0.3 to 0.7 m of the site is a loose, fine-grained sandy soil overlying approximately 2 m of material that is partly to wholly cemented with calcite (caliche). Below the caliche layer is an 11- to 50-m-thick layer of very dense, gravelly, sandy alluvial material, which overlies the ash-flow tuff bedrock. Section 2.7.3 provides a detailed description of the geoengineering properties of the near-surface soil and rock.

A summary of the physical and engineering properties of the surface materials (Ho et al., 1986) at the proposed location of the central surface facilities is presented in Table 6-9. These properties are based on preliminary tests on samples of the underlying uncemented surface materials at the proposed surface facilities location. Samples were taken at depths of 4 m or less. The combination of the paucity of data and the fact that certain of the recommended values and ranges are based entirely on engineering judgment attach a fair degree of uncertainty to the values listed. Because the facility foundations are based on soils deeper than those tested, and because confining stresses increase with depth, the estimated values of the parameters were considered suitable for the SCP-CD.

6.1.2.1.3 Stratigraphy and lithology

The stratigraphy identifies, in part, the thermal/mechanical medium for disposal of the radioactive waste. Yucca Mountain consists of a layered sequence of ash-flow tuffs that are welded, nonwelded, and bedded tuff as detailed in Chapter 1, Geology. The ranges in thickness, spatial extent, and vertical and lateral variations of the stratigraphic units as determined from core, surface mapping, and geophysical techniques are described in Section 1.2. Definition of these stratigraphic units and their variability is important on all scales in order to understand possible scenarios of thermomechanical response. Within the stratigraphic units, variations in the degree of welding, devitrification, and zeolitization have been identified. These variations in physical and mineralogic characteristics can affect the thermomechanical response. The exact manifestations of these effects as they apply to the mechanical and thermal properties are uncertain at this time.

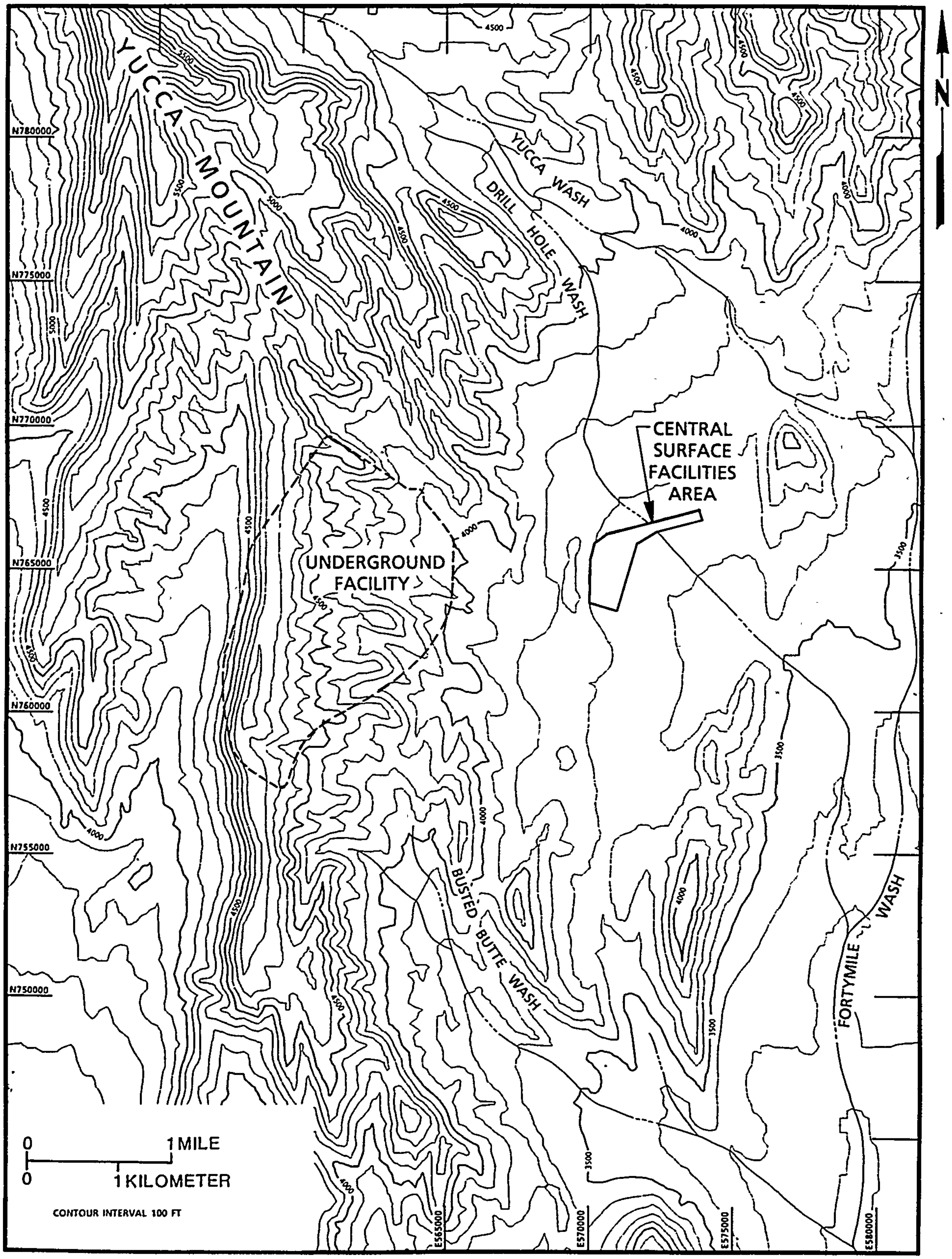


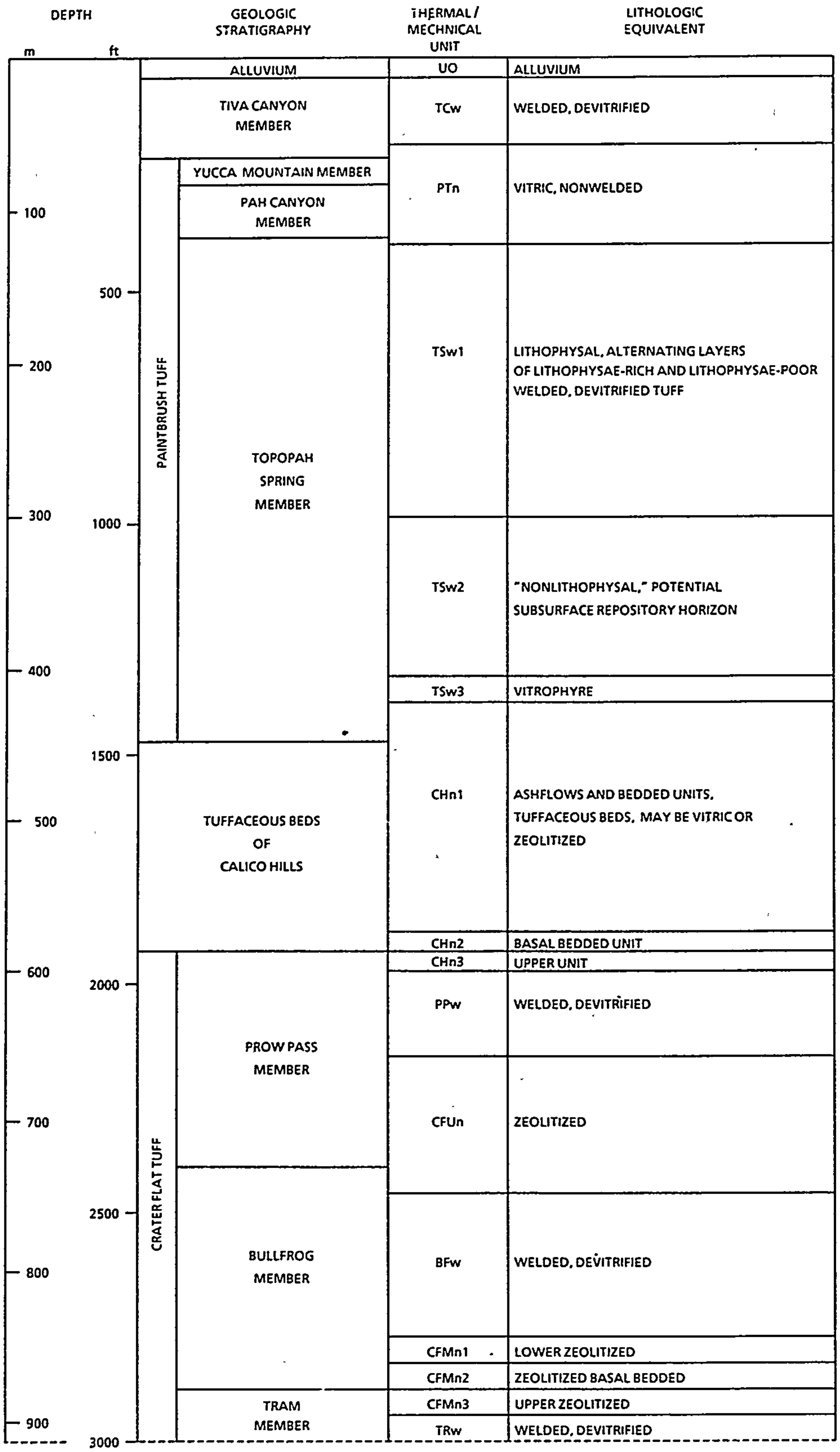
Figure 6-5. Site topographic map.

Table 6-9. Summary of physical and engineering properties of surface materials[a]

Property[b]	Value
PHYSICAL PROPERTIES	
Soil classification	GP to GM[c]
Natural moisture content	7.2%
Absorption	7.9%
In situ density	1.62-1.79 g/cm^3 (101-112 pcf)
Percent of maximum dry density	93.5-100%
Specific gravity of soil solids	2.43
Void ratio	0.37
Optimum moisture content at maximum dry density	12.0-14.7%
ENGINEERING PROPERTIES[d]	
Young's modulus	0.7-1.4 GPa (10,000-20,000 psi)
Poisson's ratio	0.30-0.35
Modulus of subgrade reaction	5,536-8,304 g/cm^3 (200-300 psi)
Cohesion	0
Angle of internal friction	33-37°
Allowable bearing pressure[e]	0.3 MPa (6,000 psf)

[a]Information taken from Ho et al. (1986).
[b]Values and ranges of physical properties are from test pit SFS-3.
[c]Unified Soil Classification: GP includes poorly graded gravels, gravel-sand mixtures, and few or no fines; GM includes silty gravels, and gravel-sand silt mixtures, which may be poorly graded.
[d]Estimated from index properties.
[e]For footings wider than 4 ft, subject to verification that settling for large structures will be tolerable.

However, redundancy in testing has made it possible to quantify probable effects of these physical and mineralogic characteristics on measured mechanical and thermal properties, as discussed under sample selection logic in the introduction to Chapter 2 and in Section 2.1.5. Furthermore, assessment of site stratigraphy coupled with mechanical and thermal properties testing resulted in the thermal/mechanical stratigraphy (discussed under stratigraphic framework for testing in the introduction to Chapter 2) presented in Figure 6-6 (SNL, 1987, Figure 2-8).



THE 'NONLITHOPHYSAL' PORTION OF THE TOPOPAH SPRING MEMBER MAY CONTAIN A SMALL PERCENTAGE OF LITHOPHYSAE

Figure 6-6. Correlation between the thermal/mechanical stratigraphy and the geologic stratigraphy. Modified from SNL (1987).

6.1.2.1.4 Structure

Structural elements such as faults and fractures are used in the empirical and analytical models presented in Section 8.3.2.4 because they are significant factors in evaluating rock mass behavior. Spatial variations in the occurrence of these structural elements introduce a degree of uncertainty in analyses addressing the thermomechanical response of the rock mass. A discussion of the history and relationship of the fractures and faults and maps showing locations and attitudes of these features are included in Section 1.3.2.2. Site-specific subsurface expression of these features is important to underground design because they introduce uncertainty in the determination of the potential thermomechanical response. Specific data on these rock discontinuities is presented in Section 6.1.2.3.4.

6.1.2.1.5 Three-dimensional thermal/mechanical stratigraphy model

The thermal/mechanical stratigraphy (discussed under stratigraphic framework for testing in the introduction to Chapter 2 and shown in Figure 6-6) for Yucca Mountain tuffs (Ortiz et al., 1985) is a stratigraphic definition based on rock properties rather than on classical geologic guidelines. It is defined through a stratigraphic breakdown of the tuffs at Yucca Mountain based on unit-specific bulk, thermal, and mechanical properties. Definition of the thermal/mechanical stratigraphy, coupled with stratigraphic and structural characterizations of the type summarized in the previous section, has permitted a three-dimensional model of the site to be developed (Ortiz et al., 1985). The three-dimensional geometric relationships of the thermal/mechanical units have been estimated using a modeling technique described by Nimick and Williams (1984). Input data for the model consist of depths of contact between thermal/mechanical units in existing drillholes at Yucca Mountain. These data permit derivation of an equation that describes a three-dimensional surface representing each thermal/mechanical unit. The surface descriptions are adjusted to account for fault offset (Section 1.3.2.2). Thus, the three-dimensional model provides the spatial distribution of each thermal/mechanical unit through both stratigraphic and structural control (Figure 6-7). Uncertainty in the model is related to sparse data and assumptions linked to the interpolation of surfaces, unit definitions, and model algorithms. As additional drillhole data are obtained, the three-dimensional model will be refined.

6.1.2.2 <u>In situ conditions</u>

In situ temperature and stress are parameters important to the design of the underground facilities because they provide definition of the initial and boundary conditions for all thermomechanical analyses. These parameters are detailed in Section 2.5.2 (Thermal and thermomechanical properties of rock at the site) and Section 2.6 (Existing stress regime). Reference values, a reasonable expected range, and uncertainties in these site characteristics, all consistent with Chapter 2, are presented in the following sections.

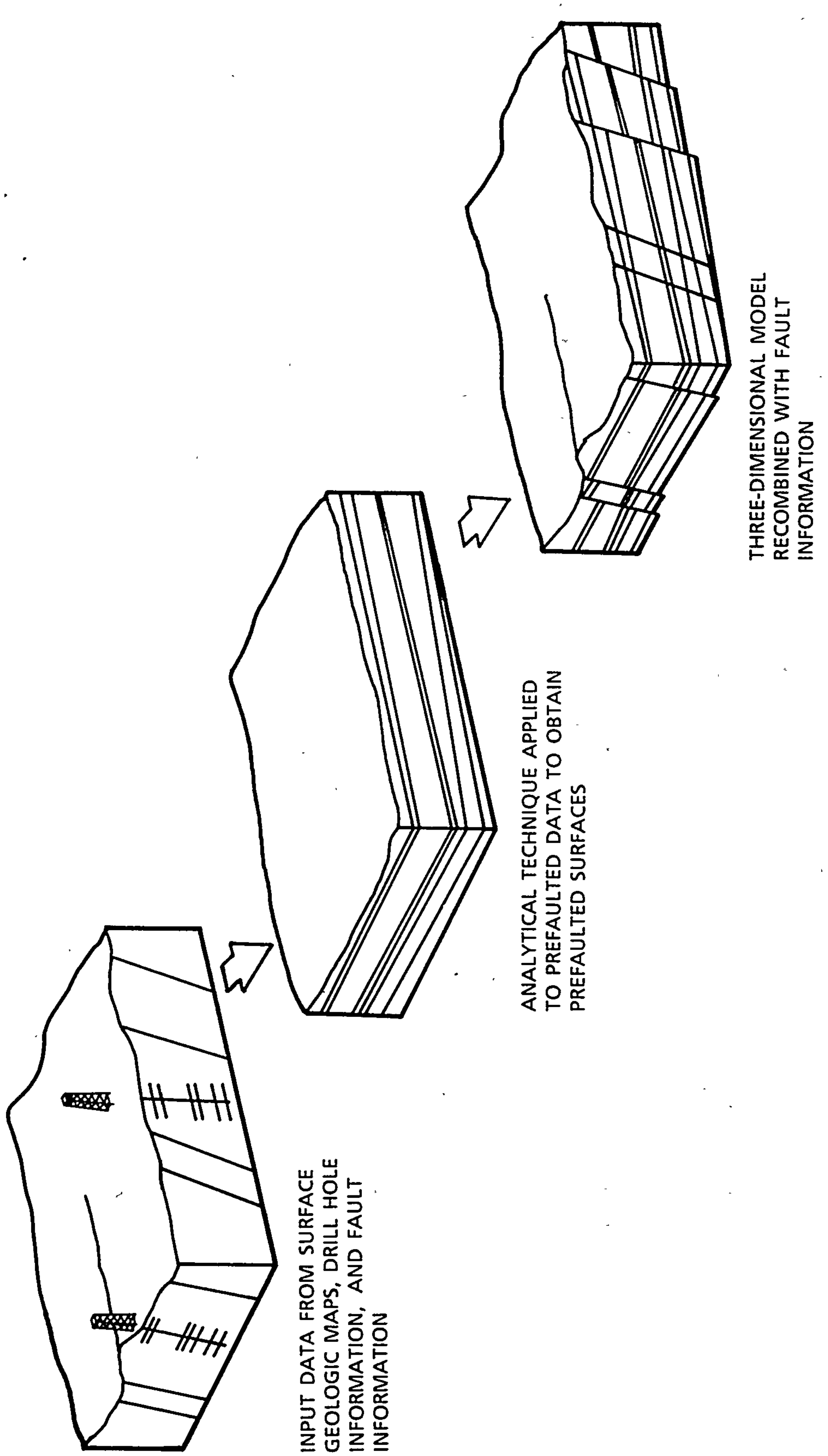


Figure 6-7. **Schematic development of the three-dimensional model. Modified from SNL (1987).**

The underground facilities would be located in the unsaturated zone above the water table. Therefore, the pore pressure was assumed to be zero for design purposes and the degree of saturation was treated as a physical property.

6.1.2.2.1 Temperature

The geothermal characteristics, including temperature, thermal properties, and heat flux, of the site are discussed in Section 1.3.2.5 (Geothermal regime). In situ rock temperatures have been measured in drillholes at Yucca Mountain (Sass and Lachenbruch, 1982). Drillholes USW G-1, USW G-2, USW H-1, and UE-25a#1 indicate an average temperature of 26°C (Table 6-10) derived from a range of 23 to 29°C temperatures observed at a repository depth of approximately 300 m (200-500 m) (Section 1.3.2.5.2). Some uncertainty in the range presented may exist and is attributed to the paucity of data.

6.1.2.2.2 Stress

The in situ stress state, characterized by the specification of the magnitude and direction of the three principal stresses, is the stress state upon which excavation and thermal stresses are superposed. Stock et al. (1985), in analyzing the stress state at the site from stress measurements at Yucca Mountain, concluded that the maximum principal stress is near vertical, and the minimum and intermediate principal stresses are near horizontal.

The magnitude of the vertical stress, which is assumed to be the maximum principal stress, is determined by multiplying the mass density of the overburden by the acceleration of gravity times the depth to the point of interest. For a given locale, the vertical stress increases nearly linearly with depth, depending on vertical variations in density related to stratigraphy and possibly degree of saturation. For a given elevation within the repository area, the vertical stress can vary horizontally by about 1 MPa because of topographic variations and differing amounts of overburden (Bauer et al., 1985a). A reasonable range in the vertical stress is approximately 5 to 10 MPa for repository horizons (Stock et al., 1984), depending on elevation and local topographic variations.

The orientation of the minimum horizontal stress ranges from N.50°W. to N.65°W. according to measurements (Stock et al., 1984) and regional structures (Carr, 1974). The magnitude of the minimum horizontal stress is often expressed as the ratio of minimum horizontal stress to the vertical stress, K_o. A working range of K_o between 0.3 and 0.8 is recommended based on regional tectonics, stress measurements, and finite-element modeling (Bauer et al., 1985a). The lower bound for K_o was established by finite-element modeling, which considered only gravity loading. A single stress measurement (Stock et al., 1984), which is inconsistent with all others at Yucca Mountain, shows the minimum horizontal stress to be equal to the vertical stress. There is additional evidence that suggests that the two horizontal stresses are not equal (Ellis and Swolfs, 1983).

Table 6-10. Mean values and ranges for principal stresses and temperature

Parameter	Mean value[a]	Range
Vertical stress (MPa)	7.0 (1,015 psi)	5.0 to 10.0
Minimum horizontal to vertical stress ratio	0.55	0.3 to 0.8
Bearing of minimum horizontal stress	N.57°W.	N.50°W. to N.65°W.
Maximum horizontal to vertical stress ratio	0.65	0.3 to 1.0
Bearing of maximum horizontal stress	N.32°E.	N.25°E. to N.40°E.
Temperature (°C)	26	23 to 29

[a]Mean value for depths of approximately 300 m.

The maximum horizontal stress (N.25°E. to N.40°E.) is oriented perpendicular to the minimum horizontal stress. On the basis of existing data and observations, it is assumed that the magnitude of the horizontal stress is less than or equal to the vertical stress. Ancillary evidence pertaining to the magnitude of the maximum horizontal stress is consistent with the range presented (Ellis and Swolfs, 1983; Stock et al., 1985).

6.1.2.3 Geotechnical data

Geotechnical data, including physical, mechanical, thermal, and discontinuity properties, are important to design because these data are the material properties used in design analyses (Section 8.3.2.5), which assess the response of the rock mass to excavation and thermally induced loads. The design values for each thermal and mechanical unit (discussed under the "current data base" heading in the introduction to Chapter 2) are presented in Tables 6-11, 6-12, and 6-13. These tables document the design values used in developing and evaluating the SCP conceptual design. The values presented in the column titled variability evaluation represent more recent results of data analyses and establish ranges for properties in some instances. Some of the information contained in these tables is not as current as the information contained in Chapter 2. These values are derived from detailed analyses of laboratory and field data and numerical analyses. The introductory section (sample selection logic) in Chapter 2 describes the details of the

Table 6-11. Physical properties of intact rock and rock mass for thermal/mechanical units[a] at Yucca Mountain[b]

Thermal/ mechanical unit	Grain density (g/cc) Design value[c]	Grain density Variability evaluation[d] value	Grain density Variability evaluation range	Porosity Design value	Porosity Variability evaluation value	Porosity Variability evaluation range	In situ saturation Design value	In situ saturation Variability evaluation value	In situ saturation Variability evaluation range	Bulk density at in situ saturation (g/cc) Design value	Bulk density at in situ saturation Variability evaluation value	Dry bulk density (g/cc) Design value	Dry bulk density Variability evaluation value
TCw	2.51	2.51	±0.04	0.114	0.11	±0.04	0.8	0.67	±0.23	2.32	2.31	2.23	2.23
PTn	2.37	2.37	±0.15	0.448	0.45	±0.15	0.8	0.61	±0.15	1.67	1.58	1.31	1.30
TSw1[e]	2.53	2.54	±0.04	0.148	0.14	±0.04	0.8	0.65	±0.19	2.27	2.25	2.15	2.15
TSw1[f]	2.53	2.53	±0.02	0.348	0.35[g]	±0.03[g]	0.8	0.65	±0.19	2.00	NA[h]	1.65	NA
TSw2	2.55	2.55	±0.03	0.121	0.12	±0.03	0.8	0.65	±0.19	2.34	2.32	2.25	2.24
TSw3	2.39	2.39	±0.02	0.043	0.04	±0.03	0.8	0.65	±0.19	2.34	2.32	2.25	2.29
CHnlv	2.34	2.34	±0.05	0.365	0.36	±0.09	0.8	0.90	±0.06	1.78	1.82	1.49	1.50
CHnlz	2.41	2.41	±0.06	0.327	0.33	±0.04	0.8	0.91	±0.06	1.96	1.92	1.63	1.61
CHn2z	2.54	2.54	±0.12	0.286	0.29	±0.06	1.0	1.0[i]	NA	1.96	2.09	1.63	1.80
CHn3z	2.41	2.41	±0.04	0.360	0.36	±0.08	1.0	1.0[i]	NA	1.96	1.90	1.63	1.54
PPw	2.58	2.58	±0.04	0.240	0.24	±0.07	1.0	1.0[i]	NA	2.20	2.20	1.96	1.96
CFUn	2.43	2.43	±0.07	0.297	0.30	±0.08	1.0	1.0[i]	NA	2.00	2.00	1.17	1.70
BFw	2.60	2.60	±0.04	0.238	0.24	±0.08	1.0	1.0[i]	NA	2.23	2.22	1.98	1.98
CFMn1	2.41	2.41	±0.06	0.246	0.25	±0.05	1.0	1.0[i]	NA	2.09	2.06	1.83	1.81
CFMn2	2.52	2.52	±0.06	0.242	0.24	±0.03	1.0	1.0[i]	NA	2.09	2.16	1.83	1.92
CFMn3	2.44	2.44	±0.07	0.267	0.27	±0.03	1.0	1.0[i]	NA	2.09	2.05	1.83	1.78
TRw	2.63	2.63	±0.04	0.188	0.19	±0.05	1.0	1.0[i]	NA	2.32	2.32	2.14	2.13

[a] Thermal/mechanical units are defined in Figure 6-6.
[b] See Appendix O of SNL (1987).
[c] Design values represent the basis for the SCP-CDR.
[d] Variability evaluation represents more recent results of data analyses and establishes ranges for properties.
[e] Nonlithophysal layers in unit TSw1.
[f] Lithophysal layers in unit TSw1.
[g] For lithophysal layers, the total porosity is $\phi_m \cdot M + \phi_A \cdot A + \phi_L$, where ϕ_m is matrix porosity, ϕ_A is the porosity of the vapor-phase-altered material, ϕ_L is the volume fraction lithophysal cavities, and M and A are volume fractions of matrix and vapor-phase-altered material, respectively (Price et al, 1985).
[h] NA = not available.
[i] These units are at least partly below the water table. For the purpose of thermal/mechanical analyses, they have been assigned a saturation of 1.0.

Table 6-12. Mechanical properties of intact rock for thermal/mechanical units[a] at Yucca Mountain[b] (page 1 of 3)

Thermal/ mechanical unit	Young's modulus (GPa)			Poisson's ratio			σ_c (MPa)[e]		
	Design value[c]	Variability evaluation[d]		Design value	Variability evaluation		Design value	Variability evaluation	
		value	range		value	range		value	range
TCw	30.8	40.0	±11.1	0.10	0.24	NA[f]	155	240	±163.5
PTn	2.2	3.8	±3.9	0.18	0.16	NA	7	19	±10.9
TSw1[g]	23.9	31.7	±17.9	0.13	0.25	±0.05	114	127	±16
TSw1[h]	15.2	15.5	±3.2	0.16	0.16	±0.03	18	16	±5
TSw2	31.1	30.4	±6.3	0.22	0.24	±0.06	171	166	±65
TSw3	25.0	NA	NA	0.11	NA	NA	46	NA	NA
CHn1v	4.8	7.1	±4.4	0.15	0.16	NA	17	27	±12.4
CHn1z	7.1	7.1	±2.1	0.16	0.16	±0.08	27	27	±9
CHn2z	8.6	11.5	±4.0	0.20	0.16	NA	34	40	±12.7
CHn3z	5.0	7.1	±4.4	0.17	0.16	NA	18	27	±11.0
PPw	12.1	16.3	±7.8	0.20	0.13	NA	51	57	±30.6
CFUn	7.6	7.6	±3.8	0.16	0.16	NA	31	31	±11
BFw	10.8	10.8	±4.7	0.13	0.13	±0.02	42	42	±14
CFMn1	11.5	15.2	±5.2	0.14	0.16	NA	48	52	±19.4
CFMn2	11.9	16.3	±3.4	0.18	0.16	NA	50	57	±13.1
CFMn3	9.9	13.2	±2.7	0.15	0.16	NA	40	45	NA
TRw	7.6	17.6	±3.8	0.18	0.13	NA	72	72	±23

Table 6-12. Mechanical properties of intact rock for thermal/mechanical units[a] at Yucca Mountain[b] (page 2 of 3)

Thermal/ mechanical unit	Cohesion (MPa)			ϕ(°)[i]			Tensile strength (MPa)		
	Design value	Variability evaluation value	Variability evaluation range	Design value	Variability evaluation value	Variability evaluation range	Design value	Variability evaluation value	Variability evaluation range
TCw	45	51.0	±20.26	29.7	44.0	±0.20	17.6	17.9	NA
PTn	3	8.0	±4.18	6.6	8.5	±0.08	1.0	1.0	NA
TSw1[g]	35	36.0	±11.40	27.4	34.9	±0.15	14.6	12.0	±4.6
TSw1[h]	8	11.0	NA	14.3	12.5	NA	1.0	1.0	NA
TSw2	50	34.0	±11.40	29.2	23.5	±0.15	16.9	15.2	NA
TSw3	NA	NA	NA	NA	NA	NA	NA	NA	NA
CHn1v	7	11.0	±4.28	13.4	12.0	±0.08	1.0	1.0	NA
CHn1z	10	10.9	±1.6	15.8	7.6	±2.60	1.0	1.0	NA
CHn2z	12	15.0	±3.33	18.5	16.4	±0.06	3.0	2.6	NA
CHn3z	7	11.0	±3.74	13.7	12.0	±0.07	1.0	1.0	NA
PPw	17	20.0	±7.04	21.4	21.0	±0.12	6.8	6.9	NA
CFUn	11	14.0	±5.51	17.8	15.6	±0.10	2.1	1.8	NA
BFw	14	20.0	±8.21	21.6	21.0	±0.14	7.0	6.9	NA
CFMn1	17	19.0	±5.21	21.0	19.9	±0.09	6.3	6.0	NA
CFMn2	17	20.0	±2.93	21.3	21.0	±0.05	6.7	6.9	NA
CFMn3	14	17.0	±2.83	19.7	18.0	±0.05	4.6	4.3	NA
TRw	23	27.0	±9.11	24.8	27.6	±0.14	11.3	11.1	NA

Table 6-12. Mechanical properties of intact rock for thermal/mechanical units[a] at Yucca Mountain[b] (page 3 of 3)

Footnotes

[a]Thermal/mechanical units are defined in Figure 6-6.
[b]See Appendix O of SNL (1987).
[c]Design values represent the basis for the Site Characterization Plan-Conceptual Design.
[d]Variability evaluation represents more recent results of data analyses and establishes ranges for properties.
[e]σ_c = unconfined compressive strength.
[f]NA = not available.
[g]Nonlithophysal portions of Unit TSw1.
[h]Lithophysal portions of Unit TSw1.
[i]ϕ = angle of internal friction.

Table 6-13. Mechanical properties and modeling parameters for fractures in thermal/mechanical units[a] at Yucca Mountain[b] (page 1 of 2)

Thermal/ mechanical unit	Unstressed aperture (μm)				Half-closure stress (MPa)				Shear stiffness (MPa/m)				Joint cohesion			
	Design value[d]	Variability evaluation[c]			Design value	Variability evaluation			Design value	Variability evaluation			Design value	Variability evaluation		
		UB[e]	RV[f]	LB[g]		UB	RV	LB		UB	RV	LB		UB	RV	LB
TCw	NA[h]	36.0	18.0	3.0	NA	2.0	1.1	0.5	NA	10^7	10^6	10^5	1.0	0.2	0.1	0
PTn	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	1.0	0.2	0.1	0
TSw1	NA	36.0	18.0	3.0	NA	2.0	1.1	0.5	NA	10^7	10^6	10^5	1.0	0.2	0.1	0
TSw2	NA	36.0	18.0	3.0	NA	2.0	1.1	0.5	NA	10^7	10^6	10^5	1.0	0.2	0.1	0
TSw3	NA	36.0	18.0	3.0	NA	2.0	1.1	0.5	NA	10^7	10^6	10^5	1.0	0.2	0.1	0
CHn1v	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	1.0	0.2	0.1	0
CHn1z	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	0.4	0.2	0.1	0
CHn2z	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	0.4	0.2	0.1	0
CHn3z	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	0.4	0.2	0.1	0
PPw	NA	36.0	18.0	3.0	NA	2.0	1.1	0.5	NA	10^7	10^6	10^5	1.0	0.7	0.1	0
CFUn	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	0.4	0.7	0.1	0
BFw	NA	36.0	18.0	3.0	NA	2.0	1.1	0.5	NA	10^7	10^6	10^5	1.0	0.7	0.1	0
CFMn1	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	0.4	0.7	0.1	0
CFMn2	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	0.4	0.7	0.1	0
CFMn3	NA	26.0	5.4	5.4	NA	2.8	1.2	1.2	NA	10^7	10^6	10^5	0.4	0.7	0.1	0
TRw	NA	36.0	18.0	3.0	NA	2.0	1.1	0.5	NA	10^7	10^6	10^5	1.0	0.7	0.1	0

Table 6-13. Mechanical properties and modeling parameters for fractures in thermal/mechanical units[a] at Yucca Mountain[b] (page 2 of 2)

Thermal/ mechanical unit	Friction coefficient: Design value[d]	Friction coefficient: Variability evaluation UB	RV	LB	JCS_o[(i)]: Design value	JCS_o: Variability evaluation UB	RV	LB	JRC_o[(j)]: Design value	JRC_o: Variability evaluation UB	RV	LB	ϕ_r[(k)]: Design value	ϕ_r: Variability evaluation UB	RV	LB
TCw	0.80	0.8	0.54	0.2	NA	406.5	243.0	79.5	NA	12	9	6	NA	38.7	28.4	11.3
PTn	0.80	0.8	0.59	0.2	NA	6.8	17.7	28.6	NA	8	5	2	NA	38.7	30.5	11.3
TSw1	0.80	0.8	0.54	0.2	NA	68.5	135.1	201.7	NA	12	9	6	NA	38.7	28.4	11.3
TSw2	0.80	0.8	0.54	0.2	NA	113.0	171.0	229.0	NA	12	9	6	NA	38.7	28.4	11.3
TSw3	0.80	0.8	0.54	0.2	NA	13.0	46.0	79.0	NA	12	9	6	NA	38.7	28.4	11.3
CHn1v	0.80	0.8	0.59	0.2	NA	39.1	26.7	14.3	NA	8	5	2	NA	38.7	30.5	11.3
CHn1z	0.55	0.8	0.54	0.2	NA	36.0	27.0	18.0	NA	8	5	2	NA	38.7	28.4	11.3
CHn2z	0.55	0.8	0.59	0.2	NA	52.6	39.9	27.2	NA	8	5	2	NA	38.7	30.5	11.3
CHn3z	0.55	0.8	0.59	0.2	NA	37.7	26.7	15.7	NA	8	5	2	NA	38.7	30.5	11.3
PPw	0.80	0.8	0.59	0.2	NA	87.6	56.6	26.0	NA	12	9	6	NA	38.7	30.5	11.3
CFUn	0.55	0.8	0.64	0.2	NA	56.0	37.5	19.0	NA	8	5	2	NA	38.7	32.6	11.3
BFw	0.80	0.8	0.59	0.2	NA	56.0	42.0	28.0	NA	12	9	6	NA	38.7	30.5	11.3
CFMn1	0.55	0.8	0.64	0.2	NA	71.9	52.5	33.1	NA	8	5	2	NA	38.7	32.6	11.3
CFMn2	0.55	0.8	0.64	0.2	NA	69.7	56.6	43.5	NA	8	5	2	NA	38.7	32.6	11.3
CFMn3	0.55	0.8	0.64	0.2	NA	NA	45.0	NA	NA	8	5	2	NA	38.7	32.6	11.3
TRw	0.80	0.8	0.59	0.2	NA	96.0	72.0	46.0	NA	12	9	6	NA	38.7	30.5	11.3

[a]Thermal/mechanical units defined in Figure 6-6.
[b]See Appendix O of SNL (1987).
[c]Variability evaluation values represent more recent results of data analyses and establishes ranges for properties.
[d]Design values represent the basis for the Site Characterization Plan-Conceptual Design.
[e]UB = upper bound.
[f]RV = recommended value.
[g]LB = lower bound.
[h]NA = not applicable.
[i]JCS_o = joint wall compressive strength.
[j]JRC_o = joint wall roughness coefficient.
[k]ϕ_r = residual friction angle.

philosophy and procedures for assessing how representative the data base is of the in situ material. The logic for determining the recommended mechanical properties and their definitions is contained in Section 2.3 (mechanical properties of rock units - large scale). The range presented for each material property warrants discussion here.

6.1.2.3.1 Physical properties

Physical properties, which include porosity, grain density, and bulk density, are important to the understanding of the physical nature of the mechanical and thermal responses of tuff and are presented in Table 6-11.

Total porosity is the ratio of the volume of void space to the total volume of a material and includes matrix porosity, lithophysal porosity, and fracture porosity. Porosity is important to the understanding of bulk density and thermal/mechanical properties. For example, porosity is used to define the maximum water content that can exist in the rock in the 100 percent saturated state. Water content is important in understanding thermal properties (Section 2.4) and in performing thermal and mechanical analyses (Section 8.3.2.2). The term "lithophysal" is applied to the TSw1 unit that contains lithophysae (lithophysal cavities and associated vapor-phase-altered material). The TSw2 unit is referred to as "nonlithophysal", although it may contain a small percentage of lithophysae (See Figure 6-6). Price et al. (1985) provides a detailed description of lithophysae and the partitioning of porosity in unit TSw1. Lithophysae can locally increase the porosity of the rock mass to values of about 35 percent. This understanding is important because the reference state, and consequent bulk and thermal properties, is based on hydrologic considerations that indicate that lithophysal voids are not saturated when the matrix is less than 100 percent saturated. For all thermal and mechanical units total porosities presented neglect the potential contribution of fracture porosity. Fracture porosity is the ratio of void-volume of fractures to the volume of the portion of the rock mass under consideration. Because the total porosities for the thermal/mechanical units are two orders of magnitude greater than the fracture porosities reported, their potential effect on thermal properties, which is a volumetric contribution, in the context of the following discussion is considered negligible. The effect of fractures is considered important to the mechanical response and is discussed in Section 6.1.2.3.3 (strength properties). Additional details of the connection between porosity and physical, thermal and mechanical properties are contained within the individual sections on these subjects.

Grain density is defined as the grain weight divided by the grain volume; whereas bulk density is defined as the sum of the grain plus pore water weight divided by the bulk volume. These two parameters are important in understanding thermal properties (Section 2.4), and performing thermomechanical analyses (Section 8.3.2.2). The range presented for bulk density represents plus and minus one standard deviation from the mean and accounts for some of the variability discussed in Section 6.1.2.1.

The saturation states for thermal/mechanical units located above the water table were obtained from laboratory measurements on cores taken from the site (Montazer and Wilson, 1984). A 100 percent saturation state was assumed for thermal/mechanical units located below the water table. Thermal conductivity for all units and thermal capacitance for the units below the water table (PPw and underlying units) are for a nominal saturation of 1.0 (100 percent saturation), whereas thermal capacitance is calculated using saturations from Montazer and Wilson (1984) for units above the water table (CHn3 and overlying units).

6.1.2.3.2 Deformability properties

Deformability properties are important parameters necessary to perform analyses (Section 8.3.2.5) that determine the mechanical response induced by excavation (mechanical) and waste emplacement (thermomechanical). In the context of the reference design data base, deformability properties are defined as those properties of intact rock, fractures, and rock mass that relate stress to strain before the onset of yield (Section 6.1.2.3.3).

For intact rock, the deformability properties of interest are Young's modulus and Poisson's ratio (Table 6-12). The design value is the mean determined from available laboratory measurements or from empirical relationships (Section 2.1.4.2) and represents the value used in the Site Characterization Plan-Conceptual Design. The value and range provided in the columns labeled variability evaluation in Table 6-12 represent more recent results of data analyses. These values were determined through analysis of data collected on approximately 300 samples deformed at a standard set of conditions (Section 2.1.2.2).

Limited data were collected to investigate potential environmental effects upon elastic and strength properties. The environmental effects investigated included strain rate (Section 2.1.2.3.1.1), temperature (Section 2.1.2.1.3), and scale (Price, 1986). The mean value of the elastic properties determined from each of these investigations falls within the range determined by the tests run at the standard set of conditions. This implies that the tuff behaves as an elastic solid for the range of conditions investigated.

Other deformability properties required for analytical techniques applied to determine the mechanical response of fractured rock in Section 6.4.10 are the shear and normal stiffness of joints. Table 6-13 provides values for some of these properties. The shear stiffness values and coefficients of friction are derived from very limited laboratory measurements on tuff (Olsson, 1987). These data contain uncertainties because the data base is small and is built on data for artificial saw-cut surfaces. However, the stiffness and coefficient-of-friction values listed on Table 6-13 fall within the range of measured values for natural fractures in other works (Sun et al., 1985). Therefore, it is not expected that the results of the design analyses that use these values will change significantly as data on natural fractures are collected. Also, it is generally predicted that slippage along joints will be small and confined to regions very near underground openings, within the influence of the proposed ground support design. The normal stiffness values define the hyperbolic normal stress versus displacement relationship for fractures (Thomas, 1982). These data were obtained from natural and artificial fracture surfaces. Some fractures observed are coated or filled. The mechanical properties of these fillings may affect the mechanical response. As such, the range of shear stiffness and coefficient-of-friction values listed for use in design sensitivity analyses is large enough to account for this potential uncertainty.

For the rock mass, the deformability properties must incorporate the sum of contributed effects from the matrix and fractures (mechanical and geometric characteristics) (Table 6-14). The rock mass deformability properties are used in analyses (Section 8.3.2.2) in the same manner as elastic material properties. The design value of the deformation modulus is one-half of the design value of the intact rock value, and the range is given by plus and minus one standard deviation to the mean. The design value and range have been chosen based on engineering judgment; therefore, a degree of uncertainty is attached to them. The design value is supported by field measurements in similar rocks (Zimmerman et al., 1984), however, which suggest that 50 percent of the intact value is representative.

The design value and range of the Poisson's ratio for the rock mass is the same as that for intact rock (Table 6-14). Field results from the heated block experiment (Zimmerman et al., 1984) performed in densely welded tuff at G-Tunnel support this recommendation. Because field measurement of this parameter at the potential site is lacking, there is uncertainty in the design value and range presented. However, it is not expected that the design will be significantly affected by minor changes in the Poisson's ratio for the rock mass.

6.1.2.3.3 Strength properties

Strength is of fundamental importance to underground design because stability assessments are based on a comparison of the stresses predicted through analyses and the strength criterion considered. The strength criteria are either incorporated in analysis methods or are used to interpret analyses (Section 6.1.3).

The Mohr-Coulomb criterion (Jaeger and Cook, 1979) is currently used to define a strength criterion for intact rock. The linear criterion may be defined by the unconfined compressive strength and the angle of internal friction (Section 2.1.2.3, matrix compressive and tensile strengths). The design value is the mean determined from available laboratory measurements or from empirical relationships (Section 2.1.2.2) and the range is given by plus and minus one standard deviation from the mean, and represents the value used for the SCP-CD (Table 6-12). These values were determined through analysis of data collected on approximately 300 samples deformed at a standard set of conditions (Section 2.1.2.2).

There is no prescribed method for determining a rock mass strength criterion nor is there a reliable field test with which the rock mass strength can be measured. It is reasoned that the strength criterion for the rock mass should incorporate the sum of contributions from the intact rock, lithophysal porosity (where present), fractures (mechanical and geometric characteristics), scale effects, and environmental conditions. The strength

Table 6-14. Mechanical properties of the rock mass for thermal/mechanical units[a] at Yucca Mountain[b]

Thermal/ mechanical unit	Deformation modulus (GPa) Design value[c]	Variability evaluation[d] value	Variability evaluation[d] range	Poisson's ratio Design value	Variability evaluation value	Variability evaluation range	σ_c (MPa)[e] Design value	Variability evaluation value	Variability evaluation range	Cohesion (MPa) Design value	Variability evaluation value	Variability evaluation range	ϕ (deg)[f] Design value	Variability evaluation value	Variability evaluation range
TCw	15.4	20.0	±5.55	0.10	0.10	NA[g]	77.5	120.0	±81.75	22.5	26.0	±10.13	29.7	44.7	±0.20
PTn	1.1	1.9	±1.95	0.18	0.19	NA	3.5	9.5	±5.45	1.6	4.0	±2.09	6.6	8.5	±0.08
TSw1[h]	15.1	15.9	±4.2	0.20	0.22	±0.05	75.0	63.5	±33.30	22.1	18.0	±5.70	29.2	34.9	±0.15
TSw1[i]	7.6	7.6	±3.2	0.16	0.16	±0.05	18.0	16.0	±5.00	7.0	5.5	NA	14.3	12.5	NA
TSw2	15.1	15.2	±4.2	0.20	0.22	±0.05	75.4	83.0	±33.30	22.1	17.8	±5.70	29.2	23.5	±0.15
TSw3	15.1	NA	NA	0.20	NA	NA	75.4	NA	NA	22.1	NA	NA	29.2	NA	NA
CHn1v	2.4	3.6	±2.2	0.15	0.15	NA	8.5	13.5	±6.20	3.4	5.5	±2.14	13.4	12.0	±0.08
CHn1z	3.5	3.6	±2.1	0.17	0.16	±0.08	13.5	13.5	±4.50	5.1	5.4	±1.08	15.8	7.6	±2.60
CHn2z	3.5	5.8	±2.0	0.17	0.20	NA	13.5	20.0	±6.35	5.1	7.5	±1.67	15.8	16.4	±0.06
CHn3z	3.5	3.6	±2.2	0.17	0.18	NA	13.5	13.5	±5.50	5.1	5.5	±1.87	15.8	12.0	±0.07
PPw	6.1	8.2	±3.9	0.20	0.19	NA	25.5	28.5	±15.30	8.5	10.0	±3.52	21.1	21.0	±0.12
CFUn	3.8	3.8	±2.95	0.16	0.16	NA	15.5	15.5	±9.25	5.5	7.0	±2.76	17.8	15.6	±0.10
BFw	5.4	5.4	±2.35	0.13	0.13	±0.02	21.0	21.0	±7.00	7.0	10.0	±4.10	21.6	21.0	±0.14
CFMn1	5.4	7.6	±2.60	0.15	0.14	NA	22.3	26.0	±9.70	7.7	9.9	±2.61	20.5	19.9	±0.09
CFMn2	5.4	8.2	±1.7	0.15	0.17	NA	22.3	28.5	±6.55	7.7	10.0	±1.47	20.5	21.0	±0.05
CFMn3	5.4	6.6	±1.35	0.15	0.15	NA	22.3	22.5	NA	7.7	8.9	±1.42	20.5	18.0	±0.05
TRw	8.8	8.8	±0.10	0.18	0.19	NA	36.0	36.0	±11.50	11.5	13.5	±4.56	24.8	27.6	±0.14

[a]Thermal/mechanical units defined in Figure 6-6.
[b]See Appendix O of SNL, (1987).
[c]Design values represent the basis for the Site Characterization Plan-Conceptual Design Report (SNL, 1987).
[d]Variability evaluation values represent more recent results of data analyses and establish ranges for properties.
[e]σ_c = unconfined compressive strength.
[f]ϕ = angle of internal friction.
[g]NA = not available.
[h]Nonlithophysal portions of unit TSw1.
[i]Lithophysal portions of unit TSw1.

criterion provided is meant to provide a working range for engineering analyses.

Two approaches were taken simultaneously to assess the rock mass strength. These approaches were either included in analyses or were used to interpret analyses (Table 6-14). One approach involved providing an estimate of the strength criterion for intact blocks of rock in situ. In this instance, a Mohr-Coulomb strength criterion was used. The rock mass strength was assumed to be 50 percent of the unconfined compressive strength of the rock matrix and the coefficient of friction was assumed to be equal to that of intact rock. The goal of this assumption was to capture the potential effects of scale, temperature, and time on the strength of intact blocks of rock. The results of the analyses were routinely reviewed to determine whether this criterion had been exceeded.

The second approach involved providing an estimate of the propensity for slip along joints. The strength criterion for joints is given by the slip condition, which is defined by the cohesion and coefficient of friction (Jaeger and Cook, 1979). The cohesion is the fracture shear strength at zero normal stress. The range of joint cohesion presented is derived from laboratory data on ground surfaces (Table 6-13). A cohesion value of zero is a realistic minimum value for the range presented. The upper bound given is that determined from experimental work. Uncertainty in the upper limit results from the lack of sufficient data on real joints.

The recommended value for the friction coefficient is derived from data provided in Morrow and Byerlee (1984). The lower value listed for the friction coefficient in Table 6-13 is a value considered representative for certain clay gouges (Shimamoto and Logan, 1981; Morrow et al., 1982). Only a small percentage (about 2 percent) of the fractures at Yucca Mountain are clay filled (Spengler and Chornack, 1984). This value is a probable realistic lower value for these clay-filled fractures. The upper value is set from the range in values listed (Olsson and Jones, 1980; Teufel, 1981; Morrow and Byerlee, 1984) as a result of examining a variation in the environmental test conditions (e.g., rate and temperature effects). The overall range presented is based on laboratory measurements of the friction coefficient for ground surfaces under various environmental conditions (Sections 2.2.2.1). Uncertainties in the range exist because of the lack of sufficient measurements on real fractures; however, the range presented and considered by design analyses is fairly encompassing for earth materials.

The potential exists for scale-dependence of the fracture slip parameters (Barton, 1982). These effects and their impact on analysis results are currently being evaluated and will be studied further during site characterization.

Increasing time (rate effects) and increasing temperature will act to decrease the strength of intact rock (Paterson, 1978). On preexisting fractures and sawcuts, increasing time (Dieterich 1972a, 1972b), decreasing rates (Scholz et al., 1972; Scholz and Engelder, 1976), and increasing temperature (Friedman et al., 1974) all act to increase the coefficient of friction (the shear stress needed to cause sliding on a shear fracture).

The dominant deformation mechanism for intact tuff at repository conditions is fracture (Price, 1983). Since fracture in rock is a time-dependent thermally activated process (Handin and Carter, 1981), increasing time (decreasing rate) and increasing temperature will both act to weaken intact rock. Limited experimental data on tuff at elevated temperatures (Olsson and Jones, 1980; Olsson, 1982; Price, 1983) and low strain rates (Price et al., 1982) are, thus far, inconclusive in quantifying potential strength changes (Section 2.1.2.3.1). Data from the 15 tests completed to date are inconclusive in quantifying changes. Variations are present not only in temperature but also in other test conditions (pressure, strain rate, and confining pressure) and in intrinsic rock properties (density and porosity). The temperatures (<200°C) are expected to dry out the rock mass, thus strengthening it. In an attempt to account for potential time effects, most laboratory tests have been performed on fully saturated rocks, incorporating potential thermomechanical effects characteristic of time-dependent thermally-activated fracture. The deformation mechanisms active on stressed fracture surfaces are undoubtedly microfracture and crystal plasticity owing to the high stress concentrations due to the relatively low real area of contact (Teufel and Logan, 1978). Since these two mechanisms are also time-dependent thermally activated processes, increasing time (decreasing rate) and increasing temperature both act to facilitate local plastic deformation at point contacts, thus causing an increase in the real area of contact. This phenomenon is manifested in tuff as an increase in the frictional strength (Teufel, 1981).

The tensile strength of a fractured rock mass can be locally very close to zero, which is the tensile strength of individual fractures. The tensile strength of the intact rock portion of the rock mass (Table 6-12) helps define the strength criterion in the vicinity of the origin of the Mohr-Coulomb diagram. Thus, the rock mass tensile strength is meant to be a modeling parameter rather than a material property. The design value for the tensile strength of intact rock is then scaled down to account for potential effects of flaws encountered in larger size samples or in blocks of rocks more representative of the field scale defined by the fracture spacing. The rock mass tensile strength value is expected to be determined by the relationship between tensile strength and unconfined compressive strength (modified Griffith criterion) presented in Jaeger and Cook (1979).

6.1.2.3.4 Geometric characteristics of discontinuities

The geometric and mechanical characteristics of discontinuities such as faults, fractures, and joints are important to design because these characteristics, coupled with intact rock properties, allow for assessment of rock mass thermomechanical response through the empirical and analytical techniques presented in Section 8.3.2.2.

The geometric description of faults pertinent to design includes distribution, offset characteristics orientation, spacing, length, and width. This information is detailed in Section 1.3.2.2 and its relation and impact on underground design is discussed in Section 6.2.6. Uncertainties exist in terms of extrapolation of faults and their description underground. A comprehensive three-dimensional subterranean geologic description of the site area (Ortiz et al., 1985) provides a means of extrapolating surface data to depth when combined with drillhole data. This type of analysis allows for an understanding of the geometrical relationships of faults to the design. Although the mechanical properties of the fault zones at depth remain an uncertainty, bounding calculations that consider realistic variations of these properties have been performed as part of the design process (Hustrulid, 1984b). Also, in similar lithology and in situ stress conditions at G-Tunnel, faults have been encountered at depth and have not presented any significant complications in terms of opening usability or ground support implementation.

Field characterization of fractures and joints (fractures without evidence of shear displacement) is included in Section 1.3.2.2 and summarized in Table 6-15. This characterization includes a map showing the location and trend of all known joint sets. For each joint set the areal distribution, attitude, length, and frequency are presented. Surface data, combined with subsurface data (Spengler et al., 1981; Maldonado and Koether, 1983; Scott and Castellanos, 1984; Spengler and Chornack, 1984), allow for reasonable estimates of the two- and three-dimensional characteristics and the variations of joints and fractures to be determined. The variations in the characteristics observed are related to the limited sampling available (vertical boreholes) and the fact that many of the fractures at the site are near vertical, and thus sampling may be biased. Also, there are very little data on the subsurface attitude of fractures. The potential combination of vertical and horizontal fractures has been considered in the design of ground support for underground openings.

6.1.2.4 Thermal properties

Thermal properties--thermal expansion coefficient, thermal conductivity, and thermal capacitance--are important in determining the time-varying extent of thermally induced stresses and displacements resulting from emplacement of radioactive waste. Theoretical considerations coupled with field measurements and analysis (Zimmerman, 1983; Blanford and Osnes, 1987) imply that, in general, laboratory-determined thermal properties may be directly applicable at the rock mass scale.

6.1.2.4.1 Thermal expansion coefficient

The design value and range for the thermal expansion coefficient for the rock mass is the same as that for intact rock for each thermal/mechanical unit. The design value is given as the mean of measurements for intact samples and the range is given in plus and minus one standard deviation to the mean (Table 6-16).

Uncertainties in the range are the result of sample-to-sample inhomogeneities. Also, calculated expansion coefficients are qualitatively consistent with measurements. The uncertainties are not likely to significantly affect the design because of the observed agreement between measured and calculated

Table 6-15. Recommended values for fracture frequency in thermal/mechanical units[a] at Yucca Mountain[b] (page 1 of 2)

Thermal/ mechanical unit	Fracture frequency (fractures per meter) at intervals of angles of inclination[c]																	
	0°-10°		10°-20°		20°-30°		30°-40°		40°-50°		50°-60°		60°-70°		70°-80°		80°-90°	
TCw[d]	0.40[e]	0.90[f] 0.05[g]	0.80	1.90 0.09	0.50	1.00 0.05	0.50	1.00 0.05	0.40	0.60 0.10	0.50	0.90 0.20	1.20	2.00 0.70	2.1	2.80 1.20	15.7	28.0 1.9
PTn	0.20	NA[h] NA	0.30	NA NA	0.20	NA NA	0.10	NA NA	0.20	NA NA	0.10	NA NA	0.40	NA NA	0.3	NA NA	2.8	NA NA
TSw1[d,i]	0.20	0.50 0.05	0.20	0.60 0.05	0.20	0.60 0.05	0.10	0.30 0.05	0.20	0.40 0.05	0.20	0.50 0.05	0.30	0.70 0.10	1.7	2.60 0.70	13.2	32.5 2.5
TSw2[d,i]	0.20	0.50 0.05	0.20	0.60 0.05	0.20	0.60 0.05	0.10	0.30 0.05	0.20	0.40 0.05	0.20	0.50 0.05	0.30	0.70 0.10	1.7	2.60 0.70	13.2	32.5 2.5
TSw3[d,i]	0.20	0.50 0.05	0.20	0.60 0.05	0.20	0.60 0.05	0.10	0.30 0.05	0.20	0.40 0.05	0.20	0.50 0.05	0.30	0.70 0.10	1.7	2.60 0.70	13.2	32.5 2.5
CHn1v[d]	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.08	0.09 0.05	0.5	1.50 0.05	0.8	1.2 0.1
CHn1z[d]	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.08	0.09 0.05	0.5	1.50 0.05	0.8	1.2 0.1
CHn2[d]	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.08	0.09 0.05	0.5	1.50 0.05	0.8	1.2 0.1
CHn3[d]	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.08	0.09 0.05	0.5	1.50 0.05	0.8	1.2 0.1
PPw[d]	0.05	0.05 0.05	0.10	0.20 0.05	0.08	0.10 0.05	0.10	0.20 0.05	0.08	0.10 0.05	0.10	0.20 0.05	0.30	0.50 0.05	0.6	0.70 0.50	1.0	1.7 0.2
CFUn	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.10	NA NA	0.01	NA NA	0.04	NA NA
BFw	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.06 0.05	0.20	0.40 0.05	0.30	0.50 0.05	0.60	1.30 0.05	1.4	3.80 0.10	6.5	18.9 0.2
CFMn1	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.1	NA NA	0.4	NA NA
CFMn2	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.1	NA NA	0.4	NA NA

Table 6-15. Recommended values for fracture frequency in thermal/mechanical units[a] at Yucca Mountain[b] (page 2 of 2)

Thermal/ mechanical unit	Fracture frequency (fractures per meter) at intervals of angles of inclination[c]																	
	0°-10°		10°-20°		20°-30°		30°-40°		40°-50°		50°-60°		60°-70°		70°-80°		80°-90°	
CFMn3	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.05	NA NA	0.1	NA NA	0.4	NA NA
TRw	0.07	0.09 0.05	0.06	0.06 0.05	0.05	0.06 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.05	0.05 0.05	0.30	0.50 0.05	0.2	0.20 0.20	3.9	7.0 0.7

[a]Thermal/mechanical units are defined on Figure 6-6.
[b]See Appendix O of SNL (1987). No specific design value was used at the onset of the conceptual design.
[c]Fracture frequencies were calculated as the average (arithmetic mean) of values from the four drillholes for which information is available. A mean value and a range are presented only for those units represented in at least two of the drillholes of Table 6 of Appendix O (SNL, 1987). Angles of inclination are presented in degrees measured downward from the horizontal.
[d]Represented in at least two drillholes.
[e]Mean value.
[f]Upper bound.
[g]Lower bound.
[h]NA = not available.
[i]Units TSw1, TSw2, and TSw3 are assumed to have the same fracture frequencies for this table.

Table 6-16. Thermal properties for intact rock and rock mass for each thermal/mechanical unit[a] at Yucca Mountain[b] (page 1 of 2)

Thermal/ mechanical unit	Thermal conductivity (W/mK)						Coefficient of thermal expansion ($10^{-6}K^{-1}$)									Thermal capacitance (J/cm^3 k)			
	Saturated[c]			Dry			Pretransition			Transition			Posttransition			Saturated[c]		Dry	
	Design value	Variability evaluation[d] mean	Variability evaluation[d] range	Design value	Variability evaluation mean	Variability evaluation range	Design value	Variability evaluation mean	Variability evaluation T(°C)	Design value	Variability evaluation mean	Variability evaluation (T°C)	Design value	Variability evaluation Mean	Variability evaluation T(°C)	Design value	Variability evaluation value	Design value	Variability evaluation value
TCw	2.00	1.84[f]	±0.12[f]	1.90	1.41[f]	±0.13[f]	10.7	8.8[f]	25-200[f]	NA[g]	NA	NA	NA	NA	NA	2.24	2.18	1.86	1.88
PTn	1.17	1.35	±0.06	1.02	1.02	±0.19	5.0	5.3[h]	25-150[h]	5.0	3.5	150-350	NA	NA	NA	2.59	2.24	1.09	1.09
TSw1[i]	2.07	2.07[i]	±0.16[i]	1.91	1.90[i]	±0.18[i]	10.7	8.8[i,f]	25-200[i,f]	31.8	31.8	200-350	NA	NA	NA	2.25	2.09[i]	1.88	1.98[i]
TSw1[j]	1.16	1.50[j]	NA	0.85	0.79[j]	NA	10.7	NA[j,]	25-200[j]	31.8	NA	NA	NA	NA	NA	1.88	1.87[j,k]	1.38	1.38[j,k]
TSw2	2.07	1.84	±0.12	1.91	1.41	±0.13	10.7	8.8	25-200	31.8	24.0	200-350	NA	NA	NA	2.25	2.16	1.88	2.17
TSw3	2.07	1.33	±0.08	1.91	1.34	±0.12	10.7	5.3	25-150	31.8	3.5	150-250	NA	NA	NA	2.25	2.04	1.88	2.45
CHnlv	1.21	1.35	±0.06	1.02	1.02	±0.19	5.0	5.3[h]	25-150[h]	5.0[h]	3.5[h]	150-250[h]	NA	NA	NA	2.46	2.61	1.24	1.26
CHnlz	1.35	1.48	±0.17	1.03	1.01	±0.14	6.7	6.7	25-T_b	-56.0	-56.0	T_b-150	-4.5	-4.5	150-300	2.46	2.61	1.37	1.36
CHn2	1.35	1.61	±0.04	1.03	1.21	±0.04	6.7	6.7[l]	25-T_b[l]	-56.0	-56.0[l]	T_b-150[l]	-4.5	-4.5[l]	150-300[l]	2.46	2.62	1.37	1.51
CHn3	1.35	1.43[m]	±0.03[m]	1.03	1.04[n]	±0.05[m]	6.7	6.7[l]	25-T_b[l]	-56.0	-56.0[l]	T_b-150[l]	-4.5	-4.5[l]	150-300[l]	2.46	2.66	1.37	1.30
PPw	2.00	2.00[n]	±0.27[n]	1.35	1.35[n]	±0.30[n]	8.3	8.3[n]	25-T_b[n]	-12.0	-12.0[n]	T_b-125[n]	10.9	10.9[n]	>125[n]	2.64	2.65	1.64	1.65
CFUn	1.43	1.43	±0.03	1.04	1.04	±0.05	6.7	6.7[l]	25-T_b[l]	-56.0	-56.0[l]	T_b-150[l]	-4.5	-4.5[l]	150-300[l]	2.67	2.68	1.43	1.43
BFw	2.00	2.00	±0.27	1.35	1.35	±0.30	8.3	8.3	25-T_b	-12.0	-12.0	T_b-125	10.9	10.9	>125	2.65	2.66	1.66	1.66
CFMn1	1.48	1.43	±0.00	1.13	1.11	±0.07	6.7	6.7[l]	25-T_b[l]	-56.0	-56.0[l]	T_b-150[l]	-4.5	-4.5[l]	150-300[l]	2.59	2.56	1.53	1.52
CFMn2	1.48	1.61[o]	±0.04[o]	1.13	1.21[o]	±0.04[o]	6.7	6.7[l]	25-T_b[l]	-56.0	-56.0[l]	T_b-150[l]	-4.5	-4.5[l]	150-300[l]	2.59	2.61	1.53	1.61
CFMn3	1.48	1.46	NA	1.13	1.11	NA	6.7	6.7[l]	25-T_b[l]	-56.0	-56.0[l]	T_b-150[l]	-4.5	-4.5[l]	150-300[l]	2.59	2.62	1.53	1.50
TRw	2.09	2.09	±0.18	1.79	1.79	±0.37	8.3	8.3[n]	25-T_b[n]	-12.0	-12.0[n]	T_b-125[n]	10.9	10.9[n]	>125[n]	2.57	2.58	1.79	1.79

Table 6-16. Thermal properties for intact rock and rock mass for each thermal/mechanical unit[a] at Yucca Mountain[b] (page 2 of 2)

Footnotes

[a]Thermal/mechanical units defined in Figure 6-6.
[b]See Appendix O of SNL (1987).
[c]Thermal conductivity data for all units and thermal capacitance data for PPw and underlying units are for a nominal saturation of 1.0, whereas thermal capacitance data for CHn3 and overlying units are calculated using saturations from Montazer and Wilson (1984).
[d]Variability evaluation represents more recent results of data analyses and establishes ranges for properties.
[e]Design values represent the basis for the Site Characterization Plan-Conceptual Design.
[f]Assumed to be the same as correlative property for TSw2.
[g]NA = not available.
[h]Assumed to be the same as correlative property for TSw3.
[i]Nonlithophysal layers in unit TSw1.
[j]Lithophysal layers in unit TSw1.
[k]For lithosphysal layers, the total porosity is $\phi_m \cdot M + \phi_A \cdot A + \phi_L$, where ϕ_m is matrix porosity, ϕ_A is the porosity of vapor-phase altered material, ϕ_L is the volume fraction lithophysal cavities, and M and A are volume fractions of matrix and vapor-phase-altered material, respectively (Price et al. 1985).
[l]Assumed to be the same as correlative property for CHnlz.
[m]Assumed to be the same as correlative property for CFUn.
[n]Assumed to be the same as correlative property for BFw.
[o]Assumed to be the same as correlative property for CHn2.

coefficients. Further, this parameter is not currently expected to vary beyond the range presented.

6.1.2.4.2 Thermal conductivity

The design value and range for the thermal conductivity for the rock mass is the same as that for the intact rock for each thermal/mechanical unit. The design value is given as the mean of measurements for intact samples, and the range presented is plus and minus one standard deviation from the mean. The conductivity at saturated conditions is calculated using methods described in Section 2.4.2.1.2. There is a difference between measured and calculated dry thermal conductivities because of mineralogic dehydration (Section 2.4.2.1.1). For certain units, there is a lack of experimental data. For these units the mean and range of units with similar physical and mineralogic properties are used.

Uncertainties in both the mean and the range for all units are the result of sample-to-sample inhomogeneities. The range captures the effect of the sample-to-sample inhomogeneities. For certain thermal/mechanical units (as noted), there is a lack of experimental data. For these units, the mean and range of units with similar mineralogic and bulk properties are used to derive values of thermal conductivity. In general, these units are either at some distance from the emplacement horizon or are not volumetrically significant in terms of accommodation of mechanical or thermal/mechanical loads. Thus, use of these derived conductivity values is considered legitimate for the far-field nature of the analyses. Further, variations in thermal conductivity for these distant, volumetrically insignificant units would not lead to conclusions different from those drawn already from far-field analyses.

6.1.2.4.3 Thermal capacitance

The design values for the thermal capacitance of the rock mass are assumed to be the mean of values calculated for intact samples, and the range presented is plus and minus one standard deviation from the mean (Table 6-16). No NNWSI Project measurements of thermal capacitance or specific heat have yet been made on tuffs. For the SCP-CD these parameters were calculated, assuming a constant heat capacity for the silicate mineral assemblage of 0.84 J/g °C, water heat capacity of 4.18 J/g °C, and air heat capacity much smaller (Tillerson and Nimick, 1984). Calculated values of the thermal capacitance (heat capacity/density product) show a broad range that depends on both porosity and degree of saturation. By considering reasonable variations in both the porosity and degree of saturation, the uncertainty in the range considered is small. This is because even with the most extreme ranges in saturation state, the variation in the volumetric heat capacity is small.

6.1.2.5 Hydrologic considerations

Hydrologic considerations, including the infiltration (water flux entry into the soil at the ground surface), percolation, (water flux through the rock units below the ground surface), hydraulic conductivity (capability of the rock to transmit water), and flood potential are factors important to design stability (Section 6.2.6.3), ventilation (Section 6.2.6.5), and sealing (Section 6.2.8). Infiltration, percolation, and hydraulic conductivity are discussed in the context of surface and ground water.

6.1.2.5.1 Surface water

The surface hydrology of the site influences the design of the surface facilities through both the location and flow frequencies of surface runoff. In addition, infiltration of surface runoff, which is a potential recharge source, influences the subsurface design. No perennial streams occur at or near Yucca Mountain. The only reliable sources of surface water are the springs in Oasis Valley, the Amargosa Desert, and Death Valley. Because of the extreme aridity of this region, where the annual precipitation averages about 20 percent of the potential evapotranspiration, most of the spring discharge travels only a short distance before evaporating or infiltrating back into the ground. Infiltration rates are low (<3 to 4 mm/yr) because of low precipitation, high runoff, and high evaporation rates.

Rapid runoff during heavy precipitation flows in the normally dry washes for brief periods of time. Local flooding can occur where the water exceeds the capacity of the channels. The potential for flooding at Yucca Mountain is described in detail in Section 6.1.2.6. In contrast to washes, the terminal playas may contain standing water for days or weeks after severe storms. Runoff from precipitation at Yucca Mountain drains into Fortymile Wash on the east and Crater Flat on the west, and both areas drain into the normally dry Amargosa River. If runoff is very high, water in the Amargosa River flows into the playa in southern Death Valley.

6.1.2.5.2 Ground water

Yucca Mountain lies within the Death Valley ground-water system, a large and diverse area in southern Nevada and adjacent parts of California composed of many mountain ranges and topographic basins that are hydraulically connected at depth. In general, ground water within the Death Valley system travels toward Death Valley, although much of it discharges before reaching Death Valley. Ground water in the Death Valley system does not enter neighboring ground-water systems.

The Death Valley ground-water system is divided into several ground-water basins. Apparently ground water moving beneath Yucca Mountain discharges at Alkali Flat and perhaps at Furnace Creek in Death Valley, but not in Ash Meadows or Oasis Valley. Yucca Mountain is in the Alkali Flat-Furnace Creek Ranch ground-water basin, at a position between the Ash Meadows and the Oasis Valley basins (Waddell, 1982).

Geologic formations in southern Nevada have been grouped into broad hydrogeologic units (Winograd and Thordarson, 1975; Montazer and Wilson, 1984; Peters et al., 1984; and Rush et al., 1984). Several of the units (aquifers) transmit water in sufficient quantities to supply water needs; whereas other units (aquitards) have relatively low permeabilities that tend to retard the flow of ground water. The geologic and hydrologic properties of the aquifers vary widely. The lower and upper carbonate aquifers and the welded tuff aquifers store and transmit water chiefly along the fractures. In contrast, the valley-fill alluvial aquifers store and transmit water chiefly through interstitial pore openings. The lower carbonate and valley-fill aquifer are the main sources of ground water in the eastern part of the NTS.

The unsaturated zone within the boundary of the primary repository area at Yucca Mountain is about 500 to 700 m thick. Within the site, the local water table slopes to the southeast and south, from an elevation of 800 m to as low as 730 m above sea level. The regional water table is 200 to 400 m below the horizon proposed for the emplacement area.

Most of the annual precipitation, approximately 150 mm (Montazer and Wilson, 1984), is returned to the atmosphere by evaporation and plant transpiration. A small part of the precipitation on Yucca Mountain percolates through the matrix of the unsaturated zone. Czarnecki (1985) calculated a recharge rate of about 0.5 mm/yr for the precipitation zone that includes Yucca Mountain. The principal source of recharge for the tuff aquifer is probably Pahute Mesa to the north and northwest of Yucca Mountain. The general direction of regional ground-water flow is south-southeast toward points of natural discharge at Alkali Flat and perhaps westward to Furnace Creek in Death Valley.

The potential repository horizon, the densely welded Topopah Spring Member, is located above the ground-water table. The in situ saturation of the Topopah Spring Member is estimated to be 65 ±19 percent and estimated percolation rates through this zone on the order of 0.5 mm/yr (Montazer and Wilson, 1984). The saturated hydraulic conductivity of the tuff rock mass (matrix plus fractures) is about 365,000 mm/yr (Sinnock et al., 1984, as derived from Thordarson, 1983). This value compares with a saturated value of approximately 0.6 mm/yr (Peters et al., 1984) for matrix flow through the Topopah Spring Member. It should be emphasized that the values of unsaturated conductivity depend on moisture content and are less than saturated conductivities. At 84 percent saturation, the upper bound, it is uncertain whether fracture flow is involved.

Thus, the magnitude of the saturated hydraulic conductivity of the rock mass is governed almost entirely by the presence of fractures. However, because the units within and above the proposed emplacement horizon are unsaturated, the in situ hydraulic conductivity of the tuff is significantly less than the saturated value.

Exploratory drilling at the site has not encountered any saturated zones above the water table that can be definitely identified as perched water, and it is not expected that any major perched water zones will be encountered. Some localized zones of saturation may exist within fault zones or beneath areas of high infiltration of surface runoff.

Ground water flowed from fractures at G-Tunnel (in Rainier Mesa) following drift excavation, but the flow eventually ceased (Thordarson, 1983). Similar localized flow from faults could occur within the prospective emplacement horizon and is being considered in design.

The depth to the carbonate aquifer beneath the primary repository area has not been determined, but can be inferred to be much more than the 1,250 m observed in drillhole UE-25p#1 located 2.5 km east of the primary area. At drillhole UE-25p#1, the hydraulic head in the carbonate rocks is 20 m higher than in the overlying tuffaceous rocks (Waddell et al., 1984). Because water cannot move in the direction of higher hydraulic head, it is concluded that ground water in the tuff aquifers beneath Yucca Mountain does not enter the carbonate aquifer.

Deep regional movement of ground water south and east of Yucca Mountain occurs chiefly through the lower carbonate aquifer. As a result of the complex geologic structures, flow paths are complex and poorly defined.

6.1.2.6 Flood characteristics

Because of the rugged terrain and meteorological conditions at the Yucca Mountain site, brief, but intense localized precipitation occurs periodically. In the vicinity of the site, Fortymile Wash and three of its principal tributaries, Yucca Wash, Drill Hole Wash, and Busted Butte Wash, have been analyzed for the 100-yr flood, 500-yr flood, and regional maximum flood (Squires and Young, 1984). Since it is shown that, for a 100-yr flood, water does not exceed the banks of the incised channels, the manifestations of a 50-yr flood are not presented. The flood zones in the surface facilities area considered in the flood analyses are shown in Figure 6-8 (Squires and Young, 1984). The flood history and potential are described in Section 3.2.1.

In the flood analyses performed thus far, except for the men-and materials-shaft area described in Section 6.2.4.2, maximum flood flows are derived from Crippen and Bue (1977), who present graphs of peak discharges versus drainage areas for measured historical floods with envelope curves above the plotted floods. The envelope curves represent the maximum potential flood for a given drainage area.

The flood flows for return periods of 100 to 500 yr are based on analysis of regional streamflow records at sites on the perimeter of the Nevada Test Site and Nellis Air Force Range, which are representative of the repository site. To determine the regional maximum flood, data were used from maximum flood flows that have been measured at other locations within the region, including all or parts of Nevada, California, Utah, Arizona, and New Mexico (Crippen and Bue, 1977). In this method, flood flows at the site are derived only on the basis of the area of the drainage basin. For example, the site-specific characteristics, such as ground slope, runoff and infiltration, are not represented.

In this preliminary analysis, it is concluded that the flood flows along Fortymile Wash would remain within the incised channel throughout the study

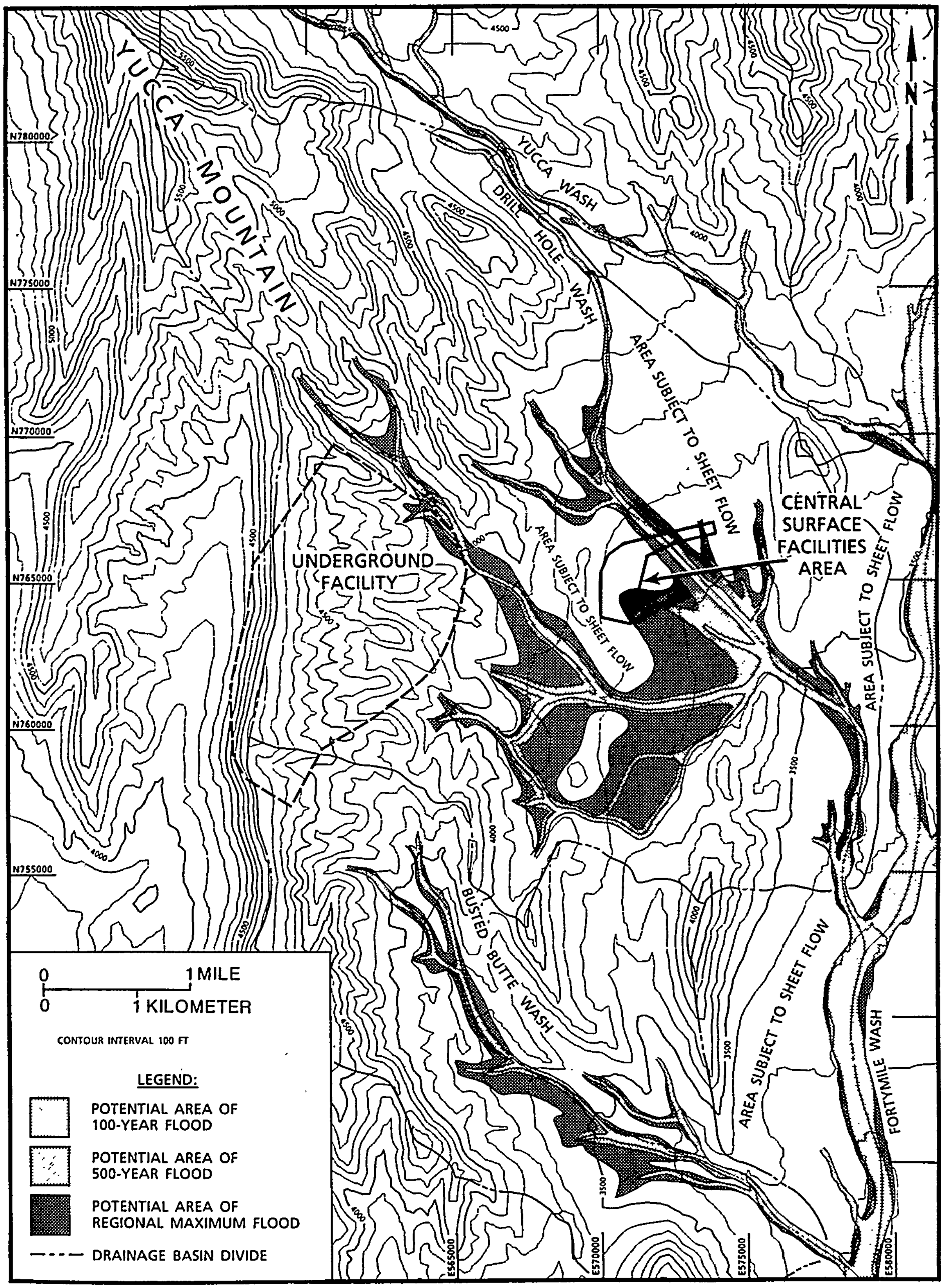


Figure 6-8. Site topography and flood potential areas. Modified from Squires and Young (1984).

area. In addition, at the Busted Butte and Drill Hole Wash drainages, the 500-yr flood would exceed stream-channel capacities at several places and the regional maximum flood would inundate sizable areas in the central parts of the watersheds. It is concluded that at Yucca Wash, flood flows of all three magnitudes would remain within the stream channel.

The later stages of design will be based on probable maximum flood (PMF) flows and levels, determined in accordance with ANSI/ANS 2.8 (1981). This method takes into account site-specific characteristics, including terrain, soil, and rock conditions of the drainage basin. This method is used by the U.S. Army Corps of Engineers for dam design and by the nuclear power industry for protection of safety-related facilities. The PMF method is more site-specific and severe than the regional maximum flood analyses.

In general, the surface facilities important to safety and underground entries at the repository site will be protected against the PMF by channels and dikes provided to divert the upland runoff and by setting finish grade elevations above the adjacent PMF levels. This design effort will be continued as more definitive topographic maps at selected locations and PMF calculations become available.

6.1.2.7 Seismic considerations pertinent to design

Evaluation of ground motion at the Yucca Mountain site must address two types of events: (1) natural seismicity (earthquakes) and (2) underground nuclear explosions (UNEs), which are conducted periodically at the Nevada Test Site (NTS). The seismic design criteria used for the SCP-CD is a vibratory ground motion input of 0.40g, developed on the basis of information contained in current documents (USGS, 1984; DOE, 1986c; URS/Blume, 1986) and seismologic and engineering judgment. This value may be revised and possibly increased as a result of ongoing studies, particularly the characterization of faults in the immediate vicinity of the site, for use in future design analysis. A seismic design criteria of 0.4g envelops the maximum ground acceleration expected from ground motion induced by a maximum yield UNE (700 kilotons) at the NTS, which is equal to 0.32g based on a mean value plus three standard deviations (DOE, 1986c).

The study supporting the SCP-CD used probabilistic methods to estimate ground motion (URS/Blume, 1986). This approach established a seismogenic zoning of the site region based on the history of the seismic events, late Quaternary strain rates, and the mode of later Cenozoic deformation in order to predict the ground motion hazard in the site region.

An occurrence model for UNEs was also established in this study, using historical data of NTS testing that occurred before the Threshold Test Ban Treaty. Probable future testing that would be nearest to Yucca Mountain will occur in the Buckboard Mesa area. This area, which is approximately 15 miles from the repository site, is closer to the repository site than any of the locations on the NTS where testing actually occurs. A maximum yield of 700 kilotons was established for a UNE at Buckboard Mesa in order to avoid significant damage in the surrounding region. The current testing limit of

150 kilotons, established by the Threshold Test Ban Treaty, produces negligible ground motion at the repository site.

The ground motions used in the SCP-CD are (1) an acceleration value of 0.4g with a return period of 2,000 yr for natural earthquakes and (2) an acceleration value of 0.15g based on the mean of observed responses plus 2 standard deviations for UNEs (Table 6-17).

In an additional study that is in process, faults in the vicinity of the site are assumed to be active. Fault-specific, random earthquake-occurrence models have been developed to predict the hazards caused by ground motion and fault displacements. The earthquake occurrence was determined from published fault-length and slip-rate information. The faults considered include the Bow Ridge, Paintbrush, Ghost Dance, Midway Valley, and Severe Wash faults.

Technology exists for designing surface facilities for accelerations much larger than those described previously; typical examples include the Diablo Canyon nuclear power plant and San Onofre nuclear power plant. Published literature also shows that structures can be designed to resist moderate surface displacement before any catastrophic failure could occur (Reed et al., 1979).

It is provisionally assumed that the peak accelerations at the emplacement level are half those at the surface. This assumption is based on an attenuation of ground motion with depth derived from UNE test data and other published information on earthquakes (Carpenter and Chung, 1985; URS/Blume, 1986). Carpenter and Chung (1985) indicate that up to surface-shaking levels (0.5g), no tunnel collapses have been observed because of shaking alone. They also point out that tunnels in poor soil and rock are more susceptible to damage than are tunnels deep in rock and that damage to all classes of deep tunnels consisted primarily of minor rockfalls and formation of new cracks, except where active faults intersected tunnels. In these instances, although severe damage occurred, the damage is localized and is readily repaired using existing technology. Hence, it can be seen that the use of current technology permits designing underground facilities that can withstand the levels of acceleration described above. The topic of borehole stability during a seismic event was not addressed as part of the SCP-CD effort. This topic will be addressed in future design activities after the impacts, if any, of borehole collapse on containment, isolation, and retrievability have been assessed.

Plans for continued work to identify design values for ground motion and surface rupture for use in more advanced design phases are discussed further in Section 8.3.1.17.

6.1.2.8 Dust characteristics

Dust characteristics, including particle size distribution, composition, and mass concentration, are important for determination and maintenance of acceptable air quality. Such information is needed for the detailed design of the ventilation systems and the surface aspects of the muck pile. Detailed calculations of dust concentration have not been performed for the

Table 6-17. Peak ground accelerations at the surface used in conceptual design[a] (SNL, 1987)

Seismic event	Acceleration (g) Horizontal	Acceleration (g) Vertical	Return period (yr)
Design earthquake	0.40	0.27	2,000
Design underground explosion	0.15	0.18	NA[b]

[a]URS/Blume, 1986.
[b]NA = not applicable.

conceptual design. Dust composition is expected to be similar to that of the rock being excavated. Data on particle size distribution and mass concentration are currently lacking.

6.1.3 ANALYTICAL TOOLS FOR GEOTECHNICAL DESIGN

The process of developing and analyzing a geotechnical design for a repository, including identifying and resolving analytical design problems, is accomplished through application of the issue resolution strategy (IRS) to the design issues identified in Section 6.4. The IRS methodology has been adopted by the DOE and is presented in detail in Section 8.1.2. As applied to the design issues, the IRS includes the use of analytical tools (or methods) and techniques in (1) identifying specific problems involved in resolving an issue; (2) separating these design problems into their component parts; (3) determining the functions, processes, performance measures, performance goals, and confidence required in meeting the performance goals; (4) selecting an empirical or numerical solution method that can be used to judge whether the goals are met; (5) identifying the parameters needed to use the selected solution method; (6) establishing the ranges and confidence levels required for these parameters; (7) obtaining these parameters from site characterization activities or other sources; and (8) using the selected solution method and the parameters to judge whether the performance goals will be met.

Analyses are presented in Section 6.4 for nine design issues. In the analytical approach subsection of Section 6.4.2.2, the analytical methods (tools) used to address specific design issues are described. The computer-aided numerical methods (codes) used during the issue resolution process are identified, and the following information is stated for each code:

1. Code name.
2. Author.

3. Ownership.
4. Design area for which the code was used.
5. A description of the calculations that the code performs.

6.1.4 STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY

Title 10 CFR 60, Subpart G, Quality Assurance, requires that the DOE apply a quality assurance (QA) program to structures, systems, and components important to safety. This QA program must also be applied to items important to waste isolation. Items (structures, systems, and components) important to safety or waste isolation are placed on a Q-list. A Q-list is a convenient way to call attention to items that require the 10 CFR 60, Subpart G, QA program. This section discusses the identification of items important to safety; items important to waste isolation are the subject of Section 6.1.5.

Items important to safety are defined as "those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure" (10 CFR 60.2). The NRC and the DOE have advocated the use of probabilistic risk assessment (PRA) techniques to determine the items important to safety. A method was developed based on a preliminary radiological safety analysis (PRSA) given in Appendix F of the SCP-CDR, which used a PRA approach. A preliminary identification of items important to safety has been performed using this method. A detailed report of the study to identify items important to safety can be found in Appendix L of the SCP-CDR. In the following discussion, a brief description of the radiological safety analysis method developed to identify items important to safety will be presented. The method is also illustrated in Figure 6-9. The results of the study will also be presented in the form of a preliminary list of items important to safety.

The PRSA method employed in Appendix F of the SCP-CDR basically follows the NRC methodology for a simplified and streamlined level 3 PRA described in the PRA Procedures Guide (NRC, 1983). The level of detail of the PRSA varies at each step, depending on the data and design information currently available. Since the primary objective of the PRSA was to provide a numerical basis for the development of a preliminary list of items important to safety, only accident scenarios resulting in public exposures were considered in detail.

After developing facility and system models, both internal and external initiating events were identified and screened by a panel of experienced design and safety analysis engineers. The basis for the screening was the potential of an event to contribute to a significant offsite release of radioactive materials. Using the event-tree technique, accident scenarios were then developed for those initiating events surviving the first screening process. Event trees are graphic depictions of the sequence of events that occurs following an initiating event. The construction of an event tree is an inductive process in that one goes from the specific (i.e., the initiating event) to the general (i.e., all the possible results of the initiating

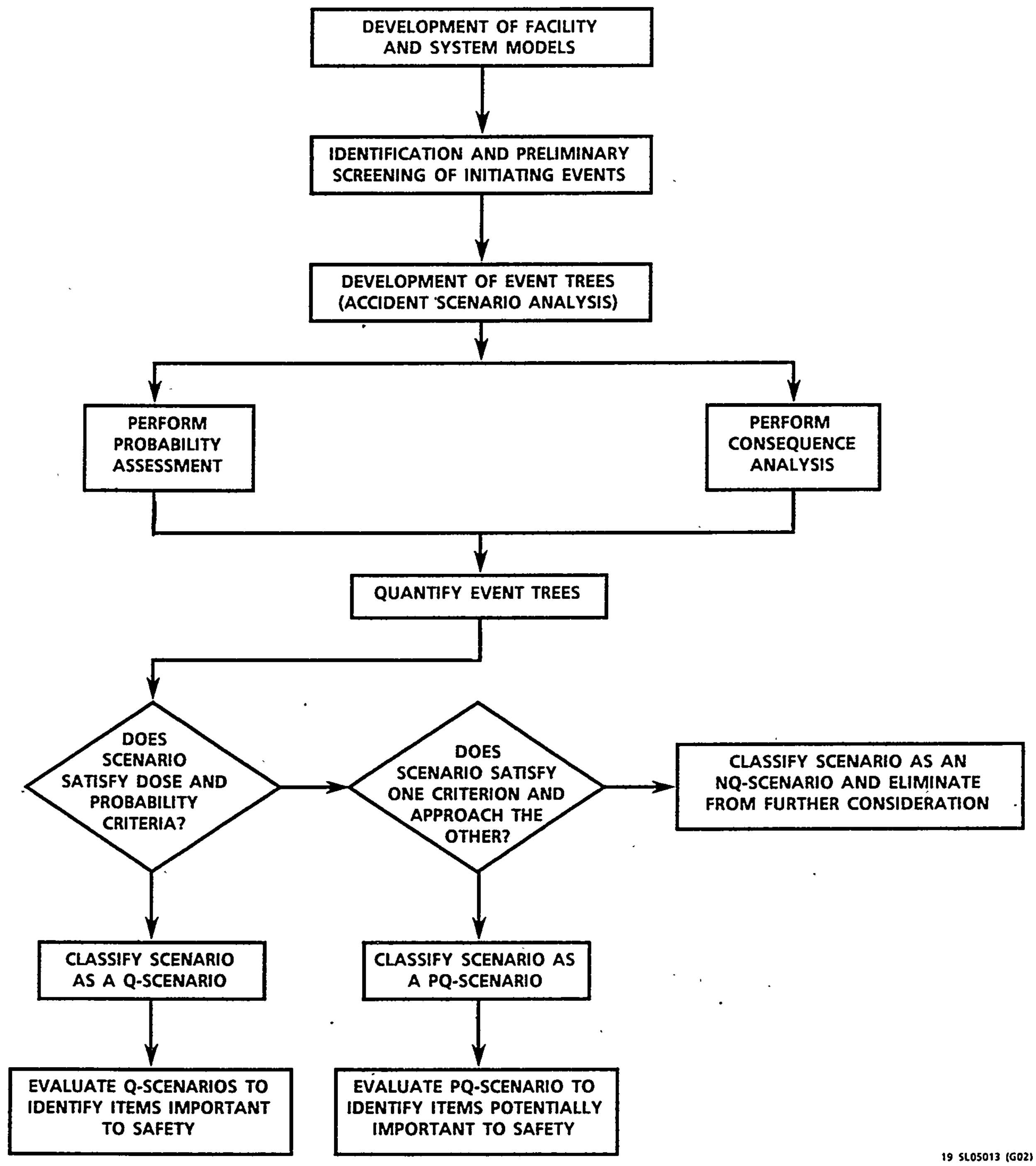


Figure 6-9. Q-List methodology for items important to safety.

event). The key factor for developing an event tree for each surviving initiating event was the selection and definition of the intermediate events. Event trees were constructed in detail appropriate to the level of design detail available and as necessary to adequately characterize the accident. Because of lack of data and design details, fault trees were not completely developed and analyzed; however, variations of conventional fault tree or fault diagrams were developed for most intermediate events. Fault trees are graphic depictions of the possible events that might lead to an intermediate event on an event tree. Constructing a fault tree is a deductive process in that one goes from the general--all possible ways for the intermediate event to come about--to the specific, the intermediate event. The use of fault diagrams provides important insight into the probabilities of intermediate events.

After event trees were developed, the probability of each initiating event and each intermediate event was evaluated, as were the consequences of accident scenarios. The probability and consequence analyses were performed in parallel. Both historical data and the judgment of a panel of engineers experienced in safety analyses were used in estimating probabilities. Consequence analyses involved the development of models and estimates of radionuclide releases, dispersion, and transport into the environment as well as calculation of doses. The results of the probability and consequence analyses were used to quantify the event trees. Briefly, the event trees are quantified by assigning probabilities to each intermediate event and consequences to each branch (or accident scenario) of the event tree.

On the basis of the results of the event tree quantifications, all accident scenarios that resulted in either dose consequences of more than 0.05 rem at the site boundary and probabilities of more than 10^{-9} per yr were selected as reference accident scenarios. The reference accident scenarios were identified by simplifying, or pruning, the event trees of all the accident scenarios that did not fall within the limitations established for the dose consequences and probability criteria. The initial list of items important to safety was derived from the reference scenarios and the numerical results of the analyses.

The reference accident scenarios were developed using the physical systems described in Section 6.2. Ramp access for waste emplacement operation, when compared to shaft access, substantially reduced the number of accident scenarios that needed to be considered. The ramp access to the underground area allows the use of a single transport cask permanently mounted on a transport vehicle to

1. Collect the waste container from the surface storage vault using the collection/emplacement mechanism contained within the transport cask;

2. Transport the waste container to the underground area by way of the access ramp;

3. Transport the waste container to the waste emplacement borehole by way of the main entry, panel access, and emplacement drifts; and

4. Emplace the waste container in the emplacement borehole using the collection/emplacement mechanism contained within the transport cask.

The use of the ramp access thus eliminates two waste container transfer operations that are normally associated with shaft access to the underground. The transfers that are eliminated are

1. Transfer of the waste container from the surface storage vault into the shaft transfer cask and

2. Transfer of the waste container from the shaft transfer cask into the waste emplacement transporter.

Ten reference scenarios associated with waste emplacement operations were identified that might lead to a release of radioactive materials from the repository facilities. These scenarios are described in the following table.

Event	Description
SURFACE STORAGE VAULT	
Container transfer mechanism (CTM) failure	During transfer an equipment failure occurs and the container is dropped, possibly causing a breach. All waste forms are considered in the scenarios.
Shielded underground transporter collision	The transporter inadvertently hits the the CTM or runs into facility wall. The CTM or transporter is carrying a a container and a breach occurs.
Shielded underground transporter moves	A transporter moves inadvertently during waste loading and a container breach occurs due to shearing.
UNDERGROUND AND EMPLACEMENT AREA	
Waste transporter coasts down ramp (run-away transporter)	Mechanical failure causes a transporter to coast from top of ramp and strike the ramp wall, particularly near the bottom where the ramp is curved. Cask breach and/or fuel ignition in the transporter is possible.

Event	Description
Transporters collide on ramp	A transporter inadvertently travels up the ramp while a second transporter is going down resulting in a head-on collision and a fire and/or an explosion. Breach of cask is possible.
Container drops into emplacement hole (vertical emplacement mode)	A transporter grapple fails or a container pintle fails during emplacement.
Unwanted movement of transporter	A transporter inadvertently moves during emplacement resulting in a shearing force and container breach.
Secondary vehicle has collision with loaded transporter	A small secondary vehicle has collision with a transporter during an emplacement operation resulting in possible container breach and secondary fire in the transporter.
Exhaust filter building fire	A fire occurs due to an electrical short or a worker accident, and HEPA filters and/or equipment for ventilation are damaged.
Loss of HEPA filter system	The radiation monitor may not activate the HEPA bypass system, or equipment failure may occur causing normal releases or accident releases given a common mode or coincidence failure.

Reference accident scenarios that could potentially lead to significant offsite releases of radioactive material and dose consequences were developed using the previously described method. The following two criteria were used to screen the reference accident scenarios for scenarios that could lead to the identification of items important to safety:

1. Dose criterion: An accident scenario could potentially lead to the identification of items important to safety if the calculated off-site public dose was greater than or equal to 0.5 rem; otherwise, the accident scenario is not significant with respect to items important to safety.

2. Probability criterion: An accident scenario could potentially lead to the identification of items important to safety if the probability of occurrence of the scenario is greater than 10^{-5} per yr; otherwise the scenario is not considered significant with respect to items important to safety.

In performing this second screening, the probability, including its uncertainty, were compared with the above criteria. If, and only if, an accident scenario passes both screening criteria, the accident scenario is classified as a Q scenario. Q scenarios then are further analyzed to determine which of the structures, systems, or components involved in the scenario are important to safety. A structure, system, or component is important to safety if it is essential to either the prevention of the scenario or the mitigation of the scenario dose consequence.

Scenarios that are not significant with respect to items important to safety are classified as either a non-Q scenario (NQ scenario) or a potential-Q scenario (PQ scenario). All NQ scenarios are eliminated from further consideration in identifying items important to safety.

Any scenario not immediately identified as a Q scenario, but which, as further study and design take place, is judged to have a reasonable potential to be upgraded to a Q scenario is classified as a PQ scenario. Two criteria were used to decide between PQ and NQ. First, a scenario was classified as a PQ scenario even if no analyses had been performed if the item or scenario was sufficiently similar to others historically classified as PQ scenarios or when practical consideration indicated it could be a Q scenario. Second, if the analysis determined that either the consequences or probability exceeded the criteria and the other was sufficiently close that a change in assumptions or data could cause the criteria to be exceeded, the scenario was classified as a PQ scenario. A variation of this second criteria was that when both consequence and probability were below the threshold but sufficiently close that a change in assumption or data could move it over, it was classified as a PQ scenario.

Once a scenario is classified as a Q scenario or a PQ scenario, that scenario is further analyzed to determine which of the items involved in the scenario should be placed on the list of items important to safety or potentially important to safety. Further analysis of the scenario involves the evaluation of the systems, structures, and components involved in the scenario to determine what role the item plays in the scenario. Items whose failure causes the loss of consequence mitigation processes or whose failure directly causes the release of radioactive materials are classified as important to safety or potentially important to safety and placed on the Q list or PQ list, depending on which type of scenario is being evaluated. Current plans will make these items PQ items, as well as items important to safety, subject to a QA level I program that satisfies the requirements of 10 CFR 60, Subpart G. A potentially-important-to-safety list is consistent with DOE guidance (DOE, 1987c).

The results to date have not identified any Q scenarios, or consequently, any Q-list items. However, this result is based on incomplete and preliminary data and design. For example, an airplane-crash scenario was not based on actual data and will have to be reexamined. Consequently, all items that have been classified as potential Q-list items will be treated as if they were Q-listed during future design until the design detail and available data support a definitive analysis and conclusion.

The preliminary PQ list is presented in Table 6-18. For a complete discussion of the methods and analyses used in classifying the items listed in Table 6-18, the reader is referred to the SCP-CDR (Appendices F and L of SNL, 1987). The work reported in these appendices includes the effects of mitigative features (i.e., radiation alarms and filtration systems) in some of the accident scenarios. Since the time these analyses were conducted, however, a decision has been made not to use mitigative features in the Q-list analyses. As a result, the analyses presented in the two appendices were re-examined to remove any reductions in radiological dose consequence or probability of occurrence that accrued from the mitigative features. This re-examination resulted in the PQ list given in Table 6-18. No Q-list items resulted from this re-examination.

In addition, as the design is developed, i.e., the reference configuration of the LAD and additional data become available, the complete sequence of the Q-list method will be implemented again to refine, correct, and validate the initial results. A detailed discussion of the methods used in determining items important to safety is given in Appendix F of the SCP-CDR, and the results of the analysis of items important to safety is given in Appendix L.

6.1.5 BARRIERS IMPORTANT TO WASTE ISOLATION

Barriers important to waste isolation are defined by the DOE (DOE, 1987b) as the barriers, structures, systems, and components that are relied on to achieve the postclosure performance objectives in 10 CFR 60, Subpart E. The engineered barriers that meet this definition are placed on the Q-list. The natural barriers that meet this definition are not placed on the Q-list, because they cannot be designed. Instead, their ability to isolate the waste is given special protection through an "activities list," which contains all the activities that might adversely affect the natural barriers and for which design criteria are not meaningful.

The identification of barriers important to waste isolation is accomplished through the performance-allocation process. Barriers at the Yucca Mountain site that satisfy the definition have therefore been identified by examining the performance allocations in Chapter 8 of this document. Each of the four postclosure performance objectives is represented by an issue in the issues hierarchy and a corresponding section in Chapter 8. In that section is a performance allocation, which selects the barriers that the DOE currently expects to rely on for demonstrating, in the license application, that the performance objective will be met. The engineered barriers named in the allocation are placed on the Q-list; the natural barriers receive protection through the activities list.

The first performance objective in 10 CFR 60.112 deals with the allowable releases of radioactivity from the repository to the accessible environment. Section 8.3.5.13, which treats this performance objective as

Table 6-18. Potential Q-list for items important to safety at the Yucca Mountain repository

Items	Locations	Initiating events
Crane, shipping cask	Cask receiving and preparation area	Crane drops a shipping cask
Hot cell structure	Packaging hot cell	Earthquake causes hot cell structure failure
Crane	Unloading hot cell Consolidation hot cell Packaging hot cell	Earthquake causes crane to drop on fuel assemblies
Vehicle stop	Cask receiving and preparation area	Vehicle with cask falls in cask preparation pit (detailed analysis not performed)
Fire protection system	Waste-handling building	Fire involving radioactive material is a dispersion promoter (detailed analysis not yet performed)
Cask transfer mechanism (CTM)	Surface storage vault	CTM drops container with consolidated fuel rods
Transport cask	Underground facility and ramp	Transporter coasts down the waste ramp and strikes the wall of the ramp or main access drift

Issue 1.1, describes the plans for demonstrating that this performance objective will be met. The performance allocation for Issue 1.1 relies largely on natural barriers: the saturated and unsaturated zones. The primary reliance is on the unsaturated zone; the principal unsaturated zone rock units in this allocation are the Calico Hills nonwelded zeolitic unit and the Calico Hills nonwelded vitric unit. The waste package, an engineered barrier, is relied on as a primary barrier only for releases of gaseous radionuclides. From these allocations, the waste package would be proposed for inclusion on the Q-list. The waste package, however, consists of two subelements: the waste container and the waste form inside the container.

The waste form does not appear on the Q-list, because it will not be engineered as part of the repository design. The waste container is therefore proposed for inclusion on the Q-list of items important to isolation. The proposed activities list includes the activities that have a potential for adversely affecting the waste-isolation capabilities of the Calico Hills nonwelded zeolitic unit, the Calico Hills nonwelded vitric unit, and the saturated zone.

The second performance objective in 10 CFR 60.113 deals with the time during which the waste package must provide substantially complete containment of the high-level waste. Section 8.3.5.9, which treats this performance objective as Issue 1.4, allocates performance to the emplacement environment of the waste package, which is the Topopah Spring welded unit in the immediate vicinity of the emplaced waste; to the waste container; and, to the waste form inside the container. This allocation suggests that the waste container should be placed on the Q-list of items important to waste isolation. For the reason given above, the waste form does not appear on the Q-list. Activities that have the potential for adversely affecting the waste-isolation capabilities of the Topopah Spring welded unit are placed on the activities list.

The third performance objective in 10 CFR 60.113 deals with the allowed releases from the engineered-barrier system. Section 8.3.5.10, which treats this performance objective as Issue 1.5, allocates performance to the emplacement environment of the waste package, which is the Topopah Spring welded unit in the immediate vicinity of the emplaced waste, and to the waste form. This allocation suggests no additions to the Q list or the activities list beyond those suggested by the first two performance objectives.

The fourth performance objective in 10 CFR 60.113 deals with the required ground-water travel time at the repository site. Section 8.3.5.12, which treats this performance objective as Issue 1.6, allocates primary performance to the Calico Hills nonwelded zeolitic unit and the Calico Hills nonwelded vitric unit. It allocates secondary performance to the Topopah Spring welded unit and to the saturated zone. Although some allocation is made to other units, the reliance on them is merely "auxiliary." The allocation in Section 8.3.5.12 suggests no additions to the Q-list or the activities list beyond those suggested by the first two performance objectives.

In summary, the proposed Q-list for items important to waste isolation contains the waste container. The proposed activities list includes activities that have the potential for adversely affecting the waste-isolation capabilities of the Topopah Spring welded unit, the Calico Hills nonwelded zeolitic unit, the Calico Hills nonwelded vitric unit, and the saturated zone.

6.2 CURRENT REPOSITORY DESIGN DESCRIPTION

This section summarizes the current repository conceptual design. The design information reflects current design concepts being considered for the Yucca Mountain repository site. These concepts include both the vertical, which is the reference configuration, and horizontal emplacement configurations. The design descriptions make reference to design documents and focus on design features that are influenced by site characteristics. Where uncertainties in site or other SCP-related design parameters are identified, plans for bounding design parameters or for performing preliminary sensitivity analyses are referenced.

6.2.1 BACKGROUND

Preliminary designs for the potential Yucca Mountain repository were schematic and had sufficient detail to formulate concepts to address feasibility and to support development of criteria for the generation of the present repository conceptual design. The design effort was divided into four distinct areas: (1) surface repository design, (2) underground repository design, (3) special waste emplacement and retrieval equipment design, and (4) design of the waste emplacement envelope. The waste package design is described in Chapter 7.

The Site Characterization Plan-Conceptual Design Report (SNL, 1987) elaborates on the repository design described herein. (This report is referred to as the SCP-CDR throughout Section 6.2.) The SCP-CDR provides a detailed description of the repository conceptual design, including discussions concerning the design methods.

The conceptual design of the surface and underground facilities depicted in this document represents the status of the work completed in May 1986. These efforts are summarized herein and include the status of work completed to date on the design for the capability of receiving, processing, and emplacing of spent fuel waste and defense high-level waste (DHLW). The effort reflects the waste delivery rates identified in Section 6.1.1 (repository design requirements), characteristics, and throughput rates resulting from new waste package designs. It is important for the reader to recognize that the repository design presented in the SCP is conceptual and will be refined as a result of site characterization activities, described in Section 8.3, and the completion of subsequent phases of repository design. Subsequent design phases include the advanced conceptual design (ACD), license application design (LAD), and final procurement construction design (FPCD).

6.2.2 OVERALL FACILITY DESIGN

The proposed repository site at Yucca Mountain is in southern Nevada, about 137 km (85 mi) by air and 161 km (100 mi) by road northwest of Las Vegas. The proposed site is located on Federal land currently under the separate control of the DOE, the Bureau of Land Management, and the U.S. Air Force.

The currently proposed highway and rail access routes to the site are shown in Figure 6-10. For the purpose of conceptual design a new access road is proposed to originate at U.S. Highway 95 approximately 0.8 km west of the town of Amargosa Valley and to extend about 27 km northward to the site. Likewise, a new railroad is proposed to originate at Dike Siding, about 18 km northeast of Las Vegas, and extend about 137 km to the site. A new bridge, or bridges, crossing Fortymile Wash would be necessary for highway and rail access to the site.

An illustration of the current repository design concept is presented in Figure 6-11. The proposed repository complex is composed of surface and underground facilities linked by a combination of shafts and ramps. Figure 6-12 shows the overall site plan of the current design, including the surface facilities, and the shafts. The location of site characterization boreholes is discussed in Section 8.3.1.4.1.

The underground facilities would be located below the ridgeline of Yucca Mountain within the Topopah Spring Member of the Paintbrush Tuff. Both vertical (Figure 6-13) and horizontal (Figure 6-14) emplacement configurations are discussed. Details pertaining to the current design of these configurations and the integration of the exploratory-shaft facility with the underground facilities are discussed in Sections 6.2.5 and 6.2.6.

The main or central surface facilities are proposed to be built on gently sloping terrain at the eastern base of Yucca Mountain. The proposed repository location was selected based on a preliminary assessment of soils, topography, and the need for an efficient interface with the ramps and shafts that provide access to the underground facilities (Neal, 1985). The main surface facilities would be segregated into three adjacent areas: (1) the waste-receiving and inspection area, (2) the waste operations area, (3) and the general support facilities area. The waste operations area would include the waste-handling buildings and other facilities where radioactive material would be handled. A ramp would be used for transporting waste from the surface to the underground disposal area. Another ramp would be used for conveying mined tuff to the surface. Four vertical shafts (two exploratory shafts, an exhaust shaft, and a men-and-materials shaft) would be located near the northeast boundary of the underground disposal area and would be used for underground ventilation and access for personnel, supplies, and equipment.

The Johnstone et al. (1984) report discusses the ranking of the rock strata at Yucca Mountain that is best suited for the underground repository. The host rock for the proposed underground repository is located within a thick unsaturated zone below Yucca Mountain. This unit is the welded, ash-flow tuff portion of the Topopah Spring Member of the Paintbrush Tuff Formation (Section 6.1.2).

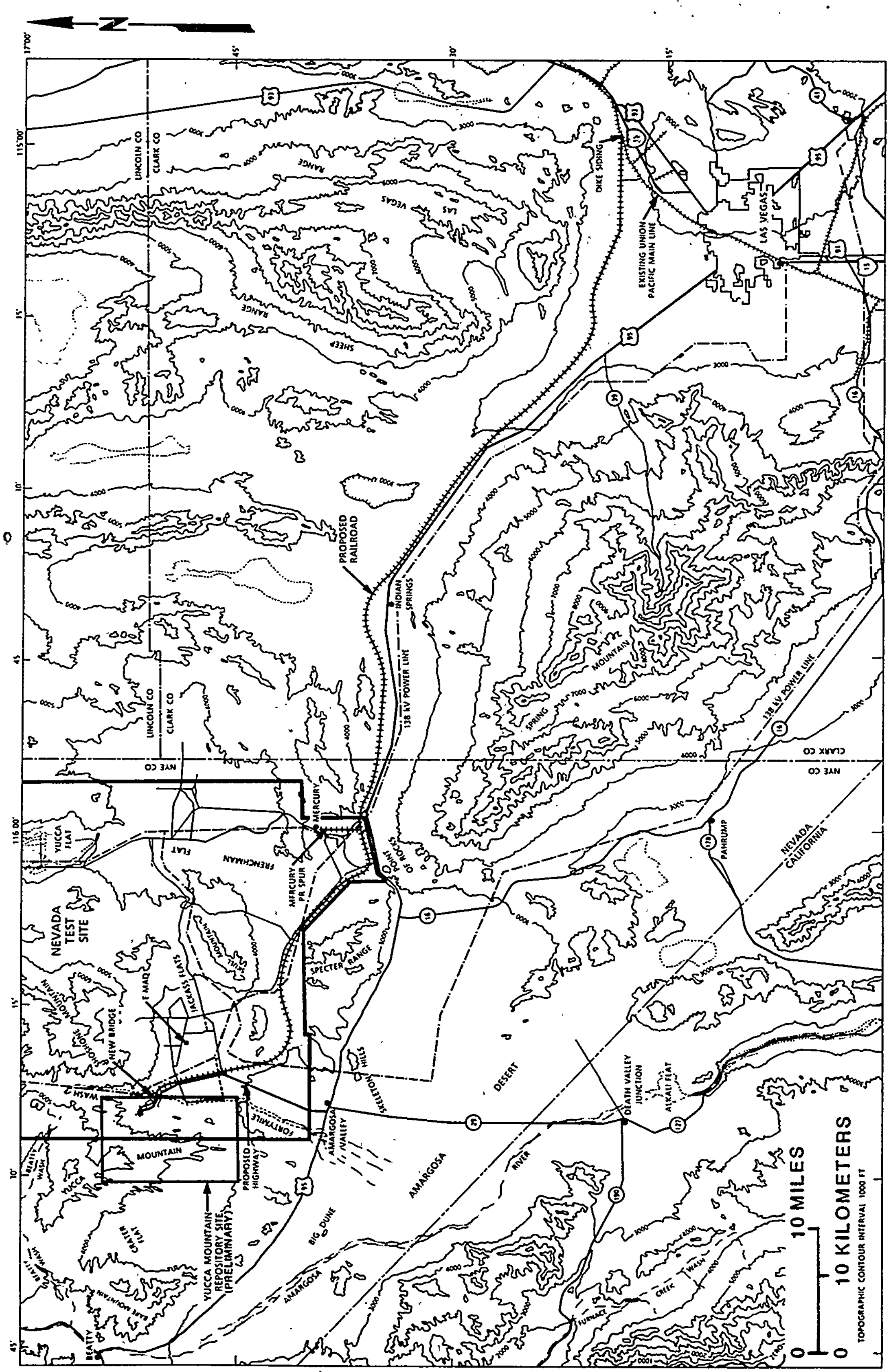


Figure 6-10. Highway and rail access routes for proposed Yucca Mountain repository.

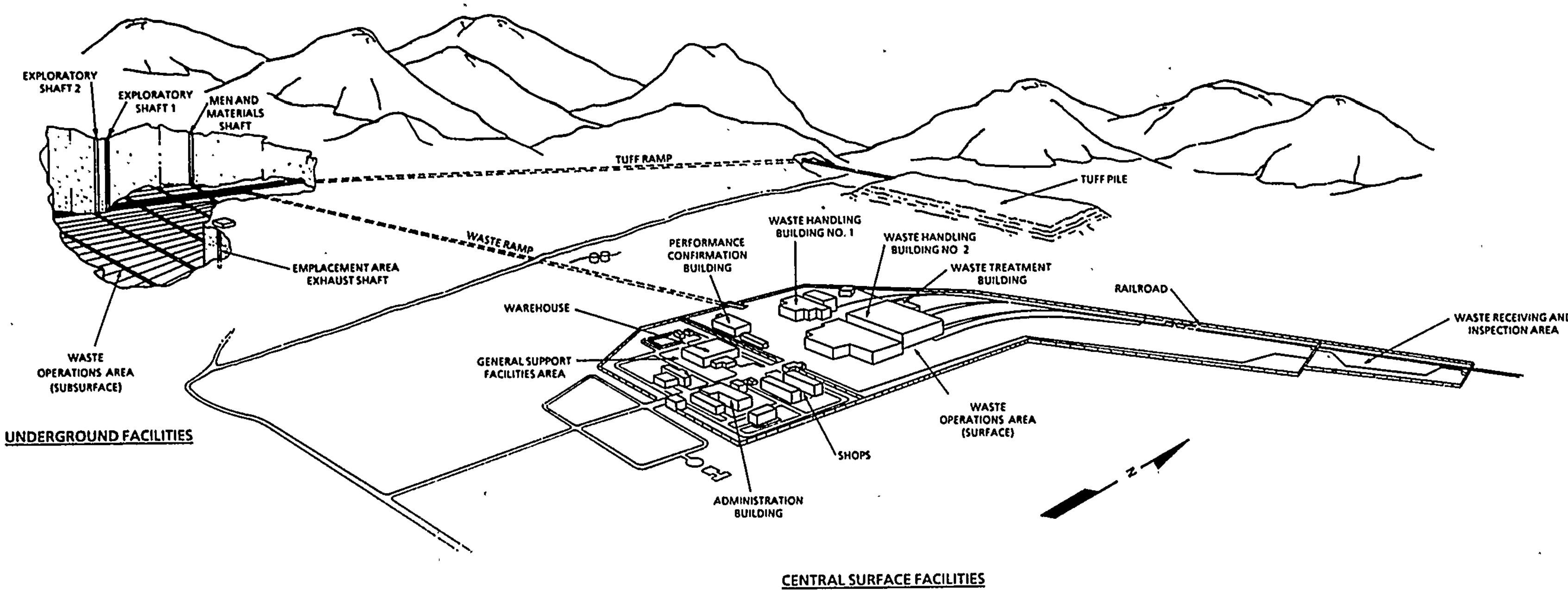


Figure 6-11. Perspective of the proposed Yucca Mountain repository.

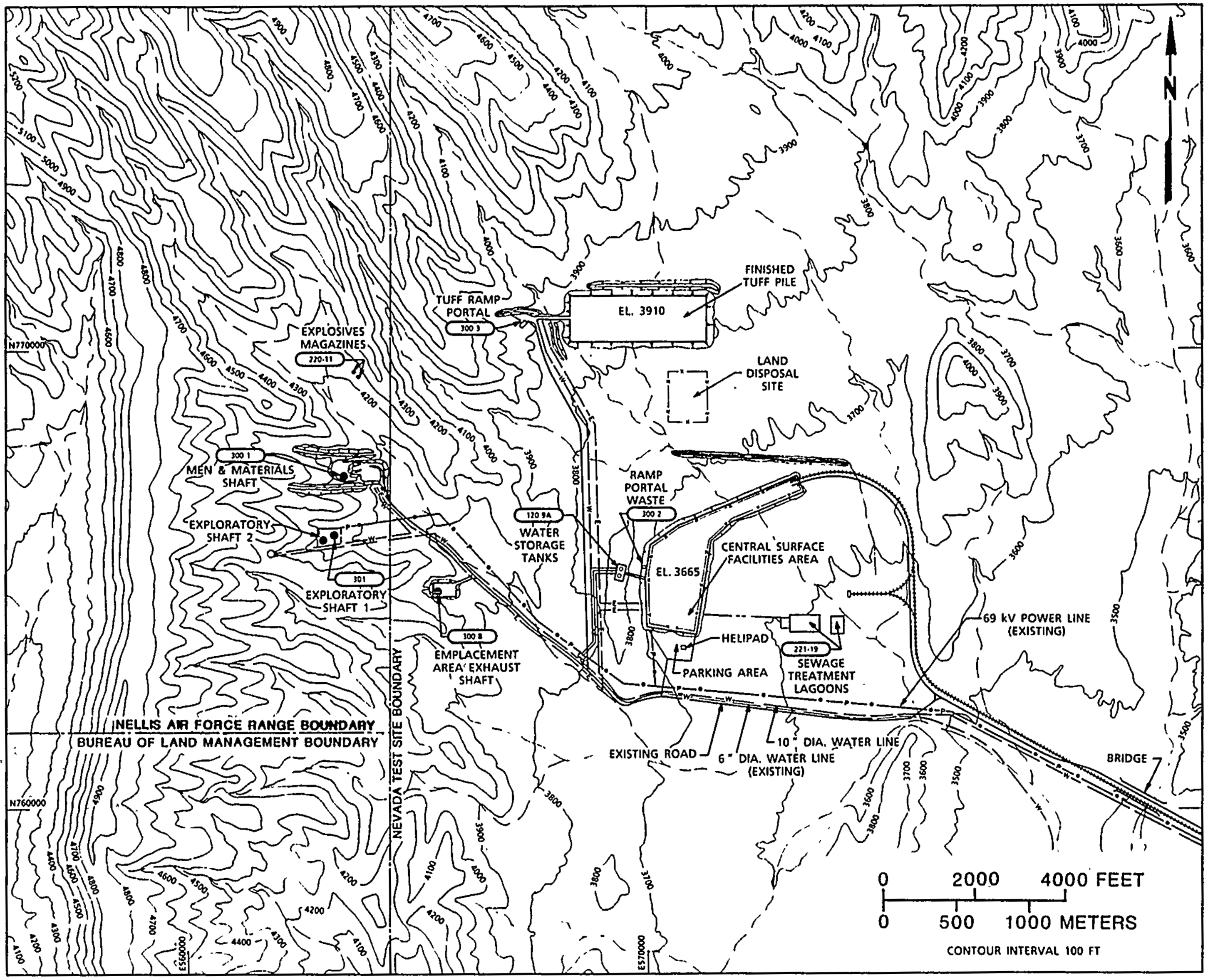


Figure 6-12. Overall site plan showing surface facilities and shafts.

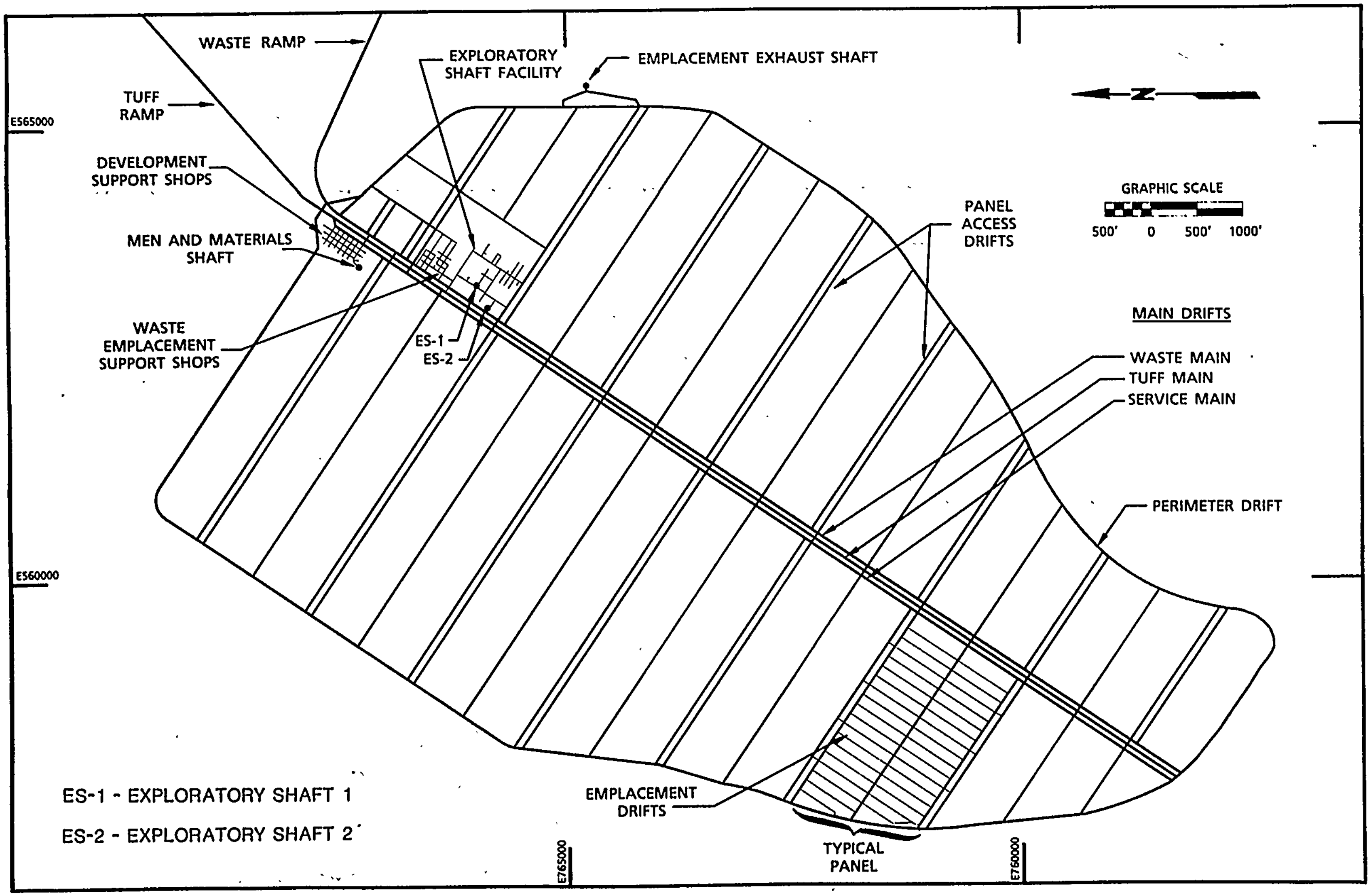


Figure 6-13. Vertical emplacement configuration.

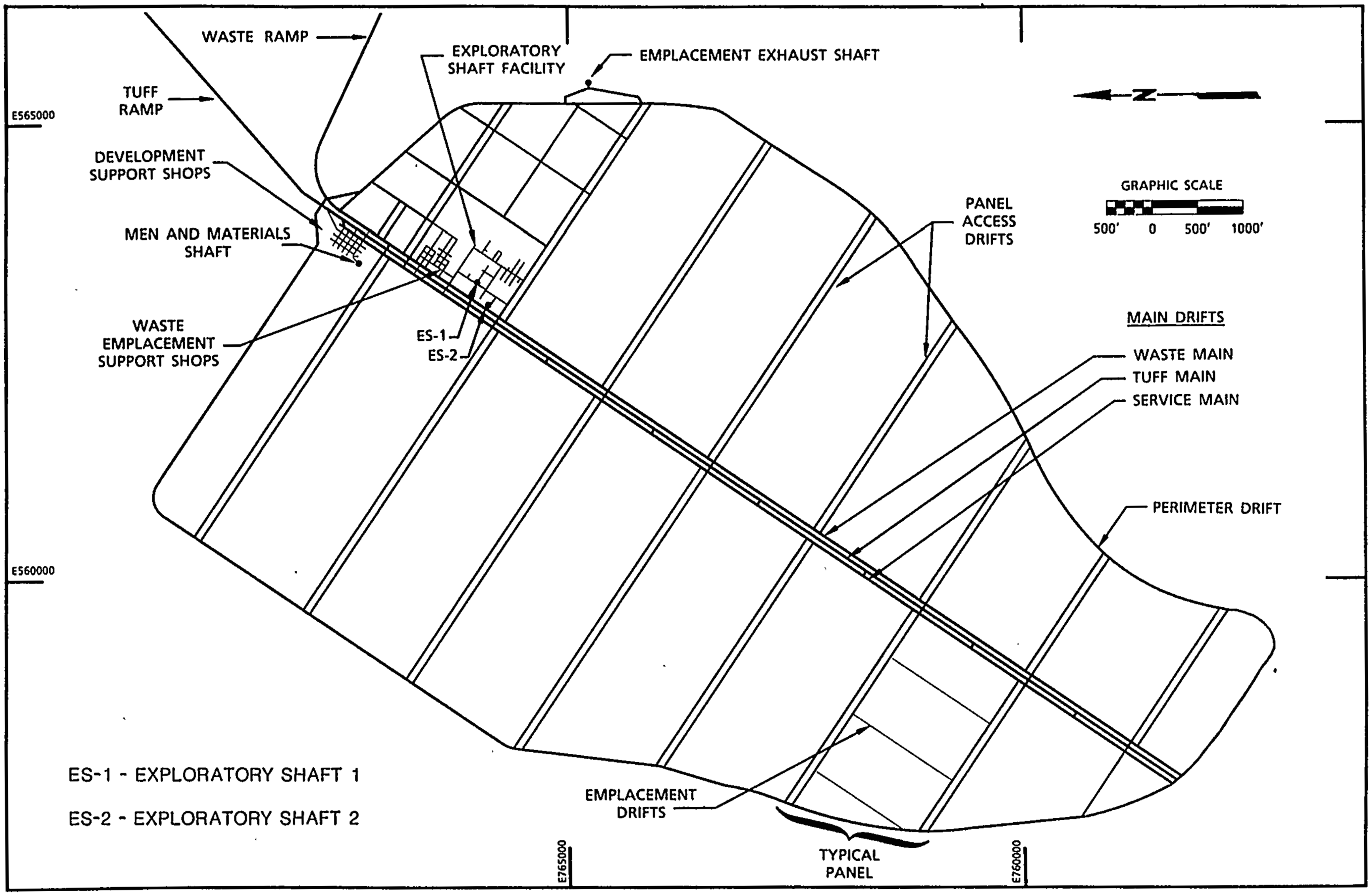


Figure 6-14. Horizontal emplacement configuration.

6.2.3 REPOSITORY OPERATIONS

This section briefly describes the principal operations that would be performed at a repository, including waste handling and disposal, waste retrieval, and support services. Chapters 3.1 and 4.5 of the SCP-CDR describe the operations, emplacement configurations, and equipment needed to perform these operations.

6.2.3.1 Waste handling and disposal operations

This section describes the current concepts for the waste receipt, and for preparation, storage, disposal, caretaker, and closure operations at the repository. This information is presented for spent fuel and other high-level waste. In addition, this section contains block-flow diagrams defining the principal operations, conceptual flow diagrams showing the equipment to be used in the operations, and lists of the major equipment illustrated in the conceptual flow diagrams. Surface and underground facility descriptions are provided in Sections 6.2.4, 6.2.5, and 6.2.6.

6.2.3.1.1 Waste handling operations

Figure 6-15 presents a block-flow diagram of the waste-handling operations. Waste shipped to the repository would be initially inspected at the gate to the waste-receiving area (Dennis et al., 1984b and 1984c). After this preliminary inspection, the waste in its shipping cask would be moved into the receiving area by the carrier. Complete radiological surveys and security-related inspections would be performed on the cask and carrier before moving them to the designated parking or waiting area near the waste-handling buildings. Figure 6-16 shows the steps that would be involved in transferring the waste to the waste-handling building and unloading the waste from the shipping cask.

The surface facilities at Yucca Mountain would be developed in two stages. During the Stage 1 operation, waste handling building 1 (WBH-1) would be used for the preparation of spent fuel for disposal (Figures 6-17, 6-18, and 6-19). During Stage 2, waste would be prepared in both WHB-1 and waste handling building 2 (WHB-2). WHB-2 is designed to have the capability of consolidating spent fuel assemblies (Figure 6-20). WHB-1 would be used during Stage 2 for preparing waste that would not require consolidation (i.e., defense high-level waste, West Valley high-level waste, and spent fuel consolidated at reactors or at an alternate facility). The waste would then be placed in a surface storage vault to await emplacement. Site-generated radioactive waste would be prepared in a separate building for offsite disposal, as shown in Figures 6-21 through 6-25. Offsite disposal of site-generated waste is a DOE design criteria (Stein, 1986).

Waste emplacement operations would begin with the removal of the waste packages from surface storage in the waste-handling buildings. Site-generated waste would be removed from the storage areas in the waste treatment building and loaded on trucks for disposal outside the repository (Figure 6-25).

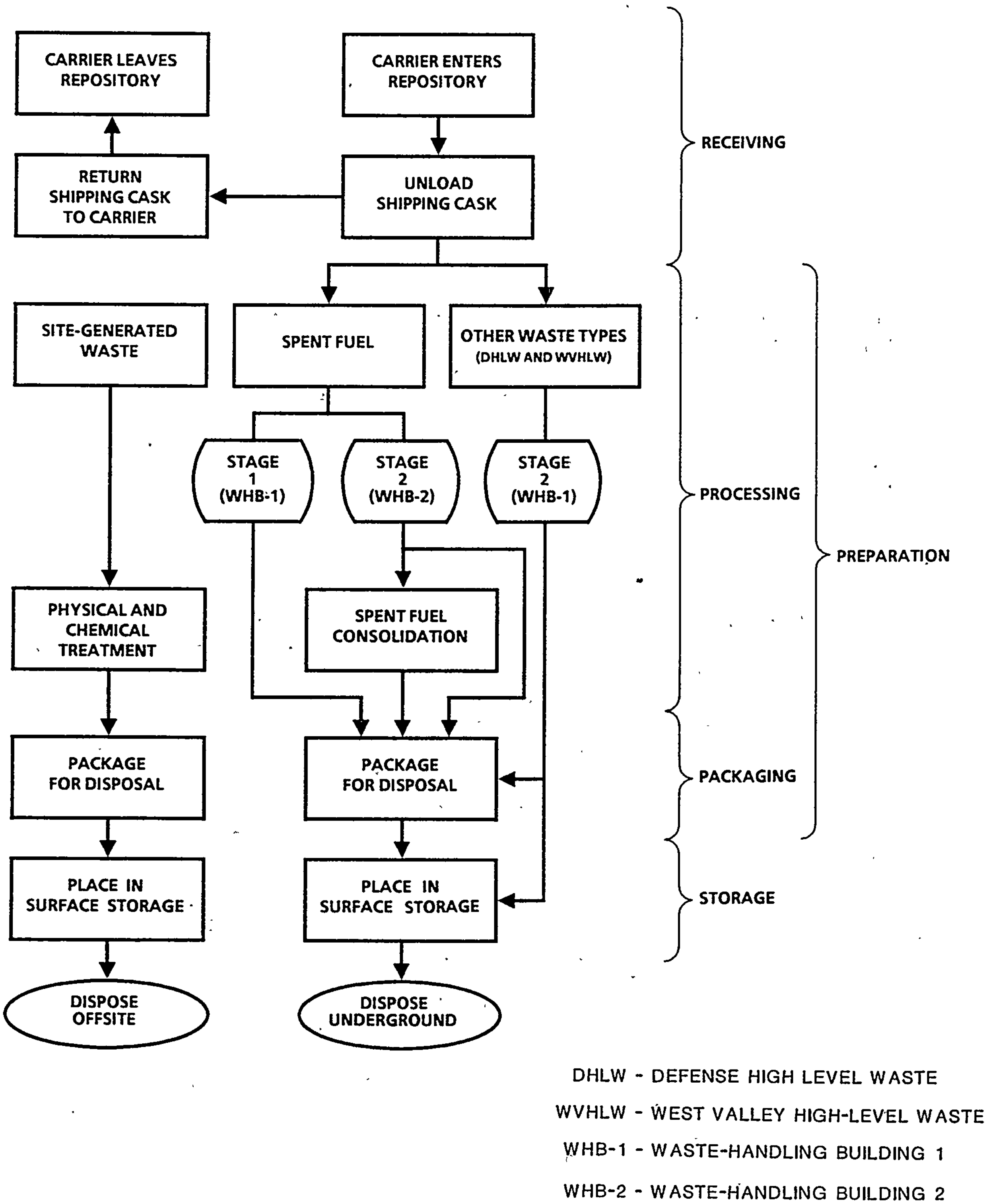


Figure 6-15. Flow diagram of waste handling.

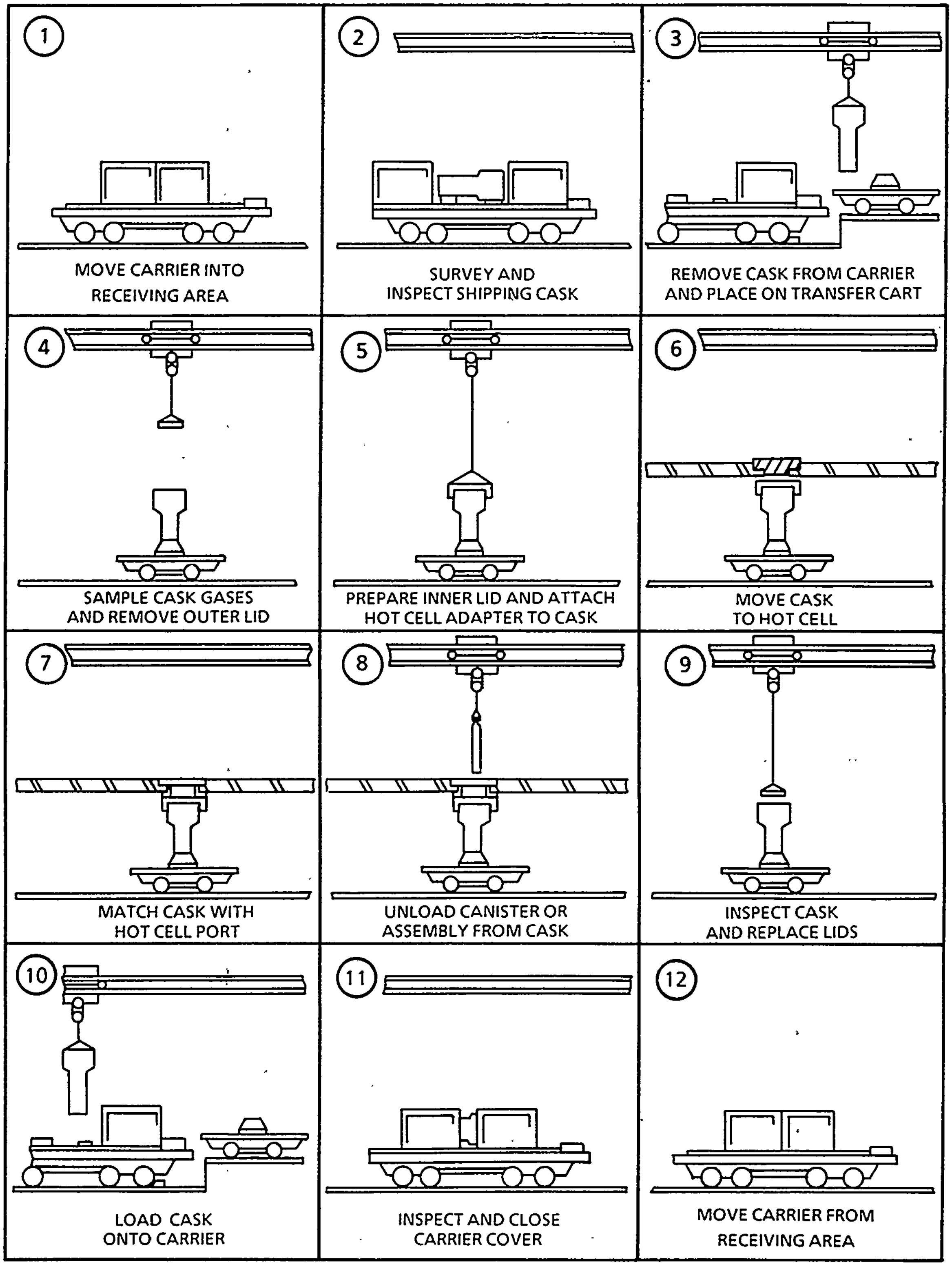


Figure 6-16. Steps involved in receiving the waste.

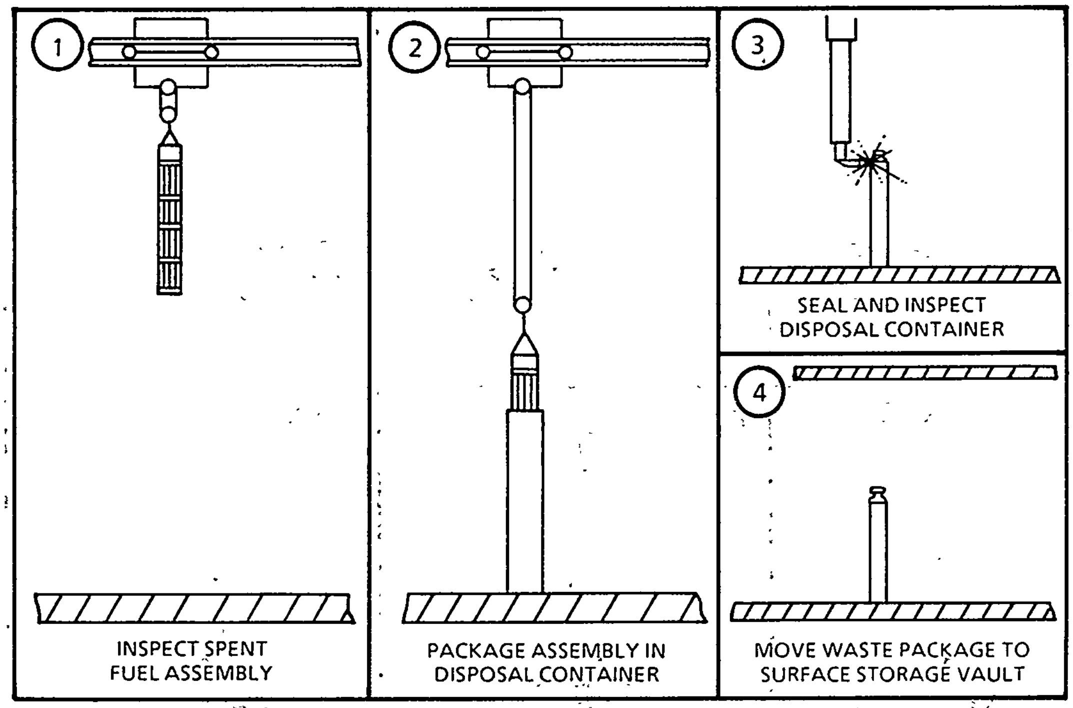


Figure 6-17. Inspection, packaging, and storage of a spent fuel assembly.

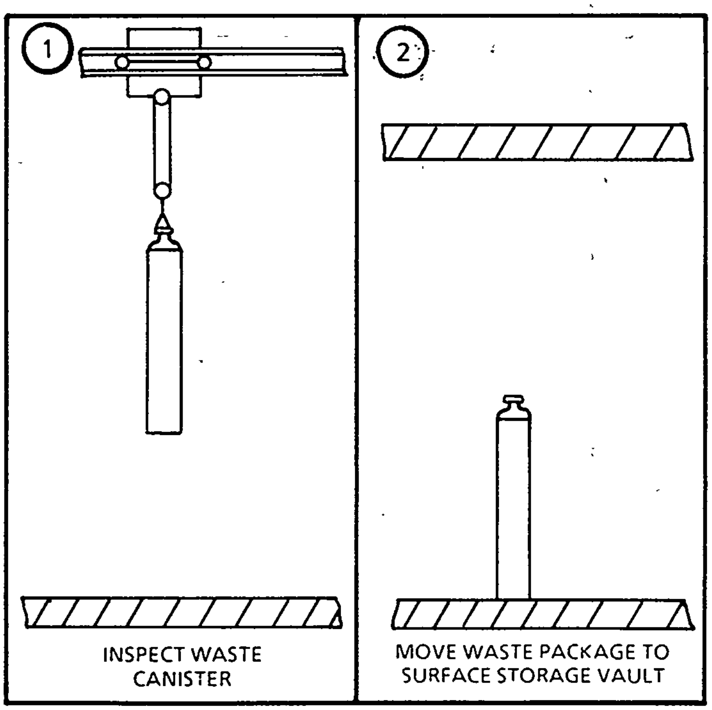


Figure 6-18. Inspection and storage of consolidated spent fuel and other high-level waste.

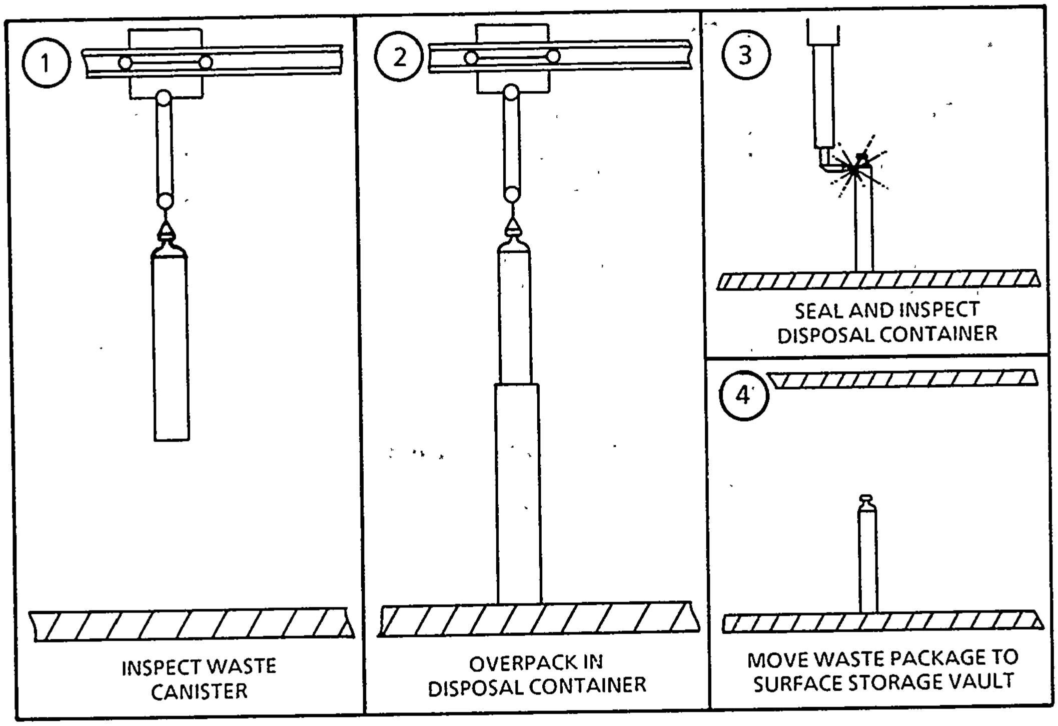


Figure 6-19. Inspection, overpacking, and storage of a canister.

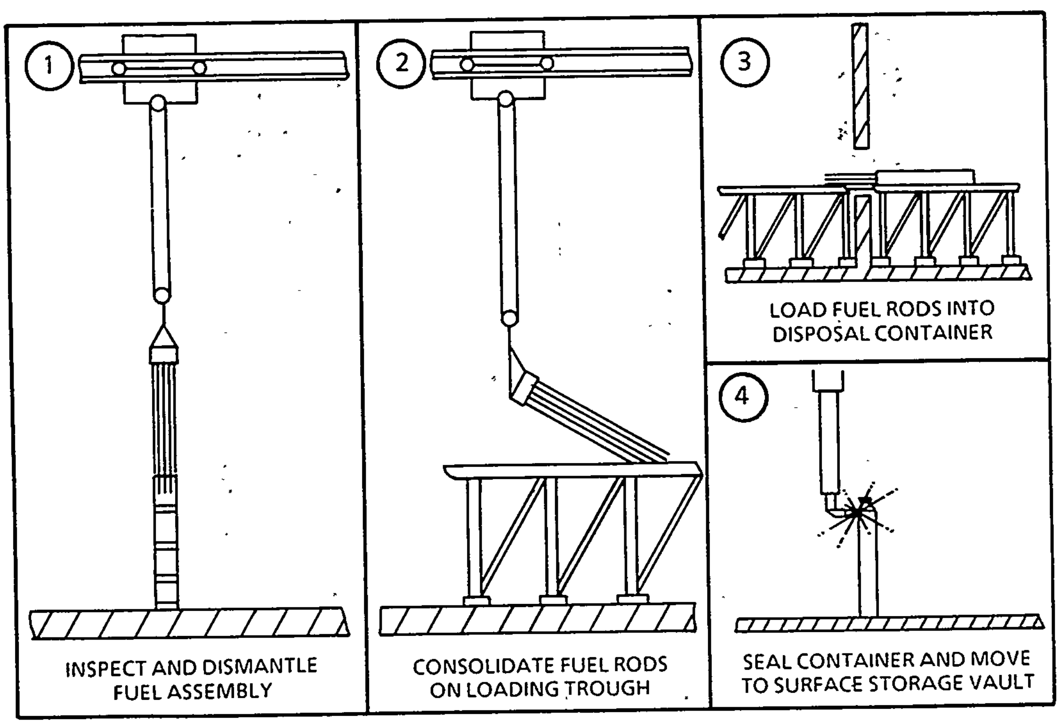


Figure 6-20. Consolidation of spent fuel assemblies in Stage 2.

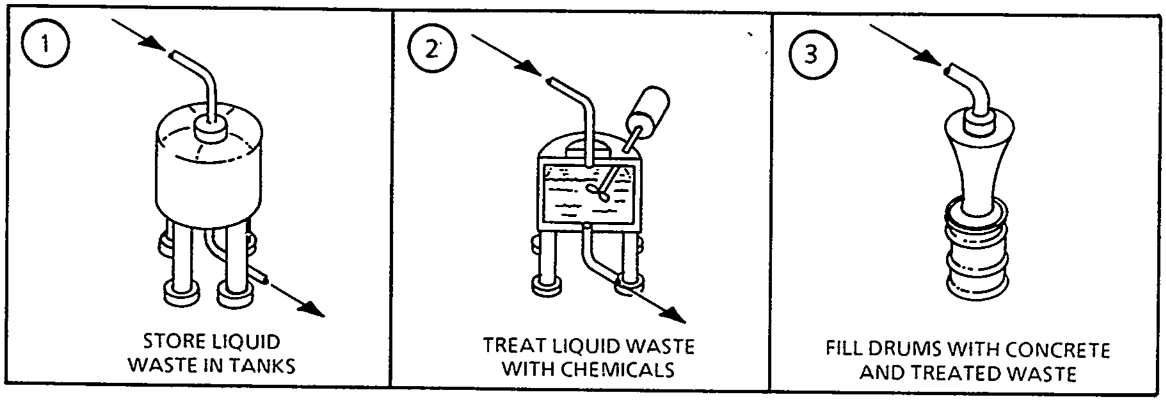


Figure 6-21. Treatment of liquid wastes.

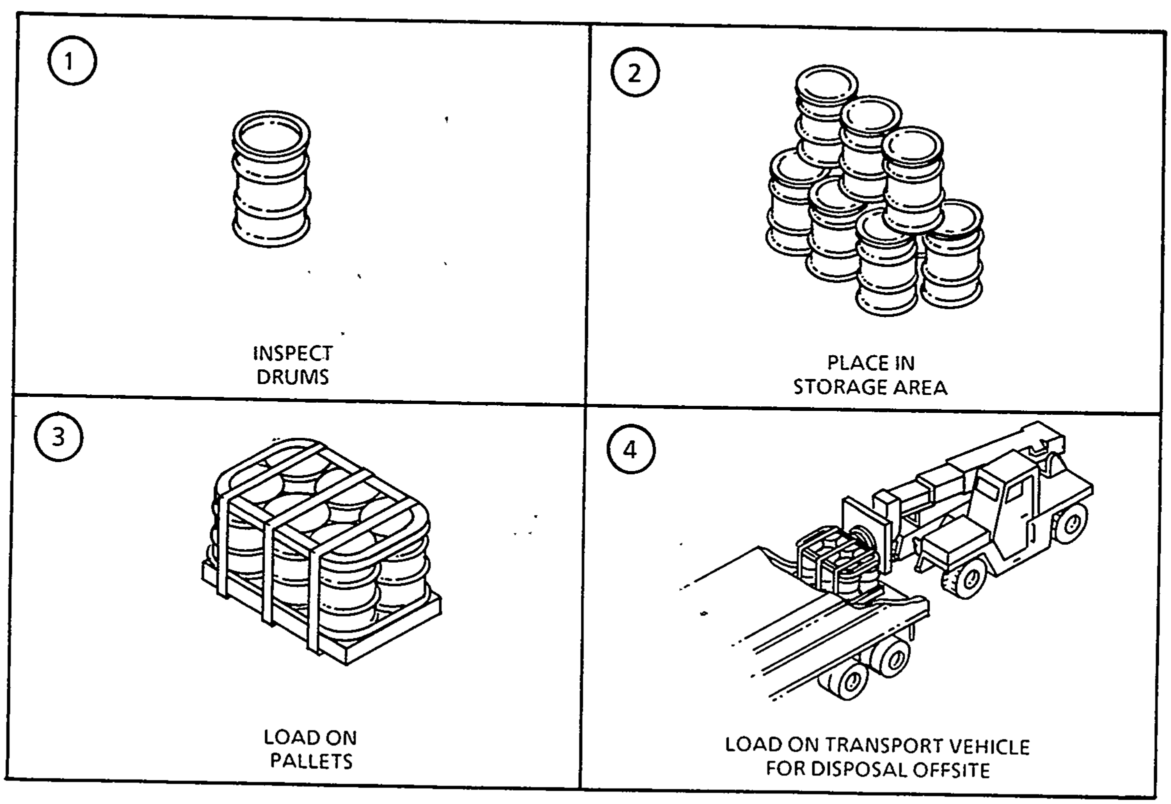


Figure 6-22. Storage and offsite shipment of solidified waste.

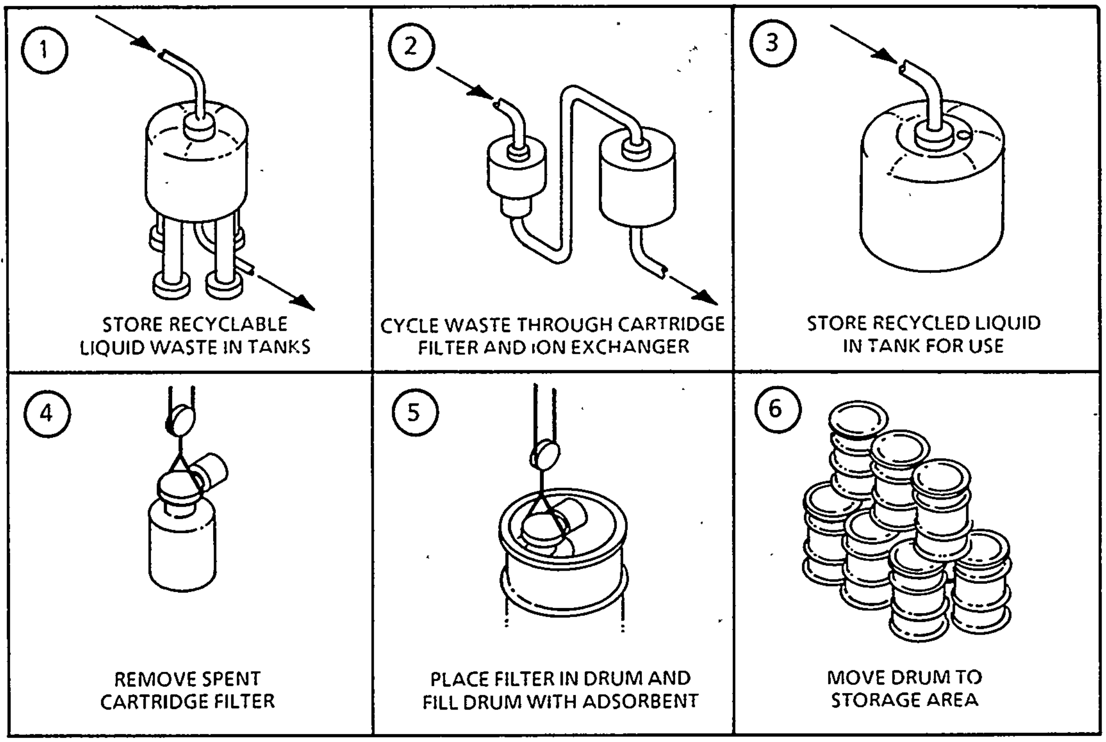


Figure 6-23. Packaging of spent cartridge filters.

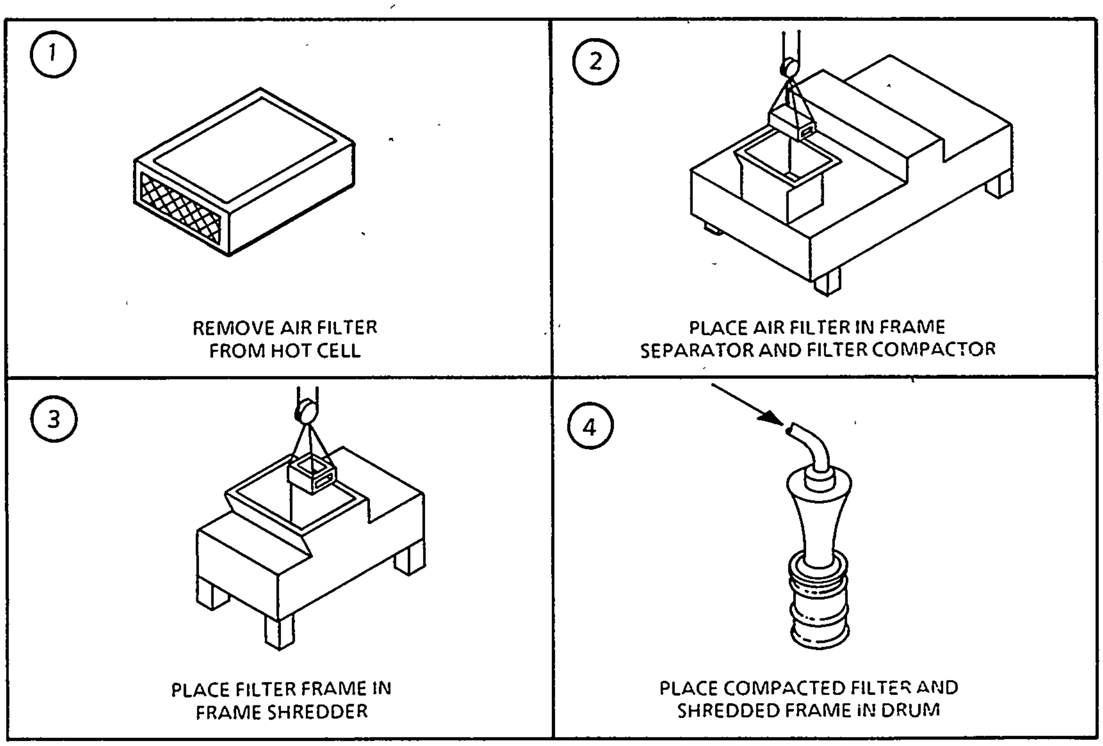


Figure 6-24. Packaging of air filters.

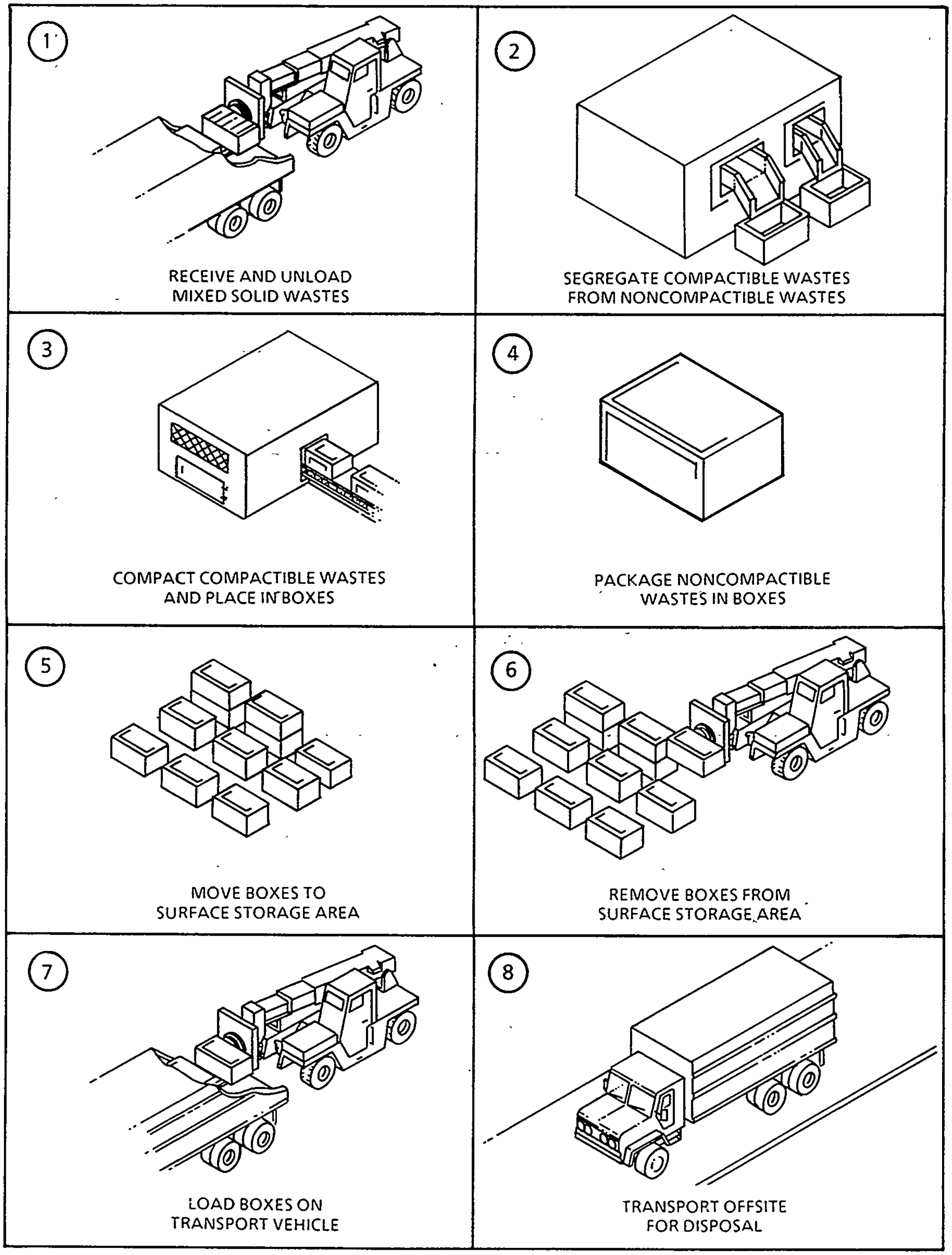


Figure 6-25. Preparation of offsite shipment of site-generated solid waste.

6.2.3.1.2 Waste disposal operations

Figure 6-26 presents a block-flow diagram of the waste-disposal operations for spent fuel and other high-level waste for both the reference vertical option and the horizontal option. Operations for both vertical and horizontal emplacement are presented as a comparison of the two configurations.

6.2.3.1.2.1 Vertical emplacement

The vertical emplacement configuration is the reference configuration for the NNWSI Project. The basic operations required for the vertical emplacement of a waste package include the following steps: (1) preparing the waste emplacement borehole, (2) transferring the waste to the emplacement area, (3) emplacing the waste container, and (4) closing the borehole. A description of these operations is provided by Stinebaugh and Frostenson (1986) and illustrated in Figures 6-27 through 6-30.

6.2.3.1.2.2 Horizontal emplacement

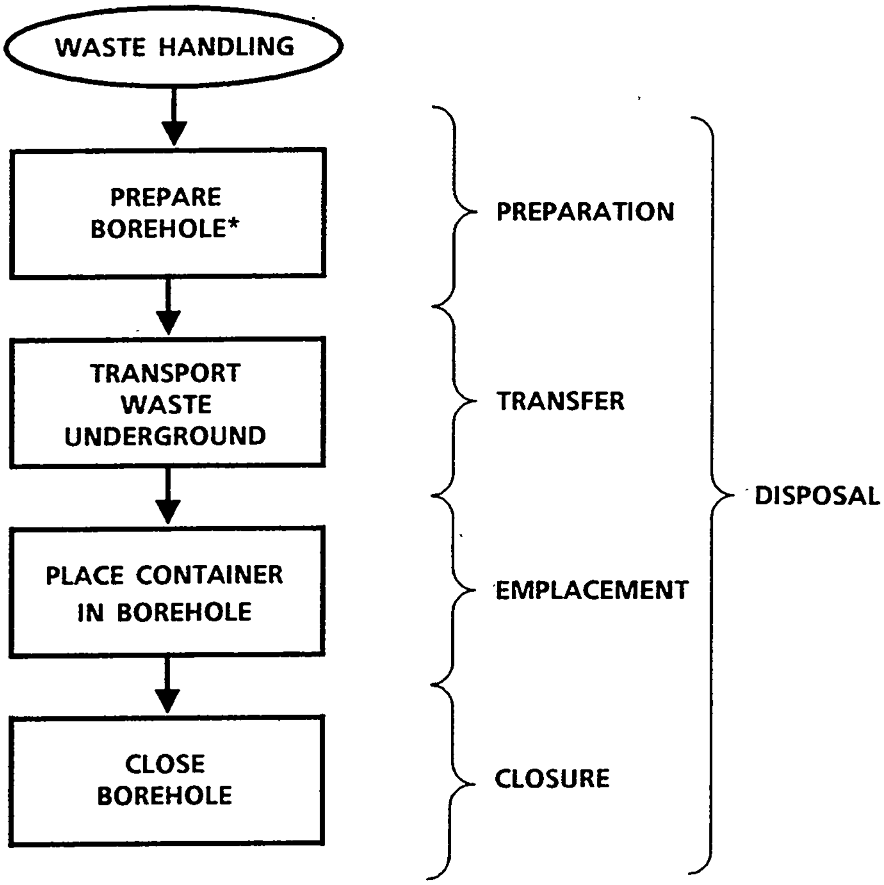
The basic operations required for the horizontal emplacement of a waste package include the following steps: (1) preparing a horizontal emplacement borehole, (2) transferring the waste to the emplacement area, (3) emplacing the waste container, and (4) closing the emplacement borehole. A description of these operations is provided by Stinebaugh et al. (1986) and illustrated in Figures 6-31 through 6-34.

6.2.3.1.2.3 Caretaker

Caretaker operations would be initiated after the last waste package had been emplaced and normal waste handling had been completed. These activities could include continued performance confirmation, radiological protection, security operations, and limited facility maintenance. Caretaker operations would continue until repository closure and decommissioning.

6.2.3.1.2.4 Closure and decommissioning

Permanent closure of the repository includes underground backfill and seals as described in Sections 6.2.7 and 6.2.8. Decommissioning activities could include decontamination and dismantling of the surface facilities and installation of facilities and equipment for a postclosure institutional barrier system (e.g., monuments).



* A VERTICAL OR HORIZONTAL BOREHOLE WILL BE PREPARED, DEPENDING ON THE EMPLACEMENT CONFIGURATION SELECTED

Figure 6-26. Flow diagram of waste disposal.

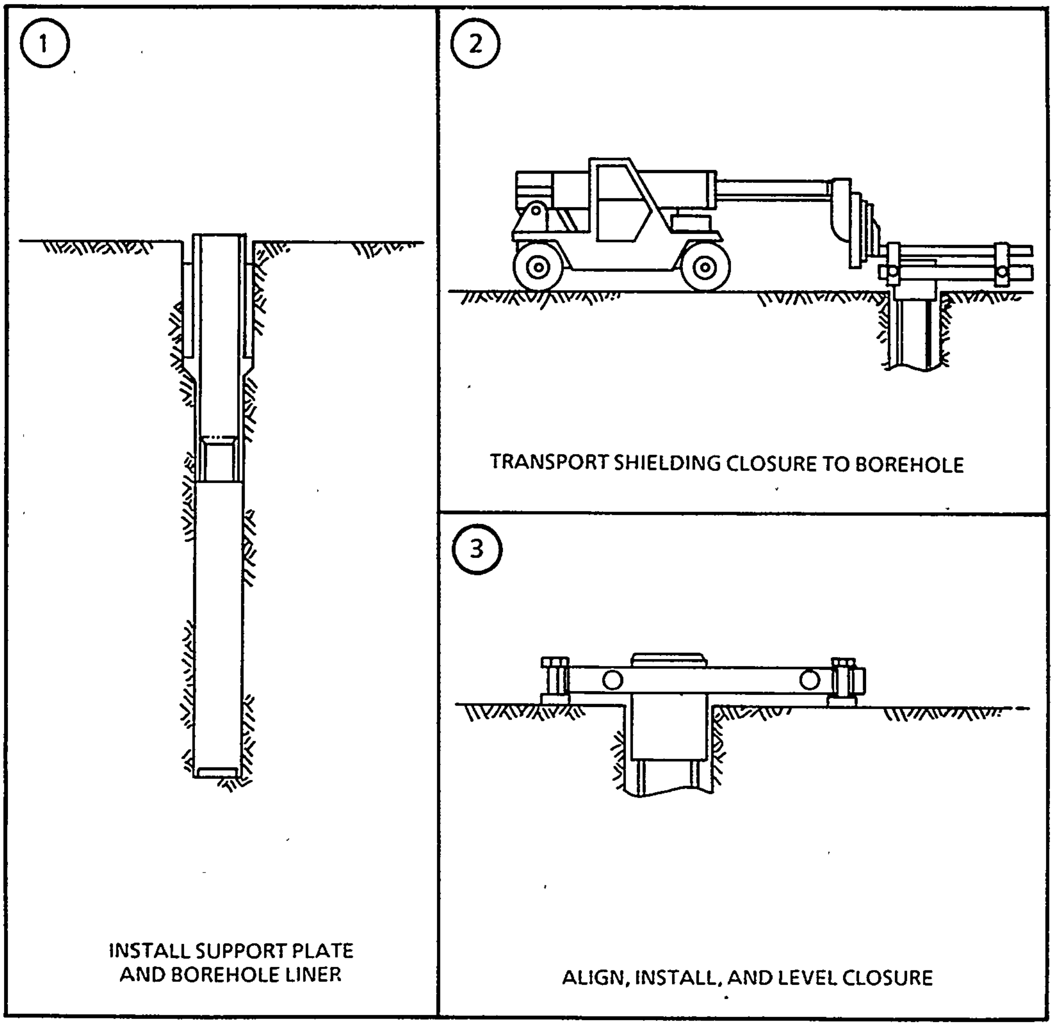


Figure 6-27. Preparation of a vertical emplacement borehole.

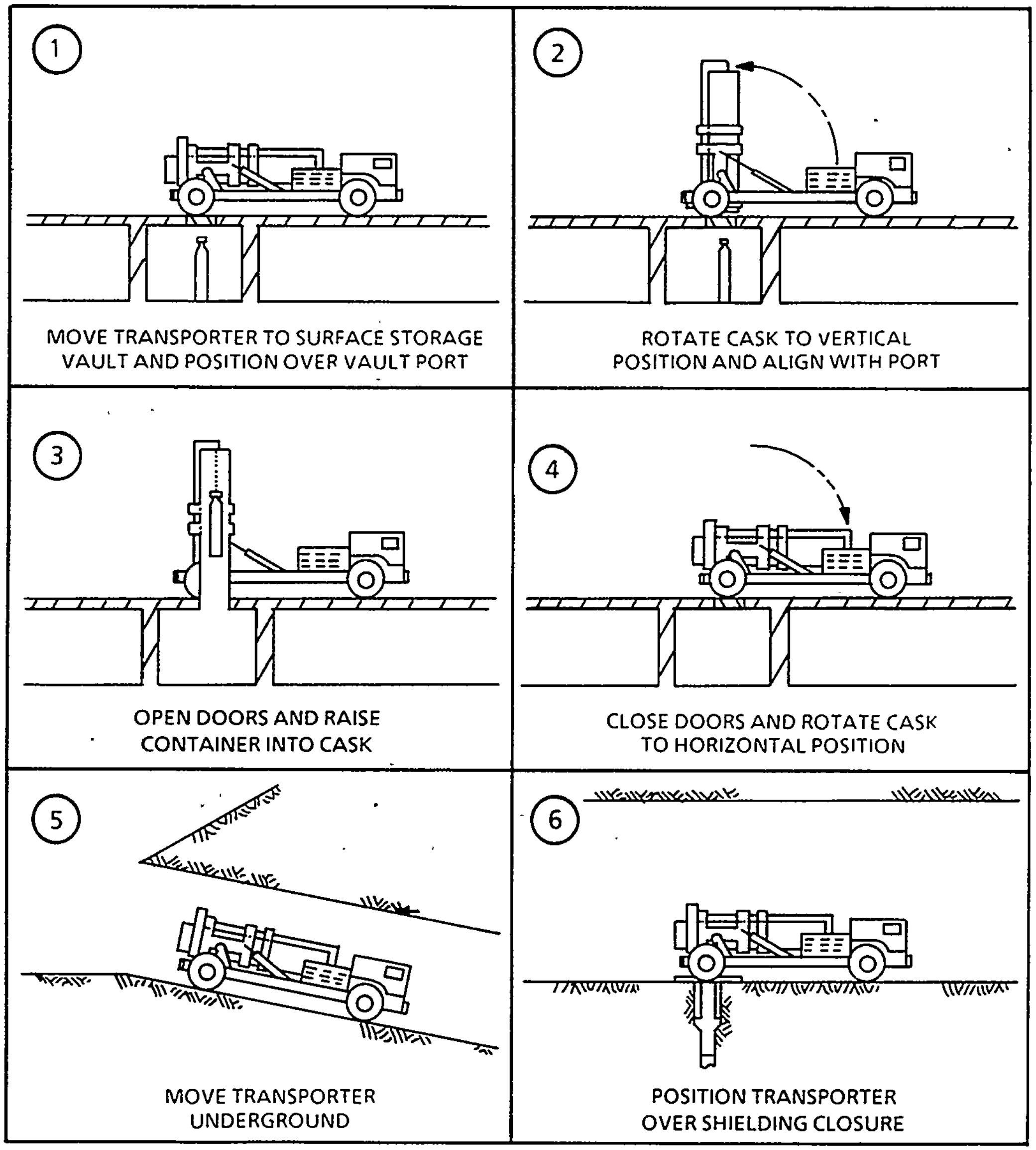


Figure 6-28. Transfer of a waste package to a vertical emplacement borehole.

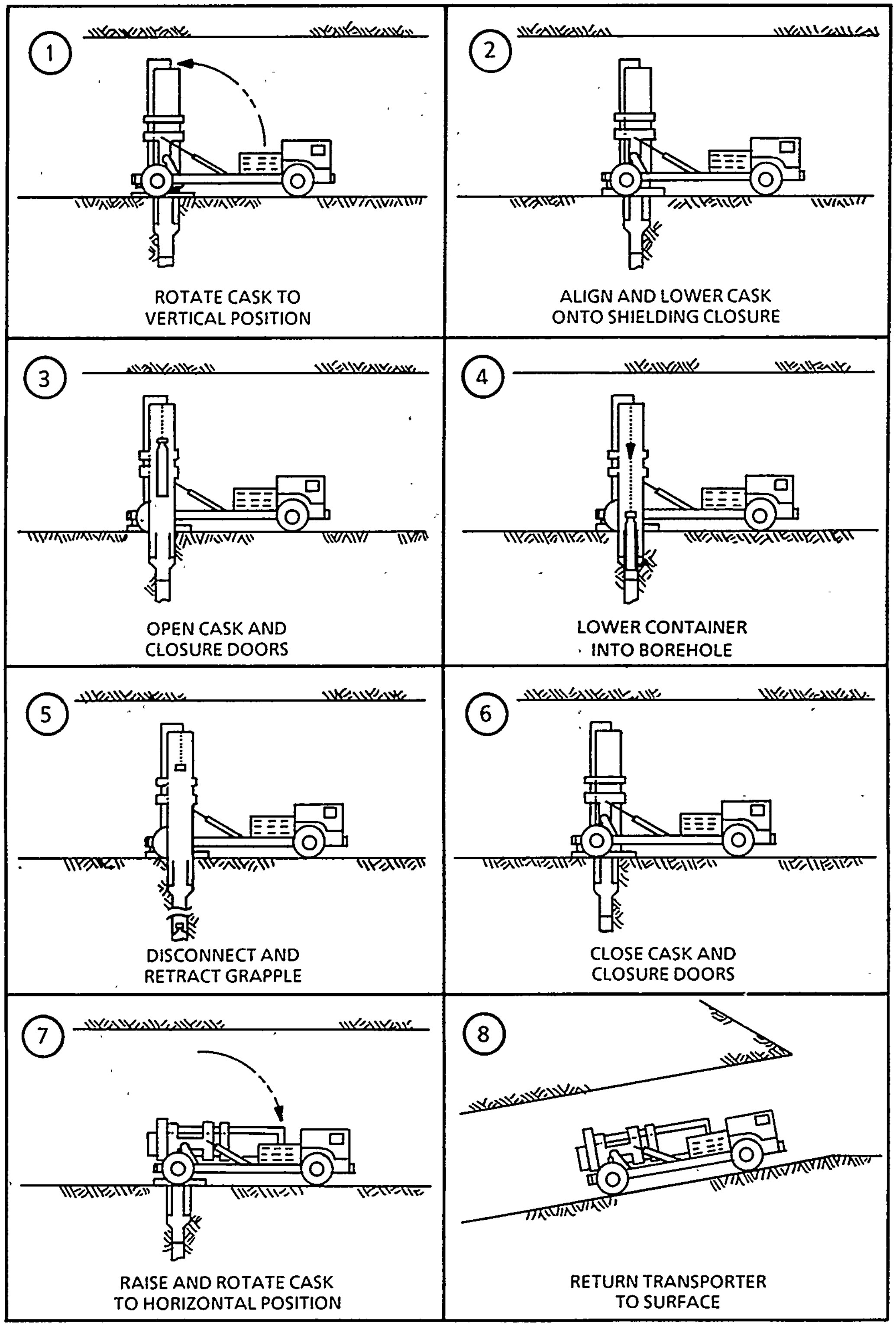


Figure 6-29. Emplacement of a waste package in a vertical borehole.

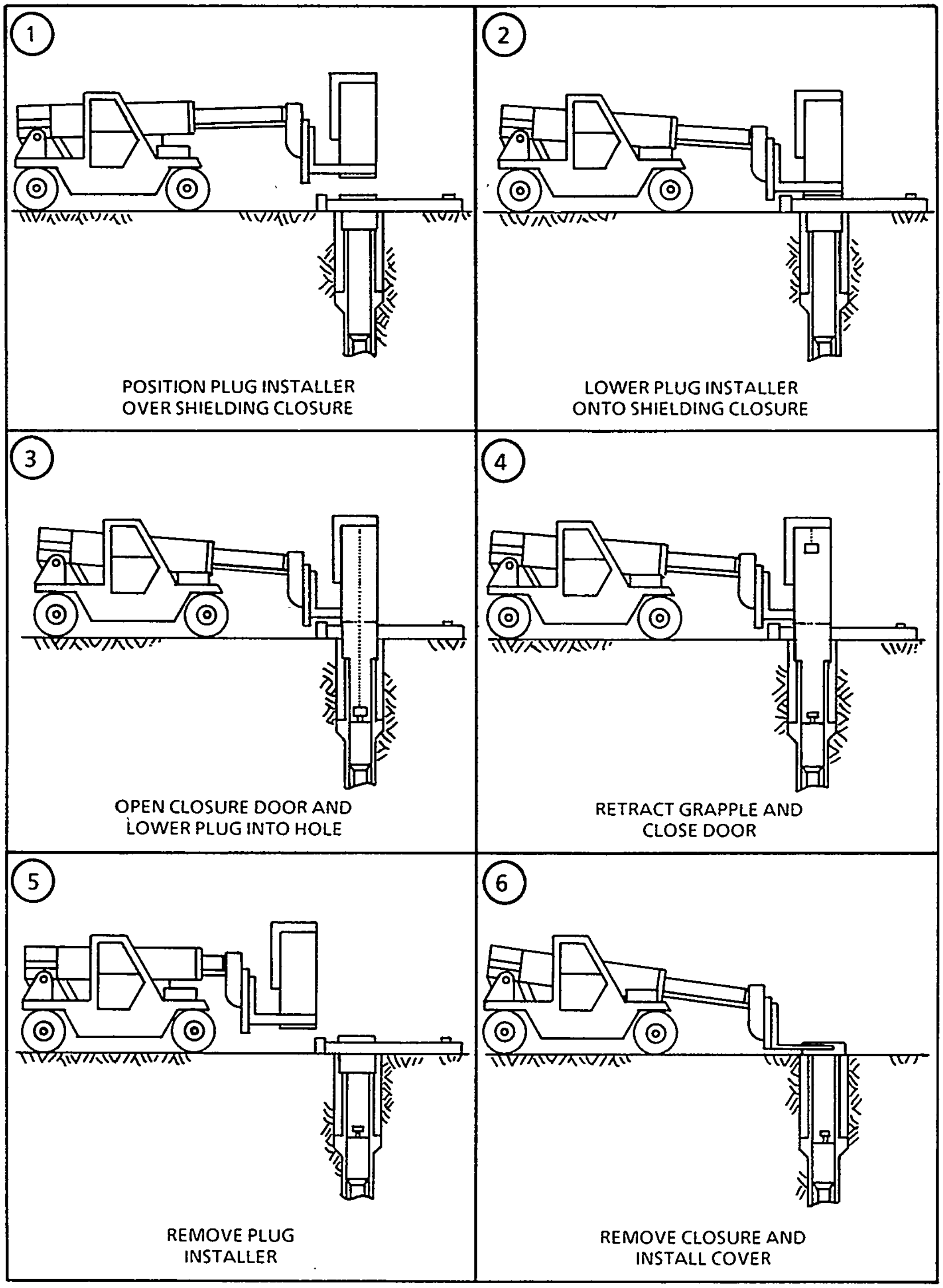


Figure 6-30. Closure of a vertical borehole after waste emplacement.

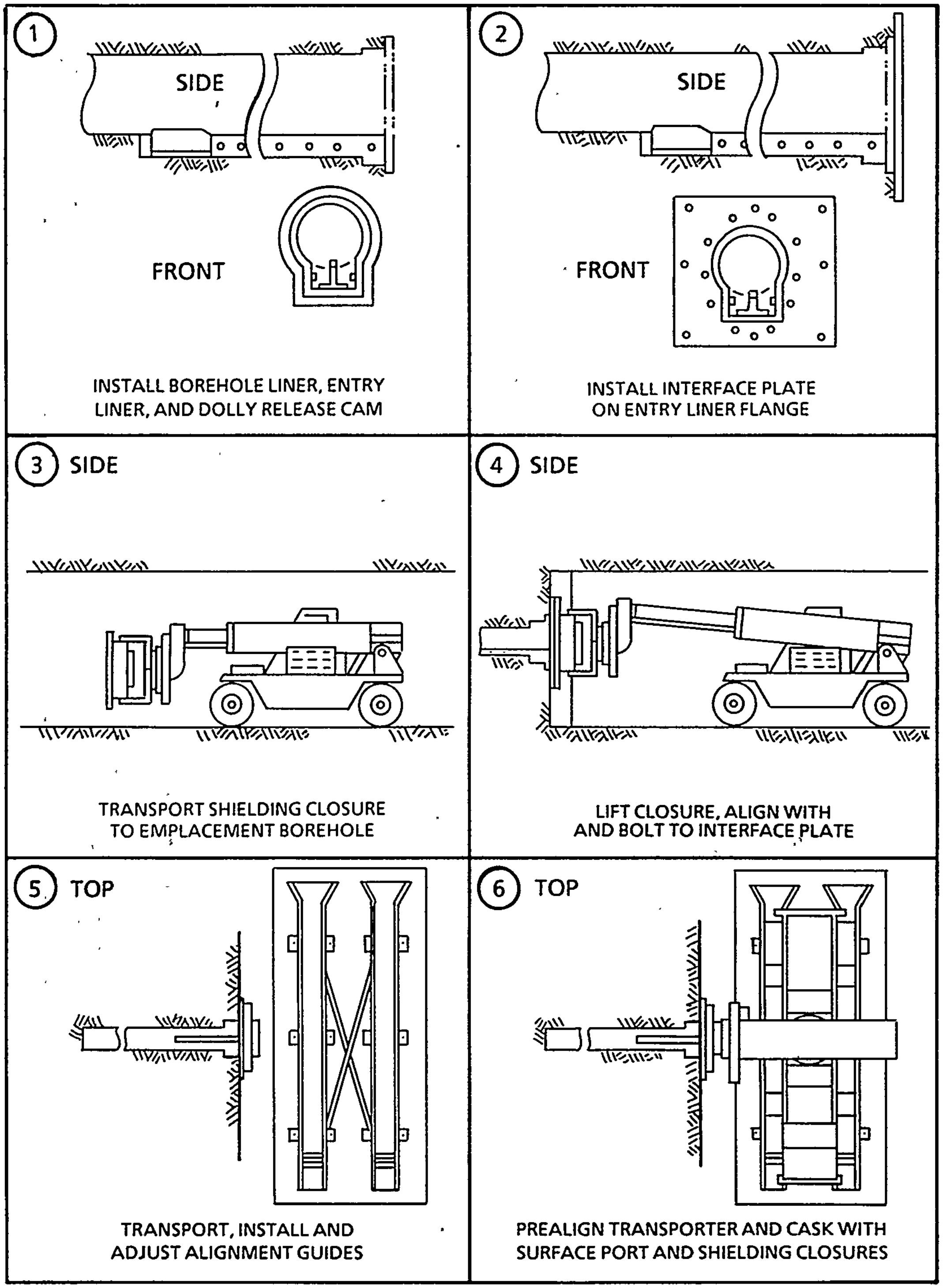


Figure 6-31. Preparation of a horizontal emplacement borehole.

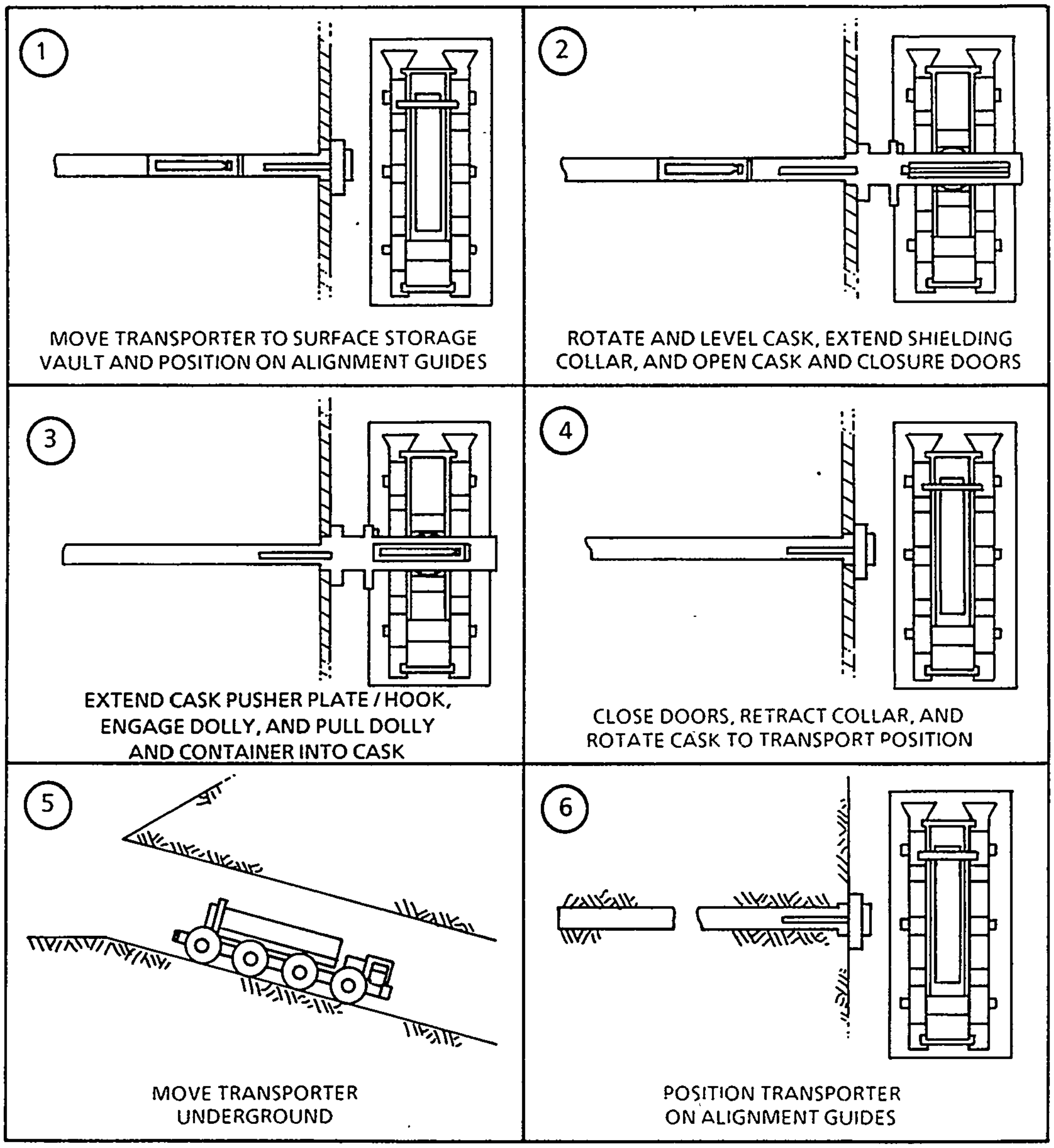


Figure 6-32. Transfer of a waste package to a horizontal emplacement borehole.

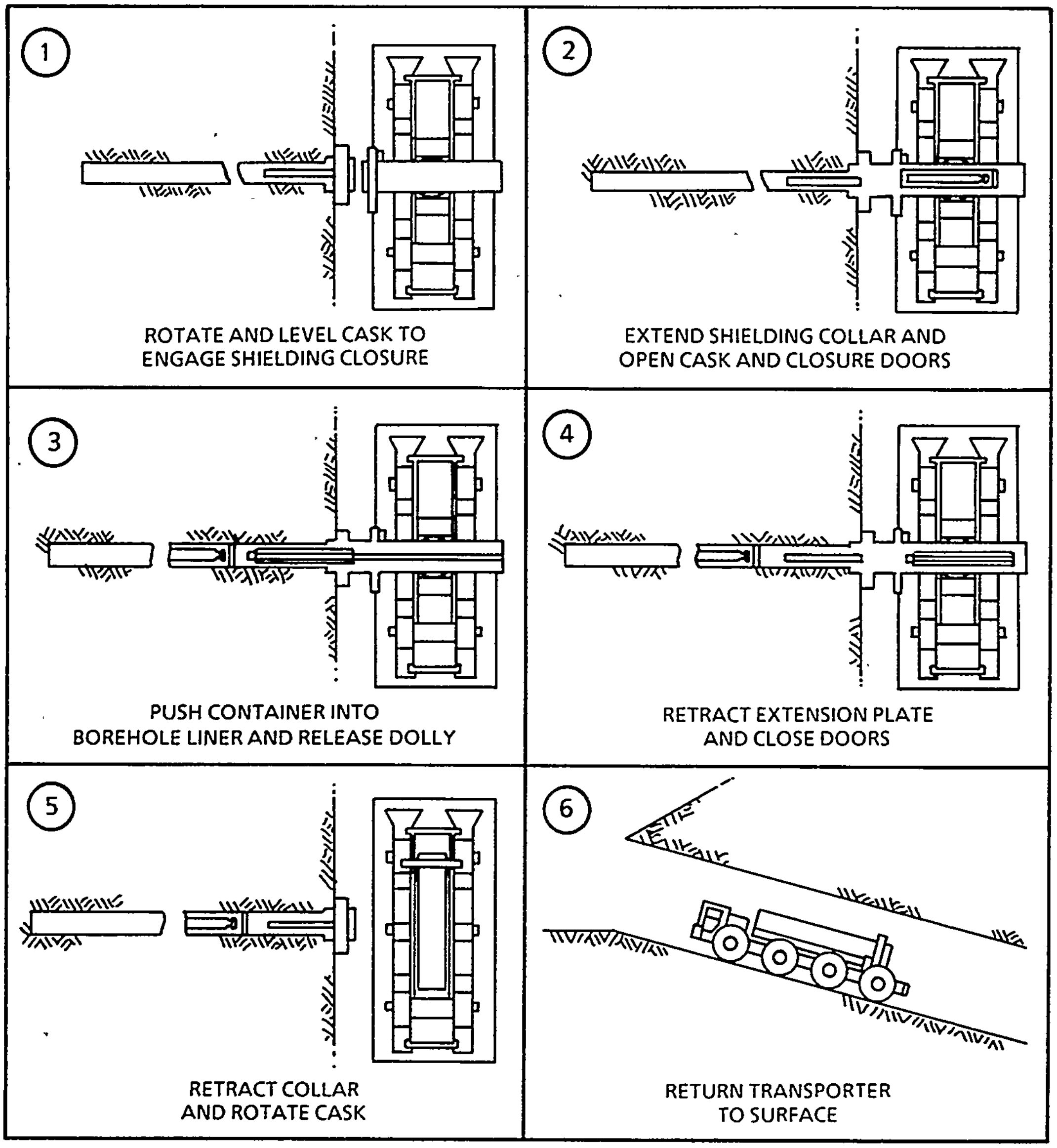


Figure 6-33. Emplacement of a waste package in a horizontal borehole.

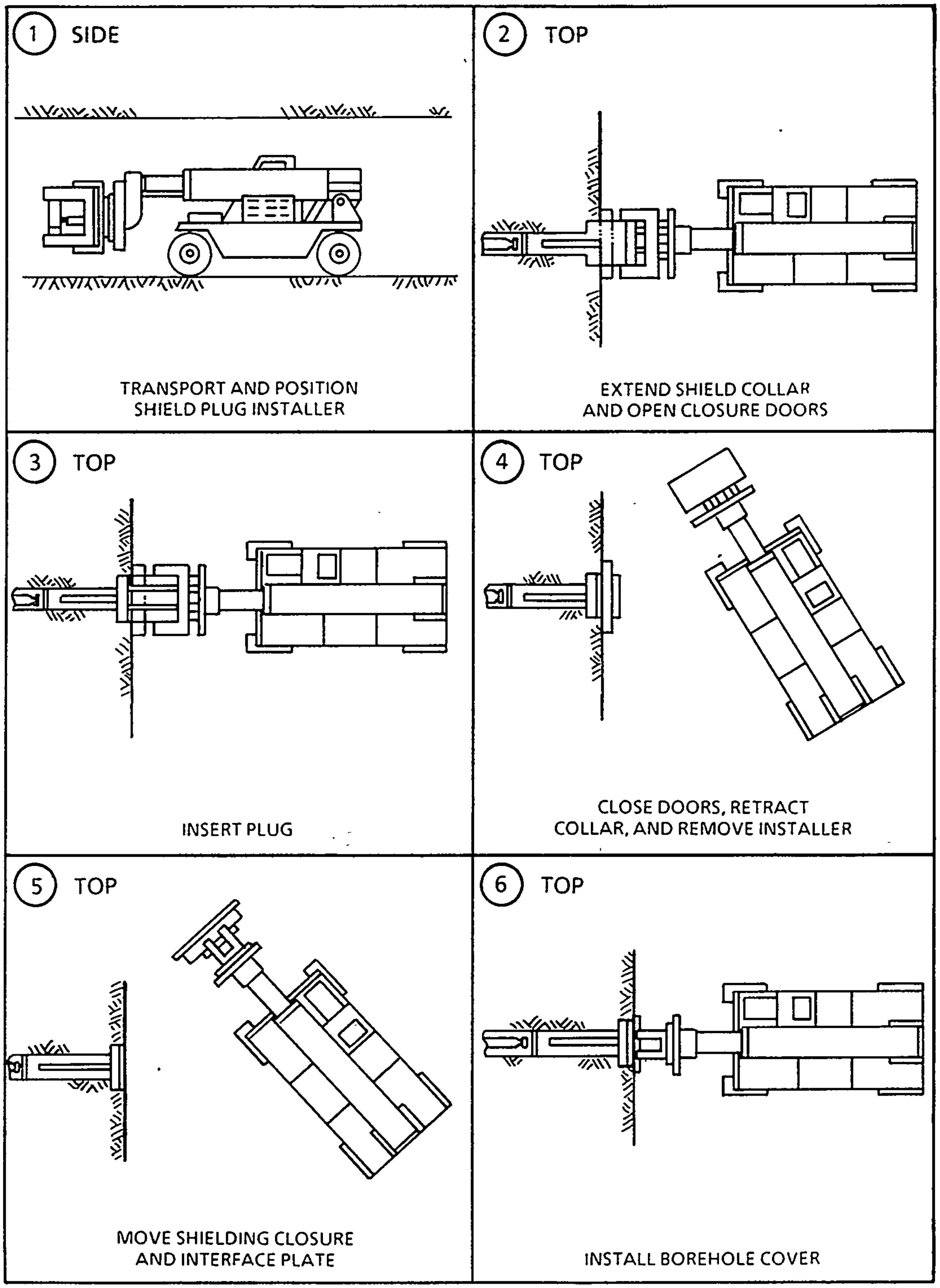


Figure 6-34. Closure of a horizontal borehole after waste emplacement.

6.2.3.1.3 Equipment

All equipment identified for waste emplacement is based on currently available technology. Conceptual designs have been developed for the equipment required for each emplacement concept. All emplacement equipment will require a detailed design, cost analysis, and development process to meet repository standards for feasibility, reliability, safety, and performance. Table 6-19 summarizes the equipment that would be used for vertical emplacement and its functions. The transporter for vertical emplacement is illustrated in Figure 6-35 (Stinebaugh and Frostenson, 1986). The equipment that will be used in horizontal emplacement is summarized in Table 6-20. The transporter for horizontal emplacement is illustrated in Figure 6-36. (Stinebaugh et al., 1986).

The drilling of vertical emplacement boreholes to contain a single waste package would be accomplished using existing mining equipment modified for this purpose. The capability for accurately drilling and lining horizontal boreholes will require the development and demonstration of prototype drilling equipment. Borehole drilling and lining is part of the subsurface excavation and development process discussed in Section 6.2.6.1.

6.2.3.2 Waste retrieval and shipping operations

This section describes the current concepts for the retrieval operations at the repository. In addition, this section contains a block-flow diagram defining the principal operations and conceptual flow diagrams showing the equipment to be used in the operations. The retrieval concept is discussed in Section 6.2.9.

6.2.3.2.1 Waste retrieval

Figure 6-37 is a block-flow diagram of the waste retrieval operations under normal conditions. Operations for both vertical and horizontal retrieval are presented for a comparison of the two systems. Waste retrieval is discussed in more detail in Section 6.2.9.

6.2.3.2.1.1 Vertical retrieval

The basic operations required for the removal of a vertically oriented waste container include the following steps: (1) preparing the vertical borehole, (2) removing the waste container, (3) transferring the waste to the surface, and (4) closing the borehole. A description of these operations under normal conditions is provided by Stinebaugh and Frostenson (1986) and illustrated in Figures 6-38 through 6-41. Off-normal events for retrieval are discussed in Section 6.2.9.2.2.

Table 6-19. Summary of vertical emplacement equipment and functions

Equipment	Function
	WASTE CONTAINER, EMPLACEMENT ENVELOPE, AND SHIELDING HARDWARE
Waste container	Waste containment
Borehole	Containment and support of waste container
Liner	Alignment of waste container and protection of borehole opening during emplacement and retrieval
Support plate	Centering and support of waste container
Plug	Radiation attenuation from borehole
Cover	Content identification and final closure of borehole
Instrumentation	Monitoring and preretrieval assessment
	WASTE TRANSPORTER
Transporter cab	Steering, controls, and monitoring
Running gear	Locomotion
Brake system	Braking
Hydraulic system	Cask support and positioning
Cask	Conveyance, handling, and shielding of waste container
	MODIFIED FORKLIFT
Forklift cab	Steering, controls, and monitoring
Running gear	Locomotion, transportation, and towing of equipment
Extending boom	Alignment, installation, and removal of shielding mechanism and equipment
	SHIELDING MECHANISM AND EQUIPMENT
Hoisting adapter	Handling of shielding closure
Shielding closure	Temporary shielding during installation and retrieval of borehole plug
	PLUG INSTALLER AND REMOVER
Housing	Radiation shielding during installation and removal of plug
Hoist	Raising and lowering of plug through shielding closure
Grapple	Attachment of hoist to pintle of plug

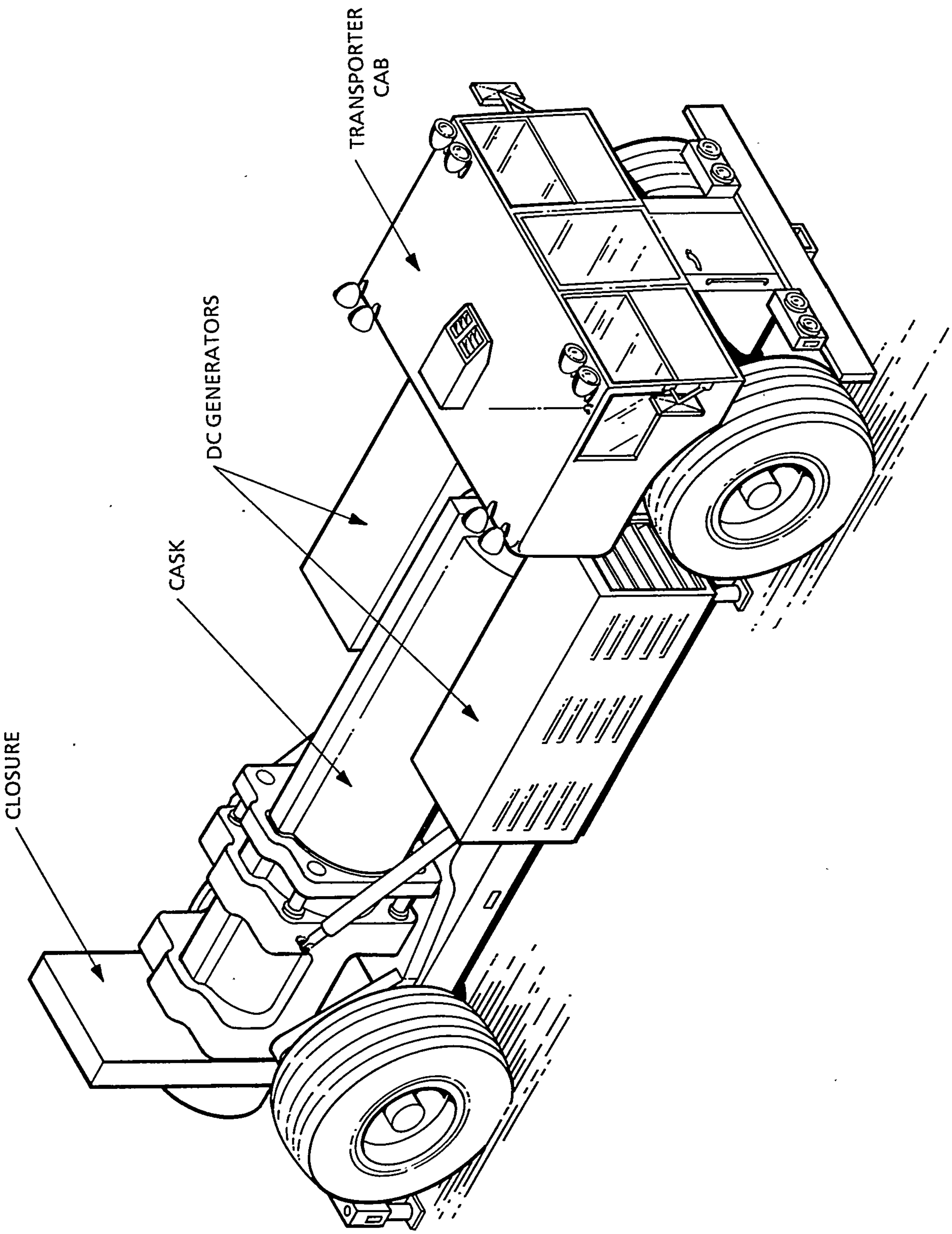


Figure 6-35. Transporter for vertical emplacement.

Table 6-20. Summary of horizontal emplacement equipment and functions

Equipment	Function
	WASTE CONTAINER AND DOLLY
Waste container	Waste containment
Dolly	Mobility for waste container
	EMPLACEMENT ENVELOPE
Borehole	Waste containment
Liner	Waste containment and support
Entry liner	Support of waste container during entry for emplacement and support for shield plug
Flange	Attachment for interface plate and borehole cover
Dolly release cam	Dolly release
Shield plug	Radiation attenuation from borehole
Borehole cover	Content identification and final borehole cover
	SHIELDING MECHANISM
Shielding closure	Temporary shielding during emplacement and retrieval of the waste container, and installation or removal of the plug
Interface plate	Attachment for shielding closure
	ALIGNMENT
Alignment guides	Waste transporter positioning
	WASTE TRANSPORTER
Transporter cab	Steering, controls, guidance, monitoring, and safety
Frame and running gear	Support and locomotion
Hydraulic leveling system	Cask support and leveling
Electrical rotation system	Cask rotation
Transporter cask	Conveyance, handling, and shielding of waste container
Cask mechanism	Emplacement and retrieval of waste container
Extension plate	Insertion of waste container in and removal from borehole
Ballscrew shaft	Drive mechanism for extension plate
Pusher plate/hook	Emplacement and retrieval of waste container and dolly
Hook release cam	Hook release during emplacement
Roller chains	Drive mechanism for pusher plate and hook
Dolly release cam	Hook release during retrieval
	MODIFIED FORKLIFT
Forklift cab	Steering controls and monitoring
Running gear	Locomotion and transportation of equipment
Extending boom	Alignment, installation, and removal of shielding, mechanism, shield plug, and borehole cover
Fork	Attachment of shielding closure, shield plug installer and remover, and borehole cover
Shield plug installer and remover	Shield plug installation and removal

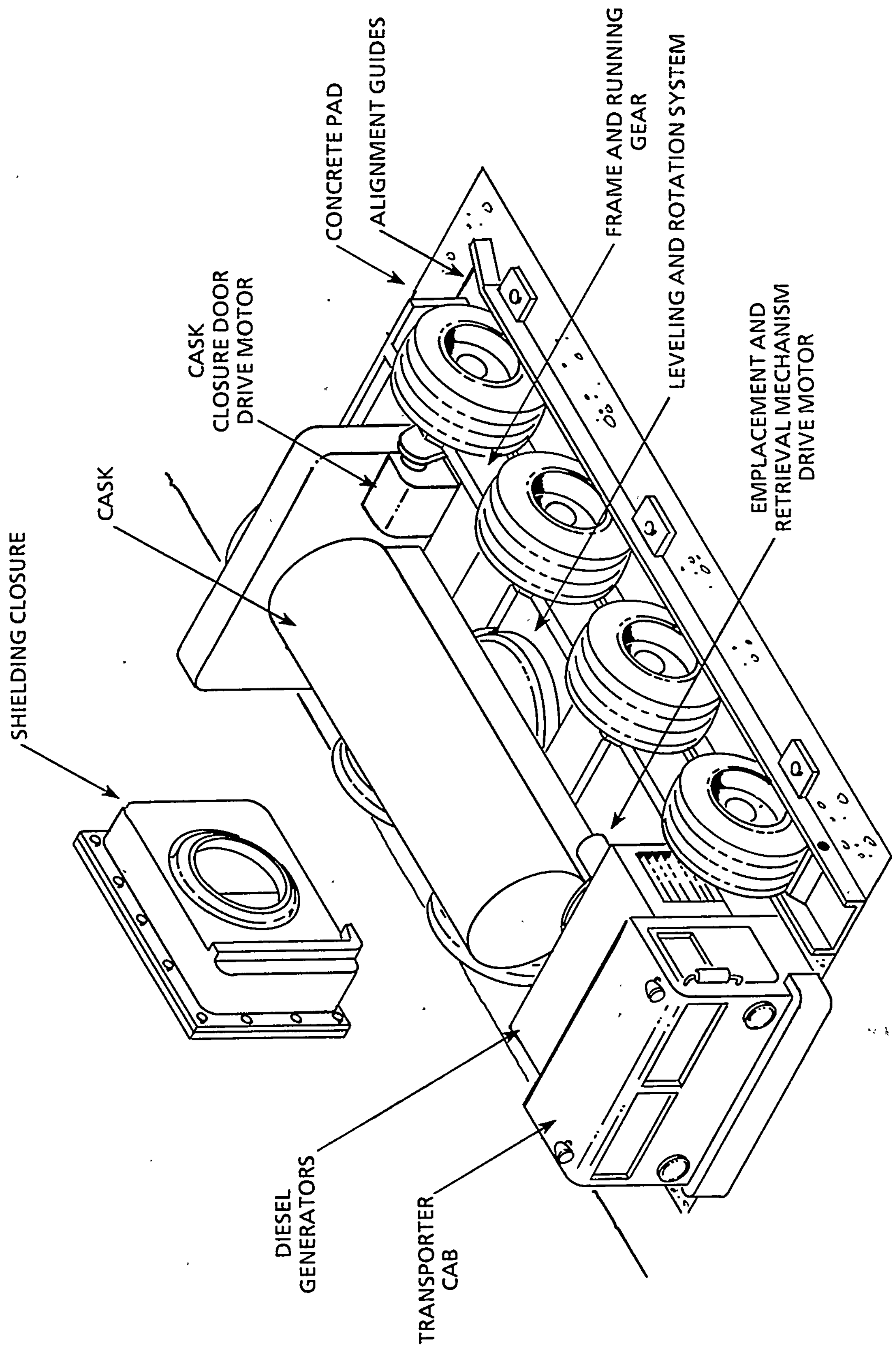


Figure 6-36. Transporter for horizontal emplacement.

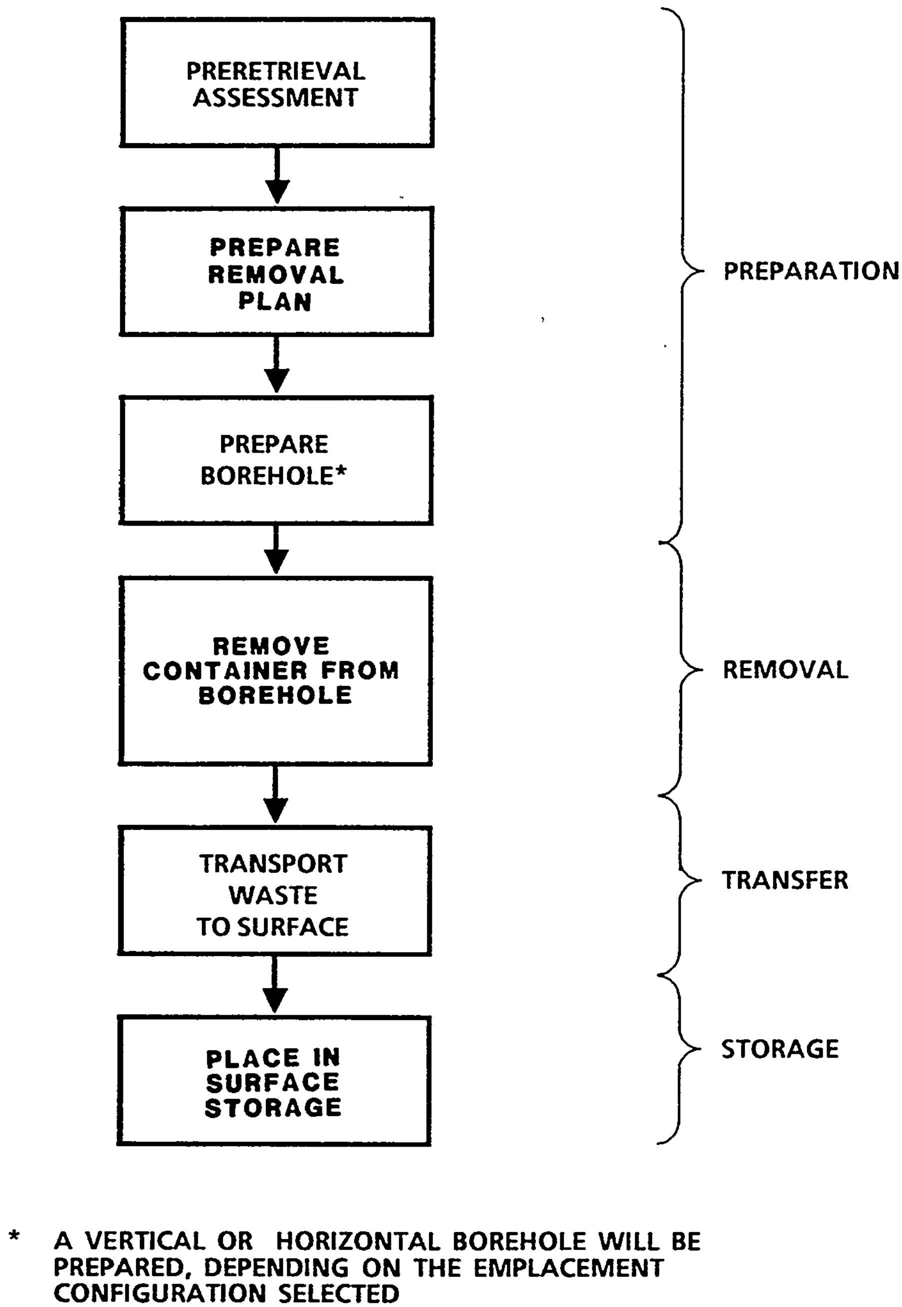


Figure 6-37. Flow diagram of waste retrieval.

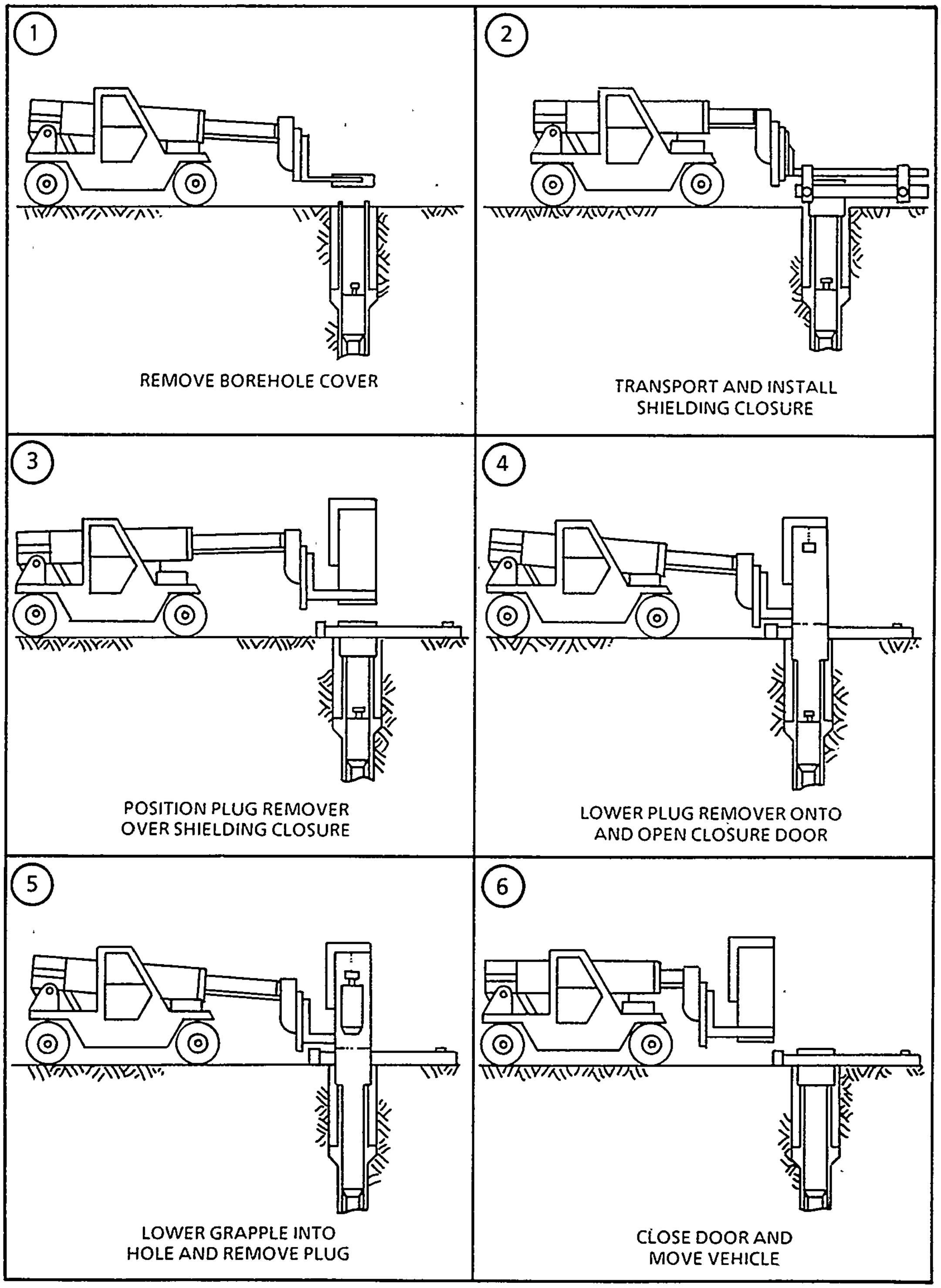


Figure 6-38. Preparation of a vertical borehole for waste package removal.

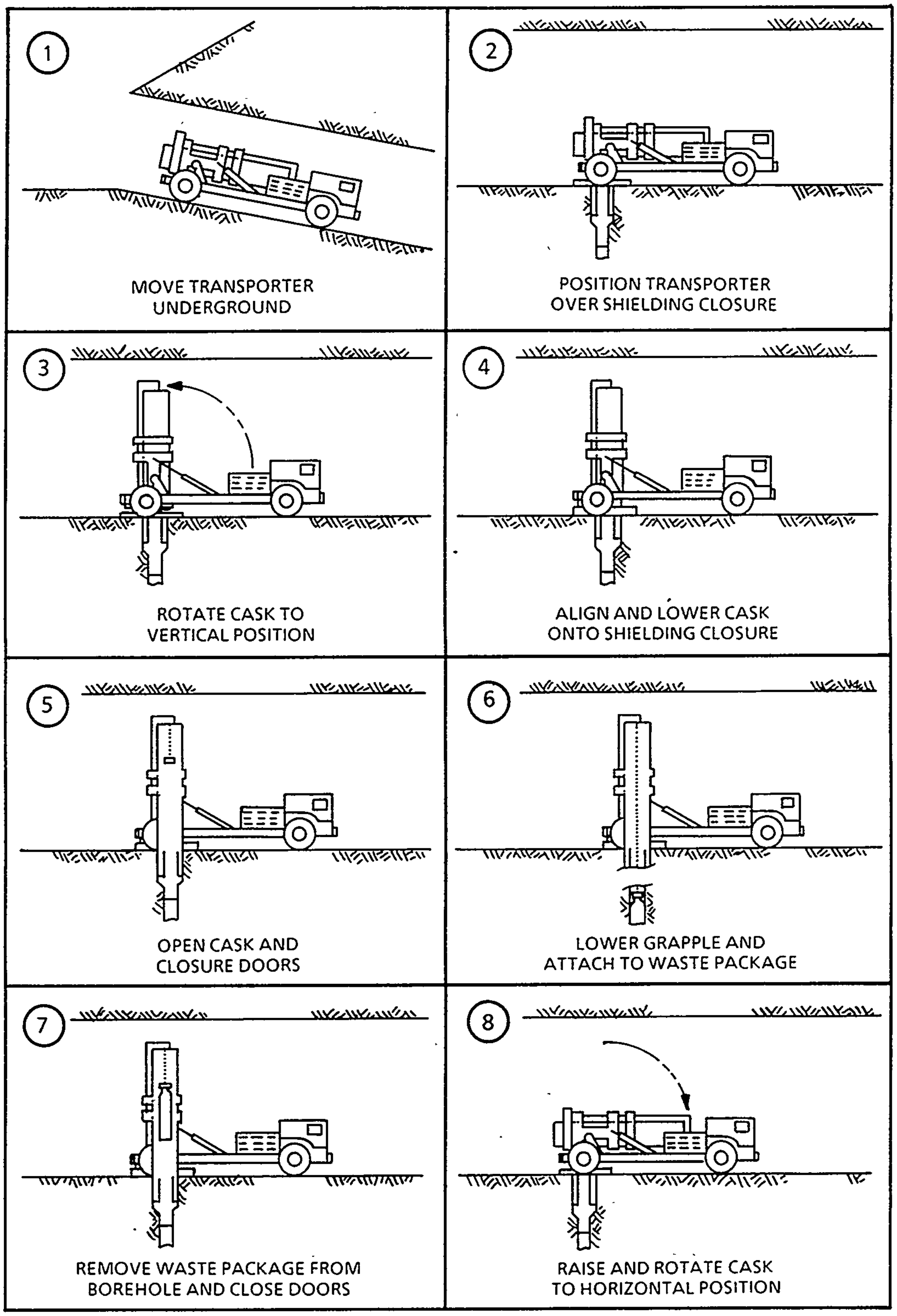


Figure 6-39. Removal of a waste package from a vertical borehole.

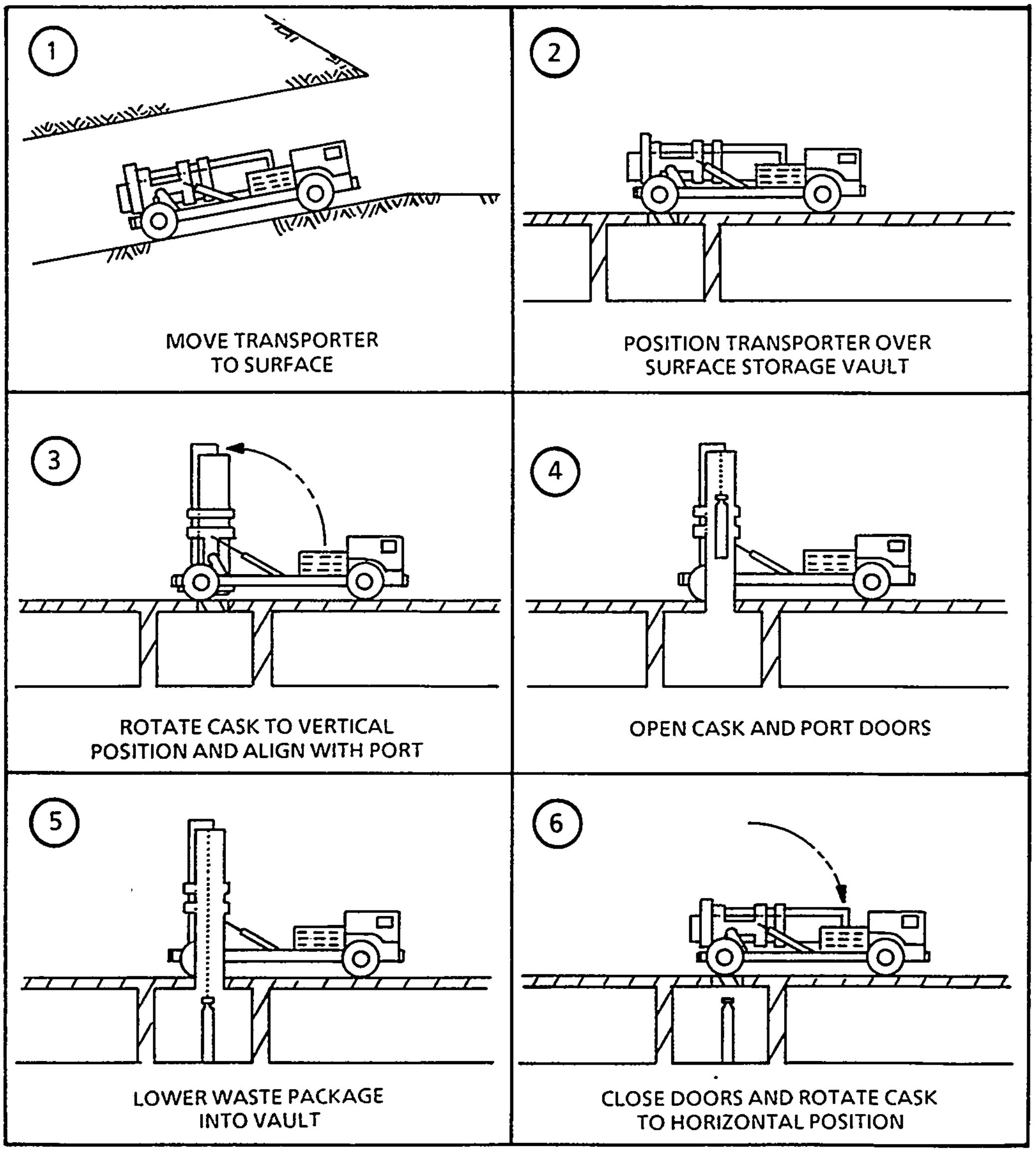


Figure 6-40. Transfer of a waste package from a vertical borehole to surface storage.

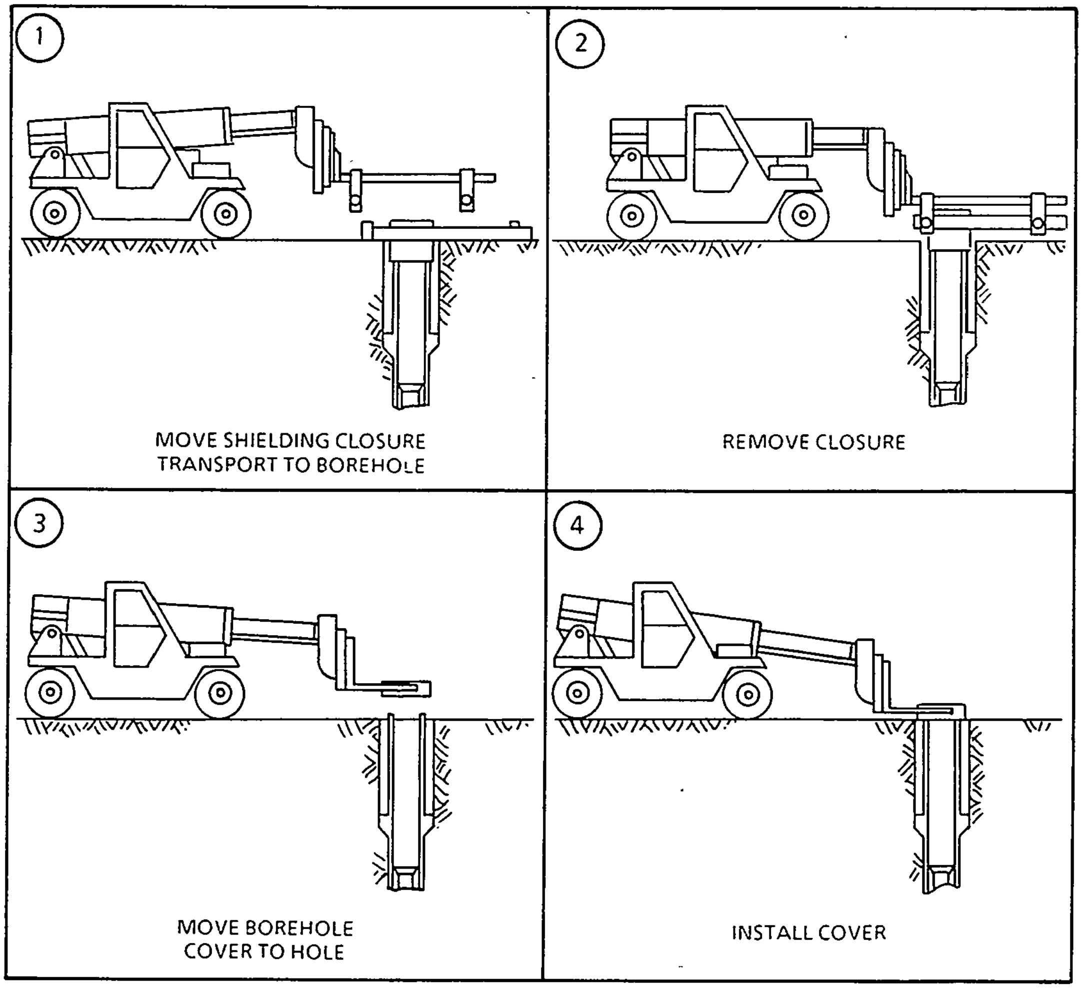


Figure 6-41. Closure of a vertical borehole after waste removal.

6.2.3.2.1.2 Horizontal retrieval

The basic operations required for the removal of a horizontally oriented waste container include the following steps: (1) preparing the horizontal borehole, (2) removing the waste container, (3) transferring the waste to the surface, and (4) closing the borehole. A description of these operations under normal conditions is provided by Stinebaugh et al. (1986) and illustrated in Figures 6-42 through 6-45. Off-normal events for retrieval are discussed in Section 6.2.9.2.2.

6.2.3.2.2 Waste shipping

Figure 6-46 presents a block-flow diagram of the planned waste shipping operations, and Figure 6-47 illustrates the shipping operations. Waste shipping operations would begin when the waste containers are removed from the surface storage vaults in the waste-handling buildings. Spent fuel and other high-level waste packages may require additional containment before loading in a shipping cask. If required, this would be done in the waste-handling building. Waste containers not requiring additional containment would be loaded directly into shipping casks, and the casks would be placed on carriers.

Radiological surveys and security-related inspections would be performed on the casks and trailers or railcars before onsite transportation vehicles would be used to move the loaded trailers or railcars to a designated shipping area. The carrier would receive a final inspection at the gate before leaving the repository.

6.2.3.3 Accident analyses

Anticipated off-normal conditions that could occur during repository operations will be assessed by the design process in accordance with the methodology discussed in Section 6.1.4. These events include maximum credible natural phenomena, mechanistic failures associated with waste-handling operations, and other man-caused events that could cause release of radioactive materials. Plans for conducting accident and safety analyses and assessments will be developed as the repository design progresses. These plans are discussed in Section 8.3.5.5.

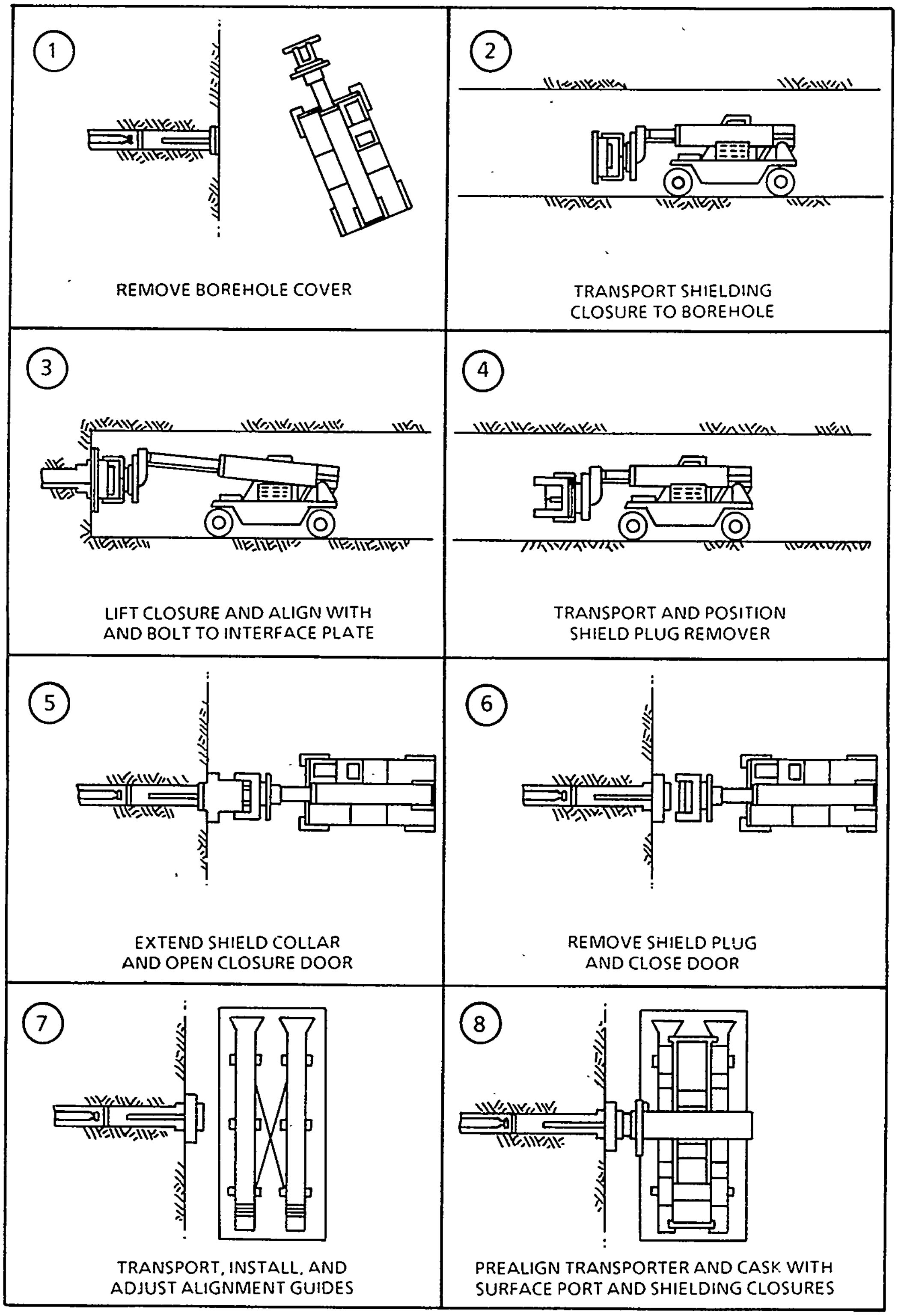


Figure 6-42. Preparation of a horizontal borehole for waste container removal.

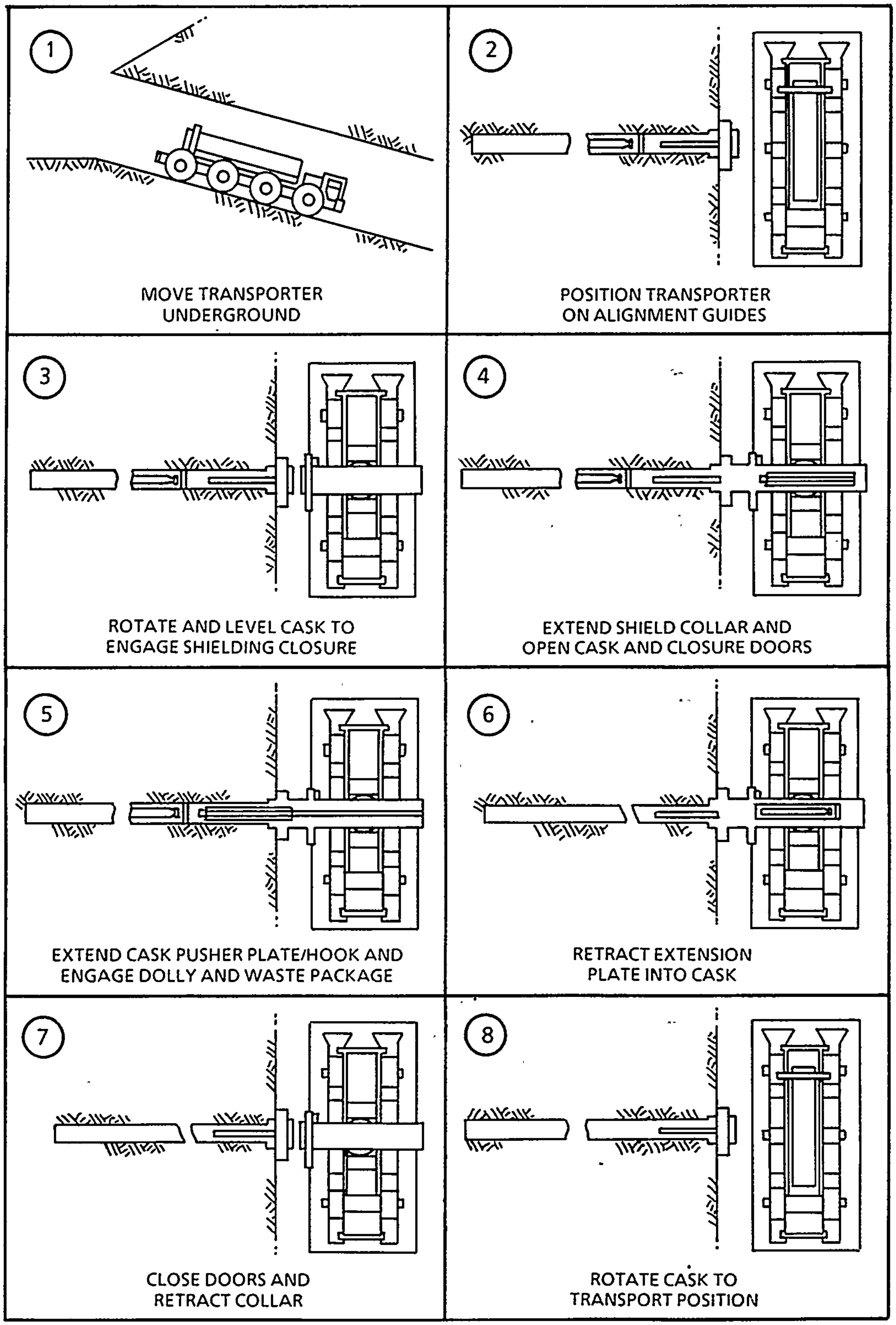


Figure 6-43. Removal of a waste container from a horizontal borehole.

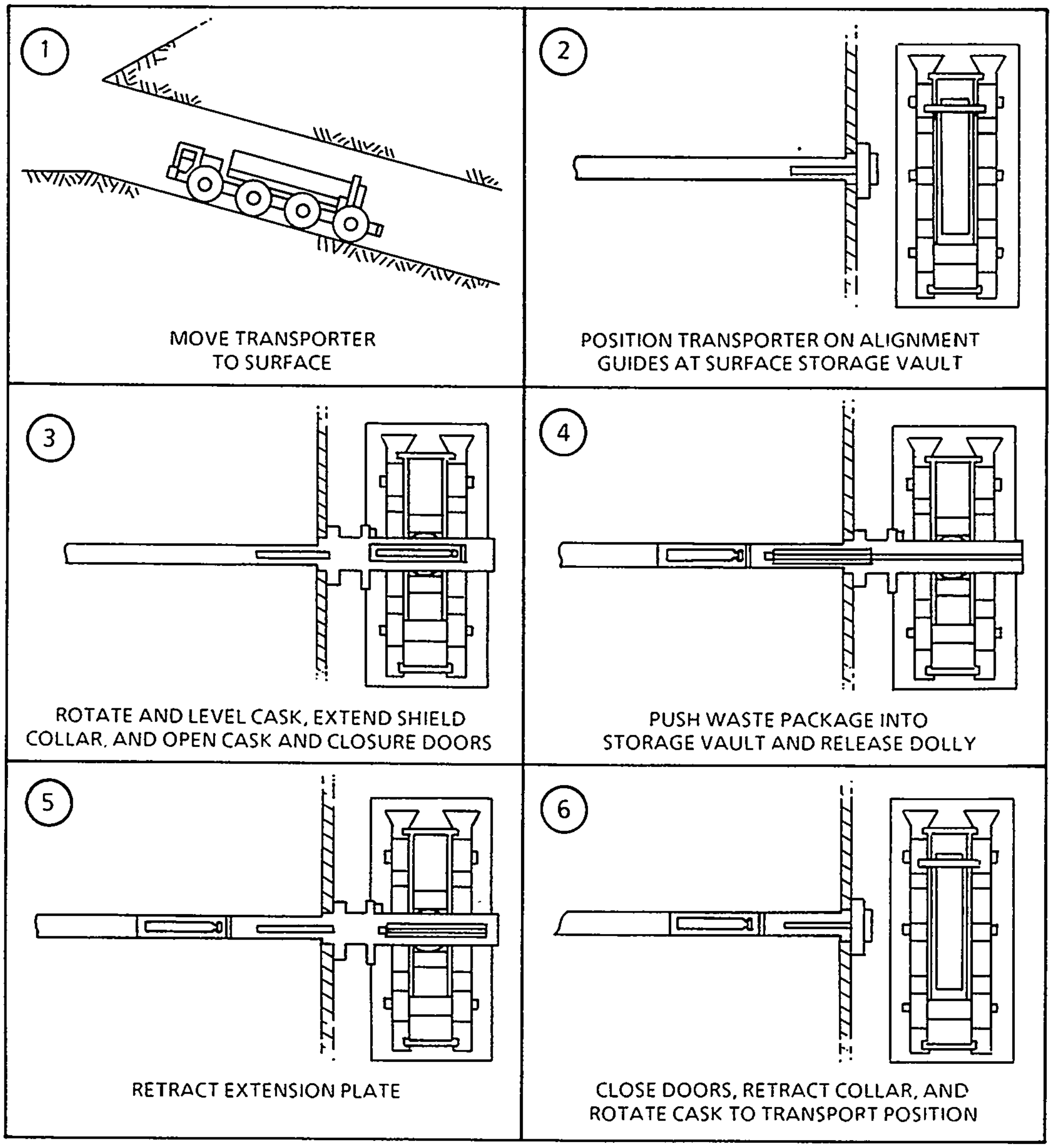


Figure 6-44. Transfer of a waste container from a horizontal borehole to surface storage.

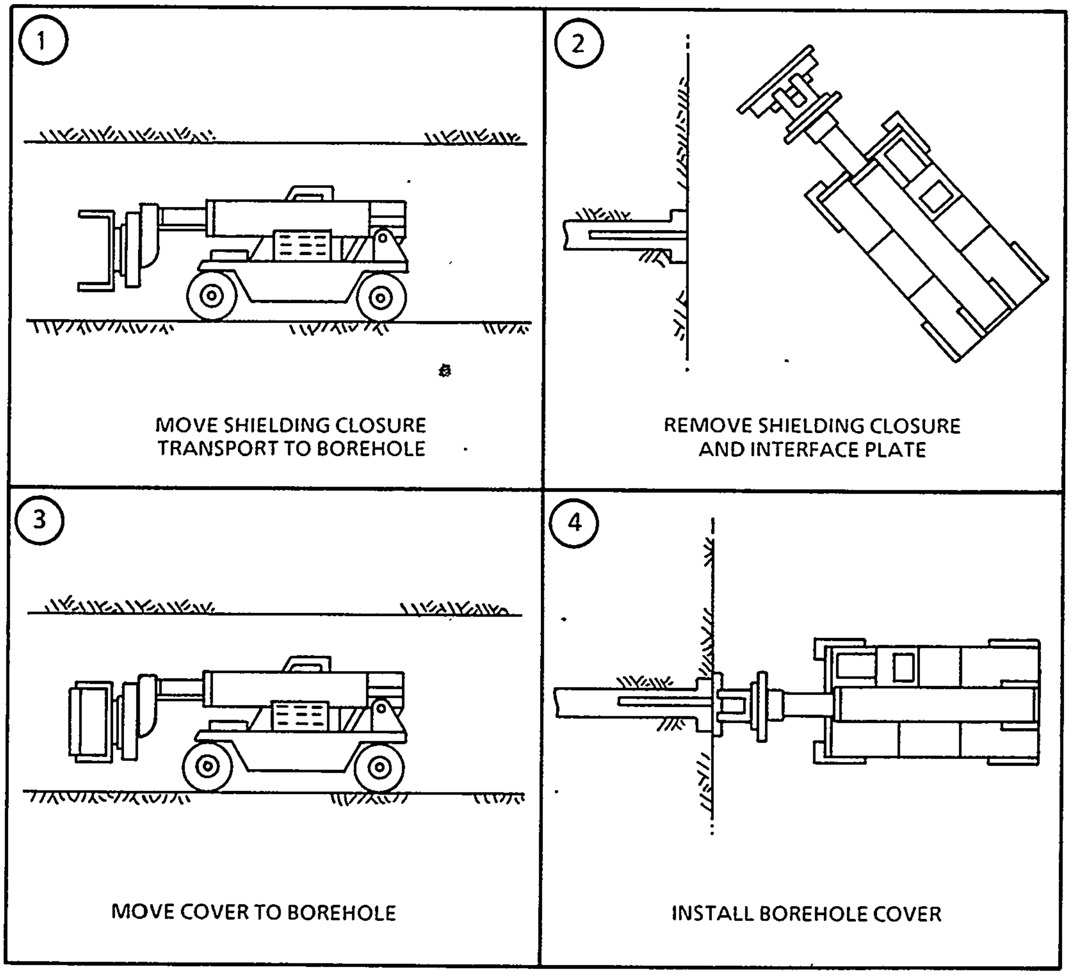


Figure 6-45. Closure of a horizontal borehole after waste removal.

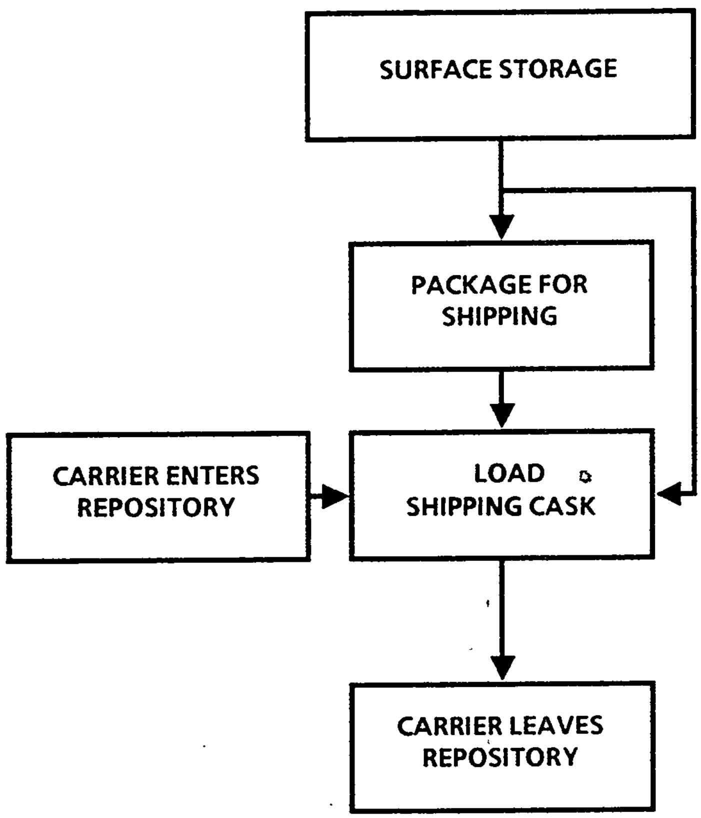


Figure 6-46. Flow diagram of waste shipping.

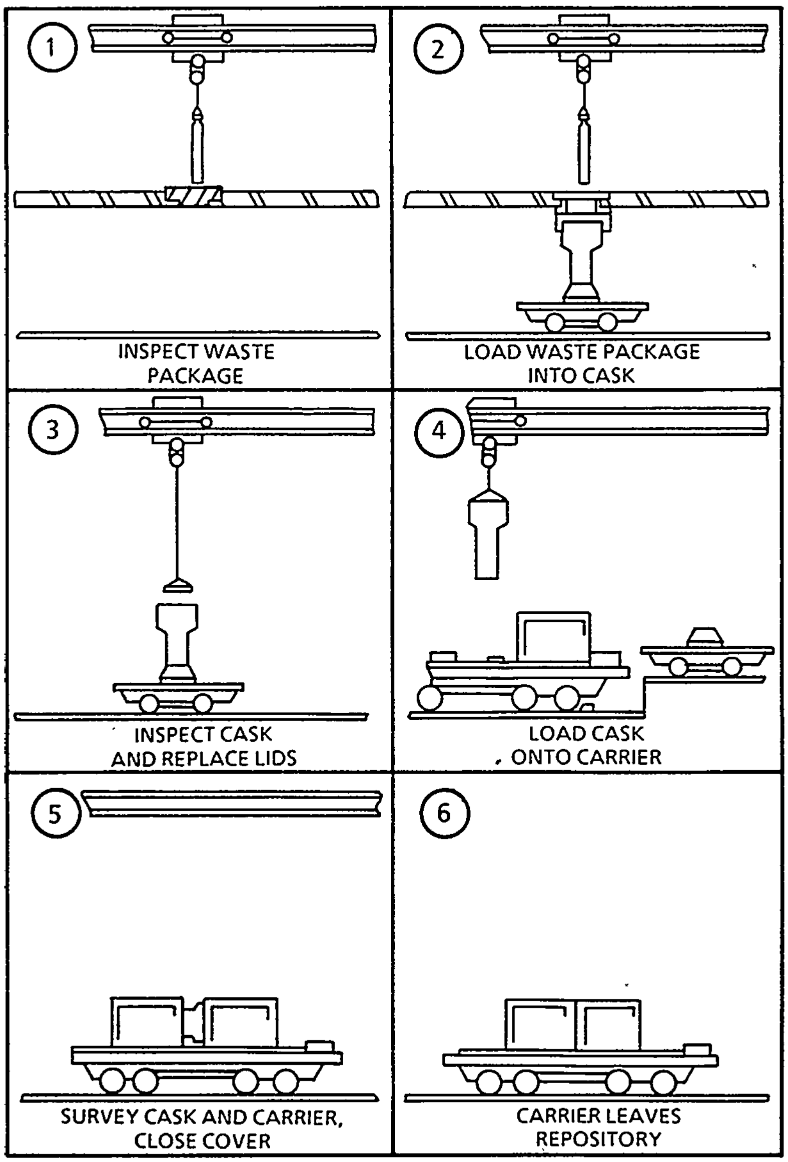


Figure 6-47. Waste shipping operations.

6.2.4 DESIGN OF SURFACE FACILITIES

The surface facilities at the repository have been designed to include a central surface facilities area, other onsite facilities that are not contiguous with the central facilities, and offsite transportation access as discussed in Section 4.2 of the SCP-CDR.

Surface facilities

The surface facilities at the repository would consist of a central surface facilities area, where waste handling and related support activities would occur, and numerous outlying support facilities and facilities that would provide access and ventilation for the underground portions of the repository (Figure 6-11, Section 6.2.2). The design and construction sequence of these facilities assumes development of the repository in two stages (Section 6.1.1.6.1). The location and layout of the surface facilities are governed by the functions they perform, by topography, and by requirements for integration with the subsurface facilities.

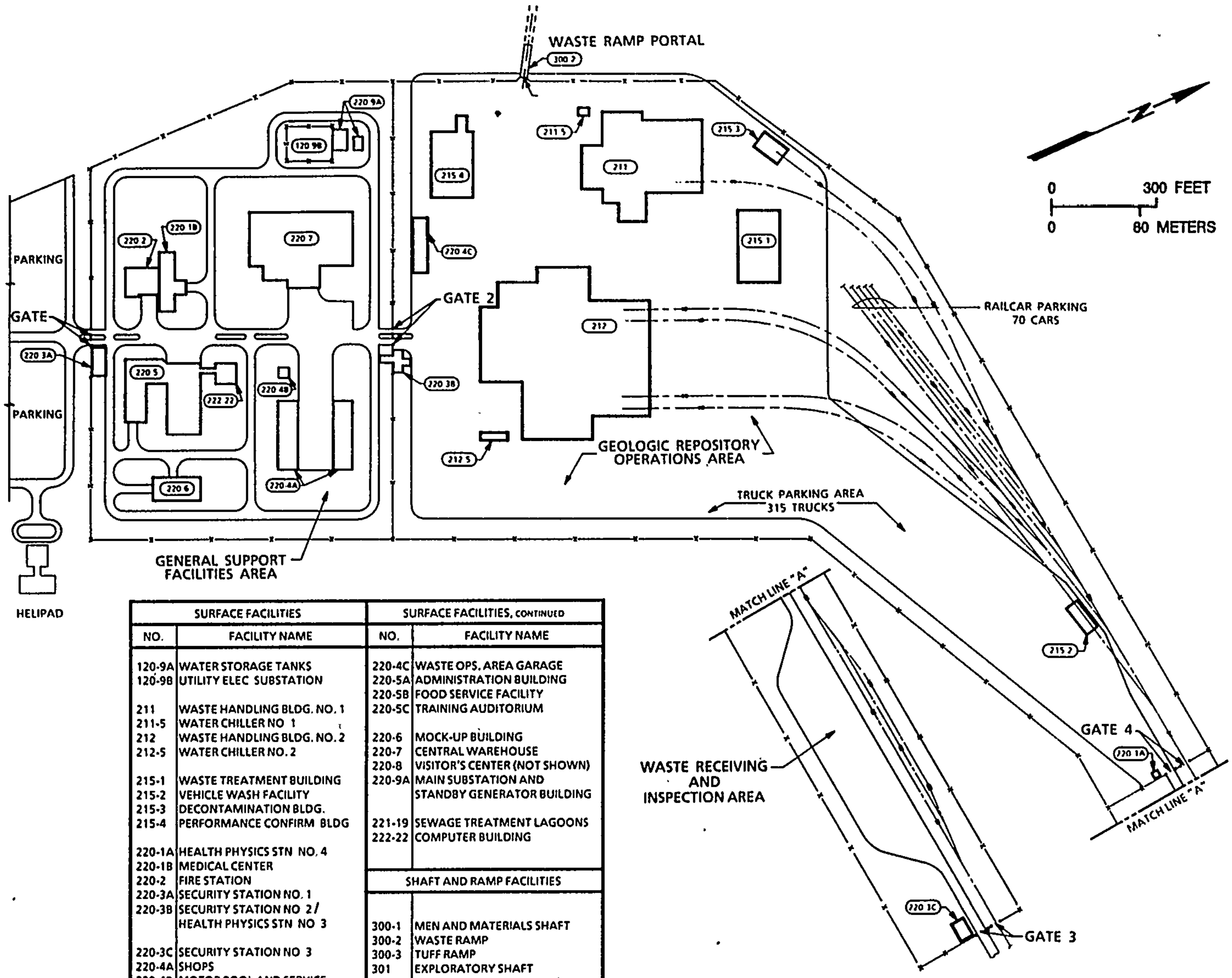
The central surface facilities area would be composed of three distinct functional areas--the waste-receiving and inspection area, the waste operations area, and the general support facilities area (Figure 6-48). Each area would be bounded by security fencing.

Radioactive waste would be shipped to the site either by rail or truck. The routes proposed for the new highway and railroad access to the site are shown in Figure 6-49.

In selecting locations for the proposed surface facilities, it was necessary to consider the siting requirements dictated by the layout and function of each facility, the location of related subsurface facilities, surface characteristics in the immediate vicinity of each facility, and general site characteristics such as access and surface drainage patterns.

A study was conducted to select a reference location for the central surface facilities to be used in developing the conceptual design (Neal, 1985). The areas considered in that study are located on the alluvial fans along the eastern base of Yucca Mountain. After an initial screening, the six areas shown in Figure 6-50 were selected for evaluation. The siting factors considered in the comparison of the six areas were identified based on the preclosure system and technical guidelines set forth in 10 CFR Part 960 for preclosure radiological safety, environmental quality, and ease and cost of construction, operation, and closure.

Numerical weighting and ranking methods were used to select the preferred site, an area east of Exile Hill (Figure 6-50). The primary advantages of this site are gentle slopes necessary for railroad construction, availability and contiguity of the area, protection from flash flooding, location adjacent to a rock outcrop suitable for constructing the waste ramp portal, and location near the northern edge of the repository, which allows flexibility for any necessary future expansion. Data obtained thus far (Neal, 1985) indicate that there are no conditions that would disqualify the area as a location for the waste-handling facilities. This is a preliminary



SURFACE FACILITIES		SURFACE FACILITIES, CONTINUED	
NO.	FACILITY NAME	NO.	FACILITY NAME
120-9A	WATER STORAGE TANKS	220-4C	WASTE OPS. AREA GARAGE
120-9B	UTILITY ELEC SUBSTATION	220-5A	ADMINISTRATION BUILDING
		220-5B	FOOD SERVICE FACILITY
211	WASTE HANDLING BLDG. NO. 1	220-5C	TRAINING AUDITORIUM
211-5	WATER CHILLER NO 1		
212	WASTE HANDLING BLDG. NO. 2	220-6	MOCK-UP BUILDING
212-5	WATER CHILLER NO. 2	220-7	CENTRAL WAREHOUSE
		220-8	VISITOR'S CENTER (NOT SHOWN)
215-1	WASTE TREATMENT BUILDING	220-9A	MAIN SUBSTATION AND STANDBY GENERATOR BUILDING
215-2	VEHICLE WASH FACILITY		
215-3	DECONTAMINATION BLDG.		
215-4	PERFORMANCE CONFIRM BLDG	221-19	SEWAGE TREATMENT LAGOONS
		222-22	COMPUTER BUILDING
220-1A	HEALTH PHYSICS STN NO. 4		
220-1B	MEDICAL CENTER		
220-2	FIRE STATION	SHAFT AND RAMP FACILITIES	
220-3A	SECURITY STATION NO. 1		
220-3B	SECURITY STATION NO 2/ HEALTH PHYSICS STN NO 3	300-1	MEN AND MATERIALS SHAFT
		300-2	WASTE RAMP
220-3C	SECURITY STATION NO 3	300-3	TUFF RAMP
220-4A	SHOPS	301	EXPLORATORY SHAFT
220-4B	MOTOR POOL AND SERVICE STATION	300-8	EMPLACEMENT EXHAUST SHAFT

Figure 6-48. Central surface facilities area.

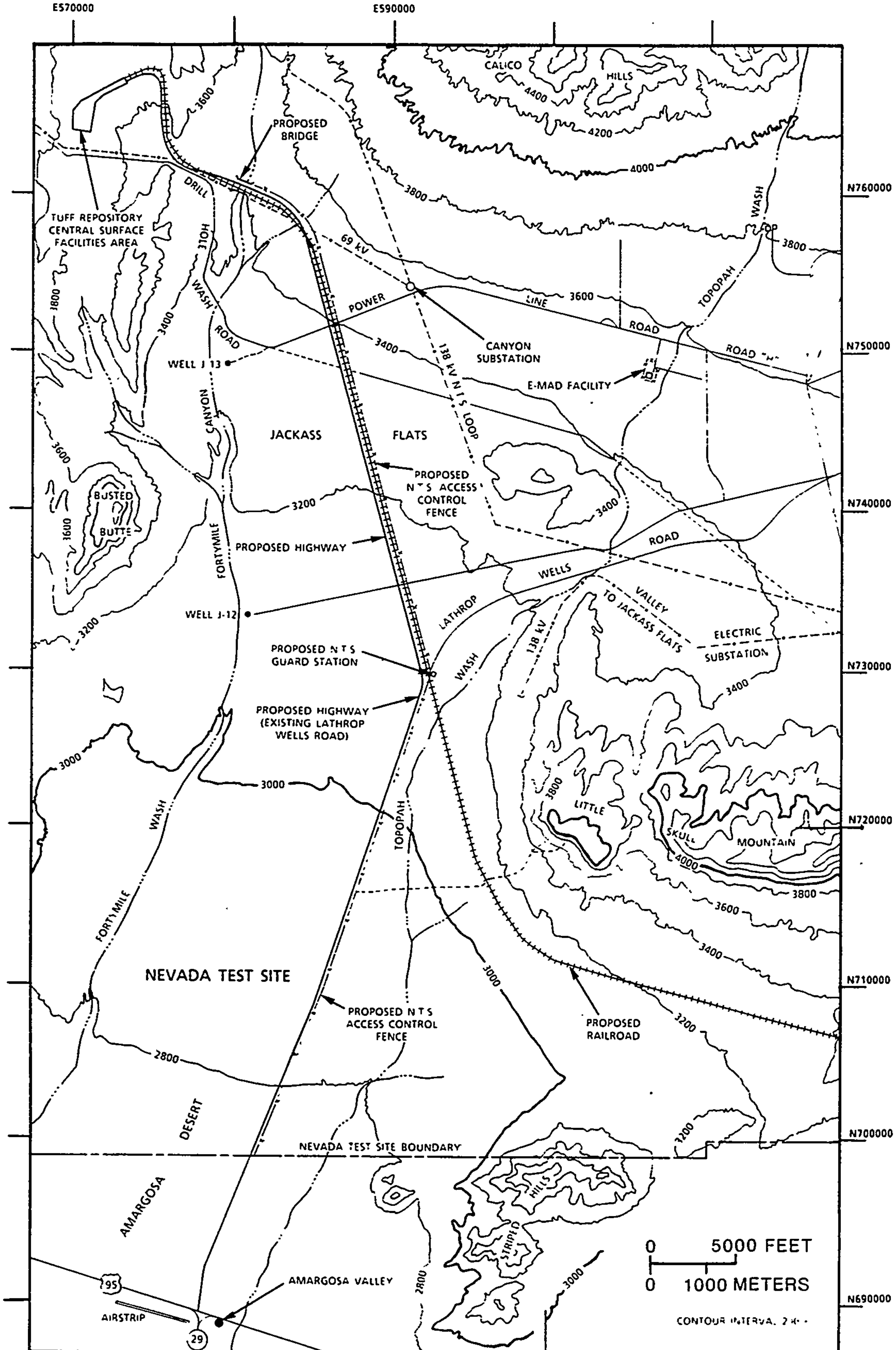


Figure 6-49. Route proposed for new highway and railroad access to the Yucca Mountain site.

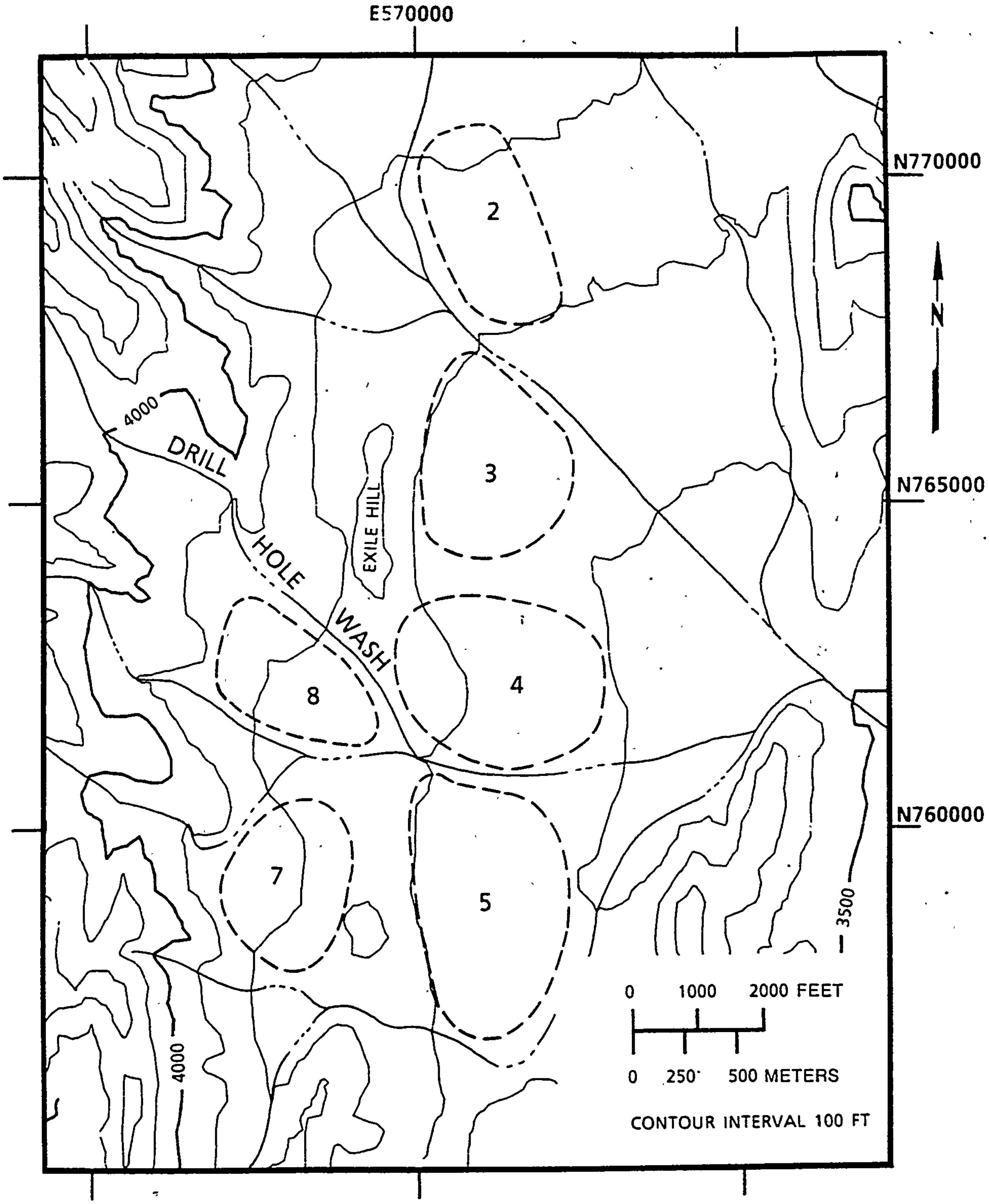


Figure 6-50. Locations of the six candidate areas for the surface facilities.

conclusion for this phase of the design and may be revised as a result of site characterization studies.

Underground accesses

The access to the underground portion of the repository would consist of two ramps and four shafts. Surface facilities would be associated with each of these accesses. The proposed location of these facilities is shown on Figure 6-11, Section 6.2.2.

The waste ramp would provide access for the waste transporter to the underground portion of the repository. The portal to that ramp would be located within the central facilities area. The proposed location of the intersection of the tuff ramp with the underground facilities was selected based on the underground layout and on proximity to a potential expansion area. The selection of preliminary locations for the portal of the tuff ramp and for the tuff pile was based on field observations of rock outcrops, which provide stable foundations for portal construction, and on the terrain in the vicinity of the portal. Runoff from precipitation would be intercepted by dikes, ditches, and liquid-collection sumps.

The men-and-materials shaft, the emplacement area exhaust shaft, the two shafts associated with the exploratory shaft facility (ESF), ES-1 and ES-2, and their related facilities, would be located 1 to 1.5 mi (1.6 to 7.4 km) west of the central surface facilities area. A road located in Drill Hole Wash would provide access to the shafts. Explosives magazines would be located approximately 0.5 mi (0.8 km) north of the men-and-materials shaft. All shaft sites would be bounded by security fencing and have level benches as shown in Figures 6-51 and 6-52.

The effect of natural forces (earthquakes, wind, tornadoes, floods) and human-induced phenomena, including underground nuclear explosions (UNEs) at the Nevada Test Site were addressed in the conceptual design process. The design criteria based on each phenomenon are given in Sections 6.1.1 and 6.1.2. More detailed evaluations of the effects of these phenomena, including more current site data, will be provided in future designs.

A two-stage approach to repository construction requires two waste-handling buildings in the waste operations area. During the first 3 yr, only the first waste-handling building (WHB-1) would be operational. During this period, construction of the second full-capacity waste-handling building (WHB-2) would be completed.

WHB-1, Figure 6-53, is designed to receive and prepare for subsurface disposal the equivalent of 400 metric tons uranium (MTU) per year of spent fuel. The two-stage approach calls for spent fuel assembly shipments to WHB-1 to be phased out when WHB-2 begins operating. WHB-1 would then handle only defense high-level waste and West Valley high-level waste and WHB-2 would be dedicated to handling spent fuel shipments. WHB-2, Figures 6-54 and 6-55, is designed to receive, consolidate, and prepare the equivalent of 3,000 MTU/yr of spent fuel for subsurface disposal and to package fuel assembly hardware generated during the consolidation process.

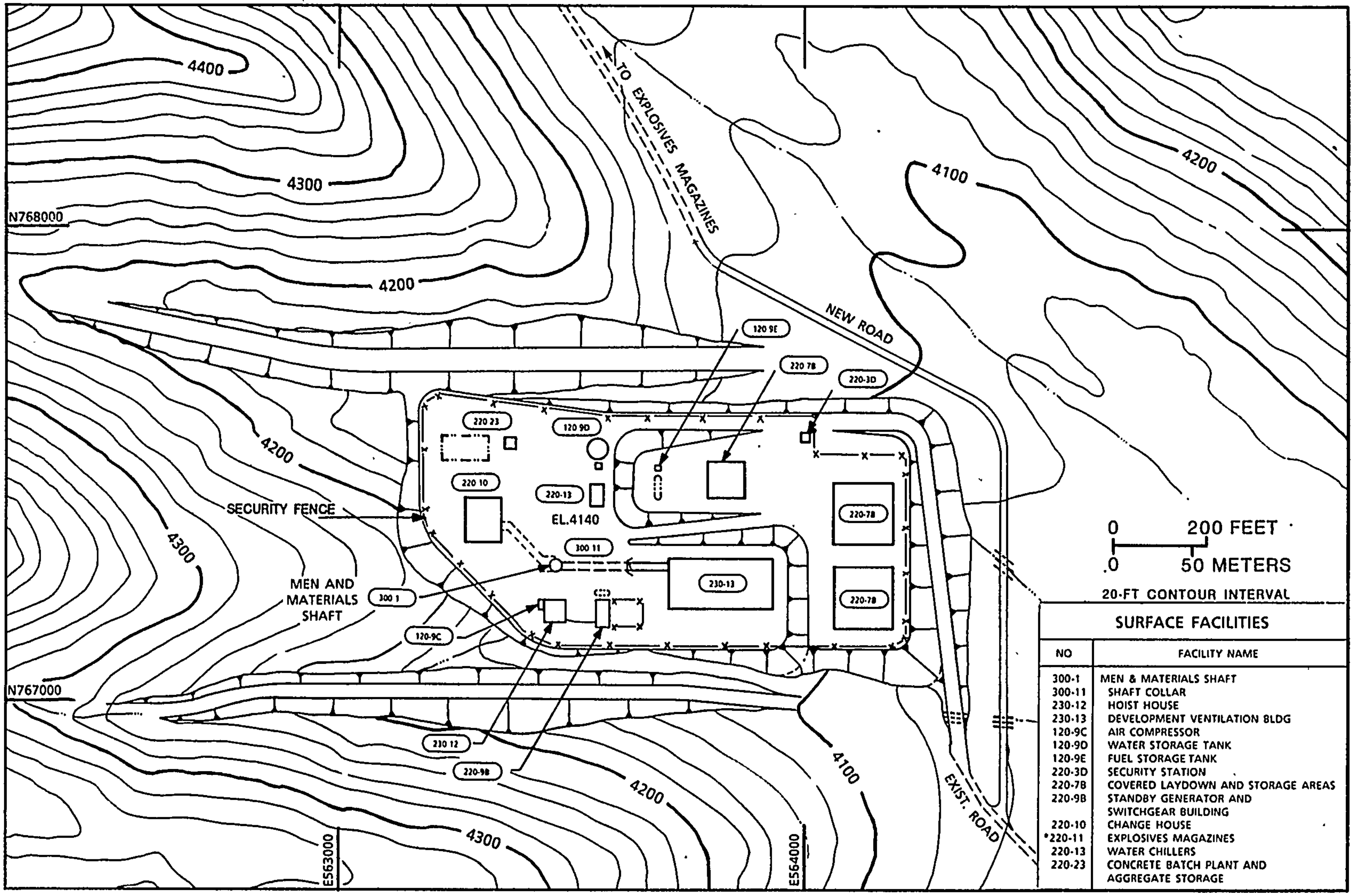


Figure 6-51. Surface facilities (large scale) of shaft sites.

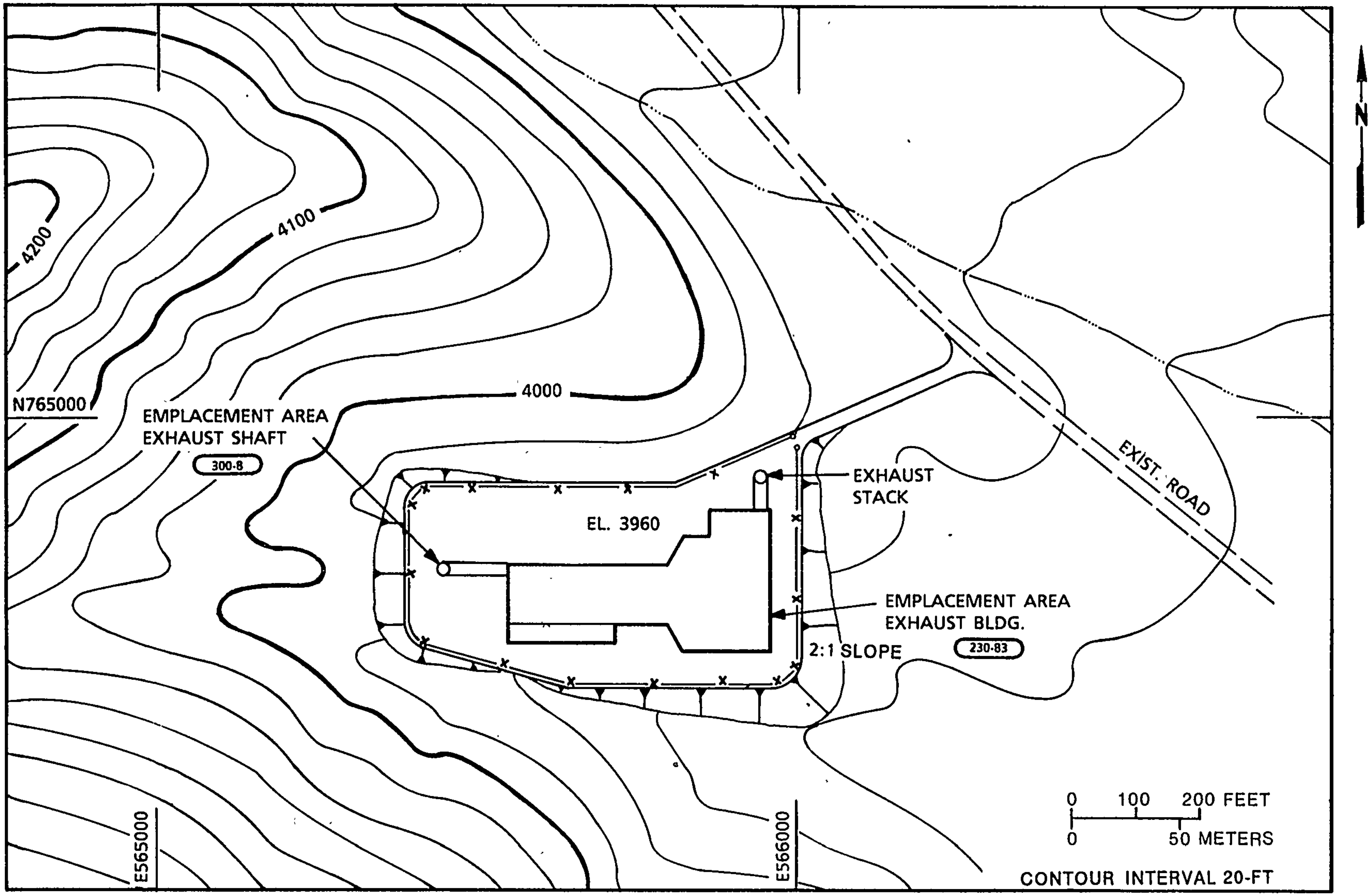
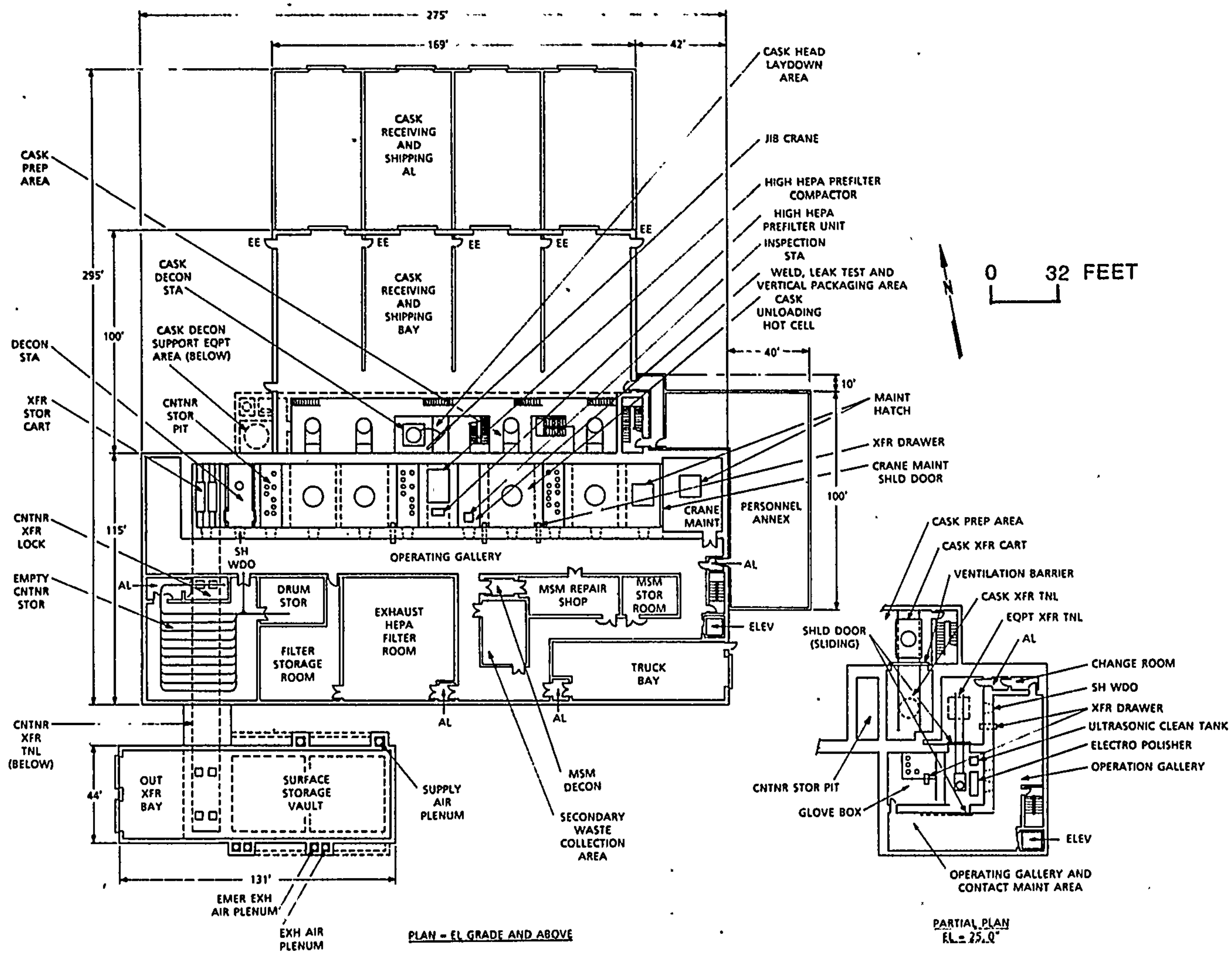


Figure 6-52. Surface facilities (small scale) of shaft sites.



KEY TO ABBREVIATIONS

AL	- AIRLOCK
ASST	- ASSISTANT
BRDG	- BRIDGE
CAL	- CALIBRATION
CHNG	- CHANGE
CNTNR	- CONTAINER
CMP	- COMPACTOR
CONT	- CONTROL
DECON	- DECONTAMINATION
EE	- EMERGENCY EXIT
ELEV	- ELEVATOR
ELEC	- ELECTRIC
ELMCH	- ELECTROMECHANICAL
EMER	- EMERGENCY
ENGR	- ENGINEER
EQPT	- EQUIPMENT
EXH	- EXHAUST
FLTR	- FILTER
HEPA	- HIGH EFFICIENCY PARTICULATE AIR
HVAC	- HEATING, VENTILATING, AND AIR CONDITIONING
INSP	- INSPECTION
INSTRM	- INSTRUMENTATION
JC	- JANITOR'S CLOSET
MACH	- MACHINE
MAINT	- MAINTENANCE
MECH	- MECHANICAL
MSM	- MASTER SLAVE MANIPULATOR
OFCE	- OFFICE
PREP	- PREPARATION
RADN	- RADIATION
RM	- ROOM
RMT	- REMOTE
SH	- SHIELD
SHLD	- SHIELDING
STA	- STATION
STOR	- STORAGE
SUPV	- SUPERVISOR
T	- TOILET
TNL	- TUNNEL
UNL	- UNLOADING
WDO	- WINDOW
XFR	- TRANSFER

Figure 6-53. Waste handling building 1, general arrangement.

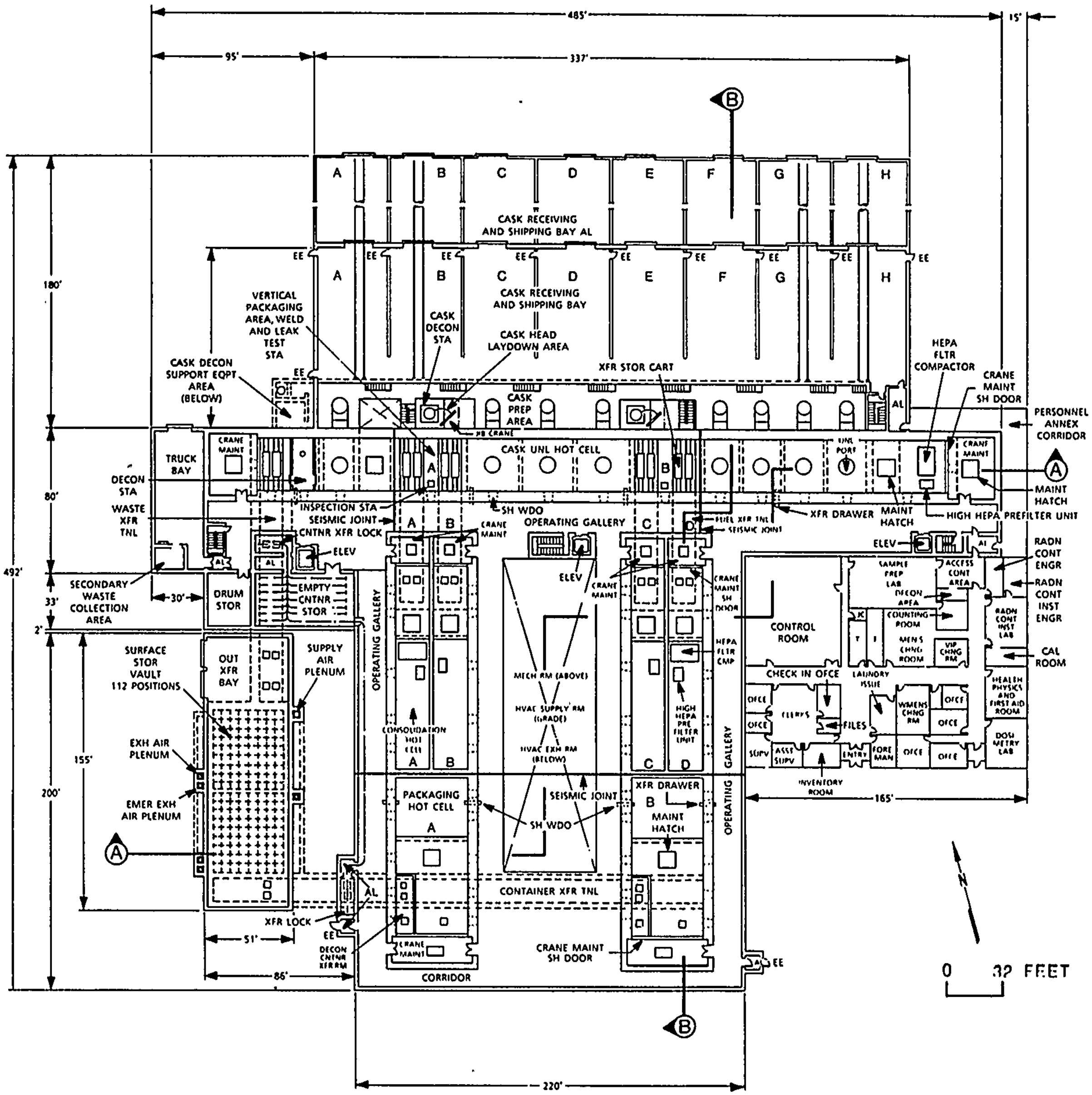
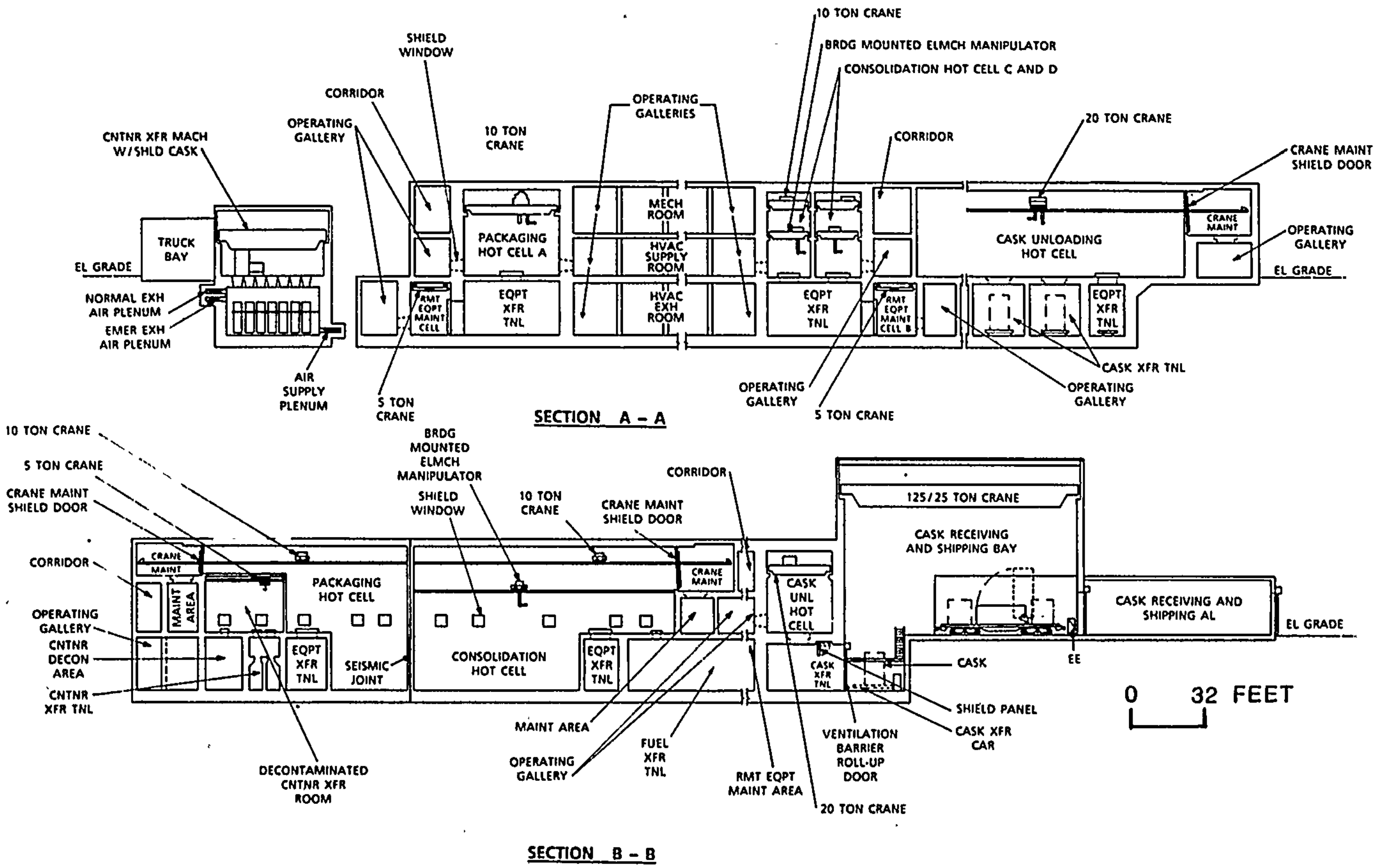


Figure 6-54. Waste handling building 2, preliminary general arrangement.

KEY TO ABBREVIATIONS

AL	-	AIRLOCK
ASST	-	ASSISTANT
BRDG	-	BRIDGE
CAL	-	CALIBRATION
CHNG	-	CHANGE
CNTNR	-	CONTAINER
CMP	-	COMPACTOR
CONT	-	CONTROL
DECON	-	DECONTAMINATION
EE	-	EMERGENCY EXIT
ELEV	-	ELEVATOR
ELEC	-	ELECTRIC
ELMCH	-	ELECTROMECHANICAL
EMER	-	EMERGENCY
ENGR	-	ENGINEER
EQPT	-	EQUIPMENT
EXH	-	EXHAUST
FLTR	-	FILTER
HEPA	-	HIGH EFFICIENCY PARTICULATE AIR
HVAC	-	HEATING, VENTILATING, AND AIR CONDITIONING
INSP	-	INSPECTION
INSTRM	-	INSTRUMENTATION
JC	-	JANITOR'S CLOSET
MACH	-	MACHINE
MAINT	-	MAINTENANCE
MECH	-	MECHANICAL
MSM	-	MASTER SLAVE MANIPULATOR
OFCE	-	OFFICE
PREP	-	PREPARATION
RADN	-	RADIATION
RM	-	ROOM
RMT	-	REMOTE
SH	-	SHIELD
SHLD	-	SHIELDING
STA	-	STATION
STOR	-	STORAGE
SUPV	-	SUPERVISOR
T	-	TOILET
TNL		TUNNEL
UNL	-	UNLOADING
WDO		WINDOW
XFR		TRANSFER

CONSULTATION DRAFT



KEY TO ABBREVIATIONS

AL	-	AIRLOCK
ASST	-	ASSISTANT
BRDG	-	BRIDGE
CAL	-	CALIBRATION
CHNG	-	CHANGE
CNTNR	-	CONTAINER
CMP	-	COMPACTOR
CONT	-	CONTROL
DECON	-	DECONTAMINATION
EE	-	EMERGENCY EXIT
ELEV	-	ELEVATOR
ELEC	-	ELECTRIC
ELMCH	-	ELECTROMECHANICAL
EMER	-	EMERGENCY
ENGR	-	ENGINEER
EQPT	-	EQUIPMENT
EXH	-	EXHAUST
FLTR	-	FILTER
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SUPV	-	SUPERVISOR
T	-	TOILET
TNL	-	TUNNEL
UNL	-	UNLOADING
WDO	-	WINDOW
XFR	-	TRANSFER

Figure 6-55. Waste handling building 2 - sections, preliminary general arrangement.

Approximate water consumption at the repository during the first 7 yr (including the construction period) is estimated to be 112,500,000 gal/yr (425,800 m^3/yr) and is expected to remain at this level for the next 25 yr. The average water demand for the following 23 yr of operation is estimated to be 2,500,000 gal/yr (9,460 m^3/yr). Present plans call for the construction of new water wells and storage provisions to be located at the proposed central surface facilities. The SCP-CD is based on the use of water, in the interim, from the existing well, J-13, located approximately 5 mi (8 km) southeast of the central surface facilities.

6.2.4.1 Foundation considerations

Foundations for the major surface buildings would be located in alluvial soil, except for some buildings adjacent to the shafts. The alluvial soil is a light tan to gray, silty to sandy gravel, with numerous blocky cobbles and boulders. These rock particles consist mostly of welded or partly welded volcanic ash-flow tuffs, derived from nearby bedrock sources.

Limited preliminary investigations of several exploratory borings and test pits were done between January and July 1984 in the six potential site areas for surface facilities, as shown in Figure 6-50.

Preliminary stratigraphic information has been developed from the exploratory boreholes and pits (Neal, 1985). The total depth of the alluvial soil at the proposed location of the central surface complex is about 90 ft (27 m); however, because the bedrock surface is sloping, the thickness of alluvium may be greater or less than this value, depending on the final location of surface structures.

The test pits were excavated in May 1984, to a depth of about 12 ft (3.7 m) below ground surface, and the soil conditions logged and sampled to obtain general physical and engineering characteristics and to estimate the possible variability among sites. Preliminary measurements of soil properties have been made on samples from the test pits. The surficial soil has been significantly modified by well-defined horizon development. The top 1 or 2 ft (0.3 or 0.6 m) (A and B horizons) of soil are loose and fine-grained; this soil would be removed during construction. The underlying material typically is partly cemented with calcite (caliche) to a depth of about 8 ft (2.4 m). Below that depth, the soil is not appreciably cemented or may be cemented only locally.

The foundations for principal surface buildings are expected to extend substantially below grade; the zone that included appreciable calcite cementation is probably too shallow to be considered for major foundations. Moreover, the degree of cementation and thickness of this zone is expected to be quite variable. Therefore, the conceptual foundation design is based on the strength and properties of the underlying uncemented material.

Preliminary measurements were taken on samples of the underlying uncemented soil from the test pits. The samples were taken at depths of 12 ft (3.7 m) or less. These measurements can be considered conservative estimates for properties of the deeper foundation soils, because soil strength normally increases with confining pressure, which increases with depth.

No direct measurements or tests of engineering properties of the proposed site soils have been completed; however, conservative estimates can be based on results of the index property tests and knowledge of the general behavior of the identified soils. The engineering properties given in Table 6-9 (Section 6.1.2.1.2) apply to uncemented soils below the zone of loose topsoil. Additional soils properties will be obtained, as described in Section 8.3.1.14 (surface characteristics) for use in surface facilities design described in Section 8.3.2.5 (preclosure design and technical feasibility).

6.2.4.2 Flood protection

Because of the rugged terrain and meteorological conditions at the Yucca Mountain site, brief, but intense localized precipitation occurs infrequently.

Flood protection was a consideration in choosing the proposed locations for surface facilities and shafts in the conceptual design. U.S. Geological Survey (USGS) topographic maps, with 20-ft contour intervals on a 1:24,000 scale, were used to choose the site. However, these maps do not provide the detail necessary for a final design based on probable maximum flood levels.

The flood history and potential are described in Section 3.2.1. In the analyses performed thus far, maximum flood flows, except for the men-and-materials shaft area described below, are derived from Crippen and Bue (1977), which contains graphs of peak discharges versus drainage areas of measured historical floods, with envelope curves above the plotted floods. The envelope curves represent the maximum potential flood for a given drainage area. The graph used for estimating maximum flows at the site is based on a region covering all or parts of Nevada, California, Utah, Arizona, and New Mexico. This methodology provides estimates of the flood discharges at the site suitable for conceptual design.

Future surface area design will be based on probable maximum flood (PMF) flows and levels, determined in accordance with ANSI/ANS 2.8 (1981). In general, the underground entries and surface facilities will be protected by providing channels and dikes to divert the surface runoff and by setting grades above the adjacent PMF levels (Section 6.1.2.6). This design effort will be finalized when definitive topographic maps and other data become available (Sections 6.3.7, 8.3.2.5, and 8.3.1.14).

A preliminary analysis of the PMF has been performed for the men-and-materials shaft area. This preliminary analysis was to evaluate the feasibility of locating the shaft and its supporting surface complex in this area. In the analysis, the PMF flows and levels were estimated, and flooding protection provisions were incorporated in the design. The men-and-materials shaft area would be benched, with the shaft entrance designed to be above the PMF levels in the diversion channels, located on the north and south sides, as well as the PMF level in Drill Hole Wash. From this preliminary analysis, it was concluded that the men-and-materials shaft may be adequately protected from PMF levels at the proposed location.

6.2.5 SHAFT AND RAMP DESIGN

Access between the surface facilities and the underground facility would be provided by shafts and ramps. The major functions of the shafts and ramps would be as follows:

1. Transfer of waste packages to the emplacement area.
2. Transfer of mining equipment to and from the underground facility.
3. Removal of mined tuff.
4. Transfer of construction materials and supplies and backfill materials to the underground facility.
5. Transfer of general supplies and test equipment.
6. Transfer of explosives.
7. Transfer of personnel.
8. Intake and exhaust of ventilation air.
9. Routing for utilities.

In determining the number, type, size, and location of the accesses needed at the Yucca Mountain repository, the following factors were considered for the conceptual design:

1. Personnel and operational safety.
2. Efficiency and effectiveness of operations, including transportation and ventilation functions.
3. Geology and natural phenomena.
4. Capital and operational costs.
5. Schedule.
6. Security.
7. Structural considerations for ramp and shaft collars.
8. Interaction between surface and subsurface facilities.

The proposed location of these accesses and their interrelationship with the underground facility are illustrated in Figure 6-56, and the number,

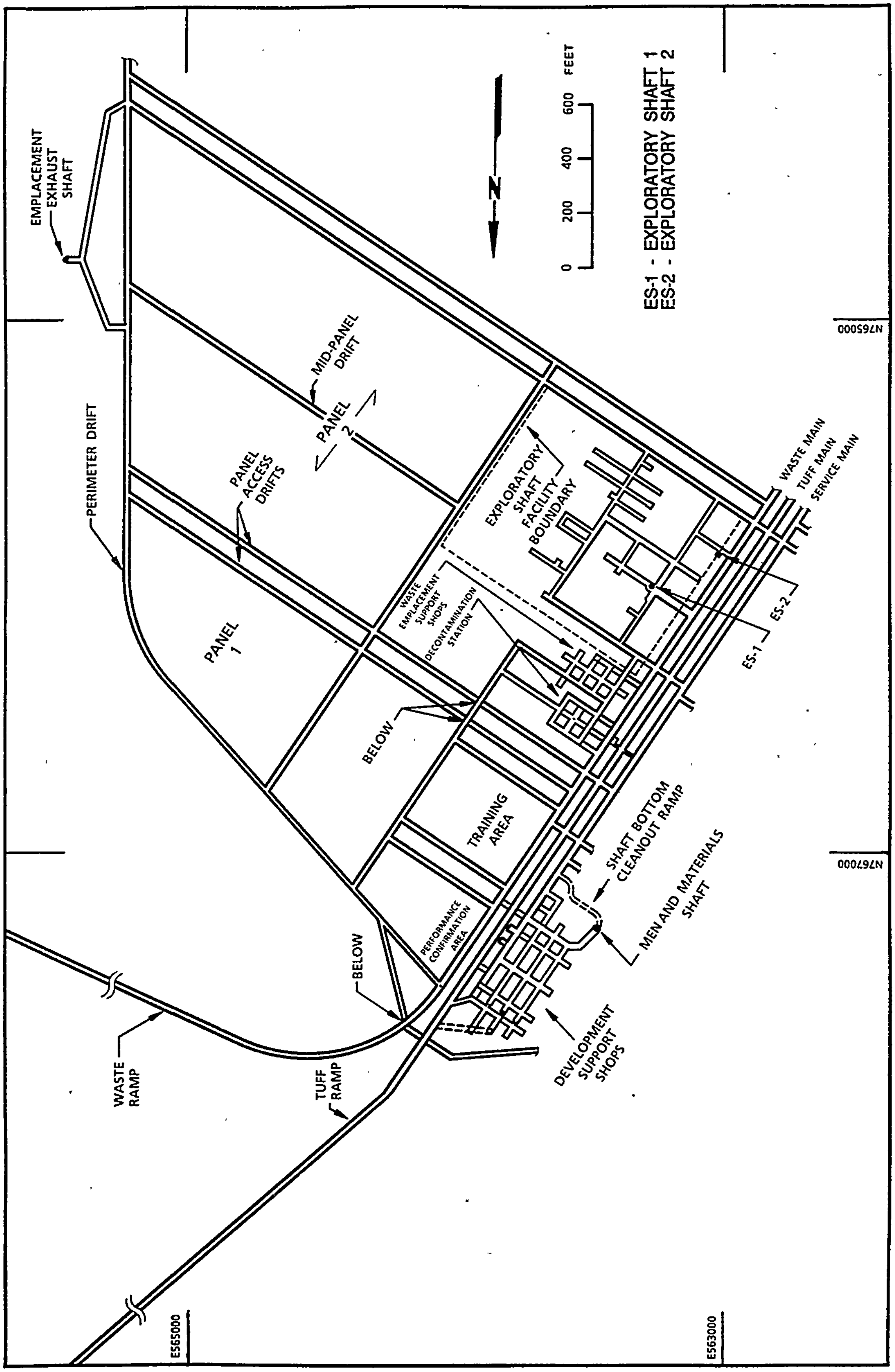


Figure 6-56. Location of shafts and ramps.

type, and size of the accesses are discussed in the following text. The types of accesses used in design are based on a study by Dennis and Dravo Engineers (1985).

6.2.5.1 Description of accesses

6.2.5.1.1 Waste ramp

The waste ramp would permit transport of the waste containers from the surface facilities to the underground facilities. The proposed location of the waste ramp was determined by grade limitations, desired location of the surface entry, and the proposed location of the surface facilities. The portal of the waste ramp would be physically separate from the waste-handling facilities on the surface. The portal of the waste ramp (Figure 6-48, Section 6.2.4) would be located in solid rock inside the boundary of the central surface facilities area. Significant data pertaining to the waste ramp are presented in Table 6-21. The waste ramp would be a fresh air intake for the waste emplacement activities.

6.2.5.1.2 Tuff ramp

The tuff ramp would be used for excavating and constructing the underground facility and for removing excavated tuff. The proposed location of the tuff ramp was determined by the desired entry point to the subsurface facilities, the proposed location of the waste emplacement area and the proposed location of the tuff pile. As currently located, the ramp portal would be in solid rock, easily accessible to the tuff pile. This location would allow the ramp to be a straight decline, minimizing transfer points on the conveyor belt between the development area and the surface. During operation of the subsurface facility, this ramp would have minimal usage for equipment transportation. The current design also calls for the tuff ramp to be the primary exhaust airway for the development area. Data for the tuff ramp are presented in Table 6-21.

6.2.5.1.3 Exploratory shafts

The locations of the two exploratory-shaft facility (ESF) shafts, ES-1 and ES-2, as used in the conceptual design are shown in Figure 6-56; their dimensions are given in Table 6-21. After completion of the site characterization program (Section 8.4), these two shafts would be used as air intakes for the waste emplacement area. ES-1 would bring fresh air to the waste emplacement area. ES-2 would serve as a fresh air intake for the shops in the emplacement area and for the underground decontamination facility; ES-2 would also serve as an emergency egress from the underground facility. Figure 6-57 illustrates the general arrangements and cross sections of all the shafts.

Table 6-21. Data for ramps and shafts

Opening	Elevation at collar (ft)	Elevation at collar (m)	Length or depth (ft)	Length or depth (m)	Slope (%)	Diameter (ft)[a]: Horizontal emplacement configuration (ft)	Horizontal emplacement configuration (m)	Vertical emplacement configuration (ft)	Vertical emplacement configuration (m)
Waste ramp	3,687	1,124	6,603[b]	2,013	8.9	21	6.4	23[e]	7
Tuff ramp	3,914	1,193	4,627[b]	1,410	17.9	21	6.4	25[e]	7.6
Exploratory Shaft 1	4,160[c]	1,268	1,480	451	(d)	12	3.7	12	3.7
Exploratory Shaft 2	4,160[c]	1,268	1,020	311	(d)	6	1.8	6	1.8
Men-and-materials shaft	4,140[c]	1,260	1,090	332	(d)	20[f]	6	20[f]	6
Emplacement area exhaust shaft	3,960[c]	1,207	1,030	314	(d)	20[f]	6	20[f]	6

[a]The dimensions of the ramps are excavated dimensions; those of the shafts are inside finished dimensions.

[b]Includes length of portal.

[c]Final construction grade elevation.

[d]All shafts are vertical.

[e]The diameter of the ramps is larger in the vertical configuration because ventilation airflow needed in this configuration is greater than that in the horizontal configuration.

[f]The diameters of the shafts are controlled by operational requirements rather than ventilation airflows.

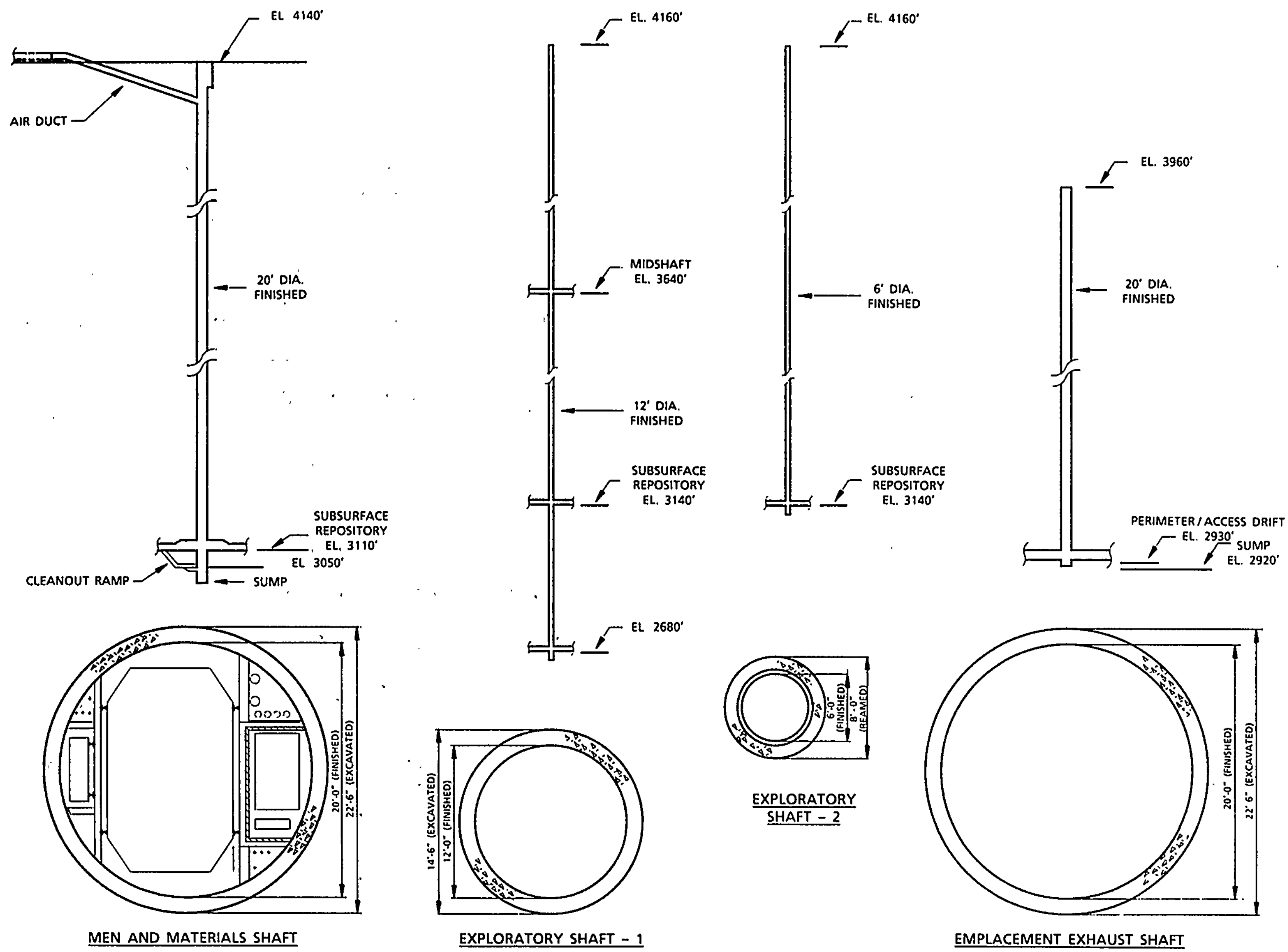


Figure 6-57. Shaft elevations and cross sections.

6.2.5.1.4 Men-and-materials shaft

The men-and-materials shaft (Figure 6-58) would provide access for men and materials and serve as an air intake for the development area. The proposed location of the shaft would provide convenient access to the development shops as well as to the remainder of the subsurface facilities. The proposed location of the surface opening of this shaft is shown in Figure 6-12 (Section 6.2.2). This shaft would be equipped with a men-and-materials cage, a service elevator, and would serve as an access for utilities. As shown on Table 6-20, the shaft is designed to have a finished diameter of 20 ft (6 m) and a total depth of 1,090 ft (330 m).

6.2.5.1.5 Emplacement area exhaust shaft

The exhaust shaft for the waste emplacement area would be located near the first panel just east of the perimeter drift, which would optimize its use as an exhaust pathway. The proposed surface location (Figure 6-12, Section 6.2.2) would provide a suitable location for the repository exhaust filtration equipment. The shaft is designed to have a finished diameter of 20 ft (6 m) and a vertical depth of 1,030 ft (314 m), Table 6-19, Section 6.2.5.1.1).

6.2.5.2 Construction and ground support

6.2.5.2.1 Sequence and methods of construction

6.2.5.2.1.1 Ramps

It is currently planned that the waste and tuff ramps will be excavated with a tunnel boring machine (TBM), which is a cost-effective method for ramp construction. After the TBM has progressed approximately two machine lengths down the ramp, the first portion of a conveyor would be installed. The conveyor would transport mined tuff from the TBM to the surface. This conveyor would be a temporary installation and would be removed after completion of the ramp. The excavated tuff would be transported to the tuff pile from the waste ramp portal by trucks and from the tuff ramp by conveyor belt. As the TBM progresses down the ramp, the conveyor would be extended, and the ground support and the ramp floor would be installed. Current ground support concepts for ramps are discussed in Section 6.2.6.1.4.

6.2.5.2.1.2 Shafts

The current design calls for the ES-1, the men-and-materials shaft, and the emplacement area exhaust shaft to be constructed using drilling and

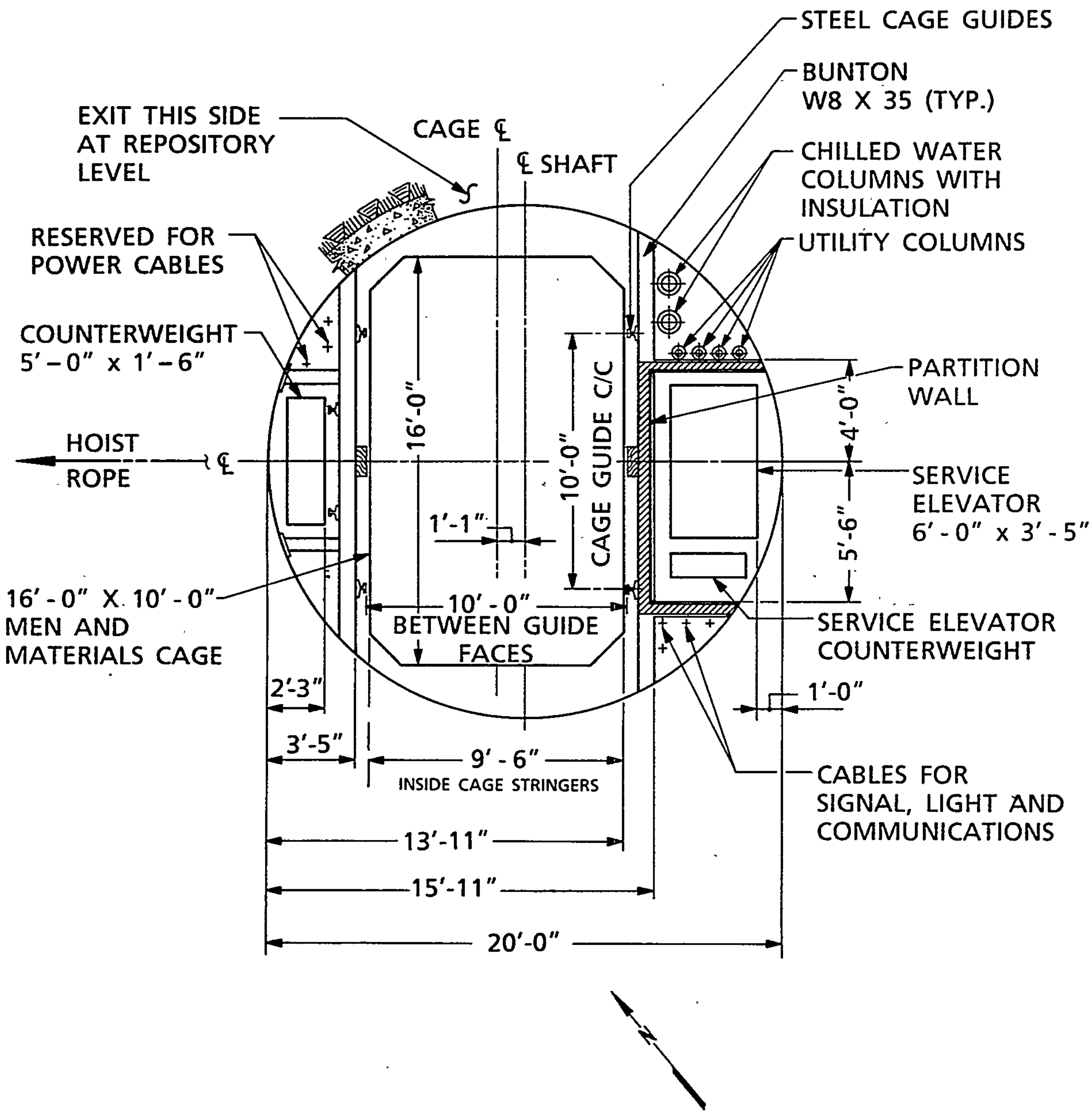


Figure 6-58. Cross-section of men and materials shaft.

blasting techniques. ES-2 would probably be constructed by raise-boring techniques. The proposed general arrangement of all shafts and current shaft lining concepts is illustrated in Figure 6-57 (Section 6.2.5.1.3).

6.2.6 SUBSURFACE DESIGN

The SCP-CDR (SNL, 1987) provides complete details about the underground repository design. The underground facility area for repository conceptual design is shown in Figure 6-59.

The conceptual design of the underground facility considered two waste emplacement orientations, vertical (the reference case) and horizontal. In the vertical orientation, a single waste container would be placed in a vertical borehole drilled in the drift floor. In the horizontal orientation, up to 14 spent fuel waste containers or up to 18 defense high-level waste containers can be emplaced in a long, horizontal borehole. The facilities for both orientations are summarized in this section and detailed in Section 4.4 of the SCP-CDR.

Description of the underground layout

The design of the underground layout is based on waste characteristics (Section 2.1 of SNL, 1987), geologic and other site characteristics (Section 6.1.2), and the requirements for total repository capacity and throughput (Section 6.2.3). A description of excavation and other development activities is presented in Section 6.2.6.1.

The proposed location for the underground facilities is within the Topopah Spring Member of the Paintbrush Tuff, specifically unit TSw2. Selection of this unit was based on hydrologic, geotechnical, and thermal criteria (Johnstone et al., 1984). The principal geologic structure considered in the conceptual design is shown in Figures 1-19 and 1-20. A primary area for the facilities was selected within the Topopah Spring Member, based on structural considerations. The boundaries for the primary area are illustrated in Figure 6-12 (Section 6.2.2). Any contiguous zones that meet regulatory, geologic, structural, engineering, and performance assessment requirements also may be used. However, the primary area contains sufficient thickness and area to accommodate the equivalent of 70,000 metric tons of uranium (MTU) waste.

The general underground facility layouts for the vertical and horizontal emplacement orientations are very similar. Figure 6-13 (Section 6.2.2) shows the underground facility layout for the vertical emplacement method, and Figure 6-14 (Section 6.2.2) shows the underground layout for the horizontal emplacement method. The emplacement panel is the primary component of the underground layout. Access to the panels would be provided by shafts, ramps, the main entry drifts, a perimeter drift, and panel access drifts. The perimeter drift, which would serve as an exhaust airway for the waste emplacement area, surrounds the entire underground facility. Shop areas areas would be provided in both the development and waste emplacement areas. The locations of the shafts and ramps are discussed in Section 6.2.5 and shown in Figure 6-56 (Section 6.2.5.1.3). The design life for all

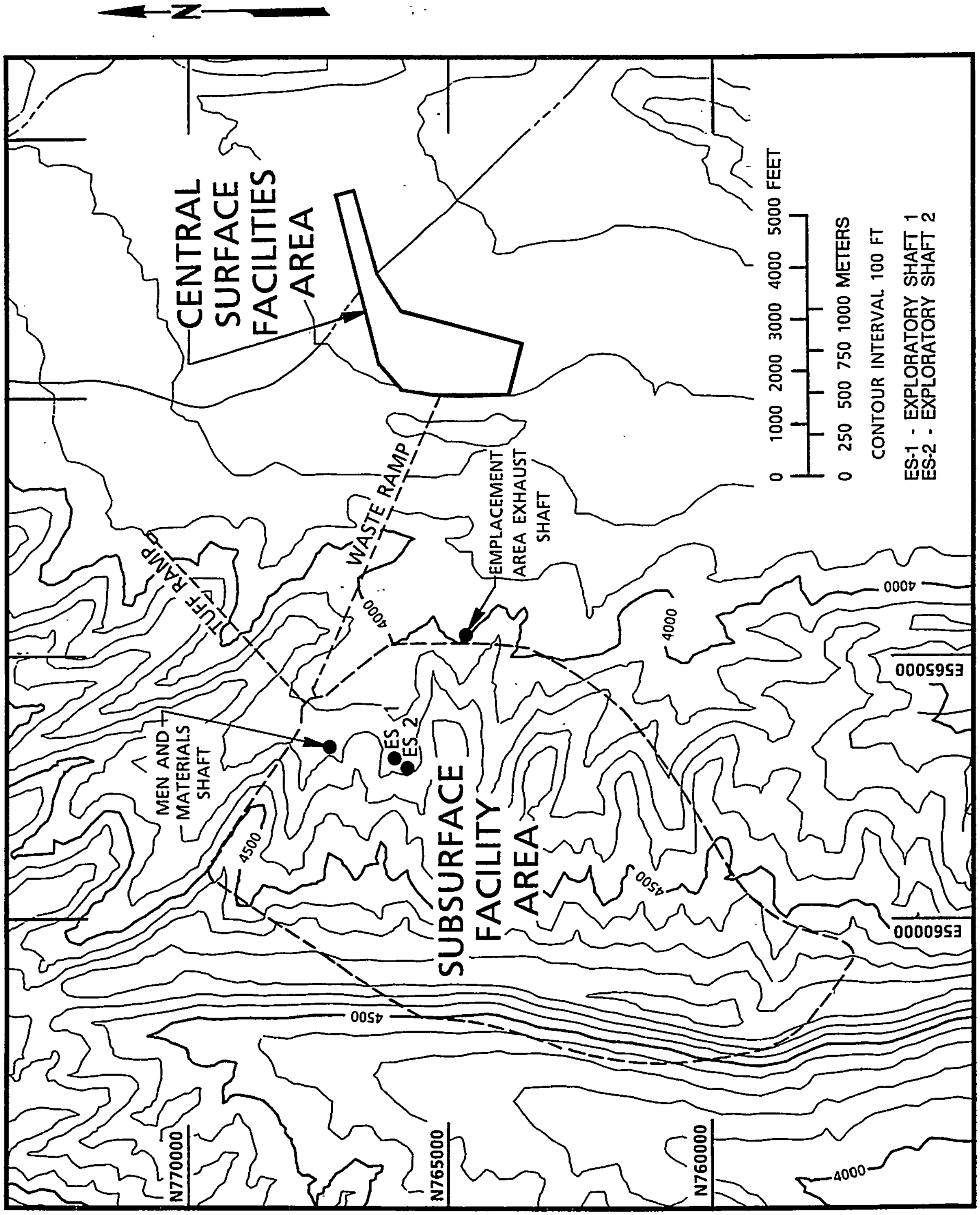


Figure 6-59. Underground facility area for the SCP-CD.

underground openings is 84 yr. The dimensions for all underground openings are shown in Figures 6-60 and 6-61, and the lining concepts for the shafts are shown in Figure 6-57 (Section 6.2.5.1.3).

Main entry drifts

Three parallel main entry drifts are planned to extend southwest through the underground facility to provide access to the emplacement panels during both the development and emplacement phases. Cross sections of these drifts are shown in Figures 6-60 and 6-61.

The waste main would be dedicated to transporting waste, the tuff main dedicated to transporting tuff and bulk materials, and the service main dedicated to ventilation. Space for electrical distribution systems would be provided in the service main. The layout and spacing of the main drifts allow separation of the development-area ventilation air from that of the waste emplacement area as detailed in Section 6.2.6.3.1. Table 6-22 presents the configurations for mains and for the perimeter drift for both vertical and horizontal emplacement options.

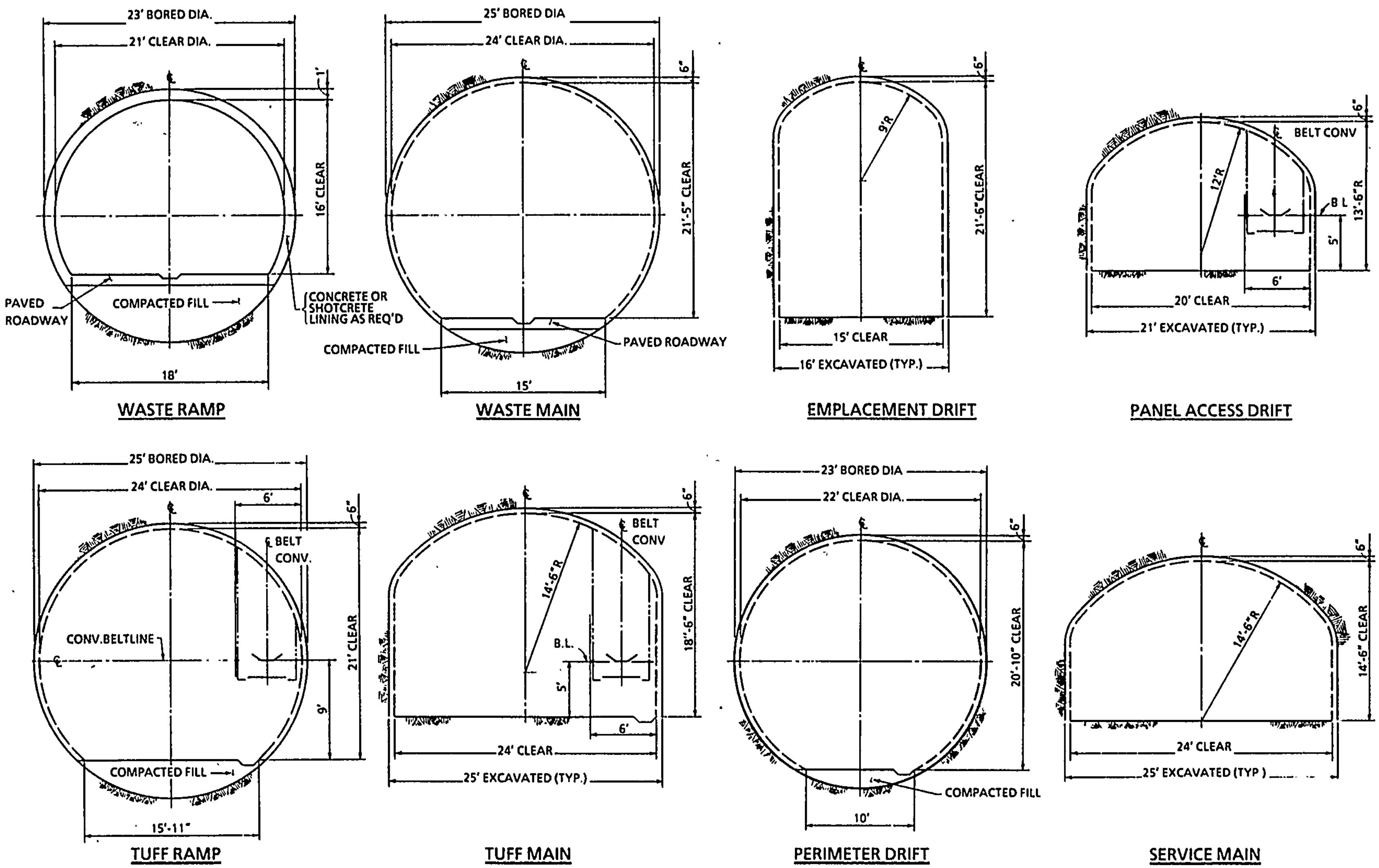
Table 6-22. Data for main and perimeter drifts

Opening	Vertical emplacement (ft)	Vertical emplacement (m)	Horizontal emplacement (ft)	Horizontal emplacement (m)
Waste main, diameter	25	8	21	6
Tuff main, width x height	25 x 19	8 x 6	25 x 19	8 x 6
Service main, width x height	25 x 15	8 x 5	25 x 15	8 x 5
Perimeter drift, diameter	23	7	23	7

The slope of the drifts was established using cross sections through the emplacement horizon. The slope of each drift was constrained by the upper and lower boundaries of the emplacement horizon and by the elevation at drift intersections.

Emplacement panels

The general repository layout is divided into panel segments. The emplacement panels are planned to be approximately 1,400 ft (427 m) wide, parallel to the main drifts, and 1,500 to 3,200 ft (457 to 975 m) long, perpendicular to the main drifts, for both the vertical and horizontal configurations. A typical panel would have an approximately rectangular shape.



DIA. - DIAMETER
TYP. - TYPICAL
CONV. - CONVEYOR

Figure 6-60. Drift and ramp cross sections for vertical emplacement.

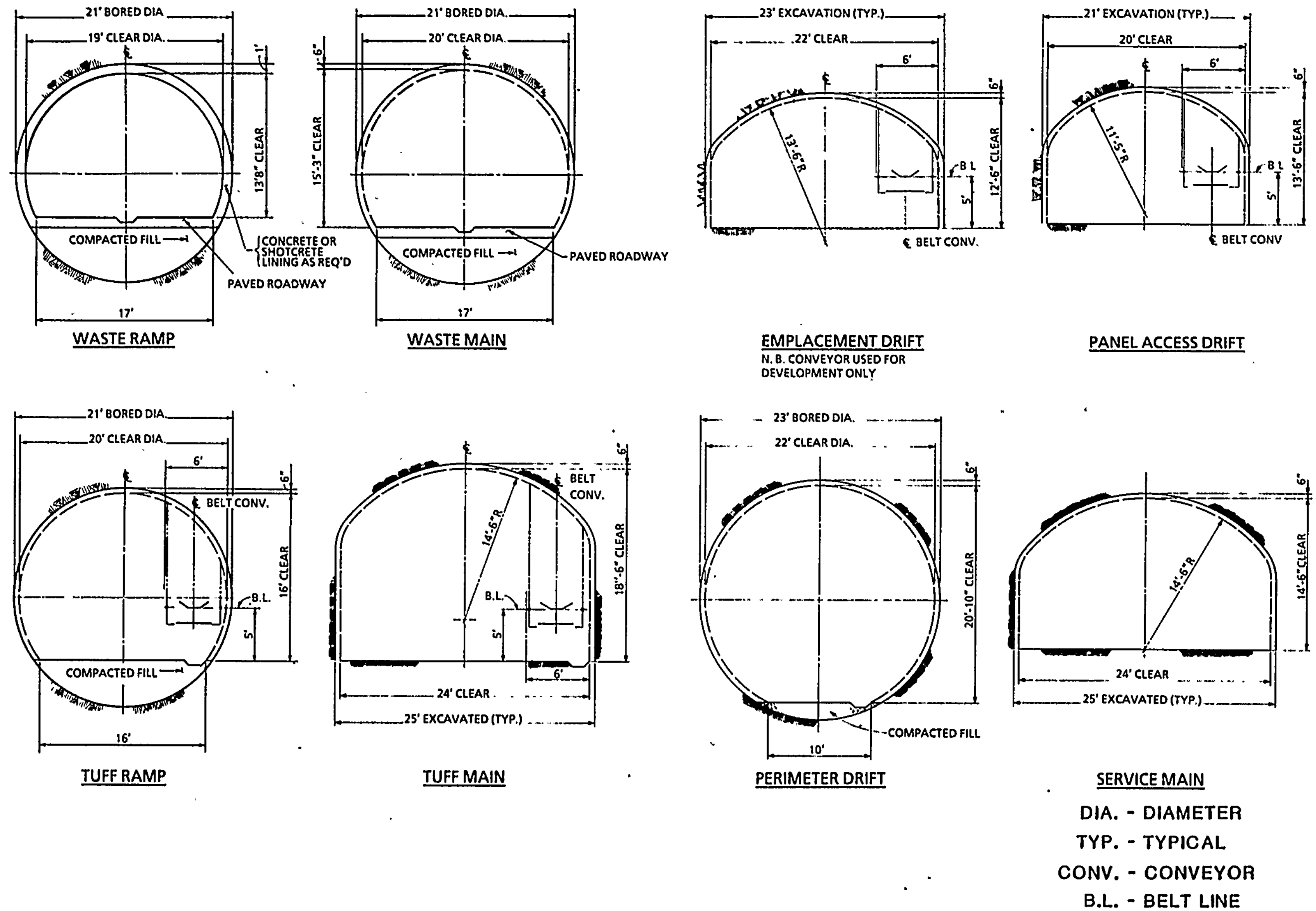


Figure 6-61. Drift and ramp cross sections for horizontal emplacement.

The methodology used to design the layout of each panel is similar in the vertical and horizontal emplacement methods. The layout for both the vertical and horizontal configurations consists of a sequence of emplacement panels. To accommodate the equivalent of 70,000 MTU of waste, the preliminary design of the panel layouts for both the vertical and horizontal waste emplacement configurations consists of 18 panels. These layouts reflect the use of an areal power density of 57 kW/acre.

Development of the panels would begin in the northeast corner and progress sequentially in a clockwise direction. Waste emplacement operations would follow in the same order. Waste emplacement would not begin until two full panels had been completely developed, providing separation between development and emplacement operations.

Description of the general layout for vertical emplacement

Figure 6-62 illustrates the details of a typical panel layout for the vertical emplacement. The panel width of 1,400 ft (457 m) was selected based on reasonable haulage distances for mined tuff removed. The design is based on an initial heat load of 3.03 kW per waste package (O'Brien, 1985). Flexibility in the methodology of the design allows for adjustments of the panel width.

A midpanel drift would serve as an airway for ventilation during development of the panel and for cooling during retrieval. Emplacement drifts (Figure 6-63) would connect the panel access drifts and would be spaced within the panel at intervals determined by the number of waste containers, borehole spacing, and standoff distances necessary to satisfy heat load criteria. The waste packages would be emplaced in 25 ft (8 m) vertical boreholes located in the floor of the emplacement drifts.

Standoffs, the distance from the access drifts to the first borehole in the emplacement drift, result from a design criterion to minimize temperatures in panel access drifts, the main entry drifts, and perimeter drifts. The current design provides for a minimum standoff of 85 ft (26 m). The access drift is currently designed to be 14 ft (4 m) high. The height of the emplacement drift is dictated by the dimensions of the equipment used for drilling the boreholes and emplacing the waste. To accommodate the transporter in the emplacement mode, the height of the emplacement drift would gradually increase from 14 ft to 22 ft (4 to 8 m) in the 85-ft (26-m) standoff distance. In the vertical configuration, the emplacement boreholes for defense high-level waste (DHLW) would typically be alternated with those for spent fuel.

Description of the general layout for horizontal emplacement

Figure 6-64 illustrates a typical panel layout for horizontal emplacement. In the horizontal waste emplacement configuration, waste packages would be in long horizontal boreholes drilled in the wall of the emplacement drifts (Figure 6-65). Up to 14 containers of spent fuel or 18 containers of DHLW could be emplaced in each borehole.

In the horizontal emplacement configuration, the panels would be 427 m wide, and the emplacement drifts would be 23 ft (7 m) wide by 13 ft (4 m)

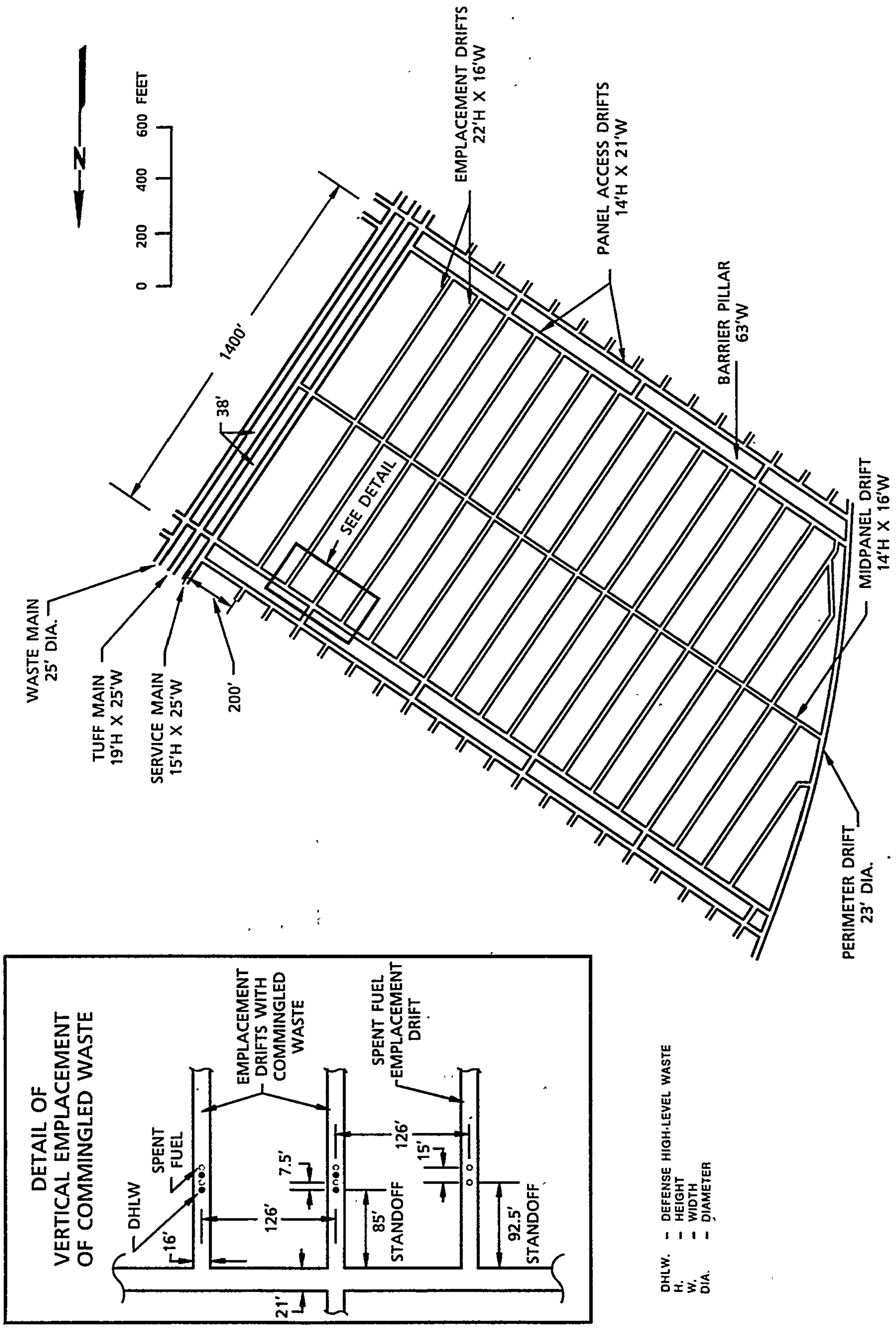


Figure 6-62. Typical panel layout for vertical emplacement.

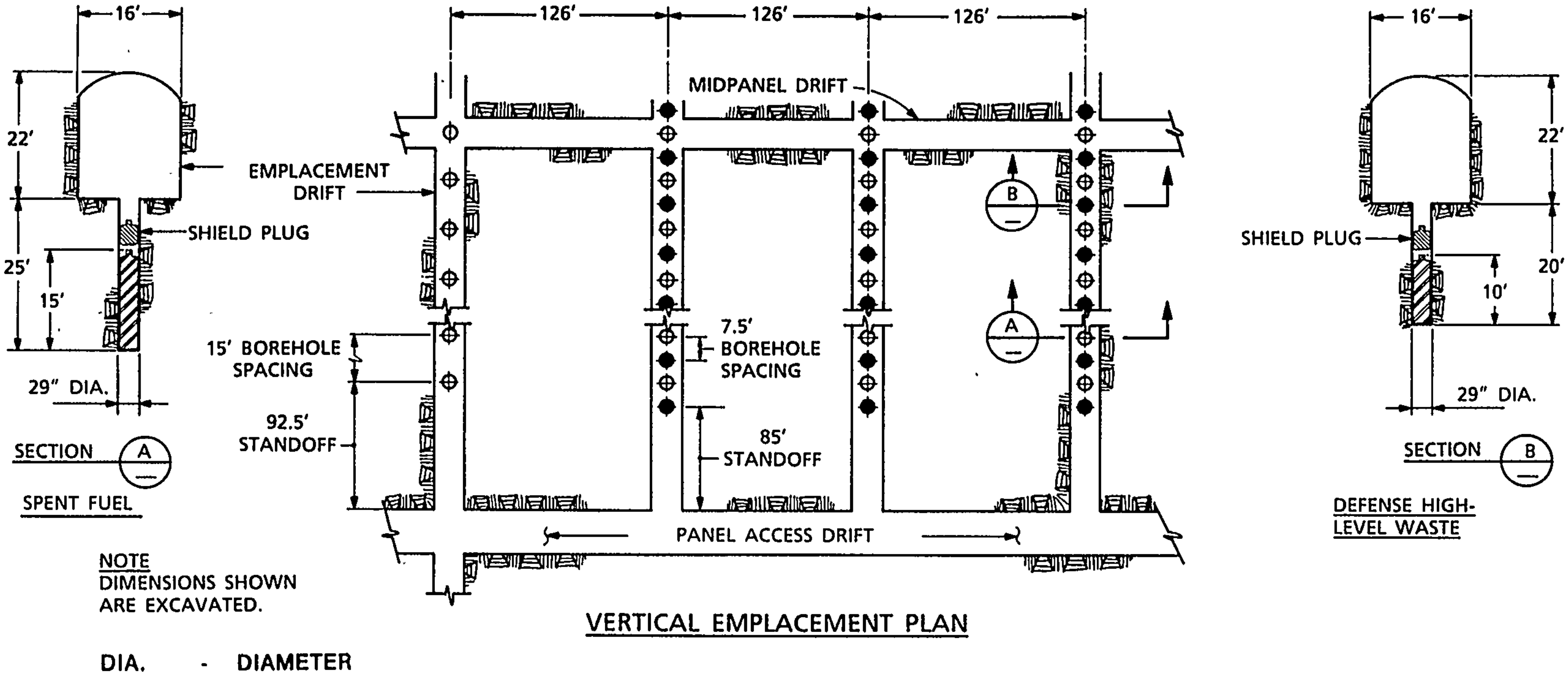


Figure 6-63. Panel details for vertical emplacement.

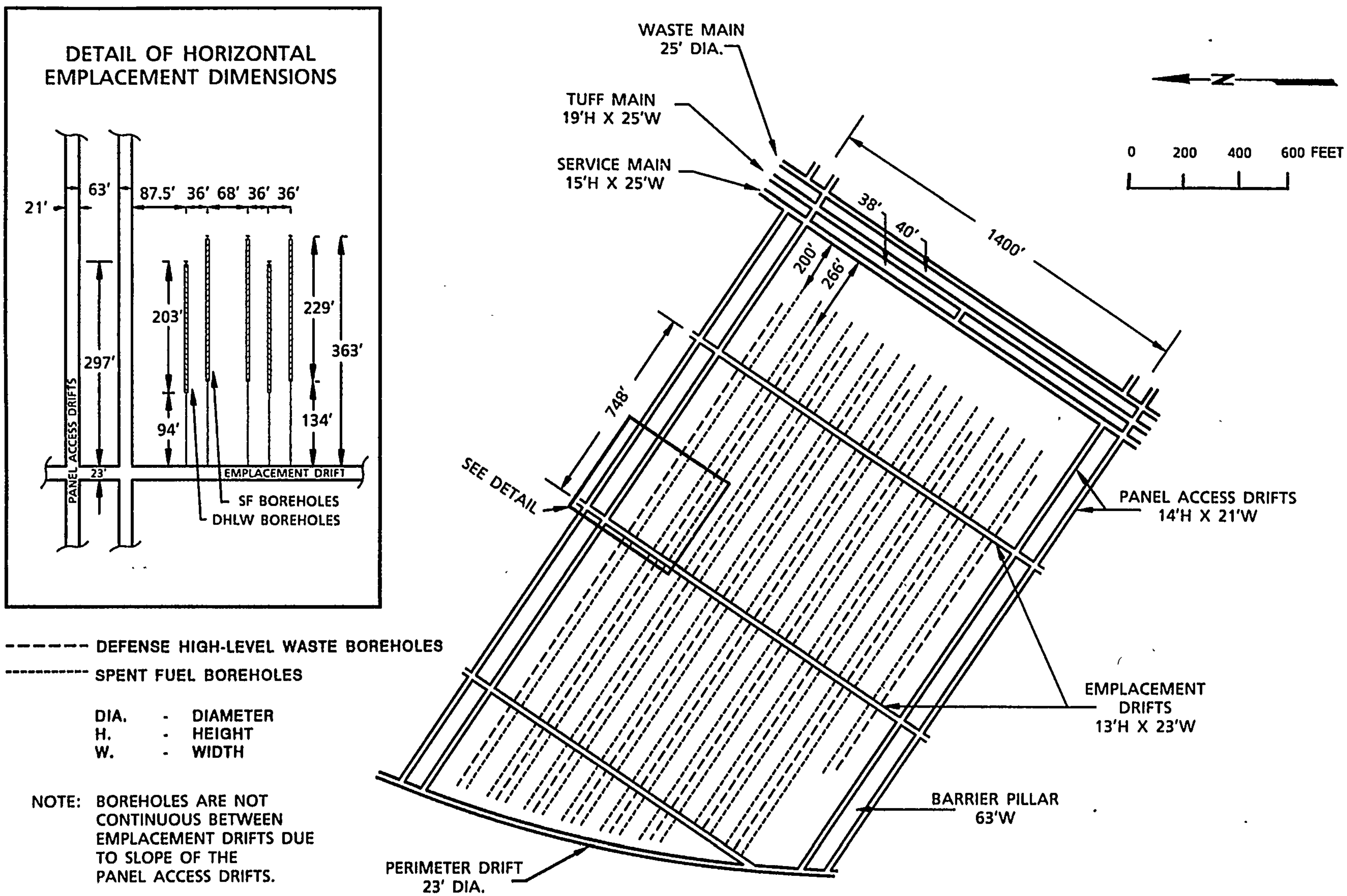


Figure 6-64. Typical panel layout for horizontal emplacement.

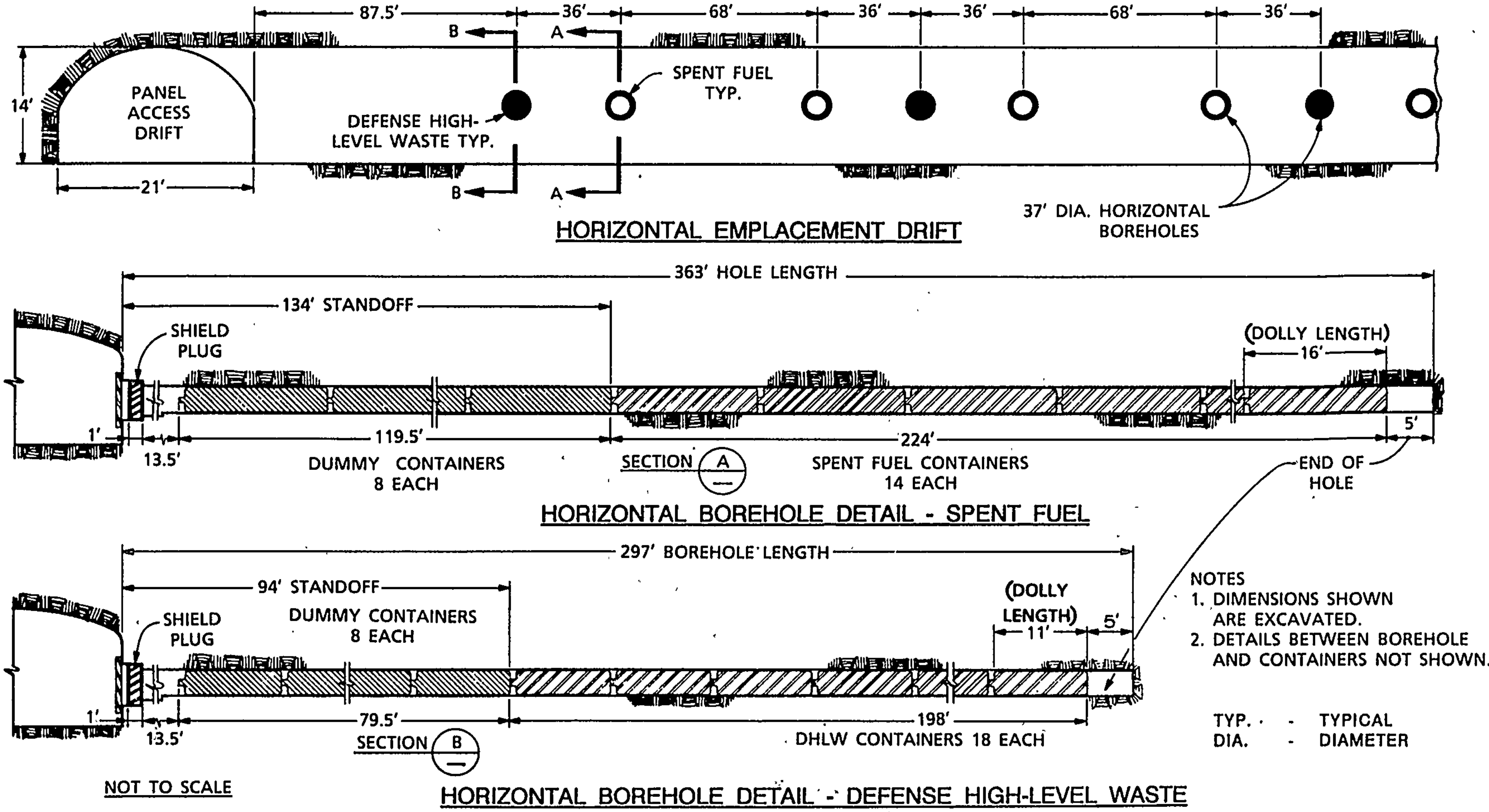


Figure 6-65. Panel details for horizontal emplacement.

high. The panel access drifts in the vertical configuration would be 21 ft (6 m) wide by 14 ft (4 m) high. No midpanel drifts are included in the design because the standoff distance between the waste containers and the drift in the horizontal configuration would reduce the temperatures in the emplacement and access drifts. Boreholes would be spaced within the panel at intervals determined by the number of waste containers and standoff distances necessary to satisfy heat load criteria. When required, the emplacement boreholes for DHLW would be alternated with the emplacement boreholes for spent fuel. The emplacement holes for DHLW and spent fuel would have the same dimension and lining.

Maintenance Shops and Service Areas

The current design calls for the shops in the development and emplacement areas to be located at the base of the waste ramp. These are shown on Figures 6-66 and 6-67. The underground service facilities would be equipped so that most preventive maintenance and minor repair functions could be performed underground. Sufficient spare parts and components for use in preventive maintenance would be stocked underground so that most repairs could be performed in a timely manner. Major rebuilding and overhauling would be performed in the maintenance facility on the surface.

An underground service area, to be located at the men-and-materials shaft, would contain a control room that houses the communications, ventilation, water, electrical, and ground-control monitoring systems. The control room would be the subsurface location for all repository monitoring and control systems. The main monitoring console for these systems would be on the surface.

Other facilities proposed to be located in the underground service areas are maintenance shops, space for equipment parking, bulk materials storage, supplies, service equipment, and space for administrative functions. Maintenance facilities, comparable to those in the underground service area, would be provided in the waste emplacement area.

Explosives would not be stored in the underground warehouse and shop area but would be stored on the surface where delivery would be controlled by warehousing personnel. Shortly before use, explosives would be brought underground and stored near the areas where they are planned to be used. Storage and transportation of explosives will follow the procedures described in 30 CFR Part 57.

6.2.6.1 Excavation, development, and ground support

Several types of openings to and within the underground portion of the repository are proposed to provide for development and waste emplacement activities. These proposed openings include two ramps, various types of drifts, four shafts, and the boreholes in which the waste containers would be placed. The proposed openings are identified as follows:

1. The waste ramp would link the central surface facilities with the subsurface facility. All waste to be emplaced in the repository

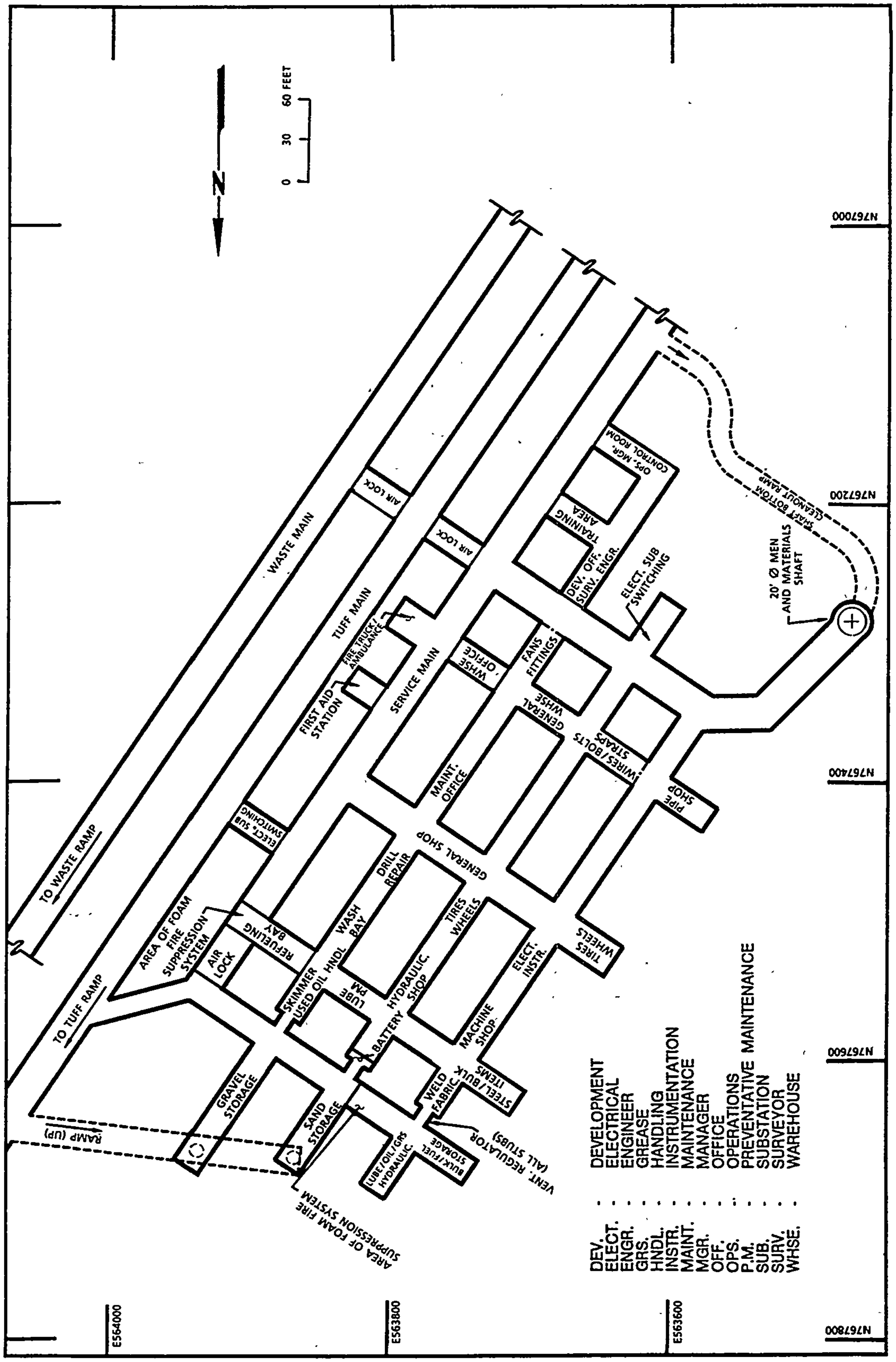


Figure 6-66. Development shops and warehousing.

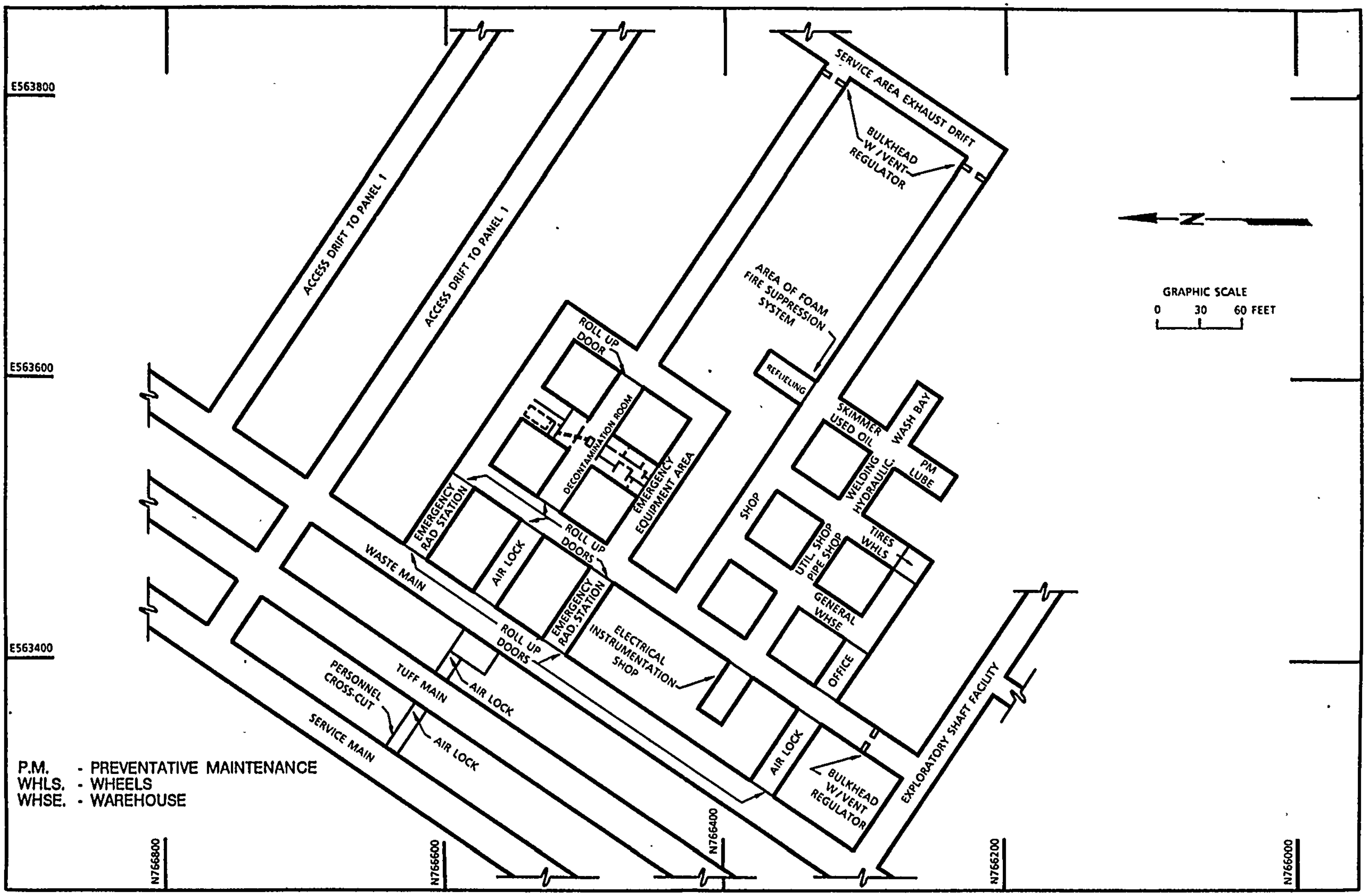


Figure 6-67. Emplacement shops, warehousing, and decontamination areas.

would be transferred from the surface to the underground facility through this ramp. The waste ramp would also serve as the air intake that supports waste emplacement.

2. The tuff ramp, which is the main haulageway for excavated tuff, would contain a belt conveyor and the main electrical feeder for the underground facility. A redundant electrical feeder would be located in the men-and-materials shaft. The current design also calls for the tuff ramp to be the primary exhaust airway for the development area. Traffic on the ramp would consist primarily of vehicles that service the belt and may occasionally be used to transport large machinery to and from the emplacement level.

3. The primary function of the service main would be to provide access for all development personnel, supplies, and machinery. The service main would also be used to carry ventilation air to the development area.

4. The waste main would connect the waste ramp with the access drifts and serve as the ventilation intake for the air circuit in the waste emplacement area. Together the waste ramp and waste main would form the sole access for the waste transporter.

5. The tuff main would be the main haulageway for excavated tuff at the emplacement level. All exhaust air from the development circuit would be exhausted through the tuff main.

6. The perimeter drift would be the exhaust airway for the air circuit in the waste emplacement area. Traffic through the perimeter drift would be restricted to inspection and maintenance vehicles.

7. The panel layout for vertical emplacement is shown in Figures 6-61 and 6-62. The layout for horizontal emplacement is shown in Figures 6-63 and 6-64.

8. Panel access drifts run perpendicular to the mains and provide access to the emplacement drifts. During development, the drifts would be used for haulage and ventilation and must accommodate mining vehicles and belt conveyors.

9. Emplacement drifts are planned to be the disposal areas, from which emplacement boreholes would be drilled, in both the horizontal and vertical emplacement modes.

10. Midpanel drifts would be used only in the vertical emplacement mode. The main purpose of these drifts is to carry exhaust air from the emplacement drifts to the perimeter drift.

11. Four shafts are proposed to lead to the underground facilities. The men-and-materials shaft would provide access for men and materials and air intake for the development area. The first exploratory shaft (ES-1) would provide intake of air for the waste emplacement area. The emplacement exhaust shaft would discharge the air from the waste emplacement area. The second exploratory shaft (ES-2)

would provide secondary intake of air for the waste emplacement area.

Boreholes would be used for the emplacement of waste containers. The vertical boreholes are designed to hold a single waste container while the horizontal boreholes are designed to accommodate up to 14 spent fuel waste containers or up to 18 DHLW waste containers.

6.2.6.1.1 Development sequence

The construction and emplacement operation of the underground repository is proposed to take place in two steps. The first step precedes waste emplacement and would consist of initial construction of accesses and underground drifts. In the second step, construction of the new emplacement drifts and emplacement of waste in previously constructed drifts would occur simultaneously.

Initial construction

In this first step of the development sequence, the entry points to the emplacement level would be constructed.

In this conceptual design, the shafts in the exploratory shaft facility (ESF) are proposed to be converted to ventilation air intakes. The 12-ft (4-m) diameter shaft, ES-1, would be stripped of all steel and internal fixtures, leaving only the concrete lining. The headframe, hoist house, and other surface facilities would also be removed, and the surface exhaust fan would be removed from the 6-ft (2-m) diameter shaft, ES-2. ES-1 would serve as the primary source of air for the waste emplacement area. ES-2 would be a secondary source of air for the waste emplacement area and the main source of the air for the shops in the waste emplacement area.

Two 20-ft (6-m) diameter finished shafts are planned. One shaft would be outfitted for use as a men-and-materials shaft; the other would be used to exhaust air from the waste emplacement area.

Current plans for initial underground development do not call for the use of the exploratory shafts, ES-1 and ES-2. However, the conceptual design does not preclude future use of ES-1 and ES-2 for underground development, if advantageous.

Concurrent development and emplacement

After completion of initial construction, the three main drifts and the perimeter drift would be developed to a point approximately two panel widths away from the point at which the first waste is planned to be emplaced. Development and emplacement is planned to begin in the northeast quadrant of the repository and proceed in a clockwise direction.

The planned operation involves concurrent activities in three panels; while one panel is being developed, one panel has been developed and is ready to receive waste, and a third panel is receiving waste. The emplacement panels would be developed by first constructing the access drifts (for vertical orientation), or emplacement drifts (for horizontal orientation) from the main drifts. The panel would be developed from the access or emplacement drift and a series of boreholes would be drilled in the required configuration and prepared to receive the waste. Liners would be placed in the horizontal boreholes during development and partial liners are planned for use in vertical boreholes. Vertical and horizontal boreholes are illustrated in Figures 6-68 and 6-69.

6.2.6.1.2 Mining methods

The excavation method for each underground opening will depend on the shape and dimensions of the opening and the properties of the rock surrounding the opening. Descriptions of the planned excavation methods for the ramps, shafts, drifts, and boreholes are provided in the following paragraphs.

Drill-and-blast methods are planned to be used to excavate most shafts. Shafts with smaller diameters, such as the 6-ft (2-m) diameter shaft in the ESF, may be raise bored.

The waste and tuff ramps are both of sufficient size and length to favor the use of tunnel-boring machines (TBMs). The ramps would be driven at a constant slope.

Long-drive drifts, the waste main and the perimeter drift, are similar to ramps in shape and dimensions. The TBM method is similarly proposed for excavating the mains. Drill-and-blast methods are proposed for excavating the remaining, shorter drifts.

Boreholes for the vertical emplacement method would be drilled using existing methods and equipment. The equipment is being developed to demonstrate the capability to accurately drill and line horizontal boreholes of the lengths proposed in the SCP-CD.

The methods proposed for excavation of each type of opening are summarized in Table 6-23.

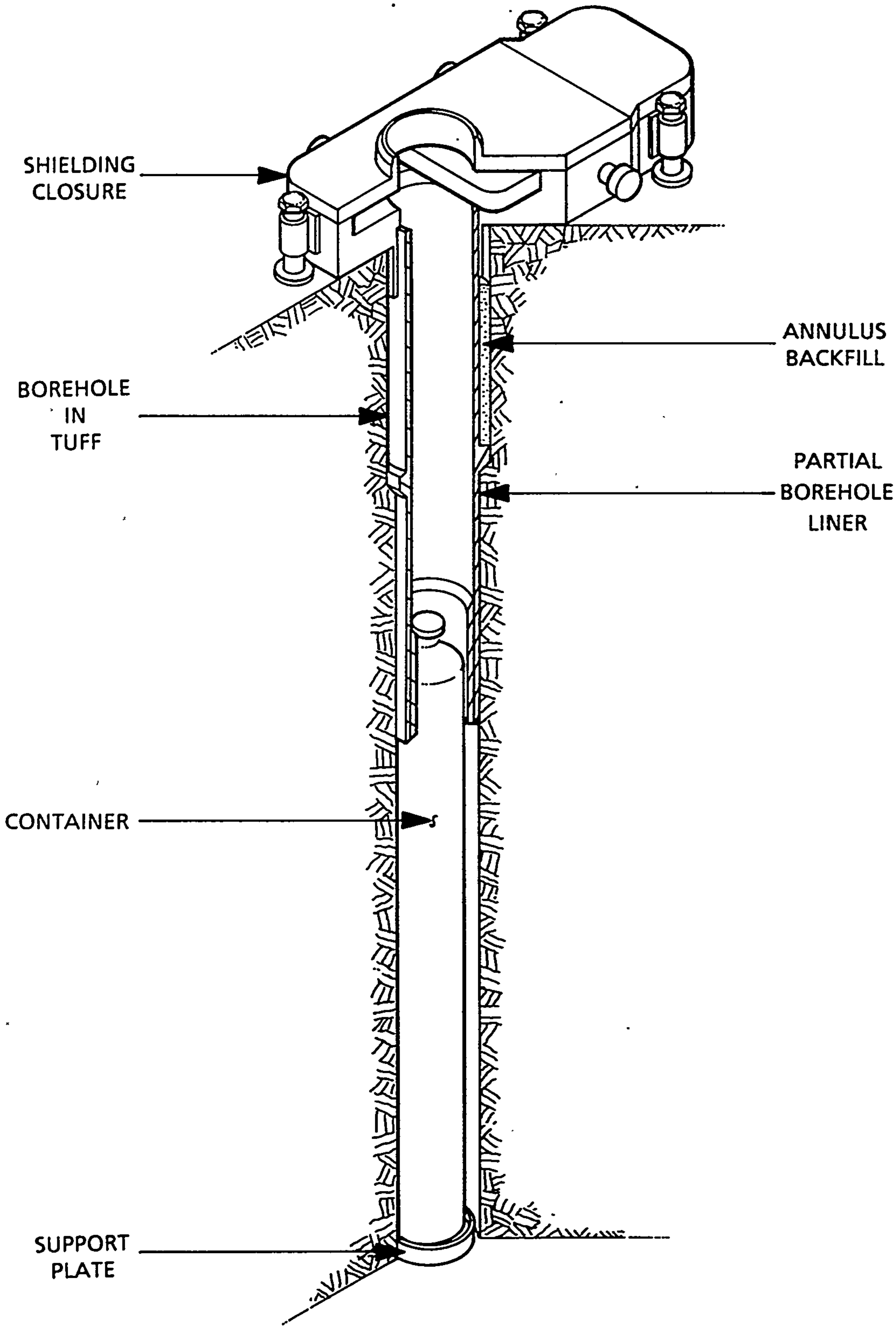


Figure 6-68. Vertical borehole.

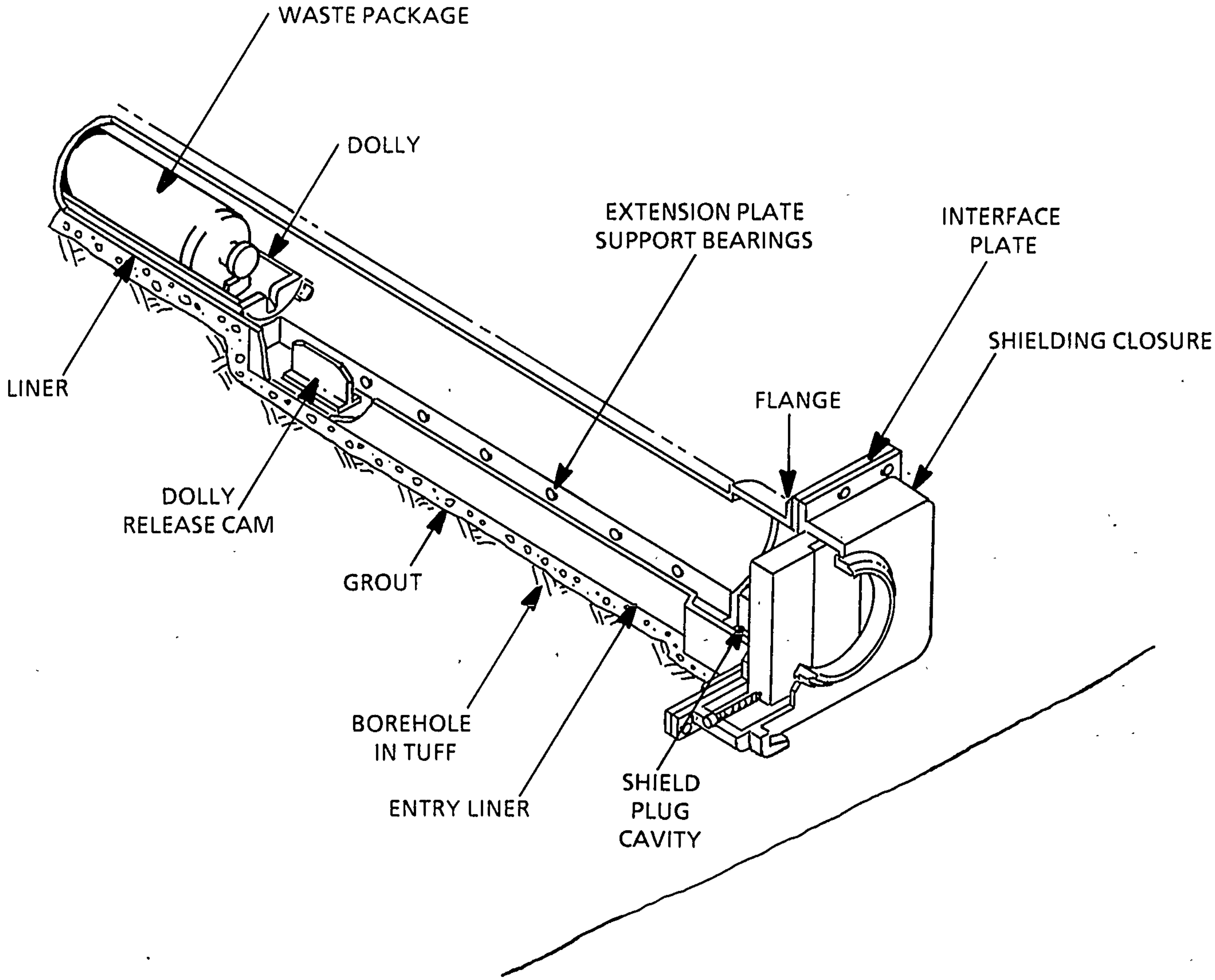


Figure 6-69. Horizontal borehole.

Table 6-23. Mining methods

Opening	Proposed mining method
Shafts	
Small diameter	Drilling, raise boring
Large diameter	Drill and blast
Waste ramp	Tunnel-boring machine
Tuff ramp	Tunnel-boring machine
Waste main	Tunnel-boring machine
Tuff main	Drill and blast
Service main	Drill and blast
Perimeter drift	Tunnel-boring machine
Emplacement drifts	Drill and blast
Panel drifts	Drill and blast
Midpanel drifts	Drill and blast
Emplacement boreholes	Drilling

6.2.6.1.3 Handling of excavated tuff

In the current conceptual design, excavated tuff would be transported from the working face to a feeder-breaker by load-haul-dump units. The feeder-breaker would crush the tuff into pieces smaller than 8 in. (20 cm) and place the crushed tuff on a conveyor belt for transportation to the surface. The conveyor series would, at its longest, be approximately 13,600 ft (4,145 m) long. At the portal of the tuff ramp, the excavated tuff would be transferred to a surface conveyor for transport to the tuff pile.

All conveyors would be controlled at the underground control center and powered by the main power distribution system. The separate units of the conveyor system would have sequential startup and shutdown for control and economy of operations. Each unit of the system would be equipped with a manually operated emergency shutdown system.

6.2.6.1.4 Ground support

This section summarizes the ground support systems proposed for all drifts and ramps at Yucca Mountain. A brief description of the design method is presented, along with a discussion of the site-specific factors that affect the choice of ground-support methods. Section 6.1.1 identifies the design requirements for opening stability, as prescribed by 10 CFR Part 60.

Ground support options

A number of ground support systems currently used in mining and civil tunneling projects are well suited for use in a repository. Although specific tunneling experience in welded tuff is rather limited, the range of expected conditions is well within the limits of historical experience. Due to the strength of the rock, depth of the openings, and ground support that is planned, no surface subsidence is anticipated. The following ground support options are candidates for use at the repository:

1. No support.
2. Friction bolts (Swellex, split-set).
3. Point-anchor bolts.
4. Resin-grouted dowels.
5. Cement-grouted dowels (pretensioned, posttensioned, and untensioned).
6. Cement-grouted cable anchors.
7. Chain-link fence materials.
8. Welded wire mesh.
9. Shotcrete with steel fiber, microsilica, and accelerators.
10. Structural-steel sets.
11. Lattice girders.
12. Yieldable steel arches.
13. Cast-in-place concrete lining.
14. Prefabricated, segmented linings.

The ground support systems selected should include the following features:

1. Compatibility with excavation methods and sequence of construction.
2. Relative flexibility to conform with thermally and seismically induced deformations.
3. Minimal site-specific prefabrication.
4. Installation methods or upgrading as needed to deal with anomalous conditions.
5. Installation that minimizes work in unsupported openings.

6. Adaptability to accommodate varying shapes of openings and intersections.

7. Durability to withstand elevated temperatures and the possible presence of water for 100 yr.

8. Cost-effectiveness for the support obtained.

9. Chemical components that are compatible with waste package isolation and containment strategies.

These containment and isolation strategies are discussed in Section 8.3.

Ground support recommendations

The empirical rock-mass classifications discussed by Langkopf and Gnirk (1986) and Dravo Engineers, Inc. (1984) recommend the use of a combination of rock bolts, grouted dowels, wire mesh, and shotcrete in varying degrees to conform with local conditions. Figure 6-70 shows typical sections of the ground support systems in the conceptual design for both bored and conventionally mined drifts. The site characterization program will provide a more detailed description of the in situ rock characteristics in the emplacement horizon. These data will provide insight into the variability of the rock characteristics encountered in a few thousand feet of drifting. Even with these data it will not be advisable or necessary to assign final ground-support designs to individual drifts. Observations and measurements made during repository construction will support the final determination of the ground support at any given location, in accordance with predetermined standards.

6.2.6.1.5 Underground development equipment

The equipment that would be used for underground development is shown on Figures 6-71 and 6-72. Except for the drill for the horizontal borehole, the equipment is currently available. Feasibility studies and conceptual designs for drilling and lining horizontal boreholes have been completed (Robbins Company, 1984a, 1984b, 1985). The design process, which is currently in the detailed design phase, will be followed by construction of demonstration equipment to be used in proof-of-concept demonstrations (SNL, 1987, Sections 6.3.2.1 and 6.3.2.2).

6.2.6.2 Ground-water control

Exploratory drilling performed to date indicates the proposed repository horizon, the Topopah Spring Member of the Paintbrush Tuff, lies within the unsaturated zone above the water table. No perched water zones have been identified within the proposed horizon, although the potential for a perched water table exists at the interface between the Topopah Spring and the less

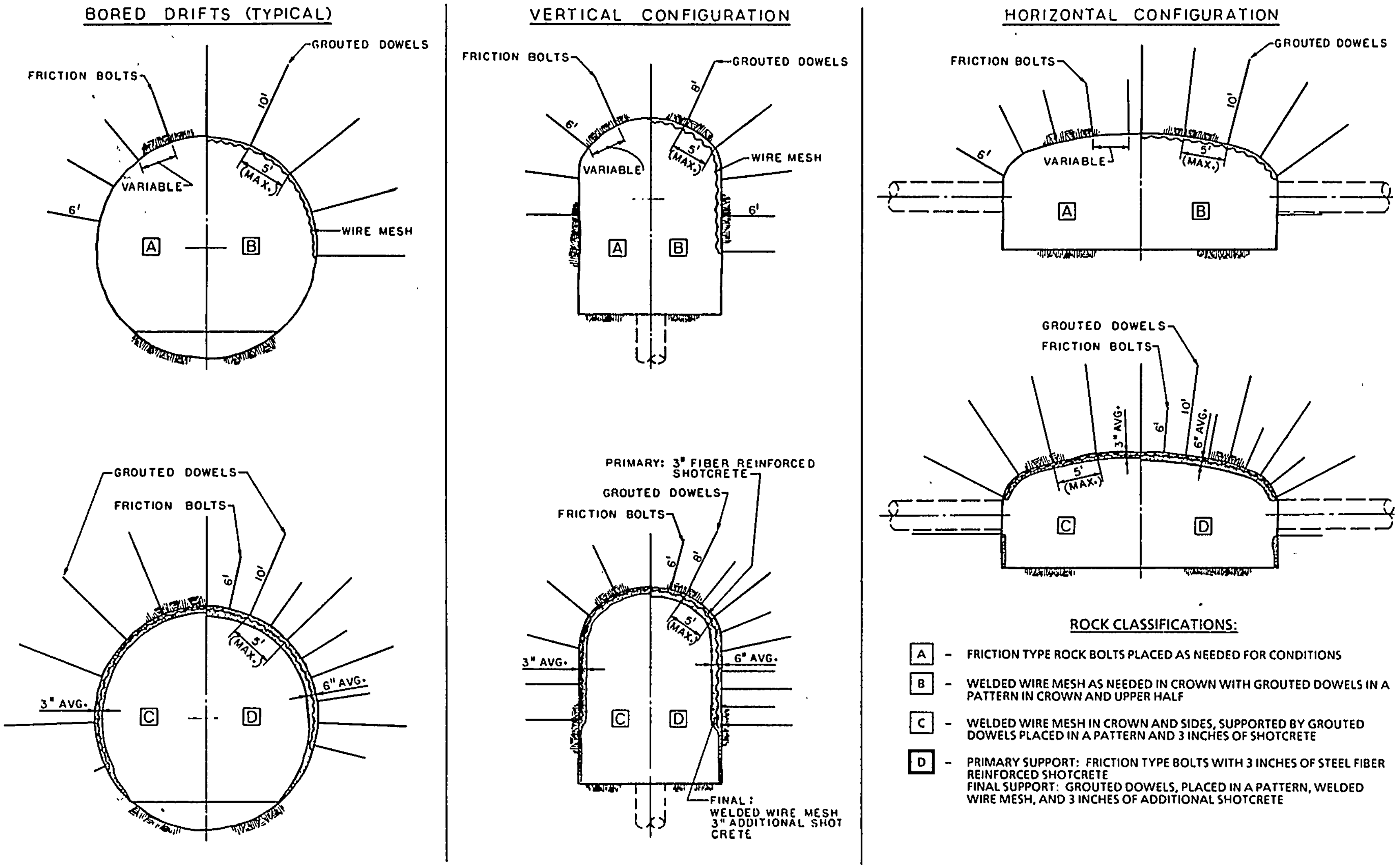


Figure 6-70. Typical ground support cross sections.

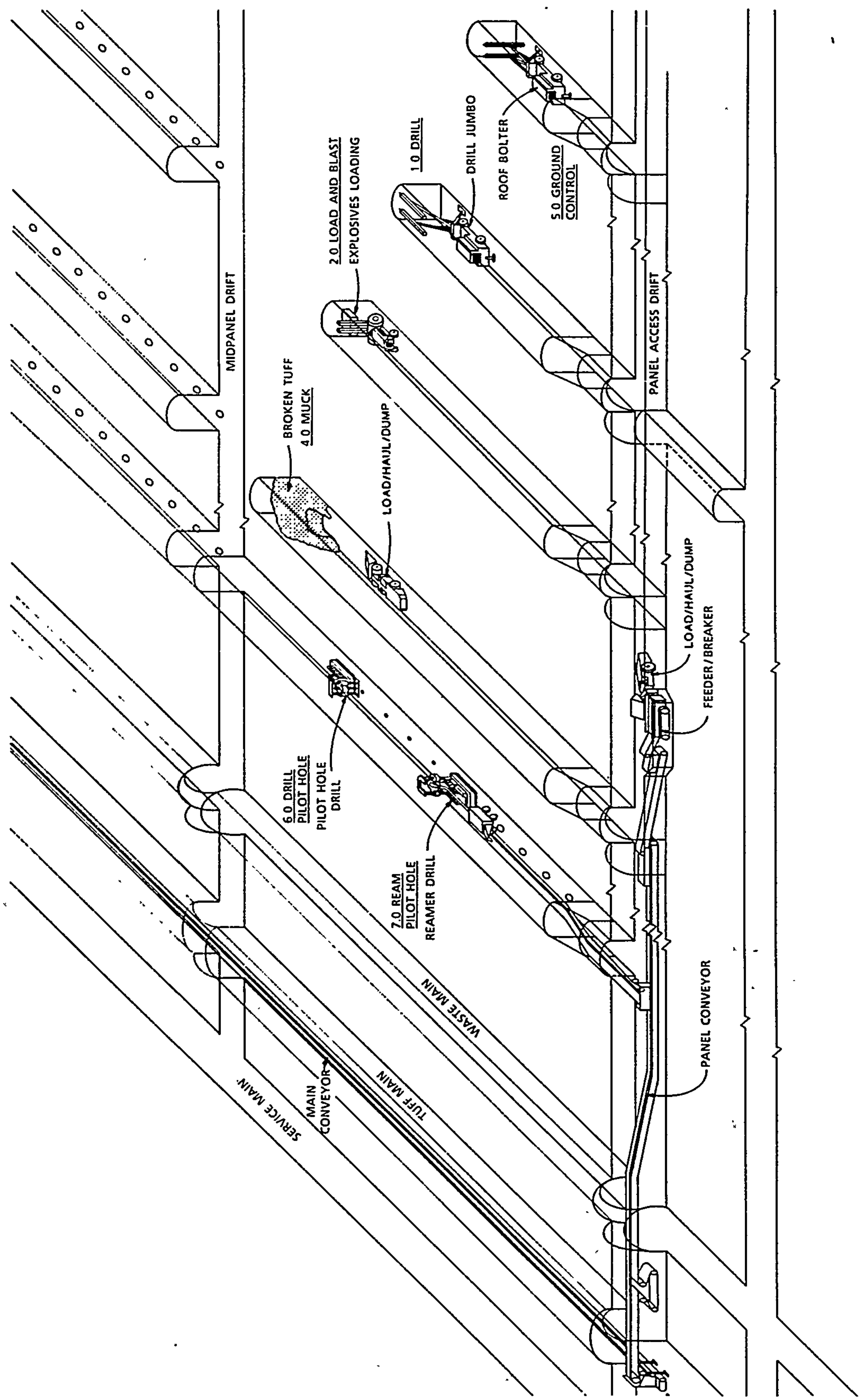


Figure 6-71. Isometric diagram of mining methods and sequence for vertical emplacement.

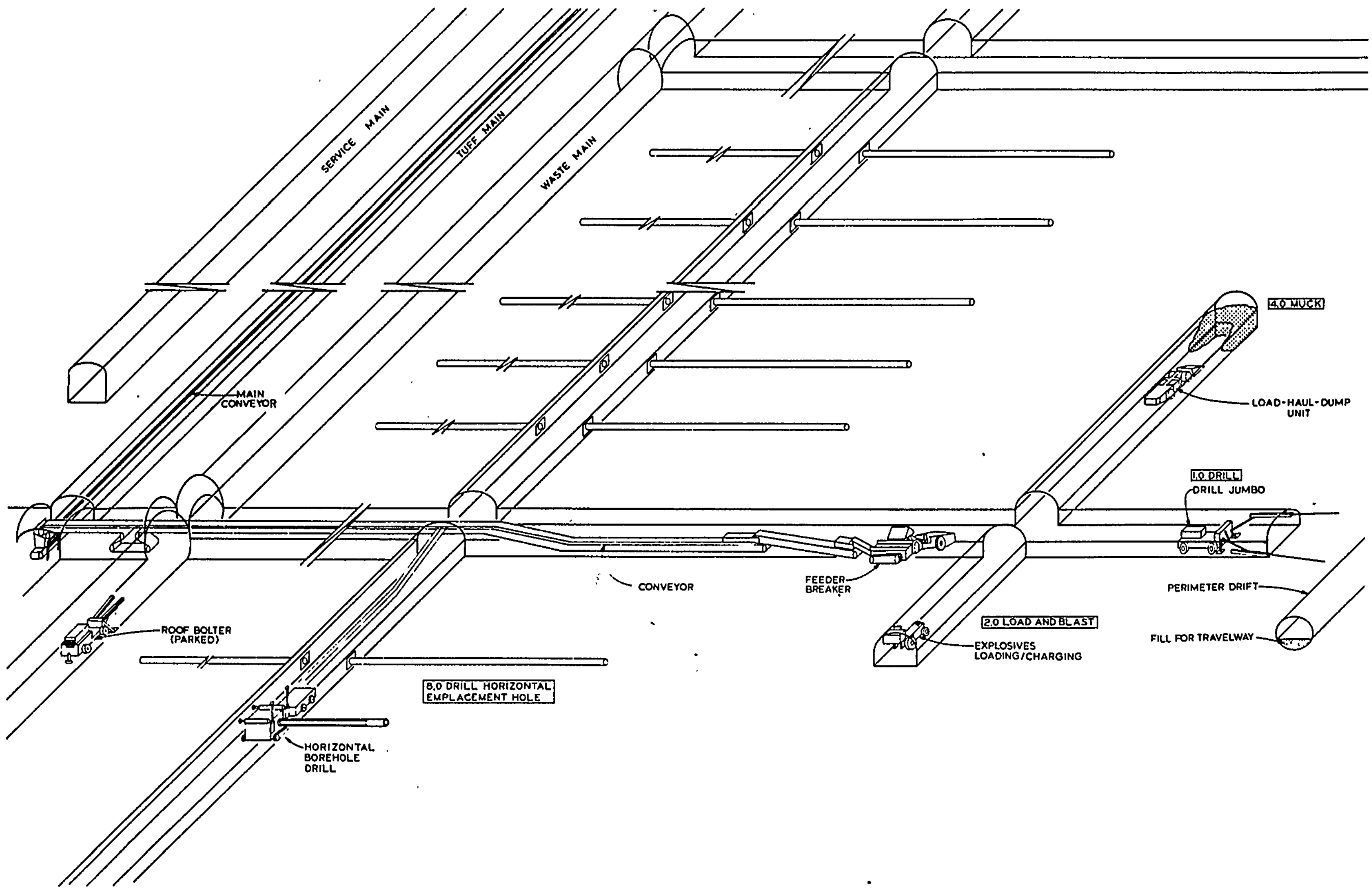


Figure 6-72. Isometric diagram of mining methods and emplacement for horizontal emplacement.

permeable Calico Hills zeolitized tuff. However, the conceptual repository layout is situated above this interface, and relatively dry conditions are anticipated throughout the repository.

Inflow of ground water is not anticipated in significant quantities, however, the drifts are designed so that any water, whether from ground water or from operations within the underground facilities, would be diverted away from the waste emplacement locations (Figure 6-73). All areas would drain in the direction of a sump located in the bottom of the emplacement area exhaust shaft, the lowest point in the underground facility. From this location, the water would be pumped to the surface through the emplacement area exhaust shaft. Backup pumps would be available to ensure adequate pumping capacity.

6.2.6.3 Ventilation

The following section describes the general ventilation system. Sections 6.2.6.3.2 and 6.2.6.3.3 describe the ventilation systems for the vertical and horizontal emplacement configurations, respectively.

6.2.6.3.1 General overview and description of the system

Two independent ventilation systems are planned to serve the underground facilities. One system would provide air for the development of the repository while the other would provide air for waste emplacement operations. Connections between the two ventilation systems would be sealed with bulkheads or air-locks. Redundancy in the ventilation fans on each side is planned. A complete description of the ventilation systems is given in Section 3.4 of the SCP-CDR.

The development and emplacement areas would be ventilated by a system with two independent air circuits. Interaction and leakage between air circuits is inevitable. To ensure that leakage would occur from the development air circuit to the emplacement air circuit, pressure differentials are established between the air circuits. A forcing or positive pressure system would be employed for the development air circuit. This positive pressure system would use forcing fans located at the collar of the men-and-materials shaft. An exhausting or negative pressure system would be used for the emplacement air circuit. This negative pressure system would use exhaust fans located at the collar of the emplacement area exhaust ventilation shaft. The conceptual design does not provide the capability for reversal of the underground ventilation, because reversal would cause leakage from the emplacement air circuit to the development air circuit. The question of reversal of underground ventilation will be addressed further as part of the advanced conceptual design studies.

The performance of the repository ventilation systems would be monitored continuously at surface and underground control centers. The goal of the planned monitoring system is threefold: (1) to maintain an immediate measure of the working environment within the repository; (2) to provide immediate notice of accidents, including fire and incidents involving radioactive

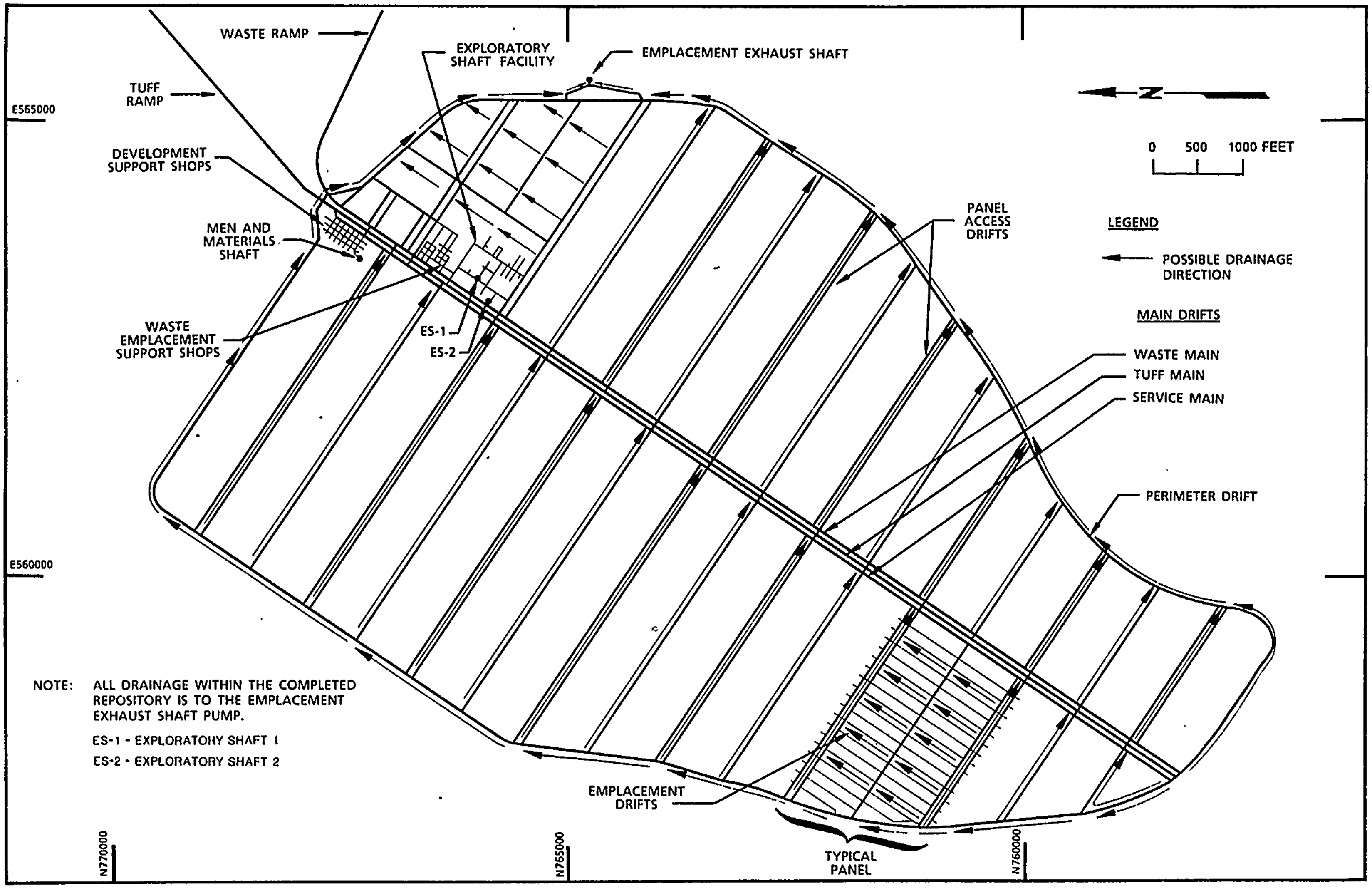


Figure 6-73. General underground facility layout showing drainage directions for vertical emplacement.

material; and (3) to activate appropriate measures, such as evacuation of personnel, redirection of waste emplacement exhaust air through filters, and dispatch of emergency crews.

During normal operations, return air from the waste emplacement ventilation system would be exhausted directly to the atmosphere. However, should the monitors detect a radiation release, the return air would be routed through a set of filters, including high-efficiency particulate air filters, before discharge.

The underground ventilation concepts are based on the proposed repository layout and development sequences presented in Section 6.2.6.1. The basic ventilation layout consists of the following: (1) four shafts, (2) two ramps, (3) three main airways, (4) emplacement areas on either side of the main airways, and (5) a perimeter airway that encircles the repository (Figure 6-74).

The main components of the planned development area ventilation system are as follows:

1. Men-and-materials shaft for air intake from surface.
2. Service main for air intake to development areas.
3. Tuff main as return for air from development areas.
4. Tuff ramp as return for air to surface.

The main components of the planned waste emplacement area ventilation system are as follows:

1. Exploratory shafts and waste ramp for air intake from surface.

2. Waste main for air intake for waste emplacement, emplacement room cooling, or caretaking operations.

3. Perimeter drift as return for air from waste emplacement area.

4. Emplacement area exhaust shaft as return for air to surface.

The waste emplacement area ventilation system is designed to ensure safe working conditions during waste transport, emplacement, or retrieval operations. Both spent fuel and defense high-level waste may be emplaced in the same drift.

For the most part, the required airflows for both ventilation systems were derived from considerations of diluting diesel exhaust fumes, minimum statutory requirements for airflow, and shop demands. Computed airflow velocities at various underground locations were compared to the maximum velocity constraints shown in Table 6-24. The velocity constraints were determined from dust abatement and comfort considerations (National Materials Advisory Board, 1980).

Cooling of an emplaced drift for inspection, maintenance, or retrieval is expected to be necessary because the containers will heat the rock mass around the emplacement hole, which, in turn, will transfer heat to the

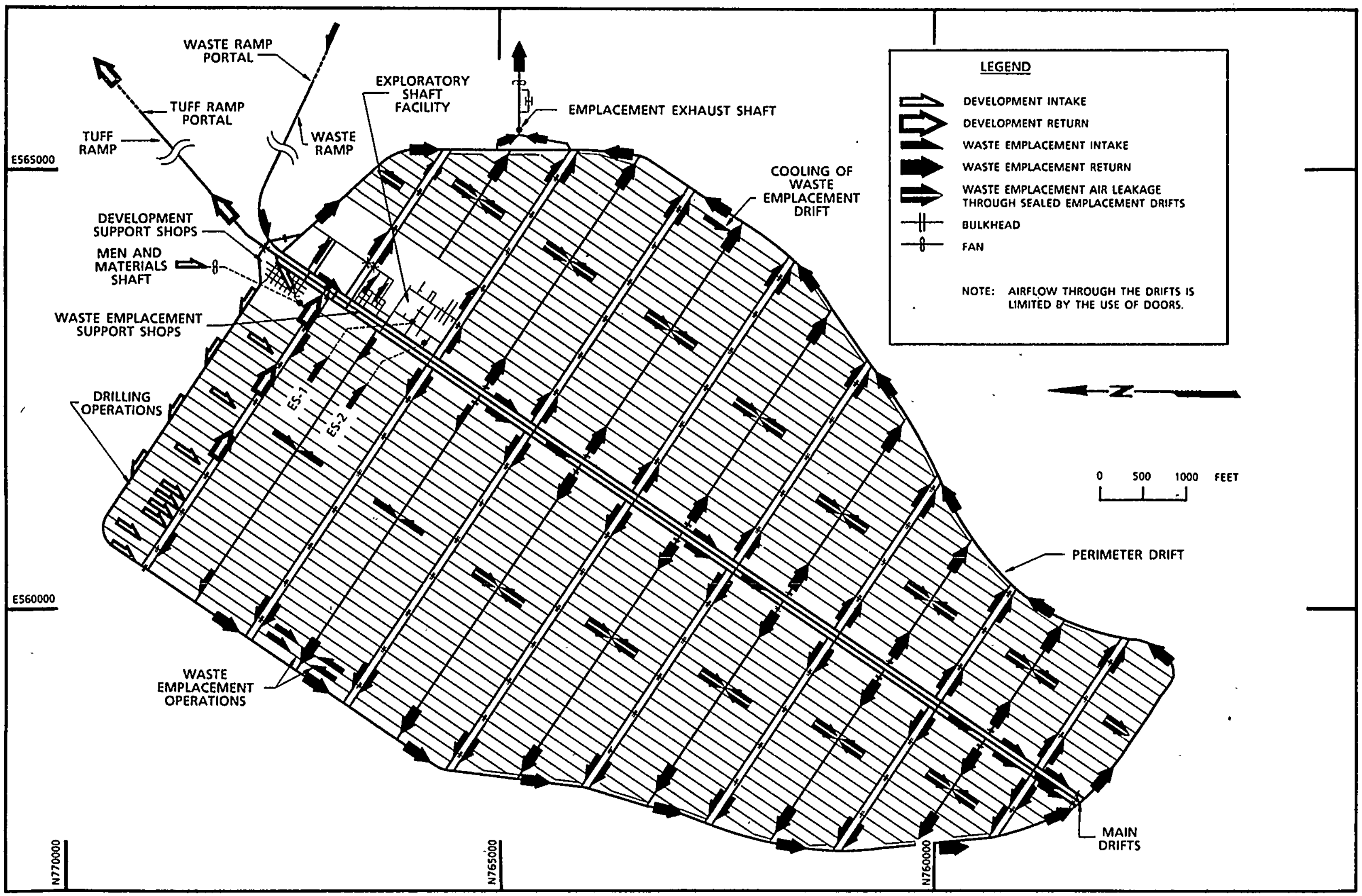


Figure 6-74. Maximum development of ventilation requirements for vertical emplacement.

Table 6-24. Maximum velocity constraints

Area	Maximum velocity[a] ft/min	m/min
Intake shafts (unobstructed)	4,000[b]	1219
Return shafts (unobstructed)	4,000[b]	1219
Waste transport ramp	1,500	457
Tuff ramp or shaft	1,500	457
Men-and-materials shaft	2,300	701
Perimeter airway	2,000	610
Main entry drifts	1,500	457
Main return drifts	1,500	457
Haulage airways (no conveyor)	1,200[b]	362
Haulage airways (conveyor)	1,000[b]	305
Emplacement drifts	1,500	457
Development areas (drilling, etc.)	600	183

[a]Maximum shaft velocities assume that the shafts are dry and unobstructed.
[b]National Materials Advisory Board (1980).

emplacement drift airflow. Because the pre-waste-emplacement rock temperature is expected to be low at Yucca Mountain, no air cooling is expected to be required in the development ventilation system.

6.2.6.3.2 Vertical (reference) emplacement configuration

Vertical development area

Figure 6-74 shows the main air intake and return flow directions and quantities, the temperature distribution, location of main fans, main fan pressures, and ventilation controls throughout the vertical emplacement configuration at the maximum development airflow demand. The main fan requirements for this layout are shown in Table 6-25.

Vertical emplacement area

The maximum airflow demand on the ventilation system for the waste emplacement area would occur when the repository is nearly fully developed and when emplacement and inspection or maintenance are occurring simultaneously. At this time, vertical borehole drilling would be the only operation in the development area. This set of conditions would require the largest waste emplacement fan capacities because the air exiting the panel in which

Table 6-25. Maximum airflow requirements for the development area in the vertical emplacement configuration

Ventilation system	Main fan requirements: Pressure (in. w.g.)[a]	Airflow (cfm)[b]
Development	9.0	411,800
Waste Emplacement	3.25[c]	481,300

[a]Inches water gauge.
[b]Based on standard air density, cubic feet per minute.
[c]Pressure required at the collar of waste emplacement area exhaust shaft.

waste is being emplaced would have to travel a great distance around the repository in the perimeter drift to the waste emplacement area exhaust shaft.

Figure 6-75 shows the expected distribution of the airflow and air temperature, as well as the proposed main fan locations throughout the repository. Nearly 40 percent of the intake air for the waste emplacement area would enter through the waste ramp. The remaining intake air would enter through the two exploratory shafts. The basic ventilation system shown for the waste emplacement area would give acceptable airflows throughout the repository. Table 6-26 identifies the fan requirements for this layout time phase.

Table 6-26. Maximum airflow requirements in the waste emplacement area in the vertical emplacement configuration

Ventilation system	Main fan requirements: Pressure (in. w.g.)[a]	Airflow (cfm)[b]
Development	1.50	209,400
Waste emplacement	13.75[c]	837,200

[a]Inches water gauge.
[b]Based on standard air density.
[c]Pressure required at the collar of waste emplacement area exhaust shaft

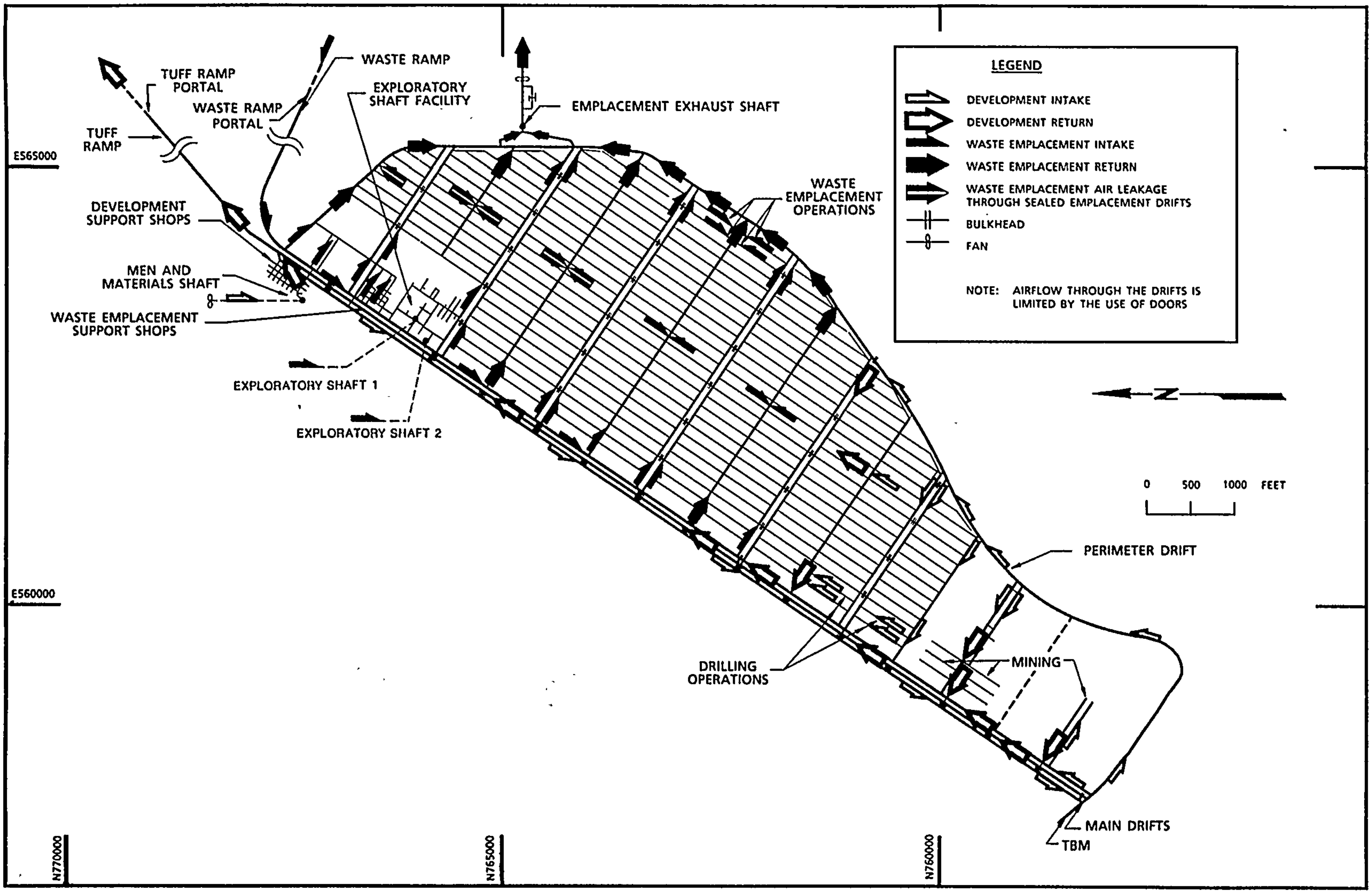


Figure 6-75. Maximum airflow directions, quantities, and temperatures for the waste emplacement area for vertical emplacement.

6.2.6.3.3 Horizontal emplacement configuration

The horizontal emplacement configuration is very similar to the vertical emplacement configuration with the exceptions that the horizontal emplacement drifts would be spaced much farther apart and no vertical midpanel access drift would be required. Therefore, the basic ventilation system planned for the horizontal configuration is nearly identical to that planned for the vertical configuration. The key differences between the two systems are (1) the emplacement drifts in the horizontal configuration would be twice as long as those in the vertical configuration and (2) the alternating sets of panel access drifts would act as returns for the horizontal emplacement drifts on each side. Therefore, the ventilation of a panel would be similar to that planned for the vertical system except that the panels in the horizontal system would be effectively twice as large.

Horizontal development area

As in the vertical emplacement configuration, the maximum airflow requirements for the development ventilation system are expected when mining operations are occurring at the greatest distance from the base of the tuff ramp. Given the reduced mining activities, the airflow requirements for horizontal waste emplacement also would be less than those for vertical waste emplacement.

The airflow and temperature distributions, ventilation controls, and main fan locations are shown on Figure 6-76.

Horizontal emplacement area

The maximum airflow demand on the ventilation system for the waste emplacement area for the horizontal configuration would occur when the repository is nearly fully developed and only emplacement borehole drilling operations are being conducted.

Figure 6-77 illustrates the expected airflow directions and quantities, air temperatures, ventilation controls, and main fan requirements throughout the repository. As in the vertical emplacement configuration, a significant amount of air for the waste emplacement area would enter the facility through the waste ramp. Table 6-27 shows the main fan requirements for the layout.

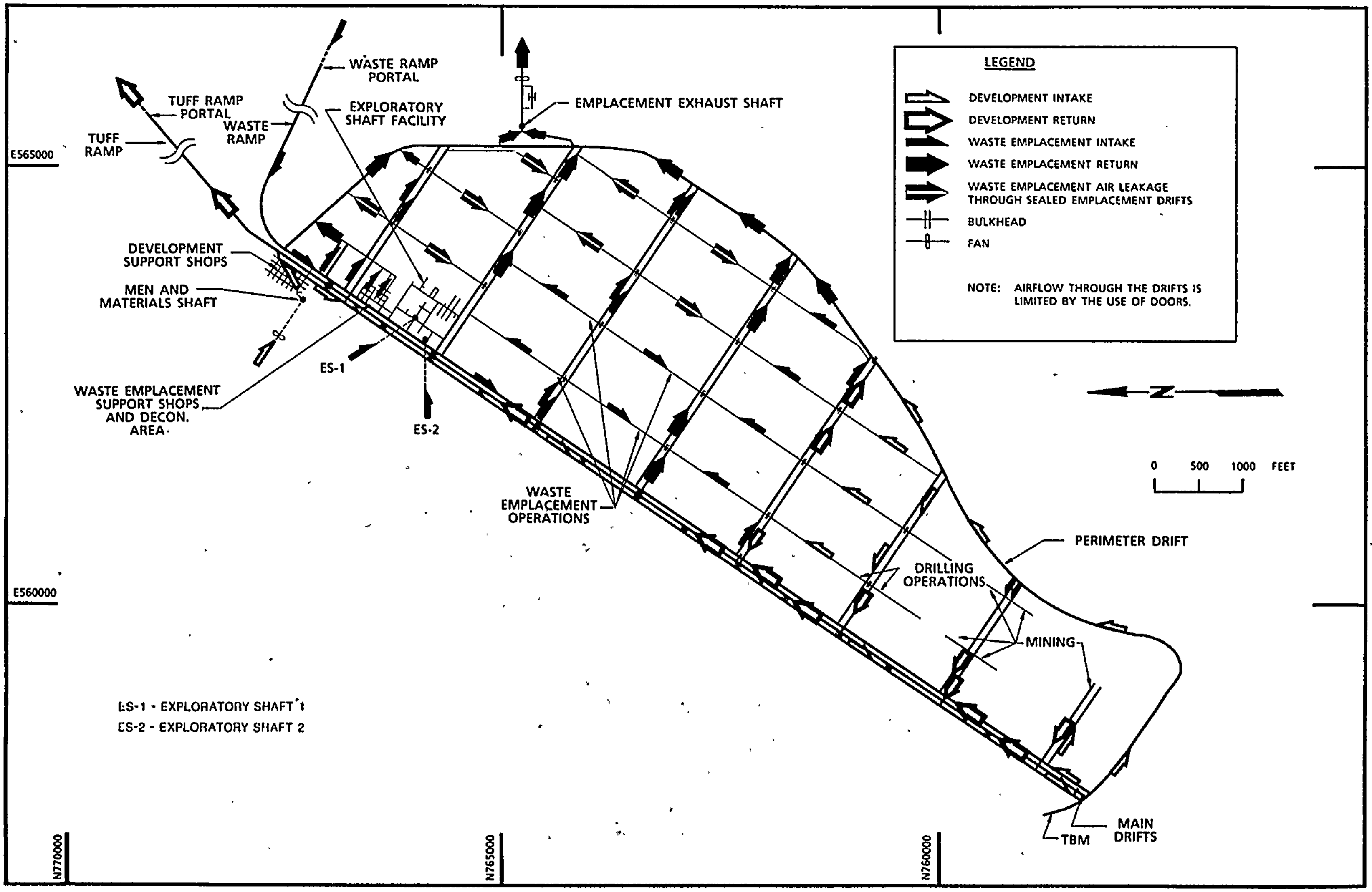


Figure 6-76. Maximum airflow directions, quantities, and temperatures for the development area for horizontal emplacement.

6-175

CONSULTATION DRAFT

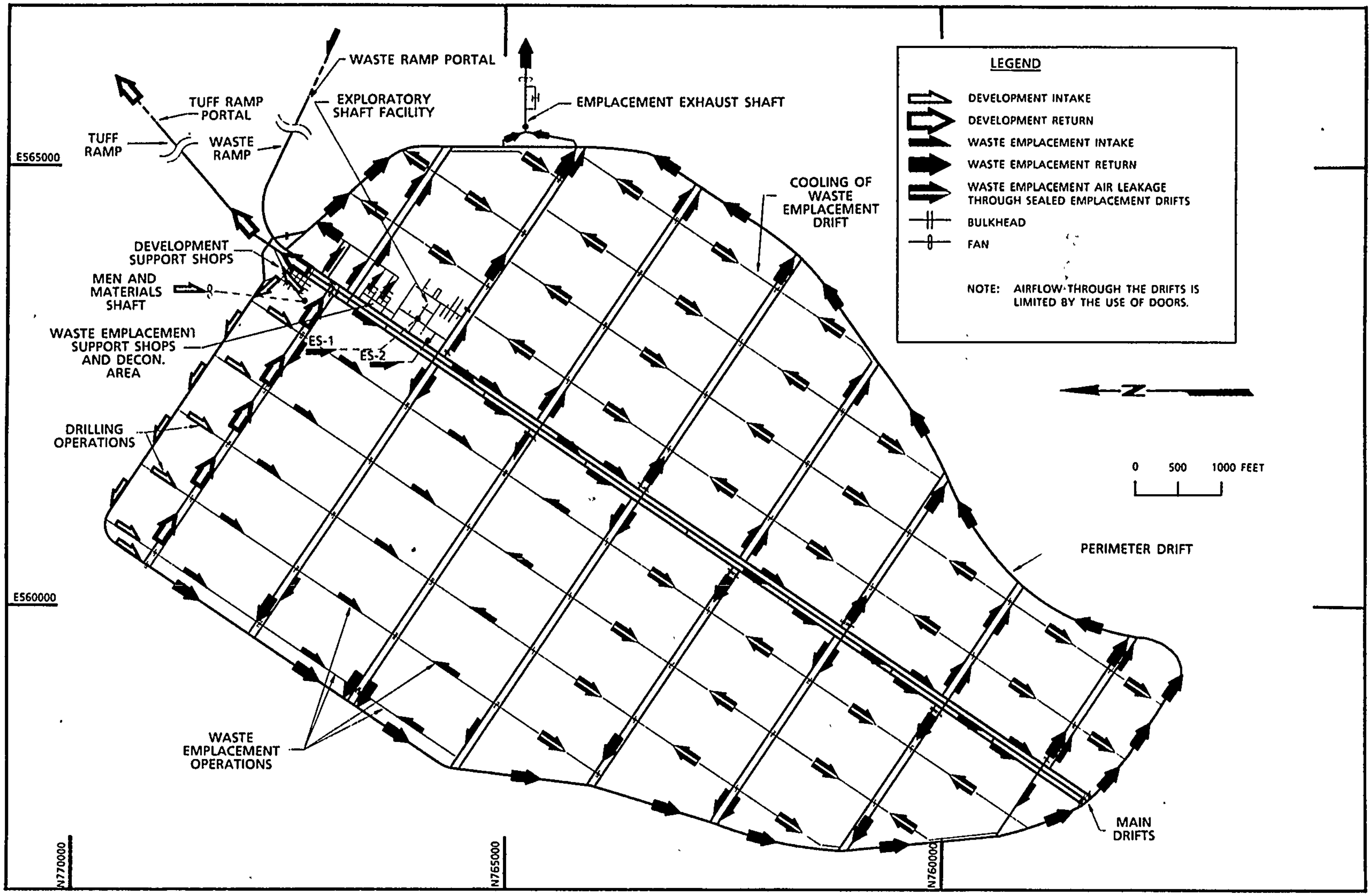


Figure 6-77. Maximum airflow directions. quantities. and temperatures for the waste emplacement area for horizontal emplacement.

Table 6-27. Maximum airflow requirements in the waste emplacement area in the horizontal emplacement configuration

Ventilation system	Main fan requirements	
	Pressure (in. w.g.)[a]	Airflow (cfm)[b]
Development	1.00	117,200
Waste emplacement	5.00[c]	517,200

[a]Inches water gauge.
[b]Based on standard air density, cubic feet per minute.
[c]Pressure required at the collar of waste emplacement area exhaust shaft.

6.2.7 BACKFILL OF UNDERGROUND OPENINGS

Section 60.133(h) of 10 CFR Part 60 states, "Engineered barriers shall be designed to assist the geologic setting in meeting the performance objectives for the period following permanent closure." Backfill is considered part of the underground facility, which is part of the engineered-barrier system.

Backfilling following emplacement and backfilling at closure are the two options considered. The need for backfill must be assessed based on the stability analyses of the underground openings and the analyses of the hydrologic conditions within the repository.

In addition, the need for backfilling following emplacement must not preclude retrieval of emplaced waste. If backfill is shown to enhance the performance of the overall repository system, requirements for backfilling will be developed. The properties of the backfill material, emplacement concepts, and specifications for backfilling will be established on the basis of these requirements.

6.2.7.1 Backfilling following emplacement

In the current conceptual design, backfilling of mined openings is not planned immediately following emplacement. The reasons for this position are that backfill is not required for structural support before closure and backfilling following emplacement would unnecessarily complicate potential retrieval operations.

Design calculations have been performed evaluating the stability of the rock mass at the underground horizon. These calculations are discussed in

Section 6.4.8.2.4. Because these calculations did not take into consideration backfill, it can be concluded that backfilling is not necessary to ensure mechanical stability throughout the retrievability period. Stability through the retrievability period is planned to be further enhanced through the use of ground support, such as rock bolts and wire mesh. More details of these calculations are reported in the SCP-CDR.

6.2.7.2 Backfilling at closure

Hydrologic calculations were performed to determine whether backfill could assist the geologic setting in meeting the performance objectives. These calculations evaluated the water flow in the vicinity of a vertically emplaced waste package located in the unsaturated zone of Yucca Mountain (Fernandez and Freshley, 1984; Freshley et al., 1985a).

The objective of these hydrologic calculations was to determine whether the selection of a drift backfill could influence the flow of water past waste packages and, thus, potentially influence the release of radionuclides. The calculations focused on determining the type of materials that might be useful as backfill. Coarse materials such as moderately to lightly crushed tuff or fine-grained backfill consisting of highly crushed tuff could be used. Sand and clay were the materials simulated in the calculations. These materials hydrologically represent a broad range of potential backfill material.

The basic approach to the modeling just described was to assume a continuum approach to water flow in the matrix of the fractured tuff of Yucca Mountain. The flux in the rock media was assumed to be less than the saturated hydraulic conductivity of the matrix. Hence, it is reasonable to assume the fractures do not transmit water, and they were not explicitly included in the analysis.

Conclusions from these hydrologic analyses indicate that from a hydrologic perspective, by approximating an open drift with coarse sand in numerical simulations, backfill in the drifts is not likely to influence flow significantly around waste packages. The results of the numerical simulations indicate that a coarse material will perform more satisfactorily as a capillary barrier to matrix water flow through drifts than will a fine material. The excavated tuff removed during development of the underground facility would be a source for coarse backfill. This material is planned to be stored in stockpiles on the surface and could be transported to the underground facility. Because of a bulking effect, only a fraction of the rock that was originally removed could be replaced as backfill.

The basis for current planning is that the underground facility will be backfilled at closure using the tuff that was excavated during development.

Water used during backfilling operations to control dust, improve compaction, or both will be introduced in limited quantities. Excess water will be removed in the same way as water introduced during development activities.

6.2.8 SEALS

Sealing refers to all activities associated with the permanent closure of the shafts, ramps, exploratory boreholes, and underground facility. Sealing includes emplacing backfill and sealing elements in shafts, ramps, drifts, emplacement holes, and exploratory boreholes (Fernandez and Freshley, 1984).

According to 10 CFR 60.134, the seals for shafts and boreholes shall be designed so that following permanent closure they do not become pathways that compromise the geologic repository's ability to meet the performance objectives for the period following permanent closure. Materials and placement methods for seals shall be selected to reduce, to the extent practicable (1) the potential for creating a preferential pathway for ground water or (2) radioactive waste migration through existing pathways.

The four functional requirements (Fernandez and Freshley, 1984) for sealing were developed based upon physical processes involving radionuclide transport through the geologic system to the accessible environment by ground-water flow. If ground-water flow near the waste packages can be inhibited or controlled, the potential for radionuclide transport can be reduced.

The first requirement (containment and isolation) addresses this concern. It is the intent of requirement 1 to preclude ground water from reaching the waste package as follows: (1) by preventing water from entering the underground facility through vertical shafts, ramps, or other vertical or horizontal penetrations; and, (2) if water does enter into the vicinity of the waste package, by diverting the ground water around the waste package. If radionuclides should enter the ground-water system, it would be desirable to retain radionuclides in the geologic system by retarding flow and absorbing radionuclides downgradient from the waste container. However, with the predominant vertical gradient in the unsaturated zone, it is not anticipated that radionuclides contained in ground water could reenter drifts.

The second requirement (human intrusion) addresses limiting radionuclide release as a result of either deliberate or inadvertent human intrusion. This objective can be achieved by closing all large openings, shafts, and ramps in a manner that would deter reentry. Small openings, such as exploratory boreholes, are not expected to present a safety hazard because seals acceptable by today's standards sufficiently deter reentry of the wells.

The third requirement (longevity of components) addresses the concern that sealing components must perform acceptably and with a sufficient degree of confidence over a required period. Their long-term performance may include a progressive but acceptable deterioration with time. An increase in confidence can be achieved in the following three ways: (1) by properly designing sealing components to static and dynamic loadings, (2) by reducing the uncertainties associated with material properties and emplacement techniques, and (3) by selecting different materials and designs serving the same or overlapping functions. The need for redundancy in seal functions will be determined through engineering analyses.

The fourth requirement (cost) also must be considered for investigating seal materials and their emplacement. When possible, complex designs and materials should be avoided because increased design, emplacement, and performance verification efforts may be required. Implicit in the verification requirements is the possibility for additional laboratory and field testing for more complex designs.

The sealing concepts proposed for the proposed repository in the unsaturated tuff of Yucca Mountain were developed by Fernandez and Freshley (1984) and provide the basis for continuing NNWSI Project repository sealing activities (Fernandez, 1985). These concepts were developed considering the hydrology of Yucca Mountain, the functional requirements discussed previously, preliminary repository concepts, federal and state regulations, preliminary performance criteria for sealing, and hydrologic calculations. The concepts are briefly described in Section 6.2.8.1 and are described in more detail in the SCP-CDR.

6.2.8.1 Shaft and ramp seal characteristics

6.2.8.1.1 Shaft seals

Four shafts and two ramps are proposed to penetrate the underground horizon at Yucca Mountain. Only the exploratory shaft is planned to extend below the repository horizon into the zeolitized tuff of the Calico Hills.

Sealing concepts for shafts, shown in Figure 6-78, can include a surface barrier, shaft fill, settlement plugs, and station plugs. Beneath the surface barrier, appropriately graded and unreactive fill, such as crushed tuff, settlement plugs, and station plugs, are proposed for the lower portion of the shaft. The surface barrier is proposed to consist of a shaft cover, a collar core, and an anchor-to-bedrock plug-seal.

The lower shaft sealing components are planned to consist of shaft fill and settlement plugs. The shaft fill can be permeable to allow water entering the shaft to drain to the bottom of the shaft where the water can infiltrate the surrounding rock below the underground horizon. Settlement plugs also can be designed to support the shaft fill and prevent the development of a surface depression, that could lead to ponding of surface water or create a safety hazard. The station plug, to be emplaced at the intersection of the shaft and the drifts within the repository horizon, can be designed to resist the lateral forces exerted by the shaft fill and, thus, control settlement of the shaft fill.

6.2.8.1.2 Ramp seals

The concepts for sealing a ramp (Figure 6-79) are similar to those for sealing a shaft (Fernandez and Freshley, 1984). The major differences in the ramp seal concepts are proposed periodic installation of dams designed to

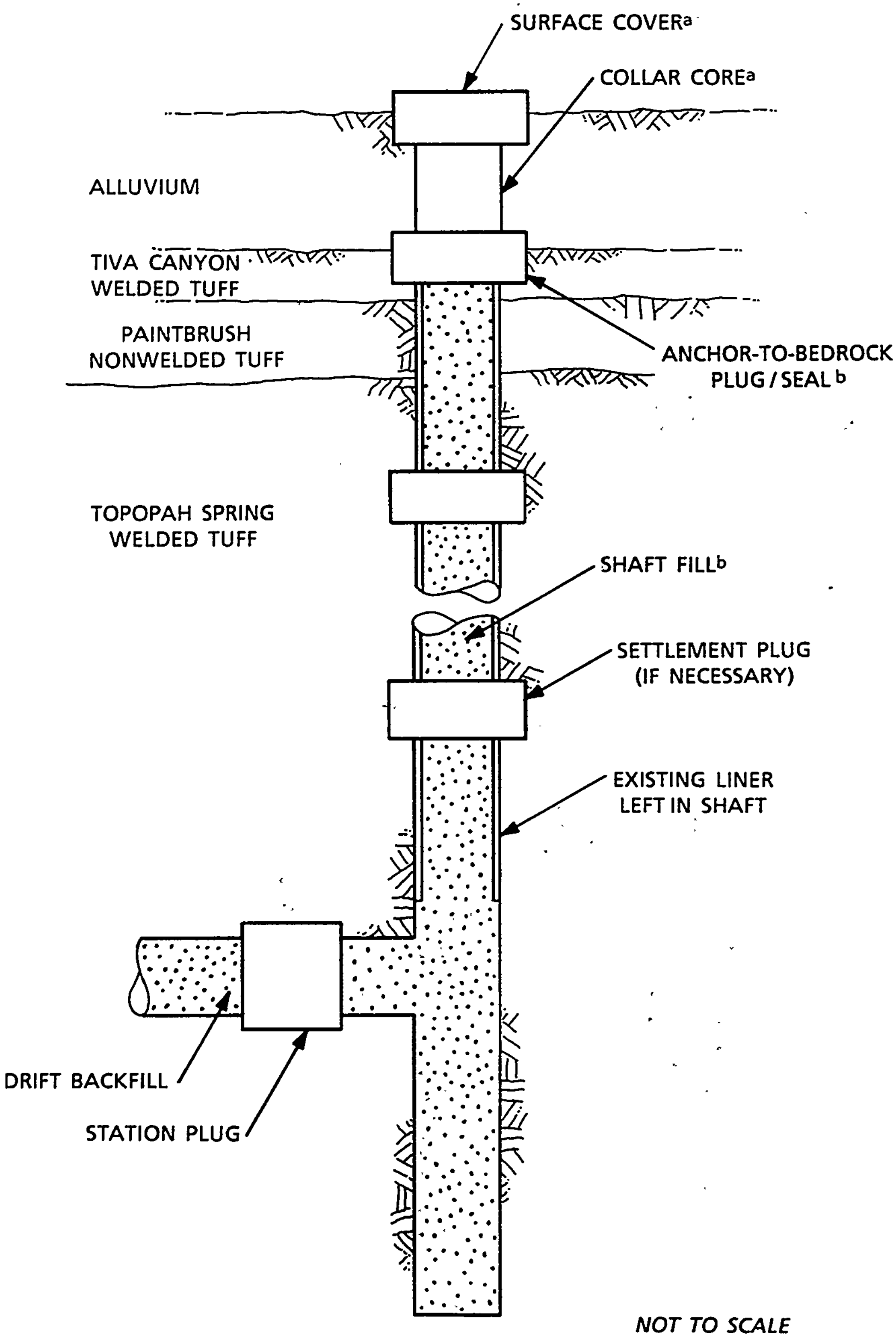


Figure 6-78. General arrangement for shaft seals.

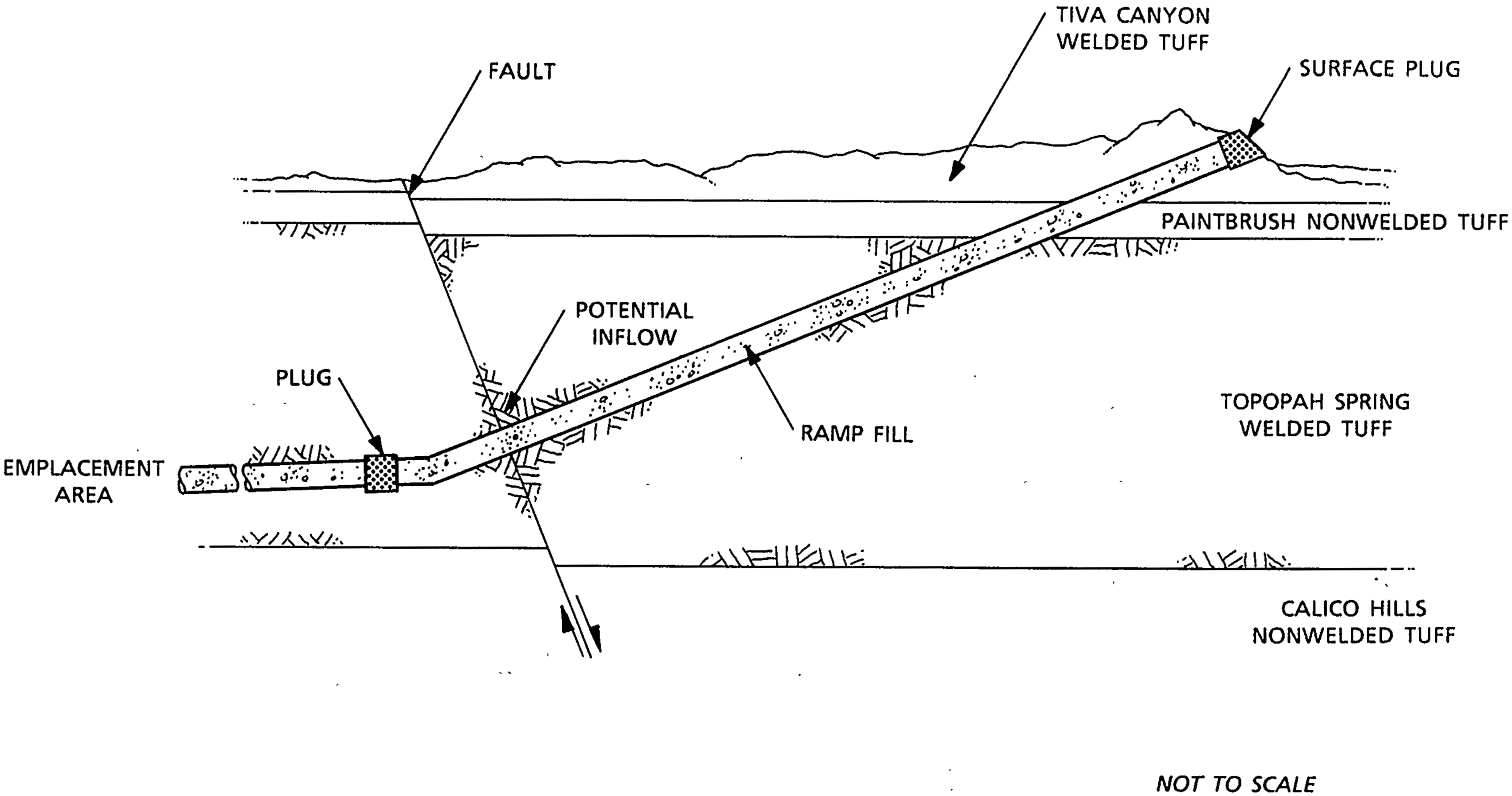


Figure 6-79. General arrangement for ramp seals.

encourage downward flow of water through the tuff rather than allowing flow down the ramp. The frequency of and necessity for dams depend on the water flow into and down the ramp. This flow is expected to be negligible. If these dams are needed, they should have a permeability that is less than the effective permeability of the undisturbed rock.

If a discrete fault or fracture zone that provides a continuous supply of water is encountered in a ramp, one of the concepts for sealing faults (Section 6.2.8.6) can be installed. If extensive concrete or grout is placed on the floor or ribs of the inclined ramp, it can be removed or perforated, partially or totally, to enhance the ramp's drainage capability. Concrete or grout at the roof of the ramp may act as a diversion shield for water and, therefore, need not be removed.

6.2.8.2 Shaft and ramp seal emplacement

Before emplacing the shaft fill, it may be necessary to remove all shaft outfitting steel that is anchored to the concrete liner. Steel left in the shaft could hamper backfilling operations. At the bottom of the shafts, below the repository station, the concrete liner may be removed or perforated to permit drainage through the walls of the shaft. The shaft should then be backfilled. The backfill, as determined by the preliminary calculations (Fernandez and Freshley, 1984), could be a coarse, well-graded, unreactive material (e.g., crushed tuff), to reduce settlement and to permit the drainage of water. Further analysis and testing are required to define the grading as described in Section 8.3.3.2.

Settlement may be further reduced by selecting the proper emplacement technique, installing settlement plugs, and allowing settlement to occur before abandoning the shaft. If it is necessary to control settlement and if settlement plugs are selected, it may be desirable for each plug to have a high permeability or be designed to drain water that may collect above it. This could be accomplished by placing tubes through the plug or emplacing a no-fines concrete. The strength, spacing, and placement of the plugs require knowledge of the load exerted on each plug and the competency of the rock into which it is placed. The plugs may be keyed into the shaft for additional support. This would require removal of the liner and excavation of additional rock.

The design of shaft and ramp seals is not well enough defined at this stage of design to permit discussion of construction details. Hence, the construction method and the general construction sequence for each component of the shaft seal are not addressed. If shaft or ramp seals are incorporated into the repository design, the construction details will be provided by the advanced conceptual design and license application design. Section 8.3.3.2 provides details on establishing the need for seal components.

6.2.8.3 Borehole seal characteristics

The primary purpose of borehole seals is to ensure that the boreholes do not become preferential pathways to the accessible environment. This function applies particularly to the following:

1. Any exploratory borehole within the perimeter of the underground facility that penetrates the water table.

2. Boreholes that penetrate to the water table down-dip from the emplacement horizon and inside the boundary of the accessible environment; these boreholes could provide a preferred pathway for contaminated ground water that drains vertically from the emplacement area through the rock mass and then flows down-dip along a capillary or permeability barrier to intersect the boreholes.

The existing boreholes that will be considered for sealing are discussed in Section 8.3.3. Exploratory boreholes drilled as part of site characterization will be added to this list.

The objectives of sealing boreholes are to (1) control preferential water movement through the tuffaceous beds of the Calico Hills and (2) dissipate into the densely welded, highly fractured tuff any water entering the boreholes.

For those boreholes that could potentially act as a preferential path for any radionuclide release to the accessible environment, emplacement of a seal in the zone penetrating the tuffaceous beds of Calico Hills is suggested. A schematic drawing of the borehole sealing concept is presented in Figure 6-80. Holes may be sealed by conventional cement plugging and emplacement of a granular material.

More details for borehole sealing (i.e., boreholes that require special sealing methods, seal properties, and the types of seals important to repository performance) will be established as the conceptual design progresses. These details will be provided in the advanced conceptual design and license application design.

Information necessary to describe the key features, types of seal materials, seal material properties, and properties of the rock and ground water surrounding the boreholes that are relevant to borehole seal design is not currently available. Plans to obtain this information are presented in Section 8.3.3.2.

6.2.8.4 Borehole seal emplacement

Commercially available technology will be used to install borehole sealing components. Alternative concepts consist of grouting the entire borehole, as described previously, or grouting only in the critical zones (i.e., the Calico Hills nonwelded tuff) with granular material placed in other zones (Fernandez and Freshley, 1984).

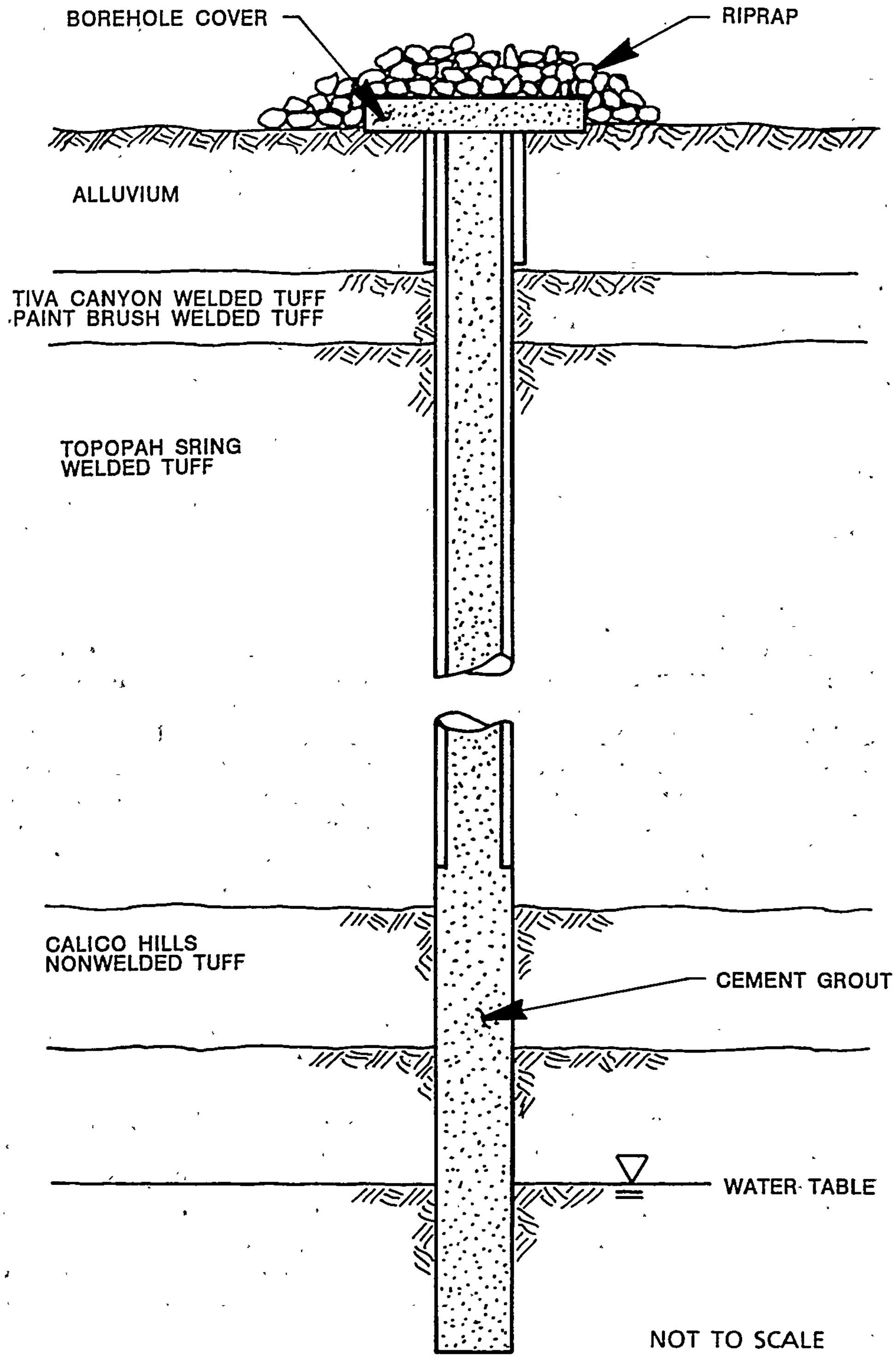


Figure 6-80. Borehole sealing concepts.

6.2.8.5 Sealing in the vicinity of waste package

Information available to date from investigations at Yucca Mountain suggests that a significant number of water-bearing faults or fractures are not likely to be encountered at the repository horizon. This information is not, however, sufficient to rule out the presence of water in fractures. Accordingly, concepts have been developed as part of the conceptual design to deal with water-bearing fractures should they be encountered in repository development.

In the vertical emplacement mode, the presence of water-bearing fractures would most likely be detected during emplacement drift development. In the horizontal emplacement configuration, information about the possible presence of water bearing fractures will be obtained as mining activities develop the access and emplacement drifts surrounding a panel.

If a water-bearing fracture with significant inflow is encountered using horizontal emplacement activities, several concepts are available for contingency planning. The first and most likely concept would be to abandon any borehole in which a significant inflow is encountered before waste emplacement.

In some instances, it may be possible to grout the permeable zone and to install a plug so that some of the borehole can be used for waste emplacement. Grouting could be performed through small-diameter boreholes drilled parallel to and around the emplacement borehole. In this instance, the borehole would be grouted after the liner had been installed. An alternative concept involves isolating the fracture zone by means of a grout plug emplaced at or near the end of the liner. The plug would be placed on the drift side of the potential inflow zone by injecting grout between packers or bridge plugs. This scheme would involve abandoning the hole beyond the plug and would only be considered if the fracture zone occurred close to the far end of the borehole.

6.2.8.6 Options for sealing a discrete fault or fracture zone in an access or emplacement drift--vertical emplacement

The prospective underground facility is located in the unsaturated zone. The semiarid conditions at Yucca Mountain, and the fact that much of the rainfall occurs in intense events of short duration, should ensure that relatively little water reaches the emplacement horizon. Preliminary information suggests that ground-water flux is low and primarily through the matrix. Nevertheless, if discrete water-producing zones are encountered several design options can be implemented to control water in the vicinity of the waste packages.

Water-producing zones in drifts can be isolated by drains or dams (Figures 6-81 and 6-82) to increase the drainage of the drift floor and to control the lateral migration of water in the drift. The drifts also can be isolated by grouting the rock above the drift (Figure 6-83) or by employing massive bulkheads to isolate large flows, if encountered.

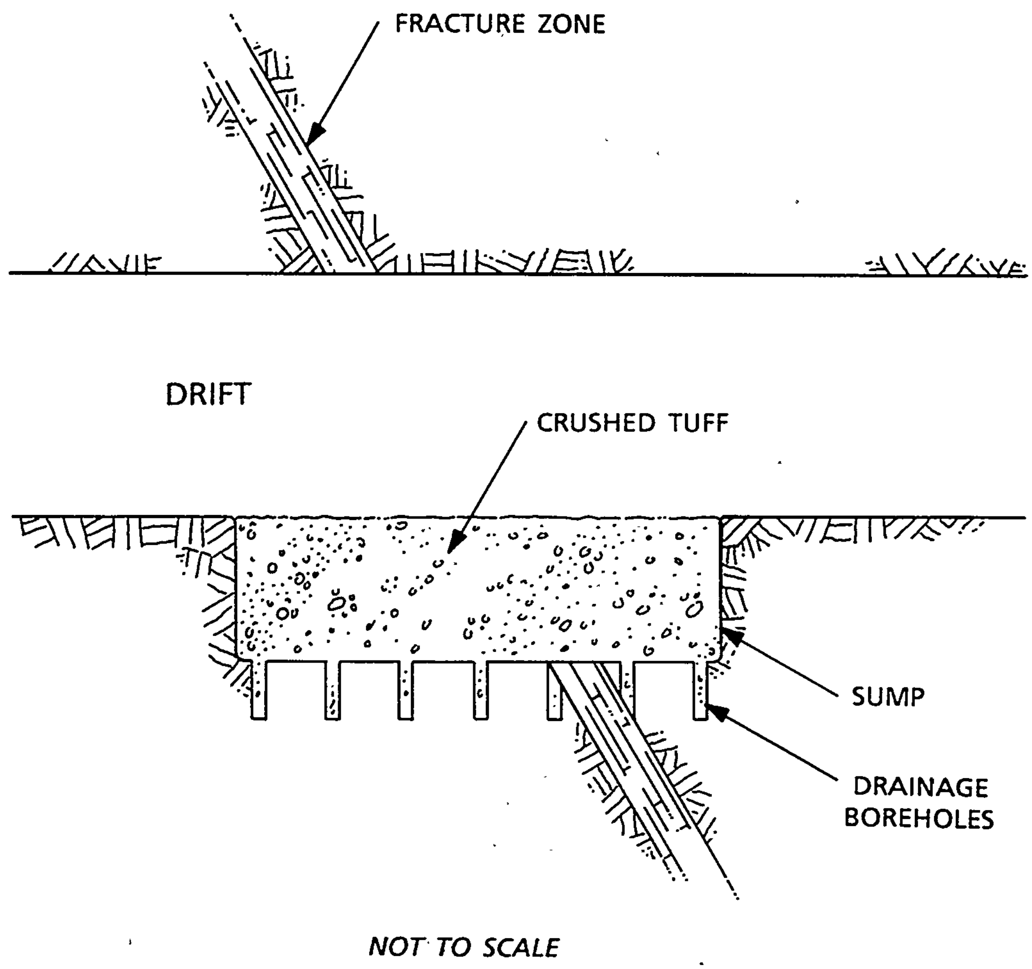
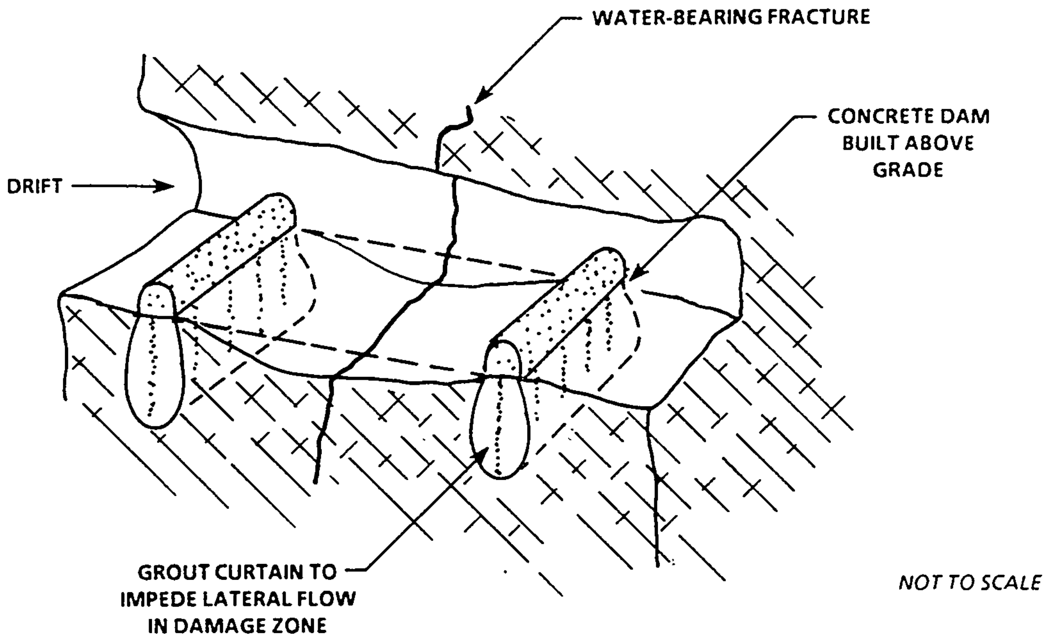


Figure 6-81. Concept for impounding and diverting water inflow using sumps and drains.



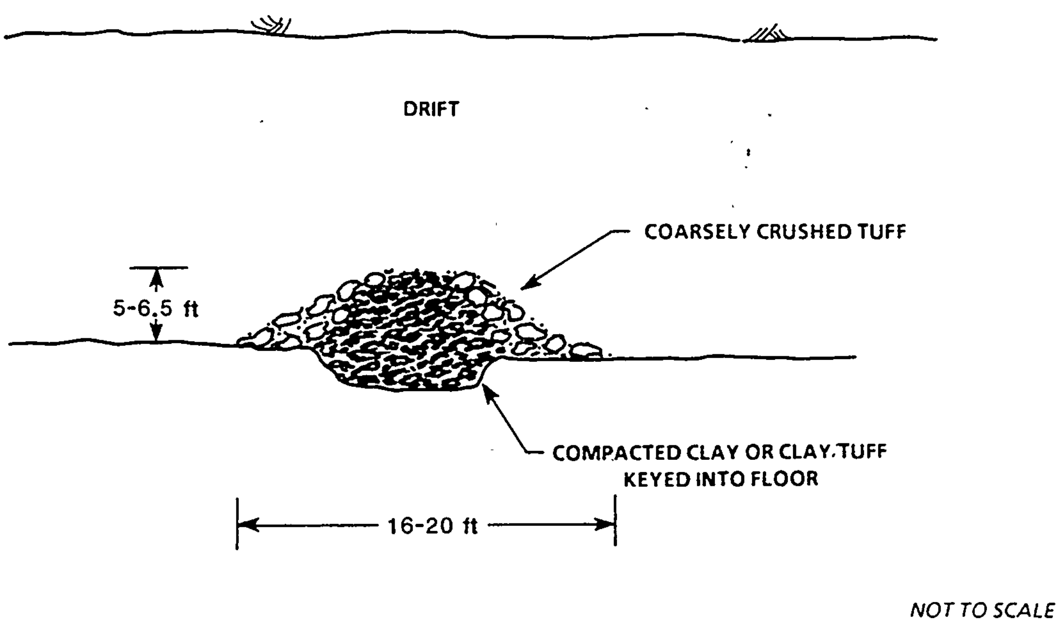
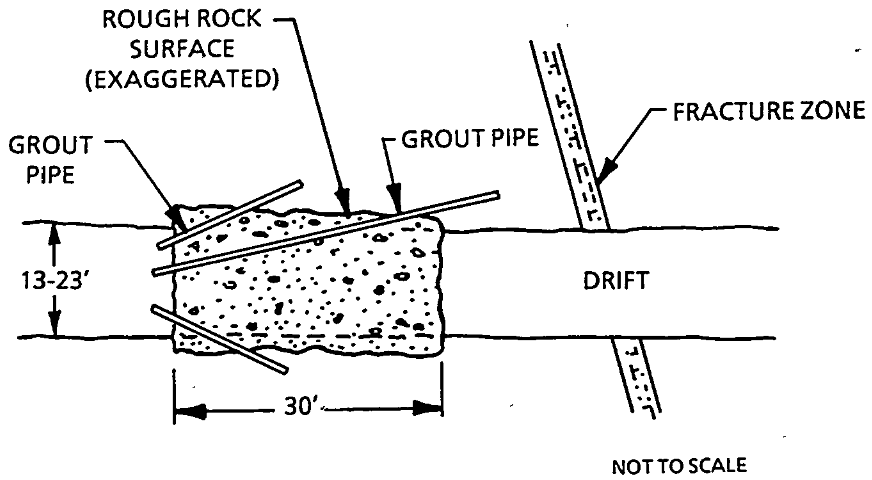


Figure 6-82. Concept for controlling water inflow with small dams.



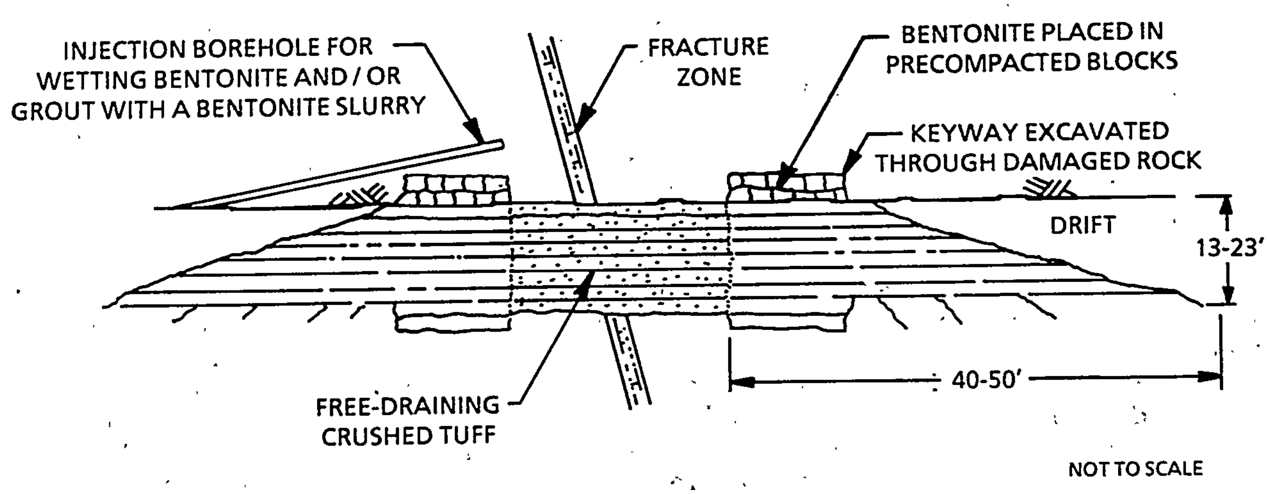


Figure 6-83. Concepts for isolating major inflows with grouting or drift bulkheads.

These options are more appropriately considered for vertically emplaced waste packages than for horizontally emplaced packages. In the latter case, the emplacement boreholes would be located at the midheight of the drift walls so that water entering from one hole could not enter other holes. The choice between the options will be determined by the expected inflow volume. Options 1 and 2 (drains and dams) are more appropriate for small inflows, whereas options 3 and 4 (grouting and bulkheads) are more appropriate for large inflows.

6.2.9 RETRIEVAL

6.2.9.1 Retrieval requirements and planning-basis time periods

The preservation of the ability to retrieve emplaced high-level radioactive waste is a federally mandated requirement. This is stated as follows: "any repository constructed on a site approved under this subtitle shall be designed and constructed to permit the retrieval of any spent nuclear fuel placed in such repository, during an appropriate period of operation of the facility ..." (NWPA, 1983, Section 122); and the geologic repository operations area shall be designed to preserve the option of waste retrieval...." for up to 50 yr after waste emplacement operations are initiated, unless the NRC specifies a different time (10 CFR Part 60.111(b)).

To comply with these retrievability requirements, the Yucca Mountain repository is designed to include the option of retrieving the emplaced waste as a planned contingency (DOE, 1986a). Inclusion of the retrieval option in the design is to be done so that it will not compromise the safety of the repository, nor will it compromise the ability of the repository to isolate the emplaced waste (Flores, 1986). Retrievability-related design criteria have been identified in Section 6.1.1.7.

The time periods used in the current conceptual design that are related to retrieval are shown in Figure 6-84. The terms "period of retrievability" and "actual retrieval period" are used to describe the time periods related to retrieval (DOE, 1986a). The period of retrievability is assumed to be 50 yr for design purposes. The actual retrieval period, the time to complete the retrieval operations after the decision is made to retrieve the waste, is assumed to be 34 yr. The construction and operating periods proposed in the Generic Requirements document (GR) (Appendix B of DOE, 1986d) are 6 yr and 28 yr, respectively.

For purposes of design the time period for which the emplaced waste must remain retrievable is the sum of the period of retrievabilty and the actual retrieval period. By combining the 50-yr period of retrievability and the 34-yr actual retrieval period, the maximum time period for design purposes from emplacement of the first waste to complete retrieval of waste is 84 yr.

In this chapter the term off-normal is used to identify conditions that are anticipated to occur infrequently. In future documents the term off-normal will be replaced with the term abnormal.

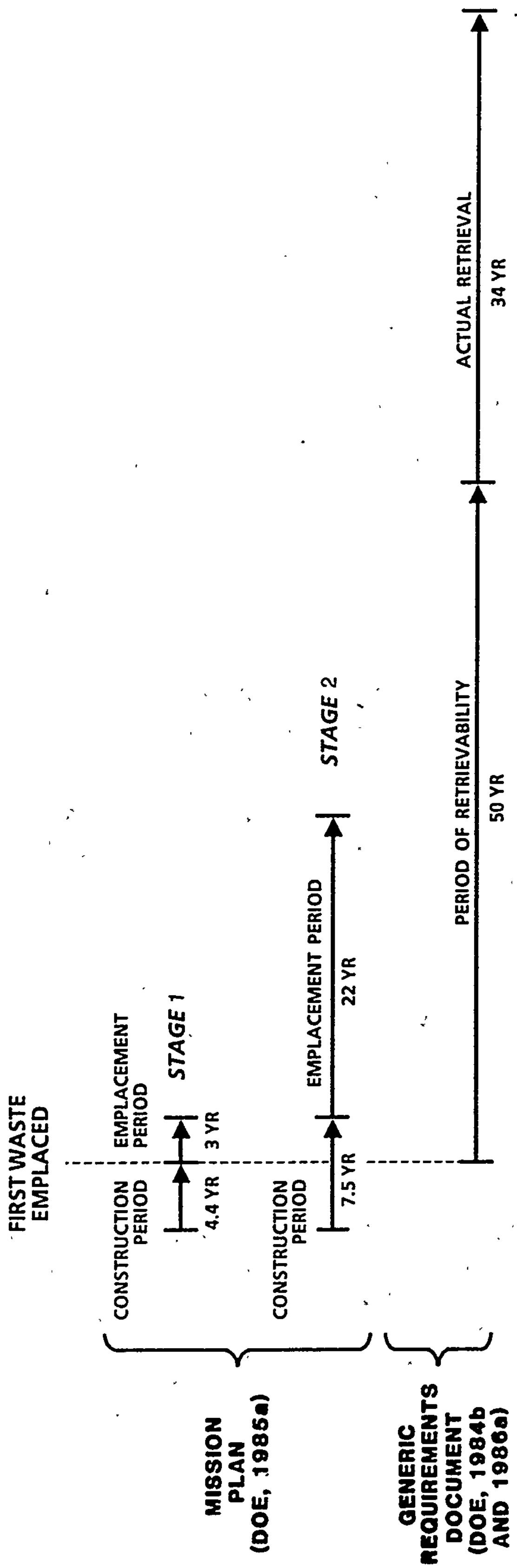


Figure 6-84. Retrieval time frame for design purposes.

6.2.9.2 Retrieval conditions

In the design work completed to date, waste retrieval conditions and, hence, operations and equipment, have been divided into two categories: normal and off-normal. Normal conditions are considered to be those conditions under which the retrieval process can be performed using standard equipment and procedures. In this discussion, standard equipment is considered to be essentially the same equipment used for waste emplacement. Minor maintenance and cleanup may be required (Flores, 1986). Off-normal conditions are considered to be those conditions under which the retrieval process must be performed using nonstandard equipment or procedures. It is important to note that the existence of an off-normal condition does not mean that the retrieval process will be particularly difficult or hazardous; in general it means that modified equipment and procedures may be used. For example, if excessive temperatures are encountered in an emplacement drift, a decision may be made to extend the cooldown period, perform the retrieval using additional thermal protection for workers, or to use a combination of these alternatives.

6.2.9.2.1 Normal retrieval conditions

The performance allocation process described in SCP Section 8.3.5.2 identifies repository system elements whose performance and, as a result, condition could affect the ability to retrieve. These elements include accesses and drifts, emplacement boreholes, ventilation system, waste-handling building, retrieval equipment, and the waste container. The normal conditions for these elements are summarized in this section; more detail is provided in Appendix J of the SCP-CDR.

Normal condition of accesses and drifts for retrieval

The normal conditions expected for retrieval operations in the accesses and drifts are characterized in terms of the following parameters:

1. Rock temperatures in the drifts.
2. Condition of the openings.
3. Radiation environment.
4. Air quality.

The normal conditions are based on the design basis that the drifts will not be backfilled until closure. Furthermore, current plans indicate that ventilation will be provided to emplacement drifts only until the emplacement process for that drift is completed. Ventilation would be reinstated for periodic inspection, maintenance, performance confirmation purposes, or retrieval until closure of the repository. Continuous ventilation of the waste emplacement ramp, access drifts, and service areas is planned until repository closure.

The surface rock temperature in the drifts is an important consideration in establishing the retrieval environment because it not only influences the ventilation requirements but also affects opening stability and retrieval equipment design. It is expected that the temperature in access drifts for vertical emplacement (emplacement drifts in horizontal) will not exceed 50°C before the end of the period of retrievability. With only brief ventilation, these drifts can be cooled to less than 40°C (the design basis temperature used for planning retrieval operations). For vertical emplacement, the emplacement drift floor temperature rises very quickly (94°C at 5 yr). The maximum temperature predicted for the floor (during the period of retrievability) is about 130°C (Section 6.4.8). Therefore, within a short period of time after waste emplacement, extensive ventilation cooling would be required to return the surface rock temperature to the environment required for retrieval.

Under normal conditions, the ramps, ventilation shafts and drifts are expected to remain stable and usable with only minor maintenance required before initiating retrieval operations. The basis for this expectation is threefold. First of all, periodic maintenance of the openings is planned throughout the period of retrievability. This implies that if areas require major maintenance, this need would have been identified and repairs made prior to the start of retrieval. Second, numerous sets of thermomechanical calculations have been performed that predict that the drifts will be stable throughout the period of retrievability (Section 6.4.8). Finally, observations and experience in the miles of drifts affected by underground testing of nuclear weapons provide indications that drifts in bedded tuffs can be maintained. It is recognized that these tuffs differ from the Topopah Spring tuff, but the experience is considered a preliminary indication of feasibility.

Under normal conditions, the radiation environment for retrieval is considered to be essentially the same as that considered for emplacement. This implies that both naturally occurring radon- and waste-related effects will be considered. The air quality requirements for retrievability operations will be the same as those for the emplacement operations.

Since the requirements for air quality are the same for retrieval and emplacement, it is expected that the actual environment (in the accesses and drifts) will be similar. Possibly the most notable exception will be related to the increased air temperature. However, precooling of the drifts before retrieval is planned so that significant differences are not likely to exist.

Normal condition of emplacement boreholes for retrieval

The normal conditions expected for retrieval operations in the boreholes are characterized in terms of the following:

1. Rock temperature.
2. Condition of the boreholes.
3. Condition of the borehole liner.
4. Radiation.

The predicted temperatures in the emplacement boreholes are presented in Figure 6-85. As shown in the figure, the maximum predicted temperatures for the wall in the vertical and horizontal emplacement boreholes are approximately 227 and 214°C, respectively. In addition, the temperature of the borehole wall is expected to remain above 140°C for 100 yr after waste emplacement. Additional detail is given in Appendix J of the SCP-CDR.

In the vertical emplacement concept, the boreholes are expected to be stable and the amounts of loose rock in the boreholes are expected to be negligible. Similar conditions are expected for the horizontal borehole. The status of supporting thermo/mechanical analyses is discussed in Section 6.4.10.

The liner is expected to remain intact and provide acceptable access to the emplaced waste containers throughout the design-basis, 84-yr period. Both corrosion rates and rockfall-induced loading are predicted to be small (Appendix J of SNL, 1987).

Normal condition of ventilation system for retrieval

The entire ventilation is planned to be maintained in fully operational condition throughout the caretaker period; hence the system will be available for use if retrieval operations are necessary.

Normal condition of waste-handling building for retrieval

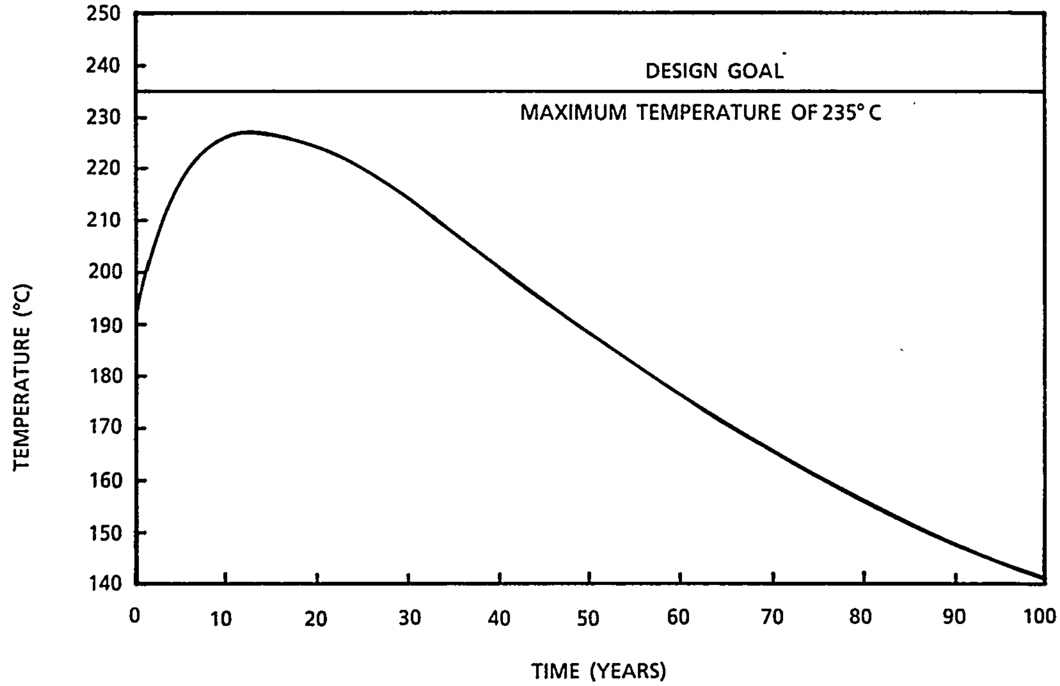
If waste emplacement operations are in progress, the waste-handling building is expected to be in operable condition. However, the current planning basis is that the building will not be constructed for reverse operations for full retrieval and, therefore, extensive modifications and additional construction would be necessary to accomplish full retrieval. In the current plan, the equipment located in the waste-handling building will not be maintained in an operational state during the caretaker period. It is planned that maintenance and repair will have been performed to maintain the structure during this period.

Normal condition of equipment for retrieval

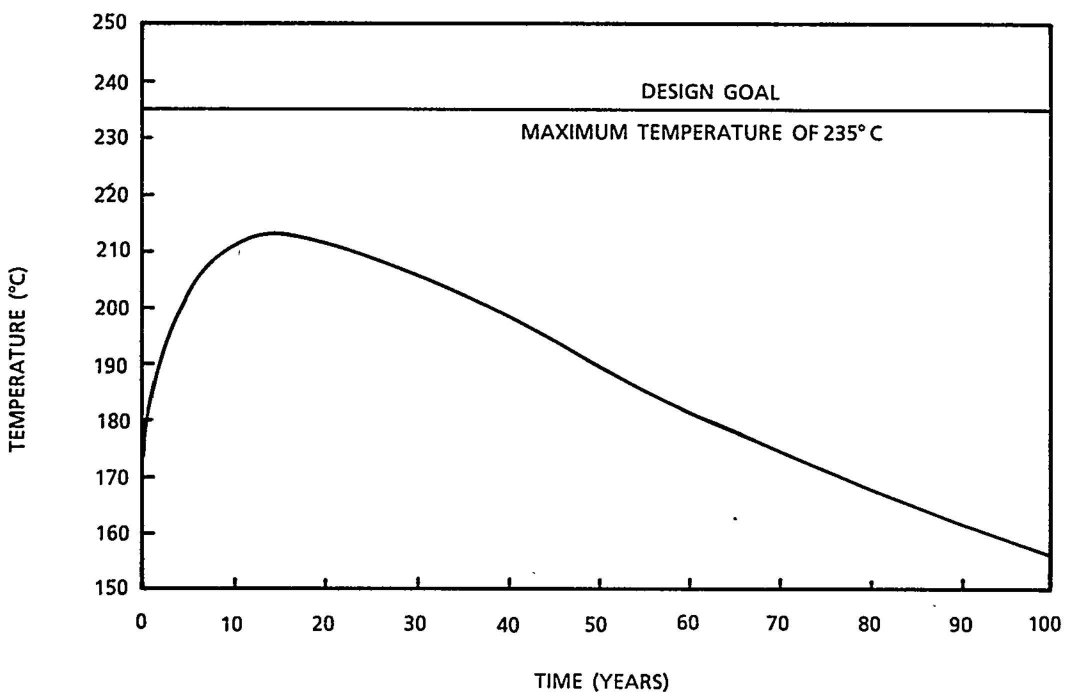
If waste emplacement operations are in progress, the equipment required for waste removal under normal conditions will be in operational condition because the current design basis includes using the same equipment (transporter, auxiliary equipment, and shield collar) for both emplacement and retrieval operations. During the caretaker phase, two sets of equipment are planned to remain operational and two other sets would require maintenance and possibly repair before starting to retrieve waste.

Normal condition of waste container for retrieval

Under normal conditions, the waste container is expected to remain intact up to and during the retrieval process. This expectation is based on the low corrosion rate and the low loading expected on the container.



BOREHOLE WALL TEMPERATURE – VERTICAL EMPLACEMENT



BOREHOLE WALL TEMPERATURE – HORIZONTAL EMPLACEMENT

Figure 6-85. Predicted temperatures for emplacement boreholes.

6.2.9.2.2 Off-normal retrieval conditions

Off-normal conditions exist when the retrieval process must be performed with nonstandard equipment or procedures. The existence of an off-normal condition does not necessarily mean that retrieval is impossible or particularly hazardous; in general it means that modified equipment and procedures may be used. For example, if higher than expected temperatures are encountered in an emplacement drift, a decision may be made to extend the planned cooldown period, perform the retrieval using additional thermal protection for workers, or to use a combination of these alternatives.

The off-normal retrieval conditions presented here were developed as part of evaluation completed in support of the development of the SCP-CDR (Appendix L of SNL, 1987). This development began with a comprehensive list of approximately 75 processes and events of potential concern. From the initial screening, the following processes and events, which could affect the ability to retrieve, were identified:

1. Tectonics.
2. Variability in rock characteristics.
3. Human error.
4. Aging and corrosion of equipment and facilities.
5. Radiolysis.

With the use of engineering judgment, these events and processes were evaluated relative to the SCP-CD. The four functions that must be performed to successfully complete the retrieval operation are: (1) provide access to the emplacement boreholes, (2) provide access to the waste containers, (3) remove the waste containers, and (4) transport and deliver the waste to the surface facilities. As a result of this evaluation, the off-normal conditions were identified. A list of the potential off-normal conditions is provided in Table 6-28.

The derivation of this list of off-normal conditions consisted of a preliminary screening to estimate the frequency of occurrence of various events and processes. The analysis was not intended to be extremely detailed. The intention was to use the results to provide guidance relative to areas needing further investigation. The potential for combined effects (i.e., the occurrence of two or more of the events or processes at the same time) was not considered in this development. In the future, the table will be used as a starting point for the application of probability evaluations and for quantification of the effect of the conditions. This is intended to aid in (1) the development of more detailed design and operational criteria for equipment and facilities, (2) the more accurate guidance for equipment development requirements and demonstration needs, and (3) a firm basis for deciding what equipment will be on hand at the repository.

Table 6-28. Potential off-normal conditons for retrieval (page 1 of 4)

Cause of condition	Component	Postulated result
Tectonics--seismic (0.45 to 0.60 g event)	Ramp	Localized rockfall within ramp
	Ventilation system	Malfunction of ventilation system surface damage of ventilation equipment
	Transmission line	Loss of offsite power
	Waste container--vertical orientation	Rockfall around waste container in vertical emplacement borehole
	Waste container--horizontal orientation	Lateral movement of waste container on the order of a few centimeters
	Waste container--vertical orientation	Movement (tilt) of waste container within borehole
	Shield plug	Jamming of shield plug resulting from borehole distortion
	Transporter cask and emplacement collar	Binding of transporter cask and emplacement collar during removal operation
Tectonics--faulting	Ramp	Localized rockfall within ramp
	Drift	Localized rockfall within drift
	Borehole and liner-horizontal orientation	Excessive liner deflection. Shearing of borehole and liner (although the probability of occurence for a shear of the borehole and liner is extremely low, it is included because of the potential consequences

Table 6-28. Potential off-normal conditons for retrieval (page 2 of 4)

Cause of condition	Component	Postulated result
Variability of rock (area of reduced mechanical strength characteristics)	Ramp	Localized rockfall within ramp
	Drift	Localized rockfall within drift
	Borehole--vertical orientation	Rockfall in borehole around vertically emplaced waste container as a result of reduced mechanical strength
Human error during fabrication	Heating, ventilating, and air conditioning (HVAC) system	Failure of ventilation system
	Liner	Excessive deflection of liner
	Collar	Malfunction of collar
	Auxiliary equipment	Malfunction of auxiliary equipment
	Shield plug	Jamming of shield plug
	Dolly - horizontal orientation only	Failure of dolly
Human error during maintenance	Ramp	Rockfall within ramp
	Drift	Rockfall within drift
	HVAC system	Malfunction of HVAC system
	Collar and collar attachment	Malfunction of collar or collar attachment
	Transporter (removal)	Malfunction of transporter during waste removal

Table 6-28. Potential off-normal conditons for retrieval (page 3 of 4)

Cause of condition	Component	Postulated result
Human error during maintenance (continued)	Transporter (transport)	Malfunction of transporter during waste transport
	Transporter (unloading)	Malfunction of transporter during waste unloading
	Surface facility interface	Malfunction of surface facility interface during unloading of waste
Human error during operation	Transporter	Collison of transporter with ramp while transporter is moving up or down the ramp
	Transporter	Collision of transporter with auxiliary equipment
	Transporter	Collision of transporter with another transporter
	Transporter cask-borehole collar	Alignment error between transporter cask and borehole collar
	Waste container	Excessive thermal loading of waste container resulting from incorrect determination of thermal output of waste container
	Waste container	Incorrect emplacement of waste container in borehole

Table 6-28. Potential off-normal conditons for retrieval (page 4 of 4)

Cause of condition	Component	Postulated result
Human error during operation (continued)	Waste container	Alignment error during removal of waste container
	Transporter cask-surface facility loading port	Alignment error during alignment of transporter cask and surface facility loading port
	Waste container	Unloading error at surface faciilty
Agining or corrosion of equipment of facilities	Rockbolt	Failure of rockbolt resulting from corrosion
	Dolly (horizontal emplacement)	Failure of waste container dolly resulting from corrosion
Radiolysis	Liner	Corrosion rates of liner above expected levels
	Dolly (horizontal orientation only)	Corrosion rates of dolly above expected levels
	Waste container	Corrosion rates of waste container above expected levels

6.2.9.3 Equipment development

In early design stages, the feasibility of developing the equipment necessary to perform the retrieval operations is an important consideration in evaluating the ability to perform the operations. As more detailed designs are developed, proof-of-principle testing can be planned for completion before license application. At present, design concepts exist for the retrieval equipment being considered for both the vertical and horizontal emplacement options. Operations under both normal and off-normal conditions have been considered in these concepts.

The design discussions presented in this chapter focus on the design aspects that are most related to the site. Therefore, the development to date regarding retrieval-related equipment proposed for use in a repository will be only briefly summarized here. Substantially more detail is provided in Sections 3.2, 4.5, 6.3, and related appendices of the SCP-CDR.

Lists and estimated quantities of equipment needed to perform the retrieval operations for both emplacement operations (under normal and off-normal conditions) are provided in Section 3.2 of the SCP-CDR. Additional information is provided to indicate whether the equipment is either currently available or may require some development for its intended use in the NNWSI Project. Both baseline and alternative concepts are described and operations are identified that will enhance maintaining the ability to retrieve the waste. Operations that could be carried out to facilitate retrieval under off-normal conditions are described considering both access to the emplacement boreholes and waste removal from the holes. Conditions affecting access or removal are identified and then procedures for overcoming the conditions are described.

The equipment intended for use in the operations are described in Section 4.5 of the SCP-CDR for both emplacement options. These descriptions reflect conceptual designs for the equipment planned for use under normal conditions and design concepts being considered for selected off-normal conditions. It is important to note that many of the operational features of equipment planned for use in handling of the waste have been demonstrated at the Climax facility on the Nevada Test Site (Patrick, 1985) for the vertical emplacement option. Some aspects of the equipment planned for use in the vertical emplacement option will need to be developed further and may require some demonstration. For the horizontal emplacement option, essentially no field demonstrations have been completed. A one-twelfth scale model has been developed (White et al., 1986) as part of ongoing evaluations of the feasibility of horizontal emplacement and retrieval. If the horizontal emplacement option is selected for use in the proposed repository, more equipment development and demonstration would be required than would be necessary for the vertical option.

6.3 ASSESSMENT OF DESIGN INFORMATION NEEDS

6.3.1 INTRODUCTION

To ensure the completeness of the site characterization program for the Yucca Mountain mined geologic disposal system (MGDS), the DOE has chosen to develop this program around an issues hierarchy. The bases for this issues hierarchy are the four key issues identified in the DOE Mission Plan (DOE, 1985a). Each of the four key issues is divided into two components: (1) system performance issues, and (2) facility design issues. Summaries of the strategies for resolving each of these issues are presented in Section 8.2. The complete issue resolution strategy (IRS) for each issue that requires site characterization information is presented in Section 8.3.

The purpose of Section 6.3 is to provide a bridge between the topics identified in Chapter 6 of the NRC Regulatory Guide 4.17 (NRC, 1987) and the issues that address these topics. In each instance where a NRC Regulatory Guide 4.17 topic is identified, the subsection headings of 4.17 are used as the subsection headings for this section. In addition the following information is provided:

1. Identification of the subsection of Section 8.3 that (a) identifies the concern in the context of the resolution of the associated issue and (b) presents the data requirements, including the current and needed confidence.

2. Identification of the subsection of Section 6.4 that (a) summarizes the analytical work completed, and (b) presents plans for future analytical work.

This method of presentation was selected to provide a perspective for the treatment of the individual NRC Regulatory Guide 4.17 (NRC, 1987) topics within the context of the DOE issue resolution strategies.

6.3.2 DESIGN OF UNDERGROUND OPENINGS

6.3.2.1 Exploratory shaft facility

The exploratory shaft facility (ESF) will be located near the eastern edge of the MGDS underground facilities as shown in Figures 6-13 and 6-14 (Section 6.2.2). The ESF will consist of a surface facility that provides office, shop, and warehouse space; a 4-m- (12-ft-) diameter shaft; a 2-m- (6-ft-) diameter shaft; and the underground drift complex. Further discussion of the ESF is provided in Section 8.4 of this document. The general arrangement of the shafts is shown in Figure 6-57 (Section 6.2.5).

The relationships of the ESF to the MGDS underground facilities, shafts, and ramps are shown in Figure 6-56 (Section 6.2.5). The known or inferred geologic and hydrologic conditions at the site, as determined from preliminary site investigations, are discussed in Chapters 1, 2, and 3. The associated design parameters are summarized in Section 6.1.2 (reference design data base). For design purposes, the geologic and hydrologic properties at the ESF location are expected to be representative of those obtained during preliminary site investigations.

6.3.2.2 Layout of the mined geologic disposal system underground facilities

The proposed layouts of the underground facilities for both the vertical and horizontal configurations are discussed in Section 6.2.6 (subsurface design), and are shown in Figures 6-59 through 6-64 in Section 6.2.6.

The design criteria for the MGDS underground facilities have been developed under the relevant performance and design issues. The following list provides the section in Chapter 8 where the issue resolution strategy and data requirements for each issue are discussed and the section in Chapter 6 where the completed work, future work, and the data needs are presented:

Issue	Subject	Chapter 8 section	Chapter 6 section
1.11	Configuration of underground facilities (postclosure)	8.3.2.2	6.4.2
2.1	Public radiological exposures --normal conditions	8.3.5.3	6.4.4
2.2	Worker radiological safety --normal conditions	8.3.5.4	6.4.5
2.3	Accidental radiological releases	8.3.5.5	6.4.6
2.4	Waste retrievability	8.3.5.2	6.4.8
2.7	Repository design criteria for radiological safety	8.3.2.3	6.4.7
4.2	Nonradiological health and safety	8.3.2.4	6.4.9

The initial layouts for the underground facility were developed based on the additional design criteria for the underground facility--10 CFR 60.133a (1) and (2), and on the criteria for thermal loads--10 CFR 60.133(i). The following factors control these initial layouts:

1. Waste quantities.
2. Waste container thermal output.
3. Maximum waste container temperature.

4. Near-field thermal and mechanical constraints.
5. Far-field thermal and mechanical constraints.
6. Separation of waste emplacement and mine development areas.

6.3.2.3 Shafts, ramps, drifts, and waste emplacement boreholes

The design concepts for the shafts and ramps are presented in Section 6.2.5 (shaft and ramp design). The general arrangements of the shafts and ramps are shown in Figures 6-56 through 6-60 of Section 6.2.5. The data for the shafts and ramps are summarized in Table 6-19 (Section 6.2.5).

The design concepts for the drifts and waste emplacement boreholes, both vertical and horizontal, are presented in Section 6.2.6 (subsurface design). Typical waste emplacement drift and ramp cross-sections for the vertical waste emplacement configuration are shown in Figure 6-60. Similarly, typical waste emplacement drift and ramp cross-sections for the horizontal waste emplacement configuration are shown in Figure 6-61. Typical emplacement drift cross-sections with alternative ground support systems are shown in Figure 6-70. The mining methods for the drifts are summarized in Table 6-23. The underground development equipment is shown in Figures 6-71 and 6-72. All these figures and Table 6-23 are in Section 6.2.6.

The following basic criteria govern the design of the underground openings:

1. Equipment clearances.

2. Utility clearances.

3. Ventilation flows that were used to establish the minimum requirements for the cross-sectional area of the shafts, ramps, and drifts.

4. Opening stability.

5. Maximum duration required for access to the underground.

6. Drainage and material property constraints.

The discussion of completed and future work, and the identification of data needs is presented in Section 6.4.10 for Issue 4.4--(preclosure design and technical feasibility). The discussion of data requirements is in Section 8.3.2.5.

6.3.2.4 Worker safety

The following table provides the section in Chapter 8 where the data needs for each worker safety issue are discussed and the section in Chapter 6 where the completed and future work and the data needs are presented.

Issue	Subject	Chapter 8 section	Chapter 6 section
2.2	Worker radiological safety --normal conditions	8.3.5.4	6.4.5
2.3	Accidental radiological releases	8.3.5.5	6.4.6
4.2	Nonradiological health and safety	8.3.2.4	6.4.9

Section 6.2.3.3 (accident analysis) identifies the design approach currently employed.

Air quality is the principal industrial health and safety concern for underground facilities. The function of the underground ventilation systems is to provide quality air to workers in the underground facilities. There are two separate ventilation systems, one for the waste emplacement area and one for the underground development area. These systems are described in Section 6.2.6.5 (ventilation).

6.3.3 BACKFILL

The backfilling of the MGDS drifts is discussed in Section 6.2.7 (backfill of underground openings). This discussion supports the DOE program position regarding backfill at the Yucca Mountain site as follows:

1. The option to backfill the repository drifts will be maintained throughout the retrievability period.

2. The reference case for planning the closure and decommissioning operations, cost, and schedule includes backfill.

Current planning calls for backfilling of the shafts and ramps during the closure and decommissioning of the mined geologic disposal system facilities. The discussion of backfilling shafts and ramps is included in Section 6.3.5 (seals).

Backfilling is being addressed under the issues in the following table. The table also gives the section in Chapter 8 where the data needs for each issue are discussed and the section in Chapter 6 where the completed and future work and the data needs are presented.

Issue	Subject	Chapter 8 section	Chapter 6 section
1.12	Seal characteristics	8.3.3.2	6.4.3
4.4	Preclosure design and technical feasibility	8.3.2.5	6.4.10

6.3.4 STRENGTH OF ROCK MASS

The test requirements necessary to supplement or confirm the preliminary design values used for the thermal and mechanical properties of rock are described in Section 8.3.1.4 (rock characteristics) and Section 8.3.1.15 (thermal and mechanical rock properties). The rock mass information presented in Section 6.1.2 (reference design data base) has been used in the development of the MGDS conceptual design. The site characterization information, which will be obtained during site characterization activities, will be recorded in the NNWSI Project Reference Information Base (RIB) as it is revised periodically (Appendix Q of SNL, 1987). The RIB is a controlled information base that will be used as a data source by the designers of the MGDS facilities during license application design activities.

The information needed to validate the analytical methods used to predict preclosure and postclosure MGDS performance relative to the relationship between intact rock properties and rock mass properties is identified in Issue 1.11 (configuration of the underground facilities) and Issue 4.4 (preclosure design and technical feasibility). The experiments that will be conducted in the exploratory shaft facility to obtain information relative to the relationship between intact rock properties and rock mass properties are identified in Section 8.3.1.4 and Section 8.3.1.15. In particular, discussions include data needs and experiments that address the following: (1) elastic and inelastic behavior of rock mass, (2) thermomechanical behavior of rock mass, and (3) mechanical behavior of rock discontinuities. The effects of radiation on thermal and mechanical rock properties have been identified as needed information in Issue 4.4.

The issues covering strength of rock mass and the corresponding Chapters 8 and 6 sections are summarized in the following table.

Issue	Subject	Chapter 8 section	Chapter 6 section
1.11	Configuration of underground facilities (postclosure)	8.3.2.2	6.4.2
4.4	Preclosure design and technical feasibility	8.3.2.5	6.4.10
	Rock characteristics (postclosure)	8.3.1.4	6.1.2

Issue	Subject	Chapter 8 section	Chapter 6 section
	Thermal and mechanical rock properties	8.3.1.15	6.1.2

The reader is directed to the discussions of future work and the identification of data needs in Section 6.4.2 for Issue 1.11 and in Section 6.4.10 for Issue 4.4. The experiments that will be conducted to obtain this information are identified in Section 8.3.1.4 and Section 8.3.1.15.

6.3.5 SEALING OF SHAFTS, EXPLORATORY BOREHOLES, AND UNDERGROUND OPENINGS

The conceptual design for sealing of shafts, ramps, exploratory boreholes, discrete faults, and fracture zones are discussed in Section 6.2.8 and are addressed under Issue 1.12 (seal characteristics). Design concepts reflect the fact that the underground facilities are located above the static water table (Figures 6-78 through 6-80 in Section 6.2.8).

The discussion of the completed and future work, and the identification of data needs is provided in Section 6.4.3 for Issue 1.12, and the discussion of data requirements is in Section 8.3.3.2.

6.3.6 CONSTRUCTION

The construction sequence and excavation methods are described in Section 6.2.6.1 (excavation, development, and ground support), and the construction of the facility is addressed under Issue 4.4 (preclosure design and technical feasibility).

The exploratory shaft facility (ESF) will be incorporated into the underground facilities and will become an integral part of the underground facility during the first phase of underground facility construction. Thus, the ESF facilities, the 4-m (12-ft) shaft, the 2-m (6-ft) shaft, and those drifts that directly tie the shafts to the underground facilities will be considered as part of the facilities. The following actions are planned:

1. Impose a quality assurance program on the design and construction of the ESF compatible with licensing requirements.

2. Impose a quality assurance program, compatible with licensing requirements, on all construction and maintenance activities conducted in the ESF before incorporation into the facilities.

3. If required at the time of license application, incorporate plans in the final procurement and construction design (FPCD) to rework the ESF design to meet licensing needs.

These actions will be taken to ensure that site integrity is maintained during the initial period when the ESF is incorporated into the MGDS underground facilities.

The construction methods being considered for development of the underground facilities are identified and discussed in Section 6.2.6.1.2 (mining methods). Preliminary site investigations indicate that site conditions are favorable for underground development and that drill-and-blast mining techniques can be used. No known or inferred site conditions that would require specialized construction techniques have been identified. However, tunnel boring machines are planned to be used to construct the waste ramp, the tuff ramp, the waste main, and the perimeter drifts to minimize the damage zone associated with the development.

Ground support methods are discussed in Section 6.2.6.1.4. Only conventional ground support methods are planned. Reinforced concrete liners are planned to be used in the shafts, and friction-type bolts, grouted dowels, and wire mesh are the principal support components in the balance of the underground facilities.

The control, collection, and disposal of ground water are discussed in Section 6.2.6.4 (ground-water control). The facilities are proposed to be located in the unsaturated zone above the static water table. The preliminary site investigation has not identified any perched water within the horizon selected for the underground facilities. Therefore, control, collection, and disposal of ground water are not expected to pose any significant problems at the Yucca Mountain site.

The construction plans for the facilities will be reevaluated after information is available from the ESF and after the advanced conceptual design, the license application design, and the FPCD are completed.

The reader is directed to the discussion of completed and future work, and the identification of data needs in Section 6.4.10 for Issue 4.4 (preclosure design and technical feasibility) and to the discussion of data requirements in Section 8.3.2.5.

6.3.7 DESIGN OF SURFACE FACILITIES

The design of surface facilities is described in Section 6.2.4 (design of surface facilities) and under Issue 4.4 (preclosure design and technical feasibility). The discussion presented in this document is limited to general information and directed toward those design features requiring information that will be obtained during the site characterization activities. A more detailed description of the surface facilities is contained in the SCP-CDR.

The site characterization activities for surface facility design are principally directed toward obtaining parameters that relate to the following:

1. Surface materials and soil characterization.

2. Surface flooding potential.
3. Surface topography.
4. Sources of water.
5. Seismic design parameters.
6. Characteristics of mined tuff.

The reader is directed to the discussion of completed and future work, and the identification of data needs in Section 6.4.10 for Issue 4.4 and to the discussion of data requirements in Section 8.3.2.5.

6.3.8 MINED GEOLOGIC DISPOSAL SYSTEM COMPONENT PERFORMANCE REQUIREMENTS

Wherever possible preliminary numerical values for the performance goals have been established and these values are stated in Section 8.3. In some instances (e.g., the use of reasonably available technology), numerical goals have not been and cannot be established. The reader is reminded that early assignment of numerical goals for systems and components cannot be accomplished with a high degree of accuracy and that revisions of these numerical values will occur as the design of the mined geologic disposal system (MGDS) matures.

The reader is directed to the discussion of the performance goals contained in Section 8.3 and in particular to the following sections:

SCP section	Subject	Issue
8.3.2.2	Configuration of underground facilities (postclosure)	Issue 1.11
8.3.2.3	Repository design criteria for radiological safety	Issue 2.7
8.3.2.4	Nonradiological health and safety	Issue 4.2
8.3.2.5	Preclosure design and technical feasibility	Issue 4.4
8.3.3.2	Seal characteristics	Issue 1.12
8.3.5.2	Waste retrievability	Issue 2.4
8.3.5.3	Public radiological exposures --normal conditions	Issue 2.1
8.3.5.4	Worker radiological safety --normal conditions	Issue 2.2
8.3.5.5	Accidental radiological releases	Issue 2.3

6.4 SUMMARY OF DESIGN ISSUES AND DATA NEEDS

6.4.1 PURPOSE AND ORGANIZATION

The purpose of Section 6.4 is to (1) describe the current status of the studies and analyses that have been completed as part of the facility design activities and (2) summarize the future studies and analyses necessary to complete the design activities and the additional site data needed to support these studies and analyses. The organization of Section 6.4 is based on the design-related issues that are part of the NNWSI Project issues hierarchy.

The NNWSI Project issues hierarchy is described in detail in Section 8.2 of this document and summarized by shortened titles in Figure 6-86. Briefly, the highest level of the hierarchy consists of four key issues, which were first defined in the DOE Mission Plan (DOE, 1985a). Issues form the second level of the hierarchy and are grouped as performance assessment and design issues under each key issue. Regulatory and functional requirements imposed on the MGDS are embodied in the issues.

The third level consists of information needs. Information needs are convenient groupings of activities and data needs appropriate to the resolution of an issue. Examples include such activities as determining detailed characteristics of the site, designing the engineered subsystems and components, analyzing performance of the natural and engineered subsystems and components, as necessary for the resolution of each issue.

Section 8.1 describes the generic strategy for resolving design and performance issues. Briefly, the issue resolution strategy for design and performance issues uses the following five-step procedure:

1. Identify the system elements of the Yucca Mountain MGDS that participate in meeting the regulatory (performance) requirements addressed by the issue. Performance allocation will be applied to these system elements and to the functions and processes applicable to these system elements, as identified in the following steps. The hierarchy of system elements for the Yucca Mountain MGDS is given in Section 8.2.1.

2. For each system element, identify the function(s) that the element must perform for the MGDS to meet the specified requirement(s). Note that in several design issues, it was found to be more convenient to define the functions in step 1, and then identify the numerous system elements that participate in performing each function. In those issues where this alternative approach is taken, the reader is appropriately alerted.

3. For each function, the processes used to perform the function are identified.

4. For each process, performance measures are defined. A performance measure is an indicator that will be used to evaluate the performance of a process.

KEY ISSUE 1 POSTCLOSURE PERFORMANCE

PERFORMANCE ISSUES
- 1.1 TOTAL SYSTEM PERFORMANCE
- 1.2 INDIVIDUAL PROTECTION
- 1.3 PROTECTION OF GROUND WATER
- 1.4 CONTAINMENT BY WASTE PACKAGE
- 1.5 ENGINEERED BARRIER SYSTEM RELEASE RATES
- 1.6 GROUND-WATER TRAVEL TIME
- 1.7 PERFORMANCE CONFIRMATION
- 1.8 NRC SITING CRITERIA
- 1.9 HIGHER LEVEL FINDINGS - POSTCLOSURE

DESIGN ISSUES
- 1.10 WASTE PACKAGE CHARACTERISTICS (POSTCLOSURE)
- 1.11 CONFIGURATION OF UNDERGROUND FACILITIES (POSTCLOSURE)
- 1.12 SEAL CHARACTERISTICS

KEY ISSUE 2 PRECLOSURE RADIOLOGICAL SAFETY

PERFORMANCE ISSUES
- 2.1 PUBLIC RADIOLOGICAL EXPOSURES -- NORMAL CONDITIONS
- 2.2 WORKER RADIOLOGICAL SAFETY -- NORMAL CONDITIONS
- 2.3 ACCIDENTAL RADIOLOGICAL RELEASES
- 2.4 WASTE RETRIEVABILITY
- 2.5 HIGHER LEVEL FINDINGS -- PRECLOSURE RADIOLOGICAL SAFETY

DESIGN ISSUES
- 2.6 WASTE PACKAGE CHARACTERISTICS (PRECLOSURE)
- 2.7 REPOSITORY DESIGN CRITERIA FOR RADIOLOGICAL SAFETY

KEY ISSUE 3 HEALTH, SAFETY, ENVIRONMENT, SOCIOECONOMIC, TRANSPORTATION

KEY ISSUE 4 PRECLOSURE PERFORMANCE

PERFORMANCE ISSUE
- 4.1 HIGHER LEVEL FINDINGS - EASE AND COST OF CONSTRUCTION

DESIGN ISSUES
- 4.2 NONRADIOLOGICAL HEALTH AND SAFETY
- 4.3 WASTE PACKAGE PRODUCTION TECHNOLOGIES
- 4.4 PRECLOSURE DESIGN AND TECHNICAL FEASIBILITY
- 4.5 REPOSITORY SYSTEM COST EFFECTIVENESS

Figure 6-86. Nevada Nuclear Waste Storage Investigations (NNWSI) Project issues hierarchy.

5. For each performance measure, performance goals and associated current and needed confidence levels are assigned. Performance goals reflect the regulatory or functional requirements as allocated to the system elements, functions, and processes. The confidence is either a numerical level or nonnumerical level, such as high, medium, or low, that indicates the importance (from an issue resolution standpoint) of an individual performance measure meeting its assigned goal.

Using this generic approach, Sections 8.3.2 through 8.3.5 describe the specific strategies and plans for resolving each issue requiring information about site characteristics. The details of the above five-step process are described at the issue level for each design and performance issue in Section 8.3. The discussions of the information needs for each issue describe how each information need is related to the processes or functions of that issue and how the activities undertaken to satisfy the information need contribute to the resolution of the issue.

Several of the issues found in Section 8.3.2, Repository program, 8.3.3 Seals Program, and Section 8.3.5, Performance assessment program, are directly related to the design of the surface and subsurface MGDS facilities, the subject of Chapter 6. While Section 8.3 gives the specific plans for resolving each of these issues, Sections 6.4.2 through 6.4.11 give the status of work already completed relative to these issues. This relationship is shown in the following table.

SCP section 6.4	Issue number	Short title of the issue	Related 8.3 section
6.4.2	1.11	Configuration of underground facilities (postclosure)	8.3.2.2
6.4.3	1.12	Seal characteristics	8.3.3.2
6.4.4	2.1	Public radiological exposures --normal conditions	8.3.5.3
6.4.5	2.2	Worker radiological safety --normal conditions	8.3.5.4
6.4.6	2.3	Accidental radiological releases	8.3.5.5
6.4.7	2.7	Repository design criteria for radiological safety	8.3.2.3
6.4.8	2.4	Waste retrievability	8.3.5.2
6.4.9	4.2	Nonradiological health and safety	8.3.2.4
6.4.10	4.4	Preclosure design and technical feasibility	8.3.2.5
6.4.11	4.5	Repository system cost effectiveness	8.2.2.3.2.4

6.4.2 ISSUE 1.11: CONFIGURATION OF UNDERGROUND FACILITIES (POSTCLOSURE)

6.4.2.1 Introduction

The question asked by Issue 1.11 is

> Have the characteristics and configurations of the repository and repository engineered barriers been adequately established to (a) show compliance with the postclosure design criteria of 10 CFR 60.133, and (b) provide information for the resolution of the performance issues?

The regulatory requirements addressed by this issue contained in 10 CFR 60.133 are the parts that address postclosure performance. The other parts of 10 CFR 60.133 that regulate preclosure performance are addressed by other issues. The specific parts of 10 CFR Part 60 addressed by this issue are as follows:

133(a)(1) General criteria for the underground facility.

> The orientation, geometry, layout, and depth of the underground facility, and the design of any engineered barriers that are part of the underground facility shall contribute to the containment and isolation of radionuclides.

133(b) Flexibility of design.

> The underground facility shall be designed with sufficient flexibility to allow adjustment where necessary to accommodate specific site conditions identified during in situ monitoring, testing, or excavation.

133(e)(2) Underground openings.

> Openings in the underground facility shall be designed to reduce the potential for deleterious movement or fracturing of overlying or surrounding rock.

133(f) Rock excavation.

> The design of the underground facility shall incorporate excavation methods that will limit the potential for creating a preferential pathway for groundwater to contact the waste packages or radionuclide migration to the accessible environment.

133(h) Engineered barriers.

> Engineered barriers shall be designed to assist the geologic setting in meeting the performance objectives for the period following permanent closure.

133(i) Thermal loads.

> The underground facility shall be designed so that the performance objectives will be met taking into account the predicted thermal

and thermomechanical response of the host rock, and surrounding strata, groundwater system.

The proposed strategy for resolution of this issue is presented in Section 8.3.2.2. In that section, the issue resolution strategy for Issue 1.11 is presented and interrelationships between Issue 1.11 and other issues are addressed. Readers unfamiliar with the issue resolution strategy for this issue should review the contents of Section 8.3.2.2 before continuing. In this section, the current status of resolution of the issue is reported.

Summary information describing the computer codes used in the analyses supporting the work completed is contained in Table 6-29.

The postclosure design criteria of 10 CFR 60.133 addressed by this issue require that the underground facility and engineered barrier system be designed to

1. Contribute to containment and isolation.

2. Assist the geologic setting in meeting performance objectives and limit the potential for deleterious rock movement or preferred pathways.

3. Account for the thermal and thermomechanical response of the host rock and the need for sufficient flexibility of design to accommodate site-specific conditions.

The underground facility, as referred to in this issue, includes the underground structure, drifts and emplacement boreholes including all materials used in construction of these openings. The underground structure includes the volume of rock adjacent to the excavation that sustains the load of the surrounding rock. Drift seals are part of the underground facility; however, they are explicitly addressed by Issue 1.12, and, thus, are considered to be an interface to this issue.

The postclosure design issue provides the mechanism for identification of repository design characteristics and configurations important to the resolution of Key Issue 1 (postclosure containment and isolation), quantification of how these characteristics and configurations are in compliance with 10 CFR 60.133, and incorporation of postclosure performance concerns into the design.

The characteristics and configurations important to containment and isolation on the scale of the whole underground facility are best understood by considering the information needs for this issue and their associated products. Listed in the following pages are the information needs and their associated products. The section of Chapter 8 is identified in which the information need is discussed in detail. In each instance, the name and number of the product is stated and the product is briefly described. The products are numbered in a manner that corresponds to the numbering used in Section 8.3.2.2. For example, product 1.11.1-2 is the second product identified under the first information need in Issue 1.11.

Table 6-29. Codes used to support work completed for Information Need 1.11.6 of Issue 1.11 (page 1 of 4)

Product number	Code name	Author	Ownership[a]	Design parameter	Analysis description
1	ADINAT	K.J. Bathe	MIT	Areal power density. Determine if thermal loading meets near- and far-field constraints.	A finite-element heat transfer program. 2-D, 3-D; for automatic dynamic nonlinear heat conduction; convective and adiabetic boundaries; constant or decaying heat source.
1	ADINA	K.J. Bathe	MIT	Areal power density. Determine if thermal-induced stresses meet near- and far-field constraints.	A finite element, stress analysis program, 2-D, 3-D; elastic, elastic/plastic ubiquitous joint model; accepts precalculated temperature history.
1	SPECTROM-41	D.K. Svalstad, RE/SPEC Albuquerque, NM	RE/SPEC	Areal power density. Determine if thermal loading meets near- and far-field constraints.	Finite element, heat transfer program. Nonlinear heat conduction; convection and adiabetic boundaries; constant or decaying heat sources.

Table 6-29. Codes used to support work completed for Information Need 1.11.6 of Issue 1.11 (page 2 of 4)

Product number	Code name	Author	Ownership[a]	Design parameter	Analysis description
1	SPECTROM-11	RE/SPEC Albuquerque	RE/SPEC No documentation available	Areal power density. Determine if thermal-induced stresses meet near- and far-field constraints.	Finite-element stress analysis program. Elastic, elastic/plastic ubiquitous joint model; accepts precalculated temperature history.
1	SPECTROM-349	D.K. Svalstad RE/SPEC Albuquerque, NM	RE/SPEC	Far-field temperature distribution, design of underground facility.	A linear superposition program. For three-dimensional heat conduction solutions for constant or decaying heat source from parallelopiped in a semi-infinite homogeneous medium.
1	ARRAYF	R.D. Klett	SNL	Borehole spacing strategy. Determine if thermal loading meets near-field constraints	A linear superposition program. For three dimensional heat conduction solutions for constant or decaying cylindrical heat source.

Table 6-29. Codes used to support work completed for Information Need 1.11.6 of Issue 1.11 (page 3 of 4)

Product number	Code name	Author	Ownership[a]	Design parameter	Analysis description
2	SIM	Ahmad Badie	Raymond Kaiser Engineering, Parsons Brinckerhoff, San Francisco, CA	Borehole spacing strategy. Determine if thermal loading meets near-field constraints	A linear superposition for three dimensional heat conduction solutions with constant or decaying line heat sources.
2	HEFF	B.H.G Brady, University of Minnesota	Public domain	Stresses around a borehole, emplacement drift, or repository	A boundary-element stress analysis program. 2-D, thermoelastic analysis of constant or decaying thermal load.
2	STRES3D	C. St. John M. Christianson University of Minnesota	Public domain	Stresses around a borehole, emplacement drift, or repository	Computer program for determining temperatures, stresses, and displacements around single or arrays of constant or decaying heat sources. Based on closed form solution.

Table 6-29. Codes used to support work completed for Information Need 1.11.6 of Issue 1.11 (page 4 of 4)

Product number	Code name	Author	Ownership[a]	Design parameter	Analysis description
2	DOT	Polivka, Wilson	University of California	Temperature distribution around a borehole, or emplacement drift.	A general purpose heat transfer program. Linear and nonlinear steady-state or transient-heat transfer. Input for VISCOT.
2	VISCOT	ONWI/OWID	Public domain	Stresses around a borehole, or emplacement drift.	Finite-element stress analysis or program. Thermovisco-elastic, thermo-viscoplastic; accepts precalculated temperature history.

[a]MIT = Massachussetts Institute of Technology; SNL = Sandia National Laboratories.

Information Need 1.11.1 Site characterization information needed for design (Section 8.3.2.2.1).

Number	Description
1.11.1-1	The data requirements list identifies the site data needed from site characterization to (1) support the postclosure design of the MGDS underground facility and (2) determine the contribution of the MGDS underground facility to containment and isolation.
1.11.1-2	The reference thermal/mechanical stratigraphy of Yucca Mountain is described.
1.11.1-3	The reference thermomechanical rock properties document will describe the conversion of measured rock properties data to reference rock properties for thermomechanical units other than the Topopah Spring Member. Reference rock properties will be recommended for incorporation in the NNWSI Project Reference Information Base (RIB) (Appendix Q of SNL, 1987).

Information Need 1.11.2 Characteristics of the waste package needed for design of the underground facility (Section 8.3.2.2.2).

Number	Description
1.11.2-1	The waste package characteristics for design of the underground facility will be obtained from Issue 1.10 (waste package characteristics--postclosure) and are recorded in the subsystems design requirements (SDR) document (Appendix P of SNL, 1987).

The characteristics of the waste package needed for design of the underground facility are identified in Section 8.3.2.2.2. The waste package characteristics used during the development of the conceptual design of the surface and subsurface facilities are presented in SCP-CDR Section 2.2, SCP-CDR Appendix G, and the SDR (Appendix P of SCP-CDR, SNL, 1987).

Information Need 1.11.3 Design concepts for orientation, geometry, layout, and depth of the underground facility that contribute to waste containment and isolation, including flexibility to accommodate site-specific conditions (Section 8.3.2.2.3).

Number	Description
1.11.3-1	The area-needed determination will establish the required area for the underground portion of the MGDS at the Yucca Mountain site.

Number	Description
1.11.3-2	The usable area and flexibility evaluation will (1) establish the boundaries of the area available for the MGDS underground facility at the Yucca Mountain site and (2) evaluate the flexibility of the site based on a comparison of the design for the MGDS underground facility layout and the area available for these facilities as determined by using a 3-D graphics model of the geologic structure of Yucca Mountain.
1.11.3-3	The vertical or horizontal emplacement orientation decision will document the decision and supporting logic for either (1) emplacing a single waste container in vertical boreholes in the floor of the drifts (current reference emplacement orientation) or (2) emplacing one or more waste containers in horizontal boreholes in the walls of the drifts.
1.11.3-4	The drainage and moisture control plan will present a plan for limiting the amount of water in contact with the containers to provide a favorable containment and isolation environment by promoting the migration of water away from the waste containers.
1.11.3-5	The criteria for contingency plan will provide (1) the criteria that can be used to identify underground emplacement areas that have geologic and hydrologic characteristics, conditions, or both within the ranges anticipated in licensing, and (2) criteria for modification of the MGDS underground facility baseline design based on the geologic characteristics encountered.

Information Need 1.11.4 Design constraints to limit water usage and potential chemical changes (Section 8.3.2.2.4).

Number	Description
1.11.4-1	The material inventory criteria will provide criteria for the inventory of materials (type and quantity) proposed for use in the subsurface facility.
1.11.4-2	The water usage criteria will establish criteria relating to the use of water during the construction and operation of the MGDS underground facilities.

Information Need 1.11.5 Design constraints to limit excavation-induced changes in rock mass permeability (Section 8.3.2.2.5).

Number	Description
1.11.5-1	The excavation methods criteria will (1) establish criteria for the allowable size and extent of the damage caused by excavation of the boreholes and drifts, (2) establish criteria for the allowable amount of alteration of in situ rock properties (e.g., permeability), and (3) establish criteria for excavation methods.
1.11.5-2	The long-term subsidence control strategy will provide design guidance to ensure that the design of the MGDS underground facilities will limit the potential for (1) subsidence and (2) creating preferential pathways for radionuclide migration.

Information Need 1.11.6 Repository thermal loading and predicted thermal and thermomechanical response of the host rock (Section 8.3.2.2.6).

Number	Description
1.11.6-1	The allowable areal power density chosen as a criteria on the layout of the MGDS underground facility and the logic supporting this choice will be documented.
1.11.6-2	The borehole spacing strategy will establish a plan for how to distribute the waste containers so that the allowable areal power density constraint and temperature criteria are met, considering the thermal characteristics of the waste.
1.11.6-3	The sensitivity studies will evaluate and document the effects of uncertainty in the description of the waste type and the geologic setting on the MGDS underground facility design and the thermal and thermomechanical response of the host rock.
1.11.6-4	The strategy for containment enhancement will document work done to evaluate alternative ways of distributing the waste containers so as to increase the number of containers that remain dry and the time the containers remain dry.
1.11.6-5	The reference calculations will (1) predict thermal and thermomechanical responses of the host rock on a container, drift, and far-field scale and (2) document the results of these calculations for use by other issues.

Information Need 1.11.7 Reference postclosure repository design (Section 8.3.2.2.7).

Number	Description
1.11.7-1	The reference postclosure design will document the reference design of the repository will form the basis for postclosure performance assessment of the MGDS facilities.

Number	Description
1.11.7-2	The documentation of compliance will document the compliance of the postclosure design with the requirements of 10 CFR 60.133.

The approach to resolution of Issue 1.11 (1) emphasizes ensuring that the postclosure waste disposal system element performs the functions identified in Section 8.3.2.2, and (2) includes developing a reference postclosure design of the repository.

The functions were derived directly from 10 CFR 60.133. The proposed strategy for resolution of Issue 1.11 used the three-step process identified in Section 8.3.2.2 (processes, performance measures, and performance goals and confidence) as a means of establishing that the functions are performed. The process step describes how the function will be accomplished. The performance measure step identifies the measure that will allow a determination of whether the process is being performed as required by the function. Each performance measure has an associated goal and confidence step. The goal is the value for the performance measure that will be adequate for the issue to be favorably resolved, and the current and needed confidence provides an indication of the importance assigned to meeting the goal.

The reference postclosure design will be discussed in future major design documents and will be documented in the Reference Information Base for use in future postclosure performance assessments.

Preliminary assessments and design concepts are contained in this chapter. Current goals are given in Chapter 8, Section 8.3.2.2. Updating of this information will continue through the advanced conceptual design and the license application design.

6.4.2.2 Work completed

Twenty products have been identified in Section 8.3.2.2 as important to the resolution of Issue 1.11 (see Section 6.4.2.1 for a complete listing of the products). Significant results have been reported to date for the following six products:

1.11.1-2 Reference thermal/mechanical stratigraphy

1.11.1-3 Reference thermomechanical rock properties document

1.11.3-1 Area-needed determination

1.11.3-2 Usable area and flexibility evaluation

1.11.6-1 Allowable areal power density

1.11.6-2 Borehole spacing strategy

The approach and data used and the results obtained for the six products are summarized in the following discussions.

1.11.1-2 Reference thermal/mechanical stratigraphy

The reference thermal/mechanical stratigraphy documents the three-dimensional thermal/mechanical stratigraphy of Yucca Mountain as contained in the Interactive Graphics Information System (IGIS) as the reference basis for design and performance assessment.

Analytical approach

The documented thermal/mechanical stratigraphy (Ortiz et al., 1985) provides a geometric representation of the rock units at Yucca Mountain. This representation, with associated material properties for each of the units, is being used as the reference stratigraphy in the design and performance assessment of the underground facility. This reference stratigraphy provides a consistent reference representation of the stratigraphy, thus addressing the use of conflicting stratigraphic descriptions through a change control process. The reference stratigraphy will be updated as information is provided from the site characterization. Final performance assessment will be based on the reference stratigraphy.

An IGIS was used to develop and display a three-dimensional model of Yucca Mountain. The model is a collection of smooth three-dimensional surfaces based on interpolation between sparse and irregularly spaced data points. Each of the smooth three-dimensional surfaces represents the base of a thermal/mechanical and hydrological reference unit. Faulting of the units is incorporated in the model. A complete description of the approach has been published (Ortiz et al., 1985).

Data

The reference thermal/mechanical stratigraphy is a compilation of site data, including surface mapping data and data obtained from boreholes drilled at the site. Current reference data and measured site data (raw data) used to establish the reference data are documented in Ortiz et al. (1985).

Specification of the accuracy of the model is difficult. However, within the primary area, the surfaces are sufficiently accurate to perform the needed conceptual design (see Section 8.3.2.2) for current accuracy, required accuracy, and additional site data needed). Confidence in the model inside the primary area is high. Additional data collected during site characterization will be used to improve the confidence in the geometric model outside the primary area.

Results

The thermal/mechanical stratigraphy (Ortiz et al., 1985) is in contrast to the previous geologic stratigraphy (Nimick and Williams, 1984) in that the geologic division of stratigraphic units does not lend itself readily to describing material properties. This is because a stratigraphic unit may contain more than one type of rock. Indeed, most stratigraphic units at Yucca Mountain include at least two different types of rock--welded ash-flow tuffs and bedded tuffs.

Field information and laboratory data were used in the development of the three-dimensional model. These data, concerning the nature and distribution of rock units at Yucca Mountain, are limited to surface geologic maps and drillhole logs. A method of analytically interpolating between sparse and irregularly spaced data source locations is used to generate a continuous analytical surface from a collection of three-dimensional coordinates.

Faulting effects are handled interactively on a case-by-case basis. The removal of fault movement from input data or its reinsertion into calculated surfaces has not been automated, because surface mapping of faults does not provide a comprehensive three-dimensional description of the area-wide fault system.

Referenceable products of the thermal and mechanical stratigraphy include cross sections (including faulting), isopach maps, contour maps, thickness or distance between features, and surface features (topography, outcropping, and faulting).

1.11.1-3 Reference thermomechanical rock properties document

This work will describe the conversion of measured rock properties data to reference rock properties for all thermomechanical units other than the Topopah Spring Member. Reference rock properties will be recommended for incorporation in the NNWSI Project RIB.

Analytical approach

Rocks are composed of crystals and grains in a fabric that frequently includes cracks and fissures. The selection of laboratory-sized specimens for testing excludes larger cracks and fractures that exist in the rock mass. Loads in the rock mass will be transmitted across these larger cracks and fractures. Laboratory strength tests on smaller specimens usually provide values greater than the actual strength of the rock mass. Thus, rock properties are dependent on sample size. Rock mass properties that are representative of a volume or mass of rock will be determined from measured data. The rock mass properties for the Topopah Spring Member are determined as part of preclosure analysis under Issue 4.4. A description of how the reference rock mass properties were determined from measured properties has been published with the recommended properties (Appendix O of SNL, 1987). Additional information also is given in Chapter 2.

Data

Rock properties are derived from laboratory measurements of thermal and mechanical rock properties. Current rock mass properties are given in

Section 6.1.2. (reference design data base). These data are derived from the site data given in Chapter 2, (Geoengineering).

Results

A consistent set of reference properties (physical, mechanical, and thermal properties and in situ conditions) for the thermal and mechanical stratigraphy at the Yucca Mountain site has been established. These reference properties have been derived from analyses of laboratory and field data (i.e., both intact rock data and rock mass data) currently existing for the site. These reference properties are contained in Section 6.1.2, in Chapter 2 and are included in the NNWSI Project Reference Information Base (Appendix Q of SNL, 1987). References for the sources of the laboratory and field data are cited in Chapter 2 and Section 6.1.2.

Analyses of the data have resulted in the derivation of methods to better understand and extrapolate both field and laboratory data. For example, a method for relating porosity to mechanical and strength data has been derived (Price and Bauer, 1985). Zimmerman et al.(1986a) show the relationship between laboratory and field determinations of the thermal, mechanical, and thermomechanical response of rock.

1.11.3-1 Area-needed determination

The area-needed determination will establish the required area for the underground facility at Yucca Mountain.

Analytical approach

Mansure (1985) gives a complete description of how the area needed was determined. Basically, the area for high-heat-producing waste has been determined by dividing the thermal output of the waste by the design basis areal power density (APD). The area for low-heat-producing waste was determined on the basis of operational and safety constraints. The area needed for the shops and other support facilities has been added to the areas needed for waste emplacement. The area needed is an important input into the usable area and flexibility evaluation discussed for the next product (1.11.3-2).

Data

The area-needed determination depends on the allowable APD. The APD, in turn, depends on site data. For the site data related to APD, refer to the discussion of that product in 1.11.6-1. The nonsite-related data required to determine the area needed include (1) the waste inventory, (2) the space for shops and support facilities, and (3) the size and spacing of the drifts.

Results

A preliminary determination of the area needed for a 70,000 metric tons uranium (MTU) underground facility has been completed. The results of this study have been used for (1) the planning of the site characterization program and (2) the preliminary evaluation of compliance with 10 CFR Part 960 as given in the environmental assessment (DOE, 1986c). Both of these uses of the area needed depend on comparing the area needed to the thickness and lateral extent of the host rock. The current value of the area needed is based on the layout of the underground facility presented in Section 6.2.

The current layout occupies 1,420 acres (Section 6.2). This is based on an inventory of 62,000 MTU of spent fuel and 8,000 MTU of defense high-level waste (DHLW) and West Valley high level waste (WVHLW), of an APD of 57 kW/acre. This is less than the area-needed given in the Environmental Assessment because the acreage reported there was for an all-spent-fuel repository. The uncertainty in the area-needed is judged to be ± 210 acres, based on uncertainty in the final basis APD of 40 to 80 kW/acre (Appendix M of SNL, 1987).

The analyses to determine the area-needed assume that the waste is emplaced at the equivalent energy density of the design basis APD. These analyses (Mansure, 1985) have resulted in two significant conclusions: (1) commingling (the placement of DHLW and WVHLW with spent fuel in the same emplacement panel) will not significantly (less than 10 percent difference) change the area needed, and (2) horizontal and vertical waste emplacement options do not require significantly different areas (less than 50 acres difference).

1.11.3-2 Usable area and flexibility evaluation

The usable area and flexibility evaluation will (1) establish the boundaries of the area available for the underground facility and (2) evaluate the flexibility of the site by comparing the layout with the area available for these facilities.

Analytical approach

The usable area and flexibility evaluation began with the selection of the primary and adjacent areas. Once these areas were selected, the preferred horizon for waste emplacement was chosen. A computer graphics model (CAD/CAM-like system) was used to display a three-dimensional picture of Yucca Mountain and compare underground facility location to constraints (required overburden, etc.). This approach is described in reports by Nimick and Williams (1984) and Mansure and Ortiz (1984).

Data

The data base used by the IGIS for this study was reported by Ortiz et al. (1985). The three-dimensional model of Yucca Mountain is based on geologic data from surface mapping of outcrops and faults and from unit contacts determined using core and cuttings taken from wells drilled at the site.

Results

Screening of the Nevada Research and Development Area of the Nevada Test Site and nearby areas for favorable locations for the permanent disposal of radioactive waste in a mined geologic disposal system (MGDS) resulted in the selection of Yucca Mountain as the primary area for location of the underground facility (Sinnock and Fernandez, 1982). Four geologic units at Yucca Mountain were compared--the Topopah Spring Member, the tuffaceous beds of Calico Hills, the Bullfrog Member, and the Tram Member. The portion of the Topopah Spring Member containing relatively few lithophysae was recommended (Johnstone et al., 1984). Subsequent evaluations of usable area and flexibility have been limited to the relatively low lithophysae portion of the Topopah Spring Member. Although the preferred horizon is expected to have low lithophysae content, this does not imply that the underground facility must be placed in low lithophysae host rock but only that host rock with lower lithophysae content may be preferable (Section 6.3.3.2.3 of DOE, 1986c).

Analysis (Mansure and Ortiz, 1984) of the output from a three-dimensional computer graphics model of Yucca Mountain prepared by Nimick and Williams (1984) indicates that Area 1, identified as the primary area and shown in Figure 6-87, contains approximately 2,200 acres. Approximately 1,850 acres of Area 1 are potentially usable on the basis of the disqualifying condition for erosion, which requires a 200-m overburden (DOE, 1986c).

Area 1 contains relatively few faults and rare fault breccias (Scott and Bonk, 1984). The surface and subsurface geologic exploration of Yucca Mountain has concentrated in this area and in the immediately surrounding area that has a relatively low fault density. Available site data indicate that rock with acceptable characteristics may be present within areas 2 through 6, and perhaps even outside these areas (Mansure and Ortiz, 1984; Sinnock and Fernandez, 1982).

If one considers only the primary area, the usable area (1,850 ± 140 acres) is more than the area-needed (1,420 ± 210 acres). Because of the irregularities of the shapes and the uncertainty in the size of the area needed and the area available, there is limited lateral flexibility. The other areas identified outside the primary area may contain over 5,000 acres (areas 2 to 6 on Figure 6-87); however, at this time, there are insufficient data to qualify most of these areas. Current understanding of design concepts and conclusions about offsets from site features suggest that there may be a need for as much as 300 additional acres to ensure adequate flexibility (Appendix M, SNL, 1987). Figure 6-88 shows two proposed expansions to the reduced primary area: (1) 2 EA and 2 EB and (2) SE. These areas could add at least an additional 750 acres. Note that the narrow, southern portion of the primary area, Area 1 (Figure 6-87), is not included in the revised, usable portion of the primary area shown in Figure 6-88 because it cannot be efficiently developed as part of the underground facility. The proposed site characterization program (Sections 8.3.2.2.1 and 8.3.2.2.3) includes plans to gather the data necessary to qualify these expansions.

Basic requirements for the thickness of the potential host rock are (1) the presence of sufficient overburden to ensure a low probability of uncovering the waste by erosion and (2) sufficient thickness of suitable host rock to provide the volume of rock required for construction of the underground facility. Mansure and Ortiz (1984) show that the approximate thickness of the preferred host rock is about 100 to 175 m within the primary area

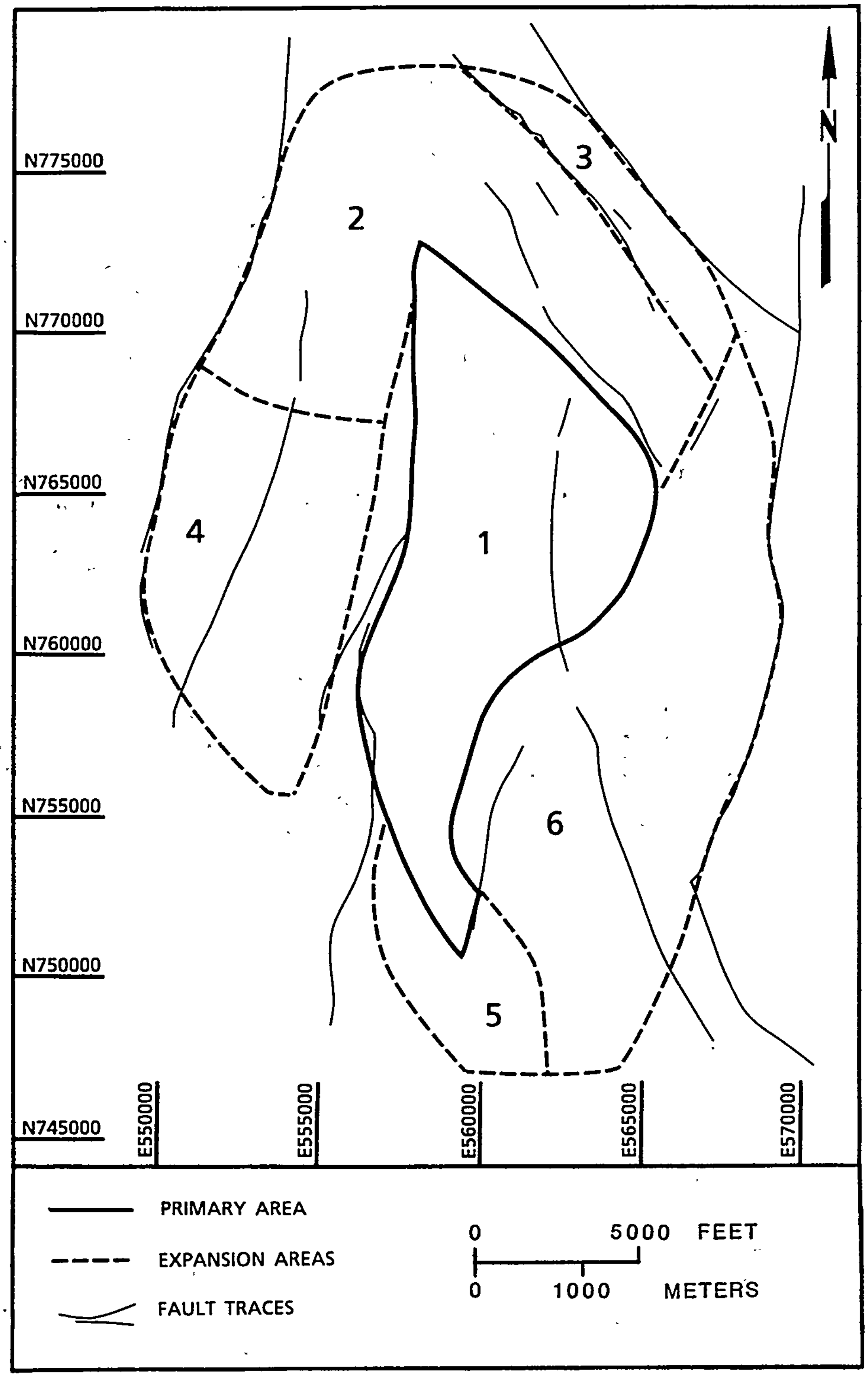


Figure 6-87. Primary area (area 1) for the underground repository and potential expansion areas (areas 2 through 6).

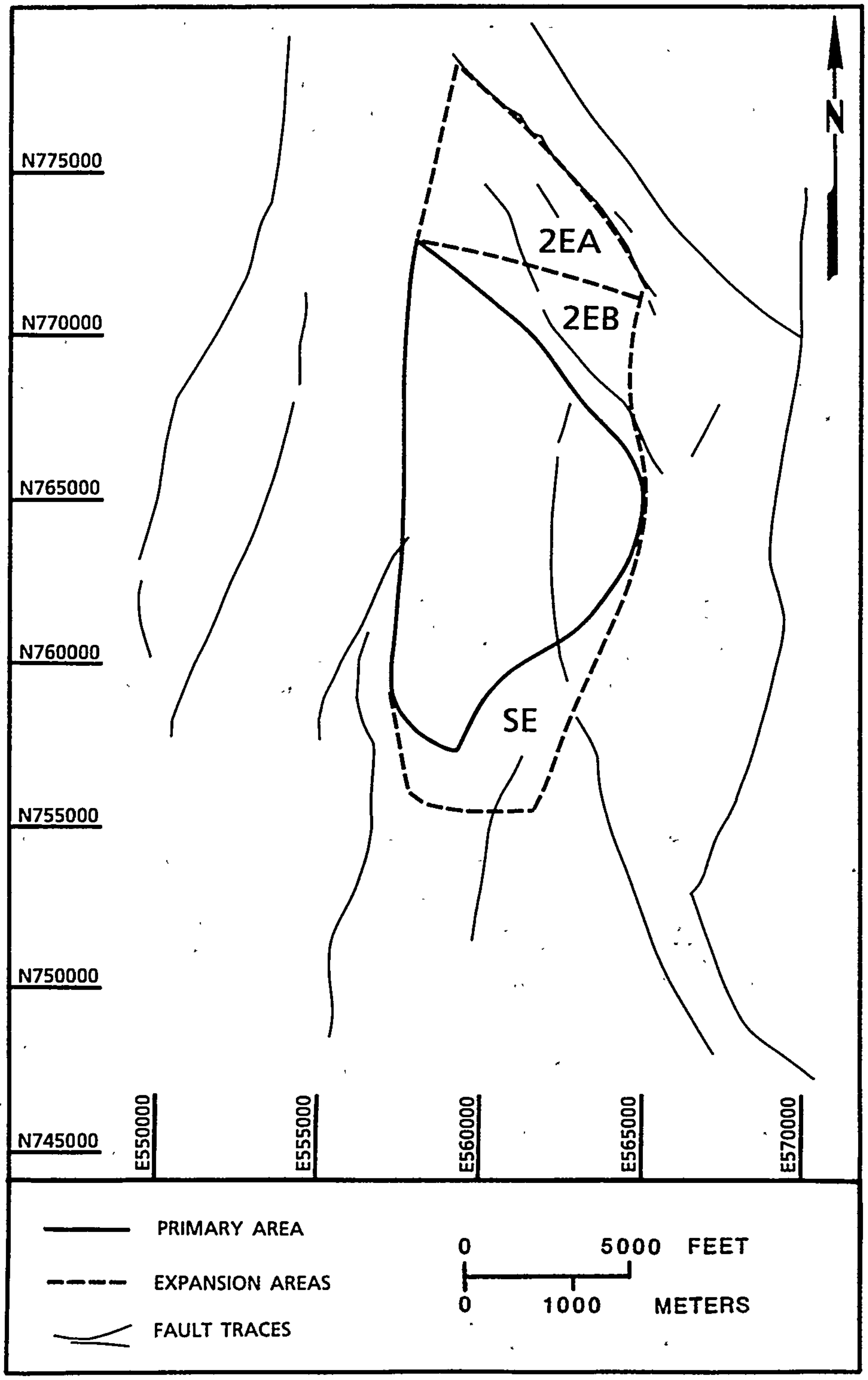


Figure 6-88. Revised usable portion of the primary area and expansion areas.

or about three times the thickness of the 45-m slab assumed for the underground facility envelope or more than 15 times the height of the drifts. The overburden at Yucca Mountain is more than 300 m thick over about 50 percent of the primary area and is over 200 m everywhere above the usable portion of the primary area. Thus, to date, exploration of the primary area has revealed sufficient thickness of potential host rock for flexibility in design of the underground facility.

1.11.6-1 Allowable areal power density

The APD (kW/acre) is a criterion placed on the design of the underground facility layout to ensure that the thermomechanical effects of the heat released by the waste meet performance allocation goals. This criterion is applied on a per panel basis (i.e., average output of a panel is divided by size of the panel).

Analytical approach

The approach used to determine the current design basis APD (Johnstone et al., 1984) was first to find the APD that resulted in a 100°C floor temperature and then to determine if that loading meets near- and far-field constraints. The determination of whether the loading meets near- and far-field constraints was done with the thermomechanical codes given in Table 6-29 (Section 6.4.2.1).

Data

Data used in this study include in situ initial conditions (temperature and stress), thermal and mechanical properties of the rock, the stratigraphy, and the thermal output of the waste. The initial in situ conditions and rock thermal and mechanical properties used in this study were reported by Tillerson and Nimick (1984). Stratigraphy used is given in Nimick and Williams (1984). The APD determined by the unit evaluation study (Johnstone et al., 1984) has been adopted as the current design basis. That study considered a range of parameter values sufficient to include the expected values of the site properties given in Section 6.1.2. The reference design data base, Section 6.1.2, contains values that are different from those used by Johnstone et al. (1984). The design basis APD will be revised before the advanced conceptual design and license application design to be consistent with the RIB data.

Results

The unit evaluation study (Johnstone et al., 1984) determined the value of 57 kW/acre for the thermal loading of the underground facility in the Topopah Spring Member. This value was computed using average waste age and burnup characteristics. This value has been the design basis used for developing the layouts reported in Section 6.2 and in Chapter 4 of the SCP-CDR. As noted in the analytical approach section previously, this loading was based on a maximum floor temperature for vertical emplacement of 100°C (Johnstone et al., 1984). This is a constraint that was assumed in the evaluation studies. Changes in ventilation system concepts have resulted in the issue resolution process replacing this constraint with a design goal of

50°C at 50 years (see Section 8.3.2.5, for an explanation of where and how the 50/50 constraint is applied). With the change to this constraint, the allowable thermal loading could increase above the current design basis of 57 kW/acre.

In establishing the allowable APD for the advanced conceptual design, a tradeoff study will be performed and an allowable APD higher than the current design basis may be adopted if it meets all criteria. Note that higher loadings are not necessarily detrimental to performance. A higher APD may make it possible to keep the waste containers dry for a longer period of time and, thus, assist in meeting the requirements of Issue 1.10, Waste package characteristics (postclosure). A higher APD also will reduce the area needed for the underground facility and, thus, increase lateral flexibility. On the other hand, a higher APD may have undesirable performance effects such as higher stresses and temperatures.

A procedure has been developed (Appendix G of SNL, 1987) based on the equivalent energy density concept (O'Brien and Shirley, 1984) to apply the design basis loading (57 kW/acre) to other than average waste ages and burnups. This procedure allows the design of the underground facility to accommodate the variability in thermal output of wastes of different ages and burnup rates.

1.11.6-2 Borehole spacing strategy

The borehole spacing strategy will establish a plan to distribute the waste containers such that the allowable APD constraint and temperature criteria are met, considering the thermal characteristics of the waste.

Analytical approach

The typical panel design, reported in Section 6.2 and in Chapter 4 of the SCP-CDR, distributes the waste to meet thermal constraints given in Section 8.3.2.2.6 (thermomechanical effects). This typical panel design is the first step in developing a borehole spacing strategy. The approach was to use actual drift and borehole dimensions to establish the standoff distances, which are used to control drift temperatures. Then, the borehole (and drift for vertical emplacement) spacings were varied to simulate emplacement of the waste at the design basis (57 kW/acre) loading, while limiting the extraction ratio to no more than 30 percent and meeting the temperature constraints (SNL, 1987; SCP Section 6.4.2). Various typical panel layouts were analyzed using the heat conduction code SIM (Table 6-29, Section 6.4.2.1). Professional judgment was used to pick the most practical panel layout.

Data

Data for these calculations were taken from reference properties (Appendix O of SNL, 1987). These data are consistent with the design data given in Section 6.1.2. Data required included rock thermal and mechanical properties and initial in situ conditions.

Results

The typical panel design presented in Section 6.2 and in the SCP-CDR is the first step in the borehole spacing study. Future work will determine practical development strategies based on sensitivity studies that consider waste type, age, and burnup. The current typical panel design contributes to the development strategies by demonstrating the design steps necessary to ensure that all criteria and design goals are met. That is, calculational procedures were developed to lay out a typical panel that met constraints. Thus, the typical panel demonstrates that reasonable waste distributions exist that meet all criteria and design goals. The typical panel presented includes commingling of waste types and should be sufficiently flexible to incorporate expected variations and uncertainty in the waste characteristics, inventory, and receipts.

The process developed results in a typical panel, reported in Section 6.2 and in the SCP-CDR, that does the following:

1. Uses the design basis APD.

2. Incorporates standoffs of the waste packages from drift walls to control drift wall temperatures.

3. Uses drift dimensions consistent with equipment, ventilation, mining, operation, and retrieval system requirements.

4. Meets borehole and rock mass thermal constraints.

Further, the design of the typical panel considers the goal (Section 8.3.2.2.6, thermomechanical effects) of enhancing containment of the waste by maintaining the temperature around the container above the boiling point of water for 300 yr. Thus, in determining the waste distribution, peak temperatures were held below constraints on the temperature of the waste package while consideration was given to maintaining the temperature of the rock surrounding the waste container above the boiling point of water for as long as possible.

In addition to the above six products where significant progress has been documented in published reports, progress has been made on understanding several of the other products.

1.11.1-1 Data requirements list

The current data requirements list is given in Section 8.3.2.2.1 (site characteristics needed for design). The list will be updated based on additional studies, changes to waste package characteristics, changes in design or design basis, or any changes to the goals of Issue 1.11.

1.11.2-1 Waste package characteristics for design of the underground facility

The current waste package characteristics for design of the underground facility are given in section 8.3.2.2.2. Those characteristics will be updated based on additional studies, any changes in the goals of Issue 1.10 (Section 8.3.4.2), or changes in design or design basis.

1.11.3-3 Vertical or horizontal emplacement orientation decision

The initial evaluations of horizontal and vertical emplacement are given in SCP-CDR (Appendix E of SNL, 1987). Potential discriminating factors in the decision process are containment, waste isolation, retrievability, worker radiologic safety, hydrologic character, and underground facility cost. Preliminary results based on normal operations indicate that the preclosure performance of the two emplacement options will be essentially the same except for worker radiologic safety and underground facility cost. The horizontal emplacement option appears to offer both lower worker exposure to penetrating radiation and lower underground facility costs. More equipment development and demonstration for retrievability would be required prior to license application for the horizontal emplacement configuration. However, both emplacement options are judged to perform in an acceptable manner; further investigations will be conducted to assess the effects of off-normal conditions.

1.11.3-4 Drainage and moisture control plan

A plan for drainage control has been incorporated in the design presented in Section 6.2 and in the SCP-CDR. This plan precedes both performance allocation and establishing goals on drainage. Future work on drainage will include establishment of performance and sealing requirements to limit the amount of water reaching emplacement areas during the postclosure period.

1.11.3-5 Criteria for contingency plan

Detailed work on the contingency plan will be part of the advanced conceptual design activities. However, some concepts on how to accommodate site-specific conditions have been incorporated in the current design. Specifically, the current design provides for a different ground support system to accommodate changes in underground conditions (SNL, 1987). During development, unexpected conditions like small zones of perched water, localized heavily fractured zones, water recharge pathways, or localized lithophysae-rich zones may be encountered. The contingency plan will provide the means to accommodate such regions.

Development of the contingency plan will begin with the assumption that all of the target area will be considered acceptable (except for possible local regions identified during the site characterization phase as being unacceptable). If local conditions are not consistent with performance objectives and regulatory requirements using the baseline design, then design modifications would be required. The procedures for implementing such design modifications would be licensed as part of the contingency plan, implemented by the performance confirmation program, and reported to and reviewed with the NRC. These procedures might include the following:

1. Continued development with design revisions like increased ground support and reduced thermal loading.

2. Skipping and isolating an area unfavorable for development.

Section 8.3.2.2.3, Underground facility orientation and layout, has a more detailed discussion of contingency plan concepts.

1.11.5-1 Excavation methods criteria

The excavation method design bases are (1) for emplacement drifts, conventional mining with controlled blasting techniques and controlled water usage, and (2) for boreholes, drill with vacuum chip removal and incorporating moisture for dust control only. These methods are documented in Section 6.2, Section 6.4.3 of the SCP-CDR, the subsystem design requirements document (Appendix P of SNL, 1987), and Section 8.3.2.2.5 (excavation methods for construction). These methods should result in excavations that meet performance goals (Section 8.3.2.2). The conventional mining of welded tuff has been demonstrated with blast control in G-Tunnel at the NTS (Section 7.3.2 of SNL, 1987) as discussed in Zimmerman and Findley (1987). Demonstration of the drilling equipment for the emplacement boreholes is currently planned for G-Tunnel (Section 8.3.2.5, preclosure design and technical feasibility).

1.11.5-2 Long-term subsidence control strategy

The long-term subsidence control strategy is to limit the extraction ratio and thermal load. The extraction ratio is to be less than 10 percent for horizontal emplacement and less than 30 percent for vertical emplacement (Section 8.3.2.2.5). Current design (Section 6.2; Section 6.4.2 of SNL, 1987) falls within this guideline for the emplacement areas. Stability at the design basis thermal loading of 57 kW/acre has been demonstrated analytically by thermomechanical calculations (St. John, 1987d). The calculations performed to assess long-term drift stability did not indicate any potential for appreciable subsidence. The design basis presented in Section 6.2 calls for backfill to be installed in all openings at the end of the retrievability period.

1.11.6-3 Sensitivity studies

The far-field unit evaluation study (Johnstone et al., 1984) considered the Topopah Spring Member, the tuffaceous beds of Calico Hills, the Bullfrog Member, and the Tram Member. Each unit is at a different depth; two are saturated and two are unsaturated, three are welded tuffs and one is a nonwelded tuff. This evaluation addressed a range of values for almost all parameters except horizontal to vertical in situ stress ratio. Results of the unit evaluation study did not show significant differences in the system response to the parameter variations. As a result, it is expected that the postclosure design will not be very sensitive to uncertainties in site data. Additional work will be needed to confirm this conclusion if the in situ stress ratio is substantially different from that assumed in the analysis done by Johnstone et al (1984). This conclusion will be evaluated in future studies (Section 8.3.2.2, configuration of underground facilities--postclosure).

1.11.6-4 Strategy for containment enhancement

Issue 1.10, waste package design--postclosure, (Section 8.3.4.2) incorporates a goal to enhance containment by keeping the container dry. Issue 1.11 has established a design goal of maintaining the majority of the waste containers at temperatures above the boiling point of water for 300 yr. Calculations of how long the containers remain above this temperature (Appendix K of SNL, 1987) have shown the (1) need for using the explicit geometry of waste distribution including the effects of the finite size of the underground facility (i.e., boundary effects) and (2) difference between local area power density and areal power density. The time duration that the containers remain above this temperature was shown to be very sensitive to thermal loading and position within the underground facility.

1.11.7-1 Reference postclosure repository design

The postclosure aspects of the conceptual design are summarized in Section 6.1.1.8 and given in detail in the SCP-CDR. This conceptual design will be used to evaluate MGDS performance and site characterization plans. The future versions of the Reference Information Base will contain reference postclosure repository design information and will be updated periodically using NNWSI Project change control procedures.

Work has not begun yet on four of the products identified in Section 8.3.2.2 for Issue 1.11. These are 1.11.4-1 (material inventory criteria); 1.11.4-2 (water usage criteria); 1.11.6-5 (reference calculations--for postclosure design); and 1.11.7-2 (documentation of compliance).

6.4.2.3 <u>Future work</u>

6.4.2.3.1 Analysis needs

The logic used in identifying the analyses required to resolve Issue 1.11 is presented in Section 8.3.2.2 and Sections 8.3.2.2.3 through 8.3.2.2.6 (information needs corresponding to issue resolution functions). The analyses are the approaches or methods described in those sections that will be used to calculate or otherwise establish that the anticipated or actual performance will meet the performance goals stated in Section 8.3.2.2. In general, the analyses are organized into activities to produce the products of the issue. For more information about these products and the analysis needs, see Sections 8.3.2.2.3 through 8.3.2.2.6.

6.4.2.3.2 Development needs

The reader is referred to Section 8.3.2.2 for discussions of future work on the products. In general, development needs do not exist for the products of this issue except the following:

1.11.3-4 Drainage-moisture control plan
1.11.3-5 Criteria for contingency plan
1.11.6-5 Reference calculations

6.4.2.3.3 Site information needs

The logic used in identifying the data important to resolution of Issue 1.11 is presented in Sections 8.3.2.2.3 through 8.3.2.2.6 (information needs corresponding to issue resolution functions). Section 8.3.2.2.1 (Information Need 1.11.1) lists in detail the site data needed to resolve this issue. It also defines how well the data need to be known to resolve the issue, and provides the link to the site characterization issues under which the test plans for obtaining the data will be discussed. The data required include rock properties data, summarized in Section 8.3.2.2, the geologic data necessary to develop the three-dimensional graphics model of Yucca Mountain, and the effects of mining on the rock mass. However, for the reasons noted previously in the Section 6.4.2.3.2, Development needs, additional data may be required to support establishing standoffs from unfavorable areas and to quantify the amount of water in contact with the container.

6.4.3 ISSUE 1.12: SEAL CHARACTERISTICS

6.4.3.1 <u>Introduction</u>

The question asked by Issue 1.12 is

> Have the characteristics and configurations of the shaft and borehole seals been adequately established to (a) show compliance with the postclosure design criteria of 10 CFR 60.134 and (b) provide information for the resolution of the performance issues?

The regulatory basis for Issue 1.12, the postclosure design criteria of 10 CFR 60.134, requires that

1. Seals for the shafts and boreholes shall be designed so that following permanent closure, they do not become permanent pathways that compromise the geologic repository's ability to meet the performance objectives for the period following closure.

2. Materials and placement methods for seals shall be selected to reduce, to the extent practicable, the potential (1) for creating a preferential pathway for ground water to contact the waste packages, or (2) for radionuclide migration through existing pathways.

A brief summary of the plan for assessing the performance of the seal system is described in Section 8.3.5.11. The complete discussion of the proposed strategy for resolution of this issue is presented in Section 8.3.3 (seal systems). Readers unfamiliar with the issue resolution strategy for this issue should review the contents of Section 8.3.3 before continuing. Issue 1.12 has been subdivided into four information needs and the completed work has been identified under the associated information need. The completed work is numbered in a manner that corresponds with the information need numbering system. For example, 1.12.4-1 is the completed work identified under the fourth information need of Issue 1.12. The information needs are as follows:

Information Need 1.12.1 Site, waste package, and underground facility information needed for design of seals and their placement methods.

This information need consists of the review and compilation of information associated with the site, the waste package, and the underground facility design. This work is ongoing and is not completed. The site characterization information needed for seal design is identified in Section 8.3.3.2.1.

Information Need 1.12.2 Materials and characteristics for seals for shafts, drifts, and boreholes.

This information need consists of the review, compilation, and development of information associated with the material and characteristics of seals for shafts, drifts, and boreholes. This work is ongoing. The site characterization information needed for seal design is identified in Section 8.3.3.2.2.

Information Need 1.12.3 Placement methods for seals for shafts, drifts, and boreholes.

The development of placement methods for seals has not begun; this information need is discussed in Section 8.3.3.2.3.

Information Need 1.12.4 Reference design of seals for shafts, drifts, and boreholes.

Number	Description
1.12.4-1	In the repository sealing concepts (hydrologic analysis 1), the concepts for sealing a nuclear waste repository in unsaturated tuff are presented. These concepts provide the basis for all future design activities in the NNWSI Project repository sealing program. As part of the development of these concepts, shaft and drift drainage calculations are performed.
1.12.4-2	In the modification of rock mass permeability in the zone surrounding a shaft (hydrologic analysis 2), analyses were performed to assess how the rock mass permeability around a vertical shaft excavated in a densely welded, highly fractured tuff might change. The modification of the rock mass permeability is due to the effects of stress distribution and blasting. From these analyses, a modified permeability zone model is presented.
1.12.4-3	In the hydrologic calculations to evaluate backfill of shafts and drifts (hydrologic analysis 3), hydrologic calculations were performed to assess the need for and extent of sealing the drifts and shafts.

1.12.4-4 In the numerical analysis to evaluate backfilling repository drifts (hydrologic analysis 4), hydrologic calculations evaluating the influence of backfilled drifts on flow through the surrounding unsaturated tuff matrix are presented.

1.12.4-5 In the vadose water flow around a backfilled drift (hydrologic analysis 5), the magnitude and direction of the ground-water flow in the vicinity of a vertically emplaced waste container were calculated. The waste container was situated in an unsaturated tuff environment.

Additional information on Information Need 1.12.4 is presented in Section 8.3.3.2.4.

Summary information describing the computer codes used in the analyses supporting the completed work just identified is contained in Table 6-30.

Summary of status of issue resolution

A brief summary of the status of issue resolution for sealing is presented below. This will aid in establishing a perspective for how the results of the individual analyses and evaluations completed to date (Section 6.4.3.2) contribute to the definition of planned future activities.

Sealing of a potential repository at Yucca Mountain involves emplacement of sealing elements in the shafts, ramps, boreholes, and underground facility. Design development of sealing elements first required establishing sealing concepts. Sealing concepts were developed with an understanding of the hydrogeology at the site and selected numerical calculations.

The following site conditions guided the concept development as well as the types of calculations to be performed.

1. The repository would be located in the unsaturated zone, 200 to 400 m above the water table.

2. The repository would be located in the Topopah Spring Member, which is a highly fractured, welded tuff unit.

3. Water flow at the repository horizon could occur within the matrix or discrete, water-producing zones.

The NNWSI Project repository sealing program is currently using these concepts to develop more specific sealing designs. Because specific seal designs have not been identified, it can be concluded that additional work is necessary to resolve this issue. Additional work will fall into the following categories:

1. Develop a complete design for the sealing subsystem. Design includes not only selection of the appropriate geometries for seals but also selection of materials and development of an appropriate emplacement strategy. Currently, the efforts within the NNWSI Project repository sealing program are focused on establishing the need for sealing and the appropriate design requirements. Both of

Table 6-30. Codes used for analyses addressing Issue 1.12

Code name	Author	Code location(s)	Design parameter	Analysis Description
TRUST	A. E. Riesenauer K. T. Key T. N. Narasimhan R. W. Nelson (Reisenauer et al., 1982) NUREG/CR-2360	Pacific Northwest Laboratory, Battelle Memorial Institute	Shaft drainage potential-performance evaluation of sealing components.	Finite difference; determines fluid flow past sealing elements in variably saturated porous media.
SAGUARO	R. R. Eaton D. K. Gartling D. E. Larson (Eaton et al., 1983) SAND 82-2772	Sandia National Laboratories	Shaft drainage potential-performance evaluation of sealing components.	Finite difference; determines fluid flow past sealing elements in variably saturated porous media.

these efforts will help to establish the suitable characteristics of sealing elements in shafts, ramps, boreholes, and the underground facility. These characteristics then will support the development of the configuration of sealing elements. First, selection of the appropriate design option will be made as part of the advanced conceptual design.

2. Document the hydrologic conditions encountered while excavating the exploratory shaft and the exploratory shaft facility. This information is necessary because the sealing concepts are based on the current understanding of the hydrologic conditions. This information will be obtained by participants involved with the testing in the exploratory shaft facility.

3. Assess the performance of select sealing designs to confirm acceptable performance and to arrive at preferred designs through design tradeoff studies. These tradeoff analyses will include performance, cost, and environmental concerns.

The resolution of this issue is believed (based upon preliminary evaluations completed to date) to be possible because of the following:

1. Conceptual designs have been defined as presented in Section 6.2 and in Chapter 5 of the SCP-CDR (SNL, 1987). These conceptual designs were developed assuming reasonably available technology would be used to construct these designs. Further, these designs form the basis for the site data needed.

2. Performance goals have been developed (Section 8.3.3) that can be used to evaluate the suitability of seal designs.

3. Preliminary calculations indicate that even if more water than anticipated is encountered at the repository horizon, these waters can effectively be isolated and drained through the repository drift floors.

However, before demonstrating the resolution of this issue, a total seal system design will have to be proposed and its performance evaluated. This includes the effects of environmental conditions on the performance of sealing materials. Currently, the design requirements are being developed and potentially suitable materials are being selected. The information obtained from this effort will support development of the ACD.

6.4.3.2 Work completed

Significant results have been reported to date for the work under Information Need 1.12.4. The approach used, data utilized, and the results obtained are summarized in the following section.

1.12.4-1 Repository sealing concepts (hydrologic analysis 1)

Analytical approach

Calculations evaluating shaft and drift drainage were presented in Fernandez and Freshley (1984). The purpose of these calculations was to determine if water entering the shafts or drifts can be drained at those locations. The possibility of using a shaft for drainage below the underground facilities was investigated using methods proposed by the U.S. Bureau of Reclamation (USBR) for boreholes and was summarized and critiqued by Stephens and Neuman (1982a,b). Two of the steady-state methods summarized by Stephens and Neuman (i.e. the methods by Glover and by Nasberg-Terletskata) were used directly in evaluating the shaft drainage potential. Flow into a drift floor was evaluated to determine the extent of the floor used to dissipate water from a discrete fault or fracture zone. Flow was computed using the equation for parallel plate analogy for flow in fractures.

Data

Drainage through a highly fractured, welded tuff was computed in this analysis. Two locations of welded tuff were considered: drift floor and the base of shafts. To compute drainage, the effective hydraulic conductivity was required. Values for fracture aperture width, hydraulic conductivity of fractures, and fracture frequency were used to compute the effective hydraulic conductivity.

From Zimmerman and Vollendorf (1982), two sets of values were selected for hydraulic conductivity and aperture width. One set represented the lowest measured hydraulic conductivity with an associated aperture width. The second set represented an arithmetic mean of 12 hydraulic conductivities with an associated mean aperture width. A conservatively low value for fracture frequency was assumed based on information discussed by Scott et al. (1983). Fracture frequencies presented by Scott et al. (1983) represented fracture frequencies of cores and outcroppings of densely welded tuff in the vicinity of Yucca Mountain.

Results

The performance of two sealing system elements (Section 8.3.3.2) was evaluated in this hydrologic analysis. The sealing system elements evaluated were the unsaturated Topopah Spring Member (TSw2) at the base of shafts (shaft drainage analysis following) and the drift floor within the Topopah Spring Member to accommodate net flow from faults (drift drainage analysis below).

From the results of the shaft drainage analysis, it was concluded that an estimated inflow of approximately 100 to 150 m^3/yr can be effectively drained through the bottom of the shaft, even when considering conservative values of fracture spacing and permeability. A geologic unit having bulk rock hydraulic conductivity of 5 x 10^{-6} cm/s could potentially drain 150 m^3/yr for a 8-m (22-ft)-diameter shaft and 120 m^3/yr for a 4-m-diameter shaft. Both conditions assume a modest buildup of water (i.e., about 15 m) at the base of a shaft. Because of the conservative nature of the calculations presented for drainage at the base of shafts (Fernandez and

Freshley, 1984) and because the expected value for bulk rock hydraulic conductivity of welded tuff is likely to be higher than 5 x 10^{-6} cm/s, yearly inflows of water (100 to 150 m^3/yr) into shafts are expected to drain through the bottom of shafts.

From the drift drainage calculations, it was concluded that considering the possible effects of fracture permeability, an inflow of 2.8 m^3/wk into a drift can be drained through a 6-m length of drift floor.

The ability to achieve the performance goals for the underground facility will depend on the frequency of water occurrences in the underground facility and the design options available to reduce or control water flow into the waste disposal rooms. Tradeoff analyses performed as part of advanced conceptual design and license application design will be performed to select the preferred design option.

1.12.4-2 Modification of rock mass permeability in the zone surrounding a shaft (hydrologic analysis 2)

Analytical approach

An analysis was performed to determine the modification in rock mass permeability resulting from stress redistribution and blast damage around a vertical shaft excavated through fractured, welded tuff (Case and Kelsall, 1987). To assess the permeability changes due to stress redistribution, elastic and elastoplastic stress analyses were performed to estimate the stress distributions after excavations for a wide range of rock properties and in situ stress conditions. Changes in stress are related to changes in rock mass permeability using stress-permeability relations for fractures obtained from laboratory and field testing. Coupling the information from the stress analysis with the relationship established between stress and permeability, the permeability enhancement due to stress redistribution was calculated.

The second half of this analysis involved performing an assessment of the increased permeability due to blast damage adjacent to the wall of the shaft. Both case histories and theoretical relationships between explosive charge weight and the particle velocity required to produce fracturing were evaluated to determine the potential extent of damage. This assessment of blast damage together with the analysis of stress redistribution effects on permeability were combined to develop the modified permeability zone model.

Data

The purpose of this analysis was to determine the changes in permeability around a shaft excavated in fractured, welded tuff. These changes were caused by stress redistribution in the area surrounding the excavation and by damage to this area due to blasting. The result of this analysis was the development of a modified permeability zone model. The data used to develop the model included the following:

1. Compressive strength of welded tuff (Price, 1983).

2. Tensile strength of welded tuff (Nimick et al., 1984).

3. Rock mass rating values (Langkopf and Gnirk, 1986).

4. Laboratory investigations of the influence of effective confining stress on fracture permeability (Peters et al., 1984).

5. Field data associated with the G-tunnel heated block test, specifically, permeability versus effective normal stress from a single fracture (Zimmerman et al., 1985).

6. Theoretical relationship between the charge weight and particle velocity required to produce fracturing (Holmberg and Persson, 1979).

Results

The modified permeability zone model was developed so that the performance of the shafts and drifts, as excavated, could be evaluated. This model then could be used in addressing the need for sealing. If it is determined that sealing is needed or desired, this model could be used in developing specified designs to achieve a desired performance.

The assumptions and data used in this analysis were varied to address potentially varying field conditions. Because of this variation in input parameters, multiple results were obtained. Two models were developed: one for a 100-m depth location in welded tuff and the second for a 310-m depth location in welded tuff. Expected conditions and upper bound changes also were evaluated. Finally, three conditions concerning blast damage were evaluated: no blast damage, a 0.5-m blast damage zone, and a 1-m blast damage zone from the shaft wall.

To compare the relative changes in rock mass permeability, the permeability was averaged over an annulus 1 radius wide around a 4.4-m (14.5-ft) diameter shaft. By performing this averaging, it was shown that permeability changes could range from 15 to 80 times the undisturbed rock mass averaged over an annulus 1 radius wide from the shaft wall. This model will be used in future analyses to determine the performance of the overall sealing system.

1.12.4-3 Hydrologic calculations to evaluate backfill of shafts and drifts (hydrologic analysis 3)

Analytical approach

Hydrologic calculations were performed to assess the need for and extent of sealing shafts and drifts. Two geometries were evaluated using the computer code TRUST (Reisenauer et al., 1982):

1. A drift with vertical emplacement of waste packages to determine water flow near the waste package and through the drift.

2. A shaft penetrating a slightly inclined contact between welded and nonwelded tuff units to determine the water flow into the shaft.

TRUST (Reisenauer et al., 1982), an integrated, finite difference code for unsaturated ground-water flow, was used for the drift and shaft analyses. Indi-vidual subanalyses were evaluated for each geometry. For the drift geometry, four subanalyses were performed. All subanalyses assumed that a drift was located in a welded tuff unit. The drift backfill, either clay or sand, and the saturated permeability values for the welded tuff unit were varied. Five subanalyses were evaluated for the shaft analysis. All drift subanalyses assumed that the shaft penetrated an inclined, welded-nonwelded tuff contact. It was assumed that the shaft was backfilled with either a clay or a sand material. The relative positions of the nonwelded and welded tuff units were varied together with the saturated permeability values. Additional details of the subanalyses evaluated are given in Freshley et al. (1985a) and in Fernandez and Freshley (1984).

Data

As indicated under the approach section for this analysis, water flow under unsaturated conditions was evaluated for two geometries. The first geometry involved a drift located in an unsaturated, welded tuff. The second geometry involved a shaft penetrating unsaturated welded and nonwelded units. Both of these geometries assumed that the drift or shaft was backfilled with a sand or clay. To assess water flow in the vicinity of the shaft or drift, it was necessary to obtain hydrologic properties of the materials used in the analysis. The permeability versus pressure-head relationships for sand, clay, and the welded and nonwelded tuff units were of primary concern. Knowledge of porosity of the materials also was required. The hydrologic properties and porosity of sand (Crab Creek sand) and clay (Chino clay) were taken from Mualem (1976). The hydraulic properties of selected welded and nonwelded units were determined from core taken from well USW GU-3. Under some instances, the permeability versus pressure-head curves were scaled up or down to provide a broad range of input parameters. All the data used in this analysis are included in Freshley et al. (1985a). The only other datum used in this analysis was the assumed input flux of 0.4 cm/yr.

Results

This analysis addressed the performance of two sealing system elements, shaft fill and drift backfill. The function of the shaft fill and drift backfill is to reduce the amount of water entering the waste disposal rooms. The analysis goes further to determine if the type of backfill can significantly influence the flow past waste packages for the drift backfill portion of the analysis. The conclusions given below are taken from Fernandez and Freshley (1984) and Freshley et al. (1985a).

The following conclusions were derived from the drift analysis:

1. From a hydrologic perspective, backfilling of the repository drifts is not essential. This conclusion is based on the observation that varying the backfill in drifts does not significantly influence the flow rates in the vicinity of the waste packages.

2. Water flow past horizontally emplaced waste packages cannot be altered by varying drift backfill. The standoff zone (i.e., the zone between the drift and the first waste package) is sufficiently

large to negate the effect backfill materials have on ground-water flow past the waste packages.

3. If backfilling is necessary, coarse, rather than fine, materials are more satisfactory because of their capacity to drain and act as a capillary barrier.

4. Greater flow of water into drifts may occur when saturation in the surrounding rock formation is high (98 to 99 percent). However, this level of saturation is unlikely to occur in the horizon being considered for the repository (Topopah Spring Member).

The conclusions from the shaft analyses were

1. From a hydrologic viewpoint, assuming porous matrix flow, backfilling the shafts is not essential. This conclusion is based on the prediction that the amount of water entering the shafts will be insignificant.

2. If backfilling is required for other reasons, the shaft should be filled with a material that behaves hydrologically like a sand.

The conclusions from the drift analysis stated above suggest that where free-flowing water from discrete, water-producing zones is not encountered, backfilling is not essential. Therefore, if sealing is required in the underground facility, emphasis will be placed on controlling water that enters the underground facility. In the current conceptual design, drainage paths exist from the emplacement drifts to the access drifts, then into the mains and finally to the base of the emplacement exhaust shaft. The ability to achieve the performance goals established for the underground facility will be evaluated, considering the alternative sealing components that could be emplaced in the underground facility.

Based on the shaft backfilling evaluation, assuming porous matrix flow, it was concluded that backfilling is not essential for hydrologic reasons, however, for safety reasons, backfilling of shafts will occur. In determining how the performance goals for shaft sealing components can be met, shaft backfill, the modified permeability zone, and other shaft sealing components will be evaluated.

1.12.4-4 Numerical analysis to evaluate backfilling repository drifts (hydrologic analysis 4)

Analytical approach

Additional hydrologic calculations evaluating the influence of backfilled repository drifts in a welded tuff unit were performed following the completion of the hydrologic calculations described under: hydrologic analysis 3. The TRUST code (Reisenauer et al., 1982) was used in this analysis. Both fine- and coarse-grained materials were assumed as the backfill material in the drift. The primary difference between this calculation and the hydrologic analysis 3 was the selection of a different permeability versus pressure-head relationship for the welded tuff unit. Details are given in Freshley et al.(1985b).

Data

This analysis evaluated the water flow, under unsaturated conditions, in the vicinity of a drift that was backfilled with a sand or clay material. This analysis (Freshley et al., 1985b) differed from that presented in Freshley et al. (1985a), by the selection of the hydraulic properties of the welded tuff unit modeled and by the flux imposed at the upper boundary. This analysis used data on the hydraulic properties of welded tuff that were considered more representative of the Topopah Spring Member than that used by other analyses. The data used are included in Peters et al. (1984). The moisture retention characteristics and unsaturated permeabilities from sample G4-6 were used in this analysis. Data obtained from sample G4-6 were used because the porosity and permeability of sample G4-6 are lower than for sample S-19 presented by Freshley et al. (1985a), and are more representative of the prospective host rock than data from other samples. Further, sample G4-6 is a densely welded tuff from the Topopah Spring Member. The flux used in this analysis was 0.01 cm/yr.

Results

This analysis evaluated the role of drift backfill on flow past a vertically emplaced waste package. Two moisture retention characteristic curves for the host rock formation were selected to perform the analysis. This analysis differs from hydrologic analysis 3 in that a second sample characteristic curve (sample G4-6) was input into the model. This second characteristic curve was more representative of the host rock formation. The conclusions from this analysis were similar to the conclusions presented in the hydrologic analysis 3. Thus, the conclusions from hydrologic analysis 3 were substantiated.

1.12.4-5 Vadose water flow around a backfilled drift (hydrologic analysis 5)

Analytical approach

Hydrologic analyses (Mondy et al., 1985), similar to those described in hydrologic analyses 3 and 4, were performed using the computer code SAGUARO (Eaton et al., 1983). SAGUARO is a finite difference code developed to model the flow of vadose water. In this analysis, the magnitude and direction of flow were determined in the vicinity of a vertically emplaced waste package below a drift backfilled with various materials. Sand and clay, representing the potential backfilled materials, were selected because of their significantly different hydrologic properties.

Data

The data used in this analysis were identical to the data used to perform hydrologic analysis 3. The hydrologic properties and porosity for sand and clay were taken from Mualem (1976). The hydrologic properties of the welded tuff unit were taken from preliminary hydrologic analyses of selected welded tuff samples from USW G-3.

Results

The sealing system element evaluated in this analysis was the drift backfill associated with the underground facility. The purpose of this analysis was to evaluate the flow past vertically emplaced waste packages using the same geometry as used in hydrologic analysis 3. However, this analysis was performed using a different computer code than that used in analysis 3 (TRUST) (Reisenauer et al., 1982). In this sense, corroboration of the results obtained in hydrologic analysis 3 would be possible. The following conclusions are based on the study in Mondy et al. (1985).

1. With the drift simulated as being backfilled with clay, the predicted water flow past a waste package is not significantly different from that predicted in areas relatively far removed from the drifts. Similarly, the analysis in which the drift was simulated as being backfilled with sand predicted that the flow near the waste package would be reduced by only 10 percent compared to that predicted when simulating clay-filled drifts. Hence, the water flow past a vertically emplaced waste package is not very sensitive to the hydrologic properties of the backfill material for the conditions simulated in this preliminary analysis.

2. The vertical water flow is diverted only one to two drift widths to the side of the drift by the drift backfill. This limited diversion implies that the drift backfill would not influence flow past a horizontally emplaced waste package if a stand-off distance of more than two drift widths is included in the design.

The conclusions from this analysis suggest, as did hydrologic analyses 3 and 4, that if sealing is required in the underground facility, emphasis will be placed on designing sealing components that will control water from discrete, water-producing zones.

6.4.3.3 Future work

6.4.3.3.1 Analysis needs

Because of the structure of this report, the analysis needs are discussed here and in Sections 8.3.3.1.3 (seal design), 8.3.3.1.4 (seal modeling), 8.3.3.2.1 (information needed for seal design under Information Need 1.12.1) and 8.3.5.11 (plans to assess seal system performance). The intent of this section is to summarize these identified sections.

The strategy used in developing seal designs is (1) to establish the need for seals through the use of analytical solutions describing unsaturated and saturated flow and (2) to account for the thermal effects of waste emplacement on the environmental conditions expected in the underground facilities, shafts, and ramps. Depending on the extent of the data base, sensitivity studies will be performed to establish a broad range of responses. Analyses will be performed on sealing elements to determine if the assigned performance goals can be achieved. This analytical effort may include the use of simple analytical solutions or complex computer codes.

6.4.3.3.2 Developmental needs

Numerical codes may be used to assess performance of sealing components, and can be used to more accurately define response of sealing components or subsystems. Responses can include hydrologic and thermomechanical behavior.

No new flow codes will be developed as part of the sealing program. Only existing codes will be used to assess the performance of sealing designs. Validation of the codes will be performed as part of the testing program associated with the exploratory shaft facility, and the strategy for this validation is described in the discussion of the ground-water travel time issue (Issue 1.6, Section 8.3.5.12).

6.4.3.3.3 Site information needs

This section presents the site characterization parameter needs to resolve Issue 1.12 (seal characteristics). The information needed to confirm design assumptions is included in tables in Section 8.3.3.2.1. Information such as the saturated hydraulic conductivity, gravitational analyses, compressibility of shaft fill, borehole construction, and geologic logs associated with specific boreholes, will support the design process in the selection of the appropriate methods to emplace sealing components. Site information needed to validate analytical methods may include hydrologic characterization of the Topopah Spring Member (TSw2). Specific properties are unsaturated matrix properties and the drainage capacity of the TSw2 unit. The prevalence of water-producing zones, if any, and the hydrologic nature of the Ghost Dance fault, the area underlying Drill Hole Wash, and the rock matrix will all be important site information needs in selecting the most appropriate sealing designs.

6.4.4 ISSUE 2.1 PUBLIC RADIOLOGICAL EXPOSURES--NORMAL CONDITIONS

6.4.4.1 Introduction

The question asked by Issue 2.1 is

> During repository operation and closure, (a) will the expected average radiation dose to members of the public within any highly populated area be less than a small fraction of the allowable limits, and (b) will the expected radiation dose received by any member of the public in an unrestricted area be less than the allowable limits, as required by 10 CFR 60.111, 40 CFR Part 191 Subpart A, and 10 CFR Part 20?

The complete discussion of the proposed strategies for resolution of this issue is presented in Section 8.3.5.3 (public radiological exposures--normal conditions). Readers unfamiliar with the issue resolution strategy for this issue, should review the contents of Section 8.3.5.3 before continuing.

Issue 2.1 has been subdivided into three information needs, and the completed work has been identified under the associated information need. The completed work is numbered in a manner that corresponds with the information need numbering system. For example, Section 2.1.2-1 is the first completed work identified under the second information need of Issue 2.1. Work that is similar to that required by this issue is discussed in Section 6.4.5 (radiological safety of workers--normal conditions) and in Section 6.4.6 (accidental radiological releases).

The Issue 2.1 information needs are as follows:

Information Need 2.1.1 Site and design information needed to assess preclosure radiological safety.

This information need consists of the review and compilation of information associated with the site; waste forms; surface and subsurface facility design; waste receiving, preparation, storage, and emplacement procedures; waste retrieval, storage, preparation, and shipping procedures; site-generated waste handling, preparation, and shipping procedures, repository caretaking procedures; and repository closure procedures. This work is ongoing. The site characterization information needed for resolution of this issue is identified in Section 8.3.5.3.1.

Information Need 2.1.2 Determination of projected releases of radioactive material from the repository to restricted and unrestricted areas under normal conditions.

Number	Description
2.1.2-1	The radioactive releases during normal operations for the surface facilities under normal operating conditions were addressed and reported in Section 6.1 of the SCP-CDR.

Information Need 2.1.3 Determination that public radiation exposure resulting from the release of radioactive material from the repository combined with exposures from offsite installations and operations meets applicable requirements.

Methodology similar to that used to obtain projections of public radiation exposure, resulting from the release of radioactive material under accident conditions, Section 6.4.9, will be used to forecast public exposures under normal conditions.

6.4.4.2 Work completed

The following work has been completed for Issue 2.1.

2.1.2-1 Radioactive releases during normal operations

Section 6.1 of the SCP-CDR, Radioactive releases during normal operations, addresses the releases that are expected to occur as a result of the waste receiving, preparation, storage, and emplacement activities. Releases of naturally occurring radiation (e.g., radon-222 and radon daughters released as a result of mining activities or released from the mined materials stored on the surface) and radiation releases from sources other than the site (e.g., radiation releases from the Nevada Test Site) have not been addressed by Section 6.1 of the SCP-CDR. This section is separated into four subsections: (1) liquid effluents, (2) solid wastes, (3) gaseous secondary wastes, and (4) site monitoring. In the first three subsections, the design concepts and approaches for collecting, monitoring, treating, and disposing of liquid, solid, and gaseous wastes are discussed. These discussions include identification of the sources, types, quantities, method of treatment, and method of disposal of these wastes. The releases of radioactive materials from the repository to the restricted and unrestricted areas under normal conditions are estimated. The fourth subsection discusses the requirements and philosophy of the site monitoring program. The site monitoring program will ensure that radioactive releases to the restricted and unrestricted areas under normal conditions are within the limits established in the regulations addressed by this issue.

6.4.4.3 Future work

Preliminary investigations are planned as part of the advanced conceptual design activities and more detailed analyses are planned as part of the license application design and licensing activities for the mined geologic disposal system.

Some of the site characteristics which have been identified in Section 8.3.5.3 as site information that should be obtained by the site characterization program are:

1. Meteorology of the Yucca Mountain site and adjacent areas.

2. Radon-222 and radon daughter emanation rate from the host rock at ambient and elevated temperatures.

6.4.5 ISSUE 2.2 WORKER RADIOLOGICAL SAFETY--NORMAL CONDITIONS

6.4.5.1 Introduction

The question asked by Issue 2.2 is

> Can the repository be designed, constructed, operated, closed and decommissioned in a manner that ensures the radiological safety of workers under normal operations, as required by 10 CFR 60.111 and 10 CFR Part 20?

The regulatory requirement addressed by this issue is 10 CFR 60.111(a). The wording of 10 CFR 60.111(a) invokes 10 CFR Part 20.

The performance objective stated in 10 CFR 60.111(a) (performance of the geologic repository operations area through permanent closure) is as follows:

> Protection against radiation exposure and releases of radioactive material. The geologic repository operations area shall be designed so that until permanent closure has been completed, radiation exposure and radiation levels, and releases of radioactive materials to the unrestricted areas, will at all times be maintained within the limits specified in Part 20 of this chapter and such generally applicable environmental standards for radioactivity as may have been established by the Environmental Protection Agency.

The complete discussion of the proposed strategies for resolution of this issue is presented in Section 8.3.5.4 (radiological safety of workers--normal conditions). Readers unfamiliar with the issue resolution strategy for this issue, should review the contents of Section 8.3.5.4 before continuing.

Issue 2.2 has been subdivided into two information needs, and the completed work has been identified under the associated information need. The completed work is numbered in a manner that corresponds to the information need numbering system. For example, 2.2.2-1 is the first completed work identified under the second information need of Issue 2.2. The information needs are as follows:

Information Need 2.2.1 Determination of radiation environment in surface and subsurface facilities due to natural radioactivity.

As part of the site characterization program, the natural radioactivity of the site will be characterized. The natural radioactivity of the site will be used in the determination of the expected annual and repository lifetime exposures of workers to natural radioactivity. No work on this information need, other than the identification of the required site characteristics, has been completed. This work, identification of site characteristics needed for determination of natural radiation environments, is described in Section 8.3.5.4.1.

Information Need 2.2.2 Determination that projected worker exposures and exposure conditions meet applicable requirements.

Number	Description
2.2.2-1	Worker exposures under normal operating conditions have been estimated and these estimates have been used in both the design and evaluation of the repository facilities.

6.4.5.2 Work completed

The completed work for Issue 2.3 is described in the following section.

2.2.2-1 Worker exposures under normal conditions

Two investigations have been conducted to forecast the expected exposures of workers to penetrating radiation during repository operations under normal operating conditions. The results of these investigations are reported in Dennis et al. (1984a) and in Stinebaugh and Frostenson (1987).

The reports by Dennis et al. (1984a) and Stinebaugh and Frostenson (1987) were prepared for use by the repository architect-engineer in the conceptual design of the waste-handling facilities and equipment. These reports list the repository operations and the estimated worker radiation exposures. All forecast annual exposures were below the 5 rem/yr permissible dose equivalent limit. However, eight worker positions were identified where the forecast exposures exceed the 1 rem/yr design objective (DOE, 1986b, Chapter 11). future design efforts will focus on reducing the exposure at these eight positions, as well as reducing general worker exposure to levels as low as reasonably achievable.

Stinebaugh and Frostenson (1987) was prepared after Dennis et al. (1984a) and it addresses the expected worker exposure under current SCP-CDR (Chapter 6-8 of SNL, 1987) expected conditions during the emplacement and retrieval of spent fuel when the vertical emplacement mode is used. Emplacement and retrieval operations and the estimated worker radiation exposures are listed. All worker exposures were found to be below the 5 rem/yr exposure limit. Only one worker position exceeded the DOE design objective of 1 rem/yr. Future design efforts will focus on reduction of exposure at this position, as well as reduction of general worker exposure to levels as low as reasonably achievable.

6.4.5.3 Future Work

6.4.5.3.1 Analysis Needs

Worker exposures resulting from the natural radioactivity of the host rock will be investigated as site information becomes available. During each subsequent design phase (advanced conceptual design, license application design, and final procurement and construction design), the expected exposure of workers under normal conditions will be forecast. The forecast will become more detailed as the supporting design, waste characterization, and site information become more detailed.

6.4.5.3.2 Site data needs

As discussed in Section 8.3.5.4 (radiological safety of workers--normal conditions), certain site data are needed to determine the radiation environment in the surface and subsurface facilities as a result of natural radioactivity. The main contribution to worker exposure from natural radioactivity is due to radon-222 and its daughter isotopes. There are other contributions from other naturally occurring radionuclides; however, these are not significant when compared with the contribution due to radon-222 and its daughters. Some of the site data needed to determine worker exposure as a result of natural radioactivity are as follows:

1. Radon-222 and radon daughter emission rates from the host rock.
2. Meteorological and environmental data.

Certain other site data are needed to estimate worker exposure from operations. These also are discussed in Section 8.3.5.4. These site data are the characteristics of the host rock required to determine the shielding properties of the host rock. Other than these site data, no further site data have been identified as necessary to determine the expected radiation exposure of the workers under normal repository conditions.

6.4.6 ISSUE 2.3: ACCIDENTAL RADIOLOGICAL RELEASES

6.4.6.1 Introduction

The question asked by Issue 2.3 is

> Can the repository 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 at the nearest boundary of the unrestricted area, or of workers in the restricted area, in excess of applicable limiting values?

The complete discussion of the proposed strategy for resolution of this issue is presented in Section 8.3.5.5 (accidental radiological releases). Readers unfamiliar with the issue resolution strategy for this issue, should review Section 8.3.5.5 before continuing.

Under this issue a list of structures, systems, and components important to safety will be developed. This list and a list of structures, systems, and components important to waste isolation combine to form the Q list.

Issue 2.3 has been subdivided into the following four information needs:

Information Need 2.3.1 Determination of credible accidents applicable to the repository.

Information Need 2.3.2 Determination of projected releases of radioactive material from the repository to restricted areas under accident conditions.

Information Need 2.3.3 Determination that projected worker exposures and exposure conditions meet applicable requirements.

Information Need 2.3.4 Determination that projected public exposures and exposure conditions under accident conditions meet applicable requirements.

These four information needs indicate the various steps (accident definition, projected releases, and predicted exposures) conducted in completing safety analyses for accident conditions. These steps have been taken in the two preliminary safety analyses completed to date for the proposed Yucca Mountain repository. The discussion for resolving the status of this issue is organized to show the progression made in moving from the analysis based on preliminary repository design concepts (Jackson, 1984) to that based on the conceptual design documented in the SCP-CDR (SNL, 1987).

Table 6-31 contains summary information describing computer codes used in the analyses supporting the completed work discussed here.

6.4.6.2 Work completed

This section discusses the work that has been performed to date to support resolution of this issue. The work has been documented in three reports:

1. Jackson, J. L., H. F. Gram, K. J. Hong, H. S. Ng, and A. M. Pendergrass, "Preliminary Safety Assessment Study for the Conceptual Design of a Repository in Tuff at Yucca Mountain," SAND83-1504, Sandia National Laboratories, Albuquerque, NM, December 1984. (Jackson et al., 1984)

2. "Preliminary Preclosure Radiological Safety Analysis," prepared by Bechtel National, Inc., for Sandia National Laboratories, Albuquerque, New Mexico, Appendix F of the Site Characterization Plan-Conceptual Design Report (SNL, 1987).

3. "Items Important to Safety and Retrievability for the Yucca Mountain Repository," prepared by Bechtel National, Inc., for Sandia National Laboratories, Albuquerque, New Mexico, Appendix L of the Site Characterization Plan-Conceptual Design Report (SNL, 1987).

The first report by Jackson et al. (1984) is of a scoping nature and based on preliminary repository concepts. Nevertheless, this work represents a significant contribution to the resolution of this issue. The second and third reports are Appendices F and L of the SCP-CDR, respectively. The second report is based on a more advanced and more complete design (although still conceptual in nature) of the repository than the Jackson et al. (1984) study and, therefore, enhances and updates some of the results of that earlier report. Also, the Jackson et al. (1984) report presented preliminary estimates of worst-case radioactive releases resulting from postulated accidents, while Appendix F of the SCP-CDR estimates radioactive releases for accidents developed using a probabilistic risk assessment (PRA) approach. Appendix L of the SCP-CDR discusses the results of Appendix F and uses the results to make a preliminary identification of items important to safety. The Jackson et al. (1984) report is discussed first, followed by an integrated discussion of Appendices F and L of the SCP-CDR.

Table 6-31. Codes used in analyses addressing Issue 2.3

Code name	Author	Code location	Design parameter	Analysis description
AIRDOS-EPA	R.E. Moore C.F. Baes III L.M. McDowell-Boyer A.P. Watson F.O. Hoffman J.C. Pleasant C.W. Miller	U.S. Environmental Protection Agency	Accident scenario. Atmospheric transport of radioactive plume. First-year and 50-yr dose commitments to maximum individual and repository personnel calculated using ALLDOS dose conversion factors.	Radionuclide releases modeled as Gaussian distributed short-duration plumes dispersed during average climatic conditions.
ORIGEN 2	A.G. Croff	Oak Ridge National Laboratory	Radionuclide source terms and release fractions (Jackson et al., 1984) 83-1504. Dose rate map extrapolation.	Calculates the radionuclide inventories for the various waste forms.

Summary of the Jackson et al. (1984) report

As just mentioned above, the Jackson et al. (1984) report presented preliminary estimates of worst-case releases resulting from postulated accidents for the repository based on preliminary repository concepts. Following is a discussion of the approach used by the Jackson et al. (1984) report and the results of that report.

The potential causes of accidental releases from repository operations that would expose the general public and repository personnel were divided into three main categories: (1) natural phenomena, (2) external manmade events, and (3) operational accidents. Three accidents were developed for the natural phenomena category: (1) flooding, (2) tornado or high winds, and (3) earthquake. Aircraft crash and ground motion resulting from underground nuclear explosion (UNE) tests were the two manmade events developed. Finally, for the operational accidents category, five accidents were developed: (1) a fuel assembly drop in a hot cell, (2) a transportation accident and fire at the loading dock that involves spent fuel, (3) a transportation accident and fire at the loading dock that involves commercial high-level waste (CHLW), (4) a transportation accident and fire in the waste-handling ramp that leads from the surface facilities to the disposal horizon, and (5) a transportation accident and fire in a waste emplacement drift in the horizontal waste emplacement concept.

Source terms for each accident were derived from the radionuclide inventory involved, the waste form, and the postulated accident. Radionuclide inventories were based on spent fuel from pressurized water reactors that had been out of reactor for 10 years, on CHLW derived from reprocessing this spent fuel, and on West Valley high-level waste (WVHLW).

The principal exposure pathway in the scenarios analyzed was the atmospheric transport of a radioactive plume. Exposures resulted from (1) radiation reflected from the plume (cloud shine), (2) radiation from fallout on the ground (ground shine), (3) direct contact (air immersion), (4) inhalation of radionuclides from the plume, and (5) ingestion of food-stuffs contaminated by radioactive fallout. In the flooding scenario, direct contact with contaminated flood water was the exposure mechanism for repository personnel.

The source terms and pathways were used to calculate the 50-yr dose commitments to the general public and the first-yr and 50-yr dose commitments to the maximum individual and repository personnel in each of the 10 scenarios. Dose commitments to the public were calculated using the AIRDOS-EPA computer code. Releases were modeled as Gaussian-distributed, short-duration plumes dispersed during averaged climatic conditions. Dose commitments to the maximum individual and to repository personnel were calculated using the ALLDOS dose conversion factors. The release plume was postulated to pass directly over the maximum individual at average wind velocity.

Dose commitments reported in this study were made up of an acute dose and a chronic dose commitment. These doses were received via external and internal exposure pathways. The acute dose was received within hours or minutes following the accidental release and was a result of external exposure. The chronic doses were received as a result of continuous exposure to radionuclides incorporated in the body after inhalation or ingestion. The calculated dose commitments were converted to health effects (excess cancer deaths), in accordance with the methodology for determining dose and health-effect relationships described in the BEIR III report (BEIR, 1980).

The Jackson et al. (1984) report presents the results of the analysis in terms of (1) doses to the repository workers, (2) doses to the maximum individual, and (3) doses to the general public. The results of the report also included the identification of accident scenarios and estimates of the probabilities of these accidents.

The Jackson et al. (1984) report analyzed ten accident scenarios. These were divided into three categories: (1) natural phenomena, (2) manmade external events, and (3) operational accidents. The natural phenomena analyzed included a flood, a 0.4g horizontal acceleration earthquake, and a tornado. The probability of these events was estimated to be 1.0×10^{-2} per yr, less than 1.3×10^{-3} per yr, and less than 9.1×10^{-11} per yr, respectively. Because of the low probability of the tornado, this event might not be considered credible. Accidents involving underground nuclear explosion (UNE) test and aircraft impact were the two manmade events analyzed. The probability of the UNE causing a radioactive release was estimated to be less than 1.0×10^{-3} for any one event. There were no data to estimate the event frequency. The probability of an aircraft impact was estimated to be 2.0×10^{-10} per year. Again, because of the low probability of the aircraft impact, this event might not be considered credible. There were five operational accidents analyzed: (1) a fuel assembly drop in a hot cell, (2) two transportation accidents and fires at the loading dock involving two different waste types, (3) a transportation accident and fire in the waste-handling ramp, and (4) a transportation accident and fire in an emplacement drift. The probabilities of these events were estimated to be 1.0×10^{-7}/yr for the transportation accidents. The transportation accidents were on the edge of what might be considered credible, taking into account the uncer-tainties of the estimates. A transportation accident with a fire can be avoided by using electric transporters and eliminating the fuel for the fire. This possibility is being considered.

The calculated first-year commitments for repository workers are below the occupational exposure limits (there is no specific accident-related exposure limit for workers) set by the NRC in 10 CFR Part 20 of 5.0 rem/yr and 3.0 rem/qtr for all accidents except for the transportation accident and fire in an emplacement drift. The dose commitment for this accident is 6.8 rem to workers in the emplacement drift. This accident was identified as being nearly credible. A major contributing factor to the dose, however, was the volatization of radionuclides caused by the fire. Since as noted earlier, all-electric transporters would remove the fuel for the fire, this accident can be eliminated or at least the consequences reduced considerably.

The calculated first-yr and 50-yr dose commitments for the maximum off-site individual were all less than the important-to-safety threshold established by the NRC (10 CFR 60.2) of 0.5 rem whole-body per accident. The greatest single first-year dose commitment for the maximum individual was calculated to be 0.055 rem and occurred in the aircraft impact scenario (recall this scenario has an extremely low probability). It should be noted that actual Air Force flight data were not used in the Jackson et al. (1984) report but are being factored into current evaluations. Similar results were calculated for the general public except that doses for the general public are always lower than for the maximum individual. The greatest single exposure to the population was calculated to be 110 man-rem (for a population of 19,908) and, again, occurred during the aircraft impact scenario. The results of this study are preliminary. Future work is expected to produce differing results based on new and more accurate data.

Summary of Appendices F and L of the SCP-CDR

The previous discussion of the Jackson et al. (1984) report presented results that were based on worst-case radioactive releases. The following discussion of Appendices F and L of the SCP-CDR presents a probabilistic-risk-assessment (PRA) approach to estimating radioactive releases from credible accidents. The methodology is related to determining items important to safety and is depicted in Figure 6-89. The complete reports are contained in Appendices F and L of the SCP-CDR while the methodology and results are summarized in the following. (This information was given earlier in Section 6.1.4.2 but is repeated here for reader convenience.)

The method used in (Appendix F of SNL, 1987) PRSA, basically follows the NRC methodology for a simplified and streamlined level 3 PRA described in the PRA Procedures Guide (NRC, 1983). The level of detail of the PRSA varies at each step, depending on the data and design information currently available. Since the primary objective of the PRSA was to provide a numerical basis for the development of a preliminary list of items important to safety, only accident scenarios resulting in public exposures were considered in detail.

After developing the facility and system model initiating events, both internal and external, were identified and screened by a panel of experienced design and safety analysis engineers. The basis for the screening was the potential of the events to contribute to a significant offsite release of radioactive materials. Using the event-tree technique, accident scenarios then were developed for those initiating events surviving the first screening process. Event trees are graphical depictions of the sequence of events that occur following an initiating event. The construction of an event tree is an inductive process in that one goes from the specific--the initiating event--to the general, all the possible results of the initiating event. The key factor for developing an event tree for each surviving initiating event was the selection and definition of the intermediate events. Event trees were constructed in detail appropriate to the level of design detail available and as necessary to adequately characterize the accident. Because of lack of data and design details, fault trees were not completely developed and analyzed; however, variations of conventional fault tree or fault diagrams were developed for most intermediate events. Fault trees are graphic depictions of the possible events that might lead to an intermediate event on an event tree. Constructing a fault tree is a deductive process in that one goes from the general--all possible ways for the intermediate event to come about--to the specific, the intermediate event. The use of fault diagrams provides important insight into the probabilities of intermediate events.

After event trees were developed, the probability of each initiating event and each intermediate event was evaluated, as were the consequences of accident scenarios. The probability and consequence analyses were performed in parallel. Both historical data and the judgment of a panel of engineers,

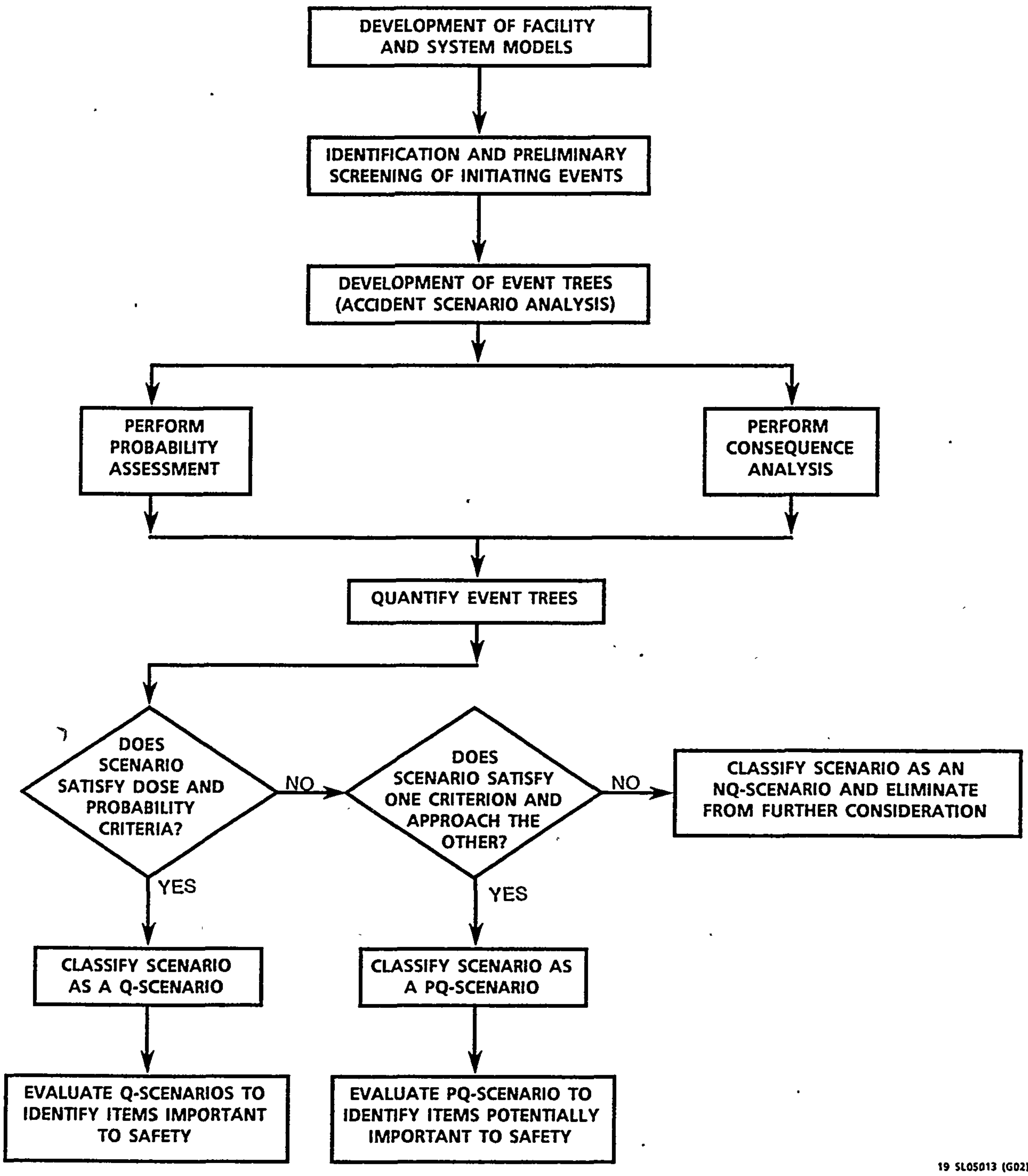


Figure 6-89. Q-list methodology for items important to safety.

experienced in safety analyses, were used in estimating probabilities. Consequence analyses involved the development of models and estimates of radionuclide releases, dispersion, and transport into the environment as well as calculation of doses. The results of the probability and consequence analyses were used to quantify the event trees. Briefly, the event trees are quantified by assigning probabilities to each intermediate event and consequences to each branch, or accident scenario, of the event tree.

On the basis of the results of the event tree quantifications, all accident scenarios that resulted in either dose consequences of more than 0.05 rem at the site boundary and probabilities of more than 1×10^{-9} per yr were selected as reference accident scenarios. The reference accident scenarios were identified by simplifying, or pruning, the event trees of all the accident scenarios that did not fall within the limitations established for the dose consequences and probability criteria. The initial list of items important to safety was derived from the reference scenarios and the numerical results of the analyses.

Reference accident scenarios that could potentially lead to significant offsite releases of radioactive material and dose consequences were developed using the previously described method. This analysis is reported in detail in the SCP-CDR (Appendix L of SNL, 1987). The following two criteria were used to screen the reference accident scenarios for scenarios that could lead to the identification of items important to safety.

1. Dose criterion: An accident scenario could potentially lead to the identification of items important to safety if the calculated off-site public dose was greater than or equal to 0.5 rem; otherwise, the accident scenario is not significant with respect to items important to safety.

2. Probability criterion: An accident scenario could potentially lead to the identification of items important to safety if the probability of occurrence of the scenario is greater than 1×10^{-5} per yr; otherwise, the scenario is not considered significant with respect to items important to safety.

In performing this second screening, the probability, including its uncertainty, were compared with the above criteria. If, and only if, an accident scenario passes both screening criteria, the accident scenario is classified as a Q scenario. Q scenarios then are further analyzed to determine which of the structures, systems, or components involved in the scenario are important to safety. Structures, systems, or components are important to safety if it is essential to either the prevention of the scenario or the mitigation of the scenario dose consequence.

Scenarios that are not significant with respect to items important to safety are classified as either a non-Q scenario (NQ scenario) or a potential-Q scenario (PQ scenario). All NQ scenarios are eliminated from further consideration in identifying items important to safety.

Any scenario not immediately identified as a Q scenario but which, as further study and design take place, is judged to have a reasonable potential to be upgraded to a PQ scenario is classified as a PQ scenario. Two criteria were used to decide between PQ and NQ. First, a scenario was classified as a PQ scenario even if no analyses had been performed if the item or scenario was sufficiently similar to others historically classified as PQ scenario or when practical consideration indicated that it could be a Q scenario. Second, if the analysis determined that either the consequences or probability exceeded the criteria and the other was sufficiently close that a change in assumptions or data could cause the criteria to be exceeded, the scenario was put on the PQ list. A variation of this second criteria was that when both consequence and probability were below the threshold but sufficiently close that a change in assumption or data could move both of them over, the scenario was classified as a PQ scenario.

Once a scenario is classified as a Q scenario or a PQ scenario, that scenario is further analyzed to determine which of the items involved in the scenario should be placed on the list of items important to safety or potentially important to safety. Further analysis of the scenario involves the evaluation of the systems, structures, and components involved in the scenario to determine what role the item plays in the scenario. Items that have a failure that causes the loss of consequence mitigation processes or have failure that directly causes the release of radioactive materials are classified as important to safety or potentially important to safety and placed on the Q list or PQ list, depending on which type of scenario is being evaluated. Current plans will make these items (PQ items), as well as items important to safety, subject to a QA level I program that satisfies the requirements of Title 10 CFR 60, Subpart G. A potentially-important-to-safety list is consistent with DOE guidance (DOE, 1987c).

The results to date have not identified any Q scenarios, or consequently, any Q-list items; however, this result is based on incomplete and preliminary data and design. For example, an airplane-crash event did not have available actual data and will have to be reexamined. Consequently, all items that have been classified as potential Q list will be treated as if they were Q list during future design, until the design detail and available data support a definitive analysis and conclusion. The preliminary list of potential items important to safety (PQ list), as developed in Appendix L of the SCP-CDR and through technical review of this appendix, is presented in Table 6-32.

In addition, as the design is developed, i.e., the reference configuration of the license application design and additional data become available, the complete sequence of the Q list method will be implemented again to refine, correct, and validate the initial results.

A detailed discussion of the methods used in determining items important to safety, is given in Appendix F of the SCP-CDR. A complete discussion of the results of the analysis to determine items important to safety, is given in Appendix L of the SCP-CDR.

Table 6-32. Potential Q-list for items important to safety at the Yucca Mountain Repository

Item	Location	Initiating event
Crane, shipping cask	Cask receiving and preparation area	Crane drops a shipping cask
Hot cell structure	Waste packaging hot cell	Earthquake causes hot cell structure failure
Crane	Unloading hot cell Consolidation hot cell Waste packaging hot cell	Earthquake causes crane to fall on fuel assemblies
Vehicle stop	Cask receiving and preparation area	Vehicle with cask falls in cask preparation pit (detailed analysis not performed)
Fire protection system	Waste-handling building	Fire involving radioactive material is a dispersion promoter (detailed analysis not yet performed)
Cask transfer mechanism	Surface storage vault	Cask transfer mechanism drops container with consolidated fuel rods
Transport Cask	Underground facility and ramp	Transporter coasts down the waste ramp and strikes the wall of the ramp or main access drift

6.4.6.3 Analysis needs

During each subsequent design phase (advanced conceptual design, license application design), the analyses described previously will be repeated. As site information becomes available and the repository design matures, the analyses will rely more on data and calculations and less on engineering judgement. The final set of analyses will be used to support the license application.

6.4.6.4 Site data needs

Immediate site data needs are meteorological data. These data include such items as wind and precipitation patterns, atmospheric-stability class, and site-boundary location. These data are used to calculate the transport of radionuclides in the atmosphere and are described in Section 8.3.1.12. Other data used by this issue are described in Sections 8.3.1.10, 8.3.1.11, and 8.3.1.13.

6.4.7 ISSUE 2.7: REPOSITORY DESIGN CRITERIA FOR RADIOLOGICAL SAFETY

6.4.7.1 Introduction

The question asked by Issue 2.7 is

> Have the characteristics and configurations of the repository been adequately established to (a) show compliance with the preclosure design criteria of 10 CFR 60.131 through 10 CFR 60.133, and (b) provide information for the resolution of the performance issues?

The general design criteria for the geologic repository operations area, (10 CFR 60.131); the additional design criteria for the surface facilities in the geologic repository operations area (10 CFR 60.132); and the additional design criteria for the underground facility (10 CFR 60.133) are presented in Table 6-33. The responsibility for meeting the criteria stated in 10 CFR 60.131 through 60.133 is divided among several issues. These issues and their associated responsibilities are identified in the table in Section 6.3.8.

To define the role and responsibilities currently assigned under this issue, an understanding of other issues is necessary. Issues 2.1 (public radiological exposures--normal conditions), 2.2 (worker radiological safety--normal conditions), and 2.3 (accidental radiological releases) address the compliance of the repository system with allowable releases of radioactive materials during preclosure. Under Issue 2.4 (waste retrievability) the retrieval option is maintained, and under Issue 1.11 (configuration of underground facilities--postclosure) the compliance with the postclosure design criteria of 10 CFR 60.133 is ensured with the exception of criteria related to sealing the repository that are addressed under the seal characteristics issue (Issue 1.12).

Issue 4.4 (preclosure design and technical feasibility) is the central or focusing issue that describes the development of the repository designs related to preclosure concerns.

With this understanding of the other issues, a detailed evaluation (Table 6-33) of the criteria specified in 10 CFR 60.131 through 10 CFR 60.133 reveals clearly the role of this issue (Issue 2.7). Under Issue 2.7 radiological-safety-related design criteria are developed and specified. The

Table 6-33. Design criteria for the geologic repository operations (page 1 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
60.131 General design criteria for the geologic repository operations area.		
(a) Radiological protection. The geologic repository operations area shall be designed to maintain radiation doses, levels, and concentrations of radioactive material in air in restricted areas within the limits of specified in Part 20 of this chapter. Design shall include		
1. Means to limit concentrations of radioactive material in air;	2.7	Yes
2. Means to limit the time required to perform work in the vicinity of radioactive materials, including, as appropriate, designing equipment for ease of repair and replacement and providing adequate space for ease of operation;	2.7	No
3. Suitable shielding;	2.7	Yes
4. Means to monitor and control the dispersal of radioactive contamination;	2.7	Yes
5. Means to control access to high radiation areas or airborne radioactivity activity areas; and	2.7	No
6. A radiation alarm system to warn of significant increases in radiation levels, concentrations of radioactive material in air, and of increased radioactivity released in effluents. The alarm system shall be designed with provisions for calibration and for testing its operability.	2.7	Yes
(b) Structures, systems, and components important to safety.		
1. Protection against natural phenomena and environmental conditions.	2.7	Yes
The structures, systems, and components important to safety shall be designed so that natural phenomena and environmental conditions anticipated at the geologic repository operations area will not interfere with necessary safety functions.		

Table 6-33. Design criteria for the geologic repository operations (page 2 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
2. Protection against dynamic effects of equipment failure and similar events.	2.7	No
The structures, systems, and components important to safety shall be designed to withstand dynamic effects such as missile impacts, that could result from equipment failure, and similar events and conditions that could lead to loss of their safety functions.		
3. Protection against fires and explosions.		
(i). The structures, systems, and components important to safety shall be designed to perform their safety functions during and after credible fires or explosions in the geologic repository operations area.	2.7	Yes
(ii). To the extent practicable, the geologic repository operations area shall be designed to incorporate the use of non-combustible and heat resistant materials.	4.2	No
(iii). The geologic repository operations area shall be designed to include explosion and fire detection alarm systems and appropriate suppression systems with sufficient capacity and capability to reduce the adverse effects of fires and explosions on structures, systems, and components important to safety.	2.7	Yes
(iv). The geologic repository operations area shall be designed to include means to protect systems, structures, and components important to safety against the adverse effects of either the operation or failure of the fire suppression systems.	2.7	No
4. Emergency capability.		
(i). The structures, systems, and components important to safety shall be designed to maintain control of radioactive waste and radioactive effluents, and permit prompt termination of operations and evacuation of personnel during an emergency.	2.7	No

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Table 6-33. Design criteria for the geologic repository operations (page 3 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
(ii). The geologic repository operations area shall be designed to include onsite facilities and services that ensure a safe and timely response to emergency conditions and that facilitate the use of available offsite services (such as fire, police, medical, and ambulance service) that may aid in recovery from emergencies.	4.2	No
5. Utility services.		
(i). Each utility service system that is important to safety shall be designed so that essential safety functions can be performed under both normal and accident conditions.	2.7	Yes
(ii). The utility services important to safety shall include redundant systems to the extent necessary to maintain, with adequate capacity, the ability to perform their safety functions.	2.7	No
(iii). Provisions shall be made so that, if there is a loss of the primary electric power source or circuit, reliable and timely emergency power can be provided to instruments, utility service systems, and operating systems, including alarm systems, important to safety.	2.7	No
6. Inspection, testing, and maintenance.	2.7	Yes
The structures, systems, and components important to safety shall be designed to permit periodic inspection, testing, and maintenance, as necessary, to ensure their continued functioning and readiness.		
7. Criticality control.	2.7	Yes
All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5% margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.		

Table 6-33. Design criteria for the geologic repository operations (page 4 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
8. Instrumentation and control systems.	2.7	No
The design shall include provisions for instrumentation and control systems to monitor and control the behavior of systems important to safety over anticipated ranges for normal operation and for accident conditions.		
9. Compliance with mining regulations.	2.7	No
To the extent that DOE is not subject to the Federal Mine Safety and Health Act of 1977, as to the construction and operation of the geologic repository operations area, the design of the geologic repository operations area shall nevertheless include such provisions for worker protection as may be necessary to provide reasonable assurance that all structures, systems, and components important to safety can perform their intended functions. Any deviation from relevant design requirements in 30 CFR, Chapter 1, Subchapters D, E, and N will give rise to a rebuttable presumption that this requirement has not been met.		
10. Shaft conveyances used in radioactive waste handling.	All waste will be transported underground in the ramp.[a]	
(i). Hoists important to safety shall be designed to preclude cage free fall.		
(ii). Hoists important to safety shall be designed with a reliable cage location system.		
(iii). Loading and unloading systems for hoists important to safety shall be designed with a reliable system of interlocks that will fail safely upon malfunction.		
(iv). Hoists important to safety shall be designed to include two independent indicators to indicate when waste packages are in place and ready for transfer.		

Table 6-33. Design criteria for the geologic repository operations (page 5 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
60.132 Additional design criteria for surface facilities in the geologic repository operations area.		
(a) Facilities for receipt and retrieval of waste.	2.7	Yes
Surface facilities in the geologic repository operations area shall be designed to allow safe handling and storage of wastes at the geologic repository operations area, whether these wastes are on the surface before emplacement or as a result of retrieval from the underground facility.		
(b) Surface facility ventilation.	2.7	Yes
Surface facility ventilation systems supporting waste transfer, inspection, decontamination, processing, or packaging shall be designed to provide protection against radiation exposures and offsite releases as provided in 60.111(a).		
(c) Radiation control and monitoring.		
1. Effluent control.	2.7	Yes
The surface facilities shall be designed to control the release of radioactive materials in effluents during normal operations so as to meet the performance objectives of 60.111(a).		
2. Effluent monitoring.	2.7	No
The effluent monitoring systems shall be designed to measure the amount and concentration of radionuclides in any effluent with sufficient precision to determine whether releases conform to the design requirement for effluent control. The monitoring systems shall be designed to include alarms that can be periodically tested.		

Table 6-33. Design criteria for the geologic repository operations (page 6 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
(d) Waste treatment.	2.7	No
Radioactive waste treatment facilities shall be designed to process any radioactive wastes generated at the geologic repository operations area into a form suitable to permit safe disposal at the geologic repository operations area or to permit safe transportation and conversion to a form suitable for disposal at an alternative site in accordance with any regulations that are applicable.		
(e) Consideration of decommissioning.	4.4	No
The surface facility shall be designed to facilitate decontamination or dismantlement to the same extent as would be required, under other parts of this chapter, with respect to equivalent activities licensed thereunder.		
60.133 Additional design criteria for the underground facility.		
(a) General criteria for the underground facility.		
1. The orientation, geometry, layout, and depth of the underground facility, and the design of any engineered barriers that are part of the underground facility shall contribute to the containment and isolation of radionuclides.	1.11	Yes
2. The underground facility shall be designed so that the effects of credible disruptive events during the period of operations, such as flooding, fires and explosions will not spread through the facility.	4.4	No
(b) Flexibility of design.		
The underground facility shall be designed with sufficient flexibility to allow adjustments where necessary to accommodate specific site conditions identified through in situ monitoring, testing, or excavation.	1.11	Yes
(c) Retrieval of waste.	2.4, 4.4	Yes

Table 6-33. Design criteria for the geologic repository operations (page 7 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
The underground facility shall be designed to permit retrieval of waste in accordance with the performance objectives of 60.111.		
(d) Control of water and gas.	4.4	Yes
The design of the underground facility shall provide for control of water or gas intrusion.		
(e) Underground openings.		
(1). Openings in the underground facility shall be designed so that operations can be carried out safely and the retrievability option maintained.	2.4, 4.2, 4.4	Yes
(2). Openings in the underground facility shall be designed to reduce the potential for deleterious rock movement or fracturing of overlying or sur rounding rock.	1.11	Yes
(f) Rock excavation.	1.11	Yes
The design of the underground facility shall incorporate excavation methods that will limit the potential for creating a preferential pathway for ground water or radioactive waste migration to the accessible environment.		
(g) Underground facility ventilation.		
The ventilation system shall be designed to		
(1). Control the transport of radioactive particulates and gases within and releases from the underground facility in accordance with the performance objectives of 60.111(a).	2.7	Yes
(2). Assure continued function during normal operations and under accident conditions.	2.7, 4.2, 4.4	Yes
(3). Separate the ventilation of excavation and waste emplacement areas.	2.7,4.4	No

Table 6-33. Design criteria for the geologic repository operations (page 8 of 8)

Design Criteria of 10 CFR Part 60	Issue that addresses the criterion	Are site data needed to address the criterion?
(h) Engineered barriers.	1.11	Yes
Engineered barriers shall be designed to assist the geologic setting in meeting the performance objectives for the period following permanent closure.		
(i) Thermal loads.	1.11	Yes
The underground facility shall be designed so that the performance objectives will be met taking into account the predicted thermal and thermomechanical response of the host rock, and surrounding strata, ground-water system.		

[a]The repository design for the Yucca Mountain site uses a ramp (instead of a shaft and hoist) through which waste is transported underground by transportors.

related design work will be done in Issue 4.4 since other issues specify requirements that must be met by the system design, e.g., the ventilation system design must meet criteria related to radiological safety (Issue 2.7), retrievability (Issue 2.4), and nonradiological health and safety (Issue 4.2).

The proposed issue resolution strategy for this issue is presented in Section 8.3.2.3 (repository design criteria for radiological safety). Readers unfamiliar with the issue resolution strategy for this issue should review the contents of Section 8.3.2.3 before continuing.

The information needs for Issue 2.7 are as follows:

Information Need 2.7.1 Determination that the design criteria in 10 CFR 60.131 through 60.133 and any appropriate additional design objectives pertaining to radiological protection have been met.

Information Need 2.7.2 Determination that the design criteria in 10 CFR 60.131 through 60.133 and any appropriate additional design objectives pertaining to the design and protection of structures, systems, and components important to safety have been met.

Information Need 2.7.3 Determination that the design criteria in 10 CFR 60.131 through 60.133 and any appropriate additional design objectives pertaining to criticality control have been met.

Information Need 2.7.4 Determination that the design criteria in 10 CFR 60.131 through 60.133 and any appropriate additional design objectives pertaining to compliance with mining regulations have been met.

Information Need 2.7.5 Determination that the design criteria in 10 CFR 60.131 through 60.133 and any appropriate additional design objectives pertaining to waste treatment have been met.

The information needs indicate that the development of the design criteria applicable for radiological safety require (1) interfaces with site, waste package, and repository designs, (2) understanding potential conditions that may exist in surface and underground facilities, and (3) establishing means of controlling releases. The discussions below will focus on the work completed to date related to criteria development and to the means for controlling releases that are part of the current design.

6.4.7.2 Work completed

The work completed in support of this issue consists of design criteria established for the use in the conceptual design presented in the SCP-CDR and in criteria prepared to support the advanced conceptual design activities. Repository design guidelines were issued in advance of the SCP-CD. As part of the SCP-CD development these guidelines were expanded by establishing additional means of limiting exposures and by more clearly defining planned operations. These refinements are part of the Subsystem design requirements (Appendix P of SNL, 1987).

Design criteria have been obtained from both regulatory guidance and from DOE Orders. The regulatory guidance is principally from 10 CFR 60.131, 132, and 133 while the DOE guidance is principally from DOE Orders 5480.1B (DOE, 1986b), 6410.1 (DOE, 1983b), and 6430.1 (DOE, 1983a). This guidance is translated into several specific points of design philosophy for the NNWSI Project work. Some of the more important points are as follows:

1. Design basis will be 20 percent of allowable release.

2. No administrative controls will be used to meet worker dosage criteria.

3. Separate underground ventilation systems will be used for excavation and waste emplacement areas.

4. Redundancy of systems and equipment will be provided.

5. Systems for mitigating disruptive events will be provided.

6. Design will consider decommissioning requirements.

7. Maintainability will be considered in facilities and equipment design.

The development of the conceptual design has led to identifying numerous means of limiting releases. For the surface facilities, these include the compartmentalization of surface facility operations, design of hot cells for negative-pressure operation, shielding of selected operations, and development of zoned ventilation systems in the surface buildings. In addition, filtration systems for gaseous effluents, strippable wall coatings in selected areas, removable liners from selected equipment, and automated systems, where possible, are used in the current design. Placement of the waste-handling building and its effluent exhaust systems is influenced by the prevailing wind direction as a means of limiting contamination of other surface facilities. Access control is also provided for the surface facilities. Collection and treatment of site-generated waste is reflected in the current design, as are specified areas for decontamination activities.

Means of limiting releases in the underground facilities rely primarily on decisions made relative to the ventilation systems and to equipment design. In the ventilation system, filters are provided on the underground exhaust building, a positive-pressure differential will exist between the development and waste emplacement areas to allow leakage to always be toward the emplacement side. For the equipment program, means of shielding are reflected in the transporter cask, transporter cab, and shield plugs designed for the waste emplacement boreholes. Additionally, speed limitations and braking criteria have been established for the transporter as a means of limiting the consequences of potential accidents. Access control is provided.

Means of limiting releases are inherent in many of the planned operations for the repository. For example, operations are sequenced so that workers are not expected to be consistently downstream from waste emplacement operations. Similarly, airflow related to muck removal from the development area will be exhausted along the tuff ramp thereby having minimal potential for ingestion by workers. Extensive inspections are planned for waste upon arrival and during processing. Maintenance operations are being designed, where possible, to be conducted outside the hot cells although obviously some maintenance will require remote systems.

6.4.7.3 Future work

As described in the previous paragraphs and reflected throughout the SCP-CDR applicable criteria have been identified as the basis for establishing means of controlling releases of radioactive materials from the repository.

The future work related to this issue consists primarily of identifying progressively more detailed design criteria for advanced conceptual design and license application design activities. This work is described in more detail in Section 8.3.2.3.

6.4.8 ISSUE 2.4: WASTE RETRIEVABILITY

6.4.8.1 Introduction

The question asked by Issue 2.4 is

> Can the repository be designed, constructed, operated, closed and decommissioned so that the option of waste retrieval will be preserved as required by 10 CFR 60.111?

In general, 10 CFR 60.111(b)(1) requires that the emplaced waste must be retrievable on a reasonable schedule until the completion of the performance confirmation program and NRC review. In addition, Section 5-1(a)(3) of 10 CFR Part 960 includes a requirement that the repository siting, construction, operation, and closure will be demonstrated to be technically feasible on the basis of reasonably available technology. These regulatory requirements form the basis for Issue 2.4. The resolution of this issue follows the issue resolution strategy (IRS) presented in Section 8.3.5.2. Readers unfamiliar with the IRS for this issue should review Section 8.3.5.2 before continuing.

The object of this issue is to ensure that the repository preserves the option of waste retrieval. To ensure that the retrieval option is maintained, this issue

1. Establishes a strategy for resolution through performance allocation.

2. Defines retrievability-related design criteria.

3. Establishes normal and off-normal conditions anticipated for retrieval operations. [Note that in this section, the term off-normal is used to identify conditions which are anticipated to occur infrequently. In future documents, the term off-normal will be replaced with the term abnormal.]

4. Identifies information (analyses, demonstrations, etc.) required from other issues to ensure compliance with retrievability requirements.

5. Assesses compliance with the regulatory requirements for retrievability.

These responsibilities assigned under Issue 2.4 are depicted in Figure 6-90, which details the strategy to be used for retrievability evaluation. The significance of Issue 4.4, (preclosure design and technical feasibility), also is evident in the figure. Under Issue 4.4 the design for facilities and equipment is developed, analyses of the design are conducted, needed tests and demonstrations are conducted, and an operations plan is developed.

In developing the strategy for resolving this issue, it was determined that the ability to perform retrieval operations is based on the ability to perform the following four functions:

1. Provide access to the emplacement boreholes.
2. Provide access to the waste containers.
3. Remove waste containers from the emplacement boreholes.
4. Transport and deliver the waste to the surface facilities.

To ensure that the design will include the ability to perform these functions under normal and off-normal conditions, it will be necessary to document the following:

1. Retrieval strategy and planning.

2. Retrieval conditions.

3. Retrievability input to repository design requirements (RDR) document.

4. Facility and equipment designs, demonstrations and design analyses.

5. Retrievability compliance analyses.

The designs, demonstrations, and supporting analyses will be documented in reports produced under Issue 4.4 (Section 8.3.2.5). Hence, discussions of the status of the retrievability issue will be focused on documentation produced to date regarding items 1, 2, 3, and 5 of the previous list.

Section 6.4.8.2 presents the work completed to date. The future work to be performed on these products is summarized in Section 6.4.8.3.

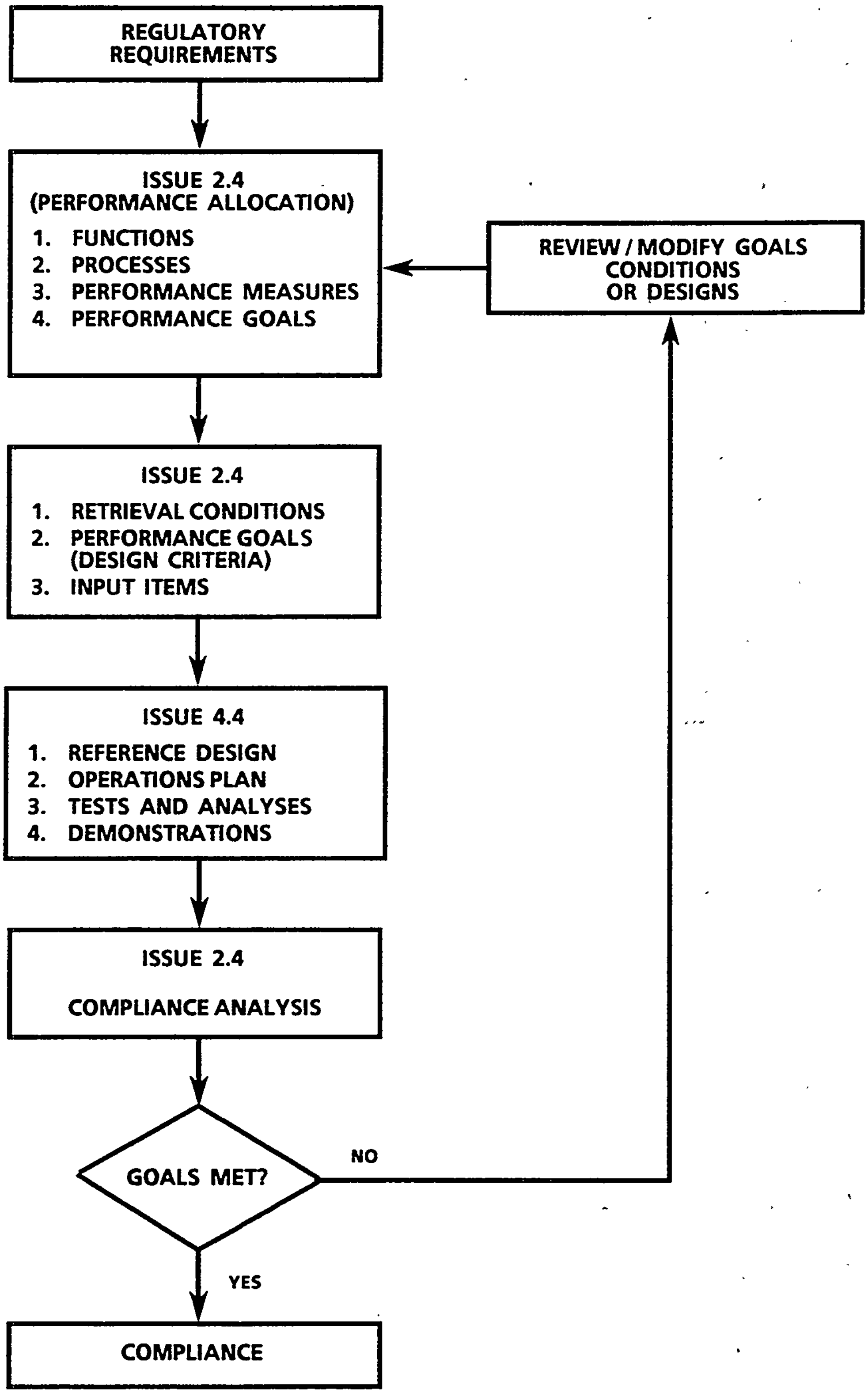

Figure 6-90. Strategy to be used for retrieval evaluation.

6.4.8.2 Work completed

6.4.8.2.1 Retrieval strategy and planning documents

The approach to development of a strategy for retrievability has been to develop a guidance paper to ensure consistency of planning assumptions concurrently with development of a strategy paper to adapt this guidance for NNWSI Project specific applications. The documents that address strategy and planning are as follows:

1. "Department of Energy Generic Requirements for a Mined Geologic Disposal Site," (DOE, 1986d).
2. "Retrievability: Strategy for Compliance Demonstration," SAND84-2242 (Flores, 1986).

The basis (or data) used to develop this strategy included the regulatory requirements, the Generic Requirements (GR) document (DOE, 1984b), and the Mission Plan (DOE, 1985a). These reports address (1) the identification and evaluation of the regulatory requirements for retrievability, (2) the establishment of a design basis for evaluating the ability to retrieve, (3) the identification of the expected repository conditions at the time of retrieval, (4) the development of the methodology for defining normal conditions and identifying off-normal conditions, and (5) the identification of the timing for design and demonstration activities that are needed to ensure that the ability to retrieve is maintained throughout the retrievability period.

These two reports will be used to form the basis for more detailed definition of the demonstrations, analyses, operations, equipment development, and anticipated conditions. Consistency in these detailed definitions requires an expansion of the strategy documented to date. This proposed strategy or approach is described in Appendix J of the SCP-CDR. This approach relies on the use of a probabilistic approach as a means for segregating conditions that require consideration from those that do not, Figure 6-91. A further segregation is identified to classify the conditions as normal and off-normal. This classification of conditions will form the basis for the segregation of items in the design approach, demonstration plans, and anticipated degree of readiness for various conditions. It is recognized that substantial engineering judgment will be necessary to implement this approach and that uncertainty will exist with regard to the exact probability of occurrence of many of the postulated conditions; nevertheless, it is planned to apply this, or a similar framework, in advanced conceptual design studies.

6.4.8.2.2 Retrieval conditions

Estimates of the repository conditions at the time of retrieval are important for use as input to the design basis and demonstration plans. The retrieval conditions are divided into two categories: normal and off-normal. Normal conditions are those conditions under which the retrieval process can

PROBABILITY		10^{-1}	10^{-3}	10^{-5}
CONDITION CLASSIFICATION	NORMAL	EXPECTED OFF-NORMAL	CREDIBLE OFF-NORMAL	NOT CREDIBLE
DESIGN APPROACH	DESIGN BASIS		CONTINGENCY	NOT INCLUDED
DEGREE OF READINESS	ESTABLISHED EQUIPMENT ESTABLISHED OPERATIONS		EQUIPMENT CONCEPTS OPERATIONS CONCEPTS	DEVELOP CONDITION SPECIFIC PLAN
DEMONSTRATION PLANS	PROOF-OF-PRINCIPLE DEMONSTRATIONS PROTOTYPE DEMONSTRATIONS (AS NEEDED)		PROOF-OF-PRINCIPLE DEMONSTRATIONS (IF NEEDED)	NONE

Figure 6-91. Classification of retrieval conditions based upon probability.

be performed using standard (essentially, the emplacement equipment) equipment and procedures. Off-normal conditions exist when nonstandard equipment and procedures are required.

The basic approach to the development of the normal conditions involves the following: (1) the identification of the repository systems important to the performance of the retrieval process and (2) the prediction of the condition of those systems at the time of retrieval. The repository systems were identified by evaluating the ability to perform the four major functions for retrieval using the SCP-CD concepts as a basis. The prediction of the condition of these systems was accomplished using current design information, completed analyses, and engineering judgment.

The off-normal conditions were identified using the approach discussed in Appendix L of the SCP-CDR. As shown in Appendix L, a short list of potentially credible processes and events was developed by screening approximately 75 events and processes contained in the master list. To identify potential off-normal conditions, the short list of processes and events were evaluated relative to the ability to perform the four retrieval functions. The potential off-normal conditions were screened on a probability basis, resulting in the determination of potentially credible off-normal conditions.

The basis for the identification of normal and off-normal conditions was the regulatory requirements, results from technical analyses, results of literature reviews, the SCP-CD, and engineering judgment.

<u>Normal retrieval conditions</u>

As mentioned previously, normal conditions are those conditions under which the retrieval process can be performed using standard equipment and procedures. The system elements whose performance and, as a result, condition could affect the ability to retrieve include ramp and drifts, emplacement boreholes, ventilation system, including the shafts, waste-handling building, retrieval equipment, and the waste container. A complete discussion of current normal conditions for retrieval is presented in Appendix J of the SCP-CDR. In this section, a brief discussion on the normal conditions for these elements is presented.

Ramps and drifts

The normal conditions within the ramps and drifts are characterized in terms of (1) rock temperature in the drifts, (2) condition of the opening, (3) radiation levels, and (4) air quality.

The basis for retrievability planning is tied to the current conceptual design results as follows:

1. The anticipated temperatures for the floor of the emplacement drifts and the wall of the access drifts for vertical emplacement and for the emplacement drift floor for horizontal emplacement are addressed in Section 8.3.5.2. As shown in Appendix J of the SCP-CDR, the goal to limit the temperature to 50°C at 50 yr in the access drifts for

vertical emplacement and in the emplacement drifts for horizontal emplacement is met.

2. Under normal conditions, current design calculations suggest that the ramp and drifts are expected to remain stable. It is anticipated that small pieces of rock will fall through the holes in the wire mesh. This will be managed with light maintenance. It is also expected that in local, more highly fractured areas, more extensive and frequent maintenance may be required.

3. For normal conditions, personal radiation protection will be the same during retrieval as that required for emplacement operations (Dennis et al., 1984a). The radiological environments for worker safety are addressed under Issue 2.2 while the design for radiological safety is addressed under Issue 2.7.

4. Acceptable air quality will be maintained in all operational areas during retrieval operations. In the ramp, service areas, and access drifts, acceptable air quality will be maintained until repository closure, through the use of continuous ventilation. However, there are no plans to ventilate the emplacement drifts during the caretaker period. Therefore, ventilation will be reestablished to ensure that an acceptable air quality exists before reentry for initiation of retrieval operations will be allowed.

Emplacement boreholes

The conditions within the emplacement boreholes are characterized in terms of (1) rock temperature, (2) condition of the opening, (3) radiation levels, and (4) condition of the borehole liner.

The basis for retrievability planning under normal conditions is tied to the current conceptual design as follows:

1. The predicted temperature histories for the emplacement boreholes for the vertical and horizontal emplacement concepts are shown in Appendix J of the SCP-CDR. The temperature remains above 100°C throughout the retrievability period, therefore, a dry environment is expected.

2. For the vertical emplacement concept, the borehole will be stable with negligible amounts of rockfall into the emplacement borehole under normal conditions. For the horizontal concept, minor rockfall against the liner is anticipated. In addition, as noted previously, a dry environment, as a result of high temperatures, is expected.

3. At the time of emplacement, order-of-magnitude estimates of the waste container surface radiation dose rates for spent fuel are 10^5 rem/h for gamma and 10^2 rem/h for neutron radiation (O'Brien, 1985). These surface radiation levels are used as the worst-case levels for shielding design.

4. Under normal conditions, the liner will be intact and provide acceptable access to the emplaced waste containers throughout the design basis 84-yr period.

Ventilation system

The ventilation system equipment (fans, regulators, chillers, etc.) will continue providing ventilation to the ramp and access drifts throughout the caretaker period. In addition, the distribution and regulation system to the emplacement drifts will be used on a periodic basis for inspection and maintenance of the emplacement drifts. As a result, the emplacement ventilation system including the shafts and ramps will be maintained in a fully operational condition throughout the caretaker period.

Waste-handling building

If waste emplacement operations are in progress, it is expected that the waste-handling building, and the equipment contained within it, will be in operable condition. However, it is anticipated that extensive modifications and additional construction would be necessary to accommodate the retrieval operations. During the caretaker period, it is assumed that maintenance and repair will have been performed only to maintain the structure; therefore, repair and maintenance may be required to bring the waste handling equipment within the building to an operational state.

Retrieval equipment

If waste emplacement operations are in progress, the equipment required for waste removal under normal conditions (the emplacement equipment) will be in an operational condition. During the caretaker phase, it is anticipated that two sets of retrieval equipment will remain operational in support of the performance confirmation program. For full-repository retrieval initiated during the caretaker phase, it is anticipated that maintenance and repair will be required to achieve an operational condition for the other two sets of retrieval equipment, assuming the current design basis that four sets will be required. In addition, training of additional operators probably will be required.

For the horizontal concept, under normal conditions, the dolly and the dolly hook, described in Section 4.5 in the SCP-CDR, are not expected to fail during retrieval operations. The current planning basis for the dolly roller system under normal conditions is that it will be operable during retrieval operations; however, sliding friction, not rolling friction is assumed in establishing design loads for the emplacement or retrieval mechanism in the transporter.

Waste container condition

Under normal conditions, the waste container is expected to remain intact throughout the retrievability period and the removal process.

Off-normal retrieval conditions

As mentioned previously, off-normal conditions exist when the retrieval process must be performed using nonstandard equipment or procedures. The current development of off-normal retrieval conditions was part of the retrievability evaluation presented in Appendix L of the SCP-CDR. As shown in Figure 6-92, the first step was to develop a comprehensive list of processes and events that could potentially lead to a delay in retrieval operations. These events and processes were categorized as naturally occurring, human-induced, and repository-induced. Input for this master list was obtained from literature surveys, engineers involved in developing retrieval equipment and operations, working sessions with engineering professionals, and peer and management reviews of the master list. The resulting master list of approximately 75 events and processes is shown in Table 3-1 of Appendix L. This master list then was screened using the set of criteria shown in Table 3-2 of Appendix L to eliminate from further consideration those events and processes that, under current design concepts and understanding of site processes, either were not applicable to the Yucca Mountain site or obviously resulted in an insignificant time delay in retrieval operations. This screening resulted in the short list of processes and events that could potentially lead to a significant time delay in performing retrieval operations. The events and processes on the short list are as follows:

1. Tectonics.

2. Variability in rock characteristics.

3. Human error.

4. Aging and corrosion of equipment and facilities.

5. Radiolysis.

Consequences of these events and processes (e.g., waste package failure) are addressed in more detail of Appendix L of the SCP-CDR. The next step involved identifying retrieval conditions that result from the events and processes identified in the short list. These conditions were developed by examining the effects of these events and processes on the ability to perform the following four retrieval functions: (1) access to the emplacement boreholes, (2) access to the waste containers, (3) ability to remove waste containers, and (4) return waste containers to the surface. Using engineering judgment and the SCP-CD as a basis, a list of approximately 110 potential off-normal conditions was developed. These conditions are presented in Table 3-5 of Appendix L. These conditions were then assigned a probability of occurrence using engineering judgment. These estimates of probability were assigned using qualitative descriptions of high, medium, low, and negligible. Negligible was considered to be less than 10^{-5}/yr. Conditions that were judged to have a negligible probability of occurrence were removed from further consideration. Retrieval conditions judged to have a medium or low probability of occurrence were classified as off-normal. A list of the 43 identified off-normal conditions is contained in Table 3-1 of Appendix J of the SCP-CDR.

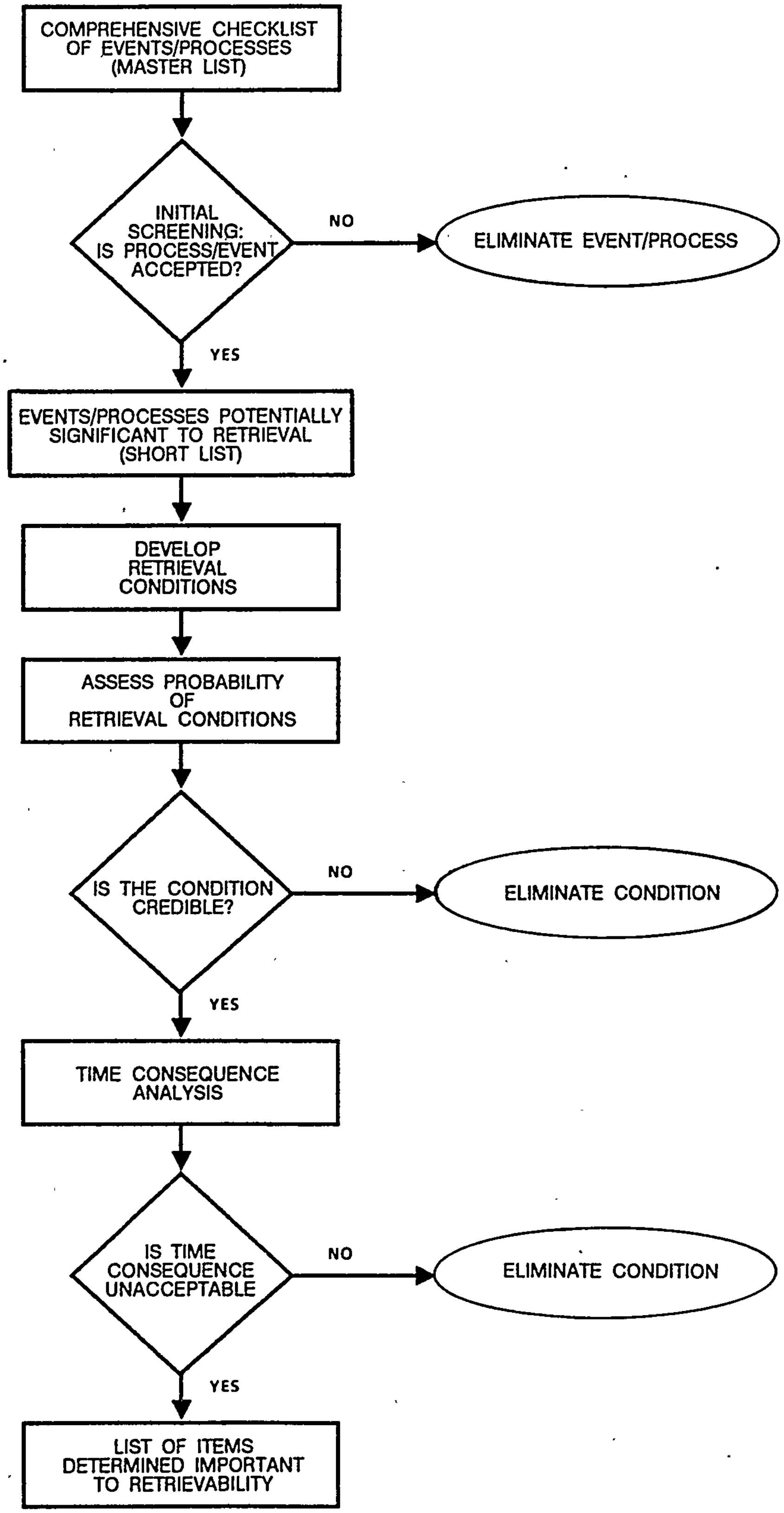

Figure 6-92. Methodology used to determine items important to retrievability.

Retrieval conditions judged to have a high probability of occurrence were classified as normal conditions and, if not already included in the design basis for retrieval, will be added. A time consequence analysis was performed to estimate the delay in total time for performing the retrieval operations for the repository resulting from the identified off-normal conditions. The following six conditions were judged to have a potential time consequence of six months or greater:

1. Ventilation equipment failure as the result of a tectonic (ground motion) event.

2. Emplacement borehole rockfall as the result of a tectonic (ground motion) event (vertical only).

3. Waste container "tilt" as the result of a tectonic (ground motion) event (vertical only).

4. Shield plug jam as the result of a tectonic (ground motion) event.

5. Ventilation system failure as the result of a human error related to maintenance.

6. Transporter collision with the ramp as the result of an operator error.

The off-normal conditions presented in Appendix J of the SCP-CDR have been used to guide the development of off-normal retrieval operations and equipment needs. In the future, Table 3-9 of Appendix L of the SCP-CDR will be used as a starting point for the application of probability evaluations shown in Figure 6-92. The application of these concepts will result in the development of more detailed design and operational criteria for equipment and facilities, more accurate guidance for equipment development requirements and demonstration needs, and a firm basis for deciding what equipment will be on hand at the repository.

6.4.8.2.3 Retrievability input to repository design requirements

The development of retrievability-related design criteria began during early design stages and has become more specific as the design has evolved. The current set of design criteria was refined as a result of the performance allocation process described in Section 8.3.5.2. In the future, this set of design criteria will be refined using the concept shown in Figure 6-91 and discussed in Appendix J of the SCP-CDR.

The basis or data used in the development of the design criteria include the regulations concerning retrievability (Section 2.6.1 of SNL, 1987), the SCP-CD, Appendix D of the GR document (DOE, 1986d), and the Mission Plan (DOE, 1985a).

The early designs that reflected the inclusion of retrievability-related design criteria were the Repository Design Concepts Report (Jackson, 1984) and the Two-Stage Repository Report (SNL, 1986).

6.4.8.2.4 Retrievability compliance analysis

Under Issue 2.4 the reference design and the results of analyses, tests, and demonstrations performed under Issue 4.4 are evaluated to determine if the goals for retrievability are met. This is accomplished during the compliance assessment activity shown in Figure 6-90. To effectively communicate between Issues 2.4 and 4.4, the concept of an input item is used. As shown in Figure 6-90, a list of input items is generated under Issue 2.4 to identify the information required from other issues to perform the retrievability compliance analysis. The current list of approximately 20 input items was developed in Section 8.3.5.2. For example, to ensure that the design will provide usable openings, a need is generated to provide opening and support system designs, perform analyses to predict their performance, develop contingency plans for off-normal conditions, and provide supporting evidence that the design will be satisfactory (i.e., the performance goals are met), which would be performed under Issue 4.4. A complete discussion of these input items is included in Sections 8.3.5.2.2 through 8.3.5.2.6.

Work toward providing these input items has been completed in the following areas: (1) underground opening design, (2) ventilation system design, (3) radiological protection, and (4) equipment development.

Underground opening design

Information on the stability of the underground openings is required both to perform the first retrieval function (i.e., to ensure that the access and drifts will be usable throughout the retrievability period) and to provide information concerning emplacement borehole stability for the second function--access to the waste container.

Evaluations of the thermal and mechanical effects on stability of shafts, ramps, drifts, and boreholes have been the focus of about 15 reports or studies synopsized in Section 6.4.10.2 and are not repeated here. These analyses have used a variety of numerical and empirical approaches as follows: (1) finite-element methods, (2) boundary-element methods, and (3) tunnel-indexing methods. Similarly, different constitutive models were used as follows: (1) elastic models, (2) ubiquitous-joint models, (3) compliant-joint models, and (4) elastic-plastic models. Other items that have been varied in some of the analyses include (1) opening sizes and shapes, (2) depths, (3) thermal and mechanical properties, and (4) fracture properties and in situ conditions. The common preliminary conclusions drawn from the approaches used to date are as follows:

1. Drifts, shafts, and ramps, as currently designed, are predicted to remain stable during preclosure.

2. Waste emplacement boreholes are predicted to remain stable during preclosure, although some potential exists for negligible amounts of rock to fall on the liner planned for use in horizontal emplacement holes.

3. Excavation-induced response of openings in Topopah Spring tuff should be expected to be similar to those in the Grouse Canyon tuff in G-Tunnel.

Ventilation system design

Evaluations of the ventilation system design are required to determine the feasibility of providing a safe working environment for retrieval operations. The status of these evaluations is presented in Section 3.4 of the SCP-CDR. Detailed analyses are documented in Appendix C of the SCP-CDR. The principal results related to retrievability are synopsized in the following text.

The design of the ventilation system has considered three different sets of operations conditions: (1) for construction and emplacement operations, (2) for reentry for inspection, and (3) for maintenance and retrieval operations. Preliminary ventilation system design analyses have been completed for both the vertical and horizontal emplacement orientations. For construction and emplacement operations, maximum velocity constraints are met and requirements to provide acceptable airflow to the ramp, drifts, and service areas are met using currently available equipment.

To ensure that continued access to the emplacement boreholes is maintained, reentry for inspection purposes is planned. The criteria for an acceptable inspection environment was defined as an air cooling power greater than 300 W/m^2 and a dry bulb temperature less than 45°C. As shown in Section 3 of the SCP-CDR, using ambient air, it would take 168 d and 14 d to cool the vertical and horizontal emplacement drifts, respectively. Using chillers, it would take 21 d and 5 d to cool the vertical and horizontal emplacement drifts, respectively. This is based on a 708-kW cooling load for vertical emplacement and a 490-kW cooling load for horizontal emplacement.

For reentry for maintenance or retrieval purposes, the criteria for an acceptable environment is defined as an air cooling power greater than 500 W/m^2 and a dry bulb temperature less than 40°C. As shown in Section 3 of the SCP-CDR, more than 560 d are required to cool the vertical emplacement drifts using ambient air. The horizontal emplacement drifts would require 70 d for cooling using ambient air. Assuming a 708-kW cooling load, the vertical emplacement drift can be cooled in approximately 37 d. For the horizontal concept, the emplacement drifts can be cooled within 11 d using a 490-kW cooling load.

The results of these preliminary analyses indicate that the requirements for providing adequate ventilation can be met for inspection and retrieval operations using currently available equipment. Future work will focus on refinement of these analyses with specific attention paid to the following items:

1. Particulates (burden and type).
2. Temperature.
3. Humidity.

4. Airborn radioactive contaminants.
5. Gaseous pollutants.

Radiologic protection

The basic approach to ensuring radiological protection is as follows: (1) identify radiation sources under normal and accident conditions, (2) establish the radiation levels for these sources, and (3) develop designs and operations plans that result in the ability to perform retrieval under an acceptable radiological environment.

The reports that address radiological concerns include the following:

1. "Reference Nuclear Waste Description for a Geologic Repository at Yucca Mountain, Nevada," SAND84-1848 (O'Brien, 1985).

2. "Preliminary Safety Assessment Study for the Conceptual Design of a Repository in Tuff at Yucca Mountain," SAND83-1504 (Jackson et al., 1984).

3. "NNWSI Repository Worker Exposure Volume I, Spent Fuel and High-Level Waste Operations in a Geologic Repository in Tuff," SAND83-7436/1 (Dennis et al., 1984a).

4. "Worker Radiation Dose During Vertical Emplacement and Retrieval of Spent Fuel at the Tuff Repository," SAND84-2275 (Stinebaugh and Frostenson, 1987).

5. "Preliminary Preclosure Radiological Safety Analysis," (Appendix F of SNL, 1987).

6. "Items Important to Safety and Retrievability at Yucca Mountain," (Appendix L of SNL, 1987).

Radiation sources in the repository are categorized as follows: (1) waste generated and (2) naturally occurring. Estimates of the source term for the waste containers are reported in O'Brien (1985). These estimates will be used as a basis for radiation shield designs for retrieval equipment. Worker dose estimates are contained in Jackson (1984), Dennis et al. (1984a), and Stinebaugh and Frostenson (1987). These studies indicate that radiation doses of less than 1 rem/yr are possible through modifications in operations or equipment. Preliminary results of studies to identify accidents during retrieval operations, which result in radioactive releases are discussed in Appendices F and L of the SCP-CDR.

Future work will include refinement of public and worker dose-rate estimates, detailed equipment shielding designs, and refinement of accident analyses. The principal concern for naturally occurring radiation sources is the potential contamination from radon-222 and radon daughters. No work in this area has yet been completed. Current work is focused on estimation of the emanation rate for radon-222 and its potential effect on repository operations. This work will include G-Tunnel tests and tests at the exploratory shaft facility (Section 8.3.1.15, rock characteristics program).

Equipment development

The development of equipment is achieved in the following phases:

1. Development of equipment concepts.
2. Conceptual design.
3. Interim design phases (proof-of-principle, prototype, etc.).
4. Final design.

Retrieval equipment development is performed under Issue 4.4 (preclosure design and technical feasibility). Work on equipment development is progressing in two areas: (1) the development of equipment for emplacement and retrieval of wastes and (2) the development of equipment to accurately drill and line long horizontal emplacement boreholes. The equipment related to drilling and lining horizontal emplacement boreholes is included here since the requirement to line the boreholes is principally related to ensuring access to the waste containers for possible retrieval. The concepts for equipment for emplacement and retrieval of waste have been developed and are presented in the following documents:

1. "Conceptual Engineering Studies and Design for Three Different Machines for Nuclear Waste Transporting, Emplacement, and Retrieval," SAND83-7089 (Fisk et al., 1985).

2. "Disposal of Radioactive Waste Packages in Vertical Boreholes--A Description of the Operations and Equipment for Emplacement and Retrieval," SAND84-1010, May 1986 (Stinebaugh and Frostenson, 1986).

3. "Disposal of Radioactive Waste Packages in Horizontal Boreholes--A Description of the Operations and Equipment for Emplacement and Retrieval," SAND84-2640, May 1986 (Stinebaugh et al., 1986).

4. "One-Twelfth-Scale Model of the Horizontal Emplacement and Retrieval Equipment for Radioactive Waste Containers at the Tuff Repository," SAND86-7135, 1987 (White et al., 1986).

Preliminary conceptual designs for retrieval equipment are presented in Stinebaugh and Frostenson (1986), Stinebaugh et al., (1986) and SCP-CDR, Appendix E. In addition, a one-twelfth-scale model of the waste emplacement and retrieval equipment for horizontal emplacement has been completed (White et al., 1986). This model is being used to gain perspective on design features that need modification. Future work will focus on developing design concepts that allow retrieval under off-normal conditions and completing conceptual designs of retrieval equipment for normal conditions during advanced conceptual design.

The work that has been completed on the development of drilling and lining equipment for the horizontal boreholes is presented in the following documents:

1. "Small Diameter Horizontal Hole Drilling---State of Technology," SAND84-7103 (Robbins Company, 1984b).

2. "Feasibility Studies and Conceptual Design for Placing Steel Liner in Long, Horizontal Boreholes for a Prospective Nuclear Waste Repository in Tuff," SAND84-7209 (Robbins Company, 1985).

3. "Installation of Steel Liner in Blind Hole Study," SAND85-7111 (Kenny Construction Company, 1987).

4. "Design of a Machine to Bore and Line a Long Horizontal Hole in Tuff," SAND86-7004 (Robbins Company, 1987).

A detailed design for the drill and lining system has been completed. Current work involves fabrication and testing of a prototype of the drill, which will be followed by additional testing in G-Tunnel at the Nevada Test Site. This work is being performed under Issue 4.4 (preclosure design and technical feasibility).

6.4.8.3 Future work

The future work that is planned under Issue 2.4 will focus on (1) further development of tactics and corresponding schedules that ensure that the work required to support resolution of Issue 2.4 is clearly identified and is completed on a schedule consistent with that for the NNWSI Project; (2) refinement of retrieval conditions, both normal and off-normal; (3) refinement of design criteria; (4) continued review of the results from Issue 4.4 for the defined input items to ensure that the established performance goals are met; and (5) development of a report that assesses compliance concerns relative to retrievability.

Retrieval planning and strategy documents

Before development of the advanced conceptual design, the strategy for meeting the regulatory requirements and for implementing the requirements contained in the DOE position on retrieval and retrievability, will be described in a future NNWSI retrieval strategy document.

During advanced conceptual design, the tactics for implementation of the retrieval philosophy outlined in Section 2.4.4 of the SCP-CDR (SNL, 1987)will be defined in more detail and presented in the retrieval implementation plan. This plan will address requirements for design development, equipment development, demonstrations, retrieval condition evaluation, and supportive studies.

Retrieval conditions

In the future, the classification of retrieval conditions will use the probability-based approach presented in Figure 6-91. Additional development of design detail for equipment, especially for off-normal operations, and additional evaluation of potential conditions is required during ACD. (In future documents, including the ACD report, the term off-normal will be replaced with the term abnormal.) This will allow classification of potential conditions as normal, abnormal (expected and credible), or not credible. The classification of these retrieval conditions will use an identification concept similar to the one used to date for identifying items important to retrievability (Appendix L of SNL, 1987). The results of future work will be presented in two reports describing retrieval conditions. The first report will be generated during advanced conceptual design; a second report is planned in support of the license application design.

Retrievability input to repository design requirements

The repository design requirement (RDR) report will be periodically updated as additional or modified requirements are developed. Most of the retrievability-related changes are expected to result from additional definition of equipment concepts and retrieval conditions. The resulting modifications to the retrievability-related design criteria will be forwarded to the RDR authors for review and publication in periodic updates.

Retrievability compliance analysis

A preliminary report addressing the status of compliance with the retrievability requirement will be issued at the completion of advanced conceptual design. An additional report on compliance will be completed as part of the license application design.

6.4.9 ISSUE 4.2: NONRADIOLOGICAL HEALTH AND SAFETY

6.4.9.1 Introduction

The question asked by Issue 4.2 is

> Are the repository design and operating procedures developed to ensure nonradiological health and safety of workers adequately established for the resolution of the performance issues?

The complete discussion of the proposed strategy for the resolution of this issue is presented in Section 8.3.2.4 (nonradiological health and safety). Readers unfamiliar with the issue resolution strategy for this issue, should review the contents of Section 8.3.2.4 before continuing.

Issue 4.2 has been subdivided into three information needs, and the completed work has been identified under the associated information need. The completed work is numbered in a manner that corresponds with the information need numbering system. For example, 4.2.3-1 is the first completed work identified under the third information need of Issue 4.2. The information needs are the following:

Information Need 4.2.1 Site and performance information needed for design

This information need consists of (1) compiling the site characterization and performance assessment information identified in the other information needs under Issue 4.2 into a single integrated list and (2) reviewing the proposed site characterization program and performance assessment program to ensure that the actions required to obtain the site characterization information and to provide the required performance assessment information are incorporated in these programs. Work completed to date has focused on the safety aspects of excavation stability. The site characterization and performance assessment information needed for design is identified in Section 8.3.2.4.1.

Information Need 4.2.2 Potential nonradiological hazards to personnel

The repository design and operating procedures will be reviewed to determine nonradiological hazards to personnel. The risk to personnel from a given nonradiological hazard then will be determined by (1) calculating the probability of occurrence of the event identified as a personnel hazard, (2) determining the consequences of the event, and (3) multiplying the probability of an event occurrence and the consequences of the event. If the risk is determined to be unacceptable, a change will be incorporated in the repository design and the repository operating procedures. The work completed to date for this information need has focused primarily on the identification of events that could have hazards associated with excavation stability.

Information Need 4.2.3 Design measures for avoiding or mitigating hazards to personnel

Number	Description
4.2.3-1	The design analysis work describes the approach being used in the design of the underground openings and the underground ventilation system.
4.2.3-2	Other work describes the approach being used to develop design criteria that address worker nonradiological health and safety concerns.

Section 6.4.9.2 summarizes the completed work that is pertinent to the resolution of Issue 4.2. Section 6.4.9.3 identifies future work that will provide additional information for use in the enhancement of worker nonradiological health and safety.

6.4.9.2 Work completed

6.4.9.2.1 Design analysis work

Work has been completed in the following areas.

Underground openings

The concern in the design of the underground openings is that they are stable and usable for the life of the repository. For personnel safety, stability implies that (1) no localized rock fall of a size sufficient to cause serious personnel injuries will occur, and (2) no catastrophic failure of the openings that could block personnel access and egress will occur.

The analyses that are being made to ensure that the design results in underground openings that are stable and, thus, safe, is the same work that is done to support the conclusion that the openings can be designed, constructed, and used for the life of the repository using currently available technology. This work is presented in Sections 6.4.10.2, Technical feasibility - work completed and 8.3.2.5.7, Design analysis. The data required to do these calculations are specified in Section 8.3.2.5.1, Site Information needed for design. The conclusion drawn from this work is that the underground openings will be stable and usable for the life of the repository; this conclusion translates further into the conclusion that the drifts will be safe for repository workers. The results of these calculations are discussed in detail in Section 6.4.10.2.

Potential hazards to excavation workers

To evaluate the potential hazards to excavation workers at Yucca Mountain, excavation experience in the welded and nonwelded tuffs at the NTS that have been used for weapons-effect testing has been examined. The safety records show that in the past such excavations have been carried out with minimum adverse effects on worker safety. To assess the relative level of safety for tunneling operations at the NTS, the incidence rates for NTS operations can be compared with injury incidence rates for similar mining operations. Such a comparison was presented in Figure 6-27 of the environmental assessment for the Yucca Mountain site (DOE, 1986c). The industry category that is most similar to excavation conditions in the tuffs at NTS is the category of hard-rock metal mining. The data presented in the environmental assessment are based upon industry average data compiled by the National Safety Council and data for NTS operations compiled by Reynolds Electric and Engineering Co. (REECo), the DOE contractor for excavation operations. The data presented clearly indicate a significantly better safety record for NTS tunneling operations than is typical of industry practice. While the industry average incidence rate is lower now than it was 20 yr ago by a factor of about two, NTS operational safety record is still lower than the industry average by a factor of about three.

Specific excavation experience in G-Tunnel at the NTS is of interest because part of the G-Tunnel experience involves a welded tuff, the Grouse Canyon Member. Engineers and geologists familiar with excavation in the welded Grouse Canyon Member have expressed the opinion that the ground support that will be required in the Topopah Spring Member at Yucca Mountain is likely to be similar to that required in the welded Grouse Canyon Member. None of the accidents identified in a search of tunnel records could be considered to be caused by unstable ground, faulting, or other such geologically related conditions. This was observed to be consistent with the period between approximately 1965 and 1985 for NTS operational experience. The one accident that involved the falling of a piece of rock was the result of an oversight in barring down loose rock before support installation. The accident report in question indicates that this accident probably would not have occurred if the correct NTS mining practice had been followed (DOE, 1986c).

Faults and shear zones that could compromise the safety of repository personnel because of construction problems or water inflow are not expected in the primary repository area at Yucca Mountain. The design and layout of the underground facility is planned to minimize contact with portions of the host rock where minor faults and shear zones are identified. There is to date no indication that pressurized brine pockets, evidence of dissolution, or significant accumulations of water or toxic gases are present in the repository horizon.

Underground ventilation system

Other significant physical or chemical phenomena known to be associated with rock characteristics are related to ventilation-system design and worker safety. The temperature increases resulting from the emplaced waste are important in designing ventilation systems and in selecting the standoff distance between the drift and the emplaced waste. Excavations at the NTS show that explosive or other hazardous gases are not to be expected. Thus, the ventilation system will primarily control dust. Hazards associated with dust and hazards associated with naturally occurring radon released during rock excavation will be mitigated by supplying adequate flow volumes to meet safety requirements. Techniques already implemented in the uranium mining industry will be considered. The proper design and operation of a ventilation system based on current technology should readily mitigate dust and radiation concerns.

The results obtained from the analyses performed to verify that the ventilation of the underground repository facility could be accomplished using reasonably available technology are applicable in the resolution of Issue 4.2. These results are given in Section 6.4.10.2. The goal imposed on the design of the underground ventilation system is that adequate air must be supplied to the workers under the most extreme operational conditions. The analyses indicate that this goal can be achieved.

6.4.9.2.2 Other work supporting the conceptual design

The work, other than analyses, that has been completed consists primarily of identifying legislative and regulatory requirements governing worker safety. These requirements have been incorporated in the subsystems design requirements document (Appendix P of SNL, 1987), the design criteria for the repository. These requirements are summarized in Section 8.3.2.5.4 (repository design requirements). The operational plan, Section 8.3.2.5.3 (plan for repository operations), also is influenced by these legislative and regulatory requirements.

6.4.9.3 Future work

Underground opening stability

Basically, two things need to be done in relation to the design of the underground openings:

1. Further work needs to be done with the rock mass classification methods. This work should be directed at obtaining additional data from comparable underground facilities to increase the data base on which projections of stability and the design of the support systems can be based.

2. Additional work needs to be done to verify the codes used to analyze the performance of the underground openings. This verification work can be accomplished in two ways: (a) by monitoring demonstration openings driven in the exploratory shaft facility, and (b) by applying the analyses techniques to previously driven openings and comparing the analytical results with the observed results.

Underground ventilation systems

Additional work needs to be done to quantify the potential radon gas burden on the ventilation system. The particulates generated during construction and operation need to be qualified and quantified.

The required efforts identified here are discussed in more detail in Section 6.4.10.3.1.

6.4.10 ISSUE 4.4 PRECLOSURE DESIGN AND TECHNICAL FEASIBILITY

6.4.10.1 Introduction

The question asked by issue 4.4 is

> Are the technologies for repository construction, operation, closure, and decommissioning adequately established for resolution of the performance issues?

The complete discussion of the proposed strategy for resolution of this issue is presented in Section 8.3.2.5 (preclosure design and technical feasibility). Readers unfamiliar with the issue resolution strategy for this issue should review the contents of Section 8.3.2.5 before continuing.

Issue 4.4 has been subdivided into 10 information needs. Several of the information needs address the concept of reasonably available technology. Evaluations of whether satisfaction of an information need can be achieved using reasonably available technology will be based upon the 10 CFR Part 960 definition: Reasonably available technology means technology which exists and has been demonstrated or for which the results of any requisite development, demonstration, or confirmatory testing efforts will be available before license application.

The completed work has been identified under the associated information need. The completed work elements are numbered in a manner that corresponds with the information need numbering system. For example, 4.4.2-1 is the first completed work identified under the second information need of Issue 4.4. The information needs and completed work are the following:

Information Need 4.4.1 Site and performance assessment information needed for design.

This information need consists of (1) compilation of the site characterization and performance assessment information identified in the remaining Issue 4.4 information needs into a single integrated list and (2) the review of the proposed site characterization program and performance assessment program to ensure that the actions required to obtain the site characterization information and to provide the required performance assessment information are incorporated in these programs. This work has not been completed; the site characterization and performance assessment information needed for design is identified in Section 8.3.2.5.1.

Information Need 4.4.2 Characteristics and quantities of waste and waste packages needed for design.

Number	Description
4.4.2-1	The preliminary reference waste descriptions document (O'Brien, 1984) contains a description of the waste type and waste containers.
4.4.2-2	The reference nuclear waste descriptions document (O'Brien, 1985) contains a description of the waste being considered for disposal at the MGDS site.
4.4.2-3	Section 2.1 of the SCP-CDR (waste form and package) contains a summary description of the waste form and waste package used as a basis for conceptual design of surface and subsurface facilities.

Additional information on Information Need 4.4.2 is presented in Section 8.3.2.5.2.

Information Need 4.4.3 Plan for repository operations during construction, operations, closure, and decommissioning.

Number	Description
4.4.3-1	The operational procedures for receiving, packaging, emplacing, and retrieving waste describe these operations as they were understood at the beginning of the conceptual design phase.

4.4.3-2	Chapter 3 of the SCP-CDR (repository) provides an overview of the principal operations and functions that will be performed in the repository.

Additional information on Information Need 4.4.3 is presented in Section 8.3.2.5.3.

Information Need 4.4.4 Repository design requirements for construction, operations, closure, and decommissioning.

Number	Description
4.4.4-1	Chapter 2 of the SCP-CDR (bases for the SCP conceptual design) contains a documentation of the reference values and design assumptions used in the conceptual design of the MGDS facilities.

Additional information on Information Need 4.4.4 is presented in Section 8.3.2.5.4.

Information Need 4.4.5 Reference preclosure repository design.

Number	Description
4.4.5-1	The repository reference designs that formed the basis for the repository conceptual design are addressed.
4.4.5-2	Chapter 4 of the SCP-CDR (design description) presents a description and discussion of the conceptual design.

Additional information on Information Need 4.4.5 is presented in Section 8.3.2.5.5.

Information Need 4.4.6 Development and demonstration of required equipment.

Number	Description
4.4.6-1	The conceptual designs for the waste receiving and preparation equipment address the equipment necessary to receive the waste and prepare the waste for emplacement.
4.4.6-2	The conceptual designs for the waste emplacement and retrieval equipment address equipment necessary to transport the waste underground; emplace the waste; and, if directed, retrieve the waste.
4.4.6-3	The conceptual designs for the waste emplacement hole boring equipment are addressed.

Additional information on Information Need 4.4.6 is presented in Section 8.3.2.5.6.

Information Need 4.4.7 Design analyses, including those addressing impacts of surface conditions, rock characteristics, hydrology, and tectonic activity.

Number	Description
4.4.7-1	Structural, thermal, and thermomechanical analyses are addressed.
4.4.7-2	Ventilation analyses are addressed.
4.4.7-3	Hydrologic analyses are addressed.
4.4.7-4	Tectonic and seismic analyses are addressed.

Additional information on Information Need 4.4.7 is presented in Section 8.3.2.5.7.

Information Need 4.4.8 Identification of technologies for surface facility construction, operation, closure, and decommissioning.

This information need will review Information Needs 4.4.3 through 4.4.6 to determine if the technology used in the surface facilities design can be considered reasonably available technology.

Additional information on Information Need 4.4.8 is presented in Section 8.3.2.5.8.

Information Need 4.4.9 Identification of technologies for underground facility construction, operation, closure, and decommissioning.

This information need will review Information Needs 4.4.3 through 4.4.6 to determine if the technology used in the underground facilities design can be considered reasonably available technology.

Additional information on Information Need 4.4.9 is presented in Section 8.3.2.5.9.

Information Need 4.4.10 Identification of technologies for emplacement of seals for accesses, drifts, and boreholes.

This information need will review Information Needs 1.12.3 and 1.12.4 to determine if the technology used in the design and placement of seals can be considered reasonably available technology.

Additional information on Information Need 4.4.10 is presented in Section 8.3.2.5.10.

Summary information describing the computer codes used in the analyses supporting the completed work elements previously identified is contained in Table 6-34.

6.4.10.2 Work completed

The following sections summarize the completed work for Issue 4.4.

Table 6-34. Computer codes used in analyses for Issue 4.4[a] (page 1 of 4)

Code name	Author	Code location	Design parameter	Analysis description
VNETPC	J. McPherson	Mining Ventilation Services, Oakland, CA	Ambient mine ventilation requirements.	Simulates underground airflow distribution and calculates fan requirements and pressure loss.
CLIMSIM 2.0	J. McPherson	Mining Ventilation Services, Oakland, CA	Drift cooling requirements.	Given wet- and dry- bulb temperatures of inlet air, the code simulates the psychrometric and environmental conditions in the airways.
ASHSD	Ghosh, Wilson	University of California at Berkeley, CA	Seismic analysis of structure-soil system.	Finite element model.
PORFLOW-R	A.K. Runchal	ACRI, Los Angeles, CA	2-D analysis of vertical emplacement drift. Thermal modeling.	Finite-element heat transfer for 2-D nonlinear heat convection and conduction in a porous medium.
THERM 3D	A.K. Runchal	ACRI, Los Angeles, CA No documentation available	Thermal modeling and 3-D analysis of vertical emplacement drift.	Finite-element heat transfer for 3-D analysis.
TEMP 3D	M. Christianson University of Minnesota	Public domain	Thermal modeling and 3-D analysis of vertical emplacement drift.	Computer program for determining temperatures around single or arrays of constant or decaying heat sources. Based on closed form solution.
ABAQUS	Hibbit, Karlsson, and Sorenson, Inc.	Hibbit, Karlsson, and Sorenson, Inc.	Displacements and stresses around underground openings.	Finite-element program for linear and nonlinear structural analysis.
SANCHO	C.M. Stone R.D. Kreig Z.E Beisinger	Public domain	Displacements and stresses around underground openings.	Finite element program to compute quasistatic, large deformation, inelastic response of planar or axisymmetric solids.

Table 6-34. Computer codes used in analyses for Issue 4.4[a] (page 2 of 4)

Code name	Author	Code location	Design parameter	Analysis description
ADINAT	K.J. Bathe	MIT	Temperature surrounding underground openings, used in conjunction with ADINA.	Finite-element heat transfer program. 2-D, 3-D; for automatic dynamic nonlinear heat conduction; convective and adiabetic boundaries; constant or decaying heat source.
HEFF	B.G.H Brady University of Minnesota	Public domain	Effects of parameter uncertainty on drift stability.	Boundary-element stress analysis for 2-D thermoelastic analysis of a rock mass subject to constant or decaying thermal loading.
ADINA	K.J. Bathe	MIT	Displacements and stresses around underground openings.	Finite-element stress analysis program. 2-D, 3-D; elastic elastic/plastic ubiquitous joint model; accepts precalculated temperature history.
STRES3D	C. St John M. C. Christianson	University of Minnesota	Stress distribution for evaluation of usable emplacement area.	Computer program for determining temperatures, stresses, and displacements around single or arrays of constant or decaying heat sources.
LINED	C. St John J.F.T Agapito and associates.	Public domain	Liner integrity due to surrounding stress.	Static analysis of a tunnel with liner or damaged annulus.
BMINES	U.S. Bureau of Mines	Public domain	Rock bolt performance.	Computer program of analytic modeling of rock/structure interaction.
VISCOT	ONWI	Public domain	Displacements and stresses around underground openings.	Finite-element stress analysis program. Thermoviscoelastic, thermoviscoplastic; accepts precalculated temperature history.

Table 6-34. Computer codes used in analyses for Issue 4.4[a] (page 3 of 4)

Code name	Author	Code location	Design parameter	Analysis description
DOT	University of California	Public domain	Used in conjunction with VISCOT. Time/temperature history (input for other stress codes).	General purpose heat conduction code for both linear and nonlinear steady-state or transient heat analysis.
SPECTROM-11	RE/SPEC Albuquerque, NM	RE/SPEC No documentation available	Displacements and stresses around underground openings.	Finite-element stress analysis program. Elastic, elastic/plastic, ubiquitous joint model; accepts precalculated temperature history.
SIM	Parsons Brinckerhoff Quade & Douglas, San Francisco, CA	PBQ&D - Parsons Brinckerhoff Quade & Douglas,	Rock temperature around borehole for comparison with thermal limit of canister.	A linear superposition program for three dimensional heat conduction solutions with constant or decaying line heat sources.
COYOTE	D.K. Gartling Sandia National Laboratories	Public domain	Temperature around underground openings (input for other stress codes).	Finite element computer program for nonlinear heat conduction problems.
SPECTROM-41	D.K. Svalstad RE/SPEC for ONWI Albuquerque, NM	RE/SPEC	Temperature surrounding an underground opening.	Finite-element heat transfer program. Nonlinear heat conduction; convective and adiabetic boundaries; constant or decaying heat source.
SPECTROM-31	D.A. Labreche S.V. Petney RE/SPEC for SNL Albuquerque, NM	RE/SPEC	Displacements and stresses around underground openings	Finite-element stress analysis program. Large deformation, static and quasi-static response of planar and axisymmetric solids; thermoelastic/plastic, ubiquitous joint model, compliant joint model; accepts precalculated temperature history.

Table 6-34. Computer codes used in analyses for Issue 4.4[a] (page 4 of 4)

Code name	Author	Code location	Design parameter	Analysis description
JAC	J.H. Biffle Sandia National Laboratories	Public domain	Displacements and stresses around an underground opening.	Finite-element stress analysis program. Nonlinear quasi-static response of solids with the conjugate gradient method; elastic/plastic, ubiquitous joint model, compliant joint model; accepts precalculated temperature history.
FLUSH	J. Lysmer T. Udada C. Tsai H.B. Seed University of California at Berkeley, CA	J. Lysmer	Soil-structure interaction, license-application design.	2-D finite element soil/structure interaction program.
CLASSI	Luco and Wong University of California at San Diego, CA	Luco and Wong No documentation available	Soil-structure interaction, license-application design.	3-D soil/structure interaction analyses using frequency-dependent impedance functions.

[a]ACRI = analytic and computational Research, Inc.; MIT = Massachussetts Institute of Technology; ONWI = Office of Nuclear Waste Isolation.

6.4.10.2.1 Characteristics and quantities of waste and waste containers

4.4.2-1 Preliminary reference waste description

In the preliminary reference waste descriptions report, O'Brien (1984) describes the reference waste types and containers for the early stages of conceptual design of a radioactive waste repository being considered for location in the tuff formations at Yucca Mountain. An assessment of the effects of nonreference waste characteristics on repository design is included.

4.4.2-2 Reference nuclear waste description

In the reference nuclear waste description document, O'Brien (1985) describes the reference wastes to be used as a basis for the conceptual design of a geologic repository being considered for location in the tuff formations at Yucca Mountain. Waste characteristics and production rates are taken from a DOE guidance document (DOE, 1984b) and the SCP-CDR. This information is recast as waste receipt and emplacement schedules to be used in the design of repository facilities and equipment and as input to the timetable for repository development.

4.4.2-3 Characteristics and quantities of waste form and waste container

For the waste form and container, Section 2.1 of the SCP-CDR summarizes the characteristics and quantities of the wastes and waste packages that were used as a basis for the conceptual design of a geologic repository being considered for location in the tuff formations at Yucca Mountain.

6.4.10.2.2 Plans for repository operations

4.4.3-1 Operational procedures

For operational procedures for receiving, packaging, emplacing, and retrieving waste, Dennis et al. (1984c) was prepared for use by the designers of the surface and underground waste-handling facilities and equipment. The report describes the radioactive waste expected at the repository; the shipping casks and the facility casks; and the waste receiving, handling, packaging, transfer, and emplacement operations. Potential waste retrieval operations, also are discussed.

4.4.3-2 Principal operations and functions

Chapter 3 of the SCP-CDR provides an overview of the principal operations and functions that will be performed at the repository. These operations and functions include waste handling and emplacement, waste retrieval, mining, ventilation, and the equipment needed to perform these operations. Equipment and concepts requiring development are identified.

6.4.10.2.3 Repository design requirements

4.4.4-1 Reference values and design assumptions for conceptual design

Chapter 2 of the SCP-CDR documents the reference values and design assumptions used by the architect-engineer in the completion of the SCP-CD. Presented are the technical requirements and assumptions that are the bases for the repository design; the site constraints, assumptions, and data that affect the repository design or the approach to the design; and the reference geologic data used in the design.

6.4.10.2.4 Reference preclosure repository design

4.4.5-1 Repository reference designs forming basis of conceptual design

Repository reference designs are the basis for performance analyses, operational plans, costs estimates, and schedules. There have been three reference designs completed and documented for the proposed repository at Yucca Mountain:

1. Jackson, J. L., (compiler), 1984. "Nevada Nuclear Waste Storage Investigations Preliminary Repository Concepts Report," SAND83-1877, Sandia National Laboratories, Albuquerque, New Mexico.

2. SNL (Sandia National Laboratories), 1986. "Two-Stage Repository Development at Yucca Mountain: An Engineering Feasibility Study," SAND84-1351, H.R. MacDougall (compiler), Albuquerque, New Mexico.

3. SNL (Sandia National Laboratories), 1987. "Site Characterization Plan-Conceptual Design Report," SAND84-2641, H. R. MacDougall, L. W. Scully, and J. R. Tillerson (compilers), Albuquerque, New Mexico.

There is only one reference design for the repository at any one time. The current reference design for the proposed repository at Yucca Mountain is the SCP-CD. The other two designs just identified have been superceded by this design and are no longer used as a basis for supporting calculations, operational plans, or cost estimates. The two obsolete designs are identified here as a part of the chronological record of the design evolution for the repository. In addition, studies supporting these two designs are often referenced in the SCP-CDR. Additional information on the philosophy for reference repository designs is given in Section 8.3.2.5.5.

4.4.5-2 Conceptual design description

The design description is in Chapter 4 of the SCP-CDR, which describes the conceptual design of the repository, with emphasis on the excavations, facilities, systems, and equipment needed to perform the operations described in Chapter 3 of the SCP-CDR.

6.4.10.2.5 Development and demonstration of required equipment

4.4.6-1 Waste receiving and preparation equipment

For waste receiving and preparation, the equipment that must be developed is primarily the remote manipulation equipment required for unloading, packaging, welding, inspection, and decontamination. This equipment is not unique in that essentially all the required components have been built and demonstrated in other applications. The design task for the repository at Yucca Mountain is to (1) configure this equipment to perform the required tasks in a safe and effective manner and (2) determine that the resulting equipment set will be capable of accommodating the projected waste throughputs. The work completed to this time relative to the receiving and preparation activity is the development of conceptual designs and the analysis of worker environments. This work is documented in the following reports:

1. Dennis, A. W., 1983. "Design Considerations for Occupational Exposure for a Potential Repository at Yucca Mountain, High-Level Waste Handling Operations," SAND83-0247C, Sandia National Laboratories, Albuquerque, New Mexico.

2. Dennis, A. W., R. Mulkin, and J. C. Frostenson, 1984b. "Operational Procedures for Receiving, Packaging, Emplacing, and Retrieving High-Level and Transuranic Waste in a Geologic Repository in Tuff," SAND83-1982C, Sandia National Laboratories, Albuquerque, New Mexico.

3. Dennis, A. W., P. D. O'Brien, R. Mulkin, and J. C. Frostenson, 1984c. "NNWSI Repository Operational Procedures for Receiving, Packaging, Emplacing, and Retrieving High-Level and Transuranic Waste," SAND83-1166, Sandia National Laboratories, Albuquerque, New Mexico.

4. Dennis, A. W., J. C. Frostenson, and K. J. Hong, 1984a. "NNWSI Repository Worker Radiation Exposure, Volume 1, Spent Fuel and High-Level Waste Operations in a Geologic Repository in Tuff," SAND83-7436/1, Sandia National Laboratories, Albuquerque, New Mexico.

Current information supports a conclusion that the receiving and preparation tasks at the repository can be accomplished using reasonably available technology. The relevant considerations are the following:

1. The equipment design and operational procedures presented in the listed documents are based on commercially available equipment, such as manipulators, remote welders, and remote operated and programmable position cranes.

2. Operations are based on demonstrated procedures for handling of radioactive materials; such procedures include using shielding, placing operators in locations remote from the material being handled, and providing for remote replacement, repair, and maintenance of cell equipment.

3. The operations to be performed in receiving and preparing of the waste at the repository closely parallel the task of refueling nuclear power reactors; thus, the experience and the equipment developed to support refueling can be incorporated in the designs for receiving and preparation equipment.

4.4.6-2 Waste emplacement and retrieval equipment

For waste emplacement and retrieval, equipment is required for loading the waste at the surface facility, transporting the waste underground, emplacing the waste in the emplacement borehole, and retrieving the waste. The concepts for the equipment required to perform these activities are based on currently available technology (e.g., the waste transporter is based on the use of a commercially available mine haulage system that has been thoroughly tested in mining applications). Requirements imposed on the design of this equipment include the incorporation of backup actuators, provision for easy access to replace key components, and simplicity. These requirements are imposed to improve the reliability of the equipment for waste emplacement and retrieval and to minimize operational malfunctions from which recovery would be difficult and hazardous to personnel. The reports that document the work completed in support of the equipment development are as follows:

1. Stinebaugh, R. E., and J. C. Frostenson, 1987. "Worker Radiation Doses During Vertical Emplacement and Retrieval of Spent Fuel at the Tuff Repository," SAND84-2275, Sandia National Laboratories, Albuquerque, New Mexico.

2. Fisk, A. T., P. de Bakker, B. J. Doherty, J. P. Pokorski, and J. Spector, 1985. "Conceptual Engineering Studies and Design for Three Different Machines for Nuclear Waste Transporting, Emplacement, and Retrieval," SAND83-7089, prepared by Foster-Miller, Inc., Waltham, Massachusetts, for Sandia National Laboratories, Albuquerque, New Mexico.

3. Flores, R. J., 1986. "Retrievability: Strategy for Compliance Demonstration," SAND84-2242, Sandia National Laboratories, Albuquerque, New Mexico.

4. Stinebaugh, R. E., and J. C. Frostenson, 1986. "Disposal of Radioactive Waste Packages in Vertical Boreholes--A Description of the Operations and Equipment for Emplacement and Retrieval," SAND84-1010, Sandia National Laboratories, Albuquerque, New Mexico.

5. Stinebaugh, R. E., I. B. White, and J. C. Frostenson, 1986. "Disposal of Radioactive Waste Packages in Horizontal Boreholes--A Description of the Operations and Equipment for Emplacement and Retrieval," SAND84-2640, Sandia National Laboratories, Albuquerque, New Mexico.

The conclusion from the work completed to this point is that the equipment required to emplace and retrieve waste at the Yucca Mountain mined geologic repository can be designed and developed by integrating available technology. Development will be required to ensure that the integration of the available technology performs as required under the conditions for emplacement and retrieval.

4.4.6-3 Waste emplacement hole boring equipment

For waste emplacement and retrieval, the vertical emplacement boreholes can be bored using existing, commercially available drills with minor modifications. This conclusion is documented in a report by The Robbins Company (1984a). The boring of the holes for horizontal emplacement will require the development of a new drill. This drill, as it is currently designed, is based on technology used in tunnel boring machines (TBMs). The boreholes required for the horizontal emplacement concept are much smaller in diameter than the drifts produced by a TBM. Thus, it is necessary to scale down the components used for TBMs to build the waste emplacement hole drill. The horizontal emplacement hole drill, because of its smaller size and the desire to drill dry, has potentially unique problems such as cuttings removal, maintenance, and control. On the larger TBMs, personnel have direct access to the machine to correct problems occurring during boring.

The work that has been completed to this time in support of the design, development, fabrication, and testing of the horizontal hole drill is summarized by the following reports:

1. Robbins Company, 1984b. "Small Diameter Horizontal Hole Drilling--State of Technology," SAND84-7103, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

2. Robbins Company, 1985. "Feasibility Studies and Conceptual Design for Placing Steel Liner in Long, Horizontal Boreholes for a Prospective Nuclear Waste Repository in Tuff," SAND84-7209, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

3. Kenny Construction, Inc., 1987. "Installation of Steel Liner in Blind Hole Study," SAND85-7111, prepared for Sandia National Laboratories, Albuquerque, NM.

4. Robbins Company, 1987. "Design of a Machine to Bore and Line a Long Horizontal Hole in Tuff," SAND86-7004, prepared for Sandia National Laboratories, Albuquerque, New Mexico.

These reports document results of drill equipment market surveys, feasibility studies for liner installation methods, and preliminary retrieval techniques.

All work completed indicates that the drilling of the emplacement holes for waste at the Yucca Mountain repository can be performed with existing drilling systems or with adaptations of existing drill systems. The verification of this conclusion will require development tests in site-specific geology.

6.4.10.2.6 Design analysis

4.4.7-1 Structural, thermal, and mechanical analyses

The results from the structural, thermal, and thermomechanical analyses, that relate to evaluating the stability of underground openings and required ground support for the MGDS are synopsized in this section. The required ground support conditions identified by these analyses were determined to be well within the capabilities of available mine support systems. Results are presented from analyses using both computer codes and empirical approaches. The published reports that document the analyses discussed are the following:

1. Hustrulid, W., 1984a. "Lining Considerations for a Circular Vertical Shaft in Generic Tuff," SAND83-7068, Sandia National Laboratories, Albuquerque, New Mexico.

2. Hustrulid, W., 1984b. "Preliminary Stability Analysis for the Exploratory Shaft," SAND83-7069, Sandia National Laboratories, Albuquerque, New Mexico.

3. Hill, J., 1985. "Structural Analysis of the NNWSI Exploratory Shaft," SAND84-2354, Sandia National Laboratories, Albuquerque, New Mexico.

4. St. John, C. M., 1987a. "Interaction of Nuclear Waste Panels with Shafts and Ramps for a Potential Repository at Yucca Mountain," SAND84-7213, prepared by Agbabian Associates for Sandia National Laboratories, Albuquerque, New Mexico.

5. St. John, C. M., 1987b. "Investigative Study of the Underground Excavations for a Nuclear Waste Repository in Tuff," SAND83-7451, prepared by Agbabian Associates for Sandia National Laboratories, Albuquerque, New Mexico.

6. Johnson, R. L., 1981. "Thermo-Mechanical Scoping Calculations for a High Level Nuclear Waste Repository in Tuff," SAND81-0629, Sandia National Laboratories, Albuquerque, New Mexico.

7. Thomas, R. K., 1987. "Near Field Mechanical Calculations Using a Continuum Jointed Rock Model in the JAC Code," SAND83-0070, Sandia National Laboratories, Albuquerque, New Mexico.

8. Johnstone, J. K., R. R. Peters, and P. F. Gnirk, 1984. "Unit Evaluation at Yucca Mountain, Nevada Test Site: Summary Report and Recommendation," SAND83-0372, Sandia National Laboratories, Albuquerque, New Mexico.

9. Svalstad, D. K., and T. Brandshaug, 1983. "Forced Ventilation Analysis of a Commercial High-Level Nuclear Waste Repository in Tuff," SAND81-7206, Sandia National Laboratories, Albuquerque, New Mexico.

10. St. John, C. M., 1987d. "Thermomechanical Analysis of Underground Excavations in the Vicinity of a Nuclear Waste Isolation Panel," SAND84-7208, prepared by Agbabian Associates for Sandia National Laboratories, Albuquerque, New Mexico.

11. St. John, C. M., 1987c. "Reference Thermal and Thermal/Mechanical Analyses of Drifts for Vertical and Horizontal Emplacement of Nuclear Waste in a Repository in Tuff," SAND86-7005, prepared by J. F. T. Agapito and Associates, Inc., for Sandia National Laboratories, Albuquerque, New Mexico.

12. St. John, C. M. and S. J. Mitchell, 1987. "Investigation of Excavation Stability in a Finite Repository," SAND86-7011, prepared by J. F. T. Agapito and Associates, Inc., for Sandia National Laboratories, Albuquerque, New Mexico.

13. Ehgartner, B. L., 1987. "Sensitivity Analyses of Underground Drift Temperature, Stresses, and Safety Factors to Variation in the Rock Mass Properties of Tuff for a Nuclear Waste Repository Located at Yucca Mountain, Nevada," SAND86-1250, Sandia National Laboratories, Albuquerque, New Mexico.

14. Langkopf, B. S., and P. R. Gnirk, 1986. "Rock Mass Classification of Candidate Repository Units at Yucca Mountain, Nye County, Nevada," SAND82-2034, Sandia National Laboratories, Albuquerque, New Mexico.

15. Arulmoli, K., and C. M. St. John, 1987. "Analysis of Horizontal Waste Emplacement Boreholes of a Nuclear Waste Repository in Tuff," SAND86-7133, Sandia National Laboratories, Albuquerque, New Mexico.

16. St. John, C. M., 1985. "Thermal Analysis of Spent Fuel Disposal in Vertical Emplacement Boreholes in a Welded Tuff Repository," SAND84-7207, prepared by Agbabian Associates for Sandia National Laboratories, Albuquerque, New Mexico.

17. Dravo Engineers, Inc., 1984. "Effect of Variations in the Geologic Data Base on Mining at Yucca Mountain for NNWSI," SAND84-7125, Sandia National Laboratories, Albuquerque, New Mexico.

18. Zimmerman, R. M., M. L. Blanford, J. F. Holland, R. L. Schuch, and W. H. Barrett, 1986b. "Final Report G-Tunnel Small-Diameter Heater Experiments," SAND84-2621, Sandia National Laboratories, Albuquerque, New Mexico.

19. Zimmerman, R. M., R. L. Schuch, D. S. Mason, M. L. Wilson, M. E. Hall, M. P. Board, R. P. Bellman, and M. L. Blanford, 1986a. "Final Report: G-Tunnel Heated Block Experiment," SAND84-2620, Sandia National Laboratories, Albuquerque, New Mexico.

20. Zimmerman, R. M., 1983. "First Phase of Small Diameter Heater Experiments in Tuff," Proc. 24th U.S. Symposium on Rock Mechanics, College Station, Texas.

21. Zimmerman, R. M., M. L. Wilson, M. P. Board, M. E. Hall, and R. L. Schuch, 1985. "Thermal-Cycle Testing of the G-Tunnel Heated Block," Proc. 26th U.S. Symposium on Rock Mechanics, Rapid City, South Dakota.

22. Chen, E. P., 1987. "A Computational Model for Jointed Media with Orthogonal Sets of Joints," SAND86-1122, Sandia National Laboratories, Albuquerque, New Mexico.

23. Labreche, D. A., and S. V. Petney, 1987. "The SPECTROM-31 Compliant Joint Model: A Preliminary Description and Feasibility Study," SAND85-7100, Sandia National Laboratories, Albuquerque, New Mexico.

24. Bauer, S. J., R. K. Thomas, and L. M. Ford, 1985b. "Measurement and Calculation of the Mechanical Response of a Highly Fractured Rock," Proc. 26th U.S. Symposium on Rock Mechanics, Rapid City, South Dakota.

25. Labreche, D. A., 1985. "Calculation of Laboratory Stress-Strain Behavior Using a Compliant Joint Model," Proc. 26h U.S. Symposium on Rock Mechanics, Rapid City, South Dakota.

26. Thomas, R. K., 1982. "A Continuum Description for Jointed Media," SAND81-2615, Sandia National Laboratories, Albuquerque, New Mexico.

27. SNL (Sandia National Laboratories), 1987. "Site Characterization Plan Conceptual Design Report," SAND84-2641, H. R. MacDougall, L. W. Scully, and J. R. Tillerson (compilers) Sandia National Laboratories, Albuquerque, New Mexico.

Relevant information from these reports is summarized in the following paragraphs grouped according to structural features of repository.

Emplacement drifts

St. John (1987c) reports the results of two-dimensional finite- and boundary-element calculations for the emplacement drifts that include thermal effects out to 100 yr. The calculations are based on reference design information using currently available information about rock and site characteristics. The thermomechanical properties are presented in Chapter 2. The design basis is presented in Section 6.1.2 of the chapter. The thermal analyses were performed using the finite-element code DOT, and a second analysis used the boundary-element code HEFF. The HEFF code resulted in temperatures of within ±1°C of those predicted by DOT. Both codes used constant thermal and elastic properties. A mixture of 60 percent PWR and 40 percent BWR waste was modeled at an areal power density of 57 kW/acre. Because the analyses were two-dimensional with cross sections through the drifts, the heat sources (i.e., waste containers) were not explicitly represented along the axis of the drift. Rather, the heat sources were equivalently represented by a plane extending into and out of the modeled drift cross sections. Both vertical and horizontal emplacement drifts were analyzed using continuously ventilated and unventilated drift conditions. The intention was to bound the problem, realizing that actual effects of drift ventilation would fall somewhere between the two ventilation extremes modeled. Drift temperatures, although not directly related to drift stability, are important for assessing environmental conditions to which personnel may be subjected if cooling is not used. Maximum drift temperatures of 58 and 109°C resulted for the unventilated condition of the horizontal and vertical emplacement drifts, respectively. The maximum drift temperatures occurred at 100 yr after waste emplacement. The large difference in drift temperature for the unventilated horizontal and vertical emplacement is a result of the difference in standoff distance of the waste from the drift. The standoff distance for horizontal emplacement was simulated as 33 m. The standoff distance for vertical emplacement was simulated as 3.1 m. The ventilated condition placed the drift temperature at 30°C for both emplacement drifts analyzed over the 100-yr period. This is an estimate based on the 23°C in situ temperature and the likely inability of ventilation to maintain the drift at that temperature once the waste is emplaced. The actual drift temperature will likely fall within the broad range of the continuously ventilated and unventilated conditions.

Thermal results for two- and three-dimensional calculations of the vertical drift are documented by St. John (1985). The two-dimensional analyses were performed by the finite-element code PORFLOW, and the three-dimensional analyses were performed by the finite-element code THERM3D and the analytic solutions contained in the TEMP3D code. Nonlinear thermal effects were not modeled; the thermal decay of the waste was modeled and all other model input parameters were held constant throughout the 100-yr period analyzed. BWR containers at 3 kW each were spaced 4.0 m apart along the drift, 3.05 m below the floor. The analyses used a rectangular drift shape, and properties differed slightly from those of the reference information base. The temperatures resulting from the two-dimensional and three-dimensional codes differed little except in the immediate vicinity of the container, and the agreement between the analytic solutions and finite-element codes was excellent. Unventilated drift conditions resulted in a maximum temperature of 133°C at approximately 50 yr after waste emplacement. The drift temperature at 100 yr was only 4°C less than that at 50 yr. These results differ slightly from those presented by St. John (1987c). The differences are mainly because of differences in the decay characteristics of the waste used and the emplacement density along the drift. The initial source strength along the vertical emplacement drift in this analysis is 28 percent higher than that used in the analyses by St. John (1987c). The ventilated drift condition assumed the drift to be maintained at 30°C as did the St. John (1987c) analyses.

The thermal modeling discussed by Johnstone et al. (1984) for drift-scale analyses was used to establish the maximum areal power density (APD) for waste emplacement such that the drift floor temperature did not exceed 100°C for times up to 110 yr after waste emplacement. This criterion sets the maximum expected temperature and is used to design the cooling system necessary to prepare the drifts for reentry for purposes of inspection, repair, or retrieval. Note that this criterion has been superseded in the SCP-CD by temperature criteria established for access drift temperature in the vertical emplacement mode (Section 6.4.2). The analyses are based on the unventilated vertical emplacement drift. St. John (1987c) shows the unventilated drift exhibiting higher temperatures than the ventilated drift and the vertical emplacement drifts higher temperatures than the horizontal emplacement drifts. Therefore, the maximum temperature criteria, if applied to the horizontal emplacement of waste result, would result in a higher allowable APD. The nonlinear thermal analyses were performed using ADINAT and SPECTROM-41. The APD of the repository was established as 57 kW/acre for the Topopah Spring tuff.

The stress results reported in St. John (1987c) were obtained from the finite-element code VISCOT, using an elastic constitutive model. The stress analyses were performed at emplacement time and 100 yr later for the horizontal and vertical emplacement drifts, assuming both ventilated and unventilated drift conditions. The temperature results are reported in the previous paragraph. VISCOT uses the thermal field generated by DOT code analyses of temperature to calculate the induced thermal loading. The induced thermal stresses and in situ stresses are combined, and the stresses around the drift are computed. Knowledge of the stress state enabled the factors of safety against localized rock failure and activation of existing vertical joints to be assessed. The highest stresses were noted at the drift crown 100 yr after waste emplacement. The magnitudes of the principal stress in the drift crown ranged from 31 to 36 MPa for the horizontal emplacement drift, depending on the drift ventilation assumed. Higher stresses occurred for the unventilated drift condition. The vertical emplacement drift had crown stresses ranging from 13 to 54 MPa for the ventilated and unventilated conditions, respectively. The effect of ventilation is much more pronounced for the vertical emplacement option, where the waste packages are located much closer to the drift. The maximum stress magnitudes are well below the average unconfined compressive strength of 150.8 MPa measured in the laboratory. If a 50 percent reduction factor is applied to the average laboratory value of strength to account for scale effects (Appendix O of SNL, 1987), the minimum safety factor for the vertical emplacement drifts is 1.4. In this cluster analysis, the minimum safety factor calculated for the horizontal emplacement drift was 2.1. These safety factors are minimal because they are based on stresses at a point on the drift boundary. Stress magnitudes in this elastic analysis decrease for locations removed from the drift. This is illustrated in Figures 6-93 and 6-94, which plot the principal stress magnitudes and their directions for the vertical and horizontal waste emplacement drifts at excavation time and 100 yr later. The ventilated and unventilated drifts are shown in the figures. The safety factors corresponding to the stress levels plotted in Figures 6-93 and 6-94 are contoured in Figures 6-95 and 6-96 respectively. The safety factor contours show an increase in magnitude as distance from the drift crown increases. The mass of rock making up the crown area of the drift has an average safety factor much higher than the boundary values at the crown. The safety factor for the drift can be obtained by integrating or averaging the safety factor values over the crown region. The crown region is chosen because it has the lowest safety factor. Interpretation of Figures 6-97 and 6-98 result in an average safety factor for the crown region of the drifts which is greater than or equal to 3.0. This safety factor is considered conservative for the drift because the crown area will contain ground support such as rockbolts, a feature not modeled in the numerical analyses by St. John (1987c).

Johnstone et al. (1984) documents the results of inelastic vertical emplacement drift calculations using the ubiquitous-joint model for times out to 110 yr. These calculations are coupled to the thermal analyses performed by ADINAT and SPECTROM-41, as previously reported. No matrix fracturing

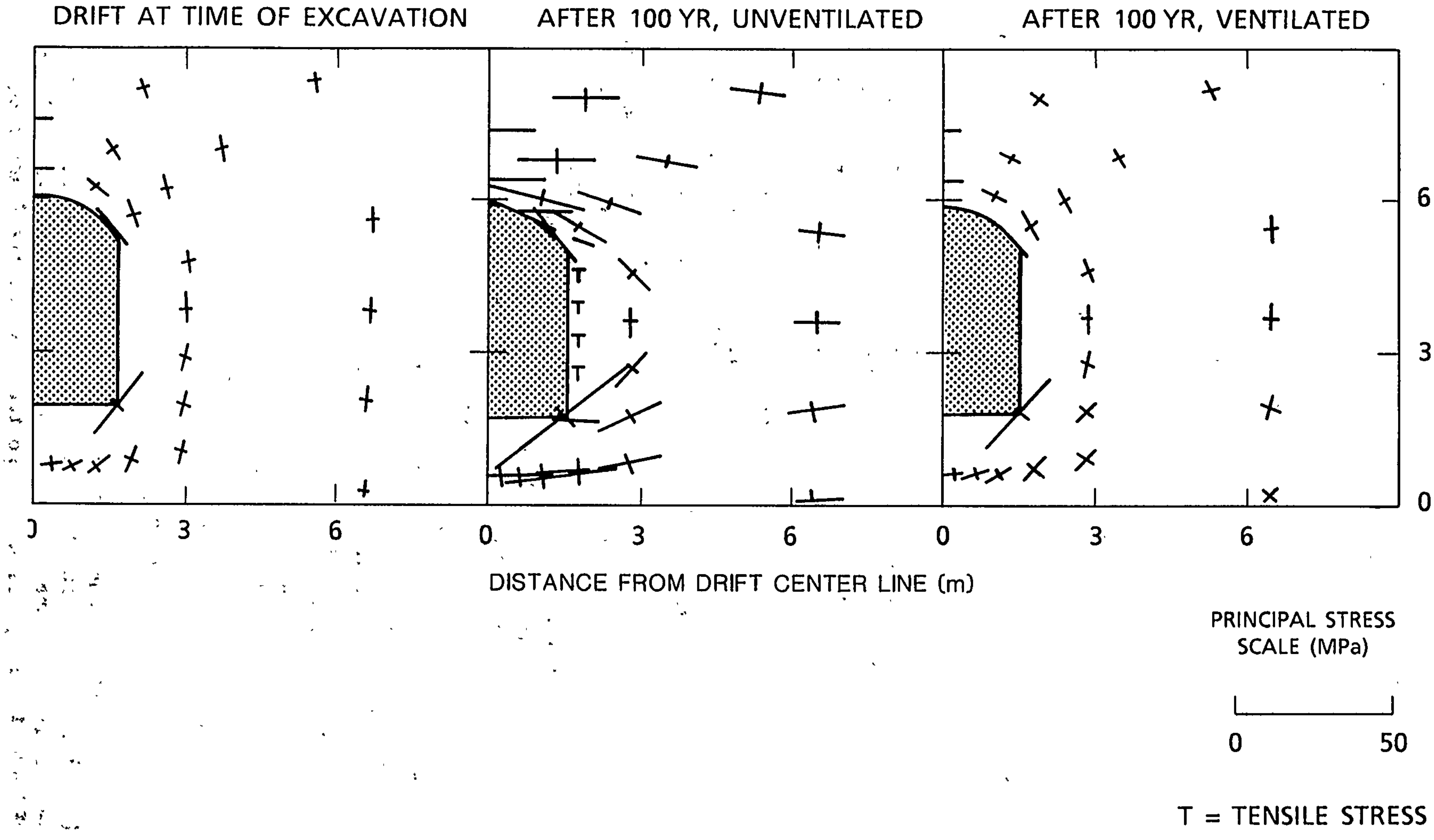

Figure 6-93. Finite-element predictions of the principal stresses in the vicinity of the vertical emplacement drift.

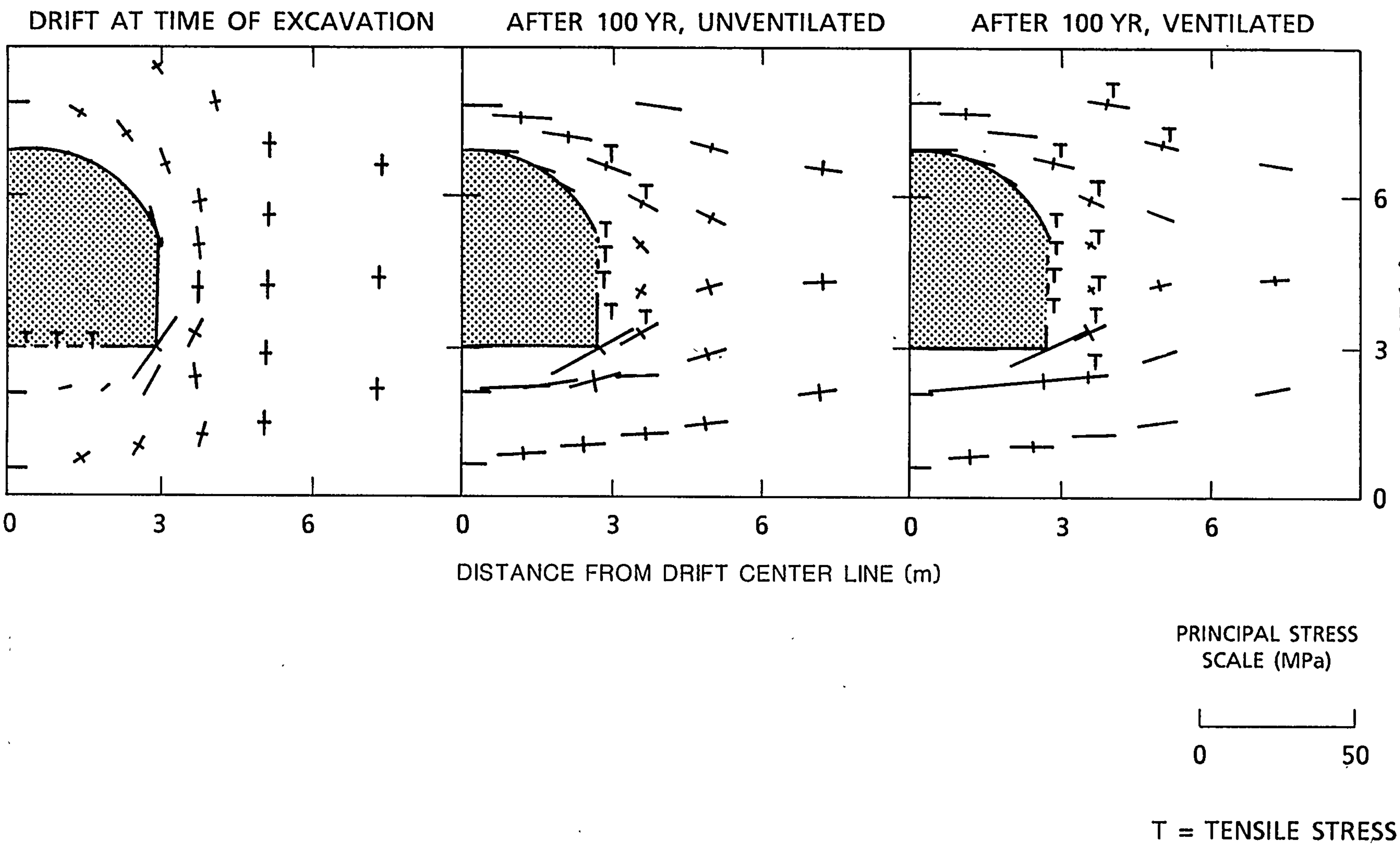

Figure 6-94. Finite-element predictions of the principal stresses in the vicinity of the horizontal emplacement drift.

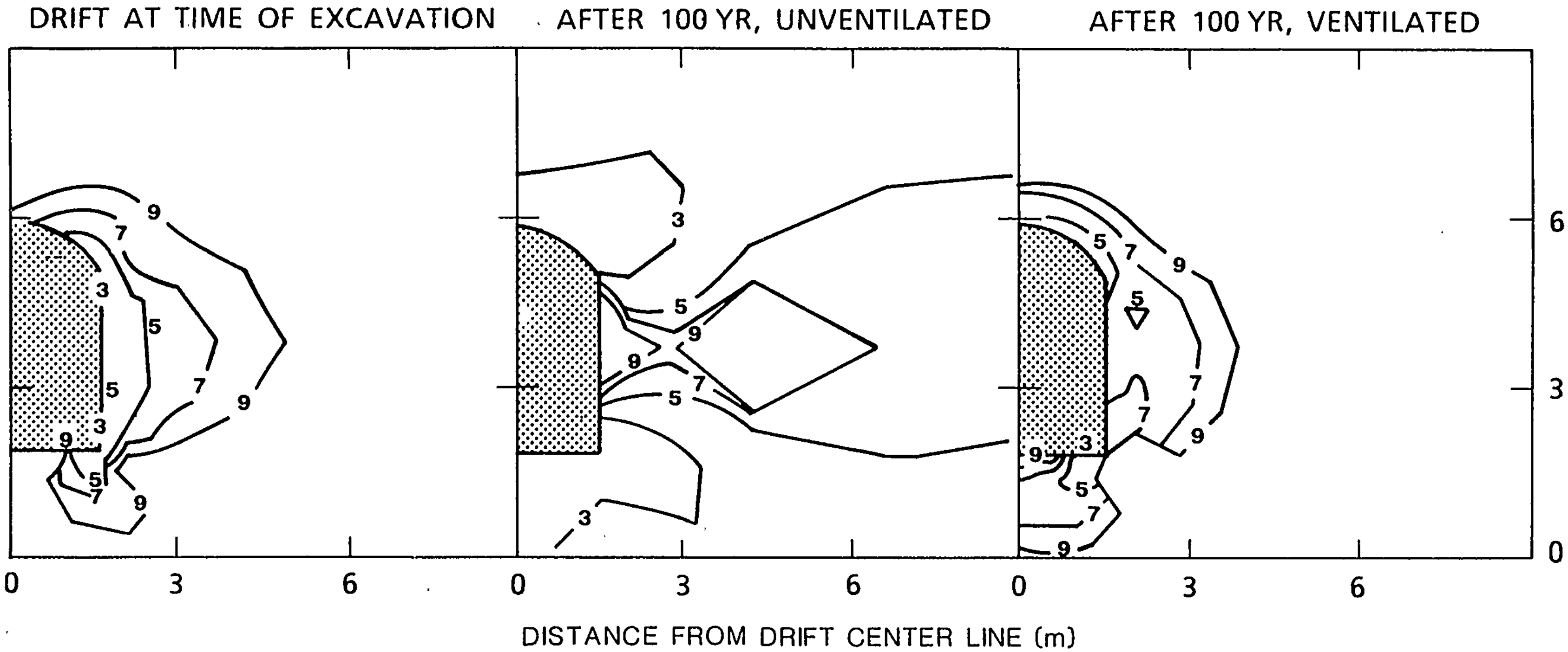

Figure 6-95. Finite-element predictions of the ratio between matrix strength and stress around the vertical emplacement drift. The numbers on the plots are ratios.

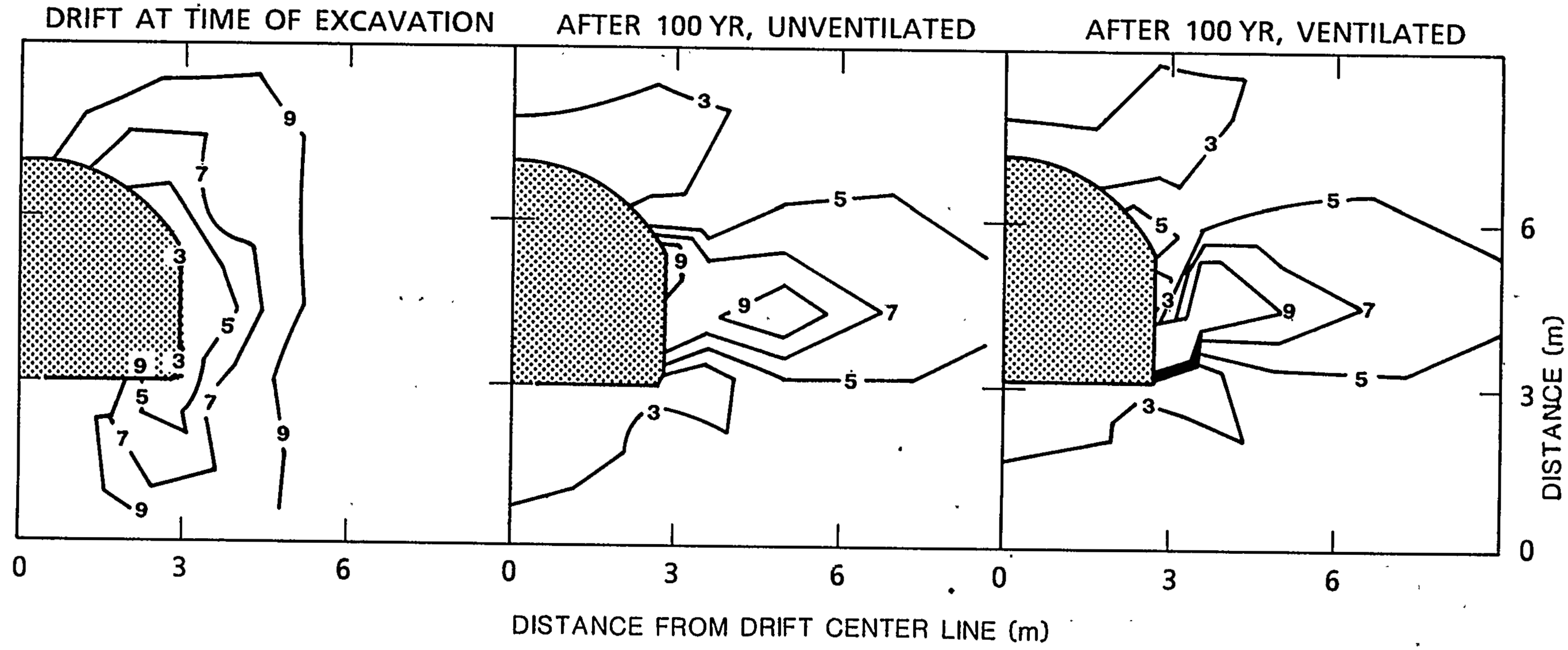

Figure 6-96. Finite-element predictions of the ratio between matrix strength and stress around the horizontal emplacement drift. The numbers on the plots are ratios.

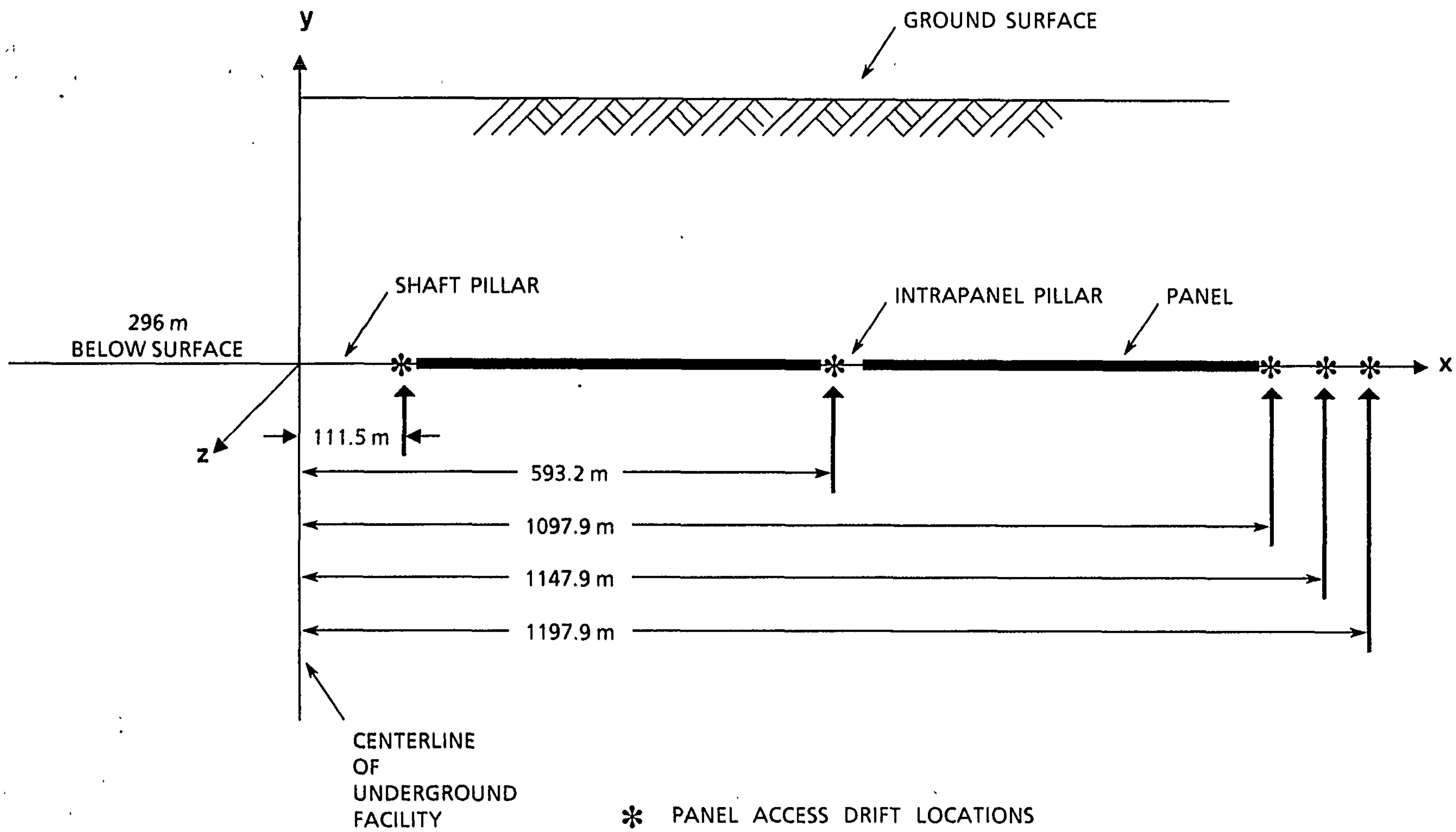

Figure 6-97. Repository cross section showing the access drift locations considered.

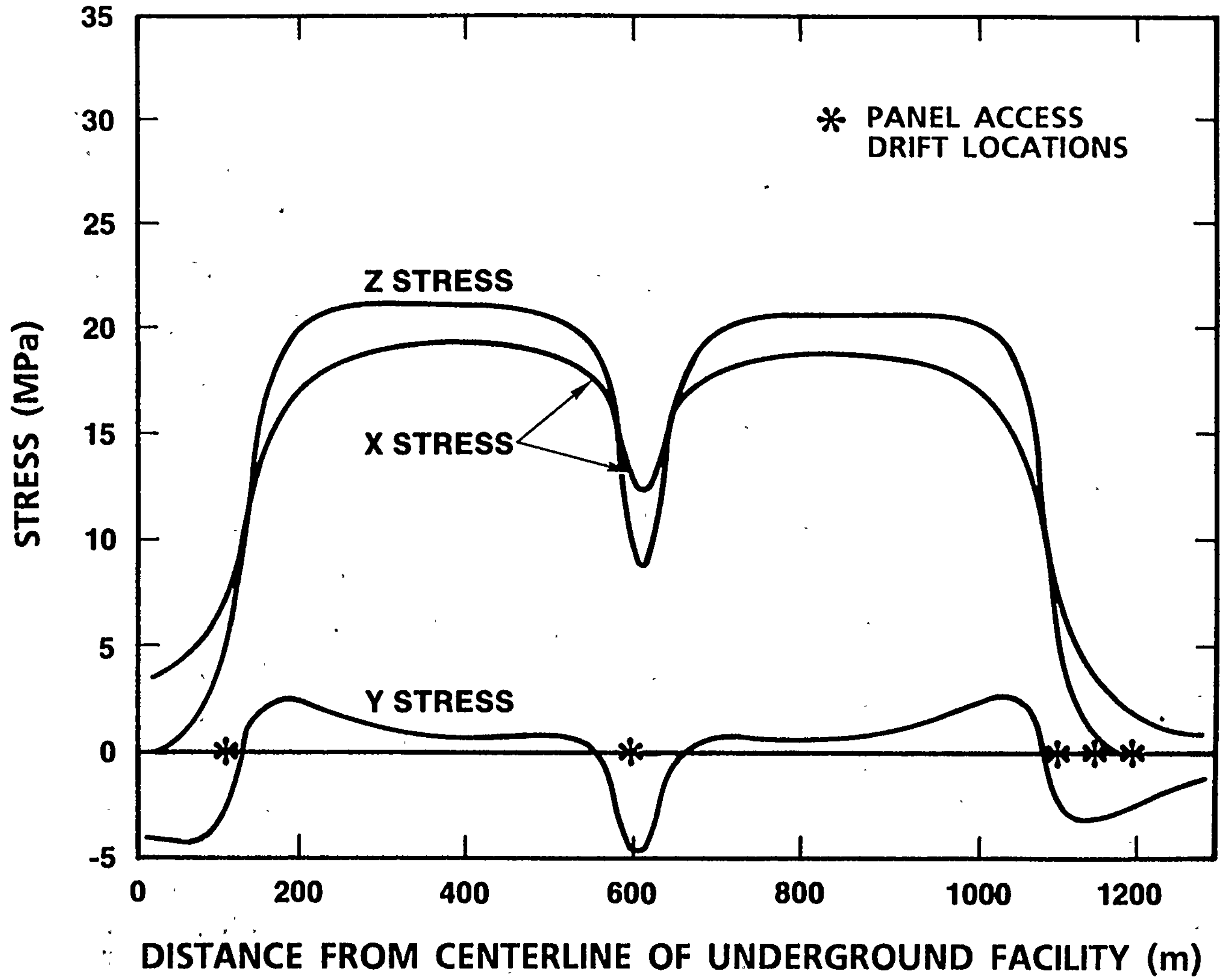

Figure 6-98. Induced stress profile on repository horizon 50 yr after waste emplacement.

occurred around the Topopah Spring drift for either the average or limiting property case. The limiting properties were taken as either plus or minus two standard deviations from average values, the sign being chosen on a worse-case basis. The elastic calculations presented by St. John (1987c) recognized the vertical emplacement drift as experiencing higher horizontal stresses than the horizontal emplacement drift for unventilated conditions. As such, the rock surrounding the horizontal drift should also remain intact when analyzed using the ubiquitous-joint model. The corresponding minimum safety factors were approximately 1.5 and 3 for the limiting and average case, respectively. Both the average and limiting safety factors indicate acceptable drift stability. Minor amounts of joint slip were noted; however, it is expected to have no consequence on drift stability--a conclusion supported by evidence from G-Tunnel. The ubiquitous-joint model predicts a slightly larger slip region for the rock surrounding G-Tunnel than for the repository drift, but no joint displacement is evident in the drifts of G-Tunnel. Limited amounts of vertical joint slip are predicted in the sidewalls of the drift both at and after waste emplacement.

Svalstad and Brandshaug (1983) report the effects of cooling a vertical emplacement drift after 39 yr of waste emplacement. The thermal portion of the analysis was performed using SPECTROM-41, and the mechanical portion used the elastic finite element code SPECTROM-11. An APD of 100 kW/acre was used to estimate the heat load in a 745-m drift. Blast cooling the rock for a year resulted in no change in the safety factor about the drift. Before and after cooling, the minimum safety factor was 2.0, and joint activation was limited to 2.0 m into the sidewall of the rectangular drift.

The results of the studies just reported indicate that the emplacement drifts will be stable and will provide a usable and safe environment for the retrievability period of approximately 84 yr. An additional positive factor in the long-term usability of the drifts is the requirement for drift support and periodic inspection and maintenance. The studies previously presented are based on thermal and mechanical properties considered to be representative of the rock mass (one exception to this is the analysis presented by Johnstone et al. (1984) that used limiting properties). The geometries and other model requirements reflected the anticipated design and environments at the time of the analyses. Because of the continual improvement in knowledge about the material properties and the design, an enlarged set of data was used in the analysis. However, the differences in the data used did not appreciably affect the results. This conclusion is based on the sensitivity studies presented below and the consistent prediction of drift stability by each of the studies.

Both general and specific sensitivity studies have been performed. Johnstone et al. (1984) documents drift conditions in not only the Topopah Spring but also in the underlying Bullfrog, Tram, and Calico Hills formations. Analyses were performed for times up to 100 yr after vertical waste emplacement for unventilated drift conditions. Thermomechanical properties for all the units differed. Johnstone et al. (1984) note that rock strength and modulus varied by a factor of three over the four units, but all units appear acceptable with regard to stability of the underground openings. Specific parameter sensitivities were investigated in Ehgartner (1987). The input parameters to the HEFF code were varied both individually and jointly to determine the effect on the horizontal emplacement drift at 50 yr. The results indicated that changes in rock strength and modulus affected the safety factors of the drift rock more than the other parameters that were varied, but in no case was the safety factor for the rock less than 1.0 over the probable range of input variables.

Ehgartner (1987) showed drift temperatures to be relatively insensitive to the thermal input variables. St. John (1987b) varied the shape of horizontal and vertical emplacement drifts over various in situ stress fields ranging from uniaxial to hydrostatic. The more rounded excavations had slightly lower stress concentrations. The elastic analyses used the boundary-element code HEFF and the elastic finite-element code, BMINES. BMINES enabled rock bolts to be included in the analyses. A damage region was modeled around the drift to simulate the impact of blasting during excavation, and rock bolts were inserted in the crown region. The calculated stresses in the rock bolts were approximately half the allowable strength of the rock bolts. The rock bolts had an insignificant impact on reducing drift closure or deformation, as compared to the unsupported drift analyses.

Johnson (1981) varied the APD for the vertical emplacement scheme to determine the effects on the drifts. The ADINAT and ADINA model incorporating ubiquitous jointing was used for analyses to 100 yr after waste emplacement. The two APDs of 75 and 100 kW/acre changed the maximum crown stress in the vertical emplacement drift by only 2 MPa. Drift temperatures were more affected by the APD change. The lower APD of 75 kW/acre resulted in a temperature of 98°C at the drift floor, whereas the 100 kW/acre APD increased the drift temperature to 107°C at 50 yr after waste emplacement. Johnstone et al. (1984) derived the maximum allowable APD for the Topopah Spring and the underlying Bullfrog, Tram, and Calico Hills units based on constraints for the vertical emplacement drift at 100 yr after waste emplacement. The APD determined for all the units including the Topopah Spring unit varied from 53 to 57 kW/acre.

In the previously mentioned sensitivity studies drift depth, shape, waste standoff, APD, and the thermomechanical properties were varied, and in each instance, it was concluded that drifts would remain stable. However, the stress ratio effects have not yet been examined.

Results of rock mass classification of the Topopah Spring tuff aid in estimating the ease or difficulty of constructing the drifts. Langkopf and Gnirk (1986) document the results of tunnel indexing or rock mass classification methods applied to the waste emplacement horizon of Yucca Mountain. Both the South African Council for Scientific and Industrial Research Classification System (CSIR) and Norwegian Geotechnical Institute Classification System (NGI) methods were applied. The CSIR gives an average standup time of 3 to 4 mo for an unsupported span of 6.1 m. The estimated standup times are based on a 6.1-m room or drift width, since this is typical in most mines. Drift widths for the vertical waste emplacement design vary from a minimum of 4.88 m (emplacement drift) to a maximum of 7.62 m for the mains. The horizontal waste emplacement design varies drift widths from a minimum of 6.40 m (access drift) to a maximum width of 7.62 m for the mains.

The NGI classification system estimates the maximum unsupported roof space from 2.3 to 9.9 m, the average being 6.0 m. The NGI system further qualifies the required support as ranging from grouted rockbolts on a 1-m spacing with chain-link mesh and shotcrete to a no-support requirement. The classification systems are based on the results of many diversified case studies, but a specific case to which anticipated repository excavation conditions can be related is found in G-Tunnel. The NGI and CSIR classifications systems both rank the welded Topopah Spring tuff and the nearby Grouse Canyon tuff almost exactly the same. This is because of the similarities in not only the geologic media but also in the in situ stress states. An underground facility (G-Tunnel complex) contains miles of drifts in a tuff unit known as Tunnel Bed 5 of the Grouse Canyon tuff; approximately 130 m of drifts in the Grouse Canyon tuff are in a welded tuff similar to the Topopah Spring unit. Drifts in this facility up to 9.2 m wide have been stable for periods up to 25 yr with a minimal amount of support.

The observations in G-Tunnel provide additional insight into the constructability and initial support requirements of the repository drifts. Even though faults and associated shear zones are expected to exist at Yucca Mountain, the preferred repository area is expected to be minable with standard equipment (Dravo Engineers, Inc., 1984). Rock with similar mechanical properties has been excavated at the G-Tunnel complex in Rainier Mesa using comparable methods of excavation and ground control.

The environmental assessment (DOE, 1986b) has documented tunneling experience in the welded Grouse Canyon Member at G-Tunnel. A nearly vertical fault with at least 1 m of vertical displacement was encountered during tunneling activities in the welded Grouse Canyon Member in G-Tunnel. No comments were noted by the mining inspector in his daily log; the lack of comments indicates that tunneling conditions had not varied appreciably. The fault zone was not noted until the tunnel had advanced about 6 m (18 ft) beyond the fault. The fault brought welded and nonwelded tuff together along a nearly vertical contact; no water influx was noted. The inspection record shows that the area of the tunnel with the fault was initially mined on November 19, 1981. Preliminary 2.5-m (8-ft) rock bolts were installed in the faulted area on November 20, 1981, and then on February 16, 1982, roughly 3 months later, 5-m (16-ft) resin-anchored hardening rock bolts were installed on a 1.3 by 1.3 m (4 by 4 foot) pattern across the back in the area adjacent to the fault. There was no record that the faulted area produced ground-support problems, and no special bolting was installed in the area of the fault. Crossing the nearly vertical fault with at least 1 m (3 feet) of vertical displacement did not result in the need for any special ground support in excess of the standard methods used in the drift where no faulting occurred. The observations at G-Tunnel and the results of rock mass classification apply to both emplacement drifts and access drifts. Panel access drifts are used between the repository mains and the emplacement drifts. The numerical analyses for the panel access drifts are discussed in the following paragraphs.

Panel access drifts

Results comparing panel access drift stability for various locations and standoff distances from the emplaced waste are documented by St. John and Mitchell (1987). The elastic two-dimensional calculations used the HEFF code for analyses of the horizontal emplacement scheme to 50 yr after waste emplacement at 57 kW/acre. Drifts were analyzed at locations in the central part and outer edges of the repository. The locations of the panel access drifts are shown in Figure 6-97. The hypothetical repository was configured of four panels, thus two of these are illustrated in the symmetric view presented in the figure. Interpanel locations also were considered. The thermally induced stresses (y stress) that correspond to Figure 6-97 are plotted in Figure 6-98. The induced stresses shown in Figure 6-98 are superimposed with the gravity induced stresses (x stress) to yield the total stress (z stress) to which the drifts are subjected. Analyses of the drifts indicated the results presented in Table 6-35 at 50 yr. A near hydrostatic in situ stress field was assumed. Although differences in results exist for the drifts at the various locations, no stability problems were identified at any of the potential locations.

The results of analyses on the intersection of the emplacement drift with a panel access drift are documented in St. John (1987d). The three-dimensional elastic calculations used STRES3D to generate the thermally induced stress field for the horizontal emplacement scheme and ADINA to elastically analyze the intersection. Stresses in the crown of the intersection reached approximately 23 MPa after 50 yr of waste emplacement. In this elastic analysis, tensile stresses approaching 9 MPa were predicted in the rib at the intersection. The tensile stresses dissipate 3 m into the rib. Note that the tensile stresses predicted in the elastic model will likely be reduced in the field because of the presence of existing horizontal fractures.

Table 6-35. Predicted stress, factor of safety, and temperatures of panel access drifts at different locations at 50 yr after emplacement

	Distance of drift from repository centerline (m)				
Parameter	115	606	1098	1148	1198
Crown stress (MPa)	36.0	48.6	34.0	24.1	19.1
Safety factor at crown	2.09	1.55	2.22	3.12	3.94
Drift temperature (°C)	56.7	71.3	55.8	28.5	23.6

Waste emplacement boreholes

The stability of waste emplacement boreholes must be evaluated to understand the loading that may be imposed on a waste package (or borehole liner) and to evaluate the environment anticipated for retrieval operations. The thermal and mechanical calculations and observations made in G-Tunnel testing related to small-scale heated borehole stability are discussed in a subsequent section on verification and validation results. The emphasis of that discussion is on model validation (i.e., the model represents the intended physical system or process). The emphasis here is to provide a synopsis of the calculations performed on the waste emplacement boreholes using configurations similar to those shown in Figure 6-97. In addition, the evaluations of potential loads on the emplacement borehole liner for the horizontal emplacement option are described briefly. The analysis results to date predict that the boreholes will be stable but that some uncertainty exists regarding whether there will be small (bounded by a few centimeters) regions where localized fracturing of the rock might occur.

Results of analyses on the horizontal waste emplacement borehole (Arulmoli and St. John, 1987) are discussed for (1) the elastic model, and (2) inelastic mechanical models. The thermal modeling used fixed thermal properties for the host rock. The calculations are based on conceptual design information and the expected rock and site characteristics. The two-dimensional finite-element calculations used boundary conditions that modeled an infinite series of boreholes of infinite length, and the thermal loading was imposed instantaneously. This approximation is considered appropriate since the focus is on very near-field effects and borehole loading rates would likely be a minor perturbation in the stresses imposed. The results reflect expected conditions in the central portion of the repository and are considered conservative near the outer regions of the repository.

The elastic horizontal borehole calculations were performed for times extending to 100 yr after waste emplacement using the DOT code with temperature constant properties and the VISCOT code with current thermomechanical properties and geometries. A maximum borehole wall temperature of 160°C occurred approximately 25 yr after the waste emplacement. Maximum borehole wall stresses ranged from 20 MPa at the sidewall to 50 MPa at the crown. Because of problem symmetry, the stresses are equivalent at the sidewalls and at the top (crown) and bottom of the borehole.

The inelastic horizontal borehole calculations used the JAC compliant-joint model, incorporating both single- and orthogonal-joint sets. The single-joint set was vertical with a strike parallel to the borehole axis. The orthogonal-joint set had both vertical and horizontal joints striking parallel to the borehole. The stresses predicted within a few centimeters of the borehole crown by the single-joint-set model were higher at the crown than those predicted by the elastic model. However, the sidewall stresses were lower for the joint model when compared to the elastic model results. The redistribution of stress is a result of joint slip in the sidewalls of the borehole and joint closure in the crown of the borehole. The joint closure is a result of increased horizontal stresses induced by the heat from the waste. The maximum borehole crown stress for the single vertical joint set was 96 MPa. This is nearly twice the magnitude of the elastic prediction of borehole crown stress.

An explanation of the difference in model predictions follows. The elastic analysis approximates the effects of joints through the use of a reduced elastic modulus meant to account for the deformation behavior of a jointed rock mass. The analyses using a compliant-joint model, on the other hand, use both the intact modulus for the rock matrix and a deformation modulus for the joints. Mechanically, as the vertical-joint set along the crown of the borehole closed, in the compliant-joint model the apparent modulus of the rock mass approached the modulus value of the intact rock.

The intact modulus used in the compliant-joint analysis was twice the rock mass value used in the elastic analysis. The thermally induced stresses are proportional to the modulus of the rock; higher crown stresses were predicted by the compliant-joint analyses. When the horizontal-joint set is included with the vertical-joint set, the horizontal joints along the crown slip, resulting in a stress redistribution and lowering the crown stress by approximately 20 MPa. The orthogonal-joint case is considered to be more representative of emplacement conditions; therefore, the higher stresses predicted by the JAC model using a single vertical joint set may be higher than would be expected to be observed underground.

In the analyses, the stress gradient is abrupt near the borehole crown. The result of this is a rapid decrease in the stress magnitude from the borehole boundary into the rock mass for all the calculations. At a distance several centimeters from the borehole boundary, the crown stresses predicted by the elastic model and those predicted by the compliant joint models are essentially the same. Thus, the higher stresses predicted by the joint models are considered a potential "skin effect." However, the localized high stresses predicted by the joint models exceed the rock mass strength. Little consequence of this small overstressed region in the crown of the borehole is expected because a liner that can withstand the loads potentially imposed by fallen pieces of rock is planned for use in all horizontal emplacement holes. Indeed, preliminary liner loading analyses indicate that such loading will not compromise the structural integrity of the horizontal borehole liner. Borehole liner loading analyses are documented in Appendix B of SNL (1987).

Shafts and ramps

St. John (1987a) contains analyses of a 6-m vertical repository access shaft at two different locations and an inclined repository access ramp. Elastic analyses of the 6-m shaft located it centrally in the repository within a central 200-m shaft pillar and alternatively at 100 m from the edge of the repository. The ramp was inclined at 10 degrees from the surface and intersected the repository at the edge of the 200-m shaft pillar. The analyses were time dependent, considering the thermally induced load up to 100 yr after waste emplacement. STRES3D generated a three-dimensional stress field of the repository superimposing both the in situ and thermally induced stresses. The stress field then was imposed on the circular shaft using LINED to calculate stresses for both the 0.5-m-thick concrete shaft liner and the rock mass surrounding it. The stress field also was used as input to the HEFF code to calculate stresses about the ramp. The alternative shaft locations at the center and edge of the repository showed slight differences, but in no instance was the rock mass surrounding it fractured because of the in situ or thermally induced loading. The concrete shaft liner was predicted to have approximately 3 MPa of tensile stress along its axis. The analysis assumed placement of the shaft in an elastic continuum, unreinforced concrete, and no expansion joints along the shaft. The transfer of the induced tensile stress from the rock mass to the liner will likely be moderate because of the presence of naturally occurring and excavation-induced joints in the rock mass surrounding the shaft liner. As shaft liner designs become more detailed, additional analyses are expected.

The ramp analyses contained in St. John (1987a) indicated no rock failure for the various cross sections analyzed along the length of the ramp from the surface to mid-repository. The minimum safety factor was 2.5, which corresponded to a maximum boundary stress of 31 MPa at 100 yr after waste emplacement.

Hustrulid (1984b) and Hill (1985) analyzed the structural stability of the exploratory shaft and facility, respectively. The analyses were time independent; therefore, the thermal effects of waste emplacement were not considered. The analyses showed safety factors greater than or equal to 3.0 for the underground facility. Both elastic three-dimensional and two-dimensional ubiquitous-jointing models were used in the ADINA code. Results were very similar between the two models. The shaft analyses implied no fracture potential for the rock at strengths corresponding to those of the Topopah Spring.

Hustrulid (1984a) considered rock units below the Topopah Spring--the Calico Hills, Bullfrog, and Tram units. Because the strength of the rock units underlying the Topopah Spring is generally lower, some rock failure was predicted but it was limited in extent. Concrete liner thickness of 0.41 and 0.30 m were recommended for the Calico Hills and Tram formations. The Bullfrog formation did not require a liner. Both elastic and plastic analyses were conducted, but no code was used. The analytic solutions are developed and applied in the text of the report.

These analyses indicate stability of both the ramps and shafts of the repository. Additional analyses are planned to evaluate seismic effects.

<u>Verification and validation results</u>

<u>Finite-element methods</u>. The finite-element method is planned to be the predominant method for license application design analyses. The status of work pertaining to qualification of models and codes in the context of verification (including benchmarking) and validation is that (1) a method of approach has been developed (Section 8.3.2.1.4, repository modeling) and (2) initial work in this area has been completed.

Part of code qualification includes the verification of the equation solver, which is the heart of a code. Two types of problems are used in verification. Problems with known analytical solutions are used to test the code's numerical solution methods. These numerical solution methods give an indication of the accuracy of the code and also may point out areas where the code may be in error. The second type of problems are hypothetical repository-type problems. These are used to determine whether the code can simulate interactions typical to the repository design. Frequently, verification of the equation solver is accomplished by comparing the computer-calculated response for a simple boundary value problem with closed-form analytic solutions. What follows is a compilation of the types of problems presented in the user's manuals for a selection of the computer codes being considered for use in design and performance assessment calculations. This compilation demonstrates the ability of the codes being considered to handle a wide field of analyses, including problems similar to those to be encountered in repository design and performance assessment.

ABAQUS is a general purpose finite-element computer program for linear and nonlinear structural analyses (Hibbitt, Karlson, and Sorenson, Inc., 1982). A theoretical development of the appropriate governing equations and a description of the numerical algorithms are presented along with a user's guide that includes several sample problems and the solutions. The sample problems, which include comparisons with closed-form analytic solutions, are briefly summarized as follows:

1. Analysis of a uniformly loaded, elastic-plastic plate is performed to serve two functions: to verify the coding of the rate-independent plasticity theory and to assess the accuracy of the time integration of this form of plasticity theory, especially in the form of nonproportional stressing.

2. Elastic analysis of the barrel-vault-roof problem is performed and is considered to be one of the standard shell-element convergence tests.

3. Elastic analysis of the pinched, open-ended circular-cylinder problem is performed and is considered to be one of the standard test cases used to evaluate the performance of shell-element formulations.

4. Elastic analysis of the cantilever beam to evaluate the accuracy of one of the beam elements in a single, large displacement case.

5. Analysis of the pressurization of a cylinder and a sphere are performed, with elastic and elastic-plastic material behavior. The structures are assumed to be quite thin, so that membrane analysis may be used to verify solutions obtained with the program. Further, the strains are quite large so that for the elastic-plastic cases, rigid-plastic analysis provides an accurate comparative result. The main purpose of the examples is to verify the capabilities of the axisymmetric shell elements at finite strains (Hibbitt, Karlson, and Sorenson, Inc., 1982).

SANCHO is a finite-element computer program designed to compute the quasi-static, large deformation, inelastic response of planar or axisymmetric solids (Stone et al., 1985). Finite-strain constitutive theories for plasticity, volumetric plasticity, and metallic creep behavior are included. A constant bulk strain, bilinear displacement isoparametric finite element is used for the spatial discretization. The solution strategy used to generate the sequence of equilibrium solutions is a self-adaptive, dynamic-relaxation scheme based on explicit central difference pseudo-time integration and artificial damping. A masterslave algorithm for sliding interfaces also is implemented. A theoretical development of the appropriate governing equations and a description of the numerical algorithms are presented along with a user's guide, which includes several sample problems and their solutions. The sample problems, which include comparisons with closed-form analytic solutions, are briefly summarized here:

1. The free thermal expansion of an infinite cylinder was included to demonstrate the input for a thermal stress problem and to demonstrate the ability of SANCHO to solve problems involving thermal loads.

2. The problem of an infinite cylinder loaded in the plastic range by an internal pressure serves as a good check of the elastic-plastic material model (Stone et al., 1985).

3. The stress relaxation of a single element is used to demonstrate the accuracy of the elastic creep model.

The last example problem is much more complex than the preceding examples and, therefore, relies on comparison with other finite-element programs for solution verification. The problem is a complex geotechnical analysis of an underground drift in a multilayered geologic medium, principally rock salt. It is characterized by creep and contains clay seams characterized with sliding interfaces and a friction coefficient of zero. Elastic anhydrite and polyhalite layers also are interspersed. The problem was specified as part of the Waste Isolation Pilot Plant (WIPP) Project code comparison activity called Benchmark II (Morgan et al., 1981). The problem involves determining the response of an infinitely long array of parallel drifts (Stone et al., 1985).

JAC is a finite-element computer program for solving large deformation, temperature-dependent quasi-static mechanics problems in two dimensions with the nonlinear conjugate gradient technique. Either plane strain or axisymmetric geometry assumptions may be used with material descriptions that include temperature-dependent elastic-plastic, temperature-dependent secondary creep and isothermal soil models. A four-node Lagrangian uniform-strain element is used with orthogonal hourglass viscosity control of the zero energy modes (Biffle, 1984). A theoretical development of the appropriate governing equations and a description of the numerical algorithms are presented along with a user's guide, which includes several sample problems and the solutions. The sample problems, which include comparisons with closed-form analytic solutions, are briefly summarized here:

1. The large deformation of an elastic cantilever beam is included since an analytical solution by Holden (1972) is available.

2. The plane strain crushing of a relatively thin tube in the diametrical direction by rigid platen represents a difficult elastic-plastic, large deformation and sliding surface problem for the finite-element technique.

3. The problem of the plane-strain extrusion of a plate (Biffle, 1984) is included to demonstrate a case where large amounts of sliding take place, along with elastic-plastic loading and unloading.

4. A laminated beam problem is used to demonstrate the behavior of multiple sets of slide lines. The problem is a simulation of the reaction of layers of material above a mine opening, as described by Sutherland et al. (1979).

5. A suddenly applied pressure is applied to the inside of a thick cylinder and the creep response is calculated (Biffle, 1984).

COYOTE is a finite-element program designed for the solution of two-dimensional, linear and nonlinear, steady- and transient-heat-conduction problems (Gartling, 1982). Available boundary conditions include constant temperature at a node, constant or time-dependent temperature along a side, adiabatic surface, forced convection, natural convection, and thermal radiation. Material properties (densities, specific heats, and conductivity tensors) may be dependent on temperature. A theoretical development of the appropriate governing equations and a description of the numerical algorithms are presented along with a user's guide, which includes several sample problems and their solutions. The sample problems, which include comparisons with closed-form analytic solutions, are briefly summarized here:

1. The problem of heat conduction in a steel bar (square cross section), with a circular hole that is subjected to prescribed temperature-boundary conditions, was performed.

2. To demonstrate the use of user subroutines for volumetric heating and generalized convection-radiation boundary conditions, a one-dimensional problem was considered. A cylindrical region of heat-generating material is encased by a thin layer of low-conductivity material and a thicker layer of material having a relatively high thermal conductivity. The outer surface of the cylinder loses heat to the surrounding environment by natural convection.

3. The finned tube radiator problem was chosen to illustrate the use of time-dependent boundary conditions.

4. Sensitivity analyses have been conducted using COYOTE and are reported by Branstetter (1983), Duffey (1980), and Gartling et al. (1981).

5. Duffey (1980) also reports the comparison of COYOTE's output with experimental results of a salt-block test. Good agreement was found for both the steady-state and transient conditions.

The SPECTROM codes solve for stresses around a repository using the finite-element method. Each of the codes can perform elastic and thermo-elastic analyses with loads due to a nodal temperature distribution, boundary stresses, and boundary displacement. A theoretical development of the appropriate governing equations and a description of the numerical algorithms are presented along with a user's guide that includes several sample problems and their solutions. The sample problems, which include comparison with closed-form analytic solutions, are briefly summarized here:

1. Analysis of a thin-walled cylinder subjected to a cooling of the interior and to an internal stress.

2. Analysis of an internally pressurized cylinder with Tresca yield criterion, compared with an analytic solution.

3. Analysis of a biaxially loaded plate with a central hole with Drucker-Prager yield criterion.

4. Analysis of a circular hole in a Mohr-Coulomb medium.

Another part of verification involves testing of components of specific material models and is often found in the detailed write-up of the material model. A general three-dimensional material model for regularly jointed media is presented by Thomas (1982). The model is composed of two parts: a continuum approximation based on average discontinuous displacements across jointing planes within a representative elementary volume, and a material constitutive description based on the linear behavior of the base material and nonlinear normal and shear behavior between jointing planes. The sample problems are briefly summarized here:

1. The dilatation response only was analyzed for a rock mass with a prescribed joint set (Thomas, 1982).

2. The shear response without coupled displacements was analyzed for a rock mass with a prescribed joint set (Thomas, 1982).

3. The shear response with coupled displacements was analyzed for a rock mass with a prescribed joint set (Thomas, 1982). The compliant-joint model was further verified through a comparison with a closed-form analytic solution and a similar compliant-joint model developed by RE/SPEC (Labreche and Petney, 1987). In the simulation, a rock joint specimen consisting of intact rock matrix and one set of joints spaced 5 m apart was inclined zero degrees from the horizontal. The excellent comparison of results is shown in Figure 6-99.

The compliant-joint model has been recently updated and modified to include a second set of joints (Chen, 1987). The sample problems, which include comparisons with closed-form analytic solutions, are briefly summarized:

1. The dilatation response only was analyzed for a rock mass with a prescribed joint set (Chen, 1987).

2. The shear response without coupled displacements was analyzed for a rock mass with a prescribed joint set (Chen, 1987).

3. The shear response with coupled displacements was analyzed for a rock mass with a prescribed joint set in an arbitrary orientation (Chen, 1987).

The results of these analyses were used to verify both the code (the equation solver) and the material model (numerical representation of the physics).

Another step in code qualification is the demonstration of the adequacy of the code and models through applications to problems in physical situations in rock. Examples follow of code-model applications to physical situations in real rock.

PROBLEM 1: 20 MPa CONFINING PRESSURE

	$G_s = 10^{10}$ MPa/m			$G_s = 10^{7}$ MPa/m		
θ	JAC	SPECTROM -31	ANALYTIC	JAC	SPECTROM -31	ANALYTIC
0°	0.166	0.166	0.166	0.166	NOT CALCULATED	0.166
30°	0.150	0.150	0.150	0.495	0.495	0.495
60°	0.122	0.122	0.122	0.467	0.466	0.467
90°	0.114	0.114	0.114	0.114	NOT CALCULATED	0.114

PROBLEM 2: LOW CONFINING PRESSURE

	$G_s = 10^{10}$ MPa/m			
	$\sigma_{cp} = 0$ MPa		$\sigma_{cp} = 1$ MPa	
θ	JAC	ANALYTIC	SPECTROM-31	ANALYTIC
0°	0.548	0.548	0.428	0.429

Figure 6-99. Compressive axial strain (%) at axial stress of 100 MPa (comparison of results of compliant-joint model (JAC), closed-form analytic solution, and a second compliant-joint model (SPECTROM-31)).

The ground support for the underground openings at the G-Tunnel underground facility are considered minimal (rock bolts and wire mesh) by rock mass classification ratings (Langkopf and Gnirk, 1986). The underground openings in welded and nonwelded tuff at G-Tunnel were the subject of finite-element analyses using the ubiquitous-joint model (Johnson and Bauer, 1987) and the compliant-joint model (Thomas, 1987). The models represent different approaches to modeling rock mass deformation. The details of accommodation of stresses and strains within the analyses performed by Johnson and Bauer (1987) and Thomas (1987) are different. Yet both models predicted stable openings at G-Tunnel, consistent with each other and with the observed physical situation at G-Tunnel. These calculation exercises provide a measure of credibility to the models-codes applied, which further justifies the concept that deformation of a rock mass may be represented by the combined deformation of the matrix plus fractures.

In a validation-type study, the mechanical response of thermally fractured granite was measured and calculated (Bauer et al., 1985b; Labreche, 1985). Analysis of the experimental results provides insight into the physical deformation of highly fractured rock, whereas the match between calculations and measurements (Figure 6-100) allows us to gauge the appropriateness of the numerical model and input parameters. In general, the calculated stress-strain behavior is in qualitative agreement with that measured. This agreement between measured and calculated response indicates a reasonable degree of validity in our modeling exercise for both the physical characterization and the numerical idealization.

Comparisons of measured and calculated thermal responses in welded and nonwelded tuff have been completed by Zimmerman (1983) and Blanford and Osnes (1987), respectively. As part of the experiment, observations were made of the borehole before and after the thermal cycling. No structural degradation was observed in the borehole. The comparison (Figure 6-101) of measured and calculated temperatures is rather good. This agreement between measured and calculated response indicates a reasonable degree of validity in the modeling exercise for both the physical characterization and the numerical idealization.

The models and codes proposed for use in thermal analyses (methods designed to model temperature-dependent heat conduction) have been subjected to numerous code qualification activities (Wart et al., 1984). Thus, the status of these codes is such that they are considered nearly ready for Level 1 analyses, pending a detailed review. The results of thermal analyses are used as input for thermomechanical analyses. Therefore, differences between results from thermal codes must be well understood.

Comparisons of measured and calculated thermomechanical response are in a preliminary ongoing phase. Initial results (Zimmerman et al., 1986a) of such a comparison are encouraging and further analyses currently are being pursued.

Calculations in support of site evaluation, repository design, and performance assessment require accurate estimates of the in situ stresses and the variability of the in situ stress state within Yucca Mountain. A modeling approach was developed to assist in understanding the in situ stress

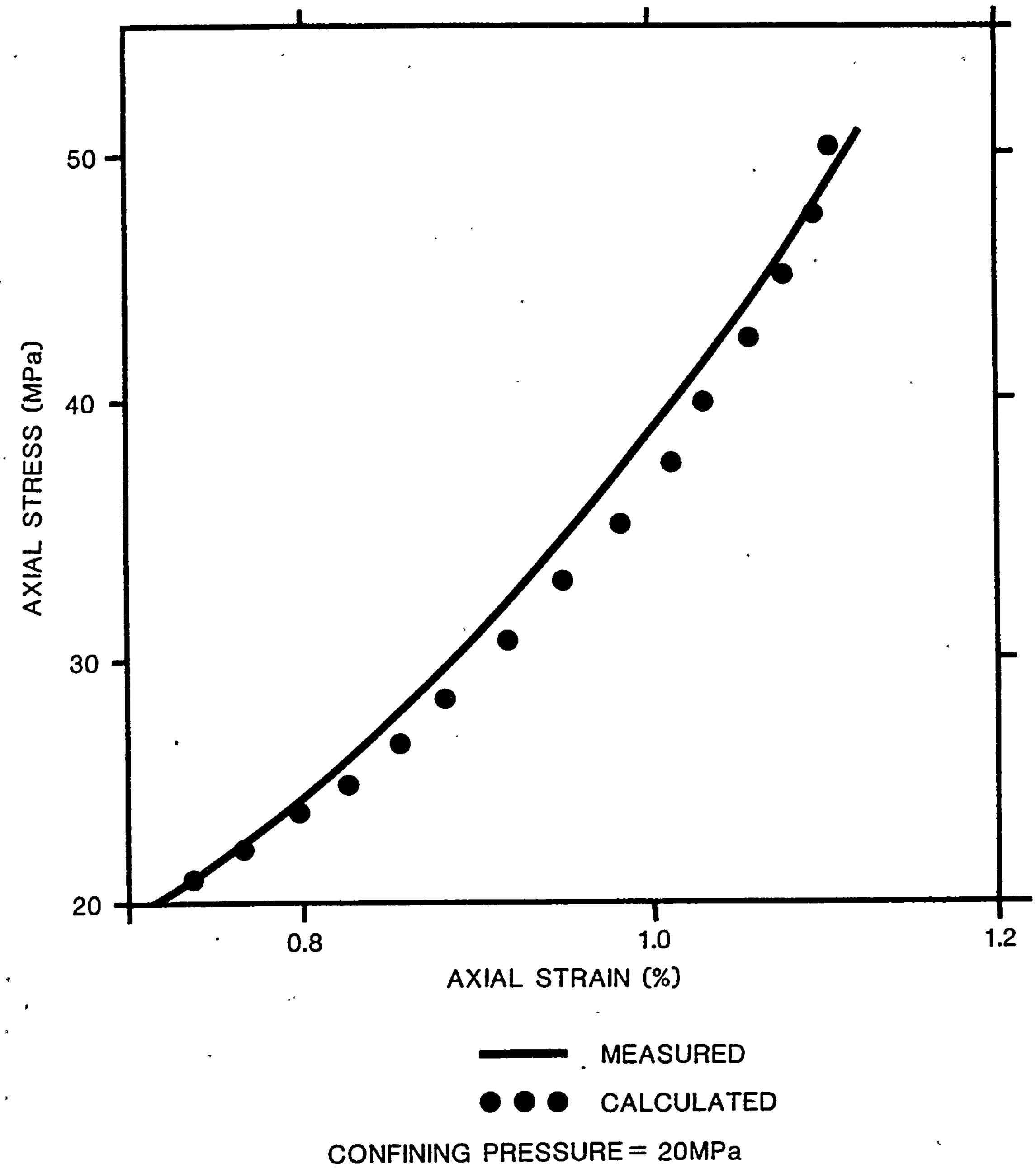

Figure 6-100. Measured versus calculated response for thermally cracked granite. Modified from Bauer et al. (1985b).

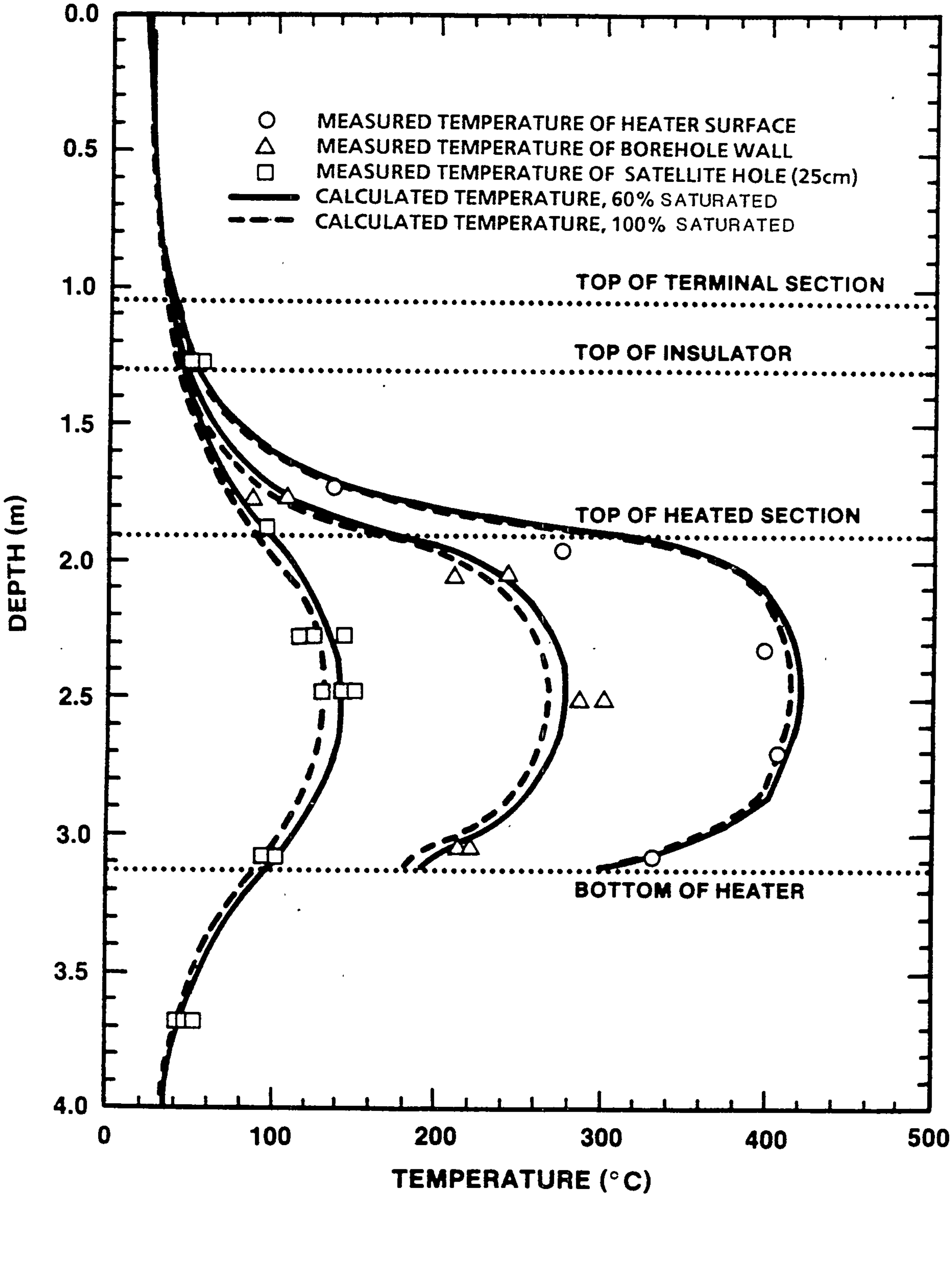

Figure 6-101. Comparison of measured and calculated temperature profiles for borehole subjected to thermal cycling. Modified from Blanford and Osnes (1987).

state at Yucca Mountain (Bauer et al., 1985a). The validation of this modeling approach is portrayed in the comparison of measured and calculated results. An analysis of the regional geologic studies that pertain to the stress state at the NTS, stress measurements in Yucca Mountain, and stress measurements in nearby Rainier Mesa was performed in conjunction with finite-element calculations to estimate the in situ stresses at Yucca Mountain (Bauer et al., 1985a).

Gravitational stress was the only loading mechanism modeled. Other assumptions used in the calculations were plane-strain conditions and linear-elastic material responses. The mechanical effects of pore pressure were not included except as pore water modifies the effective mass that induces gravitational loads. The validity of the approximations for estimating the in situ stress field were demonstrated through comparing the calculated stresses with carefully measured stresses at Rainier Mesa. Because of this favorable comparison, the same assumptions then were used for calculations that were compared with measured stresses at Yucca Mountain. The results indicate that topographic effects may result in spatial variations in the horizontal stress field at elevations of the proposed repository horizon. Also, by considering the vertical variation in mechanical properties (mechanical stratigraphy), deviations from a smooth increase in horizontal stress with depth are predicted. The combination of tectonic setting, measured values of in situ stress, and finite-element approximations of the stress state at Yucca Mountain provide a basis for estimating the lateral earth-stress coefficient that should be used in design and performance assessment calculations (Bauer et al., 1985a).

Boundary-element method. Boundary-element methods used here (e.g., Brady, 1980) are analysis methods for plane-strain thermoelastic analyses that may be applied to many problems posed by the emplacement of heat sources in a conductive stressed medium. The effects of excavation and heat may be included in the analysis. Rock strength and fracture slip may be used to interpret results. These methods allow for efficient parametric studies to be undertaken in which design tradeoffs and potential effects of parameter sensitivities to design can be evaluated through the predictions of deformations and stress at specific locations as a function of time. The usefulness of boundary-element methods will be qualified through comparison with closed-form analytic solutions, finite-element codes, and underground exploration because their applicability generally is limited to the preliminary phases of design.

Boundary-element methods are a relatively new geotechnical tool for thermal, mechanical, and thermomechanical analyses; yet the methods have been accepted for fairly wide use in underground stability analyses (Hoek and Brown, 1980).

The program HEFF is the FORTRAN code of an indirect formulation of the boundary-element method for plane-strain thermoelastic analysis. A theoretical development of the appropriate governing equations and a description of the numerical algorithms are presented in (Brady, 1980) along with a user's guide that includes several sample problems and their solutions. The sample problems, which includes comparisons with closed-form analytic solutions, are briefly summarized here:

1. Analysis of a circular hole in an elastic continuum, subject to plane strain, was performed and provides a convenient test of the segments of the code concerned only with solution of elastostatic problems.

2. Analysis was performed concerning the two-dimensional thermal stresses in an infinite medium subjected to an exponentially decaying line heat source perpendicular to the plane of analysis (Brady, 1980).

Tunnel index methods. Tunnel index methods (Barton et al. 1974a, b; Bieniawski, 1974) are empirical methods of classifying the rock mass through which ground support recommendations may be obtained. The tunnel indexing methods will be qualified through further studies of case histories specific to the repository in tuff, demonstrations in the exploratory-shaft facility, and checking against the implications of predictions made using boundary-element and finite-element analyses. The methods were each developed through an extensive study of case histories of underground openings in many types of rock, including tuff. The methods, Norwegian Geotechnical Institute Classification System (NGI) and the South African Council for Scientific and Industrial Research Classification System (CSIR) (Bieniawski, 1974), have been used to classify tuff rock masses at Yucca Mountain and G-Tunnel (Langkopf and Gnirk, 1986). These classifications have been used to recommend ground support for the repository openings (Dravo Engineers Inc., 1984). The methods were developed for a wide variety of rocks and currently do not incorporate the effects of heat in the considerations for their ground support recommendations.

4.4.7-2 Ventilation analyses

Results of the ventilation calculations completed to this time are discussed below. Both vertical and horizontal emplacement orientation have been analyzed. The SCP-CDR provides the basis for these discussions. Details are provided in Appendix C of the SCP-CDR.

Mine ventilation calculations for normal conditions

The approach to the design of an underground ventilation system starts with the determination of the required air quantities based on applicable regulations. For the repository, the selected regulation for compliance consists of the Mine Safety and Health Administration (MSHA) regulations (30 CFR Part 57) and the California Administrative Code. The ventilation criteria obtained from these documents are (1) from MSHA, the requirement to supply air for the dilution of diesel exhausts will be met by providing 125 cfm per brake horsepower of diesel equipment at a working location, and (2) from the California Administrative Code (Title 8, Division of Industrial Safety, Subchapter 17, Article 31) the minimum air velocity for work areas based on area cross section is 60 ft/min (18 m/min) and the minimum air supplied per worker is 200 cfm.

The typical emplacement drift for vertical emplacement is 16-ft (5m) wide and 21.5-ft (6.5 m) high, the maximum crew size in the drift during emplacement is 6 people, and the diesel-electric waste transporter is rated at 300 horsepower. Applying the requirements stated in the previous paragraph to the vertical emplacement system results in the following air quality requirements:

Air required to dilute diesel exhaust	37,500 cfm
Air required for 60 ft/min (18 m/min) velocity	20,640 cfm
Air required for crew (200 cfm per crew member)	1,200 cfm

The controlling requirement is the 37,500 cfm requirement for the dilution of exhausts. The design of the ventilation system is based on supplying 45,000 cfm to each active emplacement drift. The excess will provide the flexibility to compensate for changes in design requirements and philosophy and to provide for errors resulting from the estimates made for parameters, such as the roughness of mined surfaces and leakage.

The design accomplishes the goal of preferential leakage from the development (mining) side to the waste emplacement side of the repository by maintaining the pressures in the waste emplacement side lower than the development side. The pressure differential is accomplished by designing the waste emplacement ventilation system as a pull system with the suction fans located at the exhaust point and designing the development side ventilation system as a push system with the fans located at the air-intake point. The results of a point-by-point pressure calculation illustrate that this preferred pressure differential is maintained throughout the repository. These results are given in Section 3.4 of the SCP-CDR.

Table 6-36 presents the results of the normal mine ventilation calculations for horizontal and vertical emplacement under the two different scenarios. The results present the required fan airflow quantities and pressure heads to meet or exceed the criteria and boundary conditions of the problem, as described in the preceding approach and data sections. Because the mining and emplacement systems are separate, the fan quantities are given for each of these ventilation subsystems.

The quantities in Table 6-36 will be handled by the men-and-materials shaft as the intake and the tuff ramp as the exhaust for the mining ventilation system. The waste emplacement system uses the exploratory shaft and waste ramp as the intake and the waste ventilation exhaust shaft for exhaust. A more complete description of the ventilation system is provided in Section 6.2.6.3. The quantities required are presented in Table 6-36 to illustrate that the total air required is well within the capabilities of available mine ventilation equipment and is typical of the quantities required by conventional mines.

<u>Drift cooling calculations</u>

The required cooling times and loads calculated for the vertical and horizontal waste emplacement orientations are given in Table 6-37. Each orientation considers the alternatives of cooling the intake air or using ambient air. The results give the time required to achieve the conditions specified for the two different levels of activity (i.e., inspection and light maintenance, and heavy maintenance or retrieval). The calculations start with the conditions expected 50 yr after waste was emplaced.

Table 6-36. Maximum ventilation airflow requirements for two ventilation scenarios

Emplacement mode	Maximum development-side airflow scenario Vertical	Maximum development-side airflow scenario Horizontal	Maximum emplacement-side airflow scenario Vertical	Maximum emplacement-side airflow scenario Horizontal
Development side airflow (cfm)[a]	411,800	281,300	209,400	117,200
Emplacement side airflow (cfm)	481,300	446,400	837,200	517,200

[a]cfm = Cubic feet per minute.

Table 6-37. Cooling requirements for vertical and horizontal emplacement using ambient and conditioned air

Emplacement method	Inspection purposes (ACP[a]>300 w/m^2, T_{dry}<45°C)				Maintenance/partial retrieval (ACP>500 w/m^2, T_{dry}<40°C)			
	Ambient inlet temperature		Cooled inlet temperature		Ambient inlet temperature		Cooled inlet temperature	
	Time to cool (days)	Cooling load (kW)	Time to cool (days)	Cooling load (kW)	Time to cool (days)	Cooling load (kW)	Time to cool (days)	Cooling load (kW)
Vertical	168	0	21	708	>560	0	37	708
Horizontal	14	0	5	490	70	0	11	490

[a]ACP = Areal cooling power.

Table 6-37 shows that, for the vertical emplacement orientation, cooled inlet air will be required in all instances for the expedient cooling of the drifts in preparation for reentry. The table shows that ambient air can be used to cool the horizontal emplacement drifts for reentry to perform inspection, but for reentry to perform major maintenance or retrieval, cooled air also may be required if a cool-down time of 70 d is not acceptable.

4.4.7-3 Hydrologic analyses

The flood hazard for a 14-km (9-mi) reach of Fortymile Wash and its principal southwestern tributaries--Busted Butte, Drill Hole, and Yucca washes--were evaluated (Squires and Young, 1984). Data from 12 peak-flow gaging stations adjacent to the Nevada Test Site were used to develop regression relations that would permit an estimation of the magnitude of the 100- and 500-yr flood peaks.

Among seven cross sections on Fortymile Wash, the estimated maximum depths of the 100-yr, 500-yr, and regional maximum floods are 2,3, and 9 m (8, 11, and 29 ft), respectively. At these depths, flood water would remain within the deeply incised channel of the wash. Near flow velocities would be as great as 3, 4, and 9 m/s (9, 14, and 28 ft/s) for the three respective flood magnitudes.

The study shows that Busted Butte and Drill Hole washes (9 and 11 cross sections, respectively) would have water depths of up to at least 1 m (4 ft) and mean flow velocities of up to at least 2 m/s (8 ft/s) during a 100-yr flood. A 500-yr flood would exceed stream-channel capacities at several places, with depths of 10 ft and mean flow velocities at 3 m/s (11 ft/s). The regional maximum flood would inundate sizable areas in the central parts of the two watersheds.

At Yucca Wash (5 cross sections), the 100-yr, 500-yr, and regional maximum floods would remain within the stream channel. Maximum flood depths would be about 1.5, 3, and 7 m (5, 9, and 23 ft) and mean velocities about 3, 4, and 7 m/s (9, 12, and 22 ft/s), respectively, for the three floods.

The results of this study were considered in the siting of the surface facilities and in the conceptual design of the flood protection features. No explicit design calculations were performed to support the conceptual design of flood protection features. Where needed, the features (dikes, channels, etc.) were identified in the conceptual design. The final design of these features will be developed during the license application design phase and will be based on the site-specific probable maximum flood (PMF) analysis.

This flood hazard calculation is preliminary partly because the surface facility locations are conceptual and partly because some regional rather than site specific data had to be used in the supporting calculations. Regional data provides the most accurate flooding potential data that can be obtained in advance of acquisition of site-specific data through site characterization. The report documents preliminary calculations of discharge volumes as a function of time for four locations under thunderstorm and general storm conditions.

4.4.7-4 Tectonic and seismic analyses

Evaluation of ground motion at the Yucca Mountain site must address two types of events: (1) natural seismicity (earthquakes) and (2) underground nuclear explosions (UNEs), which are conducted periodically at the Nevada Test Site (NTS). The seismic design assumption for the SCP-CD is that 0.40g is the vibratory ground motion input. This value is based on information contained in current documents (USGS, 1984; DOE, 1986c; and URS/Blume, 1986) and seismologic and engineering judgment. This value may be revised as a result of ongoing studies, particularly the characterization of faults in the immediate vicinity of the site for use in future design analysis. The assumption does not account for any potential surface rupture on the faults in the site vicinity. A value of 0.40g for vibratory ground motion envelops the maximum ground acceleration expected from ground motion induced by a maximum yield UNE (700 kiltons) at the NTS, which is equal to 0.32g based on a mean value plus 3 standard deviations (Section 6.3.3.4, DOE, 1986c).

Two reports have been completed using probabilistic methods to estimate ground motion. In the first report (URS/Blume, 1986), seismogenic zones were delineated from regionalization of the southern Great Basin on the basis of historic seismicity, late Quaternary strain rates, and style of late Cenozoic deformation in order to predict the ground motion hazard at the site.

Further, an occurrence model for UNEs also was established in this study from historical data of NTS testing that occurred before the Threshold Test Ban Treaty. Any future testing was assumed to occur no closer than the Buckboard Mesa area, which is approximately 15 mi from the repository site and is closer than any of the locations on the NTS where testing actually occurs. A maximum yield of 700 kilotons was used for a UNE at Buckboard Mesa, based on the requirements for limiting offsite damage. The current testing limit of 150 kilotons, as limited by the Threshold Test Ban Treaty, produces negligible ground motion at the repository site. The ground motions calculated in URS/Blume (1986) are (1) an acceleration value of 0.4g with a return period of 2,000 yr as a result of natural seismicity and (2) an acceleration value of 0.15g based on a mean plus 2 standard deviations for UNEs.

The second study, which is in the process of completion, assumes faults in the vicinity of the site are active. Fault-specific, random-earthquake-occurrence models have been developed to predict the hazard caused by ground motion and fault displacements. The earthquake occurrence was determined from published fault length and slip rate information. The faults considered in the model include the Bow Ridge, Paintbrush, Ghost Dance, Midway Valley, and Severe Wash faults.

Preliminary evaluations indicate that several local faults range in length to 20 km or more. If the waste-handling facility were to be located in Midway Valley, the controlling earthquake source would appear to be the north-south-trending Paintbrush Canyon fault. This fault may be capable of producing a $M = 6.5$ earthquake, with an average return interval of several tens of thousands of years. Preliminary ground motion studies indicate that the most probable value for peak acceleration in close proximity to a $M = 6.5$ earthquake is about 0.5g. The probability of having a $M = 6.5$ earthquake or any significant surface offsets on the faults considered during the preclosure period is extremely small. Technology exists for designing surface facilities for the design accelerations much larger than those described above. Typical examples include Diablo Canyon Nuclear Power Plant and San Onofre Nuclear Power Plant. Published literature also shows that structures can be designed to resist moderate surface displacement before any catastrophic failure would occur (Reed et al., 1979). Plans for characterizing potential ground motion for design purposes are discussed in Section 8.3.1.17.

At the level of the underground facility, it is provisionally assumed that the peak accelerations are half those at the surface. This is based on an conservative estimate of attenuation of ground motion with depth available from UNE test data and other published information (URS/Blume, 1986; Carpenter and Chung, 1985) on earthquakes. Further investigation and data gathering during site characterization and the development of a satisfactory model for the surface and downhole data may result in revised estimates of peak accelerations. Carpenter and Chung (1985) indicate that up to surface shaking levels of 0.5g, no tunnel collapses as a result of shaking alone have been observed. Tunnels in poor soil and rock are more susceptible to damage than are tunnels deep in rock. Damages to all classes of deep tunnels consisted primarily of minor rockfalls and formation of new cracks except where active faults intersected tunnel bores. In these instances, localized severe damage was experienced. Hence, it can be seen that technology exists for designing underground facilities for the levels of accelerations described previously.

6.4.10.3 Future work

The planned future work is identified in Section 8.3.2.5.7. This work consists of (1) performing tradeoff and sensitivity studies to establish the design approach, the design configurations, and to select basic equipment types; (2) completing reference calculations; and (3) performing verification and validation of analysis codes.

For the underground facility, there are three specific things that need to be determined and assessed: (1) the potential for radon gas, (2) the impact of seismic events on the underground design, and (3) the radiation shielding characteristics of the formation.

The tunnel indexing methods will be qualified through further studies of case histories specific to the repository in tuff, demonstrations in the exploratory shaft facility (ESF), and checking against the implications of predictions made using boundary-element and finite-element analyses. The potential for studies of case histories may be limited because of the availability of underground openings in tuff. Demonstrations in the ESF will include evaluations of various ground support recommendations within the range of types of ground encountered. The boundary-element and finite-element techniques will be used to help qualify the tunnel indexing methods in at least two ways. In the first, the results of numerical analyses of openings without ground support for ambient temperature conditions will be compared with ground support recommendations using the tunnel indexing techniques to provide a second level of confidence to the initial ground support recommendations. Boundary-element and finite-element techniques (without ground support) will be applied next to determine the potential effects of heat on the opening stability. The results of these analyses will be studied to determine the potential for problems in the initial ground support design as a result of the heat. In this manner, analysis of these results will qualify the initial ground support recommendations.

Future work pertaining to method, model, and code qualification includes the following activities: (1) a review of existing methods, models, and codes; (2) verification; (3) benchmarking and parametric studies; and (4) validation. While it may appear that this work is to be completed sequentially, in reality, each progressive activity could lead to looping back through earlier ones. The following descriptions of activities apply to both boundary-element and finite-element models and codes.

At least three classes of material models (linear elastic and elastic-plastic, compliant joint, discrete discontinuities) are recommended for mechanical-structural calculations. A linear and nonlinear, steady- and transient-heat-conduction code is recommended for thermal calculations. A review of existing material models and codes will be performed to assess their applicability to repository performance, repository design, and site evaluation calculations. Selected material models and codes will be modified as necessary to satisfy requirements for analysis of repository performance and design.

Computer codes developed for engineering analysis will be verified to ensure that they correctly perform the operations specified in the numerical model. Verification will be accomplished by testing the model's numerical computations against closed-form analytic solutions. Part of the verification procedure for finite-element codes will be solutions comparisons with previously fully documented boundary-element codes.

Benchmarking is the comparison of the results on one item of software with the results of another item of software designed to solve a comparable problem to show that they produce similar results. Material models and codes will be benchmarked by cross-checking the numerical solutions to a series of well-defined thermal, mechanical, and thermomechanical boundary value problems. At least one benchmarking analysis will be run for each model for each problem scale to be encountered in repository design. Material properties, in situ conditions, boundary conditions, and loading conditions for these problems will be representative of those expected of the repository. The material models will be further evaluated through parametric studies in which input parameters are systematically varied to determine the relative significance of a parameter and to ensure that the variations impart the correct sense of change in material behavior.

It is currently planned to have models and codes qualified for Level I analyses by the end of final benchmarking activities. At that time only initial validation analyses will have been completed.

Validation is ensuring that the physical model, as embodied in software, is a correct representation of the intended physical system or process.

Validation will be accomplished by comparing the results of numerical computations with the results of field-, bench- and laboratory-scale experiments. Certain G-Tunnel, exploratory shaft, and laboratory experiments were developed for this purpose. The purpose of these physical models is to test the physics embodied in the material models. Analog material tests may be appropriate for this purpose. Validation analysis also may be conducted by comparing calculated results to experimental results available in the open literature. In general the validation process will be conducted using the following series of steps:

1. Experimental design analysis is performed to develop the experiment concept in a design that will address the phenomena of interest.

2. Site-specific data and material properties are collected for model calculations.

3. A pretest analysis is performed.

4. The experiment is conducted.

5. The pretest analysis is reevaluated in light of the actual experimental procedure.

6. A posttest comparison of experiment and analysis is conducted by a peer review panel.

Field experiments in the ESF are in the process of being designed specifically for validation of models used for thermal, mechanical, and thermomechanical analyses. The design of these experiments will include pretest analyses to optimize the experiment and analysis for the validation activity. For example, experiment location, orientation, loading conditions, and data collection arrays will be chosen such that analysis of the experiment (i.e., validation) will be facilitated.

6.4.11 ISSUE 4.5 REPOSITORY SYSTEM COST EFFECTIVENESS

6.4.11.1 Introduction

The question asked by Issue 4.5 is

> Are the costs of the waste package and repository adequately established for the resolution of the performance issues?

This section is concerned with the cost estimating activities related to establishing the cost bases necessary to perform the comparative analysis required by the performance issue under Key Issue 4. That performance issue addresses the higher level finding for the system guideline on ease and cost of construction operation, closure, and decommissioning of the mined geologic disposal system required by 10 CFR Part 960. The higher level finding is concerned, in part, with a comparative evaluation of the total system life cycle costs (TSLCC) for each of the repository siting options.

The work completed to date by the NNWSI Project has focused on the site specific aspects of the TSLCC. Specifically, cost estimates have been prepared for the construction, operation, and decommissioning phases for facilities and for waste packages. This information, which is tabulated in Table 6-38, is summed to produce the total repository life cycle cost (RLCC) estimate. The RLCC estimates are only a portion of the TSLCC; the resolution of this issue requires the consideration of nonsite specfic costs such as those associated with a monitored retrievable storage (MRS) facility, as well as transportation costs.

The discussion of the proposed strategy for resolution of this issue is presented in Section 8.2.2.3.1.1. Readers not familiar with the resolution strategy for this issue should review this section before continuing.

Issue 4.5 has been subdivided into three information needs. The text that follows briefly identifies each information need and discusses site specific work that has been completed to date for each of the information needs.

<u>Information Need 4.5.1</u> Estimate the costs of the reference and alternative Waste packages

This information need consists of preparing and compiling the costs associated with the fabrication of the reference waste package, as well as alternative designs. The waste package includes the container and the materials within the container. Under the curent NNWSI Project design, the cost of the waste package is essentially the cost of the container. Waste container costs constitute a significant portion, about 9%, of the RLCC and are allocated entirely to the operations phase. They constitute over 12% of the operating phase costs (Table 6-38). There are several waste container design variables that affect these costs: size, material type, shell thickness, capacity, fabrication method, internal separator configuration, and quality assurance (QA). Cost analysis for alternative designs contributes to the TSLCC analysis and provides input for the selection of the final container design.

<u>Information Need 4.5.2</u> Estimate the costs of the reference and alternative repository designs

This information need consists of preparing and compiling the costs associated with the construction, operation, and closure of the reference repository design, as well as alternative designs. Repository design costs allocated to the construction phase include the costs for the final design, construction, and inspection of surface and subsurface facilities, and for management and integration of these activities. The original construction and capital equipment cost estimate for the reference and alternative repository designs constitute approximately 21% of the RLCC or about $1.4 billion (Table 6-38). Repository life cycle costs for the construction phase are sensitive to the size and complexity of the physical plant required to support surface and subsurface operations.

Table 6-38. Repository Life Cycle Cost (RLCC) Comparison[a]

Cost Account Description	Construction Phase			Operations Phase			Decommissioning Phase			TOTAL RLCC	PERCENT RLCC
	Cost in $ Million	Percent Phase	Percent RLCC	Cost in $ Million	Percent Phase	Percent RLCC	Cost in $ Million	Percent Phase	Percent RLCC		
Management and integration	346	24.4	5.2	49	1.0	0.7	20	5.3	0.3	415	6.3
Architect/Engineer	217			0			14				
Construction Management	74			0			2				
Other	55			49			4				
Surface facilities	785	55.4	11.9	2,546	52.9	38.6	116	31.3	1.8	3,447	52.3
Site	167	11.8	2.5	114	2.4	1.7	38	10.2	0.6		
Waste-handling facilities	488	34.4	7.4	1,242	25.8	18.8	34	9.1	0.5		
Balance of plant	130	9.2	2.0	1,190	24.8	18.0	45	12.0	0.7		
Subsurface facilities	286	20.2	4.3	1,610	33.5	24.4	235	63.4	3.6	2,131	32.3
Shafts and ramps	62	4.4	0.9	28	0.6	0.4	3	.8	0.0		
Excavation and emplacement	136	9.6	2.1	838	17.5	12.7	101	27.1	1.5		
Service systems	88	6.2	1.3	744	15.5	11.3	132	35.5	2.0		
Waste packages	0	NA[b]	0.0	603	12.5	9.1	0	0.0	0.0	603	9.1
Spent fuel	0			351	7.3	5.3	0	0.0			
Defense High Level Waste	0			128	2.7	1.9	0	0.0			
Other	0			125	2.6	1.9	0	0.0			
Repository Life Cycle Cost	1,417		21.5	4,809		72.9	371		5.6	6,597	100.0

[a]Source: Gruer et al. (1987)
[b]NA = Not Applicable

Information Need 4.5.3 Estimate the life cycle costs of the reference and alternative total system designs

This information need consists of preparing and compiling the costs of the reference and alternative total system designs. The composite cost estimates for the reference and alternative total system designs cover all chronological phases and are referred to as the TSLCC. The TSLCC includes the costs of the waste package (Information Need 4.5.1) and the costs of the repository (Information Need 4.5.2). It also includes, however, costs for activities such as transportation and development of an MRS, which is relatively nonsite dependent. The NNWSI Project has prepared composite cost estimates for the reference and alternative waste package design, as well as the reference and alternative repository designs. These cost estimates cover only the site-specific portion of the TSLCC for three phases: construction, operation and decommissioning.

6.4.11.2 Work Completed

Information Need 4.5.1 Estimate the cost of the reference and alternative waste packages

Several preliminary waste container cost estimates have been performed for the two-stage repository study (SNL, 1986). The first preliminary cost estimate was based on a generic waste package. This design was revised to be more site-specific and the cost estimate was recalculated.

Approach

Cost estimates for the boiling water reactor (BWR) fuel rod and pressurized water reactor (PWR) fuel rod emplacement containers were determined by averaging several manufacturers' quotes. These quotes were obtained from manufacturers experienced in the fabrication of similar chemically resistant low-carbon stainless steel vessels. The technology for fabricating such stainless steel vessels is well established and is very similar to the fabrication process expected to be used for the emplacement containers.

Cost estimates for the alternative emplacement containers were determined by extrapolating the costs of the reference case containers. The similar physical parameters of the reference and alternative containers studied thus far have permitted plausible results. Shipping, handling, and quality assurance and quality control costs were also included.

Data

Data were obtained from manufacturer quotations for 2,000 units.

Results

Estimated emplacement container costs are identified specifically and included as data in RLCC reports and in the cost report by Gruer et al. (1987).

Information Need 4.5.2 Estimate the cost of the reference and alternative repository designs

Cost estimates are only approximations of value at a given time and are very sensitive to the degree of completeness of the design definition and available historical data. In the initial design stages, such as at SCP-CD, design details are not defined and large contingency factors were used to compensate for design uncertainty. As the design evolves through the conceptual phase, design options studies, development of the advanced conceptual design, and the later detailed design definition stages, the design information available upon which to base estimates becomes correspondingly more detailed, confidence increases, and contingency factors decrease.

Several traditional cost-estimating methods have been used to produce the direct (bare) labor and material costs because the amount of design detail available is not the same for each cost account considered (see Table 6-38). The design detail available for the particular category of cost account was the primary determining factor for the method used.

The subsurface cost-estimating methods were those traditionally used in the mining industry and included unit costs and itemized material takeoff. A detailed explanation of the method is described in the cost estimate for the SCP-CD (Gruer et al., 1987).

Cost markup factors were applied to the bare costs using conventional computer-oriented methods that use off-the-shelf personal computer spreadsheet and data base management software. Guidance for the format, methods, and cost account definitions was provided by the DOE/Weston cost guidelines (DOE, 1987d).

Data

Data sources include the architect-engineering (A/E) data bases, commercial cost-estimating references, and vendor's quotes. Data types include direct labor units and/or lot costs, equipment costs, and indirect labor and material costs.

Indirect labor and material factors were determined by the A/Es from their historical data bases. Other cost factors, such as engineering and contingency, were determined by professional judgment.

Results

Work completed includes nine capital cost estimates. Four cost estimates were produced for the two-stage repository report (SNL, 1986), which compared a single construction stage to a repository constructed in two stages, plus a vertical (reference) emplacement design and a horizontal emplacement design (for each construction stage). These same two emplacement configurations were reestimated for the SCP-CD. Construction costs are summarized in Table 6-38, which also shows the relative costs between the major cost accounts.

The results of the capital cost estimate are used for several purposes. During the conceptual design phase, the results are the basis for most of the input to the RLCC. This estimate is used as a factorable base from which other major project costs are derived, such as operations and maintenance, decommissioning, project management, equipment design and inspection, quality assurance, and contingency. These capital cost estimates are also used in the preparation of the project data sheets for project planning, funding, and management.

Information Need 4.5.3 Estimate the life-cycle costs of the reference and alternative total system designs

Approach

Total System Life Cycle Cost (TSLCC) estimates have been prepared to support the preliminary finding about the preclosure system guideline on ease and costs of construction (DOE, 1987e), as well as the annual assessment of waste fund fee adequacy (DOE, 1987d). The cost estimates prepared by the NNWSI Project under Information Needs 4.5.1 and 4.5.2 can be used to estimate the RLCC, which in turn is used to estimate the TSLCC. Different analytical approaches have been used to estimate the RLCC for each life-cycle phase, and several different approaches have been used within each phase. The costs for the construction phase are discussed under Information Need 4.5.2. The costs for waste packages, which are allocated to the operations phase, are discussed under Information Need 4.5.1. The operations phase, which lasts for 50 yr and accrues about 73% of the RLCC, is composed of two subphases: emplacement operations and caretaker operations. Preliminary estimates of the surface and subsurface facilities operations and maintenance cost were performed by the respective A/Es, based on preliminary information they had previously developed.

Emplacement phase operating labor, materials, and supply costs were estimated by using several methods. The labor force cost was estimated from a functional breakdown of site operations such as waste-handling equipment operation, hot cell operation, process maintenance, fire protection, training, health and safety, and quality assurance. Material and supply costs were factored as a percentage of the construction costs. Cost markup factors were determined by DOE guidance, historical data, and engineering judgment. The caretaker operations phase costs were estimated by the same method.

The final phase contains three distinct activities: closure, decommissioning, and site marking. All three costs were determined by examining plans for each activity and ascertaining sequences, material requirements, and costs, as well as labor requirements and associated costs.

Off-the-shelf personal computer spreadsheet and data base management software were used for the costs study (SNL, 1987). The cost matrix format, the cost account definitions, and the current RLCC methods were provided in the DOE/Weston cost guidelines (DOE, 1987d).

Data

The primary data sources included the A/E data bases, commercial cost-estimating references and vendor's quotes. Data types include direct labor units and/or lot costs, direct material units and/or lot costs, equipment costs, and indirect labor and material costs.

Results

The work completed for this information need includes nine RLCC estimates for various designs for a repository in tuff. An RLCC estimate was made for each of the four designs in the two-stage repository report, which are (1) a single-stage repository based on emplacement of single containers in short vertical boreholes, (2) a two-stage repository based on emplacement of single containers in short vertical boreholes, (3) a single-stage repository based on emplacement of multiple containers in long horizontal boreholes, and (4) a two-stage repository based on emplacement of multiple containers in long horizontal boreholes. The costs for current designs for a two-stage repository based on a single container in a short vertical borehole and multiple containers in a horizontal borehole were estimated for the SCP-CDR. Table 6-38 is excerpted from the SCP-CDR, and additional information may be found in Stinebaugh and Robb (1987).

The RLCC information is used primarily for project management purposes, including project funding, funding schedule, socioeconomic impact studies, and electric utility fee adequacy analysis.

6.4.11.3 Future work

Analysis Needs

Estimation of waste package costs is an ongoing activity that will support waste package design decisions during the advanced conceptual design (ACD), the license application design (LAD), and the final procurement and construction design (FPCD).

Identified future analysis needs include physical parameter sensitivity studies; reappraising quality assurance factors, especially with regards to NQA-1 criteria; and shipping and handling costs. Other specific future cost analysis needs have not yet been identified. Special NNWSI intraproject and DOE interproject studies will likely be identified as the design detail develops.

Cost needs include (1) new requests for quotation when the design, scope, or assumptions change; (2) regular reassessment of the appropriate cost factors, such as quality assurance and quality control, contingency, and engineering; and (3) the use of a consolidated data base management system to facilitate integration of new emplacement container costs in the RLCC.

Repository construction and TSLCC estimates will be updated at each design phase; the TSLCC for the ACD will form the basis for resolution of Issue 4.5 through a comparative evaluation of costs among siting options.

Alternative cost-analysis needs depend on the number of design considerations and the amount of design detail available. When more design detail is available, more detailed cost estimates with greater accuracy and smaller contingency are possible. Changes in economic conditions and labor policies that might impact the repository construction and operating costs will be considered in the analysis.

General cost-estimating analysis needs include analysis for staffing, quality assurance, operating and maintenance materials and supplies, decommissioning, site marking, and capital equipment.

Development needs

Internal to the cost-estimating process, there are no technical or process development needs. The RLCC data base will expand during the various design phases and become very site specific and detailed. Because of the large scope of this particular project, the only practical cost-estimating method requires the use of a computer.

Site information needs

There are no technical site characterization information needs related directly to the estimation of costs for resolution of this issue.

DOE/RW-0160

Chapter 6

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Chapter 7

Nuclear Waste Policy Act
(Section 113)

WASTE PACKAGE

Consultation Draft

Site Characterization Plan

Yucca Mountain Site, Nevada Research and Development Area, Nevada

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Office of Civilian Radioactive Waste Management
Washington, DC 20585

Chapter 7

WASTE PACKAGE

INTRODUCTION

This chapter describes waste package components, emplacement environment, design, and status of research and development that support the Nevada Nuclear Waste Storage Investigation (NNWSI) Project. The site characterization plan (SCP) discussion of waste package components is contained entirely within this chapter and Chapter 8. The discussion of emplacement environment in this chapter is limited to considerations of the environment that influence, or which may influence, if perturbed, the waste packages and their performance (particularly hydrogeology, geochemistry, and borehole stability). The discussion considers the near-field environment based on environment described in previous chapters (especially Chapters 2, 3, 4, and portions of 6) but does not address overall site environment as do those chapters. The basis for conceptual waste package design as well as a description of the design is included in this chapter. The complete design will be reported in the advanced conceptual design (ACD) report and is not duplicated in the SCP. The site characterization data that will provide the waste package environment design envelope information to advanced conceptual designs is covered under Issue 1.10 in Section 8.3.4.2. Discussion of studies related to ACD is covered in Section 8.3.4.3. The focus of the design discussions in the SCP is on those aspects of the design that depend on site characterization information. The relationship between this chapter and other design-related documents is shown in Figure 7-1.

Chapter 7 gives the status of the NNWSI Project's knowledge and discusses the extent to which current information is sufficient to demonstrate that objectives will be satisfied. Section 8.3 discusses how the additional information needed to resolve issues will be obtained and how each issue will be resolved so that it can be demonstrated that the performance objective will be met.

WASTE PACKAGE COMPONENTS

For the purposes of this chapter, the waste package is divided into two functional components: waste form and container. In addition, constructed features of the geologic repository immediately adjacent to the container and the near-field host rock environment are discussed.

The waste forms of concern are of two generic types: spent nuclear fuel resulting from the operation of commercial power reactors and high-level waste resulting from fuel reprocessing associated with commercial and defense activities. In addition, the nonfuel components associated with commercial spent fuel are expected to be packaged and emplaced in a geologic repository.

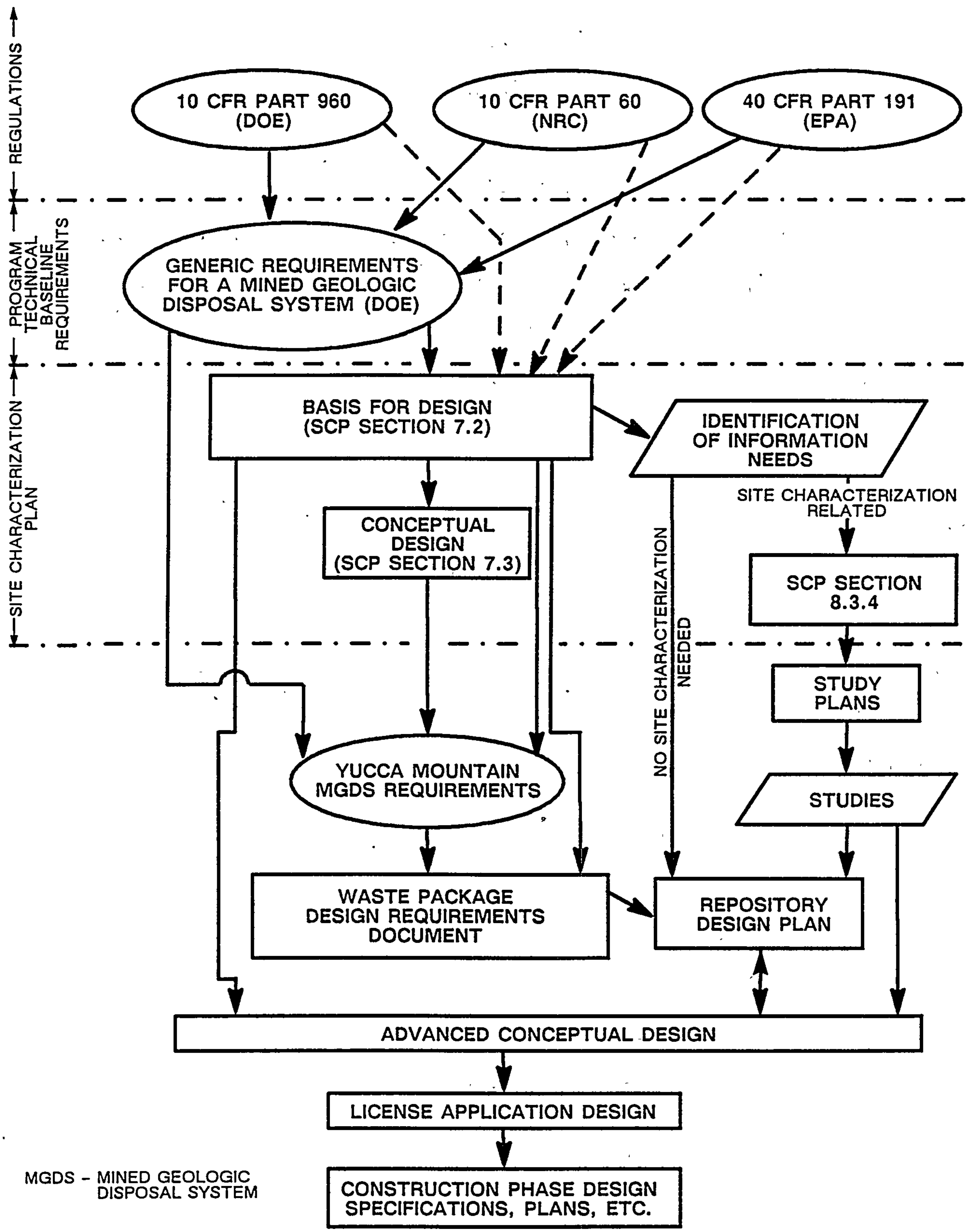

Figure 7-1. Flow chart indicating relationship of design-related documents for the waste package.

The waste package container is the component specifically designed as an engineered barrier to provide containment of the waste form during the period when the radiation and thermal conditions are dominant or are changing rapidly.

Other constructed features include the openings in the host rock, currently planned to be mechanically bored cylindrical holes, into which the waste packages are emplaced and other features such as borehole liners and any mechanical appurtenances needed to facilitate emplacement or possible retrieval of the packages.

REQUIREMENTS

The requirements for performance of the waste packages are either defined in detail or implied by three regulations (40 CFR Part 191, 10 CFR Part 60, and 10 CFR Part 960) and by a document that responds to all three regulations entitled "Generic Requirements for a Mined Geologic Disposal System" (GR) (DOE, 1984). Environmental Protection Agency (EPA) regulations (40 CFR Part 191) do not place requirements directly on the waste package. The U.S. Department of Energy (DOE) 10 CFR Part 960 contains general specifications for the waste package, deferring to 10 CFR Part 60 for details. Specific requirements for the waste package as defined in 10 CFR Part 60 and in the GR fall into two broad categories, preclosure and postclosure. Not all the requirements are appropriate for discussion in the SCP because the regulations include items that are independent of site characteristics. The specific portions of the regulations that are appropriate for discussion here are shown in Table 7-1.

Preclosure criteria or requirements are associated with the operational phases of the repository system, such as providing for waste package material compatibility, identification, waste form criteria, and handling (including emplacement and retrieval). These considerations are essentially covered in 10 CFR 60.135(b) and (c) and, except for portions of the handling, are not related to site characterization activities as defined by the Nuclear Waste Policy Act (NWPA, 1983). Therefore, these aspects will be addressed in reports on future design phases (advanced conceptual design and license application design reports) rather than the SCP. Aspects of handling require only limited site characterization information as inputs to package design and are essentially related to conduct of repository operations as described in Section 6.2.

The postclosure requirements are directed toward the long-term performance of the package components in providing containment and isolation functions. Long-term performance includes considerations of interaction between the waste package and its environment. The specific requirements are contained in 10 CFR 60.113 and 60.135(a). The requirements in 10 CFR 60.113 assign the containment function to the waste package and the controlled release of radionuclides function to the engineered barrier system, of which the waste package is a major component. The requirements in 10 CFR 60.135(a) address the waste package and its components. Therefore, this chapter will focus on the postclosure considerations of the waste package with design and other preclosure considerations summarized but not developed in detail. The

Table 7-1. Regulations that address site-specific requirements for the waste package

Regulation	Applicable sections	
	Preclosure	Postclosure
10 CFR Part 60 (NRC)[a]	60.135(b), (c)	60.113 60.135(a)
10 CFR Part 960 (DOE)[b]	960.5-1(2), (3) and Appendix II	960.5-1 Appendix I
40 CFR Part 191 (EPA)[c]	No direct requirements	Addresses overall performance, does not specify waste package directly
Generic Requirements for a Mined Geologic Disposal System (DOE)	1.3.1 1.3.2	2.2.1

[a]NRC = Nuclear Regulatory Commission.
[b]DOE = U.S. Department of Energy.
[c]EPA = Environmental Protection Agency.

detailed discussion of these nonsite-specific considerations will be contained in the advanced conceptual design (ACD).

HISTORY OF ACTIVITIES

Before 1982, waste package research and development activities were not specifically assigned to the NNWSI Project but were conducted on a generic basis by the National Waste Terminal Storage Program. The emphasis of the NNWSI Project was on identifying a suitable repository horizon in the saturated zone of Yucca Mountain, and the waste package activity was directed toward defining the applicable environment and developing design concepts appropriate to that environment. In late 1982, the emphasis of the NNWSI Project shifted to the consideration of the unsaturated zone as a potential repository setting and, specifically, to the investigation of the Topopah Spring Member of the Paintbrush Tuff as the reference repository horizon. This change was made to take advantage of the separation from the zone of saturation where the ground water provides a more direct pathway to the biosphere and also to take advantage of the expected more beneficial environment for metal barriers and for waste forms.

This represented a substantial change in the environment in which a waste package would have to exist. A different set of physical and chemical phenomena, which are useful for meeting the containment and isolation requirements, exist in the unsaturated zone. Thus the change presented an entirely new regime for testing and evaluating package components. Consequently, the specification of the environment in which the package would function was revised, and the direction of the research, development, and testing plans for evaluating the performance of package component materials was modified.

Activities that have been conducted by the NNWSI Project since 1982 regarding the waste package have centered around those necessary to support a license application for construction of a repository. The activities are grouped into four major categories: (1) waste package environment; (2) waste form and materials testing; (3) design, analysis, fabrication, and prototype testing; and (4) performance assessment. Each of these groups of activities is discussed in detail in subsequent sections of this chapter. A general description of these activities follows.

Waste package environment activities involve characterization of the variability of properties of Topopah Spring tuff in the repository area. Included are studies of (1) hydrothermal reactions between the tuff and ground water and (2) rates and mechanisms of dehydration and rehydration of rock adjacent to the boreholes.

Waste form and materials testing activities involve measurement of radionuclide release rates from spent fuel and from borosilicate glass to provide input for modeling of long-term radionuclide release rates from the waste package; determination of whether a packing material must be incorporated in the waste package design; selection of candidate metals for fabrication of containers; characterization of the corrosion rates of these metals under expected conditions; and characterization of the effect that other materials associated with the repository might have on the long-term performance of the waste form and container materials.

Design, fabrication, and prototype testing activities involve development and testing of waste package designs that are compatible with repository design.

The performance assessment activity involves the development and validation of models for use in making long-term predictions of package performance for comparison with U.S. Nuclear Regulatory Commission (NRC) performance objectives. Associated with this activity, though not an integral part of it, is the development of advanced computer codes for modeling geochemical processes in the repository environment.

UNCERTAINTIES IN WASTE PACKAGE DEVELOPMENT

Several sources of uncertainty exist in the development and analysis of performance waste packages. Some of these will be reduced significantly as a result of the investigations and tests described in Chapter 8; others are inherent in the inhomogeneous characteristics of the waste forms and the geohydrologic setting of the repository and will, therefore, remain to be considered using statistical techniques in the analyses of projected performance.

Among the types of uncertainties that exist in the waste package development and analysis activities are

1. Variability in the geohydrologic system and uncertainty in understanding the response of the system to repository thermal loads. Significant uncertainties are associated with the present information on the characteristics of the natural system, primarily due to the limitations imposed by surface-based investigations. These data will be supplemented by extensive laboratory and in situ testing in the future that is expected to substantially improve the knowledge of the nature and response of the natural system. Residual uncertainties will result from the limitations of the laboratory and in situ testing program in both areal extent and duration.

2. Identification of the mechanisms and prediction of the rates of degradation of the metallic components of the packages, especially the metal containers. The major sources of these uncertainties are the lack of a final selection of container material and the influence of the processes that will be used to form and close the containers on their long-term performance. The selection of a material and determination of the fabrication processes will allow assessment of these uncertainties by relatively short duration tests, but residual uncertainties will exist due to the lack of relevant experience with the materials over the time periods of interest in this application. These service lifetimes are far longer than any other engineered structures, and therefore beyond the limits of engineering experience.

3. Variation of the physical and chemical characteristics of the waste forms. These uncertainties can be reduced by a systematic examination of the existing and proposed wastes, but there is a substantial residual uncertainty resulting from the fact that a large fraction of the wastes to be disposed of do not exist now, and will not exist prior to the application for a license for the disposal system.

4. Limitations on the ability to model accurately the physical and chemical processes operating on and in the waste packages. The development of fully coupled models that include all the synergistic interactions that can affect the performance of a combined engineered and natural system is beyond the existing capability of the scientific community. Therefore, simplifications of these interactions will be necessary to allow the evaluation of the known uncertainties in the system. These simplifications will themselves introduce a level of uncertainty to the analyses of the predicted performance.

The present status of the evaluation of the types of uncertainty known to be associated with the various aspects of the waste packages are discussed within the appropriate sections of this chapter and summarized in Section 7.5.5.

CHAPTER ORGANIZATION

The reference waste package emplacement environment is briefly described in Section 7.1, which is intended to orient the reader to those features of the unsaturated zone that are significant in the design and performance of waste packages.

Section 7.2 specifies the design basis for the waste packages and includes the applicable regulatory requirements, as well as design assumptions that have been adopted on an interim basis for the purpose of conceptual design. These design bases will be modified in the waste package design requirements document which will be issued after the SCP.

The initial conceptual waste package designs for the reference case (including conceptual design drawings, general material specifications, and configurations) are described in Section 7.3. These design concepts were developed during 1983 and 1984. A follow-on ACD phase will be discussed in an ACD report. Alternative waste package designs, including those of copper-based alloy containers, are also discussed in Section 7.3 but not at the same level of detail, since the reference case was based on 1984 activities. Evaluations (laboratory and analyses) have continued since that time on copper-based containers. Copper and copper-based alloy containers are being considered for future work as described in Section 8.3.4.

Section 7.4 discusses the status of waste package research and development activities that have been under direct NNWSI Project auspices since late 1982. Section 7.4 is divided into five subsections, each of which addresses a component of the research and development activity.

Section 7.4.1 describes the status of investigations of the waste package environment, including thermal effects and chemical interactions in the region adjacent to emplaced waste packages and the stability of boreholes.

Section 7.4.2 discusses tests on various candidate metal container materials, with primary emphasis on the investigation of degradation modes of austenitic stainless steels. This section also briefly discusses projected container lifetimes and likely failure modes. Recently initiated testing of copper alloys is also discussed.

Section 7.4.3 describes the waste form testing activities that have been conducted on both spent fuel and high-level waste in borosilicate glass. The activities include waste form dissolution tests as well as investigations of spent fuel oxidation and cladding corrosion. This section includes a brief discussion of release models for use in developing waste package source terms.

Section 7.4.4 describes the status of development of geochemical models that will be used in the assessment of the waste package performance.

Section 7.4.5 describes the functions of waste package performance assessment and the work on the development of a waste package system model to be used to show compliance with the waste package performance objectives and assessments.

This chapter concludes with Section 7.5 which summarizes the significant results of the activities described in the preceding sections. References are given to appropriate subsections of Chapter 8 where the performance allocation, issue resolution strategy, and detailed plans are presented for acquisition of additional required information to demonstrate satisfying performance objectives.

7.1 EMPLACEMENT ENVIRONMENT

Characterization of the emplacement environment is required to meet performance assessment goals and to provide information necessary to carry out materials and design tests. This information also provides input to development of source term models.

The NNWSI Project has selected the Topopah Spring Member of the Paintbrush Tuff as the reference repository horizon for a potential repository for high-level nuclear waste sited at Yucca Mountain. The investigations that led to the choice are discussed in the unit evaluation report (Johnstone et al., 1984). The reference horizon is the welded, devitrified section c Topopah Spring tuff that lies above the basal vitrophyre of the unit. The repository would be approximately 350 m below the ground surface and 200 to 400 m above the static water table (Sinnock et al., 1984; Ortiz et al., 1985). The stratigraphy of the site is described in Section 1.2, and the structural features of the site are discussed in Section 1.3. Hydrologic characteristics of the site are summarized in Chapter 3. Data related to these characteristics have been used by Montazer and Wilson (1984) to develop a conceptual hydrologic model of flow in the unsaturated zone at Yucca Mountain.

The choice of the unsaturated zone as the location for the reference repository horizon marks a departure from the conventional environment that has been considered for repository siting. There are many characteristics of the unsaturated zone that make it particularly attractive for a high-level waste repository site. Winograd (1981) and Roseboom (1983) have discussed waste disposal in areas where thick unsaturated sequences of rock are present. The unsaturated setting simplifies the design of waste packages and provides limits on the conditions to which the waste packages will be subjected. Some of the anticipated conditions of the setting relevant to waste package design and performance are the following:

1. The waste containers would not be submerged in a continuum of water. Anticipated conditions in the unsaturated zone suggest the containers will experience high humidity during the period of substantially complete containment (Section 7.4.1). Intermittent contact with limited amounts of liquid water for some packages is possible under some conditions.

2. The pressure exerted on the containers by the environment would be approximately one atmosphere. There would be no hydrostatic pressure because of the location above the water table. The packages would not bear lithostatic load because the host rock is not

expected to creep under the postemplacement thermal regime (Section 2.5). The load-bearing requirements for the waste packages are limited to pressures caused by sloughing of rock from emplacement boreholes.

3. The environment to which the waste packages would be exposed would be a mixture of water vapor and air, with a total pressure of about one atmosphere during the period of time when the temperature is above the local unconfined boiling point of water, approximately 96°C. Only slight variation (<1.0°C) in the boiling temperature can be expected for unconfined, unbound, non-capillary pore water even if the concentration of dissolved salts is greatly increased (Section 7.4.1.2).

4. Aqueous corrosion of waste packages by liquid water and its dissolved constituents would only commence after the temperature of the container drops to the point where a liquid film could form on the container surface. For packages containing spent fuel, liquid water may not contact the container until hundreds of yr have passed (Section 7.4.1).

5. The vadose water and atmosphere of the repository would be mildly oxidizing due to the exchange of repository air with the external environment and due to radiolysis effects (Section 7.4.1.4). The degree of oxidation has yet to be established. This condition could be used advantageously in waste package design if the metal container chosen were of the type that forms a protective oxidized surface on the metal.

6. Water available for corrosion and waste form dissolution would be limited to very small amounts. The net infiltration of water at Yucca Mountain was estimated by Montazer and Wilson (1984) to be between 0.5 and 4.5 mm/yr. However, only a small portion of the net infiltration would be expected to pass through the repository horizon. An upper estimate of 0.2 mm/yr of downward flux was given by Montazer et al. (1986) for the matrix of the Topopah Spring welded unit. The estimated range of downward flux is 0.1 micrometer per yr to 0.2 mm/yr (Montazer et al., 1986). To allow for uncertainties in this estimate, 1 mm/yr will be adopted as a working value for the purposes of waste package design and testing which is the conservative estimate of transmission through the Topopah Spring unit.

7. Thermal perturbation of the near-field environment by the waste package will dehydrate the near-field host rock and will prevent liquid water from contacting the waste packages for hundreds of yr (SCP Section 7.4.1.2; Glassley, 1986). This thermal effect will diminish the potential for liquid water to corrode the waste container for several hundred yr after emplacement and will eliminate the possibility of dissolution and transport of waste during the required period of substantially complete containment.

8. Radiation effects will be limited to those resulting from interaction of gamma and neutron radiation with the moist air and rock

although neutron doses will be negligible compared to gamma doses (Van Konynenburg, 1984, 1986). The radiolysis products will depend on the temperature of the environment, which will be a function of time. By the time temperatures are low enough for liquid water to contact the waste packages, the radiation level will be diminished by about three orders of magnitude, resulting in a similar decrease in the abundance of radiolysis products in the water-air-rock system.

The host rock at the reference repository horizon is a welded, devitrified, ash-flow tuff. The mineralogy and petrology of the rock have been investigated in detail using samples obtained from drill holes sited around the edge of the exploratory block at Yucca Mountain. Detailed reports giving quantitative estimates of mineral compositions and proportions for major and trace phases have been published by Bish et al. (1981), Vaniman et al. (1984), and Warren et al. (1984). A summary of the data is presented in Section 4.1.1. The major phases in the rock are a fine-grained assemblage of alkali feldspar, quartz, and cristobalite; minor phases include phenocrysts of plagioclase, alkali feldspar, iron-titanium oxides, quartz, and biotite, and variable amounts of montmorillonite clay. Trace phases include allanite, apatite, zircon, and occasional fluorite. Vein and fracture fillings consist of various assemblages of the minerals mordenite, calcite, quartz, feldspar, cristobalite, heulandite, clinoptilolite, and smectite (Carlos, 1985).

The chemistry of water in the saturated zone beneath Yucca Mountain has been investigated by Benson et al. (1983) and Ogard and Kerrisk (1984). The water samples showed a very limited range in chemical composition for aquifers in the tuff units. The results of water chemistry investigations are summarized in Section 4.1.2.

The composition of water from these investigations is very similar to that of vadose water in tuffs from Rainier Mesa (White et al., 1980; Henne, 1982). This similarity suggests that vadose water and ground water in the silicic tuffs of the area have similar compositions. Water from well J-13, a high flow rate well east of Yucca Mountain, has been selected as the reference water for experimental studies because its producing horizon is the Topopah Spring Member of the Paintbrush Tuff and because its composition is very similar to that of previously analyzed vadose water and ground water. The uniform composition of waters in this area suggests that the composition of vadose water at Yucca Mountain is similar to water from well J-13.

Samples of vadose water will be obtained during the sinking of the exploratory shaft. Should the water composition in the vadose zone differ markedly from that in the saturated zone, some or all the previously completed testing may have to be repeated. Present indications, based on the absence of readily soluble components in drill core samples, are that the water chemistry in the vadose zone should be similar to that in the saturated zone (Oversby, 1985; SCP Section 7.4.1.3).

The composition of the pore fluid phase may be altered as a result of evaporation, fluid-migration, and condensation processes resulting from heating of the repository host rock. It is expected that two-phase flow will be initiated in regions adjacent to the dry-out zone. During this process, evaporation of water may lead to deposition of dissolved salts that may be redissolved during periods of rehydration when the repository cools. The magnitude of this process and the parameters controlling it are the subjects of current investigations.

The ambient hydrologic conditions at Yucca Mountain are discussed in Chapter 3. Montazer and Wilson (1984) used the available data to construct a model for the flow paths and fluxes to be expected in the repository region, and provided the following summary of characteristics of the rock. The Topopah Spring welded unit has a fracture density of 8 to 40 fractures per cubic meter, a mean matrix porosity of 14 ± 5.5 percent, and a mean saturation level of 65 ± 19 percent. The water content of 29 drill core samples averages 5.5 percent by weight. The geometric mean of measurements of saturated matrix hydraulic conductivity is 3×10^{-6} m/d, and the effective hydraulic conductivity of the matrix has been estimated to be 1.6×10^{-7} m/d (Montazer and Wilson, 1984).

The model for flow in the unsaturated zone provides estimates of water fluxes through the various units at Yucca Mountain. Because of the potential for lateral flow within some units and at unit boundaries and of the effects of capillary barriers present in the natural setting, the flux through any given unit may be less than the net downward infiltration rate for the mountain as a whole. Indeed, it is possible for some units to have a net upward flux caused by vapor transport of water even though the system as a whole has a net downward flux (Montazer and Wilson, 1984). Geothermal data from the air-drilled drillhole USW UZ-1 indicate an upward flux of 1 to 2 mm/yr in the Topopah Spring welded unit although calculations of flux through the welded unit using effective permeability and hydraulic gradient in Darcy's equation give an estimate of flux of 0.003 to 0.2 mm/yr downward (Weeks and Wilson, 1984). Calculations based on the in situ potential gradient measured in drillhole USW UZ-1 and the effective permeabilities measured using drill core from the adjacent drillhole gave estimates of downward flux ranging from 1×10^{-7} mm/yr to 2×10^{-4} mm/yr (Montazer and Wilson, 1984).

The thermal and mechanical properties of the rocks at Yucca Mountain are summarized in Chapter 2. These properties are used to determine the maximum heat loadings that could be put into each waste package without violating the thermal constraints established for the waste forms. The rock mechanical properties are also used in the analysis of the stability of borehole openings (Section 7.4.1.1).

Emplacement of waste packages into the host rock would cause changes in the environment due to thermal effects and rock-water interaction. Research and development work to establish the nature of the changes is described in Section 7.4.1, waste package environment. Some changes to the environment would also occur because of construction methods and materials used during the construction and operational phases of the repository. Some types of inorganic materials, such as portland cements and grouts, can cause a large change in the pH of water interacting with them. This can have a deleterious effect on the leaching of glass waste forms (Section 7.4.3.2). Other types of grouting materials are to be preferred if they maintain a near neutral pH for the local water chemistry and are structurally sound. Deleterious effects may also occur from other sources (such as anions). These other sources will be considered in waste form testing as discussed in Section 8.3.5.10. Organic materials should be avoided in the immediate vicinity of waste packages because they and their radiolysis products could provide ligands for subsequent radionuclide transport. Where the use of organic materials is unavoidable, the effects caused by their use will need to be carefully evaluated and controlled, as discussed in Section 8.3.1.3.4.

7.2 DESIGN BASIS

In this section, the waste package design requirements derived directly and indirectly from regulatory requirements and the relationships of these requirements to waste package functions are discussed. Section 7.2.1 contains the waste package design requirements, performance criteria, and constraints that have been adopted for the development of the conceptual designs described in Section 7.3. Section 7.2.2 provides a reference to the baseline characteristics and receipt rates for the waste forms. Section 7.2.3 summarizes the allocation of performance for the waste package components.

The anticipated emplacement environment is described in Sections 7.1 and 7.4.1, both for the preemplacement and postemplacement periods in terms of chemical composition of the host rock and vadose water, flux and flow regimes of water, temperature, and rock stability. The postclosure radiation levels are described in Section 7.3, and the effect of these levels on the environment is discussed in Section 7.4.1.

7.2.1 DESIGN REQUIREMENTS

The design requirements for the waste package established in this section are derived from several sources. The primary source is the baseline "Generic Requirements for a Mined Geologic Disposal System" (GR) (DOE, 1984). The NNWSI Project-level document, Yucca Mountain Mined Geologic Disposal System Requirements, exists only as a draft and is not available as a baseline source of requirements at this time. Therefore, the functional requirements, performance criteria, and constraints specified in Sections 7.2.1.1 and 7.2.1.2 are cited from the GR.

Other sources of design requirements that have been included in the basis for the conceptual designs are 10 CFR Part 60, 10 CFR Part 960, and the NNWSI Project requirements established to ensure that component performance is not compromised. These design requirements are identified in Section 7.2.1.3.

The nomenclature used in the requirements, though perhaps confusing to the reader, is not intended to draw any distinction between a waste emplacement package and a waste package. These terms are used in the GR to differentiate between preclosure and postclosure system elements, but are assigned to the same physical object once it is completely assembled. The NNWSI Project waste emplacement package or waste package consists of the waste form and the disposal containers (including any required internal stabilizing structure).

7.2.1.1 Generic preclosure requirements

The preclosure waste emplacement package functional requirements are specified as follows in DOE (1984):

"1. To contain waste during unloading, handling, storage, further packaging at the repository surface facilities, emplacement, and retrieval, if necessary (10 CFR 60.135(b)(3)).

"2. To limit the potential for criticality within waste emplacement packages.

"3. To provide a means of unique identification."

The corresponding performance criteria for these requirements are specified as follows (DOE, 1984):

1. Handling--"A. The waste emplacement package must remain intact as a unit, which contains the waste and provides for safe handling of the waste, at least to the end of the period of retrievability.

   "B. The waste emplacement package must be capable of sustaining normal handling and packaging operational loads without loss of containment, and design bases accidents either without loss of containment or with a limited release of radionuclides as required in 10 CFR 20 or 10 CFR 100 [sic] when applicable."

   (In addition, pending revisions of the Generic Requirements document include an additional criterion that the waste emplacement package shall be designed and fabricated to allow retrieval of the waste by the methods planned from the emplacement configuration used in the repository.)

2. Criticality control--"The internal waste distribution in waste emplacement packages shall be such that nuclear criticality shall not be possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5 percent margin after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation."

3. Unique identification--"Provide a label or other means of identification for each waste emplacement package. The identification shall not impair the integrity of the waste emplacement package and shall be applied in such a way that the information shall be legible at least to the end of the period of retrievability. Each waste emplacement package identification shall be consistent with the waste emplacement package's permanent written records (10 CFR 60.135(b)(4))."

In addition, the following constraints apply to the waste emplacement package and are to be accommodated in the design (DOE, 1984):

1. Explosive, pyrophoric, and chemically reactive materials--"The waste emplacement package shall not consist of [sic] (contain) explosive or pyrophoric materials or chemically reactive materials in an amount that could compromise the ability of the underground facility to contribute to waste isolation or the ability of the geologic repository to satisfy the performance objectives (10 CFR 60.135(b)(1))."

2. Free liquids--"The waste emplacement package shall not contain free liquids in an amount that could compromise the ability of the waste packages to achieve the performance objectives relating to containment of high-level waste (because of chemical interaction or formation of pressurized vapor) or result in spillage and spread of contamination in the event of waste package perforation during the period through permanent closure (10 CFR 60 135(b)(2))."

In addition to the requirements and criteria cited for the waste emplacement package, specific requirements and criteria are established for the waste form and container as individual components of the waste package. These are specified as follows (DOE, 1984):

1. Waste Form.

   Functional requirement--"1. Control release if containment is lost during handling."

   Performance criteria--"1. The encapsulating or stabilizing matrix associated with spent fuel or used with reprocessed waste shall be designed to limit the availability and generation of particulates in case of an accident occurring during preclosure (10 CFR 60.135(c)(1) and (2))."

2. Container.

   Functional requirement--"1. The container shall be capable of being handled and shipped as needed."

   Performance criteria--"1. The container shall have lifting studs, the mechanical integrity to sustain routine handling and shipping, and a readily identifiable label giving contents."

7.2.1.2 Generic postclosure requirements

The waste package postclosure functional requirements are specified as follows (DOE, 1984):

"1. To contain the radionuclides for a specified period of time.

"2. To contribute to controlling the release of radionuclides after the containment period."

The corresponding performance criteria for these requirements are specified as follows (DOE, 1984):

1. Radionuclide containment--"The waste package system shall be designed, assuming anticipated processes and events, so that containment of radioactive waste will be substantially complete for a period not less than 300 yr after permanent closure of the geologic repository.* Specific numeric criteria for 'substantially complete' and the time period for containment depend on site-specific characteristics (10 CFR 60.113(a)(1)(ii)(A) and 60.113(b))."

   (In addition, pending revisions to the Generic Requirements document include an additional criterion for radionuclide release control-The waste package subsystem shall be designed, assuming anticipated processes and events, so that the release rate of any radionuclide from all of the waste packages** following the containment period shall not exceed one part in 100,000 per yr of the curie inventory of that radionuclide calculated to be present at 1000 yr following permanent closure; provided that this requirement does not apply to any radionuclide which is released at a rate less than 0.1 percent of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per yr of the curie inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 yr of radioactive decay.)

7.2.1.3 Other design requirements

7.2.1.3.1 Waste package interactions with the emplacement environment

Packages for high-level waste shall be designed so that the in situ chemical, physical, and nuclear properties of the waste package and its interactions with the emplacement environment do not compromise the function of the waste packages or the performance of the underground facility or the geologic setting (10 CFR 60.135(a)(1)).

All the waste package designs involve materials and design configurations that interact with the near-field environment. The in situ chemical,

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* This criterion, as stated in the GR, is incomplete because it does not include the upper limit of 1,000 yr that may be imposed by the NRC as the minimum time of required substantially complete containment.

** The engineered barrier system performance objective of 10 CFR 60.113 is applied to the waste package as a goal for design purposes and a boundary for performance assessment in the GR.

physical, and nuclear properties are being quantified and evaluated for their effects on the long-term performance of the waste package and the repository. The research and development accomplished to date are described in Section 7.4; future plans for work associated with quantification and evaluation of these interactions are outlined in Sections 8.3.4, 8.3.5.9, and 8.3.5.10.

7.2.1.3.2 Technical feasibility

A waste package design requirement derives from the preclosure siting guidelines (10 CFR 960.5-1(a)(3)). This guideline requires that the repository (and therefore the waste packages produced and emplaced therein) shall be demonstrated to be technically feasible on the basis of reasonably available technology and that the associated costs be reasonable. The specific criteria that follow from these requirements are

1. Processes specified for the fabrication, assembly, closure, and inspection of waste packages shall be based on reasonably available technology. These processes need not be reduced to commercial practice in all applicable details but shall not require significant extensions of the technology.

2. Waste package designs shall not impose requirements on the repository packaging, handling, and emplacement facilities, equipment, or operations that are beyond reasonably available technology.

3. In evaluation of design concepts, materials, and process alternatives, consideration shall be given to cost effectiveness. This consideration shall be secondary to realization of designs that will be technically conservative and meet the regulatory performance objectives.

7.2.1.3.3 Waste form temperature limitation criteria

Maximum temperature of the waste forms must be maintained below limits established for them. These limits are 500°C for West Valley (WV) and Defense High-Level Waste (DHLW) glass and 350°C for spent fuel cladding. These values are considered conservative for the following reasons. For spent fuel, the spent fuel storage program report (Einziger, 1986) indicates a 380°C temperature limit for the cladding. A more conservative value of 350°C has been selected to account for uncertainties in source characteristics as well as heat transfer calculations. For the glass waste forms, the NNWSI Project has the responsibility to maintain the peak temperature below the transition temperature. The NNWSI Project has established a 500°C temperature limitation for this reason.

These limitations have been established by the NNWSI Project to reduce the potential for degradation of the waste forms. The limit for high-level waste glass has been determined to prevent devitrification, which might degrade the properties that are important in limiting radionuclide release.

The limit for spent fuel has been determined to limit degradation of the fuel cladding, which could compromise its integrity (O'Neal et al., 1984).

7.2.1.3.4 Inert cover gas in spent fuel packages

An inert gas is to be introduced to replace ambient air in the void volume of all spent fuel waste packages. This requirement is imposed to reduce the potential for oxidation of the fuel that may be exposed to the internal container atmosphere by defects in the cladding. Tests reported by the spent fuel storage program (Johnson and Gilbert, 1984) indicate that extended storage at fuel temperatures up to 380°C in an inert gas does not degrade the fuel or the cladding.

7.2.2 WASTE FORMS

The reference waste forms to be received, packaged, and disposed of in the repository are described in Appendix B, Section B.1 of the Generic Requirements for a Mined Geologic Repository (GR) (DOE, 1984). This appendix describes characteristics such as waste types and forms, repository capacity, receipt rates, spent-fuel age and burnup distribution, physical dimensions, and output characteristics such as decay heat and radiation. The relevant parts of this information are included in the tables in Section 7.3.1, along with additional descriptions of the important variations in the waste forms and the effects that these variations have on the reference designs.

7.2.3 WASTE PACKAGE COMPONENT PERFORMANCE ALLOCATION

For the purpose of allocating performance, the waste packages consist of two components: waste forms (including any structures, canisters, or other means of encapsulation or stabilization) and containers that surround an individual waste form). In addition, since assembled and emplaced waste packages are the "product" of the repository subsystem, some aspects of the preclosure performance must be allocated to the repository facilities, equipment, and operational procedures.

7.2.3.1 Preclosure performance

The preclosure performance of the waste packages in response to the design requirements is allocated as discussed below. The impact of this preclosure requirement allocation on waste package design must be evaluated to ensure that the postclosure performance is not compromised.

Waste package handling. The container is designed to be handled with repository equipment and to provide containment of the waste forms under normal handling loads during transportation (at the repository), emplacement, and retrieval. Under accident loads, the waste package as an entity will assist in the containment of radionuclides but is not designed to provide complete containment under all conditions. The repository facilities and equipment will be designed to provide the containment capability under these conditions.

Criticality control. The waste form geometry and internal container environment combined provide the control, including the required 5 percent margin, under normal conditions once the waste package is assembled (O'Neal et al., 1984). Before final assembly, the waste forms are subcritical unless aggregated and moderated to produce a critical arrangement. The repository facilities and equipment will be designed to preclude this arrangement.

Identification. Each container will provide a means of identification on the end with the lifting fixture. This identifier will be designed to be legible until repository closure and will be traceable through the repository record system to determine the identity of the contained waste forms. Traceability of the waste form identities before waste package assembly will be maintained by the repository operational procedures.

Reactive materials and free liquids. The waste forms, as described, will meet the constraints with the possible exceptions of water present inside failed fuel cladding and pyrophoric fine particulates of some nonfuel hardware components. Additional information on these possible exceptions will need to be obtained from the spent fuel owners as part of the documentation of fuel characteristics.

Waste form release control (Preclosure). For spent fuel, the requirements for acceptance are defined in the Standard Contract for Disposal of Spent Nuclear Fuel and/or High Level Radioactive Wastes (10 CFR Part 961). These contracts between the DOE and the spent fuel owners do not contain any requirement for spent fuel design or condition other than categorization into one of several classes and furnishing documentation to identify and describe the fuel. For the reprocessed wastes from DOE defense program sources, the waste form requirements are the subject of waste acceptance preliminary specifications that have been developed by the DOE Office of Civilian Radioactive Waste Management (OCRWM). Similar requirements are being developed for reprocessed wastes from commercial sources. The intent of this release control requirement will be included in the specifications for both West Valley commercial wastes and defense wastes from the Savannah River Plant. The waste acceptance process has not been initiated for other reprocessed wastes.

Container handling and shipping. The intent of the specific requirement and performance criterion defined for the waste package container and repository handling equipment will be incorporated into the design to be compatible and allow for all anticipated loads. The container will provide for a unique identifier that will provide traceability of the package contents through the repository records system.

Reasonably available technologies. The container will be designed so that it can be fabricated, assembled, closed, and inspected using reasonably available and cost-effective processes. Additional discussion of this topic is in Section 7.3.1.4.

Waste form temperature limitations. The waste package as an entity will be designed to control the internal temperatures to less than the limits under anticipated postemplacement conditions. The repository packaging, storage, and transfer facilities and equipment will provide sufficient heat dissipation to maintain the waste forms below the limits between receipt and emplacement. In addition, because the peak waste form temperatures following emplacement will occur during the preclosure period (typically within 5 to 10 yr after emplacement) the repository emplacement geometry will be designed to ensure that the temperature limits are observed.

7.2.3.2 Postclosure performance

The allocation of postclosure performance to components of the waste package and the engineered barrier system was not completed at the time that the conceptual designs described in this chapter were developed. Therefore, no quantitative allocation is presented here.

The postclosure performance of the waste package is intimately related to the anticipated near-field environmental conditions, because they are perturbed by the engineered features of the repository and the imposed thermal and radiation effects resulting from the waste emplacement. The status of the understanding of the environment is contained in Sections 7.1 and 7.4.1. The response of the package components to the environment will determine the postclosure performance. The research on response of the package components to the environment is described in Sections 7.4.2 and 7.4.3, and the approach to be used in assessing the package performance is discussed in Sections 7.4.4 and 7.4.5.

The strategy to be used in meeting the postclosure performance requirements and the design goals for future waste package and engineered barrier system design activities are discussed in Sections 8.3.4.2, 8.3.5.9, and 8.3.5.10.

7.3 DESIGN DESCRIPTIONS

In this section, the NNWSI Project waste package reference and alternate conceptual designs are described. These design concepts have evolved over time as the characteristics of the waste forms to be received and disposed at the repository have become better defined; the anticipated geologic, hydrologic, and geochemical properties of the repository have been characterized by laboratory and surface-based field investigations; and the conceptual design of the repository has been developed.

Section 7.3.1 discusses the reference designs, including waste forms, container materials, package designs, and fabrication processes. The alternative designs are described in Section 7.3.2. Finally, Section 7.3.3 briefly discusses other waste emplacement hole components that have potential impact on waste package performance.

The reference waste package emplacement environment is described in Section 7.1. The repository conceptual design is summarized in Chapter 6 and further described in the Site Characterization Plan-Conceptual Design Report (SCP-CDR) (SNL, 1987).

7.3.1 REFERENCE DESIGNS

Conceptual designs have been developed for spent fuel and high-level waste packages that represent general configurations and design concepts (O'Neal et al., 1984). Subsequent to that report, several modifications, primarily to internal geometries and stabilizing structures for various spent fuel loading configurations, have been evaluated.

The characteristics of the reference waste forms are described in Section 7.3.1.1 and the metal container materials in Section 7.3.1.2. Section 7.3.1.3 describes the reference waste packages, including their dimensions, internal configurations to accommodate the various waste forms, and their associated thermal decay heat and radiation characteristics. The processes that may be used for fabricating and assembling the waste package containers and the criteria for process selection based on performance are discussed in Section 7.3.1.4.

7.3.1.1 Reference waste form descriptions

Waste forms to be received and packaged for disposal will include both unreprocessed spent fuel from commercial power reactors and canisters of solidified high-level wastes from commercial and defense fuel reprocessing operations. The commercial high-level waste will be received from the West Valley Demonstration Project in New York (WVHLW). High-level wastes from the defense program activities at Savannah River, Hanford, and Idaho may be disposed of in the repository. Characteristics of the waste from the Defense Waste Processing Facility (DWPF) at the Savannah River Plant have been established; definitive information is not yet available for defense wastes from the Hanford and Idaho sites (DOE, 1984).

7.3.1.1.1 Spent fuel

For the purposes of this document, spent fuel is the enriched uranium oxide, the transuranic nuclides, and the fission and activation products resulting from operation of commercial light water power reactors (LWR) and the zirconium alloy or stainless steel cladding, which also contains activation products. Spent fuel may be received at the repository either as intact assemblies or canisters of fuel rods that have been consolidated at the reactors or other facilities. Intact assemblies include many other metallic components, such as end fittings, flow channels, guide tubes, springs, and spacer grids. These nonfuel hardware components will also be packaged for repository disposal. A summary of the quantities of spent fuel expected to be received at the repository is given in Section 6.1.1. After the repository startup phase, a receipt rate of 3,000 metric tons of uranium (MTU) per yr is anticipated. Table 7-2 gives a tabulation of the anticipated burnup distribution and age at repository receipt of spent fuel as a function of time from 1998 to 2020.

Standard spent fuel is defined in 10 CFR Part 961, the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Wastes, as spent fuel with a minimum age of 5 yr after discharge from the reactor. The Standard Contract specifies that the DOE will accept fuel for disposal on an oldest-first basis. During the early yrs of the repository receiving and emplacement period, the average spent fuel age will be greater than 10 yr. This average age will steadily decline and will be down to the 5-yr minimum during the last several yr of operation. The waste package and repository designs must be capable of receiving and disposing standard spent fuel on a routine basis. Fuel cooled less than 5 yr will remain in the transportation and storage system. Table 7-3 presents typical characteristics of 5- and 10-yr old spent fuel to be received at the repository (DOE, 1984).

The burnup of LWR fuel is expressed as the fission (thermal) energy released per unit of initial uranium weight loaded into a reactor core. A commonly used unit is megawatt-d per metric ton uranium (MWd/MTU). The build-up of fission products in the fuel is proportional to the energy generated in the reactor. These fission products and the actinides formed by neutron capture reactions are the principal sources of radioactivity in spent fuel. Thus the spent fuel burnup, together with the fuel age, determines the radionuclide inventory. Radioactive decay of this inventory produces the thermal energy that must be dissipated in the repository.

Fuel in pressurized water reactor (PWR) and boiling water reactor (BWR) assemblies is presently enriched to about 3.2 and 2.6 weight percent fissile U-235, respectively, and is irradiated to achieve burnups of 33,000 and 27,500 MWd/MTU, respectively (DOE, 1984). As nuclear power plants have matured, the average burnup of spent fuel assemblies has increased toward this value. The evolving economics of the nuclear power industry has produced an incentive to increase enrichments to allow higher burnups. A recent survey of fuel vendors (DOE, 1984) indicated that approximately two-thirds of the present United States nuclear power plants have made commitments toward the purchase of fuels with higher enrichments to allow longer fuel residence times in reactor cores, which will result in higher burnups.

As the repository operation draws down the backlog of spent fuel at reactors, the average burnup of the annual receipts will tend to increase, since the lowest burnup fuel is also generally the oldest in storage (DOE, 1984). Thus, the overall average of spent fuel burnup received for disposal is expected to show a generally increasing trend over the life of the repository, and individual assemblies with burnups substantially exceeding the average may be anticipated. For an industry-predicted average PWR fuel burnup of 35,000 MWd/MTU in 2020, a number of cases in excess of the average would occur. If a PWR plant were to discharge a batch of fuel with an average burnup of 42,000 MWd/MTU, a small number of assemblies would be expected to be as high as 60,000 MWd/MTU (DOE, 1984).

Table 7-2. Spent fuel burnups and ages at emplacement normalized to Energy Information Agency (EIA), 1983 midcase projections[a]

Burnup (MWD[b]/MTU) vs. year in MTUs

Discharge year	0-5000 BWR	0-5000 PWR	5000-10000 BWR	5000-10000 PWR	10000-15000 BWR	10000-15000 PWR	15000-20000 BWR	15000-20000 PWR	20000-25000 BWR	20000-25000 PWR	25000-30000 BWR	25000-30000 PWR	30000-35000 BWR	30000-35000 PWR	35000-40000 BWR	35000-40000 PWR	40000-45000 BWR	40000-45000 PWR	45000-50000 BWR	45000-50000 PWR	50000-55000 BWR	50000-55000 PWR	Age (yr) at emplacement	Receipt yr	Total MTU BWR	Total MTU PWR	Total MTU	Cumulative MTU
1961																							37	(1998)			4*	4
1962																							36	(1998)			6*	10
1963																							35	(1998)			10*	20
1964																							34	(1998)			11*	31
1965																							33	(1998)			11*	42
1966																							32	(1998)			11*	53
1967																							31	(1998)			11*	64
1968																							30	(1998)			11*	75
1969					1		7					8											29	(1998)	8	8	16	91
1970								41		8													28	(1998)		49	49	140
1971			14		6	5						40											27	(1998)	20	45	65	205
1972	142		12		31	8	4	45				9		22									26	('98-'99)	189	84	273	478
1973	9		16		36		40	24				40											26	(1999)	101	64	165	643
1974	60		6	7	122	77	31	40	4	14		74											25	('99-'00)	223	212	435	1078
1975			1	9	46	186	155	12	22	71		41		20									25	('00-'01)	224	339	563	1641
1976			49		108	17	149	147		84		91		37									25	('01-'02)	306	376	682	2323
1977			52		34		174	132	102	126		134		86		18							25	(2002)	362	496	858	3181
1978				2	25	51	119	78	206	45	28	402		73		62							24	('02-'03)	438	713	1151	4332
1979					20	26	92	124	161	55	178	279		239		32							24	(2003)	451	755	1206	5538
1980	15				3	9	46		352	55	110	304		226		29							23	(2003)	526	623	1149	6687
1981						26	73		256		162	304		428		16							22	('03-'04)	491	774	1265	7952
1982					15		41	80	30	89	300	161		361		13							22	(2004)	386	704	1090	9042
1983							36	49	42	48	292	296	16	220		51							21	('04-'05)	394	664	1058	10100
1984	14		9		13	52	3	41	49	16	262	118	14	412		94		3					21	(2005)	384	736	1100	11200
1985			55		52	21	15	70	25	11	249	205		380		193		24					20	(2005)	396	904	1300	12500
1986	14		62		57	46	69	127	7	52	382	107	38	245		266		28					19	(2005)	629	871	1500	14000
1987	29		23		34	59	102	48	127	27	255	123	13	379		349		32					18	(2005)	583	1017	1600	15600
1988			33		99	2	120	80	94	19	514	165	44	506		417		107					18	(2006)	904	1296	2200	17800
1989			44		24	27	41	27	159	64	399	165	118	501		435		95				1	17	(2006)	785	1315	2100	19900
1990			30		15		66	27	31	18	480	93	219	542		460		119					17	(2007)	841	1259	2100	22000
1991			3				47		29	46	808	104	226	629		618		190					16	(2007)	1113	1587	2700	24700
1992					3		3		152	1	470	100	272	771		588		139		1			16	(2008)	900	1600	2500	27200
1993							44	60	27	7	740	63	239	594		703		122		1			15	(2008)	1050	1550	2600	29800
1994									64	7	640	63	256	860		553		157					15	(2009)	960	1640	2600	32400
1995									50		699	73	225	825		592		135		1			14	(2009)	974	1626	2600	35000
1996									70	8	790	32	393	815		731		160		1			14	(2010)	1253	1747	3000	38000
1997									64	7	572	38	355	871		717		175		1			13	(2010)	991	1809	2800	40000
1998									56		898	71	173	848		585		169		1			13	(2011)	1125	1675	2800	43600
1999				21					68	30	638	31	379	849		728		155		1			13	(2011)	1085	1815	2900	40500
2000	1						6		31		758	41	386	1022		597		158					12	(2012)	1182	1818	3000	49500
2001						11			117	10	743	87	300	882		841		208		1			11	(2012)	1160	2040	3200	52700
2002									71	8	916	33	403	853		746		169		1			11	(2013)	1390	1810	3200	55900
2003									33		657	48	508	1167		867		219		1			10	(2013)	1198	2302	3500	59400
2004			36			27	45		109	37	844	81	281	1029		711		199		1			10	(2014)	1315	2085	3400	62000
2005									38		1120	42	515	1040		966		178		1			9	(2014)	1073	2227	3900	66700
2006		4	32	50		14	48	40	34	69	611	50	529	1127		863		228		1			9	(2015)	1254	2446	3700	70400
2007	14	17	87	56		66	93	91	114	118	1009	121	285	982		752		194		1			9	(2016)	1602	2398	4000	74400
2008	20	5	177	49	49	128	204	28	119	68	715	157	427	820		767		166		1			8	(2016)	1711	2189	3900	78300
2009		12	124	83				132	41	271	788	2	428	1120		773		226					8	(2017)	1381	2619	4000	82300
2010		18		61	37	67	144	73	80	142	910	45	273	1132		575		142		1			8	(2018)	1444	2256	3700	86000
2011				42		61		103	58	105	1236	2	481	1381		887		142		2			7	(2018)	1775	2725	4500	90500
2012		12	34	18	34	48		131	83	79	810	22	690	1216		754		167		2			7	(2019)	1551	2449	4000	94500
2013	42		160			59	160		196	83	1095	2	417	1318		787		179		2			6	(2020)	2070	2430	4500	99000
2014	34	9		43	188	99		1	244	136	1122	31	569	1108		804		110		2			6	(2020)	2157	2343	4500	103500
2015		3		63	272			63	333	186	898	63	632	1606		838		143					6	(2021)	2135	2965	5100	100600
2016			265	122	81	75	321		156	181	1225	75	277	1627		692		203					5	(2021)	2325	2975	5300	113900
2017	26			82	137			98	218		1140		404	1375		958		82					5	(2022)	1925	2575	4500	118400
2018									107		1214		547	2271		847		214					5	(2023)	1868	3332	5200	120600
2019			109		109		111		210		1591		198	1945		838		189					5-6	(24-25)	2328	2972	5300	128900
2020		27	111			129	164		113	129	1204		364	1613		1053		93					6-7	(25-27)	1956	3044	5000	133900

[a]Source is DOE (1984).
[b]MWD = megawatt-days; MTU = metric tons of uranium; BWR = boiling water reactor; PWR = pressurized water reactor.

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Table 7-3. Characteristics of spent-fuel assemblies[a]

Characteristic	Pressurized water reactor	Boiling water reactor
MECHANICAL CHARACTERISTICS		
Overall length (in.)	149-186	84-179
Width (square assemblies) (in.)	8.1-8.5	4.3-6.5
Fuel rods per assembly	100-264	48-81
Fuel rod diameter (in.)	0.360-0.440	0.483-0.570
Fuel rod length (in.)	91.5-171	80.5-165
Rod pitch (in.)	0.496-0.580	0.640-0.842
MTU[b] per assembly	0.11-0.52	0.19-0.20
Assembly weight (lb)	1280-1450	600
TYPICAL CHARACTERISTICS AS RECEIVED		
FIVE-YEAR FUEL[c]		
Burnup (average conditions) MWd/MTU	33,000	27,500
Actinides and daughters (Ci/MTU)	104,000	93,000
Fission products (Ci/MTU)	453,000	365,000
Decay heat (W/MTU)	1,800	1,400
Photon release (photons/s/MTU)	1.3×10^{16}	1.0×10^{16}
Photon energy release (Mev/s/MTU)	4.8×10^{15}	3.6×10^{15}
Burnup (high condition) MWd/MTU	50,000	
Actinides and daughters	155,000	
Fission products (Ci/MTU)	640,000	
Decay heat (W/MTU)	2,800	
Photon release (photons/s/MTU)	1.9×10^{16}	
Photon energy release (Mev/s/MTU)	7.3×10^{15}	
TEN-YEAR FUEL[d]		
Burnup (average conditions) MWd/MTU	33,000	27,500
Actinides and daughters (Ci/MTU)	83,000	75,000
Fission products (Ci/MTU)	302,000	249,000
Decay heat (W/MTU)	1,100	900

Table 7-3. Characteristics of spent-fuel assemblies[a] (continued)

Characteristic	Pressurized water reactor	Boiling water reactor
Photon release (photons/s-MTU)	7.7×10^{15}	6.2×10^{15}
Photon energy release (Mev/s/MTU)	2.6×10^{15}	2.0×10^{15}
Burnup (high condition) MWd/MTU	50,000	
Actinides and daughters	124,000	
Fission products (Ci/MTU)	442,000	
Decay heat (W/MTU)	1,800	
Photon release (photons/s/MTU)	1.1×10^{16}	
Photon energy release (Mev/s/MTU)	3.8×10^{15}	

[a]Modified from DOE (1984).
[b]Metric tons of uranium.
[c]Spent fuel out of reactor for five years.
[d]Spent fuel out of reactor for ten years.

Spent fuel may be received at the repository in at least three forms. The majority will be in the form of intact assemblies that contain undamaged fuel rods. Some of these fuel rods may have minor cladding breach defects, but would not be structurally damaged to the extent that the fuel is not safely contained. The second form is fuel that has been preconsolidated by being disassembled at the reactors or other facilities, with the rods packaged in canisters whose dimensions approximate those of an intact assembly. The reference consolidation factor is 2:1 (i.e., a canister contains the fuel rods from two fuel assemblies). No reference form for the configuration of nonfuel hardware resulting from these preconsolidation operations has been established. Hence, no provisions have been made for this material in the conceptual designs for either the waste package or repository surface facilities. The third form in which spent fuel is expected to be received is "failed fuel" that has been structurally damaged to the extent that the fuel assemblies must be placed in a canister to contain the particulate fuel materials during handling and shipment from the reactor. No reference dimensions have been established for this category, but it is assumed that the canisters will be only slightly larger than the assemblies they contain.

The reference form for disposal of spent fuel is fuel rods that are removed from intact assemblies and consolidated at the repository. The option of disposing of intact spent fuel assemblies as they are received is retained in the reference design because it may not be possible to consolidate assemblies that are received at the repository in a damaged or failed condition. In addition, current planning envisages loading fuel with very high burnup and short cooling times into containers as intact assemblies in quantities dictated by the waste form temperature limitations.

There is a wide variety of PWR and BWR fuel rod and fuel assembly dimensions. Limits are shown in Table 7-3 (DOE, 1984).

7.3.1.1.2 High-level wastes

High-level radioactive wastes from the West Valley Demonstration Project (WV) and the Defense Waste Processing Facility (DWPF) will be received in solidified form. The radionuclides will be immobilized in a borosilicate glass matrix contained in 304L stainless steel pour canisters. The 61-cm-diameter, 3.0-m-long pour canisters are nominally 1-cm-thick 304L stainless steel and essentially identical for both WV and DWPF (DOE, 1984). because of concerns that the thermal cycles associated with the glass-pouring operation will result in the pour canisters being highly stressed (Baxter, 1983), the pour canisters will not be used as the primary barrier to meet the containment performance objective.

7.3.1.2 Reference container materials

The reference waste package container material for the current conceptual design is AISI 304L stainless steel. The reference alloy system is 300 series austenitic stainless steel and alloy 825, a high nickel, iron-based austenitic alloy. Potential candidate metals initially considered for the reference material included the following alloy groups: austenitic stainless steels, ferritic stainless steels, duplex stainless steels, high-nickel alloys, titanium alloys, zirconium alloys, copper-nickel alloys, low-carbon steels, and cast irons (Russell et al., 1983). Section 7.4.2 has additional details on the candidate alloy selection. The chemical compositions of three alloys in the reference alloy system are given in Table 7-4; a comparative listing for several physical properties of these materials is presented in Table 7-5 (ASTM, 1982).

7.3.1.3 Reference waste package designs

The reference waste packages are designed as thin-walled right circular cylinders with end closures and a lifting fixture on one end. The metal containers are 66 cm diameter with a nominal wall thickness of 1 cm. The diameter was determined on the basis of the geometry of the waste forms and their thermal limitations. The wall thickness is based on structural and handling considerations as well as material degradation rates (Section 7.4.2). The length will vary from about 3.1 to 4.7 m, depending on the waste form dimensions. The total package weights will range from 2.7 to 6.4 metric tons, depending on the quantities and types of wastes in the package.

Table 7-4. Alloy compositions for candidate container materials in reference alloy systems[a]

Common alloy designation	UNS[b] designation	Chemical composition (wt % percent)[c]							
		Carbon	Manganese	Phosphorous	Sulfur	Silicon	Chromium	Nickel	Other element
304L	S30403	0.030	2.00	0.045	0.030	1.00	18.00-20.00	8.00-12.00	N: 0.10 max
316L	S31603	0.030	2.00	0.045	0.030	1.00	16.00-18.00	10.00-14.00	Mo: 2.00-3.00 N: 0.10 max
825	N08825	0.05	1.0	Not specified	0.03	0.5	19.5-23.5	38.0-46.0	Mo: 2.5-3.5 Ti: 0.6-1.2 Cu: 1.5-3.0 Al: 0.2 max

[a] Information adapted from ASTM specifications A-167, B424 (ASTM, 1982).
[b] UNS designation from Unified Numbering System for Metals and Alloys (SAE, 1977).
[c] The values given are maximums except where ranges are given.

Table 7-5. Representative mechanical properties for candidate container materials in reference alloy systems[a,b]

Alloy designation	Tensile strength MPa	Tensile strength psi	Yield strength MPa	Yield strength psi	Elongation percent	Reduction of area percent
304L (annealed)	483	70,000	170	25,000	40	40
316L (annealed)	483	70,000	170	25,000	40	40
825 (annealed)	586	85,000	241	35,000	30	ND[c]

[a]Information adapted from American Society for Testing and Materials specifications A-167, B-424 (ASTM, 1982).

[b]Minimum values are given.

[c]ND--no data specified.

For spent fuel containers, an internal structure will provide mechanical stability and facilitate loading operations. For repository-consolidated rods, preconsolidated rod canisters, or intact assemblies, the internal structure will consist of an array of internal modules fabricated from 304L stainless steel sheet metal 3 to 6 mm thick.

Figure 7-2 depicts the reference designs for a spent fuel disposal container that includes the internal configurations to accept both consolidated pressurized-water reactor (PWR) and boiling-water reactor (BWR) fuel rods and intact assemblies.

The reference waste packages include an internal structure that is configured to accommodate the rods from 6 PWR (configuration 1) or 18 BWR (configuration 2) assemblies. The configurations place the fuel rods in sectors around the perimeter of the container, with the central hexagonal space reserved for placement of compacted nonfuel hardware resulting from consolidation operations at the repository. With effective compaction, this space will accommodate the hardware from all the assemblies whose fuel rods are in the package. There is no requirement that the hardware from the same assemblies as the fuel rods be placed in a common container. This internal configuration (configuration 3) will accept six square canisters of BWR fuel rods that have been preconsolidated (2:1) before receipt, or six intact BWR assemblies. A variation of the internal structure (configuration 4) will accept three canisters of preconsolidated PWR fuel rods, or three intact PWR assemblies. The container can be fabricated to a length to accommodate fuel rods or assemblies of varying lengths, or the container length may be modified to a few standard lengths to match the fuel dimensions.

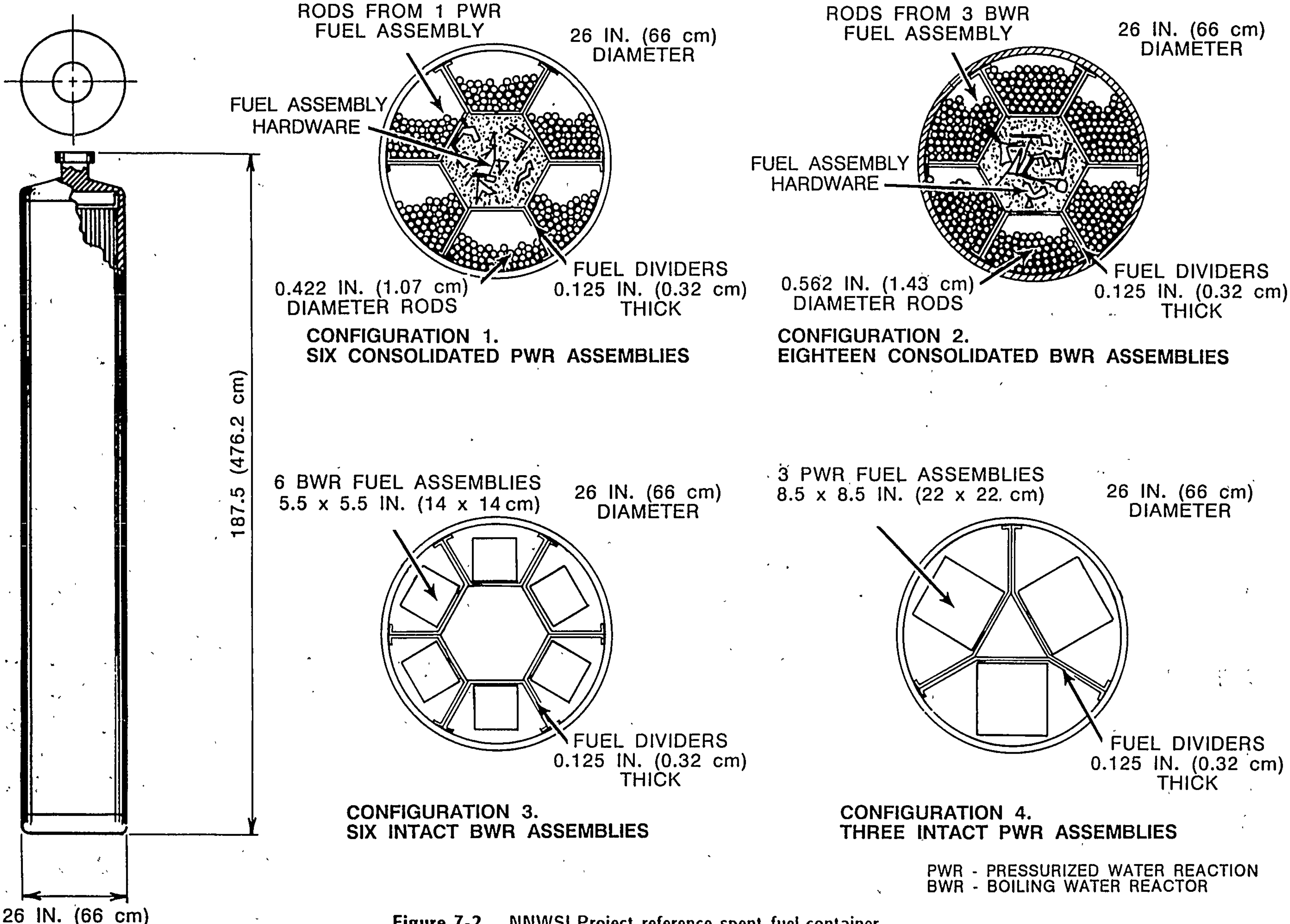


Figure 7-2. NNWSI Project reference spent fuel container.

All waste packages containing spent fuel will be filled with an inert cover gas before final closure. The design concept is to use argon as the cover gas. A small amount of helium may be included if it would be useful in closure integrity inspection.

Spent fuel packages containing nominal burnup, 10-yr-aged PWR or BWR consolidated fuel will have a thermal decay power of about 3.3 kW, based on the reference configuration (Figure 7-2) and the decay heat values from Table 7-3. Initial repository receipts of low burnup fuel aged about 27 yr would result in packages in the reference configuration with power levels of about 1.0 to 1.2 kW. Five yr minimum age high burnup BWR fuel would result in a power level of about 4.5 kW per package. This package design cannot be fully loaded with high (50,000 megawatt-d per metric ton uranium (MWd/MTU) burnup PWR fuel because this would result in a package power level of about 7.9 kW. The heat dissipation characteristics of this package configuration limits it to a maximum thermal power of approximately 5 kW per container. This limit results from the waste form temperature limitation criteria, discussed in Section 7.2.1.3. Thus in the later yrs of repository operation when high burnup, short-cooled PWR fuel will be received, the configuration that accepts intact PWR assemblies would be used. It would have a maximum package power level of about 4.0 kW. As indicated earlier, power per package will vary depending on age, burnup, and quantity of fuel in the package. The gamma dose rate at the outer surface of the package is expected to be approximately 5×10^4 rads/h for the nominal burnup 10-yr-aged fuel. This dose rate will scale approximately linearly with the thermal power output of the package. The neutron dose rate is expected to be about 1×10^4 neutrons/cm^2/s.

The reference waste package design for West Valley (WV) and Defense Waste Processing Facility (DWPF) high-level waste (HLW) is shown in Figure 7-3. This design includes a disposal container very similar to the reference spent fuel container. The 61-cm-diameter WV/DWPF pour canister filled with the glass waste form is installed in a 66-cm-diameter container that is fitted with minimal stabilizers to provide mechanical restraint during handling operations. There is no requirement for the HLW packages to contain an inert cover gas. The reference HLW package design will have a thermal power level in the range of 200 to 470 W, depending on the source and age of the reprocessing wastes in the glass matrix. The gamma radiation dose rate at the outer surface of the package will be about 5.5×10^3 rads/h. The neutron dose rate will be very low.

7.3.1.4 Container fabrication and assembly processes

A number of processes are available for fabricating the component parts of the waste package containers. These range from very conventional and widely used to recently developed, advanced metal-forming processes. An example of the former would be forming the cylindrical body by rolling flat plate and welding the longitudinal seam. The end closures could be cut from flat plate and conventionally welded to the body. More advanced processes include integral forming of the body and end closure by double extrusion, centrifugal casting, or various combinations of forging, drawing, spinning, etc. The top closure, including the lifting fixture and unique identifier,

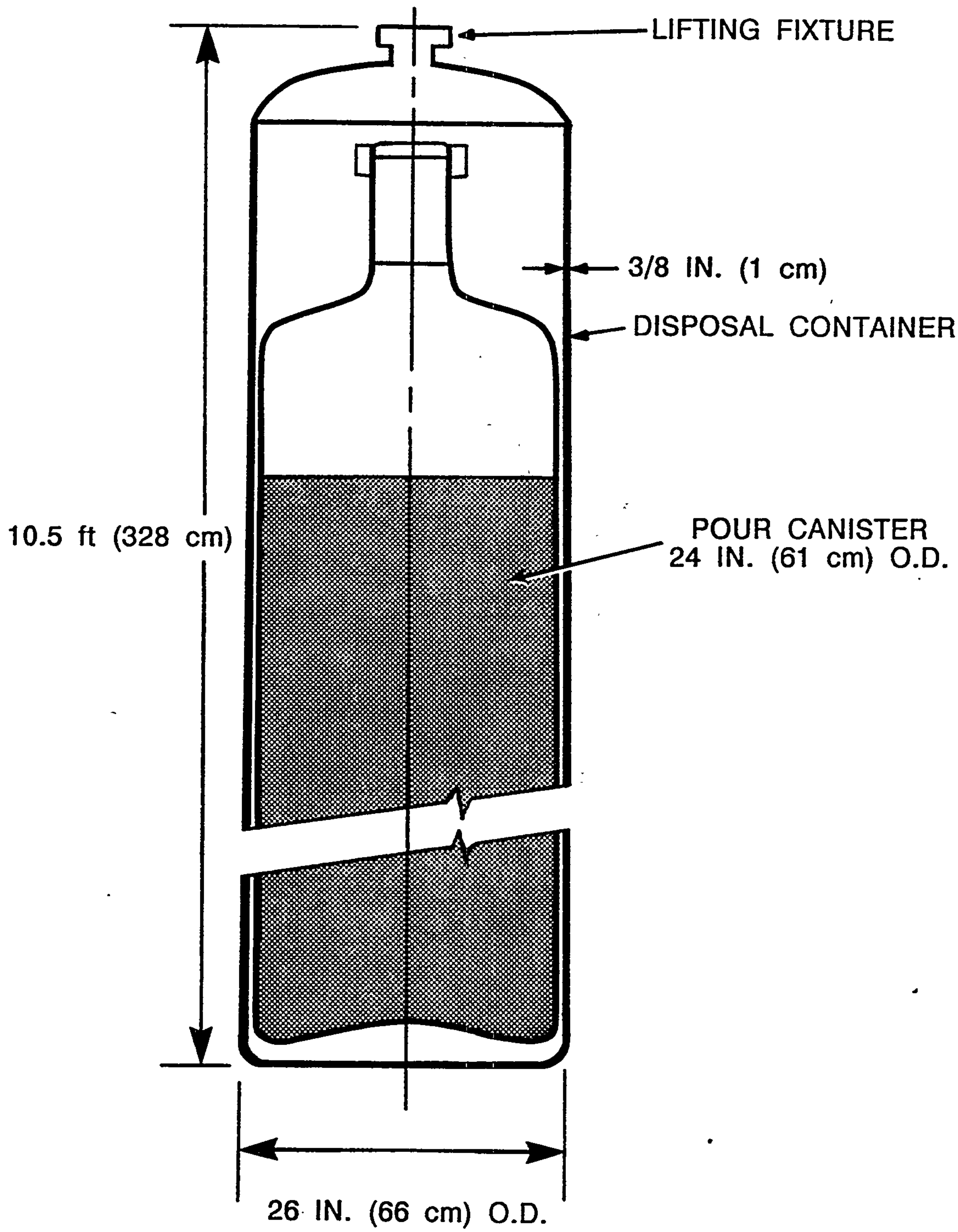


Figure 7-3. NNWSI reference West Valley and defense high-level waste package.

can be fabricated in separate pieces and joined or integrally formed by machining, forging, or casting processes. Following the forming and joining operations, the container components can be machined and/or heat treated as appropriate. Any of the conventional inspection and nondestructive evaluation (NDE) techniques can be used to verify the quality of the components. These fabrications will be produced and inspected at commercial facilities to comply with approved specifications. Finished components will be supplied to the repository for loading the waste forms and for final assembly and inspection.

The final assembly and closure of the container after loading with the waste form requires a metal-joining process. A number of potential processes are available; these range from seal-welded threaded joints to a wide variety of conventional welding processes such as gas metal arc to advanced autogenous techniques such as laser or friction welding. This closure will be made in shielded repository surface facilities. The process must be amenable to remote operation with very high reliability and low maintenance. The completed joint must also be capable of inspection by NDE techniques. The NDE technique must also be amenable to remote operation in a highly radioactive environment with high reliability and must minimize false-positive indications that would result in unnecessary rework.

Fabrication and closure processes have not been selected. The selections will depend on the details of the design as it evolves during the advanced conceptual and license application design phases. The primary criteria for process selections will be based on optimization of the long-term performance of the containers in the repository environment. This will require evaluation of the processes for ability to produce assemblies with high structural integrity, homogeneous metallurgical microstructures, and minimum residual stresses. Other criteria will include consideration of demonstrated reliability, amenability to inspection and flaw detection, flexibility to accept design modifications, vendor availability, production capacity, and cost effectiveness. Plans for evaluation, selection, and development of fabrication, closure, and inspection processes are described in Section 8.3.4.4.

Waste package assembly processes are obviously dependent on the type and geometric configuration of the waste form being packaged. Loading procedures for repository consolidated spent fuel will entail insertion of the spent fuel rods, either vertically or horizontally, into a container with a preassembled internal structure. For canistered preconsolidated spent fuel, or intact spent fuel assemblies, the individual canisters or assemblies will be inserted into the disposal container. After the inert cover gas is introduced, the final closure will be made. The WV/DWPF waste canisters will be loaded into the containers and the final container closure and inspection will be completed. Provision will be made for disassembly and transfer of the contents of completed containers that fail final inspection. Containers suspected of being damaged in handling at the repository after final inspection is completed would be reinspected and, if appropriate, disassembled, and the contents would be transferred to new containers. Additional descriptions of the packaging processes and flow sequences in the repository are found in Section 6.2.3.

7.3.2 ALTERNATIVE DESIGNS

Prior to selection of the reference alloy system for the waste package container material, a number of potential candidate metals were initially considered (Russell et al., 1983). At the time the reference conceptual designs were developed, two alternative designs were also studied. One was a conceptual design using the reference container material for a spent fuel package that supports an alternative disposal strategy for intact fuel assemblies. The other was a conceptual design that changed the container material, based on a feasibility assessment (Acton and McCright, 1986), to a different alloy system, namely a copper-based alloy. Sections 7.3.2.1 and 7.3.2.2 describe the status of these two alternative designs. Waste package design concepts, developed during the ACD phase, will be evaluated as part of the design process described in Section 8.3.4.3.

7.3.2.1 Intact spent fuel package alternative design

At the direction of DOE, the NNWSI Project has evaluated the technical and cost implications of changing the reference strategy for disposal of spent fuel from one using disassembly and consolidation of all possible fuel assemblies to one where all spent fuel received at the repository as intact assemblies is packaged as received without consolidation (O'Brien, 1986).

As an adjunct of this study, an alternative waste package design concept was developed. The approach adopted for this design concept was to retain the basic concepts of the reference design (i.e., a thin-walled cylindrical container with an internal structure to provide support for the contents). Modifications to the external diameter and internal geometry were examined to improve the packing efficiency of nominally square cross-sectional fuel assemblies in a circular cross-sectional container. A further constraint on the design is the mix of PWR and BWR fuel assemblies to be received. This mix averages approximately 40 percent PWR and 60 percent BWR assemblies.

Ideally, the waste package configuration would maintain that mix. The geometry of the situation, however, is best satisfied by the arrangement shown in Figure 7-4 as configuration 1, which places three PWR and four BWR fuel assemblies in an array. This so-called hybrid package concept leads to a surplus of BWR assemblies. Another internal arrangement, within the same 71-cm external diameter, is shown in Figure 7-4 as configuration 2. This places 10 BWR assemblies in the container and provides a means of disposing of the excess BWR assemblies. Only about 7 percent of the total number of packages would need to be of this all-BWR configuration to accommodate the excess number of assemblies.

For nominal burnup, 10-yr aged fuel, these packages would have power levels of 2.3 and 1.6 kW, respectively, for the hybrid and all-BWR configurations. These power levels would vary from about 1.0 to 4.0 kW over the reference ages and burnups projected for spent fuel receipts.

Further development of this package concept will depend on the program guidance resulting from the OCRWM consolidation strategy study in progress.

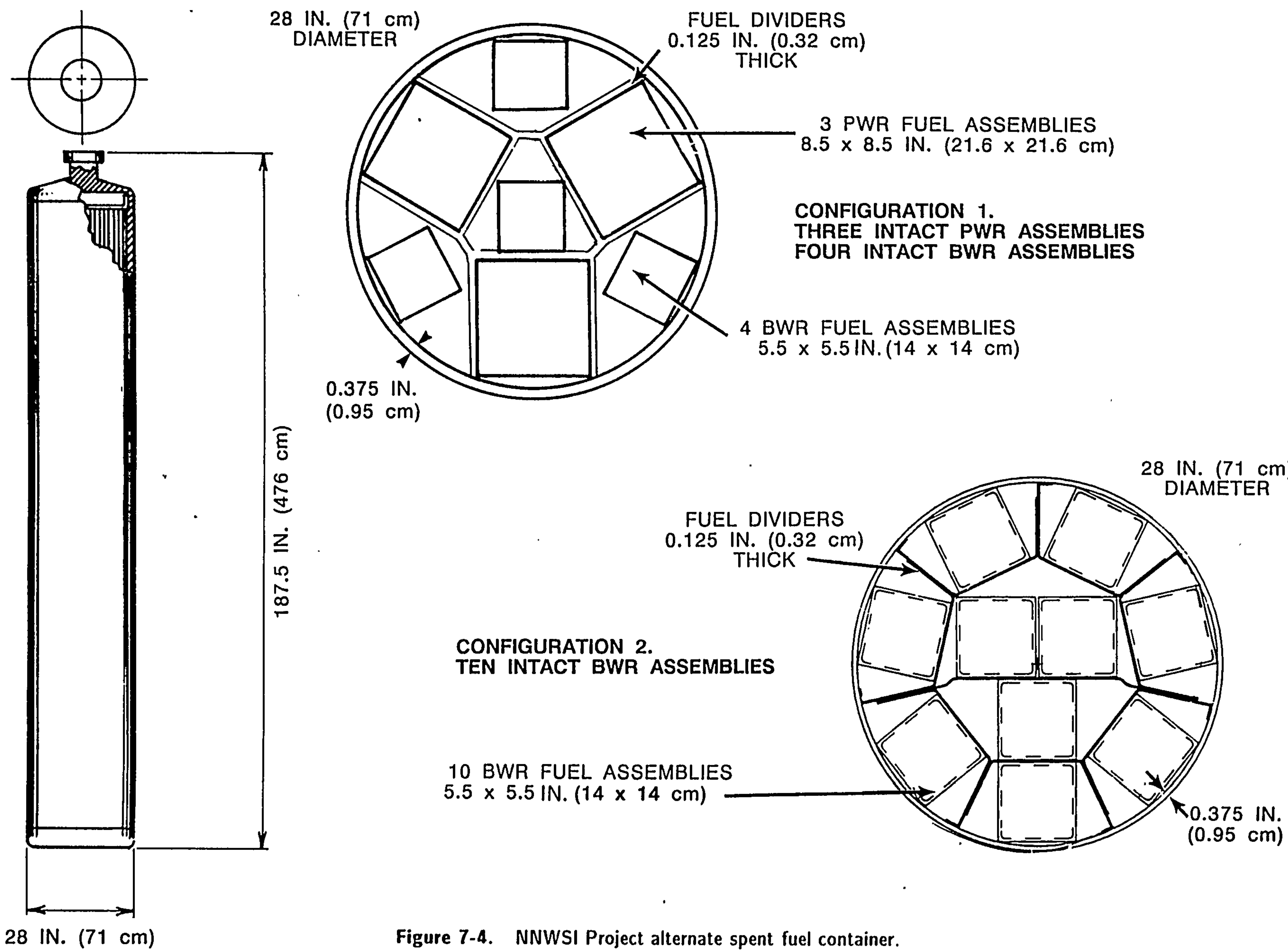


Figure 7-4. NNWSI Project alternate spent fuel container.

7.3.2.2 Copper-based alloy container design

At the time the reference designs were developed, the only other alternative design under active consideration involved the replacement of the container material with a copper-based alloy. Copper and its alloys serve as the alternative alloy system for container fabrication. A different set of corrosion-related concerns is applicable for this alloy system (Section 7.4.2.9). As with the austenitic stainless steels, several copper-based materials are being investigated on the basis of their resistance to specific forms of corrosion. The candidate materials include oxygen-free, high-conductivity copper (CDA 102), aluminum bronze (CDA 613), and 70-30 copper-nickel (CDA 715). The two copper-based alloys were selected for their expected improved corrosion resistance under oxidizing conditions. The chemical compositions and some mechanical properties of these materials are tabulated in Tables 7-6 and 7-7.

The alternative copper container designs are similar to the reference designs in Figures 7-2 and 7-3 but may require a thicker wall to compensate for the lower strength of copper, especially at elevated temperatures (O'Neal et al., 1984). Containers fabricated from copper-based alloys whose mechanical strength properties approach those of the austenitic stainless steels would be essentially the same thickness as the reference containers.

The fabrication and closure processes associated with this alternative material will be assessed as a part of the container production process, evaluation, and selection (Section 8.3.4.4).

7.3.3 OTHER EMPLACEMENT HOLE COMPONENTS

In addition to the waste packages, other components will be present in the reference vertical or alternative horizontal emplacement holes. Because of their proximity to the emplaced waste packages, they have the potential for altering the package environment and impacting the package performance. These other components, borehole liners, emplacement hole shielding plugs, and emplacement dollies are briefly described as follows.

7.3.3.1 Borehole liners

The waste packages will be placed in the emplacement holes with no additional metal containment barriers. Although borehole liners are not considered as a containment barrier, they are discussed in the Site Characterization Plan-Conceptual Design Report to facilitate emplacement and to aid in ensuring retrievability (SNL, 1987). A partial liner is proposed for vertical boreholes in which single waste packages are placed. A full liner is proposed for horizontal boreholes that would contain multiple waste packages in each hole. The presence of a liner will modify the near-field package environment and could influence the life of the waste package. A schedule for retrievability has been used that requires a nominal design service life of about 100 yr for the liners. The construction material for

Table 7-6. Alloy compositions for candidate container materials in the alternative alloy systems[a]

Alloy designation	UNS designation[b]	Chemical composition (wt %)							
		Cu	Fe	Pb	Sn	Al	Mn	Ni	Zn
CDA 102 (Oxygen-free Copper)	C10200	99.95 (minimum)	--[c]	--	--	--	--	--	--
CDA 613 (Aluminum Bronze)	C61300	92.7 (nominal)	3.5 (maximum)	--	0.2-0.5	6.0-8.0	0.5 (maximum)	0.5	--
CDA 715 (70-30 Copper-nickel)	C71500	69.5 (nominal)	0.4-0.7	0.5 (maximum)	--	--	1.0 (maximum)	29.0-33.0	1.0 (maximum)

[a]Compiled from CDA Standards Handbook Data Sheets (CDA, 1986), Copper Development Association, Greenwich, CT.
[b]UNS designation from Unified Numbering System for Metals and Alloys (ASTM, 1982).
[c]-- = Not specified.

Table 7-7. Representative mechanical properties for candidate container materials in alternative alloy systems

Common alloy designation and condition	Yield strength[a] (ksi)	Tensile strength (ksi)	Elongation (percent)
CDA 102			
Hot rolled	10	34	45
Hard	45	50	4
CDA 613			
Soft Annealed	40	80	40
Hard	58	85	35
CDA 715			
Hot rolled	20	55	45
Half hard	70	75	15

[a]0.5 percent extension under load. Compiled from CDA Standards Handbook Data Sheets (CDA, 1986).

the borehole liners will be selected to avoid adverse long-term package performance implications. The liner will probably consist of welded sections having a wall thickness of approximately 0.6 to 1.0 cm based on the expected maximum load imposed by any rock that sloughs from the borehole walls or based upon loads imposed during liner installation.

7.3.3.2 Emplacement hole shielding plugs

Regardless of the emplacement orientation, each emplacement hole will be equipped with a solid plug to prevent radiation streaming from the waste packages into the emplacement drift. The required service life of these plugs and any mating collar hardware is until permanent repository closure. A nominal design lifetime of about 100 yr is therefore appropriate. As with the liners, the construction material of the plugs must not adversely impact the long-term package performance. This is particularly important in the reference vertical emplacement orientation where corrosion products or other plug constituents could fall onto the waste packages under gravity. There will be similar constraints in material selection for the support plate planned for use in the bottom of vertical boreholes.

7.3.3.3 Emplacement dollies

For the horizontal borehole emplacement orientation, a dolly or sled type fixture will be used to permit emplacement of the waste packages without mechanically abrading, gouging, or otherwise degrading the container surface. This fixture will be designed to support the package approximately centered in the liner. This fixture is planned to provide annular air gap around the container and will help avoid contacting the waste package with liquid water should condensation occur on the interior surface of the liner. This fixture will require careful attention to design details and materials of construction to avoid adversely impacting the long-term performance of the package components.

7.4 WASTE PACKAGE RESEARCH AND DEVELOPMENT STATUS

Waste package research and development was conducted as a generic program until 1982. The studies emphasized salt emplacement media, with some work directed toward basalt. No materials performance data directly applicable to tuff as a repository host rock were obtained during the generic phase of the program. In October 1982, the materials testing work was transferred to the site-specific projects. The NNWSI Project waste package development task was then structured to provide data describing the effects of waste emplacement in tuff on the properties of the emplacement environment and on the performance of waste package components in that altered environment. The development of detailed physical and chemical models, which are needed to extrapolate measured performance of materials to the time scales relevant to waste disposal, is considered an integral part of the materials testing work under Issues 1.4 (Section 8.3.5.9), 1.5 (Section 8.3.5.10), and 1.10 (Section 8.3.4.2). The system model to be used to assess performance of the waste package will use the results from the detailed models developed under materials testing tasks, which is a need identified in Issues 1.4 and 1.5.

The status of research and development work needed to describe the post-emplacement history of the waste package environment is discussed in Section 7.4.1. The work on metal barriers is covered in Section 7.4.2. The results of waste form testing using spent fuel and borosilicate glass is summarized in Section 7.4.3. Section 7.4.4 discusses the development of a geochemical modeling code to be used in conjunction with the system model. The final section, Section 7.4.5, briefly describes the performance assessment model of the waste package system.

7.4.1 WASTE PACKAGE ENVIRONMENT MODIFICATION DUE TO EMPLACEMENT

Construction of a repository and emplacement of heat and radiation generating waste in the repository would cause changes in the physical and chemical characteristics of the environment. Design of the waste package and prediction of the performance of the package and its components can only be accomplished if there is a thorough understanding of change with time of the environment as affected by the repository construction and waste emplacement.

Definition of the waste package environment involves detailed description of the preemplacement (ambient) conditions at the proposed repository horizon and determination of the changes that will take place after the waste packages are emplaced. The effects that can be expected from development of the repository and emplacement of the waste packages are (1) physical changes in the rock unit due to mining activities; (2) changes in the physical and chemical properties of the rock and vadose water due to the waste-related radiation field; (3) heat-induced mechanical effects; (4) modification of moisture conditions due to mining ventilation; and (5) modification of the ambient rock-water system and hydrodynamic regime due to the thermal load generated by the waste packages. Research that defines the waste package environment addresses Information Need 1.10.4.

The radiation field from the waste forms will consist of alpha, beta, neutron, and gamma radiation. Alpha and beta radiation will be contained within the waste package because of the short penetration range of these particles. Neutron and gamma radiation will penetrate through the container and interact with the environment in the immediate vicinity of the waste package. Neutrons can cause damage to material by displacing atoms as a result of atomic collisions. The damage from atomic displacements in the host rock minerals would be very small at the flux levels present in the waste forms and would not significantly affect the properties of the material (Grasse et al., 1982; Van Konynenburg, 1984). The neutrons would not significantly affect the water chemistry because the flux would be small (Wilcox and Van Konynenburg, 1981). Gamma radiation interacts with matter by ionization mechanisms. The effects of gamma interaction with tuff largely will be transient and are not expected to cause significant changes in the rock or mineral properties (Durham et al., 1985). Gamma radiation can cause changes in water and vapor chemistry due to the production of radiolytic species. The research conducted in this area is described in Section 7.4.1.4.

The thermal load imposed by the waste packages on the near-field environment will be a function of the waste form and the loading density. The actual thermal output for the spent fuel will depend on burnup history, the time out of reactor, and whether consolidated fuel rods or intact fuel assemblies are stored in the container. Thermal output of the glass waste forms will depend on the age and composition of the waste and its concentration in the glass. The resulting rise in temperature of the environment would alter the local hydrologic system and would cause the pore water contained in rock in the vicinity of waste packages to vaporize. After dehydration, rehydration of the near-field environment will occur as the waste package temperature drops. Performance of the container and the waste forms is closely tied to the amount of water in the system and to the flow mechanisms operating in the vicinity of the waste packages. The rates and mechanisms of dehydration and rehydration of Topopah Spring tuff are being investigated in laboratory experiments using intact and fractured rock samples. This work is discussed in Section 7.4.1.5. Numerical modeling associated with near-field fluid flow is discussed in Section 7.4.1.6.

The elevated temperature in the host rock would also encourage rock-water interactions to occur and would thus affect the chemistry of any water in the system. To assess the potential for changes in water chemistry, a series of tests have been conducted at temperatures from 90 to 250°C using a variety of Topopah Spring tuff samples and a representative ground water. This work is described in Section 7.4.1.7. Geochemical modeling of the rock-water reactions is presented in Section 7.4.1.8.

7.4.1.1 Stability of borehole openings

Stability of emplacement borehole openings is of concern (1) during construction, (2) after emplacement but before closure when the option to retrieve the waste package must be maintained (Section 6.4.8), and (3) for the 1,000-yr period after closure when substantially complete containment must be achieved. The anticipated loads resulting from borehole instability are needed as design input to the packages and borehole liners. The stability of the emplacement borehole during postclosure is being examined as part of the definition of the waste package environment (Section 8.3.4.2). Preclosure retrievability-related analyses are discussed in Section 6.4.8 and in Section 8.3.2.5.

Emplacement hole thermomechanical analysis is conceptually much the same for either vertical or horizontal emplacement options. Loading of packages could result from either failure of intact rock (matrix) or by movement of rocks along preexisting discontinuity planes. Loading resulting from movement along discontinuities is more likely than from failure of intact rock since discontinuities are preexisting planes of weakness within the rock mass. Discontinuity planes in the rock mass intersect to form discrete rock blocks, some of which can move by sliding, falling, or toppling. Sliding and falling involve translational movement of a block into the emplacement hole, while toppling involves rotational block movement. Rotational motion requires more room for block movement than translational motion and, therefore, may not be as likely to occur in the limited confines of an emplacement hole.

An analysis of intact rock behavior based on engineering mechanics principles will be used to estimate or bound the loading condition. Analysis of deformation involving rock mass discontinuities may require three-dimensional methods because of the geometry of the problem. This analysis includes identification of block geometries that could potentially move and evaluation of their potential motion. The necessary techniques have been developed and tested (Yow, 1985) and will be applied to the Yucca Mountain site when definition of the fracture characteristics is available. The application of these techniques to borehole stability is discussed in Section 8.3.4.2; similar techniques will also be applied in designing other underground openings of the repository as discussed in Section 8.3.4.2. Analysis of thermomechanical response during the postemplacement period will consider stress changes from heating, cooling, and possibly long-term creep. A plan for how the analyses will be performed is discussed in Section 8.3.4.2.

The primary effect of temperature changes on the near-field rock will be a change in rock volume due to isobaric thermal expansion. Except for quartz and cristobalite, the isobaric thermal expansion of most silicate minerals, on average, is 3×10^{-5}/K (Helgeson et al., 1978). The maximum temperature attained by the near-field environment, based on the modeling studies discussed in Section 7.4.1.2, results in a total volume change of about 1 percent using the average thermal expansion value.

Cristobalite, which is present in the near-field rock, undergoes a structural transition from tetragonal (alpha-cristobalite) to cubic (beta-cristobalite) symmetry. This phase transition results in a volume increase of about 5 percent (Helgeson et al., 1978). The temperature at which this phase transition occurs has been measured for naturally occurring cristobalite at Yucca Mountain (Wolfsberg and Vaniman, 1984) and has been found to be 225 ±25°C. Within the experimental uncertainty, this temperature is in agreement with that obtained by other laboratories.

The alpha-to-beta cristobalite transition temperature falls within the temperature range expected for the very near-field waste package environment during the period immediately following emplacement. The effect of the associated volume change on the waste package environment has yet to be established. There is additional discussion of this subject in Chapter 2. Research addressing this issue is discussed in Section 8.3.1.4.

7.4.1.2 Anticipated thermal history

Waste package conceptual designs suggest that spent fuel waste packages generate a thermal output of between 1.3 and 3.3 kW per container, but processed high-level waste in the form of borosilicate glass may have an output of less than 0.5 kW per container (O'Neal et al., 1984; Baxter, 1983). Packages that contain spent fuel with higher burnup or younger age than that assumed in the conceptual design studies may have higher thermal output.

Figure 7-5 shows a typical modeled thermal history of a vertically emplaced spent fuel waste package and its immediate surroundings. For an unventilated emplacement drift, the rock at the borehole wall would reach a maximum temperature of 230°C at 9 yr after package emplacement. The rock temperature 1 m from the borehole wall would peak at 190°C about 10 to 20 yr after emplacement. The values of the thermal maxima and the time at which they are attained are sensitive functions of the container power output, the mode of heat transfer, the thermal properties of the near-field rock, and the specific configuration of the boreholes and emplacement drifts. The thermal maximum will be followed by an extended period of cooling that will last for hundreds of years (O'Neal et al., 1984; St. John, 1985). Other related heat transfer analyses are summarized in Section 6.4.8.

At the elevation of the repository horizon, the unconfined boiling point of pure water is approximately 96°C. The pore water in the repository rock is likely to be a dilute solution. Although concentration of the solute species may occur during dehydration and rehydration, elevation of the boiling point is slight (less than 1°C) (DePoorter, 1986; Montan, 1986), even for solutions that are as much as 100 times more concentrated than the reference ground water (well J-13; Section 7.4.1.3). Such solutions exceed the maximum concentration expected for solutions in the waste package environment (Morales, 1985).

Vaporization of unconfined and unbound pore fluid can therefore be assumed to be complete at approximately 97°C. This phenomenon will result in the development of a dehydration zone around the waste package, the width and duration of which will depend on the migration behavior of the boiling point isotherm. The model presented in Figure 7-5 suggests that during the period of substantially complete containment, which will last at least 300 yr, rock within 1 m of the spent fuel waste packages will be at temperatures in excess of the boiling point and will therefore be in the dehydration zone.

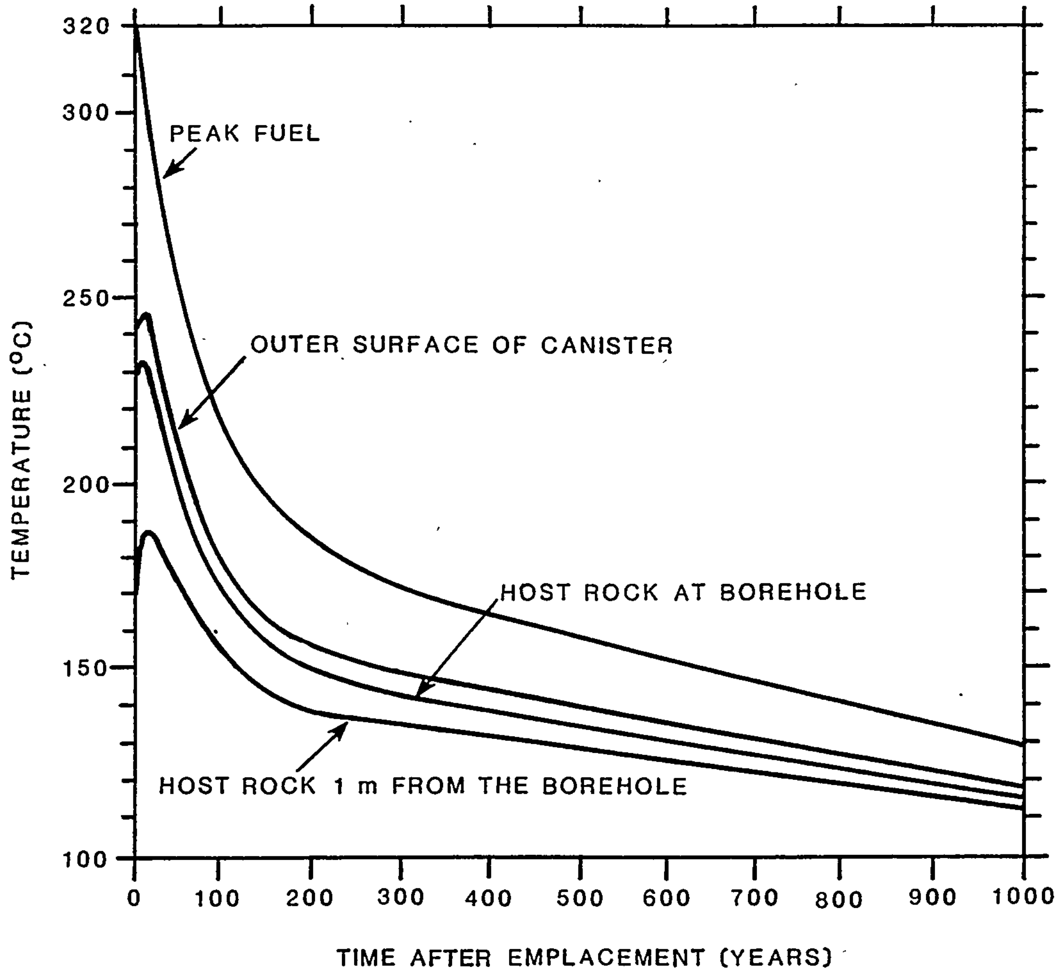
Modeling of the effect of thermal perturbation on local hydrologic transport has suggested that heat transfer through liquid-vapor cycling in convective cells (heat pipe effect) may occur in a closed system (Preuss et al., 1984). Experimental studies (Section 7.4.1.5) and numerical analyses (Section 7.4.1.6) are being conducted to determine the extent to which such behavior will occur in a natural system and the effect of this behavior on near-field water chemistry.

The scenario for thermal evolution of the near-field waste package environment is derived from computer codes for which some uncertainties remain. The scenario assumes that the waste package being modeled resides in the interior of a repository. Waste package environments along the perimeter of a repository will have a thermal history different from that depicted in Figure 7-5. The nature of the thermal history along the perimeter of the repository has yet to be defined. The codes assume no nonlinear effects in the near field and do not consider the effect of pore water vaporization. The histories presented in Figure 7-5 must therefore be considered approximations of the actual thermal field that will be attained in the waste package environment for a central portion of the repository. The greatest uncertainties pertain to the environment immediately adjacent to the borehole wall where nonlinear effects are expected to be most pronounced (St. John, 1985). Beyond a few meters from the borehole wall, the computed results are expected to closely represent the actual thermal history for the set of specified conditions.

Additional near-field thermal calculations, using techniques that improve the modeling of nonlinear effects will be undertaken in the future and are described in Section 8.3.4.2.

7.4.1.3 Reference water for experimental studies

The composition of pore fluid in the repository site will be established when the exploratory shaft test program recovers rock from the repository horizon. Evidence suggests that tests conducted prior to completion of the exploratory shaft can closely approximate repository pore fluid chemistry by use of water recovered from well J-13, which is located to the east of Yucca Mountain. The characteristics of water from well J-13 are discussed in Section 4.1.2. The elevation at well J-13 is lower than at Yucca Mountain, and the Topopah Spring tuff lies below the water table there, forming the major producing horizon for the well. The chemistry of well J-13 water is similar to that found in drillholes drilled to sample water from below the water table at Yucca Mountain (Table 7-8; Ogard and Kerrisk, 1984). Water from well J-13 is also similar to unsaturated zone water obtained from tuffs



INITIAL CONDITIONS

WASTE FORM . . . . SPENT FUEL
LOCAL POWER DENSITY . . . . 57.0 kW/acre
AREAL POWER DENSITY 48.4
AVERAGE 10-YR POWER . . . . 3.3 kW
CONTAINER DIAMETER . . . . 0.7 m
DISTANCE BETWEEN CONTAINERS . . . . 5 m
DISTANCE BETWEEN DRIFTS . . . . 47 m

Figure 7-5. Example of temperature histories of thermal waste package components and host rock for a vertically emplaced spent fuel container (O'Neal et al., 1984).

Table 7-8. Compositions[a] of various unsaturated-zone water from Rainier Mesa compared with well J-13 water[b]

Constituent	White et al. (1980) Interstitial	White et al. (1980) fracture	Henne (1982) Tunnel water	Ogard and Kerrisk (1984) (well J-13)
	Source of data			
Na	1.73	1.53	2.30	1.96
K	0.18	0.12	0.11	0.14
Ca	0.27	0.21	0.08	0.29
Mg	0.10	0.06	0.01	0.07
HCO_3	1.14	1.61	2.25	2.34[c]
SO_3	0.43	0.15	0.10	0.19
Cl^-	0.75	0.24	0.18	0.18
SiO_2	0.97	0.88	0.73	1.07
pH	7.8	7.5	7.0	6.9

[a]Values are averages of reported data.
[b]All species compositions are in moles/L.
[c]Measured alkalinity.

at Rainier Mesa, which is north of Yucca Mountain (White et al., 1980; Henne, 1982). Thus, the water chemistry for well J-13 water is expected to be close to that of the vadose water in the Topopah Spring tuff in the unsaturated zone. Until samples of water from the unsaturated zone are available, the water from well J-13 has been adopted as a reference ground water for the NNWSI Project experimental work. When the composition of pore fluid in the repository horizon is established, the need to complete additional experimental work will be evaluated.

7.4.1.4 Radiation field effects

The types of ionizing radiation that interact with the rock-water-vapor system will be neutron and gamma radiation; alpha and beta radiation will not penetrate the intact waste container (Van Konynenburg, 1984). The absorbed dose from gamma radiation will dominate over that from neutron radiation by more than four orders of magnitude (Van Konynenburg, 1984). The total radiation field will be less than 1 x 10^5 rads/h (Wilcox and Van Konynenburg, 1981). The dominant gamma radiation effects in the waste package environment will result from interaction of gamma radiation with water, steam, and air since gamma radiation at the expected doses results in negligible damage to silicate rock (Durham et al., 1985). Less than 3 percent of the total thermal energy released by the waste package will be deposited in the host rock by gamma radiation (Van Konynenburg, 1984). Over 99 percent of the gamma radiation energy will be deposited within 1 m of the borehole wall (Van Konynenburg, 1984).

The significant radiolysis products in the waste package environment will vary with time as the thermal conditions evolve. The thermal history of rock adjacent to the waste packages will be complex (Figure 7-5, Section 7.4.1.2). Vaporization of the pore water will occur in rock immediately adjacent to the waste packages where the temperature exceeds 96°C. The width of the dehydration zone will depend on the magnitude of the thermal load imposed by the waste package on the host rock. While the rock temperature exceeds 96°C and the dehydration zone extends more than about 1 m into the rock surrounding the waste package, radiolysis products will be restricted to those resulting from interaction of gamma radiation with moist air.

Radiolysis products expected in the moist air system, although known to be temperature dependent, are not well established (Van Konynenburg, 1986). Theoretical and experimental analysis of the moist air system (Jones, 1959; Tokunaga and Suzuki, 1984) and of the tuff-water-steam system (Van Konynenburg, 1986) and consideration of the thermodynamic properties of radiolytic products (Forsythe and Giaque, 1942; Beattie, 1967) suggest that, at temperatures below 120°C, the most abundant products with relatively long lifetimes are HNO_3, N_2O, and a small amount of O_3. Between approximately 120 and 135°C, NO_2, N_2O_4, N_2O, and O_3 are expected to be the dominant chemical products; above approximately 135°C, the dominant product species would be NO, NO_2, N_2O, and O_3. How these compounds would interact with rock in the waste package environment and with the waste package will be established in activities described in Section 8.3.4.2.

When liquid water and air in contact with each other are irradiated, the identities and amounts of radiolysis products depend on the identities and concentrations of solutes in the water, the radiation dose, the oxidation-reduction conditions, the pH, and the air-water ratio. In well J-13 water that has equilibrated with tuff, solute concentrations on the order of 1×10^{-3} to 9×10^{-2} M are obtained, and the dominant species is sodium bicarbonate, buffering the pH near neutral. These concentrations are sufficient to cause the solutes to dominate the reactions with radiation-produced free radicals in the water, leading to increases in hydrogen and oxygen partial pressures due to decomposition of water at high (10^5 rads/h) radiation levels (Van Konynenburg, 1986). In addition, fixation of nitrogen from the air due to irradiation causes nitrite and nitrate ions to be added to the solution. The nitrite-nitrate ratio apparently depends on the oxidation-reduction conditions and on the abundance of materials that can catalyze the breakdown of hydrogen peroxide (e.g., iron(3+) and manganese oxide). Addition of hydrogen ions to the solution also occurs, but the pH is not strongly affected provided the total dose and the air-water ratio are such that the bicarbonate buffering capacity is not exceeded. If the solution is in contact with the rock, additional buffering is provided by the feldspars (Garrels and Howard, 1957).

The chemical effects of the radiolysis products on the initially dehydrated waste package environment host rock will be limited to oxidation of mineral phases containing elements of variable valence states. The principal reactions will involve biotites, amphiboles, and oxides, which individually occupy less than 0.1 volume percent of the rock. The reaction products are expected to be various oxide phases, but little information is currently available on experimental studies relevant to this system. At any given temperature, the degree of oxidation will depend on the reaction kinetics and the concentration of the oxidizing species, neither of which can be quantified at this time (Section 8.3.4.2).

By the time rehydration of the host rock occurs within 1 m of the waste package, the gamma radiation flux will have decayed to levels three orders of magnitude below initial emplacement values (Baxter, 1983; Oversby, 1984b). Concentrations of radiolysis products will, therefore, also be diminished considerably. Further work is planned that will more thoroughly characterize the geochemical consequences of the radiolysis products for the host rock mineralogy and pore fluid chemistry (Section 8.3.4.2).

7.4.1.5 Thermal effects on water flow in the vicinity of waste packages

The present level of understanding of the ambient hydrologic system at Yucca Mountain has been discussed by Montazer and Wilson (1984) and Ortiz et al. (1985). These papers document that the potential repository horizon is located within a nonlithophysal unsaturated zone that is several hundred meters above the water table.

Water transport within this rock occurs by a combination of vapor transport, water migration through the matrix, and fracture flow (Montazer and Wilson, 1984). The relative importance of each flow mechanism is a function of the bulk saturation, the volume of water transported through the rock, the temperature gradients in the rock, the fracture characteristics, and the permeability of the matrix. It has been established that the fracture density within this lower nonlithophysal zone varies between 8 and 40 fractures per cubic meter (Scott et al., 1983), with a mean matrix porosity of 14 percent (56 samples, with a standard deviation of 5.5) and a mean saturation of 65 percent (44 samples, with a standard deviation of 19) (Montazer and Wilson, 1984).

The net flux of water through the repository is probably 1.0 to 2.0 mm per yr upward, although a downward flux of 1.0×10^{-7} to 0.5 mm/yr may occur as a result of matrix flow (Montazer and Wilson, 1984; Montazer et al., 1986). The current matric potential of the Topopah Spring tuff is approximately -112 kPa which results in negligible fracture flow (Wang and Narasimhan, 1985). For all fractures other than minor fractures, fracture saturation will not exceed 0.01 unless the matrix is fully saturated and the matric potential is in the range of -1 to -0.01 kPa which is two to four orders of magnitude greater than at present (Wang and Narasimhan, 1985). Precise statements regarding when fracture flow would become dominant cannot be made until more information is available about fracture roughness and other fracture characteristics (Section 8.3.4.2).

The emplacement of waste packages in the rock would produce a large thermal perturbation. This, in turn, would cause water to vaporize and migrate in the near field and would result in an altered hydrologic regime. There is very little information, experimental or theoretical, on thermally driven flow in partially saturated rocks. Because water is the main corrosive agent for the metal container and the main agent for the transport of radionuclides, experimental and numerical modeling studies have been initiated to characterize fluid flow and the geochemistry of water-rock interactions in the Topopah Spring tuff.

The fluid flow experiments (Lin and Daily, 1984; Daily et al., 1986) were designed to investigate the mechanism of dehydration and rehydration under a variety of thermal conditions in initially saturated intact and fractured rock. These experiments were primarily conducted to investigate and develop experimental techniques. Experiments in unsaturated tuff are under development (Section 8.3.4.2). The results described in the following paragraphs are from fully saturated samples.

Studies were conducted in two stages. The initial stage involved reconnaissance experiments in which the use of electrical resistivity measurements and P-wave velocity measurements were evaluated as means of mapping fluid distribution in the rock during dehydration-rehydration cycles at various temperatures (Lin and Daily, 1984). These experiments allowed initial characterization of the hydrologic properties of the tuff during dehydration and rehydration. The second stage of the study involved development and use of computed impedance tomography (CIT) to obtain detailed images of fluid distribution in the tuff during dehydration-rehydration cycles (Daily et al., 1986). In both stages, steam and water permeability was determined for fractured and unfractured (intact) samples of tuff.

In the initial stage of this study, three samples of tuff were used in the experiments. Two samples of Topopah Spring tuff from Fran Ridge and one sample from drillhole USW G-1 were machined to form right cylinders 9 cm long and 2 cm diameter. One of the Fran Ridge samples was intact; the remaining samples each had one longitudinal fracture.

To simulate as closely as possible the chemistry of the in situ pore fluid, well J-13 water was used as the aqueous phase in these experiments. The protocol for the initial experiments and the results of permeability measurements are summarized in Lin and Daily (1984).

The resistivity images obtained during dehydration and rehydration of the intact sample identified no preferred flow paths. In addition, the permeability of this unfractured rock was independent of temperature and thermal history. These results are consistent with those reported by Morrow et al. (1981, 1985) and suggest that the hydrologic properties of unfractured rock in the waste package environment will not be modified by the anticipated thermal perturbation.

The behavior of the fractured sample was in striking contrast to that of the intact sample. The initially high permeability decreased by a factor of about 20 after initial dehydration and by more than three orders of magnitude to values indistinguishable from that of the intact sample during successive cycles of dehydration and rehydration (Table 7-9). In addition, the initial dehydration occurred eight times faster than in the intact sample. The rate

Table 7-9. Experiment protocol and measured permeability in permeability experiments on Topopah Springs tuff[a,b]

Step No.	Experimental procedure	Measured permeability (in microdarcies) Intact sample	Fractured sample
1	Determine permeability at 21°C	0.34	850
2	Dehydrate[c] at 140°C	--[d]	--
3	Rehydrate with liquid water at 140°C with differential pressure of 2.5 MPa	--	--
4	Determine liquid water permeability at 140°C	0.31	34[e]
5	Dehydrate at 140°C	--	--
6	Rehydrate with steam at 140°C with differential fluid pressure of 0.2 MPa	--	--
7	Determine steam permeability at 140°C	1.99	3.9
8	Dehydrate at 140°C	--	--
9	Rehydrate with liquid water at 98°C with differential fluid pressure of 2.5 MPa	--	--
10	Determine liquid water permeability at 98°C	0.35	0.24

[a]Source: Lin and Daily (1984).
[b]The confining pressure was 5 MPa. The rehydration and permeability measurements were conducted by applying the indicated pressure to one end of the sample, resulting in the indicated differential pressure.
[c]Dehydration was accomplished by drying the sample in the pressure vessel, with both ends of the sample exposed to the atmosphere.
[d]Not applicable.
[e]The value of 34 microdarcies was measured after cooling the sample to 98°C.

at which drying occurred suggests that dehydration was controlled by fluid flow along the fracture.

Healing of the fracture by deposition of silica during fluid flow appears to be responsible for the change in permeability. Before the experiment, the fracture was open, but upon completion of the experiment, the two fragments of rock were bonded together. Scanning electron microscope images of the healed fracture surface showed multiple layers of silica deposited on the fracture walls (Lin and Daily, 1984).

To isolate the main factors contributing to fracture healing, the fractured sample from drillhole USW G-1 was subjected to a sequence of thermal cycles under conditions of constant saturation. Preparation of the sample was the same as for the earlier samples. Surface conditions of the fracture were closely similar to those of the fracture in the outcrop sample. Initial room temperature permeability was 600 to 700 microdarcies, decreasing to 350 microdarcies after a day under flowing conditions. Upon heating the sample to 96°C under saturated conditions, the permeability decreased by one order of magnitude. Changes in temperature up to 140°C and back to room temperature did not produce any further large changes in permeability. Examination of this sample after the experiment showed that fracture healing had occurred, but to a lesser extent than in the first sample.

Resistivity maps document that the rate of change of resistivity is much greater for the fractured samples than for the intact sample. This is consistent with the interpretation that dehydration occurs much more rapidly in the fractured sample. The distribution of resistivity along the fracture surface was not uniform, suggesting that fluid migration along the fracture surface does not take place at a uniform rate. This behavior contrasts with that deduced from resistivity maps of the intact sample that document uniform resistivity changes and, consequently, uniform fluid flow (Lin and Daily, 1984).

Attempts to use P-wave velocity measurements to characterize fluid flow did not provide sufficient resolution to be useful.

The initial results suggested that fluid flow in fractured rock at elevated temperatures would lead to a decrease in permeability of one or more orders of magnitude. Dehydration rates were significantly faster for fractured material compared with dehydration rates obtained for intact samples.

The second experimental stage was designed to more thoroughly characterize the fracture flow process by using computed impedance tomography (CIT) to obtain higher resolution resistivity images (Daily et al., 1986). In these experiments, 14 electrodes were attached to the circumference of a fractured sample, approximately perpendicular to the plane of the fracture. Four electrode pairs were also attached parallel to the long axis of the core samples, as in the previous experiments. The resulting resistivity maps provide a cross-sectional image of fluid distribution in the core, as well as low resolution images parallel to the long axis of the core. Permeability measurements, obtained simultaneously with the CIT images, allow correlation of permeability changes with changes in fluid distribution in the core. The experimental protocol in this second set of experiments was similar to that used during dehydration and rehydration cycles for the first fractured core sample (Daily et al., 1986).

The results of these experiments (Daily et al., 1986) are consistent with the previous results. Rapid decrease in permeability occurred during dehydration stages (Figure 7-6), resulting in permeability changes of more than three orders of magnitude. The final permeability obtained at the end of the experiment was indistinguishable from that obtained previously on intact samples.

The CIT images suggest that the aperture of the fracture decreases during the course of the experiment. This result suggests that, as the aperture of the fracture is diminished, ventilation is restricted, resulting in a decreased rate of dehydration. These observations are consistent with the interpretation of the previous studies that healing along the fracture during dehydration and rehydration cycles is responsible for the drop in permeability observed in the experiments using fractured samples.

These initial results suggest that the hydrologic properties of intact rock may not be modified by dehydration and rehydration cycles resulting from the thermal load imposed on the waste package environment by the waste containers. Uniform fluid flow under postemplacement conditions can be expected. The hydrologic properties of fractured samples in a thermally perturbed environment may be sensitive to solution/deposition processes and to thermal gradients. Because the mechanism responsible for this behavior is unknown, and the behavior in a two-phase system has not been characterized, further work is planned to address these questions under isothermal and polythermal conditions. Completion of the planned work will allow evaluation of the implications of the fracture healing process and will contribute to resolution of Issue 1.10 (Section 8.3.4.2). Field studies in the exploratory shaft establishing the characteristics of flow in natural systems will be used to examine the applicability of these experimental studies (Section 8.3.4.2).

7.4.1.6 Numerical modeling of hydrothermal flow and transport

Numerical modeling studies of coupled multiphase heat and fluid flow in the vicinity of waste packages are being conducted in conjunction with the laboratory activities. These studies and accompanying transport calculations will aid in the prediction of the waste package hydrologic environment and will contribute to the calculation of radionuclide release source terms. The numerical simulations focus on understanding the fundamental mechanisms governing heat and fluid flow in partially saturated fractured rock. Understanding the roles that fractures and adjoining matrix blocks play as conduits to liquid and vapor phase transport is of particular importance. This interaction will influence the extent of dry out in the surrounding host rock and the rate at which rewetting can occur as the thermal output of the waste decreases. These processes impact assessment of waste package corrosion mechanisms and rates and will influence transport rates near the waste package after loss of containment. Upon completion of the exploratory shaft, field studies will provide data to aid in verifying the results of the laboratory and numerical studies.

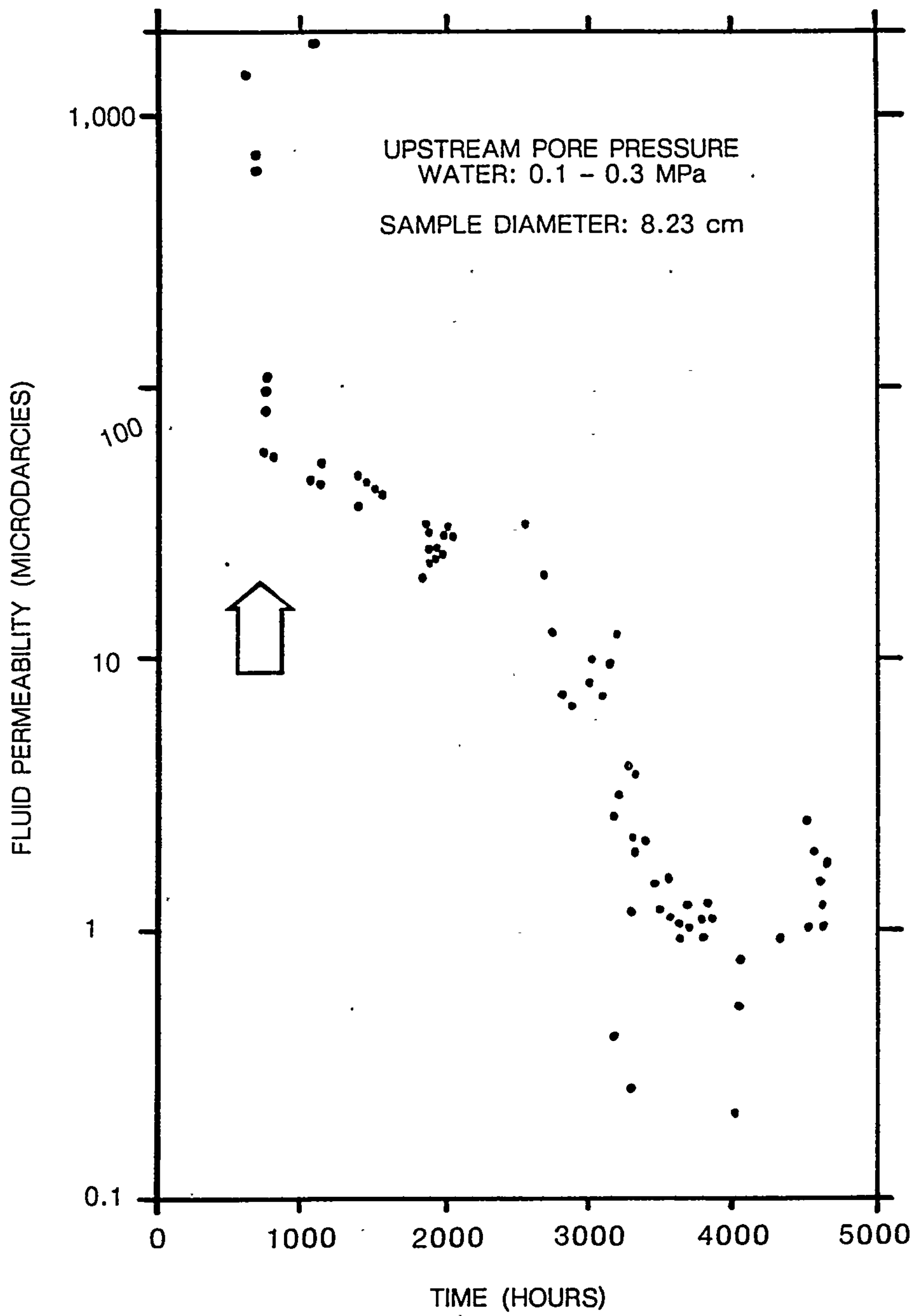


Figure 7-6. Fluid permeability versus time for fractured Topopah Spring tuff. The initial permeability was measured at room temperature. The first rapid drop in permeability, indicated by the arrow, occurred during the initial dehydration period at 90°C. The second rapid drop in permeability, at about 3,000 h, occurred during the second dehydration period. Modified from Daily et al (1986)

The hydrologic properties and ambient conditions in the host rock at the repository horizon result in a complex conceptual model of coupled hydrothermal flow and transport with several unresolved questions. The complexity of the near-field hydrothermal problem arises from two general features. First, the host rock is partially saturated with approximately 65 percent (Montazer and Wilson, 1984) of the bulk pore volume containing water held in the porous matrix primarily by capillary suction. Because of the extremely small average pore size in the matrix, capillary suction effects are very pronounced. Second, the host rock is fractured, containing both matrix and fracture porosity and permeability. The hydrologic properties of the matrix and fracture porosity differ markedly. Experimental evidence (Lin and Daily, 1984; Daily et al., 1986) indicates that the absolute permeability in the matrix and fractures differ by at least three orders of magnitude. Based on empirical evidence and thermodynamic principles, it appears that the characteristic curves (i.e., relative permeability and capillary pressure as functions of bulk saturation) for the respective porosities differ quite markedly (Klavetter and Peters, 1986). Characteristic curves for a steam-water-air system for the full range of saturation conditions in Topopah Spring tuff will be the subject of studies during site characterization (Section 8.3.4.2).

Attempts have been made to synthesize characteristic curves for the fractures using conceptual models that use simple geometric assumptions and thermodynamic principles (Montazer and Wilson, 1984; Klavetter and Peters, 1986). Because the synthesized characteristic curves have not been validated directly by experimental means and because of the inherent geometric variability of naturally occurring fractures, the characteristic curves that are synthesized for the fracture porosity will be subject to variability and uncertainty. The sensitivity of the near-field hydrothermal response to the characteristic curves in the fractures is of interest because it is the interaction of the capillary forces and relative permeability effects on either side of the fracture-matrix interface that will determine whether liquid water is mobile in the fractures under preemplacement conditions and while the system is thermally perturbed. Consequently, any near-field hydrothermal modeling effort will need to consider the range of variability of the characteristic curves in the fractures.

Implicit in the discussion of characteristic curves is the concept of a discrete fracture model. Conceptually, it is most straightforward to model discrete fractures explicitly, using small volume elements to represent fractures adjacent to porous matrix blocks. However, at the waste package scale, limitations in the dimensioning of hydrothermal computer codes on existing mainframe computers preclude the use of discrete fracture models. For near-field hydrothermal modeling, it may be necessary to develop an equivalent continuum representation of the two porosity fracture/matrix system. An equivalent continuum model has been developed, yielding results similar to the discrete fracture model of an idealized two-dimensional near-field problem (Preuss et al., 1984). For models attempting to simulate actual repository conditions, it may be necessary to consider fracture orientation and interconnectivity in three dimensions as well as the variability of these geometric parameters in building equivalent continuum models. Studies during site characterization will attempt to resolve these issues (Sections 8.3.4.2.4 and 8.3.1.2).

The essential features of a conceptual model of the near-field hydrothermal response to the emplacement of the waste packages are described in Preuss et al. (1984). The model considers multiphase heat and fluid flow in partially saturated tuff containing both fracture and matrix porosity. Preuss et al. (1984) only consider fractures that contribute to radial flow away from the waste package. Other considerations that are also potentially important are (1) the influence of fractures oriented orthogonally to the radial direction of flow and (2) the determination of whether the waste package borehole is sealed or open to atmospheric conditions. If liquid water is immobile in the fractures, then the fractures oriented orthogonally to the radial flow direction may become capillary barriers to liquid flow in the radial direction. Emplacement of the waste packages causes temperatures to rise in both the rock matrix and fractures. Most of the vapor generated in the rock matrix flows toward the fractures, which are high conductivity conduits to the gas phase.

Gas phase flow in the radial direction may be affected by whether there is a pathway for gaseous flow from the waste package to the emplacement drift. If the waste package boreholes are ventilating to the access drifts (and are therefore maintained at approximately atmospheric pressure), then some of the vapor generated in the rock matrix will likely seek fractures that intersect the waste package borehole and flow into the borehole and toward the emplacement drift. Near-field hydrothermal calculations are required to determine how much of this vapor (1) will reach the emplacement drifts and condense along the drift walls, (2) will be pulled out through the repository ventilation system, and (3) will condense along the cooler portions of the waste package borehole wall if temperature is low enough.

Assumptions regarding gas and liquid flow geometry after waste emplacement must be evaluated during site characterization. Presently, this geometry is assumed at the waste package scale to be basically radially outward from the package centerline. The reasoning for this assumption appears in Preuss et al. (1984). Because the fractured rock mass at the repository horizon is partially saturated, an interconnected gas phase may be assumed to exist and to be at a pressure equal to the local atmospheric pressure. As the rock heats due to waste decay and the pore water begins to boil, the phase change and the relatively lower gas permeability of the matrix will induce a pressure gradient between the matrix and the fracture system where pressure gradients will move the vapor away from the waste package. Since the fractures have a very large gas-phase permeability, the total gas phase pressure in the fracture system will likely remain near atmospheric. Therefore, gas should flow from the matrix to the fracture system where pressure gradients will move the vapor away from the waste package. Even though many of the fractures within the boiling zone will not communicate with the borehole, the net gas flux in the rock mass may still be directed radially outward from the package.

The outward radial flow of vapor in the fractures will result in condensation along the cooler fracture walls. An unresolved question, which is being addressed in Section 8.3.5.10, is whether the liquid condensation along the fracture walls will attain saturations sufficient to result in liquid phase mobility within the fracture or the liquid condensation will be pulled into the rock matrix by the capillary suction gradients. The answer to this question will depend on the rate of condensation, the capillary suction gradients, and liquid phase permeabilities within the respective matrix and fracture porosities.

The size of the dried-out region around the waste package is inversely dependent on the efficiency of the primary mechanism of heat transfer in the fractured rock mass. Where the liquid phase is mobile in the fractures, a highly efficient heat pipe system may be established within the fractures, carrying away most of the heat generated by the waste package (Preuss et al., 1984). For an immobile liquid phase in the fractures, a far less efficient vapor-liquid counterflow system may develop in which the radial inflow of liquid in the matrix is never able to balance the radial outflow of vapor in the fractures. Existence of fractures approximately perpendicular to the thermal and hydraulic gradients could further interfere with the vapor-liquid flow system. Within the growing dried-out region around the waste package, the only available mechanism of heat transfer is heat conduction, which is far less efficient than the balanced vapor-liquid counterflow heat pipe system established when the liquid phase is mobile in the fractures. Numerical modeling provides a means of examining these effects and will be applied in conjunction with laboratory and exploratory shaft field investigations during site characterization.

7.4.1.7 Rock-water interactions

The elemental and molecular constituents and the ionic species dissolved in water may have a significant effect on the corrosive action of the water and on the extent to which it may dissolve and transport radionuclides. Factors that influence the nature and concentration of dissolved constituents include the mineralogy of the rock in contact with the water, the temperature at which the contact takes place, the duration of the contact, and the ratio of rock surface area to water volume. The last two parameters are most important for interpretation of kinetically controlled dissolution and precipitation reactions. Performance assessment requires knowledge of these parameters, to establish corrosion behavior of the waste package under a variety of scenarios and to evaluate radionuclide release and transport (Sections 8.3.5.9 and 8.3.5.10).

As discussed in Chapter 3 and summarized in Section 7.1, uncertainties exist with regard to flux and flow mechanisms in the unsaturated zone. Low matrix flow rates will result in long contact times between rock and water. Flow through fractures may be slow or rapid, depending on the fracture aperture and the water flux. To cover the range of possible conditions, a number of tests have been conducted to determine the changes in water chemistry for different contact times, rock-to-water ratios, temperatures, and rock samples representative of the Topopah Spring tuff.

Rock-water interaction studies have been conducted using the following three experimental methods:

1. Solid core wafers or crushed tuff reacted in gold-bag rocking autoclaves (Dickson reaction vessels).

2. Solid core wafers or crushed tuff reacted in Teflon capsules contained in steel casings (Parr acid digestion reaction vessels).

3. Solid core wafers or crushed tuff reacted in stainless steel vessels.

Each of these methods has advantages and limitations; by combining data from all three methods, the artifacts of experimental technique can be removed from the data and a better estimate of the chemistry of the water under in situ conditions can be obtained.

The gold-bag autoclaves allow sampling of fluids without quenching the solution. The gold bag is inert and impervious, providing a nonreactive container that is gas tight. This feature is important because carbon dioxide (CO_2) is a reactant of interest in the system. The disadvantages of the gold-bag autoclaves are that the experiments are expensive to run, thereby limiting the volume of data that can reasonably be obtained; the rocking action of the system is not representative of expected repository conditions; and the sampling during reaction causes a change in fluid-to-solid ratio during the run.

Reactions in Teflon™-lined reaction vessels are inexpensive to run, and a large amount of data can be accumulated. The disadvantages of this method are that the Teflon™ capsule allows diffusion of CO_2 from the reacting mixture, thereby affecting the pH and alkalinity data (Knauss and Bieriger, 1984); the assembly must be cooled to near room temperature before opening and separating the fluid and solid phases; and the Teflon™ cannot be used in a gamma radiation field because it breaks down radiolytically and contaminates the fluid with hydrofluoric acid (HF). As will be discussed, the gas absorption effect can clearly be seen in the data, but the cooling to room temperature does not produce any detectable effects.

For tests conducted in a gamma radiation field, stainless steel vessels are used. Corrosion of the vessels is very slight under the reaction conditions, and the effects on solution chemistry are limited to minor amounts of iron, nickel, and chromium being added to the solutions.

The Topopah Spring tuff occurs in outcrop at Fran Ridge. Because many of the waste package materials interaction tests require rather large amounts of rock, the outcrop material has been characterized to determine whether it can be used in experiments and tests that require more rock than can be obtained readily from drill core (Oversby and Knauss, 1983; Oversby, 1984a). The petrologic characterization of the outcrop showed that there were no significant differences between the outcrop and samples obtained from the USW G-1 drill core (Knauss, 1984). There is a difference between the outcrop and drill core material with respect to rock-water interaction chemistry. The outcrop material contains a readily soluble component that appears to be caused by evaporation of surface runoff water from the pores, which leaves behind significant amounts of previously dissolved salts. The principal constituents of these salts are calcium, sodium, potassium, chlorine, nitrate (NO_3), sulfate (SO_4), and boron (Oversby, 1984a). The salt deposits are found only in near-surface samples and are not present in samples collected some distance from the surface (Oversby, 1985).

The choice of rock sample form is dictated by the type of data needed from the tests. If characterization of secondary phases and alteration products is required, it is necessary to use polished wafers of solid rock that are suitable for examination by surface analysis techniques after completion of the tests. If the desired information concerns the effects of rock-to-water ratio on final chemistry, then crushed rock is used because the weight added to the system can be more easily controlled.

Studies that involved characterization of secondary phases used rock wafers 2.54 cm in diameter by 0.25 cm thick that were prepared from drill core. One side of each wafer was hand-polished so that each could be examined by electron microscopy after completion of the experiment. Aliquots of the fluids from these experiments were removed periodically without interrupting the course of the reaction. These aqueous samples were analyzed for pH, dissolved carbonate species, cations, and anions. By the end of the reaction period, the fluid volume in the autoclave cell was about one-half the volume at the start of the reaction. At the end of each test, the rock wafer was removed from the autoclave, rinsed with distilled water, dried, and carbon coated for examination of phases by scanning electron microscopy (SEM) and electron microprobe analysis (EMP).

All cation analyses were conducted on acidified, filtered (0.1-micron Nuclepore™ filters) samples. Anion analyses were conducted on filtered samples. Unmodified samples were used for all pH measurements.

The results of tests conducted at 150 and 250°C in gold-bag autoclaves using rock wafers prepared from USW G-1 drill core have been published by Knauss et al. (1984). Results from gold-bag tests at 90, 150, and 250°C using crushed drillhole USW G-1 material are reported in Knauss et al. (1985). Long term studies (304 d) at 90 and 150°C are reported in Knauss et al (1987a). Parr reaction vessel data have been published for reaction temperatures of 90, 120, and 150°C (Oversby, 1984a,b). These tests used rock obtained from an outcrop of Topopah Spring tuff located at Fran Ridge. Results from reaction at 150°C in Parr reaction vessels of crushed rock from drillhole USW G-1, USW GU-3, USW G-4, and UE-25h#1 cores are contained in the report by Oversby (1985). Description of the outcrop locality and characterization data for the rock samples may be found in the report by Knauss (1984). These studies are summarized in Glassley (1986). The salient points from the summary are discussed in the remainder of this section.

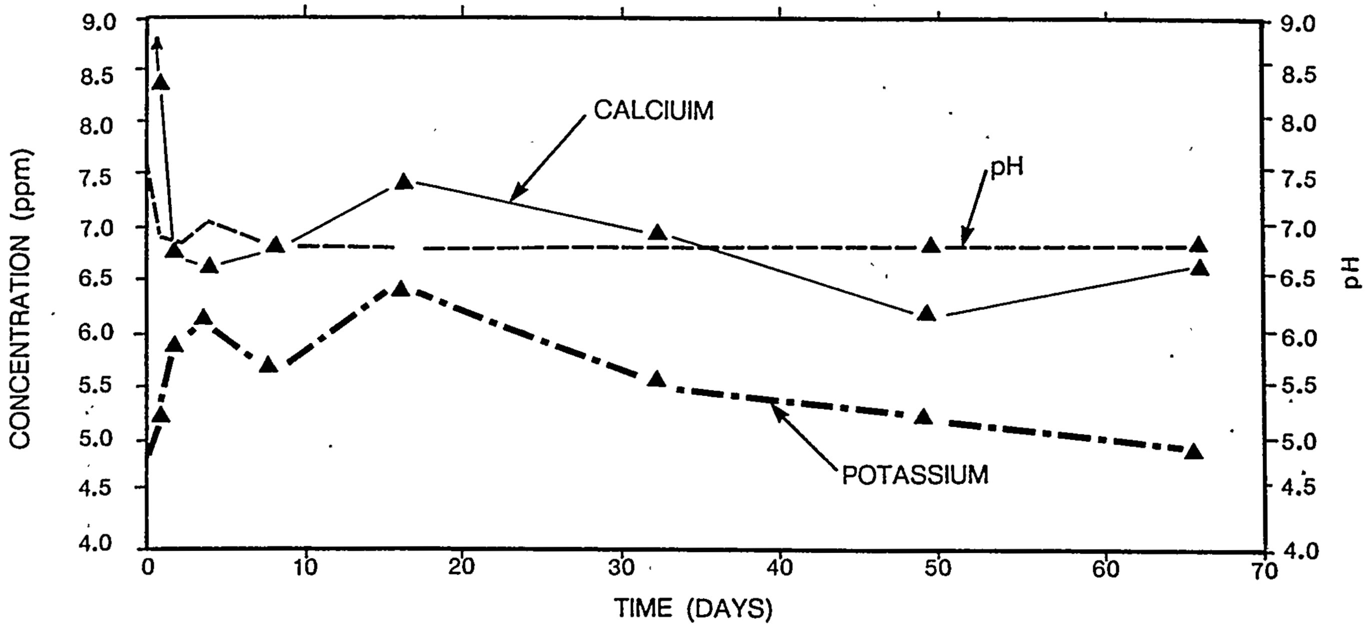
The unconfined boiling point of pure water at the repository elevation would be approximately 96°C. Small pores in the rock retain water at somewhat higher temperatures due to capillary forces. Repository-scale modeling has indicated that water can be retained in pores at temperatures up to 140°C if a restriction on venting is imposed (Travis et al., 1984). Rock-water tests were conducted at 90°C to represent conditions of the near-maximum temperature expected for the main mass of water that would flow through rock and potentially contact waste packages. Tests at 150°C represent the conditions for small volumes of water retained in pores under unusual, but possible, conditions. Some tests were conducted at 250°C to investigate the nature of secondary phases formed and to assist in providing kinetics data for reaction path modeling. Because the secondary mineral assemblage at 250°C may be different from that formed at lower temperatures, these data should be used only to guide the direction of lower temperature studies until a direct tie to lower temperature phase assemblages can be established.

The evolution of fluid composition in the rocking autoclaves for long-term and short-term experiments at 90, 150, and 250°C showed similar behavior. Figures 7-7 and 7-8 present typical examples of fluid evolution from a 150°C experiment. Silica and sodium concentrations exhibit an increase early in the experiment. During the experiment, the silica concentration approaches that expected for cristobalite saturation at the experimental conditions. Calcium, magnesium, and carbonate (not shown) exhibit an early, rapid decrease in abundance before attaining a steady-state concentration. This behavior results from the retrograde solubility of carbonate and reflects the precipitation of calcium- and magnesium-bearing carbonate early in the experiment. Potassium and aluminum tend to develop a peak concentration early in the experiments and then develop a gradual, moderate decrease in abundance before attaining a near steady-state concentration. This behavior for potassium and aluminum is believed to result from kinetically inhibited precipitation of a potassium-rich clay mineral during the course of the experiment. The concentration of anions (fluoride, chloride, NO_3, and SO_4) exhibits no significant variation for the duration of the experiment, reflecting the absence of significant sources for these anions in the rocks and the absence of precipitating phases that incorporate these constituents. The pH of the solution decreases from approximately 7.5 to 6.8 in response to the removal of carbonate from solution during carbonate precipitation.

Solid phases occurring as reaction products in the 150°C experiments were rare. A potassium-rich phase, believed to be a clay, was identified with the scanning electron microscope (SEM). This was the most abundant secondary phase observed. Also present were very small quantities of a zeolite rich in magnesium, calcium, and iron, of gibbsite, of calcite, and of a silica-rich phase believed to be cristobalite. Plagioclase feldspars, sanidine, biotite, and the fine-grained matrix in the rock wafer exhibited corroded surfaces resulting from hydrothermal reaction and dissolution. No change in the composition of the original solid phases in the rock was observed. In the 90°C experiments, only a potassium-rich clay similar to that observed in the 150°C experiments was identified. The effects of hydrothermal interaction on the primary phases in the 90°C wafer were very minor, with no observable change in the solid phase compositions. At 250°C, the zeolites dachiardite and mordenite formed in addition to previously recognized secondary phases (Knauss et al., 1987a,b).

These experiments lead to the following conclusions: (1) regardless of the temperature, the fluid in a water-saturated environment will evolve toward cristobalite and carbonate saturation; (2) secondary phases resulting from hydrothermal interaction will include sorptive clays and zeolites although the assemblage that forms is a function of temperature; and (3) the composition of the fluid phase remains benign at all temperatures, i.e., the pH is near neutral and there are low abundances of the anions chloride, fluoride, and NO_3.

Experiments to examine reaction kinetics were conducted on crushed samples of Topopah Spring tuff at the conditions just described. The fluid composition evolution in these studies was very similar to that obtained for



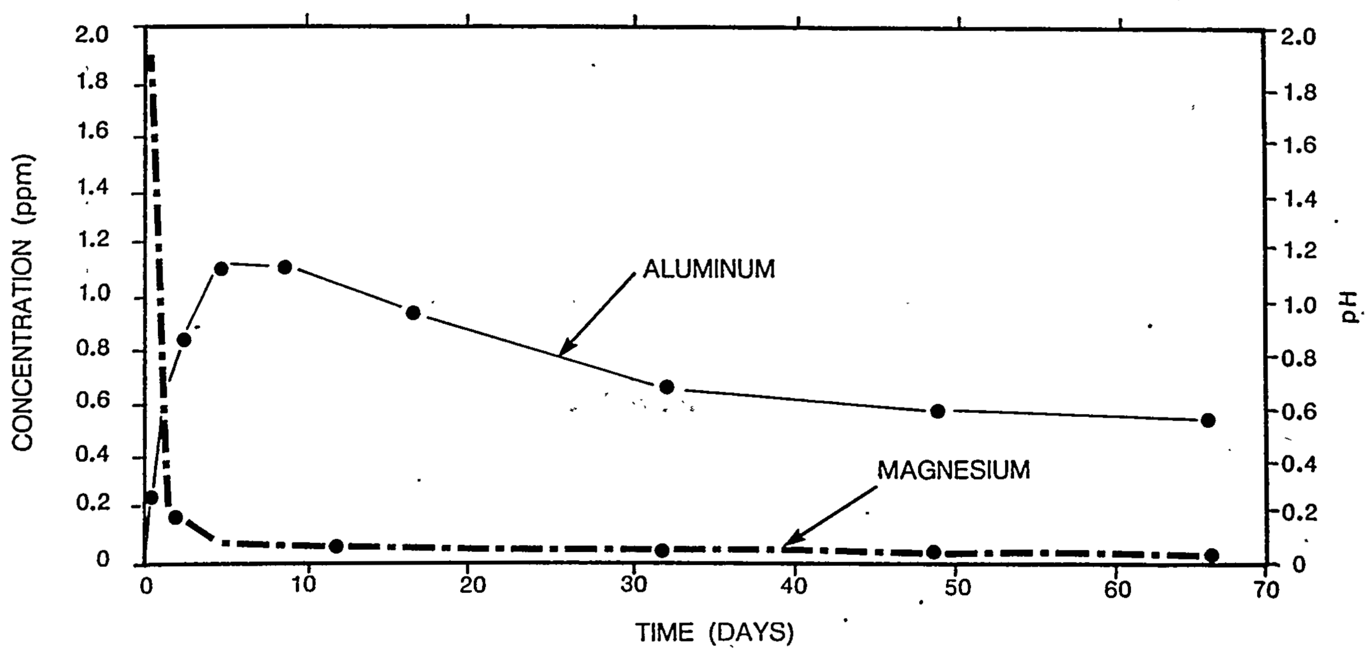


Figure 7-7. Aluminum, potassium, calcium, magnesium, and pH analyses of water from well J-13 reacted with USW G-1 core wafers at 150°C as a function of time. Modified from Knauss et al. (1987a).

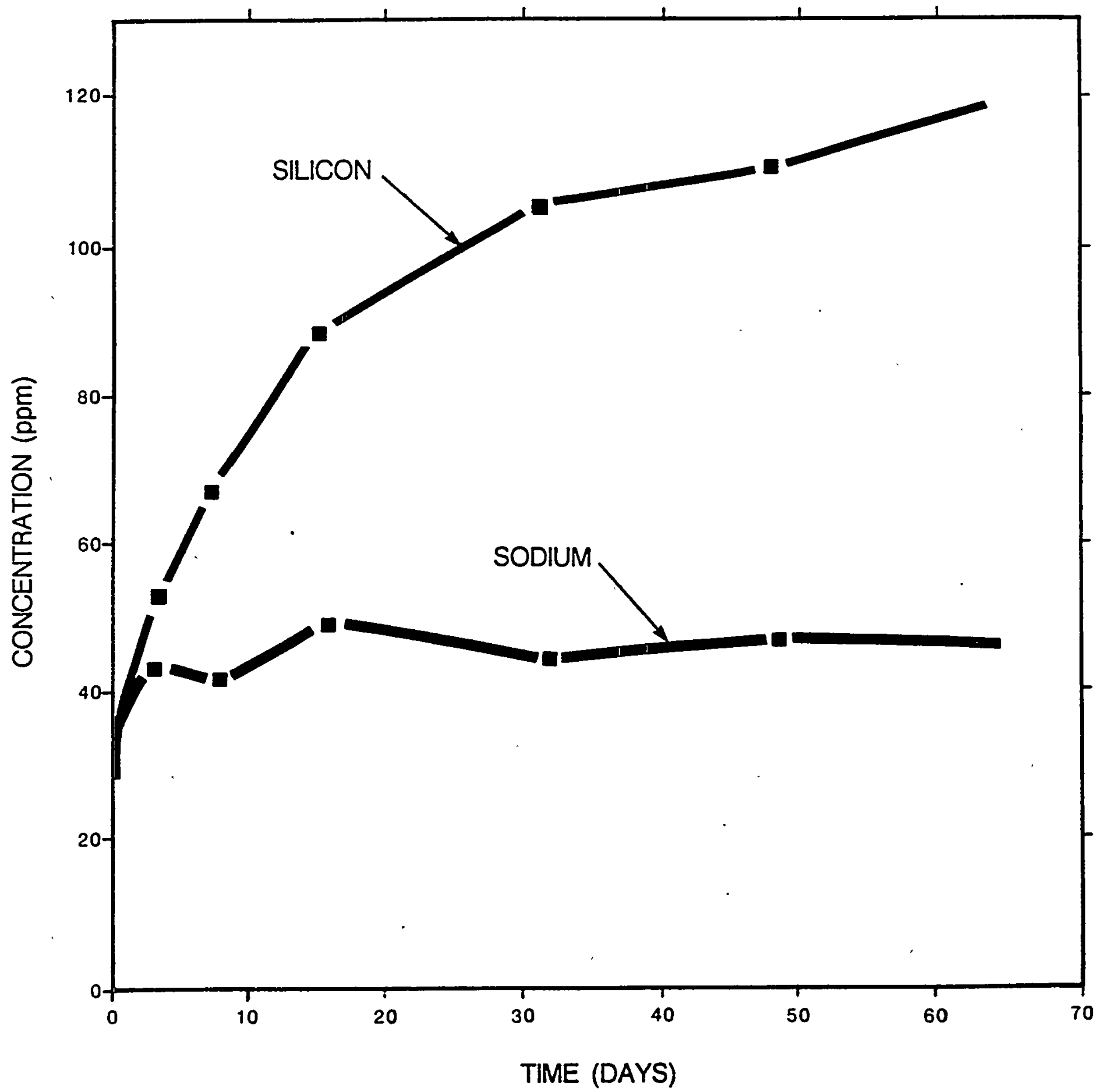


Figure 7-8. Silicon and sodium concentrations in water from well J-13 reacted with USW G-1 core wafers at 150°C as a function of time. Modified from Knauss et al. (1987a).

the solid wafer, although the rate of change for the solution was much greater during the early portion of the experiment. This result was expected since the hydrothermal reactions occurring in the experiments are expected to be kinetically controlled dissolution reactions that are dependent on the surface areas of solid in relation to fluid volume (SA:V) (Aagaard and Helgeson, 1982; Helgeson and Murphy, 1983).

Rock-water interaction tests conducted in Teflon™ capsules contained in steel casings (Parr acid digestion reaction vessels) allow comparison of the compositional evolution of fluid equilibrated with surface samples with that equilibrated with drill core obtained from much greater depth. These experiments used crushed tuff and core wafer material prepared from outcrop samples and reacted at 150°C (Knauss and Beiriger, 1984), and crushed material at 90°C (Oversby, 1984b). Tests have also been conducted using crushed outcrop material at 120°C (Oversby, 1984a). Results from the latter tests were intermediate between the 90 and 150°C results, as expected, and will not be discussed further here.

The trends of solution chemistry with time for the 150°C experiment were similar for both the outcrop material and the drillhole USW G-1 crushed rock except for pH, alkalinity, and final silicon concentration. The differences in alkalinity and pH may be due to diffusion of CO_2 out of the Teflon™ capsule, resulting in lower amounts of dissolved carbonate species and higher pH (Knauss et al., 1983). The lower silicon concentrations found by Oversby (1984a) may be the result of lower effective surface area in the crushed rock (less than 60 mesh) compared with the 100 to 200 mesh material used by Knauss et al. (1985). Results at 90°C also showed similar differences in pH, alkalinity, and silicon. In addition, the calcium from the outcrop samples was somewhat lower due to a higher solution pH, with final concentrations of about 8 ppm versus about 12 ppm for the drillhole USW G-1 samples.

Oversby (1985) has examined the water chemistry resulting from reaction of Topopah Spring tuff with well J-13 water as a function of lateral and vertical extent of the unit by using material taken from three vertical drillholes (USW G-1, USW GU-3, and USW G-4) and from the horizontal air-drilled hole at Fran Ridge. The solution chemistry after 70 d of reaction was very similar for all samples except that the silicon concentration was slightly lower for one of the Fran Ridge samples. The pH and alkalinity data show the expected differences from the gold-bag autoclave tests. Final silicon values ranged from 99 to 110 ppm (excluding the low-silicon Fran Ridge sample), somewhat lower than the autoclave results. Again, the differences may be due to the different mesh size used, which affects the surface area to volume ratio of the test material. Data for aluminum, magnesium, and sodium are very similar, but potassium and calcium concentrations were lower in the Parr reaction vessel tests compared with the values obtained in the gold-bag autoclave tests. The lower calcium concentration may be due to the higher pH, which increases carbonate relative to bicarbonate and may cause increased precipitation of calcite. The reason for the lower potassium (1 to 3 ppm versus 6 ppm) is not understood.

The drill core samples were tested for the presence of readily soluble salts by treating with water at room temperature and also by heating with well J-13 water overnight. This is the pretreatment procedure used on outcrop samples to remove evaporite salts. The resulting solutions showed no evidence of readily soluble material. This result is particularly significant for the samples from the air-drilled hole at Fran Ridge since drilling fluid that could have removed soluble salts was not used in the portion of the hole from which the samples were obtained. This result strongly suggests that the presence of soluble salts is a surface evaporation phenomenon and that such materials are unlikely to be present at the depth of the repository horizon (Oversby, 1985).

It has yet to be established whether rock-water interactions in vitric- and vitrophyre-rich units are similar to those observed for the devitrified welded tuff. It is also necessary to complete detailed rock-water studies on rock material obtained from the exploratory shaft at the repository elevation to confirm the implications of previous studies and to satisfy Issue 1.10 (Section 8.3.4.2). This work will be done using water compositions appropriate for the vadose water system of the repository horizon, which will require completion of extraction techniques and analysis of vadose water from representative near-field material in the exploratory shaft (Sections 8.3.1.3, 8.3.1.16, and 8.3.4.2).

7.4.1.8 Modeling rock-water interaction

Results of the 150°C rock-water interaction tests were modeled using the EQ3/6 modeling code (Knauss et al., 1984). The EQ3/6 code (Wolery, 1979, 1983) uses finite difference derivatives with a Taylor series expansion to predict the equilibrium distribution of components between aqueous species as a reaction progress variable is incremented. Chemical affinities for all minerals that might occur in the system are then computed using a Newton-Raphson algorithm. If saturation of any of these minerals is attained, a mass action relationship is incorporated into the array of equations. The activities and molalities of all aqueous species and the number of moles per unit mass of water of each mineral produced or destroyed are then computed for each step in the reaction progress.

The composition of well J-13 water was used as input to the EQ3NR portion of the EQ3/6 software package to obtain a model aqueous solution for the reaction simulation. For the 150°C tests, the dissolved aluminum concentration was constrained to satisfy the mineral solubility equilibria for kaolinite to reduce the number of supersaturated phases listed by the model. The rock was represented in the EQ6 model by six minerals: sanidine, plagioclase, cristobalite, quartz, biotite, and montmorillonite. Specific surface area for each phase was calculated from the measured Brunauer-Emmett-Teller (BET) surface area of the rock wafer and the estimated abundance of each phase.

The precipitation of all silica phases less soluble than cristobalite was suppressed to allow the code to reproduce the experimentally observed silicon concentrations. Clays initially present in the hydrothermal experiments were modeled by magnesium-beidellite, a smectite clay that is a close compositional and structural analog to the montmorillonite identified in the experiments.

The EQ3/6 calculations realistically predicted that none of the initial reactants were exhausted and that a relatively minor amount of the rock dissolved during the reaction interval. The final concentrations for all major cations in solution predicted by the model were close to the observed values. As silica in the form of quartz and cristobalite dissolves, its concentration increases to a steady-state value, constrained by the input parameters to the steady-state solubility of cristobalite. The observed sharp rise in aluminum can be attributed to rapid dissolution of a small amount of magnesium-beidellite included in the model rock assemblage. The initial drop in calcium is due to formation of minor amounts of calcite, but the continued decrease in calcium is due to precipitation of a calcium-bearing smectite clay. The formation of the calcium-bearing smectite clay in the model also lowers the aluminum concentration in the model reaction path. The absence of thermodynamic data for potassium-bearing clays has prevented their incorporation in the model. The initial rise in potassium in the actual reaction solutions cannot be accounted for by the model.

The kinetic rate laws included in the current version of EQ6 provide only for dissolution reactions. Precipitation appears as an instantaneous process once saturation of a particular phase has been reached. Code development work is under way to include kinetically limited precipitation reactions in the EQ6 model.

The reaction path code used to model the experimental results uses equilibrium thermodynamics to establish the phase relationships. In the experimental system, it is possible that metastable phases precipitate during reaction progress. Such behavior would be a consequence of the tendency of natural and experimental systems initially to produce phases that have free energies between those of the reactants and those of the stable products of a discontinuous reaction (Ostwald, 1887). To account for the behavior of potassium, it will probably be necessary to include a metastable potassium-bearing phase (such as illite or smectite) to model the observations. When appropriate thermodynamic data are available for these phases, they will be incorporated in the code (Section 8.3.5.10).

Delany (1985) has continued the reaction path modeling of the 150°C autoclave tests, extending the work to the results of investigations with crushed tuff. Estimates of dissolution kinetic parameters were improved from the earlier work (Knauss et al., 1985) by using a combination of published rate constant data and additional kinetic constraints. Reported dissolution rate constants that are consistent with transition-state theory and that extend to temperatures as high as 150°C are available for potassium-feldspar (Helgeson et al., 1984) and for the silica phases (Rimstidt and Barnes, 1980). Dissolution rate constants are available to 70°C for pure albite (Knauss and Wolery, 1986) and to 25°C for phlogopite (Lin and Clemency, 1981). The dependence of the rate constants on temperature was estimated from transition state theory (Delany, 1985). The results of these most recent modeling efforts are compared in Figures 7-9 and 7-10 with the observed solution compositions.

A preliminary attempt to model the long-term behavior of a rock-water system was also completed (Delany, 1985).

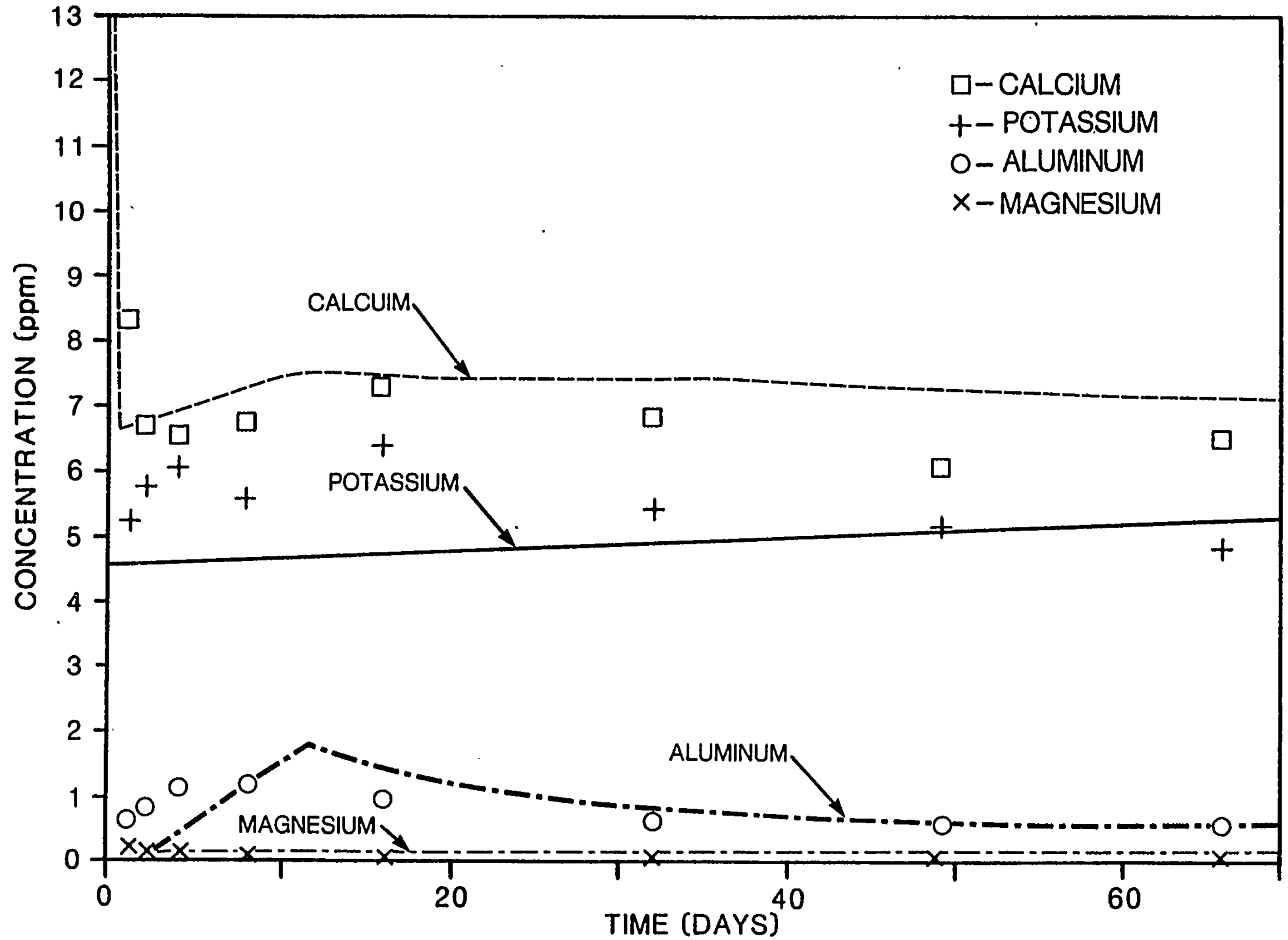


Figure 7-9. Comparison of fluid composition from EQ3/6 calculations and actual measured values for calcium, potassium, aluminum, and magnesium in water from well J-13 reacted with Topopah Spring tuff at 150°C. The lines indicate the results of the EQ3/6 calculations; the symbols represent the experimental results. Modified from Delany (1985).

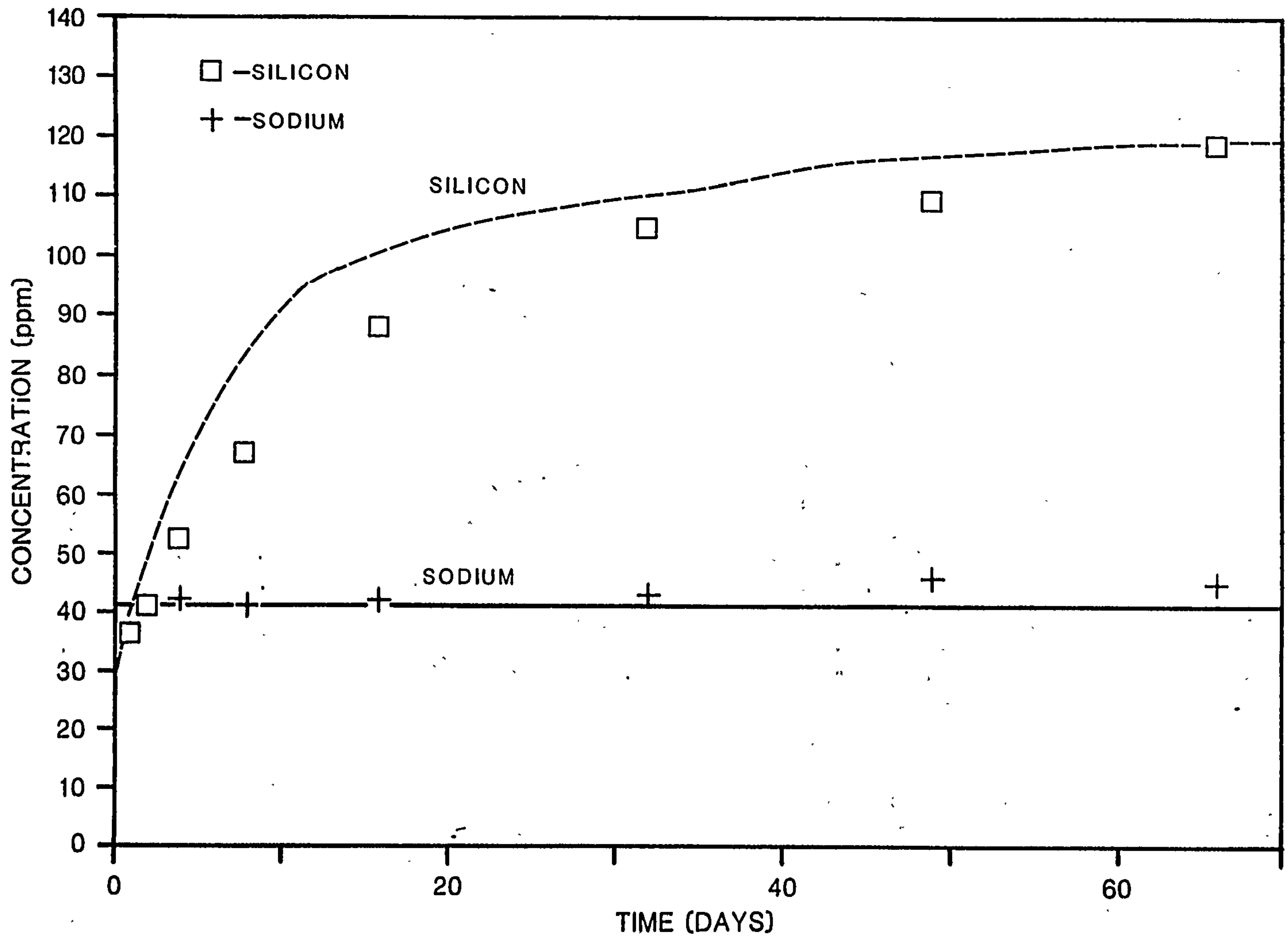


Figure 7-10. Comparison of fluid composition from EQ3/6 calculations and actual measured values for silicon and sodium in water from well J-13 reacted with Topopah Spring tuff at 150°C. The lines indicate the results of the EQ3/6 calculations; the symbols represent the experimental results. Modified from Delany (1985).

The model used a system open to the atmosphere so that carbon dioxide and oxygen were available as reactants in essentially infinite supply. The model calculation was done at 150°C using the core wafer model parameters and a reaction interval of 100 yr. EQ6 predicted that the solution composition would reach steady state at 150°C after 180 d. If adjustment is made to account for the high rock surface area-to-water volume ratios (SA:V) expected in the repository, as compared with the experiments, the dissolution that occurs during 1 yr in an experiment would be approximately comparable to that occurring during approximately 1,000 yr in the repository (Delany, 1985). The approach to steady-state concentrations should, however, be faster in the repository because of the higher SA:V.

The water chemistry measured in short-term tests and predicted through the use of reaction path modeling differed from the initial well J-13 water composition by having higher concentrations of silicon, aluminum, and sodium and lower concentrations of calcium, magnesium, potassium, and dissolved inorganic carbon species. The pH remained near neutral, and no significant source of anions has been identified in the rock. The secondary mineral assemblage produced by reaction at temperatures up to 250°C consisted of clays, zeolites, and small amounts of other minerals. The dominant secondary minerals have high surface areas, high cation exchange capacity, and positive sorption capabilities. As such, alteration of the rock by hydrothermal reaction should contribute to the ability of the site to retard migration of radionuclides.

7.4.2 METAL BARRIERS

This section discusses the results of testing several different waste-package container materials. It describes the candidate materials, the functions of the metal container, and degradation modes that can occur for the different candidate materials. This information will be used to select one candidate material that best meets the performance objectives for a waste package container, as discussed in Section 7.2. The waste package components, including the metal barrier, are described briefly in the introduction to this chapter.

7.4.2.1 Functions of the metal barrier

The function of the metal barrier will change from the first 50 yr of repository operation through the 1,000 yr containment period and the 1,000 to 10,000 yr controlled release period following repository closure. The metal barrier will function as the vessel for transporting, handling, emplacing, and retrieving (if necessary) the contents of the waste package in the repository during the operational period. During the first 1000 yr following closure, the metal barrier will be relied upon to provide complete containment for a large fraction of the waste packages and to be a substantial impediment to release of radionuclides by aqueous transport from those packages with breached containers. During the subsequent controlled release period, many containers may remain intact. Breached containers are expected to continue to inhibit transport of liquid water into, and radionuclides out of the waste packages. The performance allocated to the metal container is disucssed further in Section 8.3.5.9.

7.4.2.2 Candidate materials for waste package containers

The NNWSI Project is evaluating six candidate materials for the waste package container. These six materials fall into two alloy families: (1) austenitic materials and (2) copper-based materials. The austenitic materials are AISI 304L, AISI 316L, and Alloy 825. Because of the high nickel content of Alloy 825 (about 40 percent) compared with the 300-series stainless steels (about 10 percent), Alloy 825 (e.g., Incoloy 825) is a transition material between iron-base and nickel-base systems. Although many of the same degradation modes that commonly affect the austenitic stainless steels can potentially affect Alloy 825, the overall difference in corrosion resistance places Alloy 825 in a separate grouping.

The candidate copper-based materials are oxygen-free copper (CDA 102), 8 percent aluminum-bronze (CDA 613), and 70-30 copper-nickel (CDA 715). The industry-standard compositions and selected mechanical properties of these candidate materials are given in Tables 7-4, 7-5, 7-6, and 7-7 in Section 7.3.

AISI 304L stainless steel (SS) was selected in 1983 as the reference material for the NNWSI Project waste package designs. A thin-walled design was chosen because of the planned location of the repository above the water table and the projected absence of significant hydrostatic or lithostatic loads. A reference material was established so that the waste package and repository design tasks could proceed in parallel with the metal barriers testing. The reference material established a benchmark to which the performance of other materials can be compared. AISI 304L SS was selected because of its excellent corrosion and oxidation resistance in steam, air, and natural waters similar in composition to well J-13 water. AISI 304L SS has adequate mechanical properties and can be fabricated and welded by a variety of processes. In addition, a substantial database exists for AISI 304L SS in a variety of engineering applications. By definition, a reference material should be a viable candidate, although it may not necessarily become the finally chosen material. The planned material selection process is described in Section 8.3.5.9.

In 1984, the NNWSI Project was requested by the Office of Geological Repositories (OGR) to evaluate copper or a copper-based alloy as a waste package container material. At the end of FY 1986, the NNWSI Project determined that copper-based materials have not been shown infeasible for containers, and further study is warranted. Some virtues of copper-based materials are (1) they have electrochemical nobility, (2) they represent simpler metallurgical systems in terms of the number of components and phases, and (3) they are potentially susceptible, under repository conditions, to degradation modes that are different from those for austenitic stainless steels. The results of studies on copper-based materials are discussed in Section 7.4.2.9.

7.4.2.3 Degradation modes of austenitic materials under repository conditions

Various forms of corrosion and oxidation attack by the environment on the metal container constitute the most likely degradation modes; the process by which the container is fabricated and closed and the strains imposed by these processes and by internal and external stresses on the container can further affect the different degradation modes. Research and development activities for metal barriers are centered around these various degradation modes with the purpose of discerning which of these degradation modes are applicable and in what time period after permanent closure of the repository each particular degradation mode may be operable.

One of the most important factors in determining which degradation modes may be operable is the anticipated postclosure environment at the Yucca Mountain repository. A summary of the anticipated conditions is given in Section 7.4.1. Summaries of important results from tests in which the candidate container materials have been subjected to different environmental conditions are presented in this section.

Some of the testing has been performed under anticipated environmental conditions; however, many tests have been performed under unanticipated conditions. The reasons for conducting tests under unanticipated conditions are (1) the particular set of environmental conditions accelerates the corrosion mode, thus making quantitative assessment of the long-term effect possible in relatively short-term laboratory tests, and (2) the set of environmental conditions, in certain instances, presents a possible worst case scenario of credible, although unlikely, conditions for comparison of the performance of different candidate materials under adverse conditions. An example of a worst-case scenario is testing in saturated vapor or moist air with a high gamma radiation field.

The austenitic materials are generally corrosion resistant because of the formation of a protective, passive film on the surface. The passive film is characteristically 50 to 100 angstroms thick; structurally, the film is an amorphous chromium-rich iron oxide when the austenitic material is in contact with aqueous environments; at higher temperatures, a more stable chromium-nickel spinel structure forms. The passive film is generally stable in mildly to moderately oxidizing environments and over a wide range of pH. The low corrosion and oxidation rates of passive austenitic materials are controlled by mass transport through the passive film; continuous dissolution and reformation of the film occurs. Breakdown of the passive film exposes the thermodynamically active substrate to the environment and is the first stage of rapid general corrosion or of initiation of localized corrosion and stress corrosion cracking. The mechanical properties of the film influence initiation of corrosion under stress and rupture of the film when the chemical environment does not favor rapid repassivation of the alloy. Projections of long-term satisfactory performance of austenitic containers depend on maintenance of the passive film. Modeling of the corrosion behavior under a variety of expected and possible (though unlikely) environmental conditions is therefore based on the chemical and mechanical stability of the passive film.

AISI 304L SS, as well as the other austenitic materials, shows excellent general corrosion resistance in aerated dry steam environments (less than the saturation steam content at the relevant temperatures and pressures) as well as in wet steam and in vadose water with a composition similar to well J-13 water. Radiolysis products, such as hydrogen peroxide and oxides of nitrogen and even dilute nitric acid, are not expected to increase the general corrosion rate by a significant factor provided that the material is not sensitized (Section 7.4.2.5). The degradation mode limiting the use of AISI 304L SS and the other austenitic materials is rarely general corrosion, but rather much more rapid penetration via localized or stress-assisted forms of corrosion. The objective of the metals testing activities is to determine to what extent these forms of corrosion would occur during the containment period, the hybrid containment-controlled release period, and the controlled release period. Corrosion modes that will receive particular attention are summarized in Sections 7.4.2.3.1 through 7.4.2.3.3.

7.4.2.3.1 Corrosion forms favored by a sensitized microstructure

A sensitized microstructure could develop during fabrication and welding of the container or possibly during the very long times (decades to centuries) in the containment period because of the elevated temperature resulting from thermal output by radioactive decay of the waste form. Such a sensitized microstructure might lead to intergranular corrosion or intergranular stress corrosion cracking if a suitable aqueous environment (liquid water intrusion or condensation of wet steam on the container surface) were to come in contact with the sensitized region. These forms of corrosion are favored by oxidizing aqueous conditions.

7.4.2.3.2 Corrosion forms favored by concentration of various chemical species in well J-13 water

Continuous evaporation during downward infiltration of water through a region where the temperature rises with depth may leave a residue of ionic salts in the rock pores. Later dissolution of these salts by new water could increase the ionic content of water now able to penetrate to the repository horizon after the boiling point isotherm has retreated to a location within the container. Another way of concentrating ionic salts is by having flow paths through fractures above the repository that admit episodic surges of water that then contact the hot container surface and evaporate, concentrating the solute species. The chloride-ion concentration is of paramount concern with regard to resistance of stainless steels to pitting, crevice, and transgranular stress corrosion cracking. The other ions present in the well J-13 water may favor or retard these kinds of corrosion attacks.

7.4.2.3.3 Corrosion and embrittlement phenomena favored by transformation products from metastable austenite

This classification includes problems that are partly related to corrosion and partly related to mechanical embrittlement. The austenite in some candidate materials (304L and 316L SS) is metastable and may transform to martensite, ferrite, or sigma phase under the influence of mechanical treatment (cold work) or thermal conditions.

Brittle phases such as sigma reduce the fracture toughness so that residual and operating stresses may initiate cracks if the transformation product is more or less continuous. Other transformation products such as martensite and possibly ferrite may be more susceptible to hydrogen pickup and embrittlement. Martensite formation is favored by high plastic deformation. These effects are most likely to occur in and around welds. Gamma radiation is the only significant radiation that the container will encounter. The rates of atomic displacement in electromagnetic (as opposed to particle) radiation are too low to cause metallurgical reactions such as austenitic transformation or carbide precipitation.

Recent investigations of potential degradation modes have revealed the occurrence of severe localized corrosion and stress corrosion cracking of stainless steel caused by microbiological organisms altering the chemical nature of the environment. Microbiologically induced corrosion has occurred under circumstances where it was quite unexpected, a recent example being cited in the paper by Stoecker and Pope (1986). An evaluation of whether the microorganism can survive in a dormant state during the hot, dry period and high accumulated gamma dose rate, and then revive at a later time is planned even though the chances of this occurring are remote. These studies are addressed in Section 8.3.5.9.

7.4.2.4 General corrosion and oxidation of austenitic materials

Low temperature oxidation will occur from the time that the waste packages are emplaced in the repository (and even before). General aqueous corrosion can occur when liquid water is present in the environment, either from intrusion of ground water to the container surface or from condensation of steam on the container surface.

Coupons of the candidate materials have been exposed to environments that approach those that may be expected in the repository during the containment period, the hybrid containment-controlled release period, and the controlled release period. Test environments have included well J-13 water, wet steam, and dry steam at various temperatures. In addition, some coupons have been exposed to irradiated well J-13 water and steam (tests conducted in a cobalt-60 gamma irradiation facility). The tests have been described by McCright et al. (1983), Juhas et al. (1984), and McCright et al. (1987). The average corrosion penetration rate is calculated from the weight loss experienced during the exposure interval, and the exposed surface is examined for the pattern of corrosion attack (uniform, localized, or edge).

7.4.2.4.1 Oxidation and general corrosion test results

Results obtained from coupons immersed in well J-13 water maintained at different temperatures show that the corrosion penetration rates were low (all substantially less than 1 micrometer per yr), with little variation during the exposure period. After more than 10,000 exposure hours in well J-13 water at different temperatures, general corrosion rates in the range 0.030 to 0.225 micrometer per yr were calculated (Glass et al., 1984; McCright et al., 1987). Data obtained at shorter exposure times have been summarized and discussed by Glass et al. (1984). Little change in the general corrosion penetration rate occurred with exposure time; for example, AISI 304L SS had time-average corrosion rates of 0.094 micrometer per yr after 3,548 h and 0.072 micrometer per yr after 10,360 h in well J-13 water at 100°C. The general corrosion rate was approximately the same for all the alloys tested; alloying differences would be expected to be distinguished only in more aggressive environments. The trend of results indicates a small decrease in the corrosion rate with increasing temperature, which may be due to the formation of a slightly thicker passive film or to the decrease in dissolved oxygen and nitrate ion concentrations at the higher temperatures.

Corrosion tests performed on candidate austenitic stainless steel coupons in wet steam (saturated at the test temperature of 100°C and at atmospheric pressure) and dry steam (unsaturated at the test temperature of 150°C and atmospheric pressure) also showed expected low general corrosion penetration rates, calculated to be comparable with those in the well J-13 water (McCright et al., 1987). However, all the specimens tested in the dry steam showed weight gains, and specimens tested in the wet steam showed weight losses. For specimens that showed weight gains, the results are expressed in terms of the corresponding amount of metal that would have been converted to oxide to create the increase (McCright et al., 1987).

Corrosion tests have been performed on AISI 304L SS coupons under irradiated environmental conditions. A 1-yr test was conducted at room temperature in partially aerated (5 ppm measured oxygen content) well J-13 water containing crushed tuff, with an average gamma dose rate of 6×10^5 rads/h (Juhas et al., 1984). An identical but nonirradiated test was conducted simultaneously for control. The nonirradiated coupons showed higher corrosion rates than comparable irradiated coupons although the amount of corrosion penetration was very small in all cases. No evidence of intergranular penetration was noted on specimens that were metallographically sectioned and examined.

7.4.2.4.2 Summary and analysis of general corrosion and oxidation testing.

In summary, the general corrosion rates of the candidate stainless steels listed in Table 7-10 are quite small and, thus far, do not seem to be significantly affected by (1) alloy composition, (2) temperature (28 to 150°C), (3) exposure time (up to 11,000 h), (4) irradiation (background to more than 1×10^5 rad/h), or (5) whether the environment was wet or dry. A value of 0.2 micrometer per yr would appear to be adequate to describe conservatively the rate data so far acquired (Oversby and McCright, 1984). Extrapolation of these rates indicates that a container made of any of the

Table 7-10. General corrosion rates of candidate austenitic stainless steels in well J-13 water at different temperatures[a]

Alloy	Temp (°C)	Time (h)	Medium	Corrosion rate (μm/yr)[b] Average	Standard deviation
304L	50	11,512	Water	0.133	0.018
316L	50	11,512	Water	0.154	0.008
825	50	11,512	Water	0.211	0.013
304L	80	11,056	Water	0.085	0.001
316L	80	11,056	Water	0.109	0.005
825	80	11,056	Water	0.109	0.012
304L	100	10,360	Water	0.072	0.023
316L	100	10,360	Water	0.037	0.011
825	100	10,360	Water	0.049	0.019
304L	100	10,456	Saturated steam	0.102	(c)
316L	100	10,456	Saturated steam	0.099	(c)
825	100	10,456	Saturated steam	0.030	(c)
304L	150	3,808	Unsaturated steam	0.071	(c)
316L	150	3,808	Unsaturated steam	0.064	(c)
825	150	3,808	Unsaturated steam	0.030	(c)

[a]Source: McCright et al. (1987).
[b]Average of three replicate specimens of each alloy in each condition.
[c]Not determined.

candidate materials with a 1-cm-thick wall would not be penetrated as a result of general corrosion for well over 1,000 yr.

A model is being developed to quantify general corrosion and to allow for extrapolation of the data to long time periods (Section 8.3.5.9.3, Information Need 1.4.3).

7.4.2.5 Intergranular corrosion and intergranular stress corrosion cracking

Experience with the candidate austenitic materials has shown that they are susceptible to stress corrosion cracking. The crack propagation path (intergranular or transgranular) is often an important indication of the causative mechanism for stress corrosion and therefore serves as the basis for modeling activities for projecting container lifetimes. In some instances, both intergranular and transgranular crack paths occur but one or the other cracking pattern dominates. Cracking along intergranular paths (and intergranular attack without stress assistance) is the subject of this section.

Intergranular corrosion attack (IGA) and intergranular stress corrosion cracking (IGSCC) are most frequently associated with a sensitized microstructure as a necessary precursor to attack. A sensitized microstructure develops when chromium carbide precipitates from solid solution (austenite or gamma phase), leaving a region depleted in chromium in the vicinity of the precipitate. This precipitation occurs most frequently along grain boundaries, and it can lead to serious degradation when the precipitates and resulting chromium-depleted zones form a continuous network across the container thickness. The protective passive film on the sensitized grain boundary is not as stable as that on the bulk material because the chromium content is lower (less than 12 weight percent in the boundary compared with 18 to 20 weight percent in the bulk). When the sensitized material is exposed to oxidizing aqueous environments, the grain boundary area tends to corrode preferentially, and the attack can proceed along the continuous network of chromium-depleted zones; hence, intergranular attack occurs. In the presence of applied stress on a sensitized stainless steel, preferential intergranular attack is favored under less aggressive environmental conditions than intergranular attack without stress assistance and results in intergranular stress corrosion cracking.

There have been reported observations of IGSCC occurring in nonsensitized material. The most common occurrence is in conjunction with transgranular stress corrosion cracking (TGSCC) in chloride-containing environments where the crack propagation path has both a transgranular and an intergranular component. This has been discussed by Cowan and Gordon (1977), primarily for AISI 304 stainless steel in concentrated boiling magnesium-chloride ($MgCl_2$) solutions. The transgranular path is the dominant of the two paths in chloride-containing environments. The intergranular component appears to be favored more when the chloride concentration is low, the temperature is low, and when alloying additions are made to the base material. As discussed in Section 7.4.2.6 on TGSCC of austenitic materials, the conditions that favor development of the intergranular path over the transgranular path are the same conditions that favor an overall increase in resistance to stress corrosion cracking regardless of the propagation path.

Another environment in which both crack propagation paths can occur in nonsensitized material is in hot concentrated caustic solution (Wilson and Aspden, 1977). Polythionic acid is another reagent that occasionally causes stress corrosion cracking of nonsensitized stainless steel (Theus and Staehle, 1977). These environmental conditions appear to be extremely unlikely to develop under any scenarios envisioned for the repository.

The experience in BWR cooling water does appear to be applicable, at least as a starting point, for material testing and modeling activities. AISI 304 stainless steel is used for the recirculating piping around the reactor core and many incidents of stress corrosion cracking have occurred. All the cracking has been intergranular and all has been associated with sensitized microstructures occurring in the heat-affected zones in welded sections. The mildly oxidizing condition of both the BWR cooling water and that of ground water associated with the geological formation at the Yucca Mountain repository site suggest that the two environments are somewhat comparable.

7.4.2.5.1 Detection of sensitized microstructures

Chromium carbide precipitation is favored by exposure of austenitic stainless steels to moderately high temperatures (generally in the 500 to 800°C range) for periods ranging from minutes to many hours. The carbon content of the stainless steel is an important factor in determining susceptibility to sensitization. Time-temperature-sensitization (TTS) curves have been generated for the austenitic stainless steels; an example is shown in Figure 7-11. During post weld cooling, the heat-affected zone near the weld can cool at a rate such that it enters the "nose" of the TTS curve. Lower-carbon stainless steels such as AISI 304L with 0.03 weight percent maximum carbon have been developed to resist sensitization. For these alloys, only prolonged heating would move the material into the nose of the curve. A considerable amount of literature is available on the details of sensitization and the ways of detecting it (Clarke et al., 1978; Streicher, 1978).

Some difficulties are encountered in applying American Society for Testing and Materials (ASTM) tests to predict sensitization. More recently, tests involving electrochemical polarization of a specimen exposed to an acid thiocyanate solution have been developed. Sensitized microstructures give a polarization transient with a shape significantly different from that of nonsensitized structures, and integration of the area under the curve is quantitatively proportional to the amount of grain boundary area attacked. These electrochemical tests are called electrochemical potentiokinetic reactivation (EPR) tests (Majidi and Streicher, 1984).

Materials and processes that lead to gross sensitization would not be used in waste packages, but partial sensitization occurring along a limited number of grain boundaries or other pathways might occur. The degree of partial sensitization and the time required to achieve this condition needs to be quantified to predict the degradation mode by intergranular mechanisms. Although the EPR technique and the ASTM A 262 test methods (ASTM, 1982) use environments that are considerably more aggressive than the expected service conditions, these techniques are useful for screening materials and conditions that could lead to intergranular corrosion attack and intergranular stress corrosion cracking. Work using these techniques is planned and is discussed in Section 8.3.5.9.

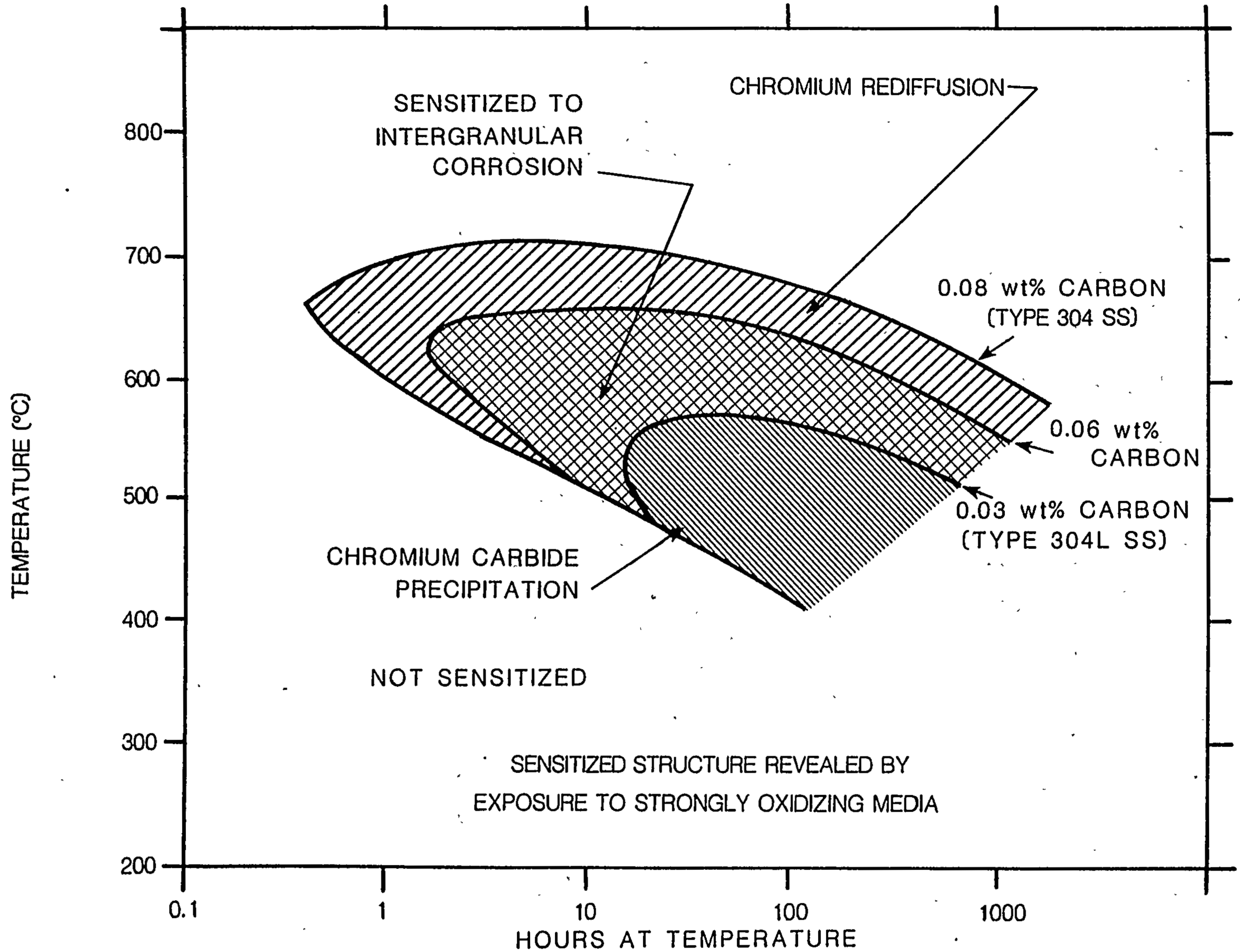


Figure 7-11. Time-temperature-sensitization curves for AISI 304 and 304L stainless steels (ss). Cross-hatched region indicates where sensitized microstructures develop as a function of heating time, heating temperature, and carbon content in the alloy. Sensitized microstructures occur when chromium carbides precipitate and deplete local areas of chromium (lower bound of hatched region); heating for longer periods of time restores original chromium content in these areas (upper bound of hatched region). Modified from Shreir (1976).

7.4.2.5.2 Tests to detect intergranular stress corrosion cracking susceptibility

Several intergranular stress corrosion cracking (IGSCC) tests are underway with specimens of AISI 304L, AISI 316L, and Alloy 825 steels in various material conditions. Four-point loaded, bent-beam specimens were machined from cold-worked welded plate of these materials. The bent-beam configuration was chosen so that the base metal, weld zone, and heat-affected zone could all be simultaneously stressed at the same nominal level. Some specimens were given intentional heat treatments to develop a sensitized microstructure; then these specimens were stressed and exposed to well J-13 water and to steam. None of the specimens exhibited cracking after 2,000 h of testing for those having no postweld heat treatment and after 4,000 h of testing (for specimens given the postweld sensitizing treatment (Juhas et al., 1984)).

Results from U-bend specimens of AISI 304 and 304L (the latter having 0.017 weight percent carbon) stainless steel exposed to irradiated well J-13 water and water vapor have been discussed by Juhas et al. (1984). The self-loading nature of the U-bend permits its use in the small available working space in the irradiated autoclaves. Maximum stress levels in U-bends substantially exceed the initial yield stress. These materials were in the solution-annealed (15 min at 1,050°C) and sensitized (24 h at 600°C) conditions. Tests were performed in two autoclaves, one at 50°C and the other at 90°C, in a cobalt-60 irradiation facility at irradiation absorbed dose rates of 6×10^5 and 3×10^5 rads/h, respectively. Each autoclave was divided into three zones: (1) water and rock (bottom), (2) rock and saturated moist air (middle), and (3) moist air only (top). The air in the test vessel was purged daily with fresh air. The operating pressure in the vessel was essentially atmospheric so that the air space in the vessel was saturated with water vapor at the operating temperature. Specimens were examined after 1 month and after 3 months of exposure. In the 50°C test, two specimen failures were recorded: one after 1 month of exposure and one after 3 months of exposure. Both specimens were sensitized AISI 304, located in the saturated moist air-only region of the autoclave. In the 90°C study, two sensitized AISI 304 SS specimens, both from the water and rock region, had failed after 1 month of exposure. The 3-month exposure inspection showed two additional failures: both were AISI 304, one in the saturated moist air only region and one in the moist air and rock region of the autoclave. Metallographic examination of the fractured specimens revealed IGSCC; an example is shown in Figure 7-12.

The observed stress corrosion susceptibility of the sensitized AISI 304 was expected. None of the AISI 304L SS specimens have cracked intergranularly, but they have exhibited transgranular stress corrosion cracking (TGSCC) in subsequent inspections (Section 7.4.2.6). The larger number of failures occurring in the irradiated moist air phase suggests that a more severe environmental condition exists there, generated by radiolysis of the environment without dilution or buffering of the species produced, than in the liquid phase.

Essentially, the slow strain rate test is a determination of the effect of the environment on degradation of mechanical properties (yield strength, tensile strength, elongation, and reduction in area). Specimens are loaded

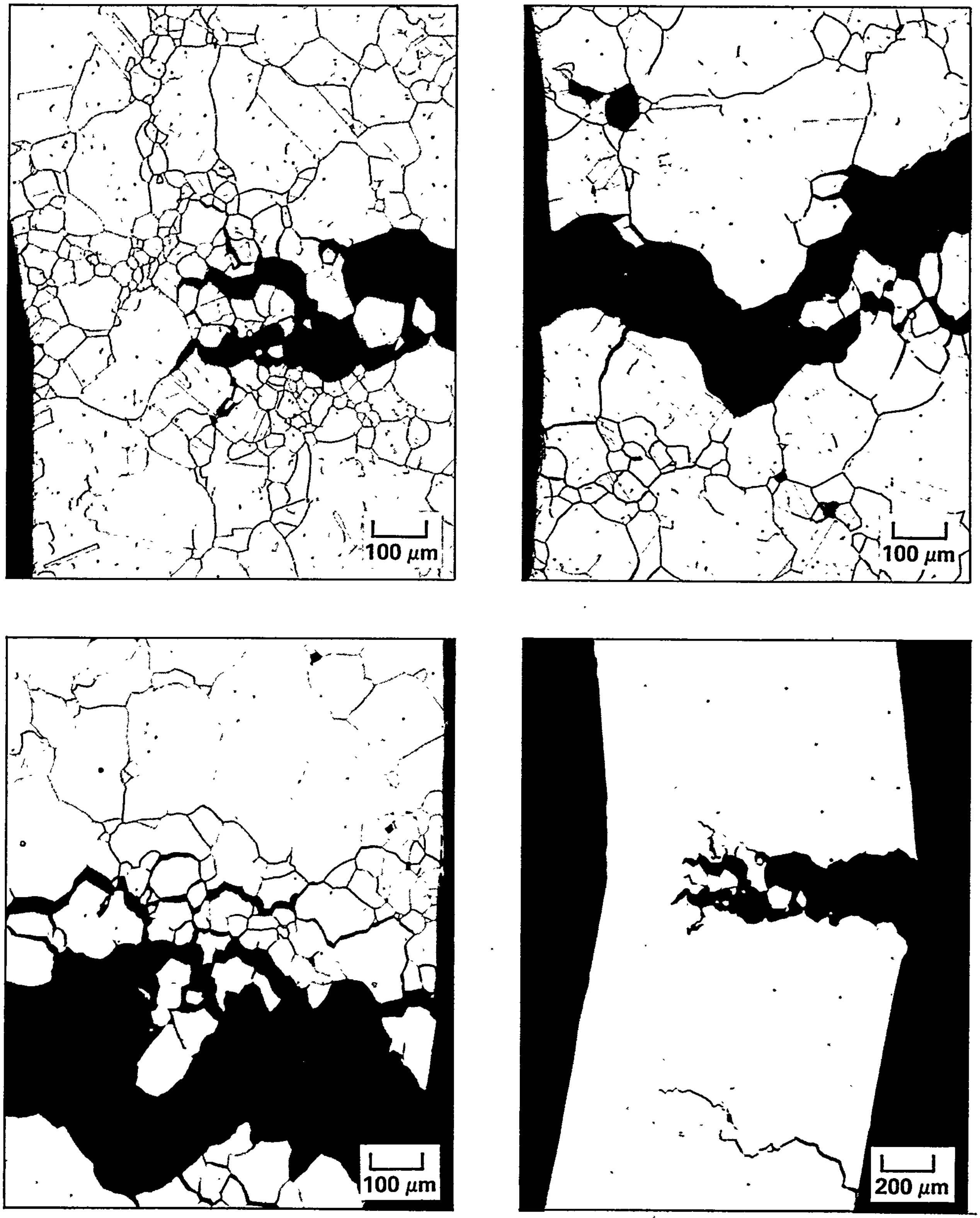


Figure 7-12. Metallographic cross sections of sensitized U-bend specimens of 304 stainless steel showing intergranular stress corrosion cracking. Modified from Juhas et al. (1984).

until fracture occurs and the fractured surface is examined, usually by scanning electron microscopy. A stress corrosion failure is indicated by a brittle fracture with evidence of grain facets and cleavage planes. In the absence of stress corrosion, the specimen fails by overload, and the failure is ductile, with characteristic plastic deformation. In many respects, the slow strain rate test resembles a common tensile test, but the specimens are strained at slow crosshead speeds (less than 1 x 10^{-4} s^{-1}). Straining at these low rates allows study of the competition between the rate of slip step emergence and the rate of environmental interaction with this new surface, the interaction causing the dissolution or the passivation of the new surface. Mechanisms of stress corrosion cracking in metals involve the transition from passive to active behavior.

Results of testing the austenitic materials in 150°C aerated well J-13 water are summarized in Tables 7-11 and 7-12 (Juhas et al., 1984). These initial tests were performed at this more elevated temperature to induce failure in some of the test specimens. The AISI 304 material contained 0.054 weight percent carbon and was tested in both the mill-annealed (as received from the vendor) condition and in the solution-annealed and sensitization-treated condition (1,050°C for 15 min and water quench, followed by 600°C for 24 h and air cooled). The AISI 304L material contained 0.024 wt. percent carbon and was tested in both the solution annealed (1,050°C for 15 minutes and water quenched) and the solution annealed and sensitization-treated condition (600°C for 10 h and air cooled). The AISI 304 specimens in the solution annealed and sensitized condition failed intergranularly with a significant drop in ductility when the strain rate was reduced from 1 x 10^{-4} to 2 x 10^{-7} s^{-1} in well J-13 water at 150°C. Cracks were found along the gage section of these specimens, and the fracture surfaces showed clear evidence of intergranular fracture. Neither the solution-annealed nor the solution-annealed and sensitization-treated AISI 304L samples exhibited stress corrosion under these test conditions. Similar tests are planned for AISI 316L and Alloy 825 materials, as discussed in Section 8.3.5.9.

7.4.2.5.3 Low temperature sensitization

Although the L grades of stainless steel do not appear susceptible to sensitization for the length of time the container might remain in the 500 to 800°C temperature range, a long-term, low temperature sensitization might occur during the tens to hundreds of yrs that the container will experience surface temperatures in the 100 to 270°C range after emplacement in the repository. This possibility of low temperature sensitization (LTS) was analyzed by Fox and McCright (1983), who concluded that heavily cold-worked AISI 304L SS (0.028 weight percent carbon) could develop a sensitized structure after as little as 10 yr at an isothermal surface temperature of 280°C. This analysis was based on extrapolating the results from the work of Briant (1982) in support of investigations on sensitized 300-series stainless steels under BWR environmental conditions. Reduction of the container surface temperature in the repository could cause a significant increase in the time required to develop a sensitized microstructure; for instance, the stainless steel in Briant's (1982) work would take 120 yr to sensitize isothermally at 250°C and 4,000 yr at 200°C (Fox and McCright, 1983). In

Table 7-11. Results of slow strain rate tests on AISI 304 stainless steel at 150°C[a]

Environment	Strain rate (s^{-1})	Reduction of area (%)	Elongation (%)	Yield strength (ksi)	Ultimate strength (ksi)
		MILL-ANNEALED SPECIMENS			
Air	$1x10^{-4}$	80.2	48.0	37.4	74.4
Air	$2x10^{-7}$	76.5	45.0	35.9	76.6
J-13[b]	$1x10^{-4}$	77.9	47.0	36.1	75.3
J-13	$1x10^{-4}$	79.6	46.0	36.3	74.9
J-13	$2x10^{-7}$	75.7	50.0	33.5	77.5
J-13	$2x10^{-7}$	76.4	47.0	35.1	77.0
		SOLUTION-ANNEALED AND SENSITIZATION-TREATED SPECIMENS			
Air	$1x10^{-4}$	72.2	50.6	21.9	68.0
Air	$1x10^{-4}$	66.5	51.5	26.0	68.8
J-13	$1x10^{-4}$	75.5	53.5	23.5	68.8
J-13	$1x10^{-4}$	74.9	51.0	23.5	69.0
J-13	$2x10^{-7}$	50.9	ND[c]	22.0	70.1
J-13	$2x10^{-7}$	26.4	ND[d]	20.7	64.5

[a]Source: Juhas et al. (1984).
[b]Air-sparged well J-13 water.
[c]ND = not determined.
[d]Broke at gage mark.

reality, the container surface will cool continuously. Therefore, predictions based on isothermal conditions are very conservative. A subsequent analysis (Oversby and McCright, 1984) indicated that with a peak surface temperature of 250°C followed by cooling at a rate predicted from a heat transfer code under certain reference waste package conditions (Hockman and O'Neal, 1984), even the high carbon, heavily cold-worked AISI 304L SS used in Briant's (1982) work would not sensitize (Fox and McCright, 1983). Results of this analysis are illustrated in Figure 7-13.

Development of LTS can be retarded in other ways beside reduction of the peak surface temperature of the container. The mechanism for development of LTS is discussed by Povich (1978) and requires a highly cold-worked structure for nucleation of subcritical carbide precipitates that are formed in the usual sensitization temperature range (above 500°C) but are not discernible by any of the American Society for Testing and Materials (ASTM) tests. These

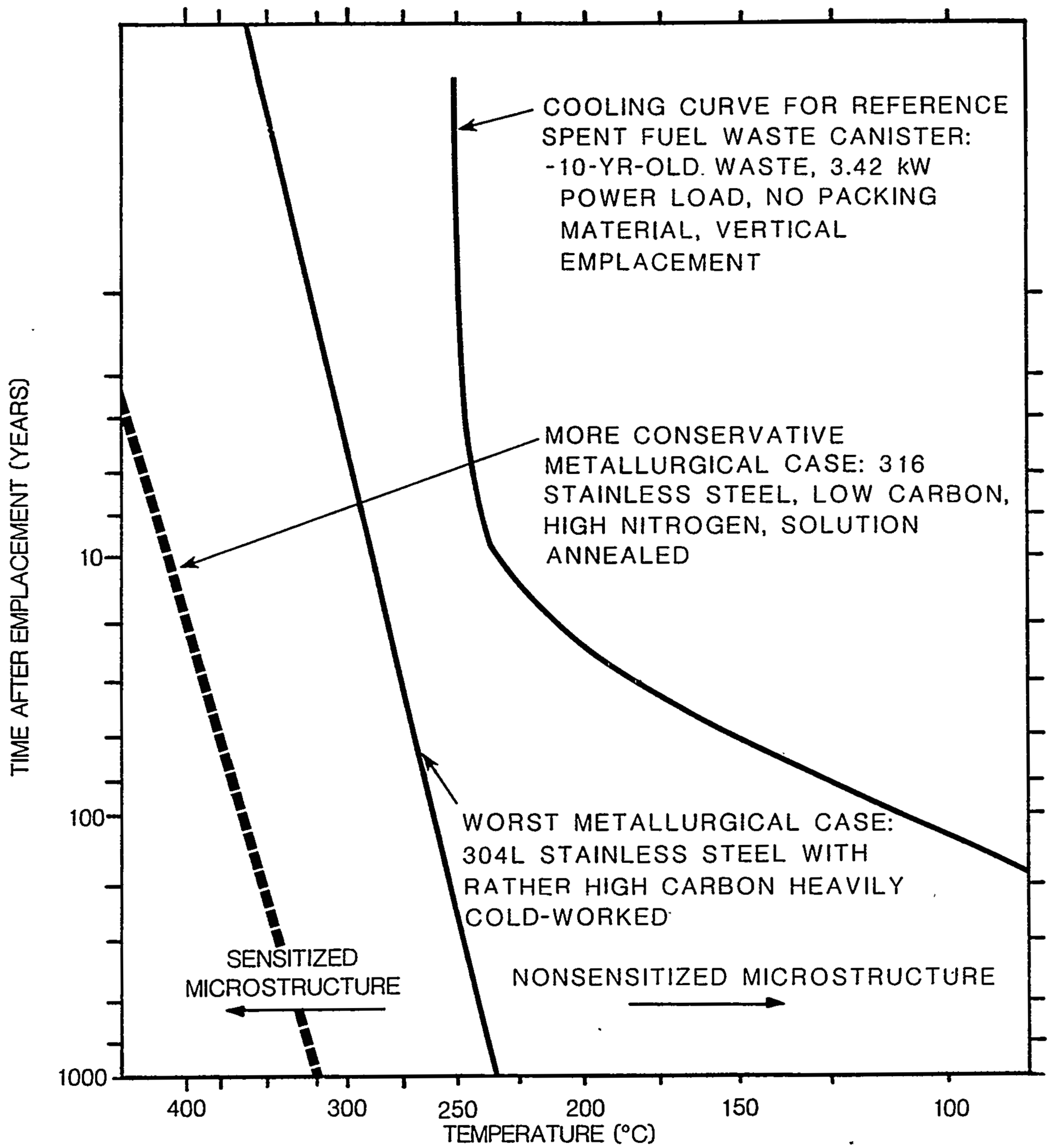


Figure 7-13. Relationship between thermal history of emplaced nuclear waste containers and long-term sensitization of austenitic stainless steels. Modified from Oversby and McCright (1984)

Table 7-12. Results of slow strain rate tests of AISI 304L at 150°C[a]

Environment	Strain rate (s^{-1})	Reduction of area (%)	Elongation (%)	Yield strength (ksi)	Ultimate strength (ksi)
		SOLUTION-ANNEALED SPECIMENS			
J-13[b]	$1x10^{-4}$	80.5	54.0	25.8	68.4
J-13	$1x10^{-4}$	78.4	52.0	27.1	68.2
J-13	$2x10^{-7}$	68.7	48.0	28.4	67.7
J-13	$2x10^{-7}$	72.9	46.3	26.7	68.2
		SOLUTION-ANNEALED AND SENSITIZATION-TREATED SPECIMENS			
Air	$1x10^{-4}$	73.7	49.0	29.4	68.6
J-13	$1x10^{-4}$	72.2	49.6	ND[c]	ND
J-13	$1x10^{-4}$	74.8	51.6	29.6	69.1
J-13	$2x10^{-7}$	76.0	49.0	26.6	68.8
J-13	$2x10^{-7}$	70.4	48.0	27.2	68.8

[a]Source: Juhas et al. (1984).
[b]Air-sparged well J-13 water.
[c]ND = Not determined.

subcritical precipitates then grow when the specimen is exposed to temperatures below 500°C for prolonged periods. Obviously, reduction of the amount of cold work in the specimen vitiates development of LTS.

Work performed in response to the concerns of intergranular stress corrosion cracking (IGSCC) in BWR piping indicates that the molybdenum-bearing AISI 316L SS is considerably more resistant to LTS (Briant et al., 1982; Mulford et al., 1983). The same group also tested special grades of 316 and 304 SS with extra low carbon (<0.02 weight percent) and with intentionally high nitrogen contents (>0.06 weight percent) and found that these compositions produced beneficial effects. Special nuclear grades of 316 SS with high nitrogen and low carbon are recommended as replacement materials for 304 SS piping in BWRs (Proebstle and Gordon, 1982; Danko, 1984). It appears that the molybdenum and nitrogen additions impede the diffusion of carbon atoms in austenitic stainless steels so that the growth rates of carbide nuclei are slowed, and materials containing these alloy additions are considerably more resistant to sensitization both above and below 500°C.

Some additional factors need to be considered to assess the possibility of developing a sensitized microstructure and resulting intergranular corrosion cracking (IGA) and IGSCC. With sufficient time and temperature, chromium could rediffuse from the bulk to the depleted regions and restore a uniformly high chromium content throughout the alloy. This condition would eliminate any preference for localized attack due to the lack of chromium and instability of the protective film at these locations. The activation energy for chromium diffusion is considerably higher than that for carbon diffusion so that the restoration process is slow (chromium rediffusion is indicated in the series of curves on the top-right side of the time-temperature sensitization curves (Figure 7-11)). At very long times at lower temperatures (200° to 300°C), it is possible that the series of curves bounding the sensitized region on the bottom and left in Figure 7-11 would meet those bounding this region on the top and right. Physically, this would mean that the carbide precipitates would be nucleated, but the growth of these precipitates would be so slow that the chromium content would never be locally depleted. This situation would not produce a sensitized microstructure.

On the other hand, situations that would enhance carbon diffusion at lower temperatures (such as movement of carbon atoms along dislocation pipes) could readily favor sensitization. Diffusion along dislocations is relatively more significant at lower temperatures where matrix diffusion of atoms is very slow. This competition of different mechanisms was considered by Logan (1983), who applied a finite element analysis and concluded that AISI 304L SS would not sensitize under the projected repository thermal conditions. However, his analysis did not consider the full range of reported activation energies for the different diffusion processes. Figure 7-13 indicates that if sensitization were to occur at all, it would be evident in a few yrs at temperatures in the range 200° to 300°C. This time period lies within the realm of experimental practicality, and experiments of this duration will provide an important way to determine the predominance of either a potentially beneficial or potentially detrimental effect. Plans for this type of testing are described in Section 8.3.5.9.

7.4.2.5.4 Environmental effects on intergranular stress corrosion cracking susceptibility

A sensitized microstructure is considered as being necessary for intergranular stress corrosion cracking (IGSCC) to occur. The chemical environmental conditions are also important. The phenomenon has been studied mostly under the thermal and chemical environmental conditions relevant to BWR coolant (250° to 300°C high-purity water of the order of 1 Mohm-cm resistivity, under pressure, with varying oxygen contents usually ranging from 2 to 8 ppm). The general conclusion from this work is that the stronger the oxidizing potential of the environment, the greater the susceptibility for IGSCC in sensitized materials. The work of Fujita et al. (1981) indicated that gamma radiation (4.5×10^4 rads/h) further increased the stress corrosion susceptibility of sensitized AISI 304 SS in high-purity water. They used the slow strain rate test to determine the stress corrosion cracking (SCC) susceptibility and found that the fractures were all intergranular.

It is not clear whether the ionic content of the water has any effect on susceptibility of stainless steels to IGSCC. Unlike other corrosion degradation modes on stainless steel, susceptibility toward IGSCC is probably not so strongly affected by the chloride ion. However, it is difficult to divorce all effects of the ions and the ionic strength of the solution from the processes of initiating and propagating a stress corrosion crack. Many theories of SCC consider a surface pit or crevice to be the site of initiation and the electrolyte content is important in establishing these sites (Staehle, 1971). It is also difficult to make clearly valid extrapolations of results obtained in the BWR environment to predict the behavior of waste package containers in the repository environment.

The analysis of Fox and McCright (1983) considered some of the mitigating effects that the environment might have, even on a sensitized stainless steel. First, the dominant environment for tens to hundreds of yrs after emplacement would be unsaturated steam at temperatures above 97°C. During this period, the gamma radiation field would be at its highest level; but without an electrolyte present, there would be high electrical resistance between local anodes and cathodes and, consequently, slow kinetics of oxidation. Mechanisms for aqueous corrosion become operable when a continuous moisture film is present, which appears to occur only at or below the boiling point and at relative humidities greater than 50 to 60 percent (Shreir, 1976). By the time the container surface temperature cooled to where condensation is possible, the radiation field for most of the packages would be a few rad/h. At this level of radiation, radiolytically induced changes in the aqueous environment may not have significant effects on corrosion. Experimental work thus far completed has not indicated any intergranular stress corrosion initiation in the wet simulated repository environment for the low-carbon austenitic stainless steels (McCright et al., 1983; Juhas et al., 1984).

7.4.2.5.5 Stress effects in intergranular stress corrosion cracking susceptibility

Stress is also an important factor in assessing stress corrosion susceptibility. Work performed in regard to stress corrosion cracking (SCC) of AISI 304/SS piping in BWR environments characteristically has shown that a threshold stress on the order of 70 percent of the yield strength is needed to initiate SCC, the actual threshold stress being a function of temperature, ionic concentrations, and degree of sensitization (Bruemmer and Johnson, 1984). The waste container body, including any assembly welds, could be annealed after forming to reduce residual stress. Reduction of the residual stress would also decrease the tendency for low temperature sensitization to occur during prolonged exposure to the thermal conditions of the repository, because the annealing or stress relief process on the container would have eliminated many potential high-diffusivity paths for carbide growth and chromium depletion. Any annealing or stress relieving to be performed after the final closure weld was made would probably be impractical, and this region might retain a residual stress approaching the yield strength of the material. However, different processes leave the weld and heat-affected zones in different states of stress, and it is possible to choose a process that would leave at least the outside surface of the container in compression, to minimize the possibility of initiating SCC. The effects of different welding processes on increasing the resistance to SCC of AISI 304 SS in the BWR environment have been discussed by Iwasaki (1982). A leading process is to inductively heat the weld on one side and cool it on the other to superpose a favorably oriented thermal stress on top of the weld stress so that the surface that is exposed to the SCC causative environment is compressed. The ready accessibility of the container assembly welds is expected to be amenable to nearly any process required to demonstrate a high level of assurance that SCC will not occur. However, the closure weld presents limitations on the choice of processes that can alleviate stress corrosion concerns. This is discussed further in Section 7.4.2.7.

7.4.2.5.6 Alloying effects on intergranular stress corrosion cracking susceptibility

Another approach to minimizing sensitization is the use of the stainless steel grades with additions of alloying elements to stabilize the carbides. Recent alloy developments combine the best features of the low-carbon grades with those of the stabilized grades of stainless steels (Abe et al., 1982). The high alloy content of Alloy 825, its low carbon content, and titanium stabilization of the carbides confer a very high degree of resistance to sensitization.

7.4.2.5.7 Summary of testing and analysis to date

In summary, experimental work performed as of June 1986 has not shown any tendency for the L grades of stainless steels to stress corrosion crack intergranularly even when specimens are stressed to and beyond the yield strength and are given heat treatments that favor carbide precipitation. Exposure conditions have also been severe, including irradiated water and saturated moist air where moderately to strongly oxidizing conditions are obtained. An analysis aimed at determining whether low temperature sensitization could occur in nuclear waste containers indicated that heavily cold-worked AISI 304L SS might sensitize given the expected fabrication, welding, and storage in the thermal environment of the repository if temperatures were higher than the reference case. Viable alternatives to AISI 304L SS would be AISI 316 SS, with extra low carbon and high nitrogen modifications, and Alloy 825. Reduction of the peak surface temperature and stresses in the container would also alleviate intergranular stress corrosion susceptibility. This can be accomplished by reducing the residual stress arising from fabrication and welding of the container and also by reducing the peak temperature of the containers by appropriate design of the waste package and arrangement of waste packages in the repository. (It is advantageous to maintain the container surface temperature above the boiling point for as long as possible for overall corrosion considerations; however, from the point of view of retarding sensitization, it is advantageous to keep the surface temperature below approximately 250°C for the more susceptible materials.)

If the low-carbon grades of stainless steel were to become sensitized, the sensitization would likely be partial in that only some of the grain boundaries would be continuously depleted in chromium. Furthermore, the areas most likely to sensitize are confined to the region around the final closure weld in the container. These points are discussed more fully in Section 7.4.2.8.

A model to predict sensitization in austenitic stainless steels and an extension of this model to high-nickel materials, such as Alloy 825, are being developed. The essential feature of this model is consideration of the chromium diffusion rate as established by the chromium thermodynamic activity gradient between the activity of chromium in the bulk and that at the carbide-austenite interface. This model considers temperature, strain, and compositional effects in materials (Mozhi et al., 1985). Plans for further development of this model are discussed in Section 8.3.5.9.

7.4.2.6 Pitting corrosion, crevice corrosion, and transgranular stress corrosion cracking

These three corrosion degradation modes are governed principally by the composition of the aqueous environment when the concentration of certain ionic species in water exceeds some threshold amount. But just as the attack itself is localized, the causative environment may be localized and considerably different from the bulk environment. In most instances, the susceptibility to these degradation modes is dependent on the chloride ion concentration in the environment. A review paper by Nuttall and Urbanic (1981) discussed the levels of chloride ion needed to initiate pitting, crevice, and transgranular stress corrosion attack in AISI 304 SS. Much of their documentation on critical chloride levels (as sodium chloride) was based on the observations of Truman (1977). Figure 7-14 was adapted from their work and the chloride content of well J-13 water (approximately 7 ppm) was juxtaposed. The low level of chloride in well J-13 water suggests that AISI 304 SS would be safe from attack by these degradation modes and that a several-fold increase in the concentration would be needed before these kinds of corrosion attack would initiate. Because of the similarity in composition of major alloying elements, Type 304L is expected to behave similarly to Type 304 SS. AISI 316L and Alloy 825 are markedly more resistant to chloride-induced corrosion degradation modes, because higher thresholds of chloride ion concentration are required to initiate and propagate the corrosion attack.

Although these degradation modes have some features in common, there are important differences, particularly in the initiation phase of each mode. A brief discussion is presented here because the testing and analyses are based on aspects of the causative mechanisms; greater detail can be found in the reference text Fontana and Greene (1978).

Crevice corrosion is favored when differential concentration cells are established on a metal surface, particularly on a metal that shows active-passive behavior, as do stainless steels in many environments. The creviced area is starved for oxygen, and the passive film breaks down and cannot readily reform. Dissolution of the metal occurs at this resulting local anode, the metal ions hydrolyzing with the water to form a more acidic

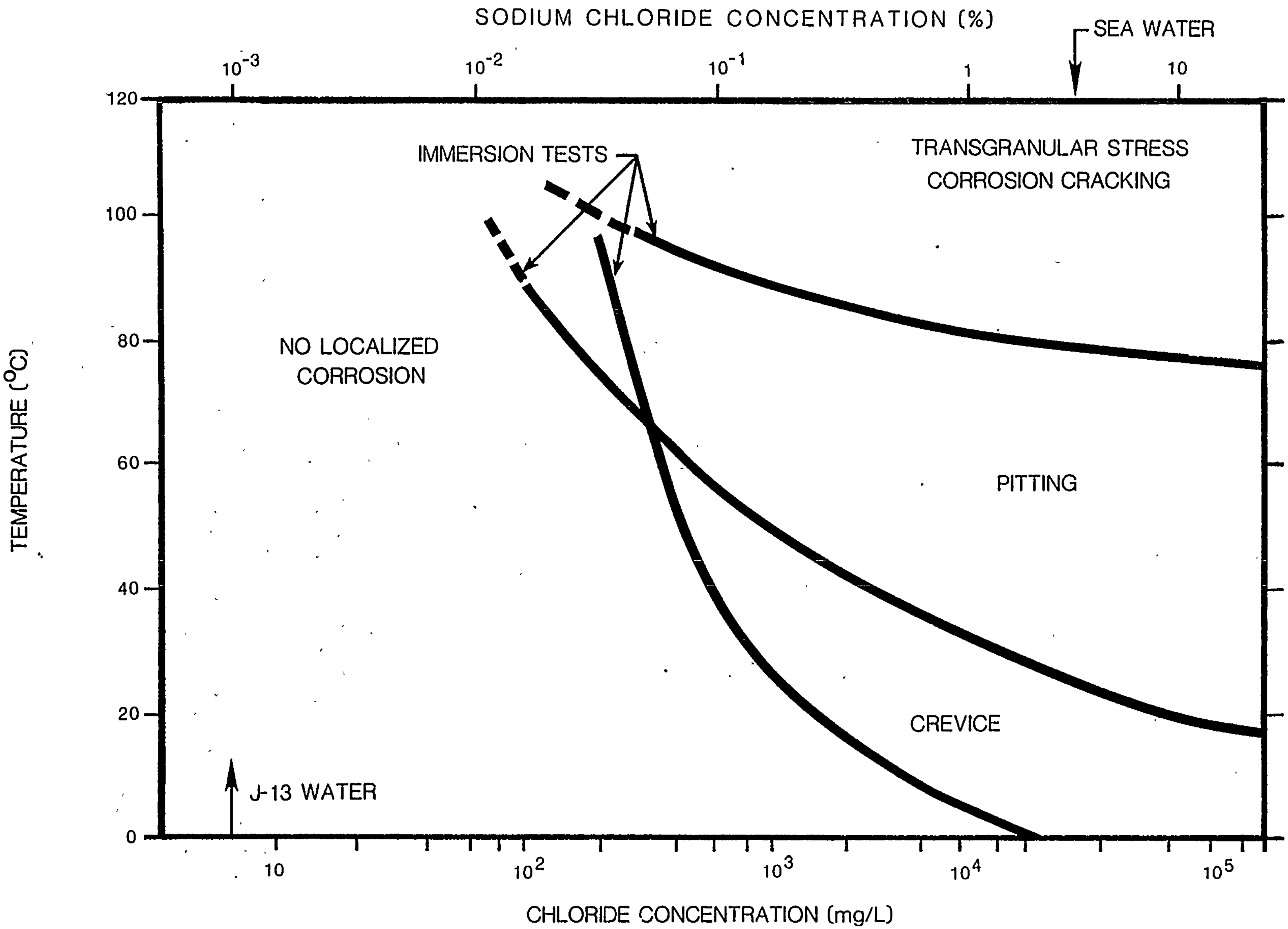


Figure 7-14. Temperature-chloride concentration thresholds for initiation of localized corrosion phenomena in sodium chloride solutions. Crevice, pitting, and transgranular stress corrosion cracking initiate to the right of their respective threshold curves as determined by immersion corrosion tests. For comparison purposes, the chloride ion concentration of well J-13 water is indicated. Modified from Nuttall and Urbanic (1981).

environment than the bulk environment. Anionic species, especially the mobile chloride ions, migrate to the anode and generally increase the ionic strength in the local environment. The local pH in the crevice region decreases, and the concentration of anionic species (such as chloride) increases. These conditions do not favor repassivation of the alloy, and a high dissolution rate of the metal in the creviced region continues.

Pitting corrosion is initiated at weak points in the passive film. Pits form preferentially at locations such as inclusions (especially certain sulfide-rich inclusions) on the exposed metal surface where the film readily breaks. Electrochemical differences between the inclusion and the bulk material stimulate local currents, and dissolution of the inclusion creates a local chemical difference that is aggressive toward the metal. Propagation of the pit occurs because of locally intense metal dissolution. The larger area of the metal surface remains passivated and cathodic. Many features of the propagation phase in pitting corrosion are similar to those for crevice corrosion.

Transgranular stress corrosion cracking (TGSCC) is due to the conjoint action of very localized anodic dissolution and applied or residual stress. The cracking frequently appears to initiate around a crevice or pit, and intense dissolution occurs at the crack tip while most of the crack wall acts as a cathode. Resistance to this kind of corrosion is obtained by adding alloying elements that decrease the electrochemical difference between "bare" areas (where the strain has transiently broken the protective passive film) and the areas still covered and protected by the film. Unlike the intergranular kind of stress corrosion cracking (SCC) discussed in Section 7.4.2.5, these cracks propagate transgranularly, and the propagation is not affected appreciably by sensitization of the material. In high chloride solutions (on the order of a few percent chloride ion), stresses on the order of 10 percent of the yield stress sustain cracking. The threshold stress decreases with increasing chloride ion concentration.

Pitting, crevice, and TGSCC are most significantly affected by the alloy composition (particularly the major alloying constituents of chromium, nickel, and molybdenum and to some extent the minor constituents of sulphur, phosphorus, nitrogen, and manganese) and the environmental composition. These degradation modes are particularly favored in concentrated ionic environments, since the high ionic strength of these environments support the electrochemical currents between local anodes and cathodes. Furthermore, species that tend to acidify the environment and species that cause breakdown of passive films favor these forms of corrosion. The primary role of the beneficial alloying additions is probably their effect on increasing the rate of film repassivation (Latanision and Staehle, 1969).

7.4.2.6.1 Electrochemical testing to determine localized corrosion occurrence

A preliminary study by Glass et al. (1984) surveyed electrochemical parameters relating to general corrosion (e.g., corrosion potential and corrosion current) and to pitting and crevice corrosion (e.g., pitting potential and protection potential). These parameters were measured for the candidate materials in aqueous solutions believed to be characteristic of the repository site. In addition to using well J-13 water, certain intentional additions were made to study the effect of chloride ion and, in some instances, the ionic species in well J-13 water were concentrated by boiling down. Test procedures and details are discussed by Glass et al. (1984). The principles and theory of electrochemical techniques for evaluating corrosion susceptibility of metals are described by Barnartt (1977), Mansfeld (1977), and Verink (1977).

To assess the susceptibility of the candidate materials to localized corrosion (pitting and crevice attack) in well J-13 water at different temperatures, cyclic anodic polarization curves were obtained. The potential of the working electrode (which is the candidate material of interest) was scanned by enforcing potentials anodic to the corrosion potential (E_{corr}) and then reversing the direction of the scan back to more negative values. (Note that the corrosion potential is the potential that the material naturally assumes in the solution as a result of corrosion processes, without having a potential artificially impressed upon it.) The scan rate was 1 mV/s (to ensure very nearly steady-state conditions), and potentials were measured relative to a saturated calomel electrode (SCE) at room temperature. Current flowing through the working electrode was continuously monitored during the scan. Such a scan, whose waveform is triangular when the impressed potential versus time is plotted, yields electrochemical values of interest such as the pitting potential (E_{pit}) and the protection potential (E_{prot}). The pitting potential is the potential above which pits spontaneously initiate and grow. The protection potential is the potential below which previously initiated pits repassivate and no new pits form. At potentials between the pitting and protection potential, new pits are not initiated but any previously initiated pits continue to grow. The values of the pitting and protection potential relative to the corrosion potential are indicative of the pitting susceptibility of the tested alloy in the testing environment.

The values of E_{pit}, E_{prot}, and E_{corr} depend on the technique used (particularly the potential scan rate). Slow scan rates approach the truly steady-state potentiostatic condition, and the pitting potential is usually observed to decrease at the slower scan rate values. For example, the pitting potential of AISI 316L SS in a 0.2 M NaCl solution is 0.429 V (SCE) at a potentiodynamic scan rate of 10 mV/s. The pitting potential decreases to 0.397 V (SCE) at a rate of 1 mV/s and to 0.391 V (SCE) at a rate of 0.1 mV/s (McCright et al., 1987). Projections of localized corrosion performance are best made on the basis of the potentiostatic determination (that is the scan rate approaching zero) of the pitting potential as well as the other important electrochemical parameters.

A typical anodic polarization curve for 304L stainless steel in 90°C well J-13 water is shown in Figure 7-15. The curve displays features common to all polarization curves obtained in well J-13 water. The important electrochemical parameters are identified on the figure. When the stainless steel is scanned anodically (to higher potentials) starting from E_{corr}, the 304L SS remains passive until the pitting potential is reached. At this characteristic potential, wholesale breakdown of the passive surface occurs and the anodic current density increases markedly by several orders of magnitude. The closer E_{pit} is to E_{corr}, the more susceptible the alloy would be to pitting in the event that an increase in the oxidizing potential of the

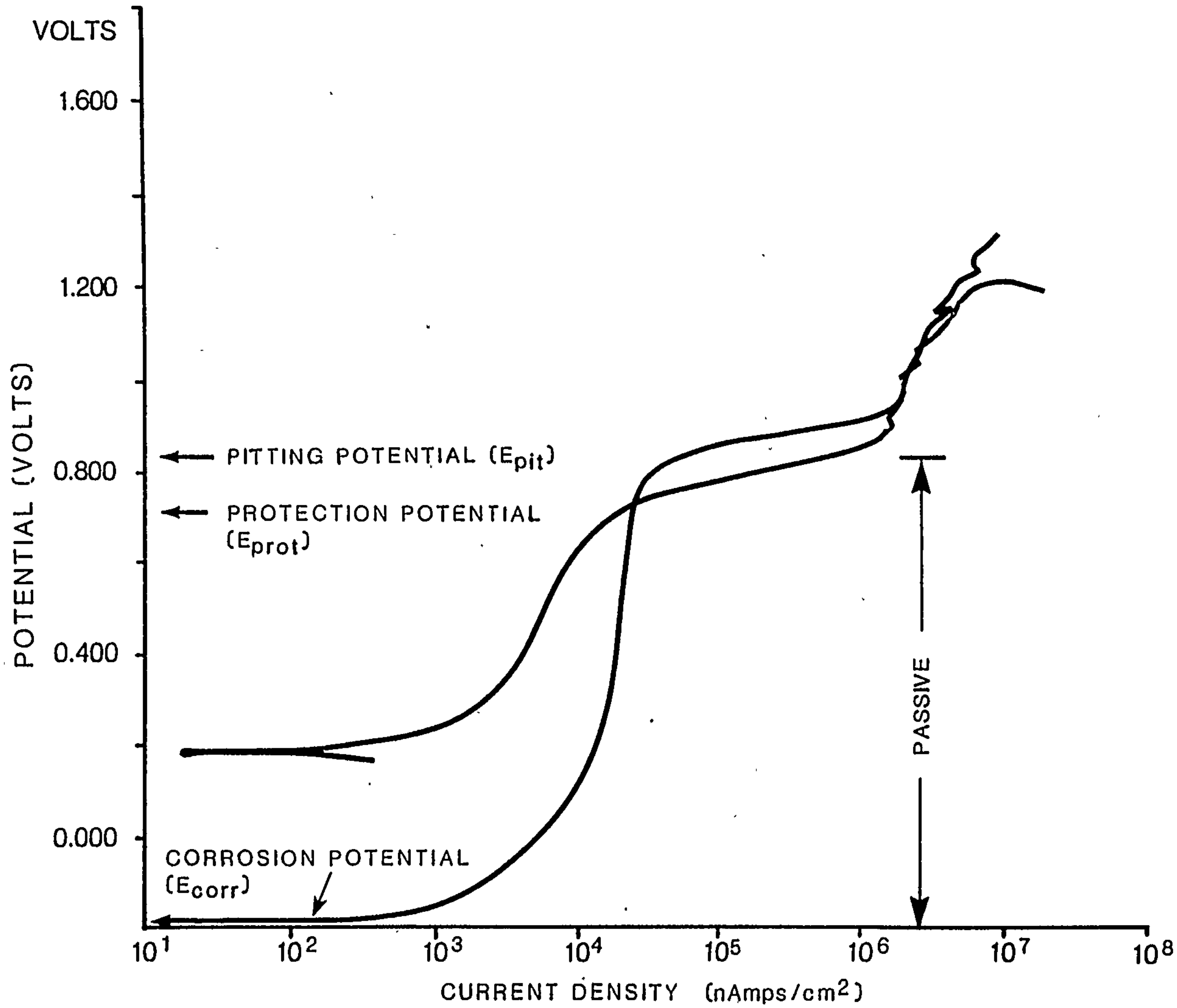


Figure 7-15. Potentiodynamic anodic polarization curve for AISI 304L stainless steel in well J-13 water at 90°C. Scan rate was 1 mV/s. Scan starts from E_{corr}. Line marked passive indicates that stainless steel remains passive until the pitting potential is reached. Modified from Glass et al. (1984).

environment occurred. This might occur, for example, in a radiation field where radiolysis products would shift E_{corr} to more positive values. From plots like that in Figure 7-15, tabulations of electrochemical parameters for some of the prospective container materials were made. The parameters for 304L SS, 316L SS, and Alloy 825 were all quite similar (Glass et al., 1984). Figure 7-16 summarizes the data for 304L SS. Over the range tested, no clear temperature dependence was found.

In all instances, the values of the pitting potential are sufficiently removed from those of the corrosion potential that no spontaneous pitting of these alloys should occur in well J-13 water. This result is the expectation from the summary figure, Figure 7-14. The other ionic species present in the well J-13 water, if they have any effect at all, would tend to retard pitting as the quantitative analysis by Szlarska-Smialowska (1974) has indicated. Although the halide ions chloride and fluoride increase the pit propagation rate, ions such as nitrate and bicarbonate tend to retard the propagation rate. The parameter (E_{pit} - E_{prot}) has been used previously by Wilde (1974) to evaluate alloys in terms of crevice corrosion resistance. As the value of (E_{pit} - E_{prot}) becomes larger, the resistance to crevice corrosion decreases. Also, greater difficulty in repassivating growing pits is indicated by larger values of (E_{pit} - E_{prot}). When the protection potential moves closer to the corrosion potential on the return scan, the metal surface is less easily repassivated allowing localized attack to continue.

As expected, when chloride ion is added to well J-13 water, the pitting and protection potentials move closer to the corrosion potential for 304L stainless steel. Glass et al. (1984) showed that an addition of 1,000 ppm by weight of NaCl to well J-13 water moved the pitting potential to within 100 mV of the corrosion potential, so that spontaneous pitting of the specimen could be observed. Their results are consistent with the data presented in Figure 7-14. The more highly alloyed 316L SS and Alloy 825 show more resistance in this environment, as would be expected, because of the well-documented role of molybdenum (and to a lesser extent nickel) in increasing the resistance to chloride-ion effects on inducing localized corrosion (Pessall and Nurminen, 1974).

Another consideration in predicting localized corrosion susceptibility is the surface condition of the alloy. The containers are expected to experience a high-temperature air/steam environment for a significant time after emplacement, with temperatures up to approximately 250°C (spent fuel packages). Some preliminary results (Glass et al., 1984) on 304L SS specimens with thermally formed oxide films indicate that these specimens were somewhat more resistant to pitting (lower current densities near the pitting potential) in 90°C well J-13 water with 1,000 ppm chloride addition than those that were not preoxidized. However, once these specimens started to pit and the applied potential was then shifted cathodically, the thermally oxidized specimens did not repassivate any more rapidly than those that were not previously oxidized; thus, the beneficial effects can be lost once pitting initiates. The work of Bianchi et al. (1974) indicates that at some temperatures oxides are formed that are more susceptible to pitting corrosion because of changes in the semiconducting properties of the oxide. Additional work is needed to resolve the effects of thermally grown films on the subsequent localized corrosion susceptibility. Plans for this work are outlined in Section 8.3.5.9.

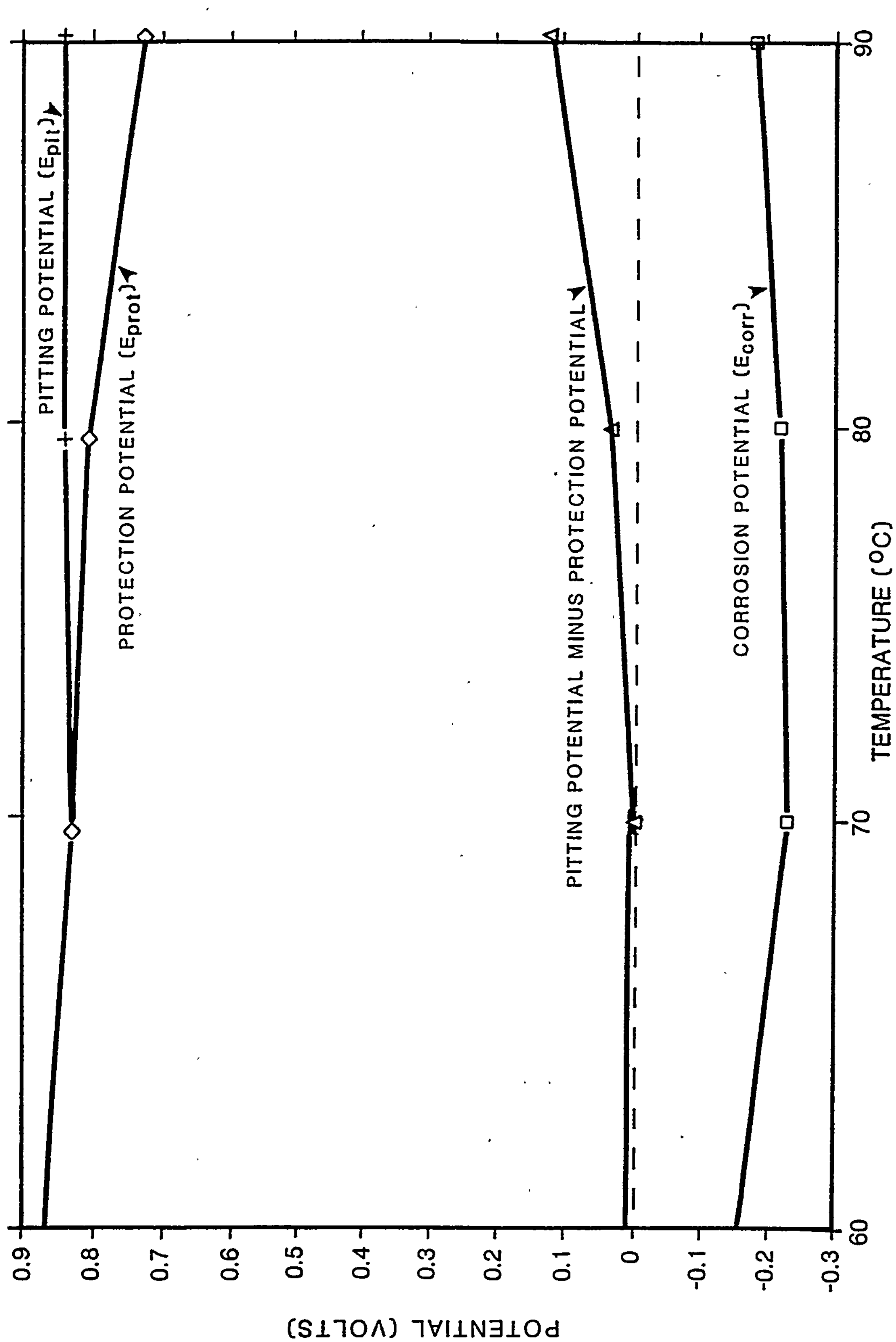


Figure 7-16. Electrochemical parameters for AISI 304L stainless steel in tuff-conditioned water from well J-13 as a function of temperature. All potentials are referenced to a saturated calomel electrode at 25°C. Modified from Glass et al. (1984).

7.4.2.6.2 Localized corrosion testing in gamma-irradiated environments

The combination of a concentrated aqueous environment and a sufficiently high gamma dose rate to promote radiolysis of the water is potentially one of the most demanding of the possible conditions for the container. Relatively little previously published work exists in this area. The co-existence of a high gamma field and a wet container surface is not an anticipated condition, because the gamma dose rate is expected to be at a low level when the container surface cools to the boiling point of water for the majority of the containers (Glassley, 1986; Van Konynenburg, 1986). However, waste packages placed at the periphery of the repository will cool more rapidly, and it is conceivable that some of these would become wet while the radiation level was high enough to radiolyze the aqueous environment.

Glass et al. (1985) measured the corrosion potentials of austenitic stainless steels in irradiated well J-13 water environments. Upon imposition of the gamma field (3×10^6 rad/h), the corrosion potentials of both 304L and 316L SS in a series of electrolytes related to well J-13 water shifted in the positive (oxidizing) direction, typically by 150 to 200 mV. The results for AISI 316L SS in a solution prepared by boiling down well J-13 water to one-tenth of its original volume are shown in Figure 7-17. In this figure, on refers to lowering the test cell into the center of the gamma source assembly (cobalt-60), and off refers to raising the cell about 1.5 m above the source assembly where the cell is shielded by the water in which the source assembly is submerged. Several on/off cycles are shown. Similar positive potential shifts upon imposition of the gamma field were observed for 316L SS in plain well J-13 water and in well J-13 water concentrated by a factor of 100, as well as for 304L SS in well J-13 water and in its concentrated versions. As expected, the gamma radiation field made the environment more oxidizing. This was shown to be very probably due to the production of hydrogen peroxide from radiolysis of the well J-13 water. Chemical analysis of the irradiated solution and a demonstration experiment in which drops of hydrogen peroxide were added to a fresh solution with simultaneous monitoring of the corrosion potential confirmed that the hydrogen peroxide concentration was in the range of 0.14 to 0.49 M and that hydrogen peroxide added by dropping produced the same potential shift as that produced by irradiation for similar concentrations.

Polarization curves were determined for some of the candidate stainless steels (304L, 316L) in irradiated well J-13 water. Results indicated that both the pitting potential and the corrosion potential were shifted in the positive direction by about the same amount. Thus, irradiation does not appear to increase the susceptibility to localized corrosion in unaltered well J-13 water. Some preliminary work by Glass et al. (1985) in which 316L SS was exposed to a solution of 650 ppm chloride (prepared by dissolving NaCl in deionized water) indicated a susceptibility of the material to pitting and crevice corrosion under those conditions. In this instance, the protection potential (on the reverse scan) lay at more negative values than the corrosion potential. In the same solution, but without irradiation, the protection potential lies at more positive potentials than the corrosion potential, a situation indicating that localized corrosion is not predicted. These polarization curves are depicted in Figure 7-18. These curves were obtained in deionized water with NaCl additions; the mitigating ionic species normally present in the well J-13 water were absent. This kind of work is being

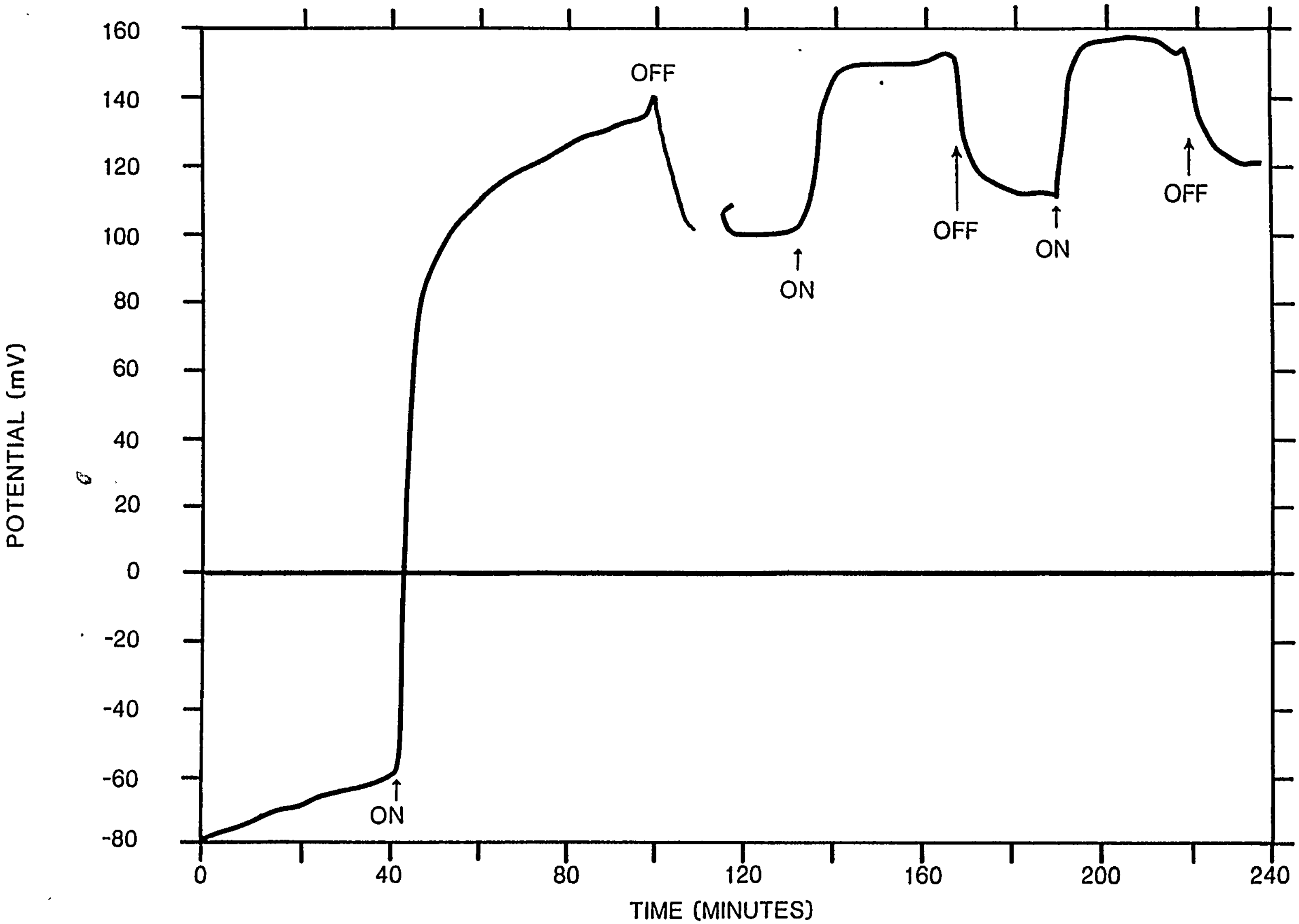


Figure 7-17. Corrosion potential behavior for AISI 316L stainless steel in water from well J-13 concentrated 10 times and under gamma irradiation. All potentials are references to a saturated calomel electrode. (The solution was not exposed to irradiation before the initiation of the first on/off irradiation cycle) Modified from Glass et al. (1985).

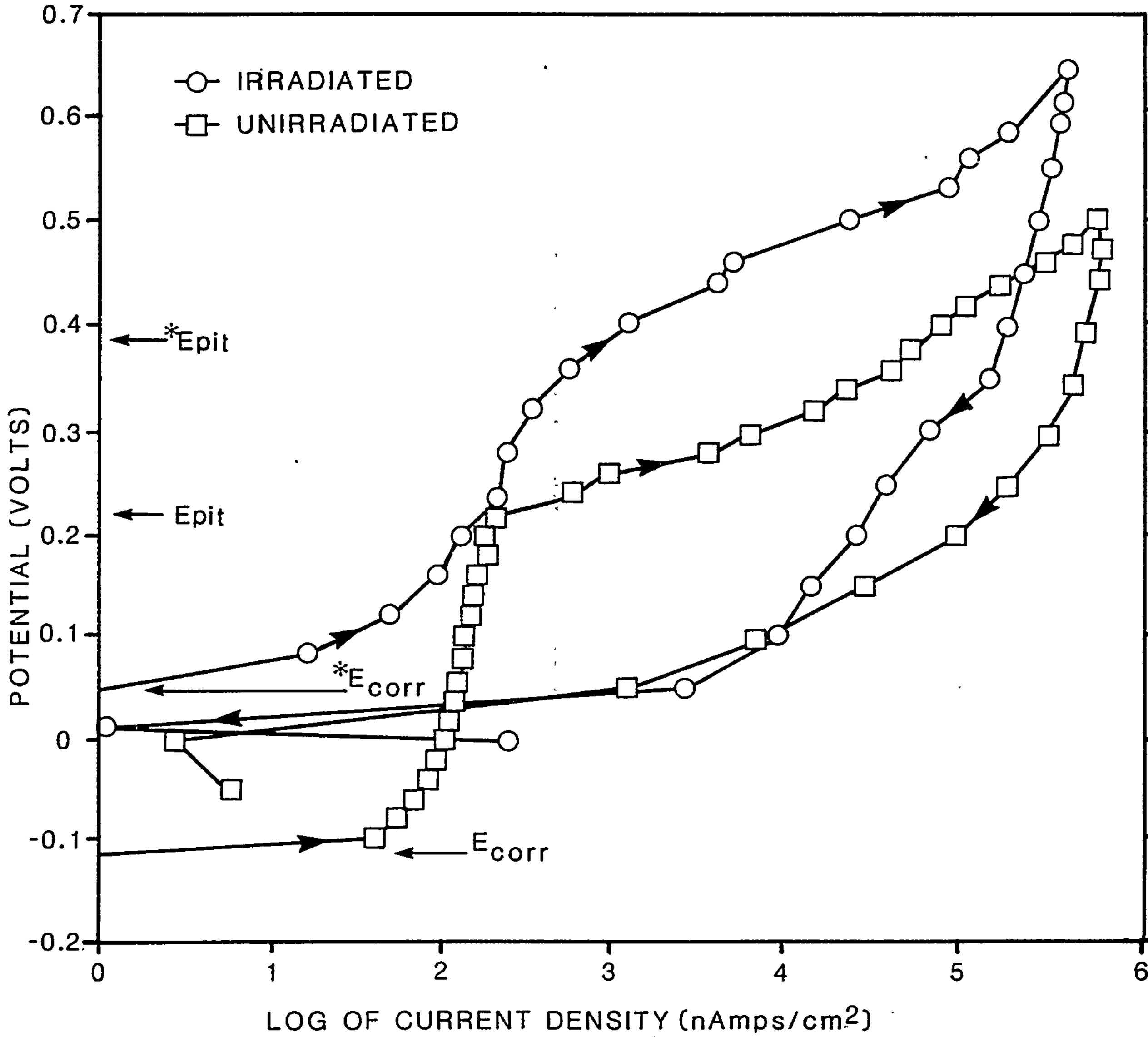


Figure 7-18. Comparison of the potentiostatic anodic polarization behavior for 316L stainless steel in 650 ppm chloride solution in deionized water with and without gamma irradiation. All potentials are referenced to a saturated calomel electrode. (The polarization curves were scanned anodically starting from the corrosion potential in each case. Upon reaching the anodic limit, the scans were reversed to more negative potentials. In this figure, E_{corr} and E_{pit} represent values of the corrosion potential and pitting potential, respectively, for the nonirradiated case. The corresponding values for the irradiated experiment are indicated on the figure as $*E_{corr}$ and $*E_{pit}$). Modified from Glass et al. (1985).

continued on other materials and at different concentrations and temperatures. Plans are outlined in Section 8.3.5.9.

An important benefit of the work described in this section is its application to conducting long-term tests at predetermined, controlled potentials. The goal of these tests is either to accelerate or to retard localized corrosion, depending on the choice of the controlled potential and its position relative to the critical pitting and protection potentials. This is an important part of the modeling effort for localized corrosion. It is discussed in greater detail in Section 8.3.5.9. This effort also relates to the work discussed under general corrosion (Section 7.4.2.4), where a model to predict the long-term changes in the corrosion potential is being developed. The chemical effects of gamma radiation can then be related to the change in corrosion potentials and critical potentials. Future work is needed at different gamma dose rates and temperatures to define the ranges of these electrochemical parameters. Plans for this work are given in Section 8.3.5.9.

7.4.2.6.3 Localized corrosion testing with creviced specimens

Metal-Teflon™-metal sandwich-type creviced specimens are being exposed for long periods of time to well J-13 water and to its concentrated versions with and without irradiation. The electrochemical polarization approach is useful to identify metal-environment combinations that might induce localized forms of corrosion, but confirming tests such as these are needed. Further, the initiation phase can require long exposure times in rather benign environments (such as what well J-13 water appears to be). An example of the type of testing used is found in Glass et al. (1984). They exposed (without any applied potential) 5 by 5 cm plates of stainless steel, bolted together with Teflon™ shims as spacers, for 54 d at 35°C in water with 10,000 ppm chloride. They found an average of 7.7 percent of the area of 304L SS specimens was attacked and 1.3 percent of the area of 316L SS specimens was attacked. The penetration depth was irregularly distributed over the attacked area and was not measured.

Along with the general corrosion tests, discussed in Section 7.4.2.4, Teflon™-metal crevices were created to observe whether preferential attack would occur there. These crevices were created with slotted washers that were used with the fastener assembly to support the coupons in the test environment. The data obtained so far (McCright et al., 1987) indicate that a tarnishing phenomenon is occasionally observed around the creviced area. After more than 10,000 h, the deepest attack in a crevice on 304L SS occurred in 90°C well J-13 water and was estimated at 0.3 micrometer while that for 316L SS occurred at 70°C and was estimated at 0.5 micrometer. Pitting attack as measured on the bold (uncreviced) surface was even less extensive than the crevice-tarnish attack. The aspect ratio (ratio of the depth of attack to the width of attack) was less than one. In true pitting of stainless steel the aspect ratio is significantly greater than one. Expressed on an annualized basis, these localized corrosion rates are of comparable magnitude to the general corrosion rate and are consistent with the conclusions drawn from the electrochemical polarization work. No enhancement of localized corrosion appears to occur in the unmodified well J-13 water under the conditions so far tested (McCright, 1985).

7.4.2.6.4 Activities to determine transgranular stress corrosion cracking susceptibility

In the discussion on intergranular stress corrosion cracking (IGSCC), (Section 7.4.2.5), it was reported that some of the U-bend specimens exposed to irradiated well J-13 water and saturated moist air showed transgranular cracking. Although this particular test had been designed primarily to show IGSCC susceptibility, broken specimens were examined to determine the crack propagation path. Test conditions are given in Section 7.4.2.5. After 14 mo of exposure in the 90°C test, two specimens of 304L SS that had been given a sensitizing heat treatment showed transgranular cracking. One of the cracked specimens was located in the moist air only zone, and the other was located in the moist air and rock zone. Subsequent inspections (after 23 mo of exposure) revealed that other specimens of both sensitization treated 304 and 304L SS and annealed 304 and 304L SS cracked transgranularly in the 90°C test. No specimens cracked transgranularly in the 50°C test, although intergranular cracking was noted for sensitized 304 material in this test. These observations are discussed in greater detail by Westerman et al. (1987).

Three major points, however, should be made before considering the results. First, when the research began, the tuff available was from the Bullfrog surface outcropping. Rock-water interaction studies by Oversby and Knauss (1983) show that this rock contains soluble salts (caliche or evaporite material) as a result of its location on the surface. The tuff taken from the depth proposed for the repository, however, did not contain such material (Oversby, 1985). Consequently, the electrolyte concentrations present in these tests (peak chloride concentration of 600 ppm) were considerably higher than would be expected in the proposed repository. This is particularly important in the case of chloride because of its role in stress corrosion cracking (SCC).

Second, these tests were viewed as corrosion tests rather than chemical experiments, and emphasis was placed upon maintaining an open system, sparged daily with fresh air, to simulate repository conditions. Attempts were not made to keep track of the total amount of water added or the final volumes of solution remaining.

Third, the temperature control in the intended 90°C experiment was unsatisfactory for the first month, which led to repeated boil-off of the solution and plugging of the access line. The temperature and the time history of the physical state of the corrosion medium are therefore uncertain for this period.

In view of these factors, it appears that these tests were considerably more severe than the repository application would be. Nevertheless, they provide information for comparison.

It appears that the transgranular stress corrosion cracking (TGSCC) resulted from the presence of a more concentrated electrolyte on the metal surface. In addition to purging the system of the radiolytically produced hydrogen and oxygen gas, it might be that the sparging (depending on how high a flow rate was used) blew the liquid up into the vapor space or produced aerosols of the dissolved solutes, which then deposited on the metal sample surfaces located in the vapor phase.

By whatever means the chloride concentrated (assuming that chlorine is the species responsible for the cracking), the effect of chloride ion in initiating cracking is increased when the environment becomes more oxidizing. Gamma radiation makes the environment more oxidizing. It is well established that the threshold levels of chloride to initiate cracking in 18-8 types of austenitic stainless steels are lowered when the environment is made more oxidizing. Williams (1957) showed that chloride levels as low as 2 ppm initiate TGSCC when the oxygen content is high (20 ppm) and that higher chloride concentrations are required to initiate cracking when the oxygen content is lower. However, his work was performed in a boiling situation of wet steam with intermittent wetting; therefore, it is likely that the real causative levels of chloride-induced cracking were higher.

In another experiment, a series of 40 U-bend specimens of 304 and 304L stainless steel, both in the solution-annealed and sensitization-treated condition, were exposed to unirradiated well J-13 water at 200°C in an autoclave (Westerman et al., 1987). The autoclave was cycled so that the specimens were exposed to the hot liquid water under pressure for 1 wk; then the liquid was allowed to boil to dryness by reducing the pressure while maintaining the same temperature. New water was then added and the system was returned to pressure and the cycle repeated. The intent of this experiment was to allow concentration of ionic species in well J-13 water to accumulate on the metal specimen surfaces. After 50 cycles (1 yr) of alternate wetting and drying, only the sensitization-treated 304 specimens had cracked, and these had cracked intergranularly, even though the experiment was planned primarily as one for investigating and accelerating transgranular cracking. Water analyses showed that, after 1 yr, the chloride content had reached 90 ppm and the fluoride content had reached 19 ppm in the water. The expectation is that some of the 304 and 304L SS specimens will eventually show TGSCC as the experiment continues, particularly those in the annealed condition.

Although rather extreme (in contrast to the expected) environmental conditions are needed to reveal localized corrosion tendencies of the candidate materials, recent publications (Bandy and van Rooyen, 1985) indicate that nitrogen additions to stainless steel and nickel-based materials combat localized corrosion. These additions are also beneficial in increasing the resistance to sensitization, and they combat forms of corrosion associated with that phenomenon.

The experiments just described were conducted in environmental conditions that are not anticipated in the Yucca Mountain repository; in particular, a concomitant high radiation field and an aqueous environment are not expected. However, the susceptibility of 304L SS to cracking due to either an increase in the electrolyte concentration or to increase in the oxidizing potential of the environment does need to be determined. Thus, testing of 316L SS and Alloy 825 to discern comparative susceptibility to TGSCC under similar accelerating conditions is planned. Future work also involves use of precracked specimens to eliminate the apparently long incubation time needed to initiate cracking in well J-13 water or in the concentrated, boiled-down modification of this water. Some of these precracked specimens will be maintained at constant applied potentials either to accelerate or to retard crack growth. Plans for this work are given in Section 8.3.5.9.2.

7.4.2.6.5 Environmental considerations in localized corrosion initiation

As discussed in Section 7.4.1, the amount of water that could enter into the near-waste-package environment is expected to be small because of the low amounts of precipitation in the Nevada desert and the small fraction of the precipitation that actually percolates to the depth where the repository would be located. The heat generated from the radioactively decaying waste would vaporize incoming water when the package environment was above 97°C. During this period, it is possible that repeated evaporative processes (i.e., a refluxing operation) could leave behind salt deposits at or near the 97°C isotherm in the rock that would later be dissolved by water at a lower temperature as the isotherm moves toward the container. Surface tension effects may locally elevate the boiling point. The result would be that water with higher concentrations of ionic species could contact the metal container. Another scenario for concentrating ionic species directly on a container would be repeated dripping on the hot container surface from a fracture located above the container so that a highly ionic residue is left on the container to become wetted at some later time. Although even more unlikely than the preceding scenarios, it is possible that flooding of a portion of the repository could occur by a series of events such as a fracture admitting a surge of water to the waste package environment combined with plugging of the fractures below the container (a bathtub effect), with consequent boiling away of the water and concentration of the ionic salts. Repeated occurrences of this event would be required to build up a significant concentration of the important species affecting metallic corrosion. Although the events that can produce concentrating effects are the subject of study discussed in other parts of this site characterization plan, performing corrosion tests in these concentrated environments is important in selecting container materials.

One analysis that has been conducted in this regard is that of Morales (1985). This analysis considered that the "dripping water on the hot metal surface" scenario could not realistically occur in the repository, because the thermal output from the container would evaporate any liquid water at a distance well away from the container. With regard to the scenario of repeated evaporation of downward infiltrating water and eventual resaturation of this water when the boiling point isotherm has moved to the container surface, this analysis concluded that as a conservative maximum a solution with 20 times the salt concentration of well J-13 water could reach the containers as a burst, or the more gradual redissolution of the accumulated salt in the rock over a longer period of time could result in a slightly more concentrated salt solution than well J-13 water reaching the containers.

To project the numbers of containers that could eventually perforate by localized corrosion attack or TGSCC, models will be built around the rates of attack for different concentrations of salt in groundwater of the well J-13 type. It is recognized that eventual cooling of the repository will allow for hydration of the environment with potential for access of the aqueous environment to the container surface. However, this does not appear to be possible until several hundred yrs after the repository closure. A more precise statement of the time depends on many features of the waste package and repository design and the thermal projections based on these designs.

7.4.2.6.6. Summary of testing for pitting, crevice, and transgranular stress corrosion cracking

In summary, the candidate stainless steels appear to be sufficiently resistant to pitting, crevice attack, and transgranular stress corrosion cracking in the unmodified and irradiated well J-13 water and in steam generated from the water. Even if the highest measured nonstress assisted corrosion rate is added to the general corrosion rate, container service life far exceeds the maximum containment requirement. However, if concentration of the ionic species in the water could occur, these would possibly result in much more aggressive environmental conditions in which the candidate stainless steels would show quite different behavior. It appears that the combination of concentrated electrolyte and irradiation is the most severe environment so far tested and even 316L SS may pit and crevice corrode under these conditions. Scenarios that could result in such environmental conditions are unlikely; however, assessing container behavior under unanticipated processes and events is necessary to develop radionuclide source term estimates for low probability scenarios.

7.4.2.7 Phase stability and embrittlement

In this section, the stability of the austenitic structure over long periods of time is discussed. Possible phase transformations of concern include the formation of ferrite, martensite, and sigma phase in some of the candidate stainless steels. The concern is whether these phases could adversely affect long-term fracture toughness and result in a more brittle structure (and in some instances less corrosion resistant) than the parent structure. Some transformation products, such as martensite, are more susceptible to hydrogen embrittlement when exposed to environmental situations where atomic hydrogen is produced. Many of these phase instability concerns are closely related to microstructural features of the weld and heat-affected zones and, thus, the processes used for fabricating and welding the container material structure.

7.4.2.7.1 Phase stability

Many austenitic stainless steels have a tendency for transformation of some austenite to ferrite. The austenite-stabilizing alloying elements are balanced against the ferrite-stabilizing elements in alloy development. The presence of delta-ferrite in the weld microstructure to prevent hot cracking can be detrimental to corrosion resistance over long periods of time or with high temperatures. According to some researchers, carbides that may have grown due to low-temperature sensitization will most likely reside at austenite-ferrite boundaries (Duhaj et al., 1968; DaCasa et al., 1969). Whether retained delta-ferrite or subsequent transformation of austenite to alpha-ferrite would be harmful to the corrosion performance of a stainless steel nuclear waste container has not yet been determined. A further concern with retained ferrite is its possible transformation to hard, brittle sigma phase when held at moderately elevated temperatures. Although most studies show the nucleation time of sigma phase from delta-ferrite to be quite long, some sigma phase nuclei have been observed after exposure at 750°C for only about 5 min (Wegrzyn and Klimpel, 1981). As with the study of low temperature sensitization, the expected prolonged exposure of the container to temperatures in the 100 to 280°C range in the repository raises the question of whether a small amount of sigma phase would form at these relatively elevated temperatures for very long times (hundreds of years). A content of just 2 weight percent sigma phase reduces the fracture toughness (impact strength) of stainless steel by one-half. Fracture toughness is most important as a material property for the first 50 yr when retrievability of the waste package is a requirement; impact loads on the container during retrieval could conceivably rupture an embrittled material. Molybdenum additions (ferrite stabilizer) as in 316L stainless steel slightly increase the tendency toward sigma-phase formation (and also enhance formation of brittle chi and Laves phases). Thus, the beneficial qualities offered by the 316 stainless steel series in improving resistance to localized corrosion, and resistance to sensitization effects may be offset by a greater tendency to form brittle phases over long periods of time. These factors need to be addressed in future work. Also, cold deformation appears to speed up the kinetics of sigma-phase transformation. Although most of the container surface could be stress relieved, the area around the final closure weld would appear to be subject to retention of residual stress.

Residual cold work in the container could also favor transformation of the austenite to martensite with attendant loss in fracture toughness. As in sigma phase formation, this transformation would be expected to occur at a local level. Further, a martensite transformation would allow more carbon to become available for chromium carbide formation, and subsequent sensitization would be more likely to occur. Generally, the molybdenum-bearing stainless steels are more resistant to martensite transformations; therefore, in this situation, the 316 series would be more resistant than the 304 series. This transformation could be avoided if the cold-worked container were annealed.

7.4.2.7.2 Hydrogen embrittlement

Although the face-centered cubic austenitic structures are ordinarily regarded as highly resistant to hydrogen-embrittling effects, the long-term interaction with a hydrogen-producing environment must be considered a possible degradation mode. Martensite transformation from the austenite results in a phase that is much more susceptible to hydrogen embrittlement, and strain-induced martensite could form in a heavily cold-worked 304L material. As discussed earlier, a source of hydrogen is the radiolytic decomposition of water vapor (during the containment period, when the radiation field is high) or the electrochemical decomposition of water resulting from corrosion reactions on a container surface. The former source would only be possible if the environment becomes saturated while the radiation field is high, which is not likely for the majority of the packages as discussed previously.

Of the candidate materials, 304L SS is the most susceptible to martensite formation if the material is highly strained. A rough calculation of the temperature at the beginning of deformation of martensite transformation, at 30 percent strain for a typical 304L SS composition, indicates that martensite could start forming when the temperature drops to approximately 85°C. This calculation was based on the expression given in the review article by Novak (1977). Higher strains will elevate the temperature, and alloying additions will lower the temperature. This result suggests that hydrogen embrittlement of a martensitic structure would be of concern at a later period when the container surface had cooled to below this critical temperature and when an aqueous environment could access the container surface. It therefore is not of concern with respect to retrieval.

One potential problem area that has not yet been explored is hydrogen embrittlement in the duplex structure of weld metal for austenitic stainless steels. The ferrite phase has the potential for trapping hydrogen. In this instance, no martensite transformation is required as a prerequisite condition. Although hydrogen could enter the stainless steel from radiolytic decomposition of the water vapor, the relatively high temperatures would tend to mitigate against trapping effects (the hydrogen could diffuse out as readily as it diffuses in). Also, the oxidizing nature of the environment will tend to combine oxidizing radicals with atomic hydrogen as it is produced. One possible source of hydrogen produced by radiolytic decomposition of water vapor could develop inside the waste package container from waterlogged spent fuel rods. Because the spent fuel waste packages will be filled with an inert gas (e.g., argon) and zircaloy may serve as an oxygen getter, atomic hydrogen may have a longer lifetime in this type of environment. Thus, there exists the possibility that this hydrogen could enter and permeate into the metal container. However, these effects will need to be investigated to determine whether significant concentrations of hydrogen atoms are produced in the gas phase and whether they can permeate into the metallic structure. Studies to address this question are discussed in Section 8.3.5.9.

7.4.2.7.3 Welding considerations

The integrity of the weld region of the container and the propensity of weld-processing variables for producing microfissures in the weld require evaluation of the state-of-the-art techniques for nondestructive postweld inspection of the container.

In welding processes using filler materials, a certain amount of delta ferrite (e.g., 3 to 5 weight percent) is generally sought in the fusion zone to mitigate against hot cracking. For this reason, the higher chromium content AISI 308L SS is often specified for the filler material in welding AISI 304L SS. However, the intentional presence of delta ferrite in the weld microstructure can be detrimental to long-term corrosion resistance. Whether the presence of delta ferrite would adversely affect container performance in the anticipated repository environment needs to be determined.

Molybdenum additions (such as those made to AISI 316L SS) slightly increase the tendency to form brittle phases such as sigma. Thus, the beneficial qualities of AISI 316L SS (including improved resistance to localized corrosion and sensitization) may be somewhat offset by the greater tendency to form brittle phases over long periods of time. These factors need to be addressed in future work. Also, cold work appears to speed up the kinetics of sigma formation. Although most of the container could be stress relieved, the area around the final closure weld would appear to be subject to retention of residual stress.

Care must be taken in welding Alloy 825. This alloy is purely austenitic and lacks the formation of delta-ferrite in the weld zone; the advantage of delta ferrite is to soak up harmful impurities (such as sulfur) that cause hot cracking. This means that more careful control of the alloy chemistry (and filler material if a weld process using filler material is eventually selected) is required. The higher-nickel materials are sufficiently unlike the austenitic stainless steels that different welding parameters must be used to ensure sound welds.

7.4.2.7.4 Summary of work on phase instability and embrittlement

Relatively little work has addressed issues of phase instability for some of the candidate materials and problems of embrittlement that may be associated with the phase instability. The embrittlement may be purely mechanical due to the loss of fracture toughness by formation of sigma phase or other brittle intermediate phases, or the embrittlement may be due to hydrogen produced by the environment, the hydrogen favoring martensite, and possibly ferrite and causing embrittlement in those phases. If the embrittled phase is continuous, then the structure suffers from a severe loss of fracture toughness. Further study of the long-term physical metallurgical stability of the different grades of austenitic stainless steels in repository-relevant thermal and chemical environments is needed. The high-nickel Alloy 825 is apparently immune to transformation of austenite, but the possibility (probably remote) of detrimental hydrogen embrittlement would need to be addressed. Phase stability and embrittlement is addressed in more detail in Section 8.3.5.9.

7.4.2.8 <u>Projections of containment lifetimes (austenitic materials)</u>

The purpose of this section is to project a lifetime for the container based on corrosion data obtained to date. The performance assessment of the container is a composite of the individual assessments of each potential mode of degradation. These individual assessments are generally not additive; environmental or metallurgical conditions will determine which mode is operative (and exclusive to the others), and, as these conditions change, different degradation modes may become operative.

7.4.2.8.1 Time periods and relevance of degradation modes

As discussed in Section 7.4.2.1, the function of the metal container depends on the time period (e.g., the period during which the repository is in operation) and the various time periods following permanent closure of the repository.

Thus far, oxidation is the only experimentally measurable degradation mode for the candidate stainless steels under anticipated environmental conditions that is expected to be significant during the minimum containment period. General aqueous corrosion will occur when the container surfaces become wet; as discussed in Section 7.4.2.4. As discussed in Glassley (1986), this condition is not expected during the minimum containment period, but it may develop around some of the containers during the succeeding post-closure period. The time for hydration of the rock to occur will depend strongly on the configuration of containers in the repository and on the average thermal loading per container and the areal power density maintained for the configuration. Typical corrosion rates in unsaturated steam are 0.07 micrometers per yr, and these increase to 0.10 micrometers per yr in saturated steam and to 0.15 micrometers per yr in water immersion. Even with full water immersion during the entire maximum containment millenium, the projected amount of metal wastage would predict complete containment for container lifetimes (assuming 10,000 micrometers thickness) in excess of 10,000 yr. If the high value of 0.51 micrometers per yr (measured on 304L SS in irradiated well J-13 water at 150°C) were used, and a high crevice corrosion rate of 0.5 micrometers per yr were coupled to this, the projected wastage would be only one-tenth of the container thickness for the 1,000 yr containment period. Sustained localized corrosion rates of several micrometers per yr would be needed before this form of corrosion would seriously limit the containment objectives. Rates of this magnitude would not be expected to occur in unsaturated steam. Because sustained immersion of substantial areas of the containers appears to be highly improbable, higher corrosion rates are not expected.

One of the nonuniform modes of corrosion that does appear to offer some limitations on containment life is development of a sensitized microstructure during the containment period and subsequent wetting of the affected area to produce intergranular stress corrosion cracking. The time-to-sensitization decreases substantially with an increase in the peak temperature of the container, a heavily cold-worked 304L SS (with 0.028 percent carbon) is predicted to sensitize in as little as 120 yr at isothermal conditions of 250°C, and in 4,000 yr at 200°C. AISI 304 stainless steel did exhibit intergranular cracking in irradiated well J-13 water and saturated vapor; the question remains as to whether the lower carbon content in the L grades can confer immunity or at least a high resistance to the phenomenon. Proper selection of the thermomechanical processing of the containers can confine this phenomenon to the area around the final closure welds; proper selection of the alloy, its physical microstructure, and control of its microconstituents can further reduce the susceptibility, until a situation compatible with substantially complete containment is achieved with a high level of assurance.

The metastability of austenite in the 300 series candidate austenitic materials may be a second issue of concern during the containment period since sigma (or another brittle) phase might form from either the austenite or from retained delta ferrite in the duplex weld material. A further embrittlement problem during the containment period might be hydrogen embrittlement at the austenite-ferrite interface or in the ferrite phase, if sufficient hydrogen forms from radiolysis of the environment and permeates the metallic structure during this time period.

The next most likely degradation modes to consider as limiting containment would be transgranular stress corrosion cracking (TGSCC), pitting, and crevice corrosion. These modes would only occur if some unusual event happened during the minimum containment period when the container was above the boiling point of the vadose water. Their occurrence becomes considerably more probable when aqueous conditions arise by eventual water infiltration to the cooling container surface in the later periods. These modes would be promoted by lengthy immersion time (to build up and sequester electrolyte concentration in the local geometry) or influx of a more concentrated electrolyte than the reference well J-13 water environment, neither of which are expected conditions for a Yucca Mountain repository.

Alloy 825 is considered to be the most resistant of the candidate materials to virtually every form of corrosion that might occur in a geological repository in tuff. The alloy is low in carbon and titanium- stabilized to render it highly resistant to sensitization. The austenitic structure is stable at all temperatures. The alloy is resistant to chloride-induced pitting, crevice attack, and TGSCC (except in very aggressive solutions that are low in pH, high in chloride, and strongly oxidizing (such as 10 percent $FeCl_3$ solutions)). Although the metallurgical stability and the chemical stability in the Yucca Mountain environment all appear to be quite favorable to Alloy 825 as a container material, some developmental work would be needed to ensure its good weldability, should this material be selected for the advanced designs.

7.4.2.8.2 Long-term performance projections and selection of container materials for advanced designs

Understanding the governing mechanisms for the different corrosion processes is the basis for any long-range predictions of performance. To this end, estimates of the changes in the corrosion potential for the different candidate stainless steels are modeled to result from changes in environmental conditions. Models of mechanisms for localized and stress-assisted forms of corrosion are based on the corrosion potential exceeding a characteristic critical potential (for each form of corrosion) that corresponds to the threshold causative environmental conditions for that particular corrosion mode. When the corrosion potential is at or above this critical potential, then the particular form of localized or stress corrosion is operative. If the corrosion potential lies below the critical potential, then general corrosion is the operating mode. In Section 7.4.5, the performance assessment of the waste package is discussed in terms of individual performance assessment models for the different waste package components. The overall "corrosion model" for the waste package container is based on which particular corrosion model (general, pitting, crevice, stress, etc.) operates for a given set of environmental conditions. During the entire

10,000-yr period of concern for characterization of the corrosion performance of container materials, different individual corrosion models can operate according to the particular environmental conditions, and the overall corrosion model will account for the situations where individual models are operative. Performance predictions ultimately will involve a mapping activity between the phenomenology observed during the testing activities and an understanding of the mechanisms that govern the phenomenology.

The research and development effort on metal barriers is driven by the need to select a container material that will provide substantially complete containment for the requisite time period. Thus, a large effort is placed on assessing the chemical stability of the different candidate metals in the projected thermal and geochemical environment of the repository, and failure of these candidate materials is assumed to occur by corrosion. A good deal of effort has been placed on the particular corrosion effects in and around the welds, and some of the special limitations imposed by making the final closure weld by remote processes.

When sufficient information has been gathered on the six candidate materials, an assessment will be made to reduce the number of candidate materials to one or two. This assessment will be made on the basis of weighted selection criteria. Additional experimental work will be done to provide model verification for the appropriate degradation modes. This work is discussed in more detail in Section 8.3.5.9.

7.4.2.9 Alternative alloy system

Copper and certain of its alloys are under consideration as a completely alternative alloy system. Copper and its alloys have many of the same advantages as the austenitic stainless steels and also offer certain unique advantages. Copper and its alloys are, like the stainless steels, ductile materials that are readily fabricable by a variety of processes. Copper, alone among the engineering materials, can thermodynamically coexist with aqueous environments under certain conditions. Thermodynamic stability may be an important argument in demonstrating that the selected waste package container material can attain the long-term performance objective. The existence of copper artifacts from earlier civilizations, as well as native copper, indicates that copper can survive for at least a thousand yrs under some environmental conditions. However, under particularly oxidizing aqueous conditions, copper readily corrodes because the various copper oxides or cations are the thermodynamically stable species under these conditions. The aerated steam that is expected to dominate the waste package environment is moderately oxidizing, as would be the vadose water. Irradiation of these environments would likely make them more oxidizing. As with the stainless steels, the question is whether copper or a copper-based alloy would passivate or corrode actively for a given environmental condition. If the metal were attacked, would the pattern of attack be substantially uniform or highly localized? The review of Nuttall and Urbanic (1981) considered the different possible degradation modes of copper in repository environmental settings but concluded that experimental work is needed in site-specific environments.

7.4.2.9.1 Candidate materials and test plan

The NNWSI Project plan for testing copper essentially parallels that for the austenitic stainless steels. The fundamental thesis for testing and selecting materials is similar, namely, certain alloys are more resistant to specific forms of corrosion and their use would be pursued if testing revealed that the less alloyed (and less costly) material were susceptible. The candidate copper-based materials, their nominal compositions, and their mechanical properties are given in Section 7.3.2.2. The materials are CDA 102 (high-conductivity copper), CDA 613 (aluminum bronze), and CDA 715 (70-30 copper-nickel). CDA 102 is the reference material in this alternative alloy system. Pure copper is a relatively weak material when fully annealed. A thicker container wall than that specified in the reference waste package designs (1 cm) is expected to be needed for a pure copper container. The copper alloys are inherently stronger materials than pure copper, and their use would probably not require much, if any, change in the dimensions of the reference designs. If copper or a copper alloy were selected for the container, required fabrication techniques may be quite different from those contemplated for stainless steel.

The test plan, candidate materials, and expected degradation modes for copper were discussed by McCright (1985). To expedite the copper testing plan, NNWSI Project staff members consulted with representatives from the copper industry on the suitability of different alloys for use in nuclear waste containment. The Copper Development Association Inc. and International Copper Research Association, Inc. recommended the alloys being considered.

7.4.2.9.2 Possible degradation modes for candidate copper and copper-based alloy materials

The types of corrosion that may develop in the tuff repository are uniform corrosion, pitting corrosion, crevice corrosion, intergranular corrosion, selective leaching, and stress corrosion cracking. Uniform corrosion is considered the most likely degradation mode. Pitting corrosion is also possible, especially because of the irradiated environment.

A review of the literature was discussed by McCright (1985). The review summarized the known oxidation and corrosion behavior of the three candidate materials in air, steam, and a variety of aqueous solutions between ambient and 300°C. The experimental program based on that review consisted of several parts: (1) electrochemical studies, (2) long-term phenomenological experiments under gamma irradiation, and (3) modeling studies. The experimental program is discussed in detail in the work of Acton and McCright (1986).

The experimental test plan for copper-based materials is based on an initial survey of the different kinds of corrosion that may occur in both the expected and the possible but not expected environmental conditions in the tuff repository. Many of the same kinds of tests, as discussed in Sections 7.4.2.5 and 7.4.2.6 are under way for the copper-based materials. The NNWSI Project has completed a 2-yr feasibility study in response to a congressional directive on whether copper or one of its alloys could be used under the environmental conditions anticipated for a tuff repository. This study was completed in September 1986. At this time, copper and its alloys have not been shown to be infeasible, based on the limited experimental work done in repository-relevant conditions. It is expected that the NNWSI Project will further test any selected material, as discussed in Section 8.3.5.

7.4.2.10 Borehole liner materials

This section is concerned with the performance of a borehole liner that can be designed to achieve approximately 100-yr service life after emplacement of the waste packages for possible retrieval of the waste packages, as required in 10 CFR 60.111. The borehole liners are discussed in Chapter 6 (repository design). In the current conceptual design, a partial liner is planned for use in the vertical-emplacement boreholes. A liner is planned for use in the entire length of the horizontal emplacement boreholes. One concern is the effect that the presence of the borehole liner could have on the integrity of the waste container. This effect could be favorable (assuming that the liner would sacrificially corrode and protect the container or resist loads that might otherwise be imposed on the waste container) or unfavorable (soluble corrosion products might migrate to the container surface and intensify the attack on this component). The effect of liner corrosion products on radionuclide transport properties is of concern. Although these effects are not included in the designed functions of the liner, its presence in the waste package environment could influence the ultimate performance of the container.

As a way of reducing the number of possible interactions among the waste package and repository components, the current NNWSI Project plans are to use a borehole liner made from a material of the same alloy family as the waste container. In this consideration, the liner material would not, however, need to be fabricated from exactly the same material as the container. This approach is discussed further in Sections 8.3.5.9 and 8.3.5.10 under Issues 1.4 and 1.5 and in Section 8.3.4.2 under Issue 1.10.

7.4.3 WASTE FORM PERFORMANCE RESEARCH AND TESTING

The waste forms being studied by the NNWSI Project to establish their expected performance in a repository sited at Yucca Mountain are of two types. The first is spent fuel from commercial power reactors. This waste form is expected to represent the majority of the waste to be disposed of in the first high-level waste repository, based on present inventories of waste and the absence of any commercial waste reprocessing plant coming on line in the foreseeable future.

The second waste type is the product of solidification of high-level waste solutions produced during the reprocessing of spent fuel. Reprocessing wastes exist in four locations in the United States (Manaktala, 1982). Three of these locations are defense facilities located at Savannah River, South Carolina; at Hanford, Washington; and at Idaho Falls, Idaho. The Savannah

River Plant has chosen a final waste form (Baxter, 1983) and is in the process of constructing the Defense Waste Processing Facility (DWPF) to convert its waste into borosilicate glass. Final choices of waste form have not yet been made for the other two facilities. The fourth site where liquid waste is located is at West Valley, New York. The solidification of this waste is controlled by the West Valley (WV) Demonstration Project. The exact formulation of the final waste form for West Valley waste has not yet been chosen, but it will be a borosilicate glass similar in composition to the DWPF glass (Oversby, 1984b; Eisenstatt, 1986).

To determine the anticipated performance of these waste forms in a repository at Yucca Mountain, an assessment must be made of the physical and chemical environment of the waste form after disposal (Section 7.4.1), the expected performance of the waste container (Section 7.4.2), and the emplacement configuration. For a repository in the unsaturated zone, the emplacement configuration may govern the geometry of the waste form-water contact. The detailed discussions of waste performance in Sections 7.4.3.1 and 7.4.3.2 are predicated upon these assessments. Most radionuclides can only be removed from the engineered barrier system through the action of water. (The principal exception is carbon-14, which could also leave a breached container in gaseous form (Van Konynenburg et al., 1986)). Even though extremely little water is expected to be available to contact the waste, waste form testing is done using various water volumes and contact mechanisms to understand how radionuclide release could occur. The results of these studies can be applied to performance assessment by calculating the fractional number of containers in which the waste will be exposed to water under those conditions.

For the purposes of designing waste form tests and calculating expected releases from the waste form, the reference container material and designs have been used. It has been assumed that the geometry of the container would be maintained throughout the 10,000-yr isolation period. The predicted uniform corrosion rate, based on presently available data, is sufficiently low that there should be a substantial thickness of stainless steel container remaining in essentially the original emplacement geometry for at least 10,000 yr. It is further assumed that some fraction of the containers would have developed perforations in the heat-affected zones around welded regions of sufficient size to allow water, if present, to flow freely through the perforations. The most likely failure mechanism for the metal container under Yucca Mountain tuff repository conditions after the containment period is stress corrosion cracking (SCC), probably of a welded or heat-affected zone. Failure by pitting or crevice corrosion would also produce localized penetrations in the container. Two emplacement configurations are under consideration by the NNWSI Project: (1) vertical, which is the reference mode, and (2) horizontal, which is an alternative mode. Waste form testing has been designed to cover the possible types of waste form-water contact that might result from both of these emplacement geometries with the waste held in a perforated, but essentially intact, container.

An essentially intact container throughout the isolation period provides a means for water to interact with the waste form for extended periods of time. This is a more realistic model for the unsaturated zone than one that assumes that the container is absent following breach. Absence of the container in the unsaturated zone would mean that water could drain away from the waste form; thus, contact times would be short and degradation of the waste form would be minimized under these conditions compared with the same amount of water being held in contact with the waste for a longer period of time. Depending on the location and number of container perforations and on the water flow regime in the borehole region, water contact with the waste could vary from none to water dripping on the waste and immediately running off to water accumulating in the container up to the level of the perforation and interacting with the waste for long periods of time.

Spent fuel canisters will contain either assemblies or consolidated rods from assemblies. In either instance, there would be a significant amount of free space distributed rather uniformly throughout the container. For a vertically oriented container with perforations in or around the top circumferential weld, water could potentially accumulate inside the container up to the level of the perforations. Testing related to this geometry forms the basis of the NNWSI Project spent fuel testing to date: spent fuel in varying conditions of degradation has been tested under semistatic conditions where the fuel is completely covered with water. For a container with perforations at different levels, the geometry for fuel dissolution and degradation is unchanged, but the volume of water in the container would be limited to that which could accumulate in the region up to the lowermost perforation. If the perforations occurred only at the bottom of the container, liquid water would probably not be able to enter the container; however, water vapor might enter the container and condense, providing a means for limited radionuclide release.

Spent fuel containers emplaced horizontally would provide similar degradation geometries. The only configuration that would provide for a significantly different waste form-water interaction mode would be that present if there were perforations in both the upper and the lower sections (as emplaced) of the container. In this instance, water could drip through the upper perforations, flow past various fuel rods, and flow out the lower perforations in the container. The unsaturated test method used for glass waste forms is being modified to use with spent fuel to test the results of water interaction under these conditions.

Containers of glass waste would have a decidedly different waste form-water interaction geometry because of the stainless steel pour canister inside the container and the presence of significant open space only at the top of the container and pour canister. Currently, the simplifying assumption is made that the inner canister provides no restriction on water access to the waste and no control on the geometry of contact. When more data are available from the metals testing program on the effects of residual stress and sensitization on canister failure rates and mechanisms, water interaction scenarios can be improved to include the effects of the inner canister on control of release. This will not change the available water contact mechanisms; it will only affect the proportion of waste packages in which there is access for water to the waste.

The location of perforations in the container would have a large effect on the geometry of water contact with glass waste forms. If perforations occurred in the container at any location below the top surface of the waste form, the ability of water to contact the waste form would be severely limited. Only perforations occurring above the glass surface would allow prolonged contact of significant amounts of water with the glass. In this instance, the headspace between the top surface of the glass and the perforation could eventually fill with water to the level of the perforation. This would result in a period of glass-water interaction without any release from the container until the water level reaches the perforated region. After this occurs, the water could potentially flow through the container at a rate up to the water flux passing the outside of the container. This scenario would provide the maximum water flux and contact time and minimum ratio of surface area of glass to volume of liquid. Glass degradation occurring under these conditions would provide an upper limit on the release rate of radionuclides from the glass.

For horizontally emplaced glass containers with perforations occurring in the upper and lower (as emplaced) regions of the top container weld, the glass-water interaction would occur as a combination of dripping water and vapor phase alteration. This type of dripping-water contact could occur in any situation where the container is perforated at two levels, but the extent of possible water flow inside the pour canister along the glass-steel interface or long cracks in the glass has not yet been determined. The NNWSI Project Unsaturated Test Method (Bates and Gerding, 1985) was developed to address these scenarios where water can enter the container, flow over the glass while reacting with it, and then leave the container.

These water contact scenarios form the basis of the NNWSI Project waste-form testing program. Significant emphasis is placed on testing corresponding to a perforated container filling with water because this would provide for the greatest waste-form degradation. For purposes of estimating release rates, it will initially be assumed that the container does not impede access of water to the waste-form except by the location of the perforations. This assumption places the burden of achieving the performance objective on control or release solely on the waste form when long time periods are considered, if one assumes that all containers will eventually develop penetrations and will have equal water fluxes. Short-term control may be provided by the containers since they are not expected to all fail at once; however, the rate of perforation initiation and growth may be difficult to establish with a high degree of confidence. Therefore, the release rate from perforated containers will be combined with the anticipated water flux in the borehole to demonstrate that the performance objective for control of radionuclide release has been met.

7.4.3.1 Spent fuel performance research and testing

The spent fuel presently in storage consists of boiling water reactor fuel discharged since 1969 and pressurized water reactor fuel discharged since 1970. The characteristics of the fuel have been summarized by Oversby (1984b) to provide a reference spent fuel composition for the NNWSI Project testing program. Before use, the fuel rods consist of a stack of uranium dioxide (UO_2) pellets encapsulated in a metal tube, which is called the cladding. Most fuel is clad in Zircaloy; however, four reactors use fuel clad in stainless steel.

The fuel rods are arranged into assemblies for insertion into the reactor. The detailed geometry of the rods and of the assemblies depends on the reactor in which they have been used. Woodley (1983) has described the variation in rod and assembly characteristics and also the evolution in rod design to improve performance of the fuel in the reactor. He also has summarized the available literature on the physical and chemical characteristics of spent fuel that are of significance to waste package design. The discussion that follows is based on his summary unless otherwise noted.

During irradiation, the physical and chemical nature of the fuel is changed. The chemical changes are caused by the fission of uranium, the production of higher atomic-number actinides by neutron capture reactions, and the production of activation products by neutron capture. The activation products are located in the fuel itself, in the cladding, in deposits on the cladding, and in the assembly parts. Since some of the activation products are in the cladding and assembly parts (such as carbon-14, a radionuclide whose release rate must be controlled to within the NRC and EPA specified limits), the waste form for spent fuel is the entire assembly (if disposed of intact) or the clad fuel rods and disassembled nonfuel hardware (if disposed of as consolidated rods).

Oversby (1986) reviewed the constraints on the release rates specified by the NRC and EPA regulations and established a list of radionuclides for which data on dissolution rates are necessary. The simplest case, which maximizes the number of such nuclides, results in the identification of 17 chemical elements for which dissolution data are needed for performance assessment. Of these, americium and plutonium are the most important; other actinides, carbon, and nickel are also important.

The abundance and spatial distribution of new chemical species formed during irradiation depends on the operating conditions to which the fuel rods are exposed. The abundance of fission products depends principally on the burnup of the fuel, that is, the amount of energy produced per unit weight of initial uranium. Burnup is usually expressed in megawatt-days per metric tons of uranium (MWd/MTU). The abundance of activation products depends on the composition of the starting materials, particularly with respect to trace impurities such as nitrogen in the fuel and cladding, and on the neutron fluence and spectrum. The transuranic actinide content will depend on burnup and on the neutron fluence and spectral conditions in the fuel (Roddy et al., 1985).

The spatial distribution of fission products in spent fuel depends on the chemical interactions between the matrix and the fission products and on the reactor operating conditions, which can strongly affect the ability of fission products to segregate from the matrix. Gaseous fission products (krypton and xenon) accumulate in the matrix as small gas bubbles (Baker, 1977; Woodley, 1983). If temperatures become high enough, the gases can diffuse to grain boundaries or to the gap between the pellets and the cladding. The degree of migration that has occurred for a given spent fuel rod can be determined by sampling the gas contained in the pellet-cladding gap. This is referred to as fission gas release and is usually expressed as a percentage of the inventory calculated to be present in the rod.

During reactor operation, the centerline temperature of the fuel rods can be between 800 and 1,600°C. Some of the other fission products that are not gaseous at lower temperatures will be volatile at the higher temperatures. Elements that have been identified as mobile under reactor operating temperatures include cesium and iodine (Johnson et al., 1983a). These elements migrate to grain boundaries and from there to the pellet-cladding gap. Other fission products are metallic under the oxidation-reduction conditions inside the fuel rod and form metallic segregations. These may be located within the grains or at the grain boundaries. The degree of fission gas migration was large in early designs that allowed the fuel to reach very high temperatures. Fission gas release of 15 to 25 percent has been measured for spent fuel from several reactors. Improved fuel rod and assembly design has resulted in lower operating temperatures for the fuel and much lower fission gas release. Gas release for modern fuels is generally less than one percent.

Migration of fission products can occur by volume and grain boundary diffusion at operating temperatures if they are sufficiently high. However, measurements of internal pressure in an instrumented assembly indicate that most of the migration occurs during reactor power changes, when the rapid temperature changes stress the fuel and induce cracks, thereby allowing enhanced migration of volatile species. Thus, the number of shutdowns and the rates of power changes during operation can influence the gas release for a fuel rod.

The temperature in the central portion of a fuel rod during irradiation may be high enough for grain growth to occur by a mechanism analogous to sintering. This phenomenon in fuel is referred to as restructuring and is accompanied by segregation of insoluble fission products such as metals into secondary phases. Operating temperature and starting grain size are both important factors determining the degree of restructuring. Extensive restructuring does not usually occur in LWR fuels because most fuel operates with a centerline temperature less than 1,300°C. Extensive restructuring is observed above 1,500°C and is common in liquid metal fast-breeder reactor (LMFBR) fuels that have peak center-line temperatures of approximately 2,000°C; fission product elements that may show larger early releases during the leaching of highly restructured fuels include cesium, iodine, selenium, strontium, and technetium (Wilson, 1985a; Olander, 1976).

Physical examination of fuel after irradiation shows that the pellets are extensively cracked. If fuel is removed from its cladding, a wide range of particle sizes is obtained, including some very fine fragments (Katayama et al., 1980). Zircaloy cladding develops an oxidized layer on the outer surface during reactor operation. Some materials precipitated from the water during operation may be present on the cladding surface; these deposits are called "crud." Some fuel rods may have small defects such as pinholes or cracks in the cladding. Approximately 1 percent of the rods developed defects in the cladding during reactor operation in the United States from 1969 to 1972. Improvements in fuel rod design and in reactor operating procedures have reduced the rod failure rate for modern fuels to less than 0.1 percent (Woodley, 1983).

The information on spent fuel characteristics just described and the data obtained by the Swedish (KBS) and Canadian (AECL) workers on the release of radionuclides from spent fuel in ground waters of composition similar to well J-13 water were used as the basis for planning the NNWSI Project spent fuel testing program. The results from both the KBS and AECL programs showed that the release rate of some elements from spent fuel pellets in contact with water would be very rapid during the first few months of contact with water (Ekland and Forsyth, 1978; Johnson et al., 1982). The initial release of cesium from spent fuel correlates with gas release, which itself is a function of linear heat generation rate in the fuel. Even for fuel with low power rating and low gas release (Johnson et al., 1983a), several tenths of a percent of the cesium inventory was released within a few days of leaching with air-saturated distilled water.

Other fission products that migrate to the pellet-cladding gap or accumulate on grain boundaries within the fuel, such as iodine-129, might also be expected to show a similar initial release pulse. The release amounts, the rapid nature of the release, and the high solubility of the elements under expected NNWSI Project disposal conditions led to a consideration of the role of the cladding material in providing control of release of radionuclides from the spent fuel pellet and gap inventories. The review by Woodley (1983) indicated that only a small fraction of spent fuel rods delivered to the repository would have defects in their cladding. If the intact condition of the cladding were to continue throughout the disposal period of concern with respect to control of release rates, then the rather large initial cesium release could be controlled easily to a maximum population-averaged release rate of less than 1 part in 100,000 per year. A 1 percent per year release rate of cesium from breached fuel rods could be tolerated provided that no more than 0.1 percent of rods contained breaches. The rapid release fraction of cesium is exhausted after a few months (Johnson et al., 1982; Forsyth et al., 1985). Thus, the cladding could provide a control on cesium release even if it were to develop defects after disposal, as long as the rate of defect generation is low and cladding failure does not occur nearly simultaneously in all rods.

Based on these considerations, a three-part testing program for spent fuel was established. The heart of the program is a series of fuel dissolution tests that examine the rate of release of radionuclides from bare spent fuel, fuel segments with intentionally defected cladding, and segments with intact cladding. Series 1 tests (Wilson, 1985a,b) used deionized water and ambient hot cell temperature (≈25°C); series 2 tests used well J-13 water and ambient hot cell temperature. The test vessel and experimental configuration of the series 1 and 2 tests are shown in Figure 7-19. Results of the series 1 and 2 tests are given in the reports by Wilson (1985a,b, 1987). Series 3 tests used well J-13 water and sealed stainless steel vessels at 85°C. Details of the test plan are described in Wilson (1986). At the time of this writing (August 1987), the series 3 tests are still in progress and published results are not yet available.

The second area of testing related to spent fuel is an experimental study of the rate of oxidation of spent fuel under repository disposal conditions. When uranium oxidizes, there is a large volume increase at the stage when uranium oxide (U_3O_8) is formed. This has been the cause of large-scale rupture of cladding that originally contained only small defects. The available data on oxidation rates of spent fuel were obtained at higher temperatures than those relevant for long-term disposal conditions. Data are needed

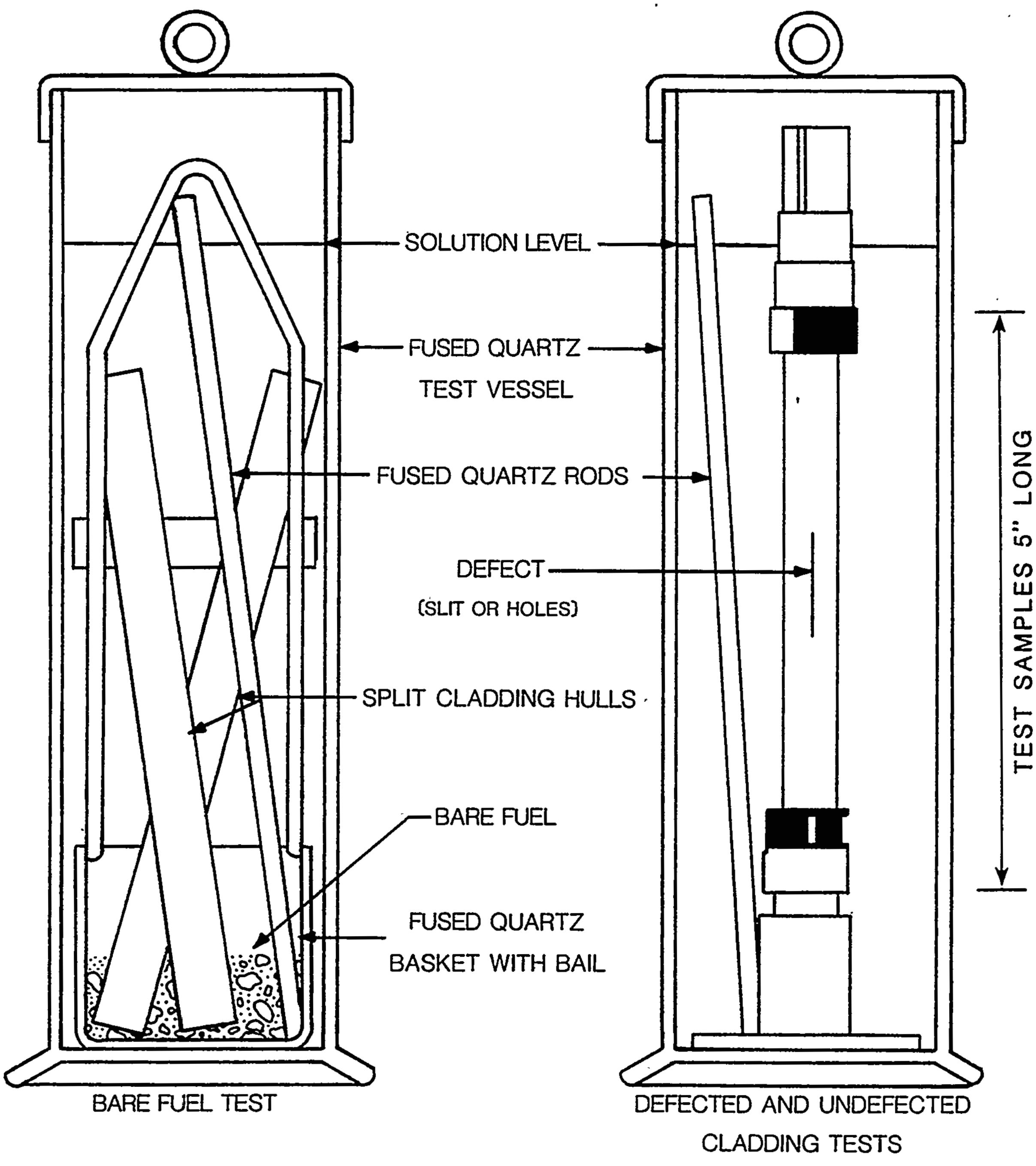


Figure 7-19. Test vessel and experimental configuration for spent fuel dissolution experiments at ≈ 25°C. Modified from Wilson (1985a,b).

at lower temperatures to determine whether the rates extrapolated from the high temperature data are valid at low temperatures and whether the oxidation mechanism is the same at the lower temperatures. Einziger and Woodley (1985a) have evaluated the potential for oxidation of spent fuel to occur under NNWSI Project disposal conditions. Their evaluation has led to development of a testing program (Einziger, 1985) that combines thermogravimetric analysis (TGA) studies for short periods (Einziger and Woodley, 1985b; 1986) with longer conventional oven-soak oxidation measurements (Einziger, 1986). The TGA studies allow continuous monitoring of weight changes, thereby providing detailed information about the oxidation mechanism and rate. The oven oxidation studies complement the TGA work by allowing longer experimental runs with multiple samples that are larger than those used in the TGA apparatus.

The rate of corrosion and degradation of Zircaloy under disposal conditions will have a profound influence on the ability of the cladding to control the release rate of nuclides from the interior of the fuel rods. An examination of the available information on corrosion of Zircaloy suggested that cladding would not develop a large number of defects under NNWSI Project disposal conditions (Rothman, 1984). A few areas were identified where further work on Zircaloy corrosion is needed to determine the expected rate of defect generation under repository conditions. Accordingly, a program of corrosion testing for irradiated Zircaloy cladding was initiated. At present, scoping experiments and tests are being conducted on two potential modes of cladding failure: electrochemical corrosion and stress corrosion cracking (SCC). An overview of the planned experiments is given in the report by Smith (1985), and the criteria used in the choice of samples for these experiments is discussed in Smith (1984b). Test plans for the initial electrochemical corrosion scoping experiments at 90°C and one atmosphere, and additional tests at 170°C and 0.83 MPa (120 psia) water are given in Smith (1984a, 1986a). Preliminary observations from the 90°C corrosion experiments are given in Smith and Oversby (1985). The test plan for the initial scoping experiments on SCC is presented in Smith (1986b).

In the discussion that follows, the work done on spent fuel dissolution by the NNWSI Project is described and the results are compared with those in the published literature. The oxidation of spent fuel both in solution and in air is discussed. A brief discussion of Zircaloy corrosion follows. The section closes with a description of the release rate model being developed for spent fuel.

7.4.3.1.1 Spent fuel dissolution studies

The purpose of the NNWSI Project testing is to determine the fraction of the sample inventory mobilized under test conditions and the physical nature of the mobilized components. That information will then be related to expected release rates at the end of the containment period. These release rates are referenced to the inventory present at 1,000 yr after repository closure. Table 7-13 gives the calculated inventory of radionuclides in PWR spent fuel at 1,000 yr for all isotopes with activity greater than or equal

Table 7-13. Radionuclide inventories at 1,000 years postclosure for pressurized water reactor spent fuel assemblies[a,b]

Radionuclide[c]	Percentage of total 1000-yr activity	Cumulative percentage
Am-241	51.84	51.84
Am-243	1.75[d]	53.59
Pu-240	26.87	80.46
Pu-239	17.37	97.83
Pu-242	0.1	97.93
Pu-238	0.06	97.99
Tc-99	0.77	98.76
Ni-59	0.252	99.01
Ni-63	0.021	99.03
Zr-93	0.181	99.21
Nb-94	0.074	99.28
C-14	0.076[e]	99.36
U-234	0.113	99.47
U-238	0.018	99.49
U-236	0.015	99.64
Np-237	0.058	99.70
Sn-126	0.045	99.74
Se-79	0.023	99.76
Cs-135	0.022	99.78
Sm-151	0.013	99.79
Pd-107	0.006	99.80
I-129	0.0018	99.80

[a]Source: Wilson (1987)
[b]Based on ORIGEN data reported in Alexander et al. (1977) pressurized water reactor assemblies with 33,000 MWd/MTU burnup.
[c]Radionuclides with 1,000-yr activity less than iodine-129 or half-life less than 1 yr omitted.
[d]Includes activity of neptunium-239 daughter products.
[e]Carbon-14 activity may vary considerably depending on as-fabricated nitrogen impurities.

to that of iodine-129. The nickel activity is mainly associated with stainless steel and Inconel assembly parts that were not part of the NNWSI Project test apparatus.

The spent fuel samples used in the NNWSI Project testing program have come from two PWR reactors, Turkey Point Unit 3 and H. B. Robinson Unit 2. Detailed characterization data are available for sibling rods from the same assemblies that these materials were obtained from in the cases of H. B. Robinson fuel (Barner, 1984) and from the same rods in the case of Turkey Point (Davis and Pasupathi, 1981). The Swedish testing program has used fuel sections from the Oskarshamn I BWR (Forsyth et al., 1984). The Canadian testing program uses CANDU natural uranium spent fuel (Johnson et al., 1982). In the CANDU reactor, where bundles of single enrichment natural UO_2 are irradiated in an array of process tubes extending through the core, greater variations in fuel rod linear power and fuel operating temperatures occur than in LWR reactors, which use variable enrichments and have more uniform neutron flux across the core. As a result, fission gas release, fuel restructuring and segregation of fission products from the oxide fuel matrix generally occur to a greater extent in CANDU fuels than in LWR fuels. A summary of the characteristics of the spent fuel samples used in the three testing programs is given in Table 7-14.

The three testing programs also use different ground-water compositions for the leaching solutions. Although the compositions are similar, differences in some components might affect fuel leaching. The Canadians have used a water composition, referred to by them as KBS ground water, which is higher in dissolved carbon species than that reported by Forsyth et al. (1984). To distinguish between the two, the Swedish composition will be designated KBS and the Canadian composition will be designated AECL-KBS. The Canadians also use a composition called "granite groundwater," which will be referred to as AECL-GR. The compositions of these three waters and the possible effects on spent fuel dissolution of solution composition differences are discussed in Oversby and Shaw (1986). The composition of well J-13 water is discussed in Section 7.4.1.

Laboratory testing cannot be done under repository conditions because the water-flow rates required are too low to obtain measurable results in the time period available. To overcome this experimental limitation, dissolution testing uses a higher ratio of water to fuel than is expected under repository conditions. The spent fuel dissolution testing procedure is a semi-static one in which samples of fluid are periodically taken for analysis. The fluid volume is then returned to the starting volume by adding sufficient fresh leachant to make up for the sample taken and for any evaporative losses. In this respect, the test resembles a slow flow-through test. Each test series consists of several cycles. This is done so that the size of the rapidly released gap and grain boundary inventory of fission products can be assessed. These nuclides are released very quickly in the first cycles and the levels attained in subsequent cycles show a progressive decrease. At the end of each cycle, the spent fuel specimens are rinsed and transferred to clean leaching vessels and the test restarted for the next cycle using fresh leachant. After the remaining leach solution is removed at the end of each cycle, the used leaching vessels are rinsed with leachant and stripped with 8 M nitric acid to remove any solid, precipitated, or adsorbed material. Series 1 and 2 tests were conducted in loosely covered, fused quartz vessels

Table 7-14. Characteristics of spent fuel samples used in release rate testing[a]

Characteristic	NNWSI Project: Turkey Point unit 3	NNWSI Project: H. B. Robinson unit 2	Sweden (Oskar-shamn I)	Canada (CANDU)
Fuel type	PWR 15 x 15	PWR 15 x 15	BWR	BWR Natural U
Cladding	Zircaloy-4	Zircaloy-4	Zircaloy	Zircaloy-4
Sample weight UO_2 (g)	43	80	16	72
Estimated burnup (MWd/kgU)	27	31	42	7.9
Initial enrichment (wt%)	2.559	2.55	--[b]	0.71
Fission gas release (%)	0.3	0.2	0.7	1 to 9
Peak linear heat generation rate (kW/m)	32.7	32.7	<30	42 to 58
Initial pellet density (% Total density)	92	92	--	97
Pellet diameter (cm)	0.9	0.9	--	1.4
Grain size (μm)	25	6	---	7 to 10
Discharge date	11/25/75	5/6/74		

[a]Data from Johnson (1982), Barner (1984), Johnson et al. (1983a), Forsyth et al. (1984), and Wilson and Oversby (1985).

[b]No data.

at approximately 25°C. The series 1 tests were conducted using deionized water as the leachant and in the series 2 tests well J-13 water was used. These tests also included fused quartz rods, which were removed at intervals to monitor plate-out. Series 3 differs from the previous two in that the tests are being conducted in sealed 304 (cycle 1) or 304L (cycle 2) stainless steel vessels at a higher temperature (85°C). The leachant used in series 3 is well J-13 water. One bare fuel specimen at 25°C is included in series 3 so that the results from the series 2 tests can be compared and so that the effect of changing the experimental vessel on the observed release of radionuclides, exclusive of any temperature effect, can be evaluated.

Each test series includes several types of test specimens, each a 5-in. segment cut from a full-sized rod: (1) bare fuel with the emptied cladding hulls, (2) fuel rod segments fitted with watertight end caps and with laser-drilled holes through the cladding, (3) fuel rod segments with watertight end caps and with a machined slit through the cladding, and (4) undefected fuel rod segments with watertight end caps. These various specimen types approximate fuel rods with differing degrees of cladding failure. The end caps prevent accidental access of water to the fuel by means other than the induced defect. The exterior surfaces of all specimens are cleaned of radioactive contamination before used in a test. Series 1 tests used spent fuel from the Turkey Point Unit 3 reactor. Series 2 and 3 used spent fuel from two sources: fuel from the H. B. Robinson Unit 2 reactor obtained from the Pacific Northwest Laboratory Materials Characterization Center and designated as ATM-101 (Barner, 1984) and fuel from the Turkey Point reactor. The two fuels are similar PWR fuels from the same vendor and are of approximately the same vintage. As can be seen from Table 7-14, both fuels have low gas release and similar burnup. The grain size of the Turkey Point fuel, however, is significantly larger than that of the H. B. Robinson fuel.

Data were obtained on all samples for americium isotopes, plutonium isotopes, curium-244, uranium, and cesium isotopes. Cobalt-60 was measured in most test solutions. Selected samples were analyzed for technetium-99, strontium-90, neptunium-237, selenium-79, carbon-14, and iodine-129. In addition, selected samples were analyzed in three fractions: (1) unfiltered, (2) filtered through 0.4 micron filters (Nucleopore 110407 polycarbonate), and (3) filtered through 1.8 nm filters (Amicon CTS-1 membrane cone centrifuge filters).

The following discussion concentrates on the results of the series 2 tests (Wilson, 1987) because they are the most relevant data available for the conditions in the NNWSI Project proposed repository. Significant differences were noted in the behavior of spent fuel in the series 1 and 2 tests because of the use of well J-13 water in series 2. These differences are discussed when they provide insight into the dissolution process or release mechanism. Complete results for the series 1 tests are contained in Wilson (1985b).

Figure 7-20 shows the H. B. Robinson uranium data for unfiltered solutions for series 2, cycles 1 and 2. Data points with a downward pointing arrow are data reported as below detection limits. The line marked 10^{-5} inventory is the solution concentration that would result if 1 part in 100,000 of the test specimen were to dissolve in the leachant. This level is indicated in the figures and tables simply as a convenient reference point for comparing the release of various radionuclides from specimens of different sizes and for comparing the release of radionuclides normalized to their abundance in the fuel. This value cannot be directly related to the NRC release rate limit of 1 part in 100,000 per year of the 1000-yr inventory.

Conversion of solution concentrations to release rates requires a model for spent fuel dissolution in the repository setting. This is discussed in more detail below and in Section 8.3.5.10.

Solution uranium concentrations reached relatively stable levels after a few days. The H. B. Robinson cycle 1 bare fuel test, which required more than 60 d to reach a stable level, was a notable exception to this. Uranium concentration data for the bare fuel tests are plotted on a linear scale in Figure 7-21. The H. B. Robinson cycle 1 test peaked at 4.5 micrograms per milliliter on day 6 and decreased to 1.2 micrograms per milliliter at the

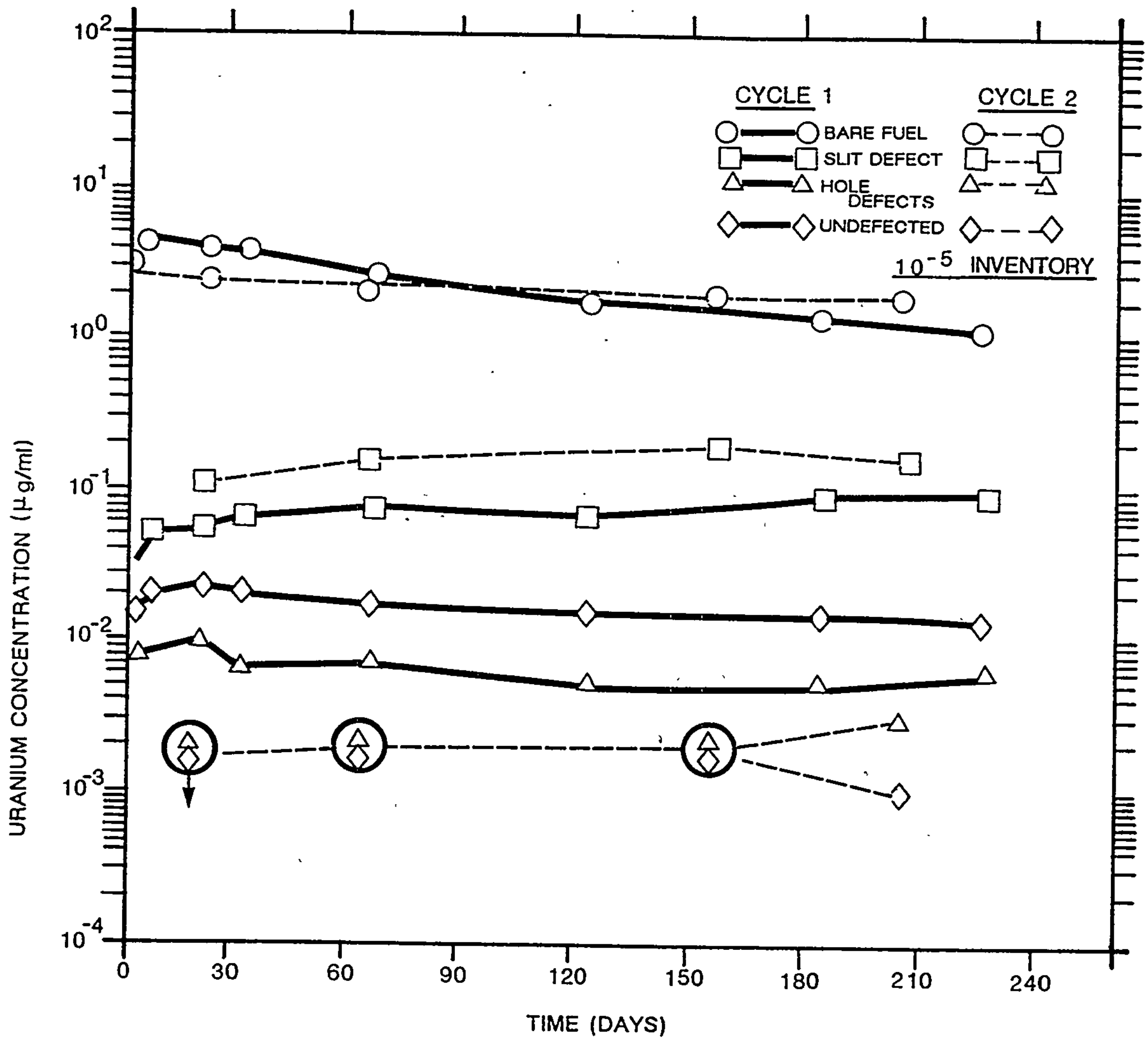


Figure 7-20. Uranium concentrations in unfiltered solution samples. series 2. cycles 1 and 2 (25°C. well J-13 water) H. B. Robinson unit 2 fuel. Circled data points have identical reported values but are offset for clarity. Modified from Wilson (1987)

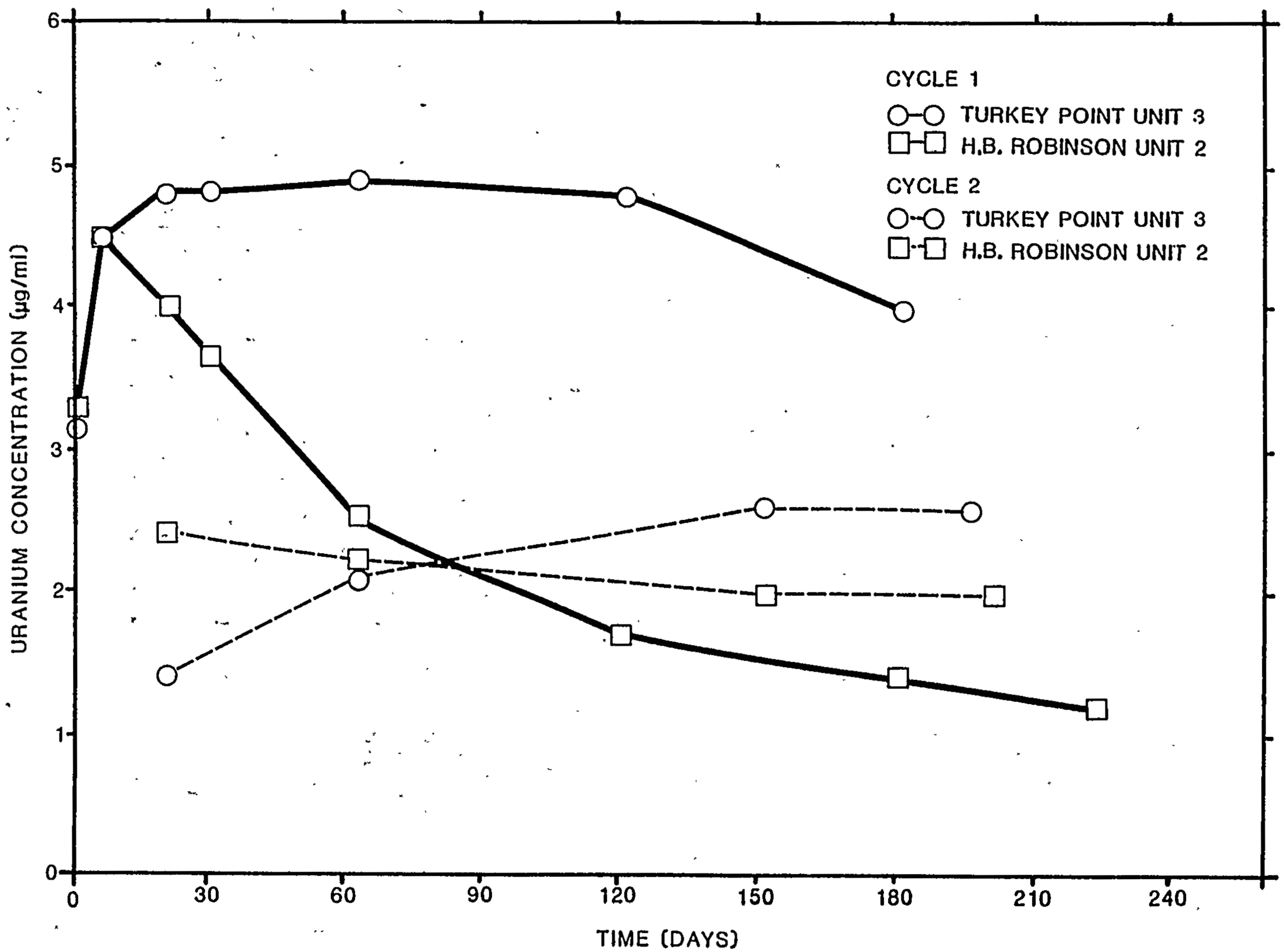


Figure 7-21. Uranium concentrations for bare fuels in well J-13 water, linear scale, series 2, cycles 1 and 2 (25°C, well J-13 water). Modified from Wilson (1987).

termination of cycle 1 as the solution apparently equilibrated with a phase having lower solubility than that initially present on the fuel surface. Uranium in the cycle 1 Turkey Point test began to decrease only after 120 d dropping from a peak value of 4.9 micrograms per milliliter on day 30 to 4.0 micrograms per milliliter at the cycle termination, suggesting that more of a higher solubility phase was initially present. The greater amount of a high solubility phase in the Turkey Point fuel may be related to this fuel's greater exposure to air before testing, which may have resulted in a more extensively oxidized fuel surface. During cycle 2, the solution concentration levels of uranium were similar for both fuel types. Uranium concentration dropped from a peak of 2.4 micrograms per milliliter in the day 20 sample to 2.0 micrograms per milliliter in the H. B. Robinson bare fuel test. In the Turkey Point test, uranium concentration peaked at 2.6 micrograms per milliliter in the day 154 sample and dropped to 2.4 micrograms per milliliter in the final sample. Data from day 224 of the third test cycle (not shown) for these specimens show uranium concentrations dropping to 1.4 micrograms per milliliter and 1.2 micrograms per milliliter for the H. B. Robinson and Turkey Point fuels, respectively (Wilson, 1987).

Table 7-15 summarizes the uranium release results for series 2, cycles 1 and 2 of the spent fuel dissolution tests. These data show that the uranium release for the bare fuel specimens was greater than that of any of the defected cladding specimens. The higher fractional release from Turkey Point fuel relative to H. B. Robinson fuel in the three defected cladding configurations may reflect the presence of a more highly oxidized fuel surface on the Turkey Point fuel. Higher uranium concentrations in the undefected test versus the drillhole defect test for H. B. Robinson cycle 1 specimens is likely due to the presence of residual contamination on the cladding exterior of the undefected specimen; in cycle 2, the uranium concentrations were more nearly the same for the drillhole defect and the undefected specimens.

The uranium solubility behavior observed in the series 2 tests using well J-13 water was significantly different from that seen in the series 1 tests that used deionized water. In series 2, essentially all the uranium measured in the unfiltered solution samples passed through the 0.4 micrometer and 1.8 nm filters, indicating that the uranium was in true solution. In contrast in series 1, much of the uranium was trapped on the 0.4 micrometer filter, indicating that much of the uranium in the aqueous phase was in a particulate or colloidal state. The real uranium solubility in the deionized water used in series 1 was at or below the detection limit of the technique used to measure uranium in that test (≈0.001 microgram per milliliter). The difference between the two test series is attributed to the presence of ≈120 micrograms per milliliter of bicarbonate ion in the well J-13 water that complexes with uranium in solution and increases its solubility (Wilson, 1985a; 1987).

Table 7-16 lists the total fractional release of the actinides measured in the series 2 tests along with the percentage of the release accounted for by the activity present in the aqueous phase (unfiltered solution activity). Because of the very low activities in solution, reliable neptunium data were not obtained for most of the sample configurations; however, the data are consistent with the results for the other actinides.

Table 7-15. Uranium release data for series 2, cycles 1 and 2 experiments with H. B. Robinson unit 2 and Turkey Point unit 3 fuels (25°C, well J-13 water). Units are micrograms unless otherwise noted[a] (page 1 of 2)

Parameter	Bare fuel HBR[b]	Bare fuel TP[c]	Slit defect HBR	Slit defect TP	Hole defect HBR	Hole defect TP	Undefected HBR	Undefected TP
				CYCLE 1				
Solution samples	253	351	6.59	35.8	0.63	2.27	1.67	0.76
Final solution	300	1,000	23.80	212.5	1.50	7.25	3.25	2.00
(ppm U)	(1.2)	(4.0)	(0.09)	(0.85)	(0.006)	(0.029)	(0.013)	(0.008)
Rod samples	36	15	0.31	0.54	<0.18	<0.19	<0.22	<0.10
Rinse	660	366	1.80	10.2	0.60	<0.60	<0.60	<0.6
Acid strip	2,700	960	1.50	15.9	0.60	2.70	0.6	<0.3
Total release	3,949	2,692	34.00	274.9	3.51	<13.01	<6.34	<3.76
x 10^5 inventory	5.66	11.67	0.047	0.662	0.005	<0.030	<0.009	<0.009[d]
% in aqueous phase	14.00	50.19	89.38	90.32	60.86	73.17	77.60	--
				CYCLE 2				
Solution samples	142	135	10.15	22.1	<0.13	0.22	<0.13	0.18
Final solution	500	600	40	125.0	0.75	1.00	0.25	0.75
(ppm U)	(2.0)	(2.4)	(0.16)	(0.50)	(0.003)	(0.004)	(0.001)	(0.003)
Rod samples	18	3	<0.08	0.054	<0.06	<0.05	<0.06	<0.05
Rinse	102	39	1.20	3.6	0.60	<0.60	0.60	<0.06
Acid strip	300	156	1.20	0.3	4.50	0.30	--	<0.3
Total release	1,062	933	52.63	151.0	6.04	<2.17	<1.04	<1.88
x 10^5 inventory	1.54	4.13	0.073	0.363	0.008	<0.005	<0.0015	<0.004
% in aqueous phase	60.45	78.78	95.29	97.42	14.57	56.22	36.54	--

Table 7-15. Uranium release data for series 2, cycles 1 and 2 experiments with H. B. Robinson unit 2 and Turkey Point unit 3 fuels (25°C, well J-13 water). Units are micrograms unless otherwise noted[a] (page 2 of 2)

Parameter	Bare fuel		Slit defect		Hole defect		Undefected	
	HBR[b]	TP[c]	HBR	TP	HBR	TP	HBR	TP
				SUMMARY, CYCLES 1 and 2				
Total release	5,011	3,625	86.63	425.9	9.55	<15.1	--	<5.64
$\times 10^5$ inventory	7.20	15.80	0.120	1.025	0.013	<0.035	<0.010	--

[a]Source: Wilson (1987).
[b]H. B. Robinson unit 2.
[c]Turkey Point unit 3.
[d]-- = Not applicable.

Table 7-16. Summary of the measured fractional release for series 2, cycles 1 and 2 experiments with H. B. Robinson unit 2 and Turkey Point unit 3 fuels (25°C, well J-13 water). Units are parts per 100,000 unless otherwise noted[a] (page 1 of 2)

Parameter	Bare fuel HBR[b]	Bare fuel TP[c]	Slit defect HBR	Slit defect TP	Hole defect HBR	Hole defect TP	Undefected HBR	Undefected TP
Uranium								
Cycle 1	5.66	11.67	0.047	0.662	0.005	0.030	0.0090	<0.009
Cycle 2	1.54	4.13	0.073	0.363	0.008	0.005	0.0015	<0.004
Sum	7.20	15.80	0.120	1.025	0.013	0.035	0.0105	<0.013
% in aqueous phase:								
Cycle 1	14.00	50.19	89.43	90.32	60.68	73.17	75.69	--[d]
Cycle 2	60.45	78.78	95.29	97.42	14.57	56.22	60.58	--
Both Cycles	23.96	57.66	93.02	92.85	32.30	70.80	73.53	>66.04
Pu-239 + Pu-240								
Cycle 1	7.18	7.31	0.009	0.1090	0.002	0.022	0.0020	<0.005
Cycle 2	1.28	1.57	0.008	0.0147	0.012	0.004	0.0008	<0.002
Sum	8.46	8.88	0.017	0.1237	0.0157	0.0255	0.0028	<0.007
(Sum Pu)/(Sum U)	1.17	0.56	0.14	0.11	1.21	0.73	--	--
% in aqueous phase:								
Cycle 1	1.56	12.61	17.10	6.25	48.57	8.01	23.15	≈13
Cycle 2	4.12	19.84	11.62	47.31	1.20	17.15	23.35	≈22
Both Cycles	1.95	13.89	14.52	11.13	14.05	9.40	23.21	>13.70
(aq[e] Pu)/(aq U)[d]	0.10	0.13	0.02	0.01	0.53	0.10	--	--
Am-241								
Cycle 1	8.04	6.88	0.009	0.126	0.0028	0.023	0.0031	<0.003
Cycle 2	0.77	1.33	0.005	0.010	0.0138	0.004	<0.0003	<0.002
Sum	8.81	8.21	0.014	0.136	0.0166	0.027	<0.0034	<0.005
(Sum Am)/(Sum U)	1.22	0.52	0.12	0.13	1.28	0.77		
% in aqueous phase:								
Cycle 1	1.57	14.04	20.53	6.16	47.95	5.86	--	--
Cycle 2	1.00	5.15	2.00	17.35	<0.66	13.65	--	--
Both Cycles	1.52	12.60	13.68	6.95	≈8.63	6.96	≈14.69	≈19.77
(aq Am)/(aq U)	0.08	0.11	0.02	0.01	0.34	0.08	--	--

Table 7-16. Summary of the measured fractional release for series 2, cycles 1 and 2 experiments with H. B. Robinson unit 2 and Turkey Point unit 3 fuels (25°C, well J-13 water). Units are parts per 100,000 unless otherwise noted[a] (page 2 of 2)

Parameter	Bare fuel HBR[b]	Bare fuel TP[c]	Slit defect HBR	Slit defect TP	Hole defect HBR	Hole defect TP	Undefected HBR	Undefected TP
Cm-244								
Cycle 1	8.64	7.79	0.010	0.137	0.0018	0.027	0.0024	0.0045
Cycle 2	1.61	1.42	0.010	0.010	0.0170	0.003	<0.0012	<0.0014
Sum	10.25	9.21	0.020	0.147	0.0188	0.030	<0.0036	<0.0059
(Sum Cm)/(Sum U)	1.20	0.58	0.16	0.14	1.45	0.87	--	--
% in aqueous phase:								
Cycle 1	2.29	18.13	19.23	7.34	60.29	5.05	11.20	7.87
Cycle 2	1.04	6.73	2.19	24.44	0.19	3.35	--	--
Both Cycles	2.09	16.37	10.71	8.50	5.94	4.86	>8.70	>6.83
(aq Cm)/(aq U)	0.12	0.17	0.02	0.01	0.27	0.06	--	--

[a]Wilson (1987).
[b]HBR = H.B. Robinson, unit 2.
[c]TP = Turkey Point, unit 3.
[d]-- indicates no data.
[e]aq = aqueous.

The H. B. Robinson bare fuel test yielded total fractional releases (summed over both cycles of the test) of approximately 8 to 10 parts in 100,000 of the inventory for all the actinides, including uranium. The analogous Turkey Point specimen yielded similar releases for the higher (i.e., transuranic) actinides but had a uranium release approximately twice that of the other actinides. As noted previously, this may be due to the presence of a more extensively oxidized surface on the Turkey Point fuel.

As for uranium, the fractional release of higher actinides in the bare fuel test was much greater than for any of the other sample configurations. The results listed in Tables 7-16 indicate that for each defected specimen type, the release of the higher actinides occurred congruently, with the Turkey Point specimens having fractional releases from 2 to 10 times that observed for the analogous H. B. Robinson specimens. The total fractional uranium release for both hole defect specimens is similar to that observed for the other actinides. In the slit defect specimens, the fractional release of uranium was approximately 8 to 10 times the release of the other actinides for both fuel types.

The fractional releases just discussed above refer to all the material mobilized from the test specimens and include material plated out on the quartz rods and on the test vessel walls, recovered by acid stripping the vessel. Of greater interest is the portion of the material present in the aqueous phase since it is this radioactivity that has the potential to be transported from a failed disposal container by fluid entering and leaving the container. This material may be either in true solution or present as suspended particles or colloids; thus the entries in Tables 7-16 for the percentage in the aqueous phase are based on unfiltered solution analyses. The percentages of the higher actinides that were in the aqueous phase were approximately the same for each specimen type; however, there were significant differences between the different specimens. These differences do not appear to be systematically related to either the cladding configuration or the fuel type. In all instances, the fraction of the uranium in the aqueous phase significantly exceeded that of the other actinides. Higher actinide to uranium ratios (normalized to the same ratio in the fuel) in the aqueous phase ranged from 0.06 to 0.43. It appears that uranium has the potential to be transported from a failed disposal container preferentially to the other actinides.

Most of the aqueous phase activity due to the higher actinides was also measured in the 0.4 micrometer-filtered fractions (58 to 96 percent). Much of the plutonium activity also passed through the 1.8 nm filter (30 to 76 percent); however, the majority of the americium and curium was stopped by this filter (only 3 to 12 percent of the unfiltered activity was found in the 1.8 nm filtrate). The americium and curium may have been present as large complexes or colloids, preventing them from passing through the finer filter. Alternatively, the concentrations of americium and curium were only ≈100 pg/g and ≈3 pg/g, respectively. At such low concentrations, these elements may have been removed from solution by adsorption onto the 1.8 nm filters.

The fission product cesium, which is partially released from the oxide fuel matrix during irradiation, segregates to the grain boundaries and the fuel-cladding gap. This gap and grain boundary inventory dissolves immediately upon contact with water. As shown in Figure 7-22, cesium was rapidly

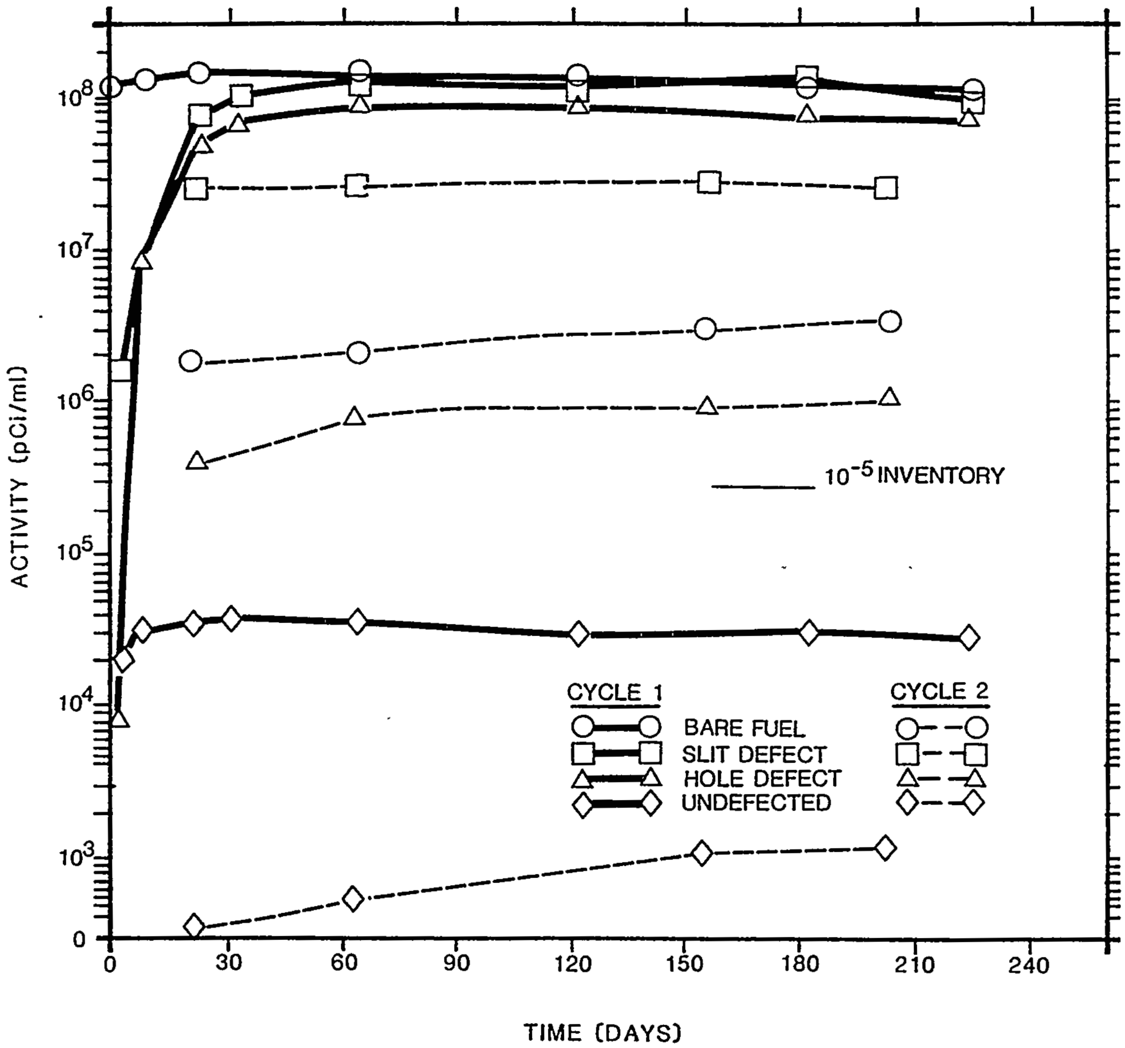


Figure 7-22. Cesium-137 activities in unfiltered solution samples for series 2. cycles 1 and 2 experiments with H. B. Robinson unit 2 fuel (25°C. well J-13 water). Modified from Wilson (1987).

released in the NNWSI Project series 2 dissolution tests. Relatively constant solution levels followed this initial release. Most of the release occurred in cycle 1 (Table 7-17 and Figure 7-22), and the mobilized cesium was retained in solution. For the defected cladding configurations, more than 97 percent of the cesium released was in the aqueous phase. Slightly lower percentages for the bare fuel tests (Table 7-17) probably reflect the mobilization of small fuel particles that quickly settle. Unlike the case of the actinides, the presence of defected cladding did not act as a significant barrier to the release of cesium. The Turkey Point hole defect test was a exception to this; the fractional release of cesium for this test (< 0.01 percent) was much lower than expected and is not currently understood.

Fractional cesium releases for the Turkey Point bare fuel and slit defect tests were 0.3 and 0.2 percent, respectively. This is close to the reported fractional fission gas release for this fuel (Table 7-14). Fractional releases for the H. B. Robinson bare fuel, slit, and hole defect specimens were 0.8, 0.8, and 0.4 percent, respectively. The reported fractional fission gas release for this fuel is only 0.2 percent. The ≈250 percent greater fractional cesium release for the H. B. Robinson fuel compared with the Turkey Point fuel is thought to be a reflection of the much smaller grain size of the former. The larger grain boundary area and shorter diffusion path lengths to grain boundaries likely results in a greater grain boundary cesium inventory as cesium is transported along grain boundaries during reactor operation. Given the similarities in the fuels (same manufacturer, same vintage, similar burnup, similar gas release), the difference in cesium release clearly shows that more work will be needed to establish the factors that control cesium distribution in the fuel before general conclusions regarding cesium release can be substantiated.

The behavior of cesium in well J-13 water was similar to that observed in deionized water. The main difference in behavior in the two leachants was that no decrease in cesium levels in the aqueous phase occurred with well J-13 water. In the series 1 test, cesium levels over bare fuel began to decrease after 60 d. This was thought to be due to the precipitation of some cesium salt, possibly $Cs_2U_2O_7$, the cesium analogue of a sodium salt that has been suggested by Johnson (1982) to be the solubility control for uranium in solutions over spent fuel. The well J-13 water contains a significant amount of sodium (50 ppm); this may cause sodium uranate to form at the expense of cesium uranate, leaving the cesium in the solution phase. To date, neither of these alkali uranate phases has been identified during post-test examination of the specimens.

The release of technetium is of particular interest since the pertechnetate ion is expected to be quite soluble in the mildly oxidizing ground water at the NNWSI Project candidate repository. Like cesium, technetium tends to separate from the oxide fuel matrix during reactor operation. Technetium should exist as a metal at the oxygen fugacity present in fuel during irradiation. Hence, it is associated with other phase-separated metallic fission products in spent fuel.

The detection limit of the radiochemical procedure used for the technetium analyses is just adequate to detect 1×10^{-5} of the specimen inventory in solution. Significant activities of technetium-99 were measured in most solution samples taken from the bare fuel and slit defect tests (Table 7-17).

Table 7-17. Summary of the measured fractional release of Cs, Tc, and I for series 2, cycles 1 and 2 experiments with H. B. Robinson and Turkey Point fuels (25°C, well J-13 water). Units are in parts per 100,000 unless otherwise indicated[a] (page 1 of 2)

Parameter	Bare fuel		Slit defect		Hole defect		Undefected	
	HBR[b]	TP[c]	HBR	TP	HBR	TP	HBR	TP
Cs-137 + Cs-134								
Cycle 1	776	308	664	144	425.0	7.8	0.19	0.11
Cycle 2	19	16	140	75	5.2	1.5	ND[e]	ND
Sum	796	324	804	219	430.2	9.3	--[f]	--
(Sum Cs)/(Sum U)[d]	110.55	20.50	6700.00	213.66	33092	264.8	--	--
% in aqueous phase								
Cycle 1	94.88	81.52	98.97	96.95	98.80	98.01	95.88	92.90
Cycle 2	92.80	89.31	98.90	98.05	97.77	98.09	ND	ND
Both cycles	94.84	81.95	98.96	97.33	98.79	98.02	95.88	92.90
(aq Cs)/(aq U)	438.0	28.6	7127.7	224.0	101196	367	24	ND
Tc-99								
Cycle 1	≈23	≈32	≈2.8	<15.3	ND	ND	ND	ND
Cycle 2	<8.6	<8.3	<2.1	<6.6	ND	ND	ND	ND
Sum	32	41	<4.9	<22	ND	ND	ND	ND
(Sum Tc)/(Sum U)	4.44	2.59	≈40.83	≈21.46	ND	ND	ND	ND
% in aqueous phase								
Cycle 1	77.00	79.32	≈43	--	ND	ND	ND	ND
Cycle 2	>86.49	>63.13	--	--	ND	ND	ND	ND
Both cycles	>79.55	>75.98	>48.70	>84.47	ND	ND	ND	ND
(aq Tc)/(aq U)	14.66	3.40	21.11	19.50	ND	ND	ND	ND
I-129								
Cycle 1	10.5	29.2	5.4	7.9	ND	ND	ND	ND
Cycle 2	7.5	12.0	0.7	5.7	ND	ND	ND	ND
Sum	18	41.2	6.1	13.6	ND	ND	ND	ND

Table 7-17. Summary of the measured fractional release of Cs, Tc, and I for series 2, cycles 1 and 2 experiments with H. B. Robinson and Turkey Point fuels (25°C, well J-13 water). Units are in parts per 100,000 unless otherwise indicated[a] (page 2 of 2)

Parameter	Bare fuel		Slit defect		Hole defect		Undefected	
	HBR[b]	TP[c]	HBR	TP	HBR	TP	HBR	TP
(Sum I)/(Sum U)	2.45	2.61	50.8	13.27	ND	ND	ND	ND
% in aqueous phase	ND	ND	ND	ND	ND	ND	ND	ND
Cycle 1	ND	ND	ND	ND	ND	ND	ND	ND
Cycle 2	ND	ND	ND	ND	ND	ND	ND	ND
Both cycles	ND	ND	ND	ND	ND	ND	ND	ND

[a]Wilson (1987).
[b]H. B. Robinson, Unit 2.
[c]Turkey Point, Unit 3.
[d]See Table 7-16 for uranium values.
[e]ND = no data.
[f]-- = not available.

As with cesium, most of the technetium release occurred during the early part of cycle 1. More than 75 percent of the mobilized technetium was in the aqueous phase, consistent with the solubility expectations. The release of technetium from the H. B. Robinson specimens was greater than for the Turkey Point specimens. This is again likely due to the finer grain size of the H. B. Robinson fuel.

Iodine-129 was measured on selected solution samples. A summary of the data, given in Table 7-17, shows that the apparent release of iodine was much less than that of cesium. It is possible, however, that iodine was lost to the air since these tests were run in unsealed vessels; however, since Johnson et al., (1983b) found high concentrations of iodine in tests conducted under conditions similar to the NNWSI Project tests, it is likely that iodine loss from solution to the air is a minor effect. The series 3 tests, run in sealed vessels, should provide better data on iodine release. Nevertheless, if one assumes that iodine loss was negligible, the iodine measured in solution correlates better with the uranium total release than with either cesium or technetium. Since iodine should be present in solution as iodide, it should not plate out on the stainless steel vessel. There is some evidence that iodide is sorbed by the Zircaloy cladding present in these tests (Johnson et al., 1985). A check for plate-out components was made at the termination of the second run by measuring the iodine-129 activity in a acid strip of the test vessel; only $\approx 1.3 \times 10^{-8}$ of the specimen inventory was recovered in this manner. However, since the vessel strip used nitric acid, iodine may have been lost to the atmosphere from the strip solution. Note that only the vessel and basket were stripped; the Zircaloy cladding was not. The data suggest that iodine-129 is being mobilized from the bare fuel at a rate similar to the matrix components and not at the much higher rate exhibited by cesium. The slit defect sample shows a clear enhanced release of iodine relative to the actinides and probably represents a gap inventory release. The amount of release, however, is still two orders of magnitude below the fractional release of cesium from slit defect samples.

Technetium is released at a rate somewhat higher than the matrix dissolution rate and higher than that of iodine-129 in bare fuel. For the slit defect case, the releases of iodine and technetium are comparable.

Wilson (1987) reports the results of carbon-14 analyses for some of the series 2 solution samples. No correlation was found between defect severity and release for either the H. B. Robinson or Turkey Point samples, suggesting that the carbon-14 was released from the cladding rather than from the fuel. Carbon-14 was radiochemically separated and measured on fuel and cladding for two samples of the H. B. Robinson fuel rod used for the series 2 specimens. The average of the analyses was 0.53 microcuries per gram for the cladding and 0.49 microcuries per gram for the fuel. In general, 30 to 60 percent of the carbon-14 in the spent fuel waste form is in the nonfuel components such as the cladding. The exact amount depends on the nitrogen impurity levels in the cladding and in the structural materials. By using the measured carbon-14 inventories in H. B. Robinson spent fuel, the carbon-14 release in the H. B. Robinson bare fuel tests was estimated to represent $\approx 1 \times 10^{-3}$ of the cladding inventory or $\approx 2 \times 10^{-4}$ of the fuel inventory in each of the cycles. These fractional inventory values are lower limits on the actual release of carbon-14 since the test vessels used were capped with loose-fitting lids that may have allowed loss of carbon as carbon dioxide to the atmosphere.

The results of a test using a spent fuel assembly under an air atmosphere in a sealed container at 275°C showed that there is an initial release of approximately 0.3 percent of the carbon-14 inventory of the entire assembly (Van Konynenburg et al., 1984). The carbon-14 release occurred as carbon dioxide gas. Gas analyses of storage tests of spent fuel stored under inert atmospheres (helium and nitrogen) do not show significant releases of carbon-14. These results support the hypothesis that the observed release under oxidizing conditions is due to the removal of carbon as carbon dioxide from the outer surface of the Zircaloy cladding by reaction of the carbon with oxygen in the atmosphere (Van Konynenburg et al., 1986).

The test was continued with the air in the container pumped out and replaced several times to measure additional release. During the first part of the test, one of the fuel rods ruptured. This occurred after the initial carbon-14 release was measured. Despite the presence of a ruptured rod, the subsequent releases of carbon-14 were much lower than the initial release (Van Konynenburg et al., 1986). These data are still being analyzed; however, the observed carbon-14 release appears to be consistent with the fuel rod fill gas analyses reported by the Materials Characterization Center on similar fuel (Barner, 1984). Carbon-14 concentrations reported for the fill gas average 0.81 nCi/cm^3 (STP), which is equivalent to an average activity per fuel rod of approximately 0.3 microcuries. This is more than three orders of magnitude lower than the observed initial release reported by Van Konynenburg et al. (1984), supporting their conclusion that the initial carbon-14 release came from the cladding or metal components of the assembly rather than from the fuel.

There is presently only one data set for dry oxidative release of carbon-14 from spent fuel (Van Konynenburg et al., 1984, 1986). Additional experiments are planned to determine the release rate and fraction released under conditions expected at long times. Work is also in progress to determine the spatial distribution of carbon-14 in the cladding by means of controlled etching of the cladding surfaces. A description of the planned tests is in Section 8.3.5.10.

Four types of solids characterizations were performed at the end of series 2: (1) scanning electron microscope (SEM) examination of small, fractured fuel particles; (2) SEM examination with energy dispersive x-ray (EDX) microanalysis of the filters used to filter the sample solutions; (3) SEM and EDX analyses of residues from bare fuel rinse solutions; and (4) posttest radiometallurgical examination of polished sections from test specimens (Wilson, 1987).

The SEM examination of fuel particles did not reveal any significant change in fuel structure caused by the test. As is typical of spent fuel, the fracture surfaces of the fuel particles tended to follow grain boundaries. This contrasts with the behavior of nonirradiated fuel, which is a hard ceramic material that exhibits transgranular cleavage when fractured. Areas of mixed cleavage and grain boundary fracture were observed in some spent fuel particles thought to be from near the outer radial regions of the fuel pellet. Irradiation temperatures are lower, and less fission product migration to grain boundaries occurs in these areas as compared with the center of the pellet; thus, there is less change in the fracture properties in these regions during irradiation.

An SEM image of one of the larger particles obtained from the cycle 1, H. B. Robinson rinse residues is shown in Figure 7-23. The particle is a small fuel fragment, partially coated with a surface layer of material that EDX analyses show to be primarily silicon or silica (the EDX method does not detect oxygen or other light elements). The layer appears to be a layer of silica gel ranging in thickness from 10 to 25 micrometers. Similar layers were observed on other H. B. Robinson particles as well as on Turkey Point fuel particles. The quantity of silicon involved in the layers appears to be a significant portion of the ≈7.5 mg of silicon contained in the 250 ml of well J-13 water used in the tests. The effect of this layer on the dissolution rate of spent fuel is not known. An H. B. Robinson bare fuel specimen is being tested at 25°C in a stainless steel vessel in the series 3 tests. This specimen was included to examine the effects, if any, of the test vessel material on the results of the test. The test being conducted in the stainless steel vessel is a closer representation of the repository case.

SEM and EDX analyses of selected filters used in series 2 commonly revealed extremely small particles composed dominantly of silicon. These probably represent colloidal silica flocs.

Metallographic examination of polished, mounted fuel fragments recovered after the series 2 tests did not reveal any significant evidence of grain boundary dissolution. Particular attention was given to the examination of the particle edges; however, no unusual features were noted in the post-test samples. These observations differ from those made on the posttest series 1 fuel fragments. Significant grain boundary dissolution was noted in the series 1, H. B. Robinson bare fuel particles. This difference is probably due to the more aggressive action of the deionized water used in the series 1 tests (Wilson, 1987).

Metallographic examination of the fuel at the pellet-cladding gap in the slit defect specimens of both fuel types did not reveal any unusual features that could be related to fuel dissolution.

Solution analyses for nonradioactive components were performed on the starting well J-13 water and on selected periodic solution samples (Wilson, 1987). The solution chemistry data indicate that there was little change in the composition of the leachant over the course of the tests. The only consistent change was a shift to more basic pH during cycle 1. The barrel of well J-13 water used for these tests also shifted during this time, and the change is attributed to equilibration of the well J-13 water with the atmosphere. Silicon levels in solution were of interest since the tests were run in fused silica vessels. Except for a 20 to 30 percent drop in silicon in the 30- and 120-d cycle 1, H. B. Robinson samples, the silicon concentrations remained relatively constant. SEM characterization of residues rinsed from the fuel specimens revealed a deposit of silica on the fuel surface. Apparently, the silica lost by precipitation was replaced at approximately the same rate by dissolution of the vessel; thus a constant silica concentration was maintained in solution. Another minor change in solution chemistry was

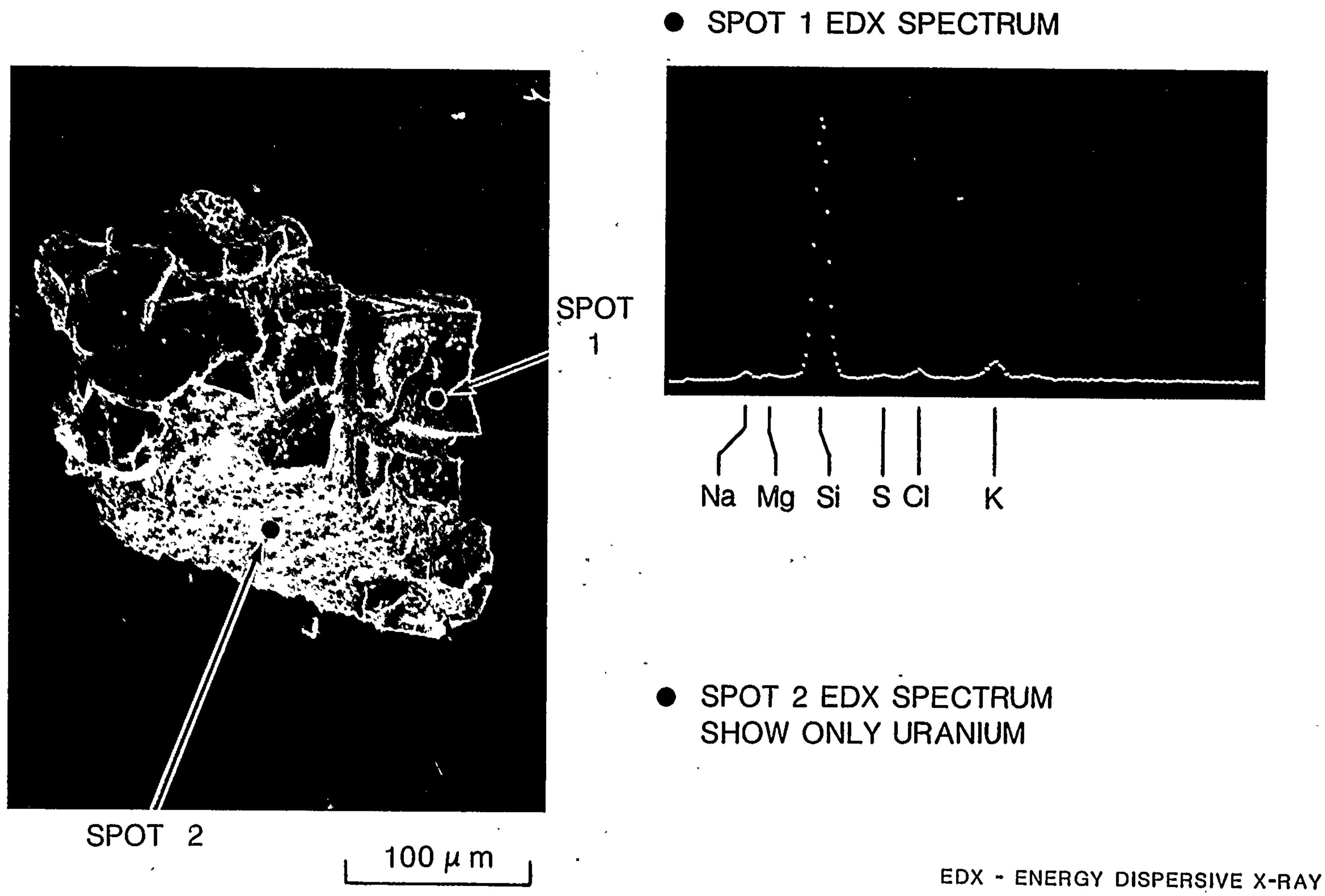


Figure 7-23. Fuel particle from the series 2. H. B. Robinson bare fuel test showing remnants of a silica layer deposited on the particle surface during the test. Modified from Wilson (1987).

the conversion of some of the initial nitrate to nitrite by radiolysis. The source of the nitrite is thought to be primarily nitrate rather than dissolved air because the measured concentrations of nitrate and nitrite in solution show an inverse correlation.

The link between the laboratory data and the expected repository performance comes through geochemical modeling calculations. The EQ3/6 geochemical modeling code is being used for this purpose. For geochemical modeling calculations to be meaningful, the solid phases that control the solution concentrations and the solubility of those phases under appropriate conditions must be identified. If the relevant thermodynamic properties of the solids are known, the solubility data obtained under one set of conditions can be used to calculate solubilities under a range of other conditions of interest. For many of the elements of interest in spent fuel dissolution, solubility data are not available for the solids that are predicted to be important in limiting solution concentrations of radionuclides.

The peak concentration of uranium in dissolution tests in well J-13 water was less than 5 ppm, and the solutions for cycle 2 of the tests were 2 to 2.6 ppm. For plutonium, the peak concentration was 5 ppb. These concentrations are much lower than the values of 50 ppm and 430 ppb, respectively, used by Kerrisk (1984) in his study of dissolution rates and solubility limits of radionuclides in a tuff repository.

Other data also point to a much lower actinide solution concentration than that predicted by the current geochemical codes. The data obtained by Forsyth et al. (1984, 1985) have a range of uranium concentrations of 0.1 to 2 ppm, with an average value of 0.8 ppm in KBS water. Forsyth et al. (1985) report a plutonium saturation value of ≈1 ppb under the same conditions. Johnson (1982) found uranium concentrations of 1.2 to 4.5 ppm in the AECL-KBS water (higher bicarbonate than well J-13 water) and 0.2 to 3 ppm in AECL-GR water. The experimental data were obtained at fuel-weight to water-volume ratios that ranged from a low of 8 g/100 ml for the Swedish tests to 72 g/100 ml for the Canadian tests. There is no indication of a correlation between solution concentration of uranium and fuel-to-water ratio. There is a weak correlation with bicarbonate content, as might be expected. Some of the variation among the three sets of results may be due to intrinsic differences in the spent fuels. Nevertheless, despite the scatter and uncertainties, there is remarkably good agreement among the maximum solution concentrations in the three data sets. The agreement among the three sets of data with respect to uranium and plutonium solution concentrations suggests that these values represent a better estimate of the solubility of uranium and plutonium over spent fuel than the estimates obtained by calculation. This, in turn, suggests that either the phases responsible for limiting the concentration of plutonium and uranium in the system were not represented in the data base used in the calculations or the thermodynamic properties for the phases were not correct. To extrapolate the results of laboratory testing of spent fuel to the time scales relevant to repository disposal, it is necessary to develop an understanding of the dissolution mechanism for spent fuel. Detailed determination of the mechanism will require the analysis of alteration products on the surface of fuel that has undergone dissolution. Identification of the alteration products on the surface of the fuel and of any secondary phases formed by precipitation from solution will allow the development of a model that will describe both the solid and solution phases during interaction of spent fuel with solutions.

Oversby and Shaw (1986) have reviewed the published work on spent fuel and unirradiated uranium dioxide (UO_2) dissolution in waters similar in composition to well J-13 water and in deionized water. The following discussion will summarize their analysis.

The methods used by the various workers for reporting the results of leach tests can cause confusion and apparent inconsistencies between data sets, especially when the solution concentrations are limited by the solubility of a sparingly soluble phase. If solution concentrations are constant and data are reported as leach rate per unit time, then the calculated rate will depend on the length of the test. Short tests will give high leach rates and long tests will give low leach rates. The ratio of fuel to liquid will also affect the calculated leach rate since, for concentrations in solution that are controlled by solubility of a phase, the amount in solution depends on the volume of solution and not on the amount of solid in the system. Thus, a fuel-to-water ratio of 10 g/100 ml will give a calculated leach rate 10 times higher than a fuel-to-water ratio of 100 g/100 ml if the rates are reported as fractional release per unit time.

To compare release rates determined under different experimental situations, the data must be normalized to a common set of conditions. Unfortunately, published results frequently do not contain all the information that is needed to reduce the data sets to a common basis. Johnson (1982) acidified solutions before he removed them from the leach container; therefore, he measured the sum of the solution plus plate-out component. Forsyth et al. (1984, 1985) decanted the solutions from the leach vessel, filtered a fraction of the sample to determine colloidal material, rinsed the vessels, and then stripped the leach vessel with acid to measure the plate-out component. The NNWSI Project solution analyses can be directly compared with those of Forsyth et al. (1984, 1985), but because they did not report the amount of material recovered in the rinse or acid-strip solutions, total release values cannot be directly compared. (They believe that all the material in the acid-stripping solution was due to fine particles of fuel and that it was inappropriate to include that material in the total release. Their interpretation is probably correct, but because of the possibility of particulate transport in a repository, it is necessary to determine the total amount of material mobilized during the test even if mobilization occurs as very fine solids.)

The procedures used for testing of spent fuel may also cause differences among the three sets of results. Johnson (1982) conducted tests using a modification of the International Atomic Energy Agency (IAEA) test. He used 100 ml of solution in a polypropylene bottle with an open-ended slice of CANDU fuel plus cladding, leaving the fuel sample in the solution for a fixed period of time. He determined that the solution had unimpeded access through the pellet-cladding gap, and so his tests are most nearly equivalent to the bare fuel tests carried out in the NNWSI Project studies. After the end of the leaching period, the fuel sample was transferred to a fresh bottle of solution, and the leaching was continued. Leachate solutions were acidified in the bottle before removal; so any plated-out material was dissolved and included in the analysis. (This should not matter for uranium in the bicarbonate-rich ground waters but will have a significant effect on the other actinides.) Johnson conducted sequential tests for periods ranging from a few days to more than 100 d, with a total leaching time of nearly 3 yr.

The Johnson tests (1982) were conducted in granite ground water and measured cesium-137, strontium-90, technetium, uranium, and plutonium. Release rates were found to decrease for the first 6 months of the test and then to level out for all elements, including cesium. The range in release rates is nearly two orders of magnitude for the samples taken at 200 to 800 d into the test, with cesium, strontium, and technetium release all higher by a factor of 5 to 10 than the uranium release rate of a few parts in 10^8.

Katayama et al. (1980) reported results of the leaching of H. B. Robinson spent fuel in deionized water, 0.03M sodium bicarbonate ($NaHCO_3$), and three other leachants. They used the same general test procedure as did Johnson (1982) but used bare fuel fragments 2 to 5 mm in size, 15 g of fuel, and 300 ml of solution. This gives a fuel-to-solution ratio of 5 g/100 ml, more than 10 times lower than Johnson's. Katayama et al. (1980) acidified the solutions in the leaching vessel, as did Johnson (1982), and so they obtained data that were the sum of the solution component plus the readily soluble plate-out component which was retained in the solution due to acidification. They then did an acid stripping of the vessel and added the amount in the acid-stripping solution to the original solution value. This total was converted into a leach rate in g/cm^2,d. There is sufficient information in the report to back out the original total amount of each element that was mobilized during each sequential leaching step, but because the leach solution and acid-stripping data were added together (and never reported separately), it is difficult to compare their data meaningfully with any other group's data. They reported rather high uranium release in deionized water, which seems to conflict with Johnson's (1982) results; but since the data contained an indeterminate amount of plate-out material, there may be no inconsistency. In the series 1 NNWSI Project work, 99 percent or more of the uranium release in deionized water was recovered from rinsing the fuel sample (which may dislodge precipitated material) or from acid stripping the vessel or glass plate-out monitors.

Katayama et al. (1980) also reported data for sodium bicarbonate water leaching. The bicarbonate concentration was approximately 1,800 ppm, a level much higher than any of the ground waters used in fuel leaching. The data given in their report can be converted into solution concentrations if it is assumed that the bicarbonate concentration is sufficiently high to ensure that no uranium has plated out. The calculated concentrations range from 1 to 6 ppm uranium, a value that is in excellent agreement with the results for groundwater leaching of spent fuel discussed previously.

Forsyth et al. (1984, 1985) reported their data (1) as the fraction of the inventory in the aqueous phase (FIAP) and based their leach rates on FIAP or (2), in some instances, as solution concentrations. Therefore, their results for BWR fuel can be directly compared with the NNWSI Project solution analysis results for PWR fuel. They use a fuel-to-water ratio of 8 g/100 ml compared with the ratio of 32 g/100 ml used for the H. B. Robinson tests in well J-13 water. They found that the fraction of cesium released rapidly in the tests was approximately 1 percent, slightly higher than the 0.7 percent fission gas release. Strontium was released at about one-tenth the rate of cesium release, and uranium release was lower than strontium release by a factor of 10. The concentration of uranium found was generally between 0.1 and 1 ppm for contact times up to approximately 3 yr. Johnson (1982) found that solution concentrations of uranium averaged 3 to 4 ppm for the 60-d leach solutions. There appears to be a real difference between the two sets of data, and the difference is probably due to the different concentrations of bicarbonate in the solutions used in the two studies. The bicarbonate content of well J-13 water is similar to the KBS water; the uranium concentrations in the NNWSI Project tests are intermediate between the other two data sets.

Johnson et al. (1983a) have established a general correlation between measured fission gas release from CANDU fuel and the fraction of rapidly released cesium found during a leach test. For very low gas release (about 0.07 percent xenon), the cesium release is approximately 10 percent of the gas release. For higher gas release, the fraction of cesium released is higher and is generally 50 percent of the gas release for all cases with more than 0.13 percent xenon release. This generally agrees with NNWSI Project results on the two low-gas-release PWR fuels. The BWR high-burnup fuel tested by Forsyth et al. (1984, 1985) showed cesium rapid release somewhat in excess of fission gas release.

Johnson et al. (1983b) have extended the study of mobile element release from CANDU fuel to include iodine. Iodine and cesium release were measured in distilled water at 25°C for up to 5 d. The same possibility of iodine loss to air as applies to the series 1 and 2 NNWSI Project iodine data also applies to this study because the Canadian tests were also run in unsealed vessels. Because Johnson et al. (1983b) found large amounts of iodine in tests conducted under conditions similar to the NNWSI Project fuel dissolution tests, the low levels of iodine found in the NNWSI Project tests are believed not to be due to loss of iodine from solution. In general, the gas release, iodine, and cesium rapid release are comparable, but differences of a factor of two or three are common. More recent data on iodine release from low-gas-release fuels shows a pattern similar to cesium release, with iodine release being about 10 percent of the gas release for gas release less than 0.1 percent (Johnson et al., 1985).

Johnson et al., (1985) found that iodine release was low if deionized water was used as a leachant. The effect was attributed to sorption of trace iodide onto the Zircaloy cladding. Most of their subsequent experiments used a solution with 0.2 g/L of potassium iodide to act as a carrier. Although this increased the reproducibility of their results, the relevance of the iodine release measured in these experiments to the release in ground water in the absence of large amounts of iodide in solution are questionable. (They do, however, provide a means of determining the location of the iodine-129 inventory within the fuel.) The low iodine-129 release rates observed in the NNWSI Project experiments may either reflect a lower gap inventory of iodine-129 in PWR fuel or the relative immobility of the iodine-129 in the absence of an iodide carrier.

Some data are available on the dissolution of spent fuel at higher temperatures. Results of testing CANDU fuel at 150°C are contained in the report by Johnson et al. (1981), with a few additional data points for longer times being reported in Johnson et al. (1982). The experiments were conducted at 150°C in 1 L titanium autoclaves using 500 ml of granite ground water (Table 7-13), saturated with air. The runs were generally 8 to 10 d in length, with the fuel samples having been preleached at ambient temperature for 100 d to remove the readily soluble cesium fraction. During the run, the oxygen content of the system was thought to have decreased by a factor of 2 to 3. Experiments were also conducted using deionized water saturated with air.

The results gave similar uranium and plutonium concentrations for deionized water and granite groundwater, with an average concentration of 0.05 to 0.1 ppm for uranium and 0.1 ppb for plutonium. The uranium concentrations were much lower than those observed at ambient temperature. Data for leaching at 260 d gave a higher uranium concentration (0.2 ppm), which is still much lower than the ambient temperature results.

The NNWSI Project series 3 dissolution tests are being conducted at 85°C. Based on the results of Johnson et al. (1981, 1982), it is not expected that a substantial increase in matrix solubility will be observed.

Forsyth et al. (1985) have studied the effect of the intensity of the alpha radiation field on the dissolution rate of spent fuel. It is possible that the dissolution rate of uranium dioxide (UO_2) could be affected by changes in the redox state of the leachant due to alpha radiolytic decomposition of the water in contact with the fuel. They conducted leaching experiments using low-burnup (≈0.5 MWd/kgU) fuel that had a fission product activity similar in size (though different in composition) to that of fuel with more typical burnup. The inventory of alpha emitters was much less, however. The dissolution behavior of this fuel was significantly different from that of higher burnup fuel, but the observations could not be attributed to the difference in the alpha field. Rather, they believe that differences in microstructure and composition between the types of fuel are the reasons for the observed difference in leaching behavior.

The most detailed studies concerning oxidative dissolution of unirradiated UO_2 have been done as part of the Canadian program. They have used natural uranium in the form of sintered UO_2, which is the fuel material for the heavy-water CANDU reactors. Enriched uranium fuel pellets of the type used in light-water reactors generally have a lower density than the unenriched pellets (see Table 7-14 for comparison). This may affect the depth of penetration of surface oxidation and the surface area readily available for oxidation dissolution but should not alter the mechanism significantly.

Sunder et al. (1981) and Shoesmith et al. (1983) report the results of electrochemical studies using an electrode fashioned from fuel pellet material. They used potentiostatic and cyclic voltametric techniques to determine the mechanism for oxidation in solution of the surface of the uranium dioxide electrode. Sunder et al. (1981) used dilute Na_2SO_4 solutions in the pH range 6 to 11, and Shoesmith et al. (1983) used aqueous carbonate solutions with a range of 0.0001 to 0.5 mole/L total carbonate. The mechanism of dissolution has been found to be different in the two systems because of the ability of the carbonate ion to complex hexavalent uranium.

In the carbonate solutions, dissolution occurred from a layer of U_3O_7 at potentials of 0.1 V or less and from a layer of U_2O_5 at applied potentials of 0.15 and 0.2 V for solutions where the carbonate content was less than 0.001 mole/L (60 ppm). As the carbonate concentration was increased, dissolution of the electrode was enhanced and oxide film formation on the surface became less important. For total carbonate concentrations greater than 0.01 mole/L (600 ppm carbonate), the oxidation of the layer did not progress beyond U_3O_7 because uranyl ions were complexed by carbonate and taken into solution, thereby being unavailable for incorporation into the oxide layer for the next step in film formation. Thus, the mechanism of dissolution has been shown to be a function of both the potential applied to the electrode and the carbonate content of the system. It is probable that for the NNWSI Project tuff repository conditions, the surface from which dissolution occurs would be composed of U_3O_7 because of the combination of mildly oxidizing conditions and moderate bicarbonate concentrations.

Wang (1981) and Wang and Katayama (1982) reported the results of studies using single crystals of UO_2. They proposed a different mechanism than did the Canadians; however, their studies are much less detailed than those discussed above, and their results are correspondingly less certain. Experiments were conducted at 75 and 150°C in systems that were pressurized to give 200 ppm oxygen dissolved in the solutions. This level of dissolved oxygen is very high and may affect the oxidation mechanism for the samples and the nature of the alteration products. The studies done by the Canadians are more relevant to the NNWSI Project tuff repository conditions than is the work referred to in this paragraph.

In summary, the results of the several studies on the oxidative dissolution of UO_2 indicate that the mechanism by which uranium is liberated to solution is a multistage process that depends on both the chemistry of the solution (particularly the thermodynamic activity of CO_2) and the Eh or oxygen fugacity of the system. The effects of raising the Eh and of raising the activity of CO_2 oppose each other. At low activities of CO_2, the UO_2 surface oxidizes to U_3O_7, U_2O_5, and UO_3, with the higher oxides only forming at high Eh (high activity of O_2). Uranium is removed to solution from this oxide surface. At a constant Eh, as the activity of CO_2 increases to the levels present in ground water at the NNWSI Project candidate repository, the oxidation of the surface does not progress beyond U_3O_7. At these activities of CO_2, uranium appears to be removed directly from this surface, with no role played by higher uranium oxides.

Plans to develop the thermodynamic data base and additional information to support the fuel dissolution model needed for geochemical code development are discussed in Section 8.3.5.10. The latter includes studies to identify fuel-surface, alteration products and secondary phases formed by precipitation from solution and experiments to ascertain the effect of the initial oxidation state of the fuel matrix on release rates. The radioactivity present in the spent fuel interferes with most available surface analysis techniques. Because of the analytical difficulties involved, current program planning is being based on the assumption that the work of others on the dissolution mechanism of unirradiated UO_2 will be relevant to spent fuel and that the work on actual spent fuel can be limited to confirmatory studies.

7.4.3.1.2 Oxidation of spent fuel in air

Most spent fuel rods emplaced into a repository will consist of fractured UO_2 pellets enclosed by intact Zircaloy cladding. A small fraction of the rods (less than 1 percent) will have cladding defects, typically in the form of small splits or pinholes (Woodley, 1983). The fuel pellet retards the release of radionuclides when the leach rate is low, and the cladding provides an additional barrier that limits the ingress of water to the fuel. In the moist air atmosphere of the NNWSI Project candidate repository, the UO_2 fuel in a rod with defected cladding in a failed container will be oxidized to UO_3, given sufficient time. When it oxidizes, the UO_2 passes through several intermediate, metastable phases such as U_4O_9, U_3O_7, and U_3O_8. There is a large increase in the molar volume associated with the transformation to U_3O_8, and, if this oxidation state is reached, the fuel pellets swell and put a tensile hoop stress on the cladding, enlarging existing breaches and in some instances creating new ones (Novak and Hastings, 1983; Johnson et al., 1984; Einziger and Cook, 1984). Thus, if there is significant oxidation of the fuel, the ability of the waste form to retard release of radionuclides may be degraded, both by the disruption of the cladding and by the formation of uranium oxides with potentially higher leach rates than UO_2.

Most of the available data on the oxidation of UO_2 were obtained above 200°C and then extrapolated to lower temperatures. The data typically use the time to spallation as a measure of the time required for the onset of U_3O_8 formation; the data provide little insight into the rate of formation of intermediate oxides or which oxides form. Under conditions expected in the NNWSI Project candidate repository, the fuel temperature will be between 160 and 110°C in a container that fails between 300 and 1,000 yr. A time-dependent extrapolation of high temperature data (Einziger and Strain, 1984) indicates that insufficient oxidation of the fuel to U_3O_8 will occur to cause additional failure of the cladding. This calculation assumes that there are no operant oxidation mechanisms at low temperatures that are insignificant at the higher temperatures at which the data were obtained but which become dominant at the temperatures present in the repository.

The results of previous studies of spent fuel and UO_2 oxidation clearly indicated that there are substantial differences in the oxidation kinetics of various fuel types and especially between irradiated and nonirradiated fuel. Consequently, a program of spent fuel oxidation studies has been initiated to investigate oxidation under the conditions identified by Einziger and Woodley (1985a) as the most relevant to the NNWSI Project disposal conditions. The program goal is to obtain oxidation rate and mechanism data on spent fuel samples at temperatures as low as achievable on reasonable laboratory time scales. The work combines thermogravimetric analysis (TGA) of spent fuel oxidation with more conventional oven oxidation work on larger samples. Some of the larger samples will then be used in fuel leaching studies since the oxidation state of the surface of the fuel may affect its leaching behavior. The results of several runs using the TGA apparatus have been published (Einziger and Woodley, 1985b, 1986) and the work is continuing. The oven oxidation work is in progress and no published data are yet available. The following discussion is based on the existing TGA data.

Small samples (200 mg) of Turkey Point unit 3 PWR spent fuel have been used in NNWSI Project TGA studies at 200 and 225°C (Einziger and Woodley, 1985b). Tests at 140 and 175°C have been completed; the results are shown in Einziger (1986). Full test results will be published in the near future. The analytical system has been shown to have excellent stability over periods of up to 2,100 h with the ability to detect a weight change as low as 10 micrograms and the ability to control temperatures to ±1°C up to 300°C (Einziger and Woodley, 1985b). The test matrix for both the completed and the planned test runs is shown in Table 7-18, and TGA weight gain curves for the tests at 200° and 225°C are shown in Figure 7-24.

The effect of particle size was examined in the 140 to 225°C range by testing a single fragment versus pulverized fuel at the same temperature. Although the pulverized fuel showed a higher initial rate of oxidation due to the larger surface area, at temperatures above approximately 200°C the fuel fragment soon caught up and eventually exceeded the weight gain recorded by the pulverized fuel. At temperatures above 200°C, the particle size does not appear to influence the long-term oxidation rate (Einziger and Woodley, 1985b). In contrast, the data at 140°C indicate that surface area may be important at lower temperatures (Einziger and Woodley, 1986).

The results to date support a two-stage mechanism for the oxidation of spent fuel in air: (1) diffusion of oxygen along grain boundaries and oxidation of the grain surfaces and (2) bulk diffusion into the grains. At temperatures above approximately 200°C, grain boundary diffusion appears to be relatively rapid, allowing access of oxygen to most of the grains in the test specimen. This effectively increases the surface area available for bulk diffusion so that the initial surface area is relatively unimportant. This period is expressed in the weight gain curves as an early period of rapid weight change. At lower temperatures (140°C), the rate of grain boundary diffusion of oxygen appears to have slowed sufficiently so that the steady-state bulk diffusion stage was not reached during the 2,100-h test.

The effect of moisture content was examined by running tests at the same temperature but with two different levels of water vapor in the air, namely 3 and 16,000 ppm water at 225°C. The oxidation curves were virtually identical (Figure 7-24, curves 4 and 5; Table 7-18), indicating that moisture content does not seem to be an important parameter in the range tested. The data obtained by TGA were in good agreement with previous work on larger fuel samples, indicating that the small sample size is still representative of the bulk fuel (Einziger and Woodley, 1985b). This is fortunate because the continuous weighing characteristic of TGA allows the oxidation process to be followed in detail.

The rate constants obtained from the bulk diffusion portion of the TGA data can be fit to a straight line in an Arrhenius plot to obtain the activation energy of the diffusion-controlled oxidation. If the results from the 200 and 225°C runs are considered alone, they yield an activation energy of approximately 107 kJ/mole. Although there is considerable uncertainty in this determination because it is based on only two points, the value is in reasonably good agreement with literature values that generally range from 85 to 125 kJ/mole (Aronson et al., 1957; Hastings and Novak, 1984, 1986). If all the data from Figure 1 of Einziger (1986) are used, the value would be 171 kJ/mole, which is higher than that found in previous work. This value may be incorrect if the effects of grain boundary diffusion have not been

Table 7-18. Fuel oxidation test parameters for spent fuel thermogravimetic analyses[a,b]

	Test 1	Test 2	Test 3	Test 4	Test 5
			COMPLETED TESTS		
Temperature (°C)	225	224	200	225	225
Duration (h)	356	408	737	387	438
Atmosphere	air	air	(c)	air	air
Dew point (°C)	14.5	14.5	14.5	-69.8	14.5
Initial weight (mg)	195.2	228.5	227.6	214.5	211.5
Sample condition	Pulverized fuel	Single fragment	Single fragment	Two fragments	Four fragments
			TESTS IN PROGRESS OR PLANNED[d]		
Temperature (°C)	140	140	140	175	175
Duration (h)	2,200	830	300	2,150	2,150
Atmosphere	air	air	(c)	(c)	(c)
Dew point (°C)	14.5	14.5	14.5	-70	14.5
Initial weight (mg)	197.9	184.2	204.2		
Sample condition	Two fragments	Pulverized fuel	Pulverized fuel		
Temperature (°C)	175	155	155	155	
Duration (h)	2,150	2,150	2,150	2,150	
Atmosphere	(c)	(c)	(c)	(c)	
Dew point (°C)	-20	-70	14.5	-20	

[a]All tests carried out using Turkey Point spent fuel (Table 7-14).

[b]Source: Einziger and Woodley (1985b, 1986).

[c]Test conducted in a mixture of 80% N_2 + 20% $^{18}O_2$.

[d]Includes tests at 140 and 175°C that have been completed but for which complete results have not yet been published.

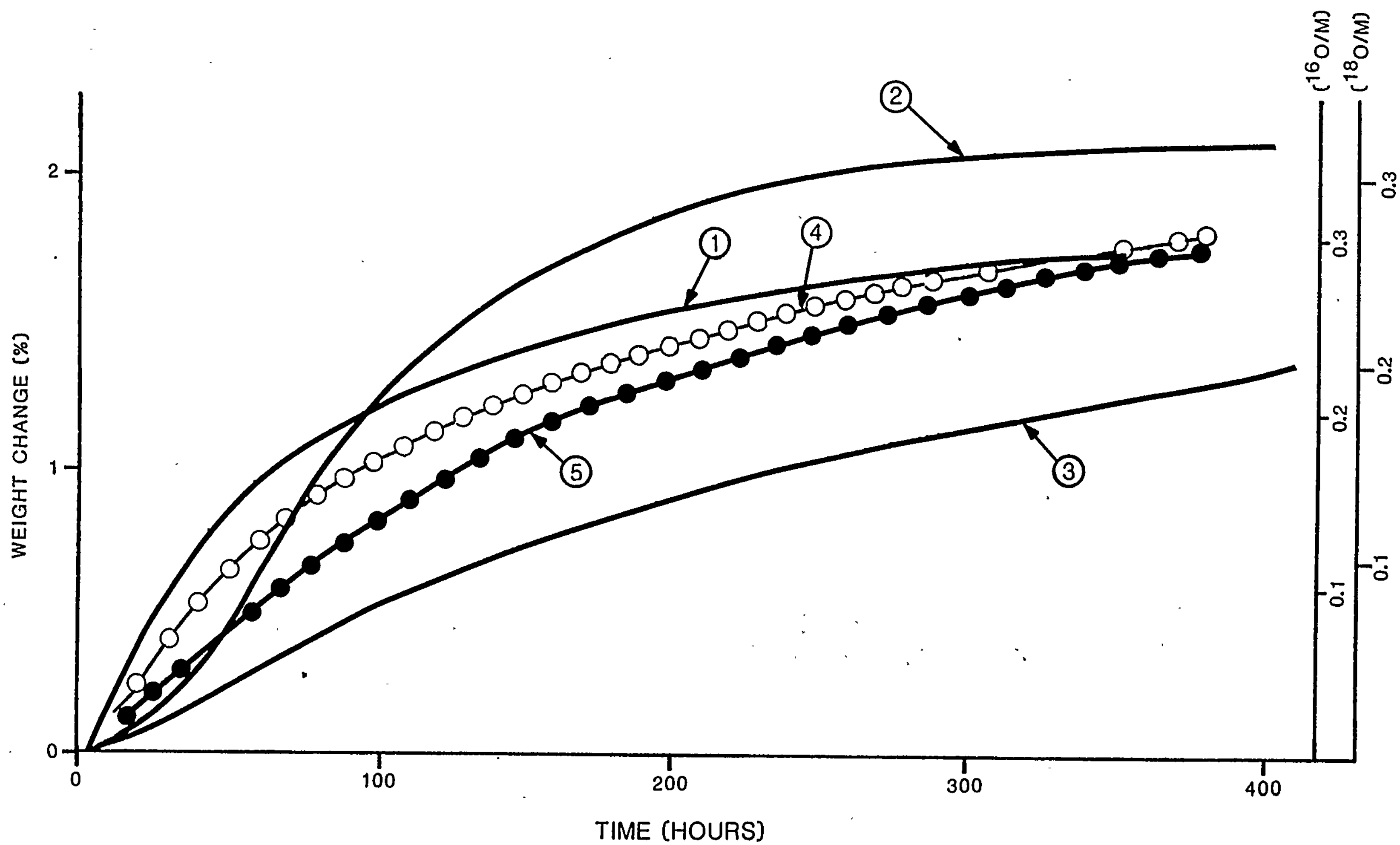


Figure 7-24. Thermogravimetric analysis test sample weight changes (percentage weight gain relative to starting weight). Numbers associated with the curves refer to the test numbers for the series of tests at 200 to 225°C in Table 7-18. Circles are actual data points. Modified from Einziger and Woodley (1985b).

accounted for adequately. The activation energy value will be refined as data from different temperatures are obtained.

A full understanding of the oxidation mechanism of spent fuel will depend on the identification of the phases formed during the process. Characterization of the posttest samples using optical microscopy, x-ray diffraction, electron diffraction, scanning electron microscopy, transmission electron microscopy, and ion microscopy (on samples oxidized in an oxygen-18 atmosphere) is planned (Section 8.3.5.10). Presently, it is not known whether the weight gains recorded in the TGA tests are caused by the formation of U_3O_8, U_3O_7, some other oxide phase, or a mixture of several oxide phases.

Other published work on the oxidation of unclad UO_2 is generally consistent with the data obtained from the NNWSI Project work. Hastings and Novak (1984, 1986) investigated the rate of oxidation of nonirradiated and irradiated CANDU pellet material in the range 175 to 400°C and found activation energies of 140 and 120 kJ/mole, respectively, for the two materials. Studies using spent fuel from the Bruce and Pickering reactors showed that the spent fuel oxidized 5 to 10 times faster than UO_2 at 215°C. They also presented data that suggested a correlation of decreasing fragment size with increasing oxidation rate; however, the scatter in the data is too large for the result to be conclusive.

White et al. (1984), from studies of nonirradiated UO_2 pellets, argue for a two-stage oxidation process with activation energy of 102 kJ/mol for the prespallation stage (corresponding to initial production of U_3O_8) and of 160 kJ/mole for the postspallation stage. There is considerable scatter in the data, and the argument for two stages is far from conclusive. If the activation energy were calculated as a single process, the result would be in reasonable agreement with that of Hastings and Novak (1984, 1986). They also tested Point Beach PWR spent fuel in gamma radiation fields of two intensities and found that the oxidation rate in a 10^5 rads/h field was less than that in a 10^3 rads/h field. This result is yet to be explained.

A large percentage of the work on oxidation of UO_2 has been done using clad fuel with induced cladding defects to study the rupture behavior of the cladding upon formation of U_3O_8. Johnson et al. (1984) have investigated the behavior of breached BWR fuel rods, which had been water logged before the start of the experiment, in air and argon atmospheres at 320°C for periods up to 2,100 h. This temperature is close to the design limit for fuel rod surface temperatures in the present NNWSI Project spent fuel container designs (350°C). They found no detectable change in the fuel rod held in the inert gas atmosphere, despite the presence of water in the system. On the other hand, the rod that was held in air oxidized extensively in the area near the original breach, which caused the cladding to split open. Before the start of the test, a drillhole was drilled near the plenum to allow moisture to escape from the rod. Air gained access to the top fuel pellet through that drillhole and oxidized the pellet, and the previously unbreached cladding split under the stress of the fuel expansion caused by oxidation.

Einziger and Cook (1984) also investigated the behavior of fuel rods in air and inert atmospheres. They used argon with 1 percent helium, a temperature of 229°C, and a test time of 5,962 h. They compared the behavior of intact rods with those in which 0.8-mm-diameter drillholes had been drilled at two locations in the rod. One drillhole was drilled near the center of the rod, while the other was located near the top. No changes were found in any of the intact rods as a result of the test, nor were there any changes in the defected PWR rod in argon or in air. The BWR rod stored in air developed a split in the cladding at the location of the top defect, which was located at a pellet-pellet interface. The splitting was due to oxidation of the fuel in the vicinity of the defect, with the final product being U_3O_8. The center defect, located in the middle of a pellet, showed the start of oxidation from U_4O_9 to U_3O_8, but the degree of oxidation and stress on the cladding was not sufficient to result in splitting.

The difference in oxidation for the two defects in the BWR rod and the absence of any indication of extensive oxidation for the PWR rod held in air were interpreted by Einziger and Cook (1984) to be due to the location of the defects. The upper BWR defect was thought to allow greater access of air to the interior of the rod due to the location of the defect at a pellet-pellet interface, allowing more extensive oxidation to occur. As oxidation progressed and the fuel swelled, the access of air was impeded until the stress was large enough to split the cladding. For the central defect, located in the middle of a pellet, oxidation probably closed the space available for air to enter the rod, thus slowing the rate of oxidation. Because of the relatively low temperature at which the oxidation and cladding split occurred, Einziger and Cook (1984) recommended that dry storage of spent fuel use an inert atmosphere. That recommendation for the atmosphere of disposal containers has been adopted by the NNWSI Project to help preserve the integrity of the cladding during the early high temperature phase in the repository.

Einziger and Cook (1984) used their data to estimate the velocity of the oxidation front inside the breached fuel rod. Their results are in good agreement with the early data on the oxidation of irradiated CANDU fuel. At the velocity calculated (2 to 3 x 10^{-5} cm/min), the fuel cladding would split from end to end in 17 yr at 230°C.

Novak et al. (1983) reported results of testing of irradiated CANDU fuel elements into which drillholes have been drilled. The specimens had either one centrally located 0.8-mm diameter hole or 4 drillholes clustered near the center and one hole near each end of the element; control samples with no defects were also tested. They found cladding splitting in the case of the cluster of 4 defects for the element, with 6 total defects tested at 250°C, but no splitting at the site of the single defects in that element. The splitting, which was inferred to be due to U_3O_8 production, occurred after 208 h at that temperature. The single defect at this temperature showed 2 percent strain at the defect site but no splitting. The control sample with no defect was unchanged.

At 230°C, no cladding split was seen by Novak et al. (1983) for any configuration, and the largest change was 0.7 percent swelling at the site of the cluster of 4 defects after 600 h. The oxidation product was determined to be U_3O_7 by measurement of the O:U ratio. The single defect locations showed lower strain, indicating that access of oxygen to the fuel was controlled by the defect geometry. The extent of oxidation was shown to be limited to the region very close to the defects, and the oxidation front was calculated to move at 2 x 10^{-5} to 3 x 10^{-5} cm/min at 230°C, which is in excellent agreement with the results of Einziger and Cook (1984). Novak et al. (1983) also measured the oxidation rate of defected nonirradiated fuel elements. Their results showed large differences between irradiated and nonirradiated fuel, a phenomenon noted by other workers studying fuel oxidation of pellets or fragments of fuel as well as defected rods (Hastings and Novak, 1984, 1986; White et al., 1984). This phenomenon may be due to the opening of grain boundaries during irradiation, allowing access of oxygen to the interior of the fuel pellets. The weight gains found by Novak et al. (1983) are 2 to 4 times less than those measured by Einziger and Woodley (1985b) for equivalent temperatures and exposure times. This phenomenon may also be a reflection of grain boundary versus bulk diffusion effects.

The short-term oxidation rate of spent fuel at intermediate temperatures of interest to the repository disposal conditions may depend on sample condition because the oxidation process occurs first by diffusion of oxygen along grain boundaries and then by bulk diffusion into the grains. For the time periods of interest in the repository, bulk diffusion should be the rate-limiting step. When measuring oxidation rates in the laboratory, care must be taken to ensure that the period of oxidation is long enough so that the grain boundary diffusion process is complete and the rate measured is that for bulk diffusion of oxygen controlling the oxidation process. Plans for work by the NNWSI Project to determine the rate of oxidation of spent fuel under conditions relevant to a repository at Yucca Mountain are given in Section 8.3.5.10, under Issue 1.5.

7.4.3.1.3 Zircaloy corrosion

Most spent fuel to be placed in a repository is clad with either Zircaloy-4 or Zircaloy-2. A small fraction of the spent fuel is clad in stainless steel (fuel from four reactors). The current NNWSI Project work on cladding involves only Zircaloy. Future work may include tests on stainless steel cladding (Section 8.3.5.10).

When considering the results of spent fuel dissolution studies, it is clear that some elements would be released into water at rates in excess of one part in 100,000 per year if the interiors of all the fuel rods were to be simultaneously contacted with water. The rapid release of cesium and associated elements would be confined to a short period and would involve the gap and grain boundary inventory of those elements. Since it will take many years for water to fill a breached container under the conditions expected at Yucca Mountain, rapid release of the gap and grain boundary inventory to water does not necessarily equate to rapid release from the waste package. If the Zircaloy cladding were to maintain its integrity for long periods of time and if cladding failure after disposal were to occur over an extended period of time rather than as a single catastrophic event, then the release of elements such as cesium would be further controlled. Preservation of cladding integrity would also prevent oxidation of the fuel to states that might be more leachable once water contact occurs.

The potential causes of failure in spent fuel rods that were intact at the time of disposal (or storage, in the case of studies conducted in support of the dry storage program) have been considered by Einziger and Cook (1984), Johnson and Gilbert (1984), and Rothman (1984). Spent fuel containers for the Yucca Mountain repository would be filled with an inert gas to prevent the oxidation of fuel contained in failed rods placed in the container. Oxidation of the cladding would occur only after the container was breached, at which time temperatures would be relatively low. The mechanism of oxidation and generalized corrosion of Zircaloy appears to be the same in air, steam, and pure liquid water environments (Rothman, 1984). Predictions of the degree of oxidation, which depend on the model used to extrapolate the available laboratory data, show a range of about a factor of ten in the predicted oxide thickness formed due to isothermal oxidation at 180°C. The greatest thickness predicted was 53 micrometers after 10,000 yr of corrosion, a value that is less than 10 percent of the cladding thickness (Rothman, 1984). Johnson and Gilbert (1984) predicted an increase in thickness of the oxide layer of about 1 micrometer after 40 yr at 250°C compared with the typical thickness of 20 micrometers of oxidation layer found on fuel rods at the end of their reactor service life.

Corrosion of Zircaloy is enhanced by the presence of some elements in solution. Under expected NNWSI Project tuff repository conditions, none of the elements known to cause accelerated corrosion of Zircaloy is present in sufficient concentration to be of concern (Rothman, 1984). Of the species present under expected NNWSI Project conditions, only the fluoride ion appears to have the potential for significantly increasing the corrosion rate of Zircaloy. Even so, an increase in concentration of fluoride by approximately a factor of 10 would be necessary before any increase in corrosion rate would be expected. Nevertheless, because of the limited data base presently available, Rothman (1984) recommended that some work be initiated on the corrosion of Zircaloy in water having various fluoride concentrations. Such work is in the planning stages (Section 8.3.5.10).

Failure of Zircaloy can occur due to stress rupture when internal pressure from the gas inside the fuel rod exceeds the strength of the cladding material. Failure occurs after a period of time under stress and depends on the stress applied, which is a function of temperature. Einziger and Kohli (1984) have analyzed the data available on stress rupture and have shown that storage of fuel rods at temperatures of 305°C for 100 yr would produce a breach due to stress rupture in less than 5 percent of the rods. For a slightly lower temperature of 288°C, an isothermal period of 1,000 yr would be required before rupture would occur in a similar number of rods. The assumptions used in the analysis were extremely conservative. In particular, it was assumed that stress relief due to cladding creep does not occur and that all rods used in experimental testing are among the 5 percent of the population able to withstand rupture. Also, the rods tested at lower temperature were altered to allow overpressurization so that the test data were for pressure levels (and hence stresses) approximately twice the normal value. No test rods failed, and so the analysis is based on the assumption that the rods were just about to fail. These are also very conservative assumptions.

Rothman (1984) extended the stress rupture analysis to include the concept of life fraction. The time-to-failure at a particular temperature was calculated to represent cladding behavior for a time interval at that temperature under disposal conditions. The ratio of the time increment for that temperature to the time-to-failure was determined. The process was repeated in steps to simulate the thermal decay curve for disposal conditions, and all the ratios were summed. If the results give a sum less than one, then failure by that mechanism is predicted not to occur. The analysis showed that a 10,000-yr period under disposal conditions is equal to only 0.1 percent of the time-to-failure by stress rupture under these conditions. Rothman (1984) concluded, therefore, that stress rupture was not a significant failure mode for fuel rods under the Yucca Mountain repository disposal conditions.

Analysis of the potential for failure by delayed hydride cracking showed that most of the data argued against this mechanism (Rothman, 1984). There is some concern that hydrides in the cladding, which are originally oriented in a circumferential direction, might reorient under storage or disposal conditions to a radial orientation. The radial hydrides might then accelerate cracking in the cladding (Einziger and Kohli, 1984). More data are needed on the statistics of crack properties in cladding and on the effects of slow cooling under storage or disposal conditions. Plans for work in the area of hydride reorientation effects are given in Section 8.3.5.10.

Stress corrosion cracking (SCC) is the final mechanism thought to be a potential cause of failure of Zircaloy under repository conditions. In an unbreached rod, SCC should be more likely on the inside than on the outside (Smith, 1985). There are two reasons for this: (1) the hoop stress due to gas pressure in the rod is greater at the inside surface than at the outer surface and (2) the chemical environment within the rod is much harsher (e.g., iodine) than that outside the rod. This is not to imply that the inside features of the fuel rod will not affect the behavior of the cladding exterior with respect to SCC. Indeed, chemical interactions within the rod could play a role in SCC from the exterior by creating inhomogenous stress distributions on the outer surface of the cladding.

Several analyses have been conducted to evaluate the potential for SCC of Zircaloy under repository conditions. Einziger and Cook (1984) concluded that SCC should not be a life-limiting degradation mechanism for fuel rods stored under dry storage or repository conditions at temperatures below 400°C. Miller and Tasooji (1984) agreed with this conclusion based on model calculation of crack propagation for typical fuel rods. They used an incipient crack depth of 20 percent of the cladding thickness and a fission gas release of one percent during reactor service and assumed a prepressurized PWR fuel rod. They found that their model predicted no failure due to SCC at any time for temperatures up to 400°C. They believed that their typical case represented 99 percent of fuel rods that are intact at the time of storage or disposal. For a worst case of 50 percent gas release (a very conservative figure), with the other parameters the same, they predicted that a crack would propagate to failure in 100 yr at 150°C. A somewhat lower, but still very large, gas release of 17 percent produced a predicted time-to-failure of 100 yr at 170°C. This is at the high end of the measured gas release fractions in reactor service for PWR rods.

Miller and Tasooji (1984) also presented an analysis of SCC based on linear elastic fracture mechanics. This analysis found a much higher limiting temperature of 290°C for an initial crack of 10 percent depth to propagate to failure in 100 yr. Below that temperature no failure would occur for cracks of that size, but larger cracks might propagate to failure.

Based on the preceding discussion, two types of information are needed with respect to Zircaloy corrosion. First, some general corrosion data for Zircaloy together with other components of the waste package in well J-13 water are needed to provide a baseline for further studies on the effect of water chemistry (especially fluoride content) on corrosion rates. Initial experiments were started in late 1984; a description of the work is contained in the reports by Smith (1984a, 1985). The preliminary results of this work are given in Smith and Oversby (1985). Initial evaluation of the test samples from the 90°C corrosion scoping experiment indicated no corrosion of Zircaloy-4 at a detection sensitivity of 1 to 2 micrometers of corrosion per year. Specimens that had been run for 2, 6, and 12 months showed no difference in appearance at that sensitivity. Work is planned to increase the sensitivity of these experiments (Section 8.3.5.10). By using STEM, electron diffraction on ion-milled specimens, and Auger surface analysis, corrosion on the scale of hundreds of angstroms can be studied.

Additional corrosion experiments at elevated temperature and pressure (170°C and 120 psia) are in progress (Smith, 1986a); data are not yet available for these experiments.

It has also been decided to investigate stress corrosion cracking using C-ring tests. An experimental apparatus has been built to allow remote testing of defueled cladding. The philosophy and execution of the tests are described in the reports by Smith (1985, 1986b). These tests are in progress and no published data are yet available (Section 8.3.5.10).

Zircaloy is also of concern because it is part of the waste form (i.e., it contains radionuclides whose release must be controlled). The most significant nuclide in Zircaloy cladding is carbon-14. The results of experiments to measure the release of this radionuclide by oxidation of the carbon in the cladding is discussed in Section 7.4.3.1.1.

7.4.3.1.4 Release model for determining the source term for the spent fuel waste form

Data from the experiments described previously and in Section 8.3.5.10 will be used to develop a model for the release of radionuclides from a waste package as a function of time. A distribution function, developed as part of the metal barrier task, will be used to determine the time-to-breach for containers under both anticipated and unanticipated conditions. The paper by Oversby and Wilson (1985) describes the methodology to be used in developing this model. Oversby (1986) examined the radionuclide inventory of the spent fuel waste form in light of EPA and NRC regulations and provides a basis for identifying which radionuclides are of the most concern.

The container breach-time distribution will give the starting point for radionuclide release from the waste package. The first radionuclide released will probably be carbon-14 from the outer oxidized skin of the cladding. This release may not require contact with water if the high temperature data are an indication of the release mechanism at lower temperatures. Following container breach, water may enter the container. The potential geometric configurations for water contact with the fuel rods are discussed in Section 7.4.3.

The source term for failed fuel rods will include five components:

1. Elements whose release is controlled by the matrix dissolution rate.

2. Elements enriched at grain boundaries and available for enhanced release.

3. Elements present in part in the pellet-cladding gap and available for rapid release.

4. Elements contained in stainless steel or in other fuel assembly components.

5. Elements contained in or on the fuel cladding.

The release of phase-segregated elements such as those present in metallic inclusions is accounted for by the first three items listed above.

The dissolution rate of the matrix will be determined using a fuel-to-water ratio that is as realistic as possible. The water volumes used in present testing for the NNWSI Project are approximately 10 times higher than those for a realistic repository ratio. The larger volume is needed to allow sampling of tests without disruption of conditions. As more data are gathered, future tests will involve less frequent sampling and higher fuel-to-water ratios. The overall dissolution rate will need to consider the distribution of the various fuel characteristics that affect the release of radionuclides such as burnup and grain size. To do this, good data on the irradiation histories and as-fabricated characteristics of the fuels to be emplaced into the repository will be needed.

The source term will include an estimate of both the number of rods initially emplaced with defected cladding and the degradation rate of the Zircaloy cladding. Distributing the breach rate of cladding over a range of time has a large effect on controlling the rate of release of elements such as cesium.

The data from each of the components of the source term will then be summed to determine the concentrations of radionuclides in solution within the container as a function of time. This information, combined with the infiltration rates for water, will then provide the maximum volume of water and amount of radionuclide that can be displaced from a breached waste package per year. At this time, it is not planned to use any restriction of flow provided by the breached container to show compliance with the release rate performance objective. The source term developed to describe spent fuel behavior will be used as input to the waste package performance analysis subtask. Plans for development of the source term model and data are given in Section 8.3.5.10.

7.4.3.2 Glass waste form performance research

The goal of the NNWSI Project glass waste form performance research is to determine release rates for waste glasses under repository conditions. To accomplish this, testing is being done under conditions designed to measure glass leaching rates, to observe component interactions, and to determine the controlling factors in glass leaching under tuff repository conditions. In this summary, general results from glass testing, including NNWSI Project testing, are presented. These results are used to define important parameters for the NNWSI Project to study. Examples of some of the important effects observed are given from NNWSI Project studies. Following that, results from recent NNWSI Project testing are presented.

7.4.3.2.1 Glass waste forms and general principles of glass performance

Two glass-based waste forms are being considered for disposal by the NNWSI Project: Savannah River Plant (SRP) defense waste to be processed at the Defense Waste Processing Facility (DWPF) and reprocessed waste from the West Valley Demonstration Project. Any SRP waste will be made into a borosilicate glass based on Savannah River Laboratory (SRL) frit 165 (Baxter, 1983; Oversby, 1984b). The waste composition at West Valley falls approximately within the range of compositions found at Savannah River, and the glass is expected to be similar to DWPF glass (Eisenstatt, 1986; Oversby, 1984b). There is currently no commercial fuel reprocessing plant in the United States and therefore no glass-based waste form of commercial high-level waste (CHLW). The NNWSI Project has used glasses based on the Pacific Northwest Laboratory (PNL) 76-68 formulation (Oversby, 1984b) in studies of the effect of compositional variation on performance. This glass was originally designed as a CHLW glass.

The composition of waste glasses will vary depending on the composition of the waste. The compositions of a SRL frit 165 based glass and a PNL 76-68 glass, both made with simulated (nonradioactive) waste, are shown in Table 7-19. Samples of these glasses were supplied by SRL and PNL, respectively, and both were used in NNWSI Project tests (Section 7.4.3.2.2). Notable differences between the glasses are the higher silica (SiO_2) content of the SRL glass; the higher uranium, transition metal, and rare earth content of the PNL glass; and the absence of Li_2O in the PNL glass. Table 7-20 lists the projected composition of West Valley glass (Eisenstatt, 1986); no glass based on this formulation is yet available to the NNWSI Project for testing. Notable in the West Valley glass is the thorium and phosphorus content.

The radionuclide inventories of the Savannah River and West Valley facilities are summarized in Tables 7-21 and 7-22 (Aines, 1986), with each waste calculated separately so that their individual contributions to the

Table 7-19. Composition of two glasses designed for high-level waste with simulated (nonradioactive) waste components

Oxides	SRL 165[a] (wt %)	UO_2-spiked PNL 76-68[b] (wt %)
SiO_2	54.90 ± 2.1	42.00
Na_2O	11.40	12.70
Fe_2O_3	10.60 ± 0.7	9.56
B_2O_3	7.20	9.00
ZnO	--[c]	5.10
La_2O_3	--	4.10
UO_2	--	3.98
TiO_2	--	3.08
MoO_3	--	1.97
Nd_2O_3	--	1.40
Al_2O_3	5.13 ± 0.18	0.59
Li_2O	5.04	--
MnO_2	2.72 ± 0.13	--
CaO	1.51 ± 0.32	2.37
NiO	0.84 ± 0.24	0.23
MgO	0.72 ± 0.04	--
ZrO_2	0.66 ± 0.07	1.89
CeO_2	0.42	0.94
K_2O	0.14	--
SrO	0.10	0.40
RuO_2	0.036	--
CsO_2	0.0028	1.10
P_2O_5	--	0.80
BaO	--	0.55
Cr_2O_3	--	0.47

[a]A borosilicate glass formulated by Savannah River Laboratory (SRL) for defense high-level waste. Values with errors were measured by Bazan and Rego (1985); others are nominal values supplied by SRL.

[b]A glass formulated by Pacific Northwest Laboratory (PNL) for commercial high-level waste. Source of data: McVay and Robinson (1984).

[c]-- = not analyzed.

Table 7-20. Actual projected composition of West Valley WV 205 glass[a,b]

Oxide	Oxide (wt %)	Oxide	Oxide (wt %)
SiO_2	44.88	BaO	0.05
Fe_2O_3	12.16	CaO	0.60
FeO	--[c]	Cs_2O	0.08
Al_2O_3	2.83	K_2O	3.57
B_2O_3	9.95	MgO	1.30
Na_2O	10.93	SrO	0.03
Li_2O	3.03	P_2O_5	2.51
UO_2	0.56	SO_3	0.22
CoO	0.00	H_2O	--
Cr_2O_3	0.31	ThO_2	3.58
CuO	0.00	CeO_2	0.07
MnO_2	1.31	La_2O_3	0.03
MoO_3	0.02	Nd_2O_3	0.12
NiO	0.34	Pr_6O_{11}	0.03
RuO_2	0.08	SmO_3	0.03
TiO_2	0.98		
Y_2O_3	0.02	Total	99.91
ZnO	0.00		
ZrO_2	0.29		

[a]Source: Eisenstatt (1986).
[b]Includes some radionuclides, but low concentration elements are not included.
[c]-- = no data.

Table 7-21. Important radionuclides in Savannah River Plant waste[a] (page 1 of 2)

Isotope	Half life (yr)	(Assumed) 1990 inventory[b] (Ci)	1,000 yr postclosure inventory[c] (Ci)	NRC release rate limit per year (Ci)	Ratio of release rate to 1 in 100,000 of 1,000 yr inventory[d]
Ni-59[e]	76,000	1.5E+04[f]	1.5E+04	1.52E-01	1.00
Ni-63	100	1.9E+06	1.2E+03	1.23E-02	1.00
Se-79	65,000	8.0E+02	7.9E+02	7.86E-03	1.00
Rb-87	4.89E+10	5.3E-02	5.3E-02	2.47E-03	4,600
Sr-90	29	3.4E+08	3.4E-03	2.47E-03	73,000
Y-90	0.0073	3.4E+08	3.4E-03	2.47E-03	72,000
Zr-93	1.5E+06	8.7E+03	8.7E+03	8.70E-02	1.00
Nb-93m	13.6	8.7E+03	8.7E+03	8.70E-02	1.00
Nb-94m	20,000	7.2E+00	7.0E+00	2.47E-03	35.4
Tc-99	2.13E+05	1.5E+04	1.4E+04	1.45E-01	1.00
Pd-107	6.5E+06	7.2E+01	7.2E+01	2.47E-03	3.4
Cd-113	9.5E+15	6.8E-11	6.8E-11	2.47E-03	**[g]
Sn-121m	55	2.3E+02	3.6E-04	2.47E-03	**
Sn-126	100,000	1.2E+03	1.2E+03	1.15E-02	1.00
Sb-126	0.034	1.2E+03	1.2E+03	1.15E-02	1.00
Sb-126m	0.001	1.2E+03	1.2E+03	1.15E-02	1.00
Cs-135	3.0E+06	6.5E+02	6.5E+02	6.52E-03	1.00
Cs-137	30.17	2.8E+08	7.5E-03	2.47E-03	32,000
Ba-137m	0.001	2.8E+08	7.5E-03	2.47E-03	32,000
Sm-151	90	1.9E+06	5.4E+02	5.37E-03	1.00
Pb-210	22.3	0.0E+00	8.2E+00	2.47E-03	0.63
Ra-226	1600	0.0E+00	1.1E+01	2.47E-03	0.67 ##[h]
Ra-228	5.76	0.0E+00	5.3E-06	2.47E-03	** ##
Ac-227	21.773	0.0E+00	2.0E-01	2.47E-03	98.4 ##
Th-229	7,300	0.0E+00	1.3E-02	2.47E-03	3,000 ##
Th-230	75,400	0.0E+00	6.8E+01	2.47E-03	0.43 ##
Th-232	1.4E+10	0.0E+00	1.4E-05	2.47E-03	** ##
Pa-231	32,800	0.0E+00	2.6E-01	2.47E-03	106.00 ##
U-232	70	1.1E+03	2.8E-02	2.47E-03	8,700
U-233	1.59E+05	1.3E-01	1.3E-01	2.47E-03	1,800
U-234	2.45E+05	3.6E+03	6.4E+03	6.35E-02	1.00
U-235	7.04E+08	1.2E+01	1.2E+01	2.47E-03	20.79
U-236	2.34E+07	2.6E+02	2.6E+02	2.58E-03	0.97 ##
U-238	4.47E+09	6.6E+01	6.6E+01	2.47E-03	0.75
Np-237	2.14E+06	6.8E+01	1.3E+02	2.47E-03	1.71 ##
Pu-238	87.74	7.9E+06	1.7E+03	1.68E-02	.00
Pu-239	24,110	7.4E+04	7.2E+04	7.20E-01	1.00
Pu-240	6,560	4.7E+04	4.2E+04	4.20E-01	1.00
Pu-241	14.35	8.9E+06	3.3E-02	2.47E-03	7,500
Pu-242	3.76E+05	6.5E+01	6.5E+01	2.47E-03	3.80
Am-241	432	8.3E+04	7.0E+04	7.02E-01	1.00

Table 7-21. Important radionuclides in Savannah River Plant waste[a] (page 2 of 2)

Isotope	Half life (yr)	(Assumed) 1990 inventory[b] (Ci)	1,000 yr postclosure inventory[c] (Ci)	NRC release rate limit per year (Ci)	Ratio of release rate to 1 in 100,000 of 1,000 yr inventory[d]
Am-242m	141	1.1E+02	5.8E-01	2.47E-03	427.71
Am-243	7,370	4.4E+01	4.0E+01	2.47E-03	6.17
Cm-243	28.5	4.3E+01	2.1E-10	2.47E-03	**
Cm-244	18.11	1.3E+03	2.1E-15	2.47E-03	**
Cm-245	8,500	5.1E-02	4.7E-02	2.47E-03	52,62.36
Cm-246	4,780	4.1E-03	3.5E-03	2.47E-03	70,459.20
Totals		1.26E+09	2.47E+05		

[a]Source: Baxter (1983); Aines (1986).
[b]An average age for waste in 1990 assumed.
[c]Assumed repository closure in 2050.
[d]For radionuclides that can be released at 0.1 percent of total release rate limit ($2.47x10^{-3}$ Ci/yr), the column summarizes the extent to which the radionuclides can be released faster than 1 part in 100,000.
[e]Radionuclides with a release that must be controlled at 1 part in 100,000 of their own 1,000-yr-postclosure inventory are underlined.
[f]E indicates exponential notation.
[g]** = Ratio exceeds 100,000. The entire inventory could be released in one year and meet the regulation.
[h]## = Grow-in affects this ratio; values shown here are from Table 7-23 later in this section.

Table 7-22. Important radionuclides in West Valley waste[a] (page 1 of 2)

Isotope	Half life (yr)	1987 inventory[b] (Ci)	1,000 yr postclosure inventory[c] (Ci)	NRC release rate limit per year (Ci)	Ratio of release rate to 1 in 100,000 of 1,000 yr inventory[d]	
C-14[e]	5,730	1.4E+02[f]	1.22E+02	1.22E-03	1.00	
Ni-59	76,000	8.2E+01	8.12E+01	8.12E-04	1.00	
Ni-63	100	6.4E+03	4.04E+00	2.13E-04	5.28	
Se-79	65,000	3.7E+01	3.66E+01	3.66E-04	1.00	
Sr-90	29	7.4E+06	6.87E-05	2.13E-04	**[g]	
Y-90	0.0073	7.4E+06	6.87E-05	2.13E-04	**	
Zr-93	1.5E+06	2.3E+02	2.30E+02	2.30E-03	1.00	
Nb-93m	13.6	2.3E+02	2.30E+02	2.30E-03	1.00	
Tc-99	2.13E+05	1.6E+03	1.59E+03	1.59E-02	1.00	
Pd-107	6.5E+06	1.2E+00	1.20E+00	2.13E-04	17.7	
Sn-126	1.0E+05	4.0E+01	3.97E+01	3.97E-04	1.00	
Sb-126m	0.001	4.0E+01	3.97E+01	3.97E-04	1.00	
Sb-126	0.034	5.6E+01	3.97E+01	3.97E-04	1.00	
I-129	1.6E+07	3.6E-01	3.60E-01	2.13E-04	59.2	
Cs-135	3.0E+06	1.6E+02	1.60E+02	1.60E-03	1.00	
Cs-137	30.17	7.8E+06	1.94E-04	2.13E-04	**	
Ba-137m	0.001	7.3E+06	1.94E-04	2.13E-04	**	
Sm-151	90	2.1E+05	5.85E+01	5.85E-04	1.00	
Pb-210	22.3	0.0E+00	1.10E-02	2.13E-04	40.8	##[h]
Ra-226	1,600	0.0E+00	1.27E-02	2.13E-04	43.9	##
Ra-228	5.76	0.0E+00	1.17E+00	2.13E-04	10.0	##
Ac-227	21.773	0.0E+00	2.00E-03	2.13E-04	956	##
Th-229	7,300	0.0E+00	9.55E-01	2.13E-04	3.5	##
Th-230	75,400	0.0E+00	6.65E-02	2.13E-04	34.3	##
Th-232	1.4E+10	1.6E+00	1.60E+00	2.13E-04	13.34	
Pa-231	32,800	0.0E+00	2.23E-03	2.13E-04	1,000	##
U-233	1.59E+05	1.0E+01	9.95E+00	2.13E-04	2.1	
U-234	2.45E+05	4.6E+00	7.16E+00	2.13E-04	2.98	##
U-235	7.04E+08	1.0E-01	1.02E-01	2.13E-04	200	
U-236	2.34E+07	3.0E-01	3.40E-01	2.13E-04	62.70	
U-238	4.47E+09	8.5E-01	8.50E-01	2.13E-04	25.10	
Np-237	2.14E+06	1.1E+01	2.33E+01	2.33E-04	0.89	##
Pu-238	87.74	7.2E+03	1.60E+00	2.13E-04	13.33	
Pu-239	24,110	1.7E+03	1.65E+03	1.65E-02	1.00	
Pu-240	6,560	1.3E+03	1.22E+03	1.22E-02	1.00	
Pu-241	14.35	8.7E+04	8.12E+00	2.13E-04	2.63	
Pu-242	3.76E+05	1.7E+00	1.70E+00	2.13E-04	12.52	
Am-241	432	7.2E+04	1.36E+04	1.36E-01	1.00	
Am-242m	141	2.1E+01	1.12E-01	2.13E-04	190.6	

Table 7-22. Important radionuclides in West Valley waste[a] (page 2 of 2)

Isotope	Half life (yr)	1987 inventory[b] (Ci)	1,000 yr postclosure inventory[c] (Ci)	NRC release rate limit per year (Ci)	Ratio of release rate to 1 in 100,000 of 1,000 yr inventory[d]
Am-243	7,370	2.4E+03	2.17E+03	2.17E-02	1.00
Cm-243	28.5	1.7E+02	9.63E-10	2.13E-04	**
Cm-244	18.11	2.2E+04	4.40E-14	2.13E-04	**
Cm-245	8,500	1.0E+01	9.17E+00	2.13E-04	2.33
Cm-246	4,780	4.3E+00	3.68E+00	2.13E-04	5.79
Totals		3.03E+07	2.13E+04		

[a]Source: Baxter (1983); Aines (1986).
[b]From actual analysis of tanks by Eisenstatt (1986).
[c]Assumed repository closure in 2050.
[d]For radionuclides that can be released at 0.1 percent of total release rate limit (2.13 x 10^{-4} Ci/yr), this column summarizes the extent to which the radionuclides can be released faster than 1 part in 100,000.
[e]Radionuclides with a release that must be controlled at 1 part in 100,000 of their own 1,000-yr postclosure inventory are underlined.
[f]E indicates exponential notation.
[g]** = Ratio exceeds 100,000 for this isotope. The entire inventory could be released in one year and meet the regulation.
[h]## = grow-in affects this ratio; values shown here are from Table 7-24 later in this section.

repository release limits can be calculated and the relative importance of the radionuclides can be assessed. Table 7-21 is based on the projected operation of the DWPF. No actual analyses or estimates of the radionuclide content of the waste tanks at Savannah River are currently available. Aines (1986) estimated the inventories by assuming that 7,000 canisters of the reference waste described in Baxter (1983) would be produced at the DWPF and that the waste from which that glass will be produced would be present, and all of the same age, in 1990. This is equivalent to giving the waste one average age. Better estimates of the radionuclide inventory at Savannah River cannot be made until analyses of the existing waste are available. Currently available analyses, such as those given by Manaktala (1982) only list the radionuclides with the highest activity at the present time. These are all short-lived, and most of the radionuclides of interest at long times are not given. Table 7-22 is based on actual samples of the West Valley tanks (Eisenstatt, 1986). The contents of the tanks may be inhomogeneous, causing these analyses not to represent precisely the bulk average; but, the inventories obtained by analysis are consistent with the operating history of the facility (Eisenstatt, 1986).

Tables 7-21 and 7-22 give the calculated inventories (Aines, 1986) at 1,000 yr postclosure, assuming repository closure in year 2050. Full decay and grow-in were considered for all radionuclides. The release rate limit from the engineered barrier system is calculated separately for each waste type from 1 part in 100,000 of the individual 1,000-yr inventory, or as 1 part in 100,000,000 of the total 1,000-yr inventory. Both values are calculated from the assumed closure time since the emplacement times required to calculate the total release rate limit are not yet known. All radionuclides with a release that must be controlled at 1 part in 100,000 of their own 1,000-yr-postclosure inventory are underlined in the tables. Most short-lived (less than 1 yr) daughter nuclides are not significant to release rate limits and are not shown. All radionuclides reported by the waste producers and whose inventory is great enough that it could not be released in a single year are included.

The final column in Tables 7-21 and 7-22 gives the ratio of the release rate limit to 1 part in 100,000 of the inventory of the individual nuclides at 1,000 yr postclosure. For the radionuclides that can be released at 0.1 percent of the total release rate limit (given in each table), the final column summarizes the extent to which these nuclides could be released at a rate faster than 1 part in 100,000 of the total waste. This can be used to eliminate many radionuclides from further consideration; any radionuclide with a value greater than 100,000 in this column could have its entire inventory released in 1 yr. Most of these radionuclides have been excluded from the table, but radionuclides with large initial inventories and those involved in actinide decay chains are included even if the ratio in the last column exceeds 100,000. These radionuclides have a double asterisk in the final column to indicate that their release does not need to be controlled.

Several radionuclides in Savannah River waste grow in to significantly higher levels at 10,000 yr such that release of 1 part in 100,000 of the 10,000 yr inventory (beginning at 1,000 yr) would slightly exceed the allowed release. The radionuclides that grow in significantly and their 10,000-yr total inventories are shown in Tables 7-23 and 7-24. Radionuclides with a

Table 7-23. Radionuclides that grow-in significantly in Savannah River Plant-Defense Waste Processing Facility waste glass[a,b]

Isotope	Half-life (yr)	1,000-yr postclosure inventory (Ci)	NRC release rate limit per year (Ci)	10,000-yr inventory grow-in (Ci)	Ratio of release rate limit to 1 in 10^5 of 10,000-yr total inventory
Pb-210[c]	22.3	8.2E+00[d]	2.47E-03	4.6E+02	0.53
Ra-226	1600	1.1E+01	2.47E-03	4.3E+02	0.57
Ra-228	5.76	5.3E-06	2.47E-03	1.7E-04	**[e]
Ac-227	21.773	2.0E-01	2.47E-03	2.5E+00	98.40
Th-229	7,300	1.3E-02	2.47E-03	8.0E-02	3,064.18
Th-230	75,400	5.8E+01	2.47E-03	5.5E+02	0.45
Th-232	1.400E+10	1.4E-05	2.47E-03	1.3E-04	**
Pa-231	32,800	2.6E-01	2.47E-03	2.3E+00	106.11
U-236	2.34E+07	2.6E+02	2.58E-03	2.7E+02	0.97
Np-237	2.14E+06	1.3E+02	2.47E-03	1.4E+02	1.71
Total		2.47E+05			

[a]Source: Baxter, 1983; Aines, 1986.

[b]Assumed repository closure in year 2050, 0.1 percent of total release rate, 2.47E-03 Ci/year.

[c]Radionuclides whose release must be controlled at 1 part in 100,000 of their own 1,000-yr-postclosure inventory are underlined.

[d]E indicates exponential notation.

[e]** = Ratio exceeds 100,000.

release that must be controlled at 1 part in 100,000 of their own 1,000-yr-postclosure inventory are underlined in the tables. The 10,000-yr post-closure inventory is calculated assuming no release over that period. The NRC release rate limit is then compared with the release of 1 part in 100,000 from the 10,000-yr inventory. Under this assumption, the ratio of allowed release to 1 part in 100,000 (the last column in Tables 7-21 to 7-24) can drop below 1.00. For example, at 1,000 yr, postclosure release of lead-210 at 1 part in 100,000 of its inventory would be below the release rate limit by a factor of 30; but if all lead-210 were retained in the waste until 10,000 yr and then released at 1 part in 100,000 per year of its own inventory, it would exceed the allowed release rate limit by a factor of 2. This is not a physically realistic case, since the grown-in radionuclide's inventory could not actually increase to the point where 1 part in 100,000 would exceed the permitted rate until close to 10,000 yr after closure. The average release over that period would be much lower.

Table 7-24. Radionuclides that grow-in significantly in West Valley waste glass[a,b]

Isotope	Half-life (yr)	1,000-yr postclosure inventory (Ci)	NRC rate release limit per year	10,000-yr inventory grow-in (Ci)	Ratio of release rate limit to 1 in 10^5 of 10,000-yr total inventory
Pb-210	22.3	1.10E-02[c]	2.13E-04	5.22E-01	40.88
Ra-226	1,600	1.27E-02	2.13E-04	4.86E-01	43.90
Ra-226	5.76	1.17E+00	2.13E-04	213E+00	10.02
Ac-227	21.773	2.00E-03	2.13E-04	2.23E-02	956.85
Th-229	7,300	9.55E-01	2.13E-04	6.00E+00	3.56
Th-230	75,400	6.65E-02	2.13E-04	6.22E-01	34.31
Pa-231	32,800	2.23E-03	2.13E-04	2.07E-02	1030.81
U-234	2.45E+05	7.16E+00	2.13E-04	6.98E+00	2.98
Np-237[d]	2.14E+0	2.33E+01	2.33E-04	2.61E+01	0.89
Total		2.13E+04			

[a]Source: Baxter (1983); Aines (1986).
[b]Assumed repository closure in year 2050, 0.1 percent of total release rate, 2.13E-04 Ci/year.
[c]E indicates exponential notation.
[d]Radionuclides whose release must be controlled at 1 part in 100,000 of their own 1,000-yr-postclosure inventory are underlined.

Tables 7-21 to 7-24 do not reflect the actual release rate limits from the entire repository because they consider the glass wastes individually, but they may be used to assess the importance of the various radionuclides. All radionuclides whose release from the engineered barrier system must be controlled at 1 part in 100,000 of their own inventory (underlined in the tables) are of equal importance. Relatively less information is required about the minor radionuclides whose release is controlled at 0.1 percent of the calculated release rate limit; palladium-107, for instance, can be released three times faster than major radionuclides from SRP waste. Matrix dissolution in glass waste forms (discussion follows) will result in an upper limit to the release rate of all radionuclides (the matrix dissolution rate), which must correspond to less than 1 part in 100,000 because of the major radionuclides; therefore, the allowed greater release of palladium-107 has the effect that the relative uncertainty in the release of palladium-107 can be three times greater than that of major radionuclides.

The release rate of glass components during storage or testing of a waste form may be expressed in a number of different ways. The currently accepted standard method is to report normalized elemental leaching (NL_i) for component i (Mendel, 1981; MCC, 1983). This corresponds to the time-integrated bulk leaching that would result if the entire sample were to dissolve at the same rate as the component i and is defined as

$$NL_i = \frac{M_i}{(f_i)(SA)} \tag{7-1}$$

where

M_i = mass of element i released from the waste form (g)
f_i = mass fraction of element i in the original sample
SA = surface area of the sample (m^2).

Alternatively, the leach rate (LR_i) may be defined as

$$LR_i = \frac{NL_i}{t} \tag{7-2}$$

where

t = time in days.

These expressions for leaching are useful because they do not involve the volume of the leachate, as does the concentration of element i. As such, the NL or LR values from laboratory tests may be scaled up to repository values directly; a NL value for a glass sample is the same regardless of the sample size. If a glass were to dissolve completely congruently, all components would have the same NL value, which would be equal to the total weight loss per square meter. The release may be calculated from the amount found in solution but is more meaningful if it includes particulate and adsorbed material on the test vessel and components. Uncertainty in NL values is a function of analytical uncertainty in both the glass analysis and the leachate-test vessel analysis.

The condition of the surface of a glass undergoing leaching can substantially change the observed leach rate simply by affecting the surface area of glass available to react with the leachant. There has been much confusion in the glass-leaching literature about rates and mechanisms as a result of this effect. Much of the initially high leach rate in static tests (Figure 7-25) may be attributed to rough, highly reactive surfaces (Mendel, 1984). This makes short-term laboratory experiments difficult to reproduce (Kingston et al., 1984). At longer times, the initial surface of the glass dissolves away and the results no longer depend strongly on the initial condition but can be affected by cracks. In 165 frit glasses, cracks can be the dominant regions where leaching occurs at short times (Mendel, 1984). Normalized loss data cited here use the geometric area of the sample, which can be measured accurately. The reproducibility of leaching results is

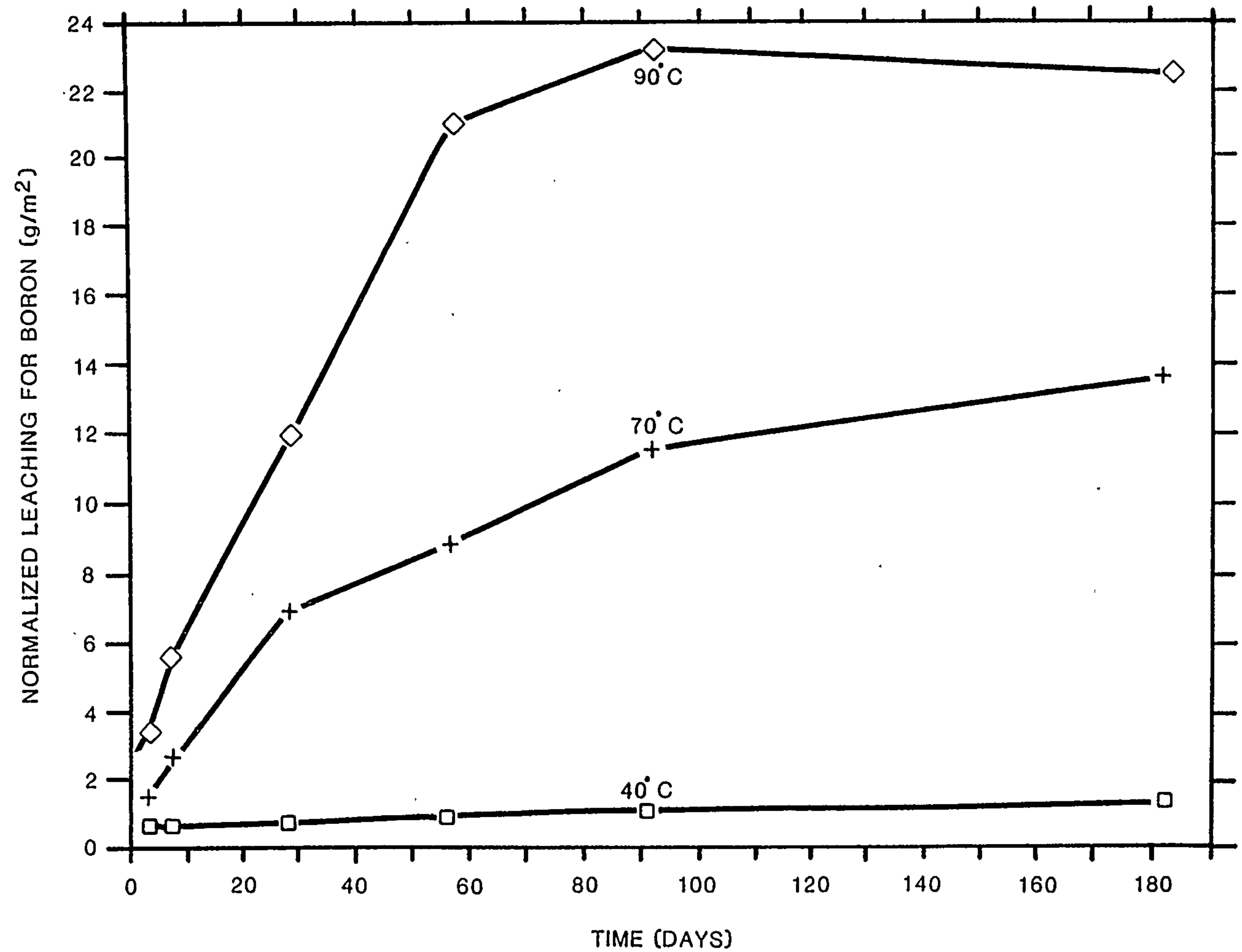


Figure 7-25. Temperature effect on leaching of PNL 76-68 glass. MCC-1 static experiments in well J-13 water. Ratio of surface area to fluid volume (SA:V) = 10 m^{-1}. Replotted data from McVay and Robinson (1984).

discussed by Kingston et al. (1984) and must be considered to be no better than 10 to 20 percent of the observed value due to the complexity of the tests and the compounding of uncertainties in calculating NL values.

The glass surfaces exposed after container and canister breach in a repository will be a combination of fractured and as-cast surfaces. Fracturing could increase the surface area by as much as a factor of 10 (Baxter, 1983), directly resulting in up to a factor of 25 increase in release if water comes in contact with fractured areas. NNWSI Project testing now includes the use of samples with as-cast surfaces and glass-304L SS surfaces formed by casting glass into stainless steel tubes. The extent of fracturing and other surface irregularities in DWPF full-scale canisters has not yet been determined. Tests reported by Bickford and Pellarin (1986) on sections cut from full-scale DWPF-type canisters indicate that the leach rate increased by up to a factor of two over the rate expected from uncracked glass. This experimental result is much less than the factor of 25 increase that might be predicted to result from surface cracking.

The second reporting method of interest is the concentration of leached elements, normally in milligrams per liter of water or milligrams per kilogram of water (ppm). This method is most useful when an element reaches a saturated concentration in the water; however, it does not normally take into account colloidal material although this can be differentiated by filtration. Solution results reported as concentrations have the advantage of not containing the compounded uncertainties inherent in NL calculations, which combine a number of individual measurements.

Current regulations regarding the release of radionuclides from the repository engineered barrier system are written in terms of individual nuclides. Therefore in NNWSI Project testing, an important goal is to determine release rates for the elements in Tables 7-21 through 7-24, which will be the major sources of radioactivity in leachate from 300- to 1,000-yr-old waste. Because of the large number of important radionuclides and their chemical diversity, it is desirable to know an upper bound on the release of any glass component. Extensive research on borosilicate glass leaching mechanisms has shown that such an approach is possible (Mendel, 1984).

The objective of the 3-yr defense high-level waste leaching mechanisms program was "to determine the dominant leaching mechanisms for defense waste glass and to evaluate the effects of some major environmental parameters upon the leaching mechanisms" (Mendel, 1984). The mechanisms involved are discussed below and are still subject to discussion in some aspects. However, it was shown that after an initial brief period of several months (depending on the leaching conditions) no element is released from borosilicate glasses at a rate in excess of the components of the frit that do not precipitate from solution under the test conditions. For PNL 76-68, molybdenum, sodium, and boron are such elements; for SRL 165 frit glass, boron, sodium, and lithium are such elements (Barkatt et al., 1983; Wallace and Wicks, 1983; Mendel, 1984). Their solubilities in water are very high under all reasonable test conditions.

Using this result, a very conservative statement can be formulated concerning the maximum leaching or leaching rate of any radionuclide: the NL_i or LR_i for any element will not exceed that of the most soluble element in the glass, commonly boron (or lithium in the case of SRL glass). Because of its conservatism, this statement should be used only as a guideline, and the NNWSI Project is therefore directly measuring individual radionuclide release values. In NNWSI Project testing, only one instance has arisen of a radionuclide having a larger long-term normalized loss than boron; in leach testing of the PNL 76-68-based glass ATM-8 inside a tuff reaction vessel, Bazan and Rego (1986) found technetium in solution at concentrations indicating that it was released at a normalized rate approximately 10 to 15 percent faster than boron and molybdenum. This may be caused by sorption of boron and molybdenum onto the tuff vessel or may be the result of an inaccurate analysis of the initial composition of the glass. If the Materials Characterization Center (MCC) analysis of the initial composition of ATM-8 is used instead of Bazan and Rego (1986), then boron, molybdenum, and technetium were released at the same rate within error.

To predict confidently the long-term behavior of waste glasses, it is essential to understand the mechanisms of glass leaching. A summary of results of current research and of theory on glass leaching has been prepared by the defense high-level leaching mechanisms program (Mendel, 1984). They determined that the important aspects of the leaching mechanism are the removal of highly soluble elements (e.g., sodium, boron, and lithium) from the glass surface, the buildup of layers of insoluble components (e.g., iron and zinc), and the gradual reprecipitation of slightly soluble components (e.g., calcium and silicon). The net result of these processes can be considered to be that the glass dissolves congruently (all elements simultaneously) but with many elements reprecipitating onto the glass or surrounding material (Strachan et al., 1984; Apted and Adiga, 1985; Barkatt et al., 1985; Grambow, 1985). This makes possible the use of a highly soluble element as an indicator of the maximum possible release.

The factors controlling the leach rate are the rate of removal of soluble species and the buildup of layers limiting that removal. These processes are dependent upon leachant chemistry; temperature; and, in particular, pH. Because the initial removal of soluble species is essentially a congruent process, restricting the solubility of the major glass component, silica, strongly limits the leach rate. The formation of surface layers may also restrict the leach rate. However, depending upon surface layers to protect glass from releasing radionuclides is a questionable practice because the layers may slough off, particularly after drying (Mendel, 1984). The most successful approach to modeling the dissolution rate of glass has been to apply transition state kinetic theory, using the dissolution of silica as the rate-limiting step. This method is described by Grambow (1984), and applications are given in numerous papers (Strachan et al., 1984, 1985; Freude et al., 1985; Grambow, 1985; Grambow et al., 1985). Although the rate of dissolution depends directly only on silica activity in solution in this modeling approach, many other factors affect that activity. The work to date indicates that many of the phenomena observed in glass leaching may be explained by this theory. The more important parameters of those phenomena are described in the following paragraphs.

Leach rates depend strongly on temperature. Results from NNWSI Project testing of PNL 76-68 glass leached at 40, 70, and 90°C in well J-13 water are shown in Figure 7-25 (McVay and Robinson, 1984). These data yield an activation energy of 16 kcal/mole, which is typical for borosilicate waste glasses (Bradley et al., 1983; McVay and Robinson, 1984; Mendel, 1984). This implies that there is an expected rate difference of about two orders of magnitude between glass leaching occurring at temperatures of 40 and 95°C, the range in which liquid water might be expected to contact the waste form under Yucca Mountain conditions (Section 7.4.1.2). Since leach rates are inevitably higher at higher temperatures, NNWSI Project tests are routinely performed at 90°C. Some future testing will also be conducted at 60°C (Section 8.3.5.10).

In glass leaching, the pH of the leaching solution is the most important parameter over which some control may be exercised in repository and waste form design. When the pH is controlled to within the range 5 to 9, leaching is at a minimum. Outside these limits, the leach rate increases dramatically. Figure 7-26 (after Plodinec et al., 1982) shows the relationship between pH and leaching in a previous SRL glass formulation, frit 131. This effect was first predicted by Paul (1977) and has since been found to occur widely in many glasses including nuclear waste glasses (Mendel, 1981; 1984).

The pH effect on leaching is subject to adjustment both by the glass and the leachant. In water in Yucca Mountain, the pH is initially buffered by the presence of dissolved CO_2 and silica. When reaction with glass occurs, the dissolved glass components will also buffer the pH of the water. The balance between acidic oxides (SiO_2 and B_2O_3) and alkaline oxides in the glass is a critical factor in determining the ultimate pH of a solution while it reacts with glass (Mendel, 1984). Proper glass formulation yields slightly more acidic oxides than basic oxides (by mole fraction), and the pH of the leachate remains near neutral and out of the rapid dissolution regime. Both SRL-165 frit-based glass and PNL 76-68 glass remain in the near neutral region during leaching. As was found by the defense high-level waste leaching mechanisms program (Mendel, 1984), however, a small increase in the amount of alkalis and a corresponding decrease in the silica content of a waste glass can lead to a large pH excursion (toward pH >10) and a concom-itant large increase in the leach rate. The buffering capacity of the ground water at Yucca Mountain would not be able to control a large excursion; glass composition must be carefully controlled to keep the pH from rising into the rapid dissolution range.

Self-irradiation of waste glass and irradiation of leaching fluids may affect release rates. However, considerable research on radiation effects on glass leaching has revealed few demonstrable effects, the most notable being acidification of heterogeneous air-water systems by intense gamma-ray fluxes. Alpha-decay damage within waste glasses produces little change in leaching behavior even at doses well in excess of that anticipated in waste glasses; alpha-radiolysis of water and dissolved components may affect leach rates by a factor of two but would be dependent upon repository interactions (Mendel, 1984; Burns et al., 1982). This type of interaction is being tested by the NNWSI Project by leaching radioactive glasses.

The radiolytic production of nitric acid from air in the presence of water can cause substantial effects (Burns et al., 1982). The effect on

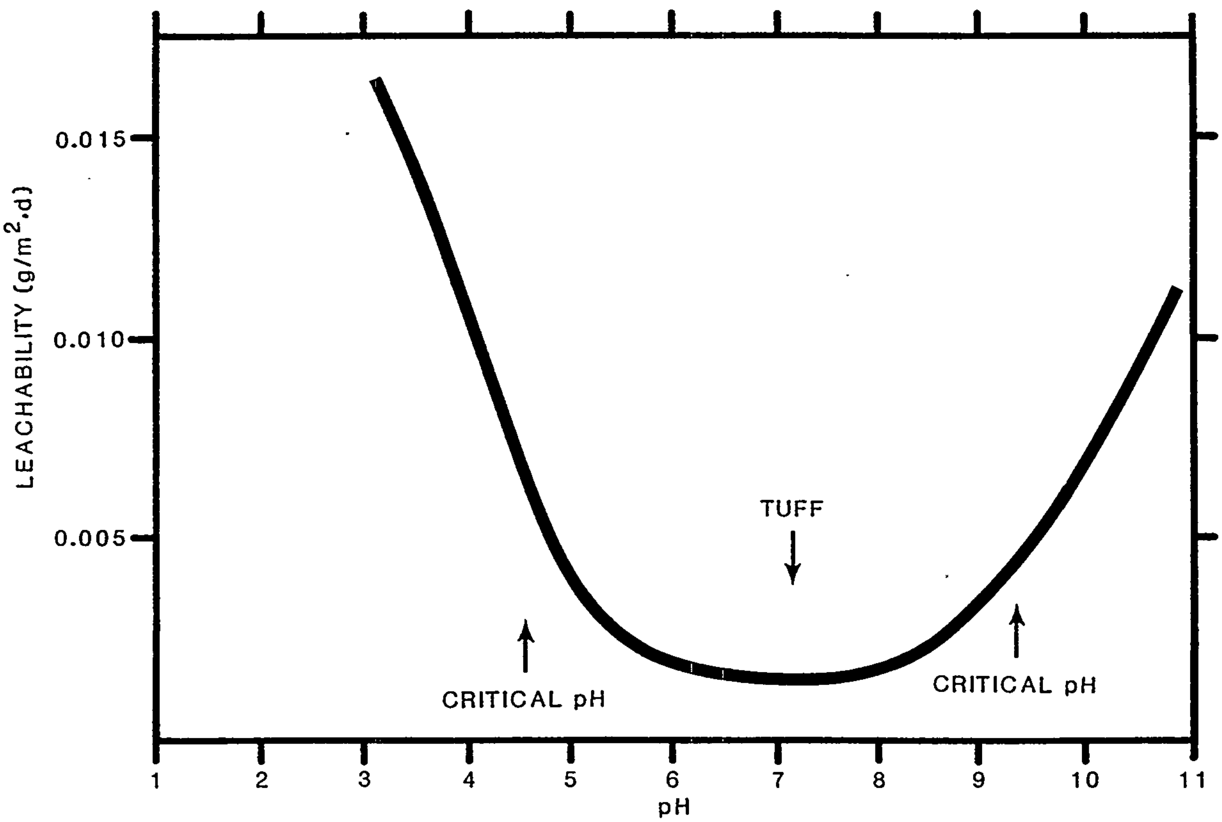


Figure 7-26. Leachability of an SRL-131 frit glass as a function of pH. Five-day static tests. ratio of surface area to fluid volume (SA/V) = 1000 m^{-1}. 23°C. Modified from Plodinec et al. (1982).

glass leaching in a repository is unknown because of uncertainties in water contact mechanisms. The dose rate from a DWPF canister would initially be approximately 2.5 x 10^3 rem/h, but would decrease by more than three orders of magnitude at 300 to 1,000 yr (Baxter, 1983). In the postcontainment period, gamma-radiolysis would not be nearly as important as it would be in the event of a premature container and canister breach; results of NNWSI Project testing in the presence of radiation fields are discussed in Section 7.4.3.2.2.

The discussion has so far been limited to factors that are independent of the amount of water or its contact mechanism with the waste form. The effects of water volume in laboratory leaching experiments are primarily on the rate of leaching. In the presence of a large amount of water, a given glass sample will lose more material than it does in a small volume of water in which the solution becomes saturated. A scaling factor that correlates this behavior over a wide range of circumstances is the ratio of glass surface area (SA) to solution volume (V), SA:V. (References to SA:V in this section are given in units of m^{-1}.) Leach rates are lowest at high SA:V where surface layers form due to precipitation of silicates and transition metal compounds. They are low primarily because the dissolution of silica is inhibited by the rapid increase of silica in a relatively small solution in contact with a large surface area of glass. The release of soluble components (boron, lithium, and radionuclides that are soluble) is inhibited by the depth of undissolved glass matrix through which they would have to diffuse to reach the solution. These elements may initially diffuse out of the glass, giving a release rate higher than that of the overall glass breakdown rate; however, diffusional release eventually slows to less than that of the overall breakdown rate, and all elements are released congruently (although many immediately reprecipitate).

Over a wide range of SA:V ratios and times (t), it has been found that a given NL_i value for a glass is reached at a constant value of (SA:V)(t) (Pederson et al., 1983; Mendel, 1984). This behavior is apparently due largely to the onset of solution saturation, but in some instances it is found in solutions that are apparently saturated with respect to some glass components. The leaching of SRL-165 glass as a function of (SA:V)(t) is shown in Figure 7-27, from Bazan and Rego (1985). Using (SA:V)(t) scaling it is possible to predict the results of long-term experiments by using the results of short-term experiments performed at higher ratios; for instance, 1 yr of leaching at 14 m^{-1} should give the same result as 51 d at 100 m^{-1}. However, the extension of this relationship into the SA:V expected at Yucca Mountain (up to 1,000 m^{-1}) must be demonstrated. This relationship is largely useful to compare leaching data obtained under different conditions.

When materials other than glass and water are added to a leaching system, they participate both by dissolving to some extent and by providing reactive surfaces. Materials that add silica to the water have been shown to be effective in reducing glass dissolution rates by slowing the removal of silica from the glass. However, even in saturated silica solutions, glass continues to dissolve and reprecipitate into more stable compounds (Grambow, 1984). Any material that aids in the formation of these compounds can accelerate leaching. NNWSI Project glass testing (described in Section 7.4.3.2.2) places a strong emphasis on materials interactions. Tests are routinely done

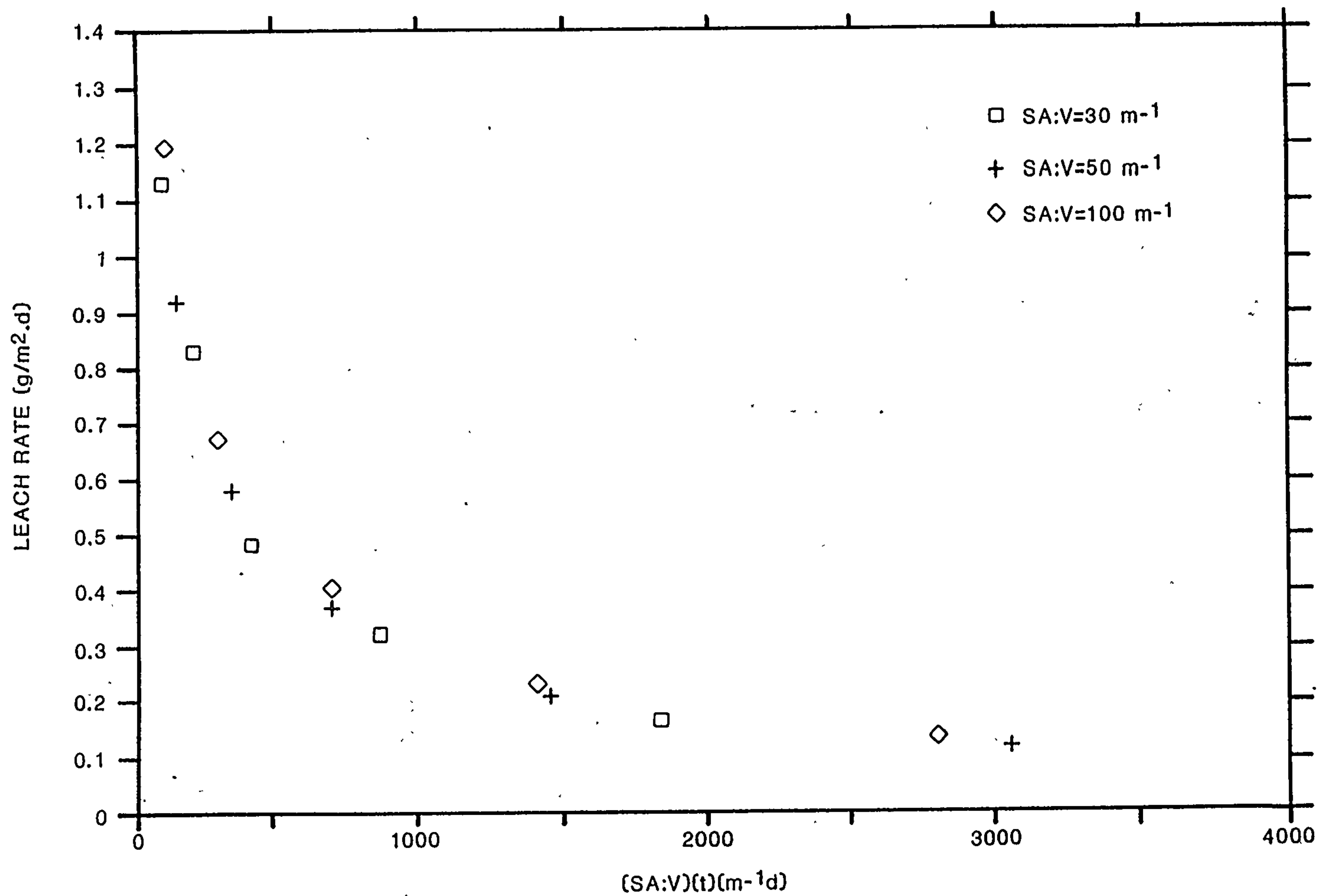


Figure 7-27. Effectiveness of surface area to volume ratios and times (SA:V (t)) scaling demonstrated by leaching of frit 165 glass in deionized water. Modified MCC-1 tests at 90°C from Bazan and Rego (1985). Tests are identical except for SA:V ratio.

using well J-13 water (Section 7.4.1.3). In this water, leach rates are considerably reduced from the rates observed using deionized water (Figure 7-28) a result that is typical for silicate rock ground water (Mendel, 1984). Figure 7-29 shows the effect of adding tuff rock to a static leaching experiment. The addition of a monolith (wafer) has a small effect, but adding rock powder dramatically reduces leaching. This is attributed to the more rapid addition of silica to the water by the rock powder relative to the monolith, which is less reactive. The reduction in leach rate brought about by tuff rock and tuff water is not fortuitous. The rock is somewhat similar in composition to the glass (both are composed largely of silicon, aluminum, alkalis, and alkaline earth oxides), but the rock has a higher silica content and it saturates the solution with silica, slowing glass dissolution. In a repository at Yucca Mountain, vadose water that has passed through heated rock in route to the waste container is expected to contain high levels of dissolved silica (Section 7.4.1.3).

Most other possible repository materials are not close to the waste glass composition, and interactions with them cause increased leach rates. Examples include mild steel and bentonite clay, which were investigated in the defense high-level leaching mechanisms program. The experimenters found that 1020 carbon steel caused the precipitation of iron silicate compounds with a resulting doubling of leaching (Mendel, 1984). Grambow et al. (1987) showed that adsorption of silica on iron or carbon steel surfaces can be responsible for increased leach rates. NNWSI Project testing has found similar results for 1020 steel in the presence of well J-13 water and tuff rock (Figure 7-30). Mendel (1984) found that bentonite increased leach rates, possibly because of adsorption of leached ions or reaction with silica. This effect was later attributed to higher pH values in the presence of the clay, which has a significant cation-exchange potential resulting in replacement of sodium in the clay for hydrogen from solution (Grambow et al., 1985). Grambow et al. (1985) predict that the long-term rate of dissolution in the presence of bentonite will be the same as in water alone, because of the onset of silica saturation in solution. This predicted effect would depend on the water staying in contact with glass and bentonite simultaneously for a long period of time (more than 1 yr).

The 304L SS is being extensively investigated but appears to have substantially no effect on static leach rates (Figure 7-30) (McVay and Robinson, 1984; Bazan and Rego, 1985, 1986). This is in accord with its low reactivity in well J-13 water (Section 7.4.2). The effect on leach rates caused by contact with heat-affected 304L SS is being investigated (Section 8.3.5.10).

Other repository materials that affect the pH of ground water are also expected to affect leach rates. An adverse effect on leach rate would be caused by material that water could contact before contacting waste glass and that causes the pH of the water to rise. Concrete containing unreacted portlandite ($Ca(OH)_2$) in the cement could raise water pH into the rapid dissolution regime. For instance, Atkinson et al. (1985) showed that water in continuous contact with concrete and clay would maintain a pH above 10.5, which is in the rapid dissolution regime, until the concrete is completely decomposed by ground water. In their example, this would take 1 million years. Nitric acid production by gamma radiolysis of air (Section 7.4.3.2.2) could result in acidic ground water during the containment period, but

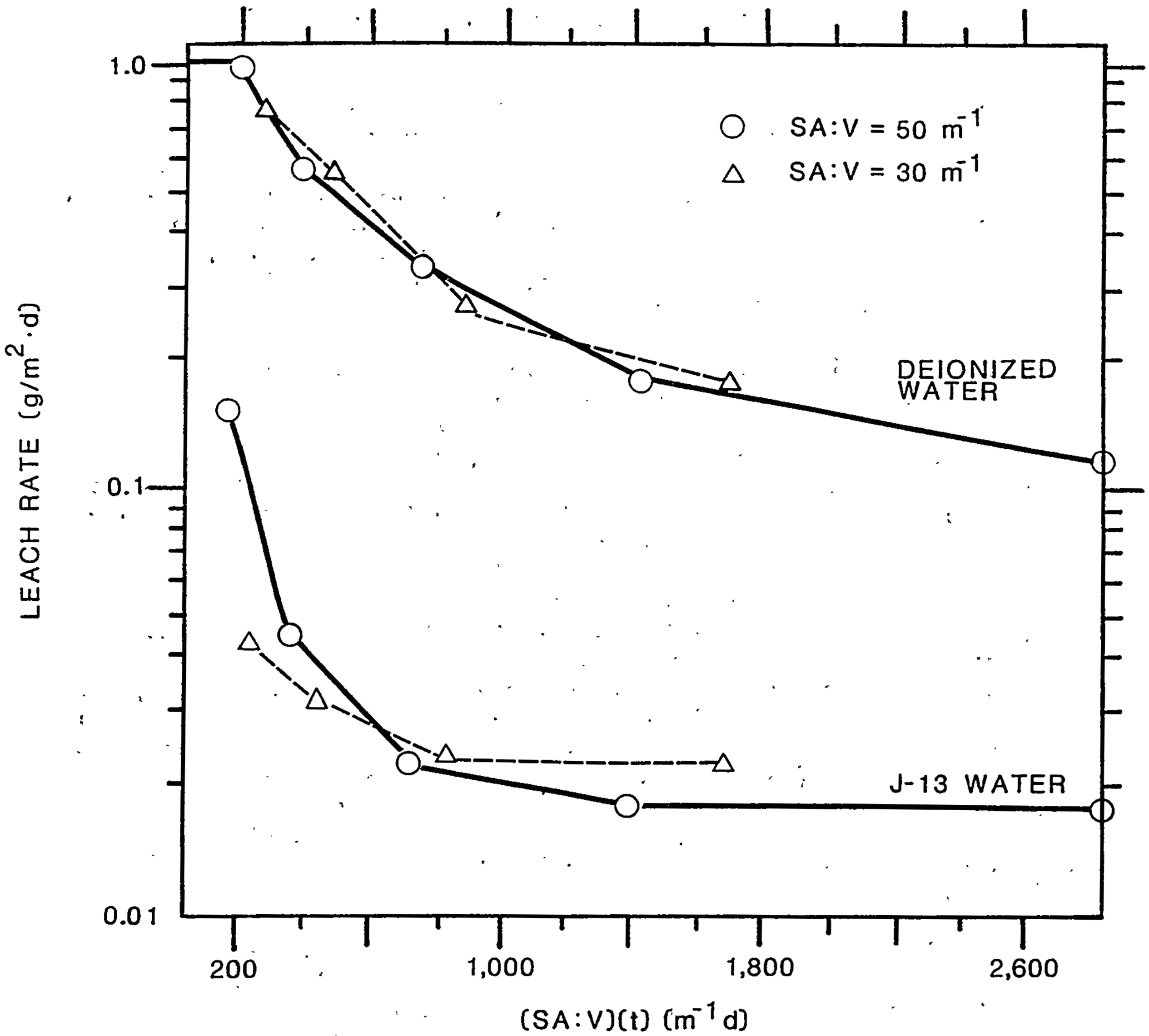


Figure 7-28. Effect of water from well J-13 on leach rates of lithium from frit 165 glass. Modified from Bazan and Rego (1985).

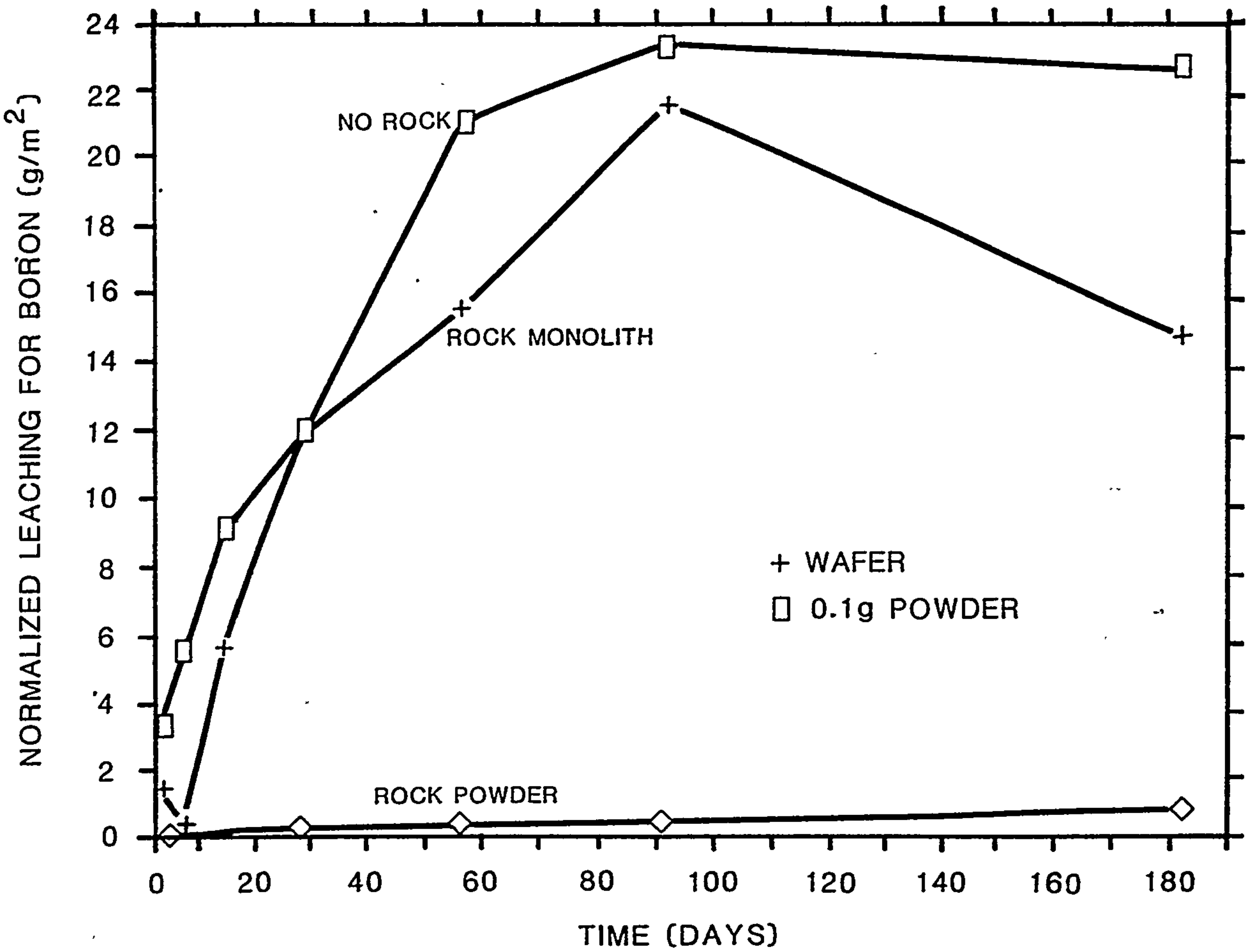


Figure 7-29. Effect of tuff rock being present in the leaching vessel on the leach rate of PNL 76-68 glass. 90°C, ratio of surface area of glass to volume (SA:V) = 10 m^{-1}. Rock monolith has same surface area as glass. Modified from McVay and Robinson (1984).

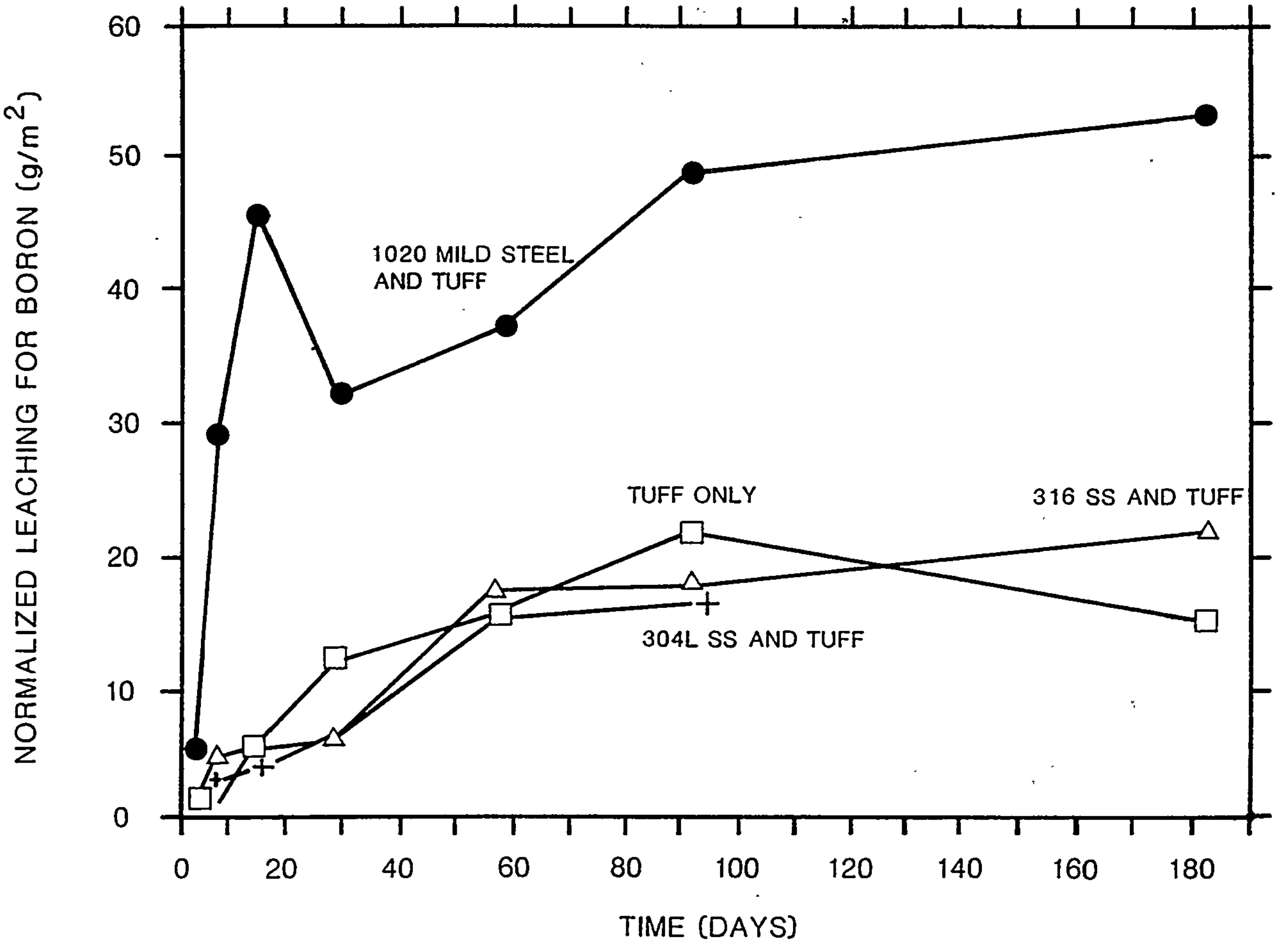


Figure 7-30. Effect of repository components on leaching of PNL 76-68 glass. All experiments included tuff monolith with same surface area as glass. Metals had same surface area (SA). Ratio of surface area to volume (SA:V) = 10 m^{-1}. The data for 304L stainless steel for 180 days were lost. Modified from McVay and Robinson (1984).

radiation fluxes will be sufficiently reduced in the isolation period that this effect is not expected to be significant (Section 7.4.3.2.2).

The results of studies on the effect of possible backfill and packing materials like bentonite have shown that increasing the complexity of the system may slow the movement of some radionuclides but that one likely result is also increased radionuclide leaching from the glass. Only materials that increase the local silica concentration or buffer the pH near the neutral point seem to effectively reduce total glass dissolution rates.

7.4.3.2.2 Results of recent NNWSI Project glass waste form testing

Research up to this point has involved parametric tests evaluating the effects of repository materials (glass, tuff, equilibrated water, and canister metals) on leach rates and developing tests designed to evaluate the effects of intermittent water contact in a water-unsaturated environment. In this section, materials used in NNWSI Project tests and their similarity to anticipated repository materials are discussed. This discussion is followed by descriptions and results of these tests.

Although different borosilicate waste glasses behave in generally similar fashions, small changes in composition can cause changes of several orders of magnitude in release rates under static conditions (Mendel, 1984). The behavior of radionuclides can be inferred from simulated waste glasses but must ultimately be measured in tests using glass containing radionuclides. The variability in composition of sludge waste streams means that waste glasses will have some variability in composition. These factors make the glass compositions used for testing important. The NNWSI Project is currently using actual sludge glasses, simulated glasses, and glasses doped with specific elements of interest in its testing program.

Table 7-25 gives the compositions of four SRL glasses used by the NNWSI Project and SRL. It also contains the composition of the reference DWPF glass used by the defense high-level waste leaching mechanisms program. The four simulated glasses in the table are very similar in composition. The actual-waste glass has a slightly lower iron content and higher alumina content. This type of variation must be expected (and its effect evaluated) due to the inhomogeneity of SRP sludge. The NNWSI Project is also using selectively doped PNL 76-68 glass in parametric tests (Table 7-19).

An important aspect of evaluating release rates is the homogeneity and quality of the waste glass. Phase separation of radionuclides, such as the technetium globules made by Bradley et al. (1979), results in erratic and nonreproducible results. The only separate phases currently expected in DWPF waste glasses are spinel, which precipitates at the canister wall, and spinel and acmite, which can precipitate near the canister centerline (Bickford and Jantzen, 1984). Spinel formation appears to have little effect on leach rates, but acmite formation may increase leach rates. No assessment of the devitrification characteristics of West Valley glass is yet available, but similar products may be expected due to the similar chemistries of the glasses. Possible precipitation of thorium-containing phases will be examined in West Valley glass when it is available (Section 8.3.5.10). Glasses

Table 7-25. Frit 165 based glasses

Oxides	Radioactive sludge tank 42 slurry fed melter (Bibler, 1986)		Simulated slurry fed melter (Bibler et al., 1984)		Simulated slurry fed melter (Bazan and Rego, 1985)		Simulated "Black Frit" (Bates and Gerding, 1985)		Simulated defense waste reference glass (Mendel, 1984)	
	Wt%[a]	Mole%[b]	Wt%	Mole%	Wt%	Mole%	Wt%	Mole%	Wt%	Mole%
SiO_2	55.40	42.17	56.20	43.44	54.90	42.44	54.10	43.22	51.60	43.33
Na_2O	11.00	16.35	10.90	16.34	11.40	17.09	10.30	15.96	7.70	12.54
Fe_2O_3	6.00	3.46	12.30	7.16	10.60	6.17	12.30	7.40	10.10	6.38
B_2O_3	8.40	11.12	7.00	9.34	7.20	9.60	6.80	9.38	7.30	10.58
Al_2O_3	9.80	8.86	4.90	4.47	5.13	4.67	4.10	3.86	5.50	5.44
Li_2O	4.90	15.11	4.70	14.61	5.04	15.67	4.70	15.10	4.10	13.85
UO_2							1.12	0.20	2.70	0.50
MnO_2	1.90	1.01	2.80	1.50	2.72	1.45	3.56	1.97	3.40	1.97
NiO	0.90	0.56	0.96	0.60	0.84	0.52	0.90	0.58	2.10	1.42
CaO	0.24	0.20	2.00	1.66	1.51	1.25	1.50	1.28	1.90	1.71
Cs_2O					.00	.00	0.11	0.04	0.40	0.14
K_2O					0.14	0.14			0.04	0.04
MgO	1.00	1.14	0.78	0.90	0.72	0.83	0.80	0.95	0.80	1.00
SrO					0.10	0.04	0.15	0.07	0.50	0.24
CeO_2					0.42	0.11			0.04	0.01
Other[c]									1.67	0.85
Total	99.2	100.0	102.54	100.00	100.72	100.00	100.44	100.00	99.85	100.00
Sum of Si and B		53.29		52.78		52.04		52.60		53.90

[a]From published analyses or nominal values from Savannah River.
[b]Mole % cation in glass (total constrained to be 100%).
[c]Defense waste reference glass also contains Cr_2O_3 0.3%, ZrO_2 1.0%, P_2O_5 0.2%, and traces of Cu, Co, Ti, Zn, and Nd.

produced in the full-size melters at DWPF and West Valley are not yet available to the NNWSI Project for testing. The quality and homogeneity of samples will have to be evaluated before placing full confidence in results from laboratory-produced glasses. Preliminary results of laboratory-scale leach testing of glasses produced by a full-scale melter and poured into DWPF reference canisters have recently become available (Savannah River Plant and Laboratory, 1984). No difference in leach rate was observed between laboratory and full-scale 165-frit glasses. When full-sized slices from the canisters were leach-tested, release increased by less than a factor of three due to cracking (Bickford and Pellarin, 1986).

Materials that are likely to be present in a repository at Yucca Mountain are used in NNWSI Project testing. The reference container and pour canister material, 304L SS, is currently being used in parametric testing and for test-vessel construction. Well J-13 water is used for testing. When used for elevated temperature testing (e.g., 90°C), well J-13 water should be re-equilibrated with tuff at that temperature. Tuff representative of that from the potential repository horizon is currently available from surface samples, conventional drill cores, and air-drilled core samples (Section 7.4.1). The core samples are expected to be representative of the rock that will be found in the repository and are in limited supply due to the expense of drilling. Surface samples suffer from contamination by caliche and soluble salt deposits (Knauss, 1984). This material is common in the arid regions of the southwestern United States (Conca, 1985) and is caused by deposition of airborne salts and surface evaporation of moisture. It is confined to the exposed area of the rock and is frequently only detectable by analysis of water that has equilibrated with a tuff sample; use of samples containing these soluble salt deposits causes elevated anion, calcium, sodium, potassium, and boron concentrations in the water (Oversby, 1984a). Boron in these surface deposits is particularly troublesome because it obscures boron release from the glass when testing is done in the presence of tuff rock. The deposits may be substantially removed from tuff samples by pre-equilibration with well J-13 water. However, the presence of this material has complicated the analysis of some of the initial NNWSI Project tests.

The NNWSI Project has completed parametric tests examining the effects of repository materials on the leaching of SRL 165 frit-simulated glass (Bazan and Rego, 1985), PNL 76-68 U-doped glass (McVay and Robinson, 1984), and PNL 76-68 actinide-doped glass (Bazan and Rego, 1986). Figure 7-31 shows nonmalized leaching (NL) values for lithium at an SA:V of 30 m^{-1} from the 165-frit study. This study showed that 304L stainless steel has no substantial effect on leaching and that the presence of tuff slightly decreases leaching rates. These results were also found in the PNL 76-68 glass study (McVay and Robinson, 1984). However, the presence of ductile iron is known to increase leach rates in static tests (Mendel, 1984). The NNWSI Project is therefore currently examining the effects of heat-affected 304L SS, which may also be more reactive. Heat-affected 304L steel will be present in the glass pour canisters. Preliminary results (Bates et al., 1986b) indicate increased glass reaction in the presence of weld-affected steel, as evidenced by the formation of nickel and chrome silicates. This effect is being examined (Section 8.3.5.10).

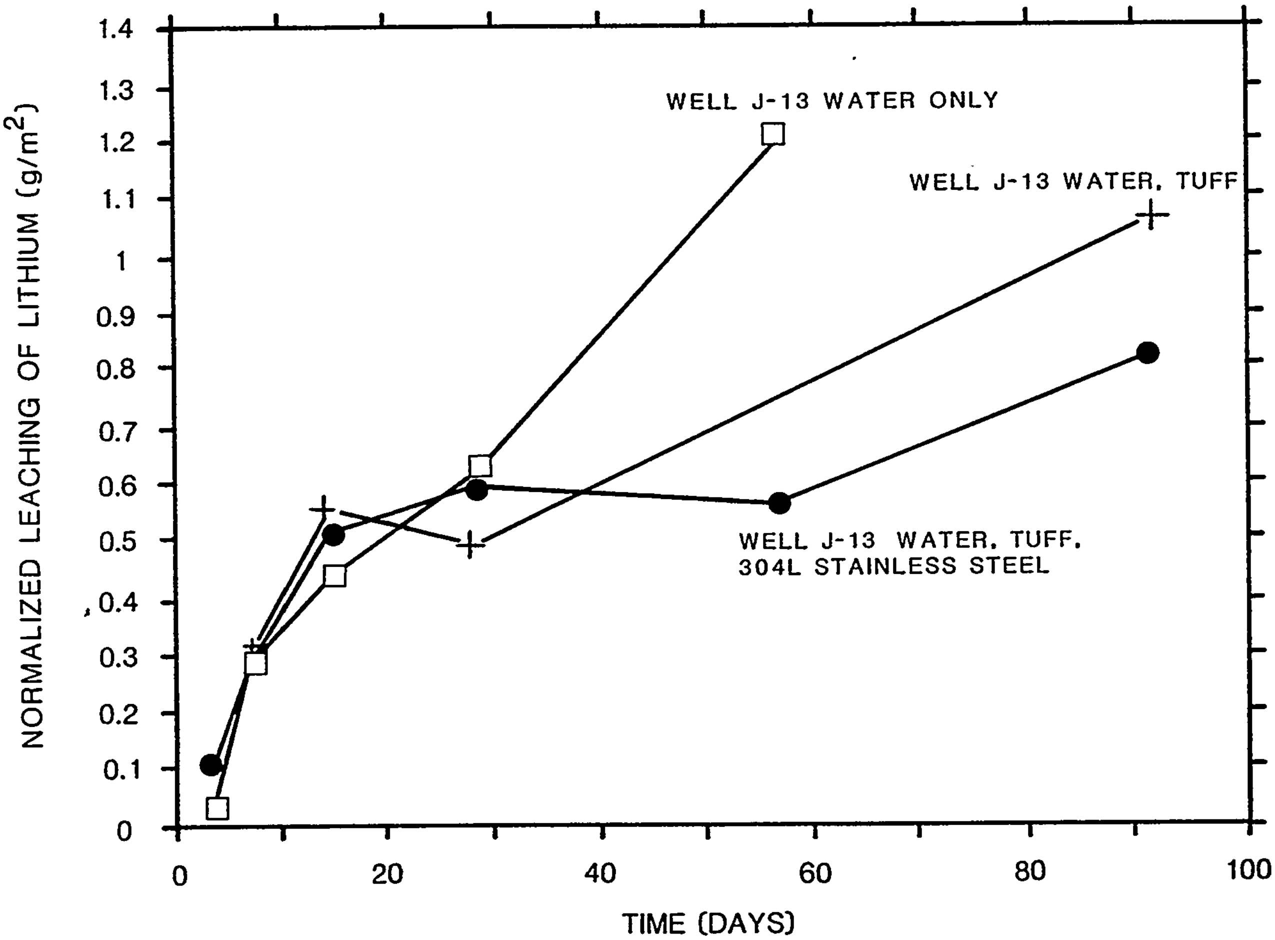


Figure 7-31. Effect of repository components on the leaching of 165-frit glass. MCC-1 tests at 30 m^{-1}, 90°C, 1 g of tuff per 20 ml well J-13 water. Note that these experiments only extend to 90 days. Data from Bazan and Rego (1985) were corrected for leachate loss.

In parametric studies, the analysis of glass release is complicated by the presence of elements occurring in both the tuff and the glass. Elements such as silicon cannot be used to measure glass release when tuff is present. Boron and, for 165 frit glass, boron and lithium can give results with minimum interference. In many instances with 165-frit glass, the glass releases so little of an element that analysis is difficult even without interferences.

The effects of gamma radiation on leaching rates of SRL 165 and PNL 76-68 glass have been investigated in a series of experiments (Bates and Oversby, 1984; Abrajano et al., 1986; Bates et al., 1986a,b; Ebert et al., 1986). Both nonradioactive and actinide-doped glasses have been leached in cobalt-60 gamma fields with dose rates of 2×10^5, 1×10^4, 1×10^3, and 0 rads/h. The primary effect of this radiation was expected to be the production of nitric acid from air in the test vessels (Burns et al., 1982). This was observed. The pH for the 165 frit glass leaching solution (well J-13 water) was 6.5 after 56 days in the 2×10^5 rads/h field (Figure 7-32). Bazan and Rego (1985) obtained a pH of 9.5 to 9.7 under similar conditions but with no irradiation. Blanks (no glass present) run by Bates et al. (1986b) yielded results of 11 ppm nitrate production, which is in excellent agreement with the predicted value of 10 ppm obtained by using the equation from Burns et al. (1982).

NL values from the gamma irradiation work at 2×10^5 rads/h are shown in Figure 7-33 for SRL 165 frit glass. Notable are the NL values for plutonium, uranium, and americium; all are below those of the most leachable elements (lithium and sodium). For PNL glass, the pH first increased, then decreased after 14 d. This was reflected in the actinide data for that glass, which show a decrease in actinide concentration in solution at the end of the 56-day period (Bates and Oversby, 1984).

Results for SRL glass at 1×10^4 rads/h (Abrajano et al., 1986) are similar to those at 2×10^5 rads/h, (Bates et al., 1986a) but show slightly increased releases at corresponding times (e.g., normalized loss of lithium at 56 d was 3.4 g/m^2 compared with 2.3 g/m^2 in Figure 7-33). The solution pH values with glass present were also slightly higher, and less nitrogen was fixed by the radiation. The increased glass reaction was interpreted as due to the increased pH, with glass dissolution (which drives the pH upward) now balancing more of the acid production by radiolysis. However, the differences between reaction rates in the irradiated samples are small compared with the difference between them and the nonirradiated samples, leading to the interpretation that glass reaction rate is fairly unresponsive to irradiation flux above 1×10^3 rads/h (Ebert et al., 1986). This is an example of the buffering effects of the solution products of glass dissolution and radiolysis; the glass reaction remains approximately constant as long as buffering species from both processes are present and is not sensitive to the quantity of the buffers (i.e., radiation flux). Experiments are under way (Section 8.3.5.10) to discover how much radiation is required to maintain the radiolysis buffer.

Actinide releases in the gamma-irradiation experiments are very sensitive to solution pH. In the experiments at 1×10^4 rads/h run for 182 d in well J-13 water only (Abrajano et al., 1986), the pH rose slightly (from 6.8 to 6.9) from the 91-day value after a continuous decrease up to that point

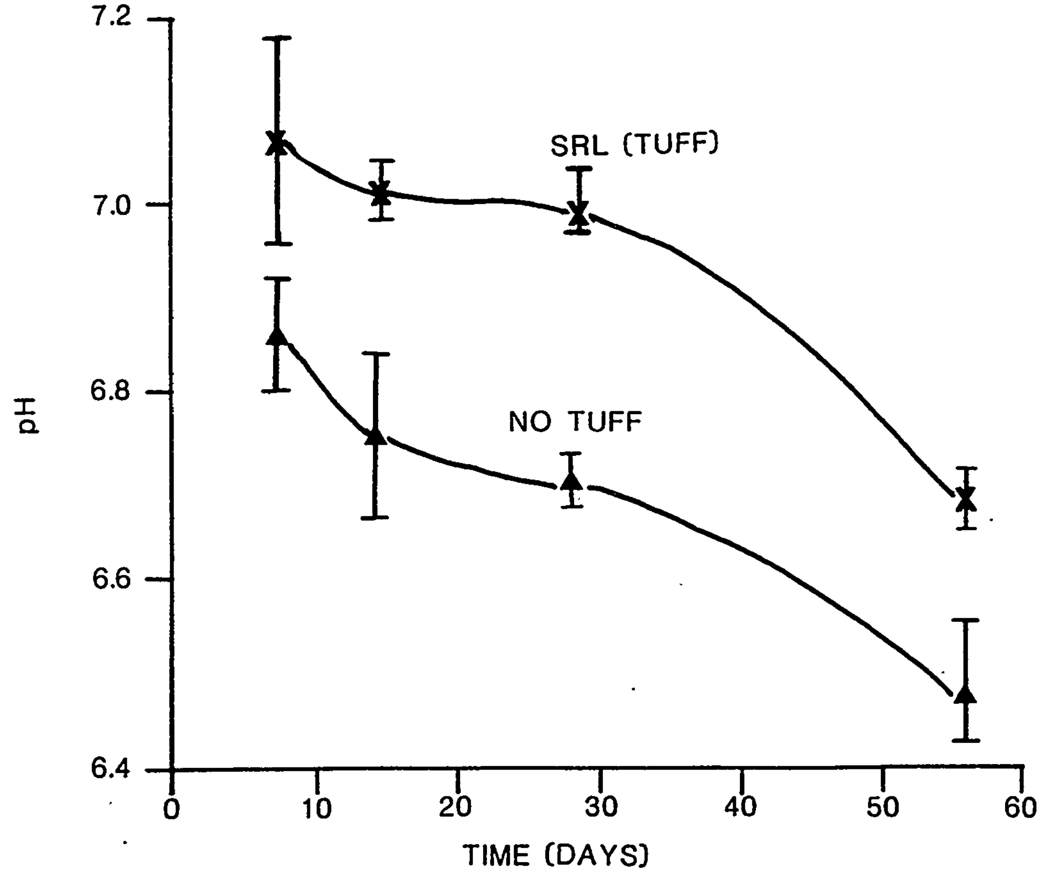


Figure 7-32. The solution pH from leaching experiments with a uranium-doped defense waste processing facility glass, from Savannah River Laboratory (SRL) in the presence of 2×10^5 rads/h gamma irradiation. Modified from Bates et al. (1986a).

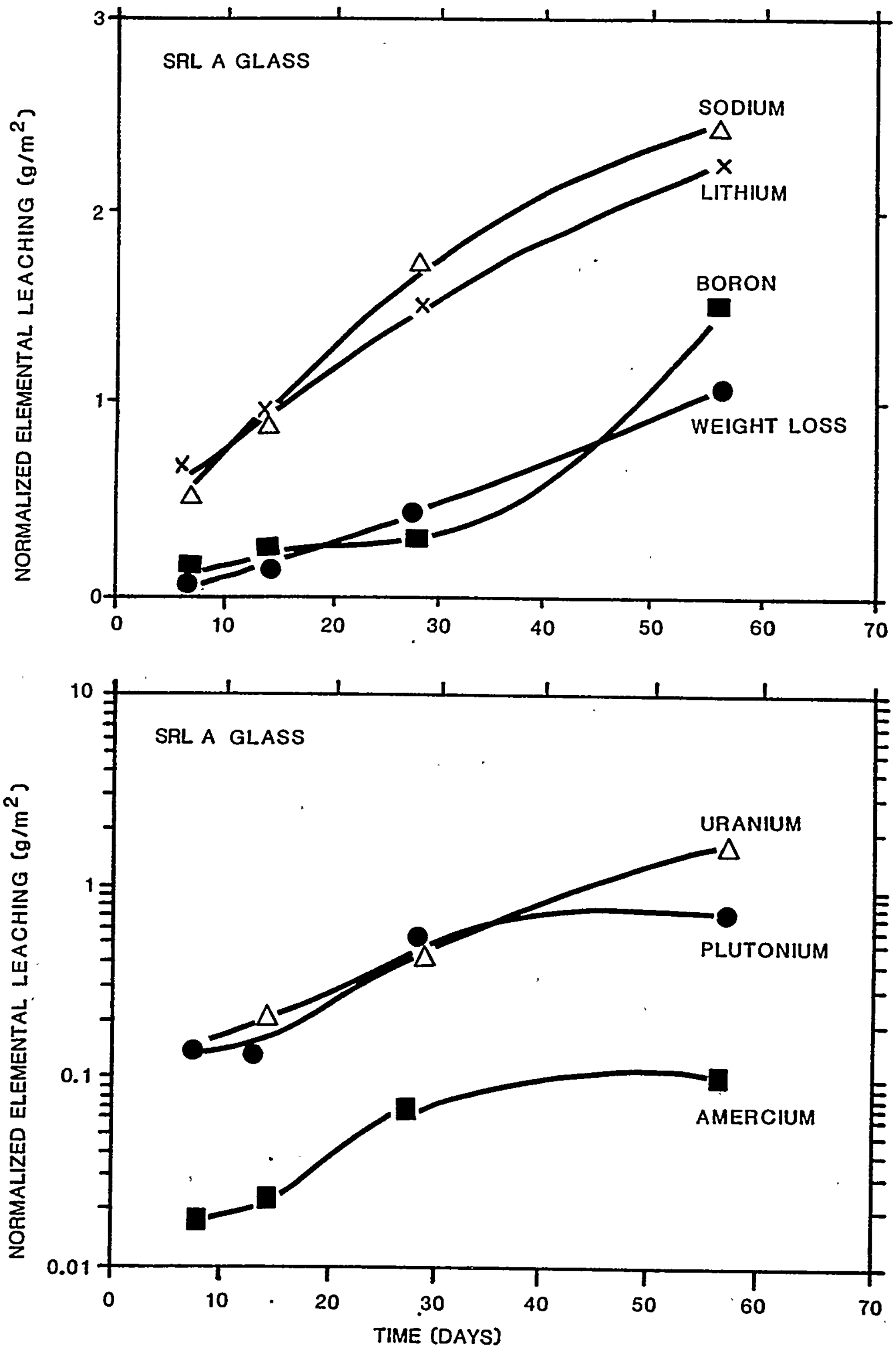


Figure 7-33. Release of actinides and frit elements from actinide-doped SRL-165 glass in the presence of 2×10^5 rads/h gamma irradiation in water from well J-13, MCC-1 type tests ratio of surface area to solution volume (SA:V) = 30 m^{-1}, temperature 90°C. Modified from Bates et al. (1986a).

(similar to Figure 7-32). Abrajano et al. interpreted this as the result of a slight loss of gas from the test vessels. It resulted in decreases in plutonium and neptunium release over the 91-day values; plutonium normalized release decreased from 2.15 to 1.67 g/m^2, and neptunium from 4.15 to 2.66 g/m^2. The normalized loss includes the concentration in solution and the amount adsorbed on the test vessel. The decrease in normalized loss must then represent precipitation onto the waste glass itself.

The gamma dose rates in these experiments were approximately 30 times, 1.5 times, and 0.15 times the maximum that would occur at the surface of fresh SRL glass. The anticipated pH change that could occur in repository water is less than that observed in the experimental work due to radioactive decay and the exclusion of water from the package vicinity immediately following emplacement when radiation fluxes are still substantial (Section 7.4.1). If a premature container breach were to occur, glass could come in contact with water while acid radiolysis was still occurring. However, because the gamma radiolysis drives the pH in the direction opposite to that of the glass dissolution, no increase in glass dissolution rate is expected because the solutions would remain at near-neutral pH. The effect of long-term buildup of nitrate before container and pour canister breach is not known. Testing at dose rates higher than those anticipated was originally intended as an accelerating mechanism (Bates et al., 1986b). They have shown, however, that this is not valid because the glass reaction changes dramatically as a function of dose rate and the accompanying pH changes. These experiments have been very useful in evaluating glass reaction mechanisms and the effect of pH on reaction rates and actinide release. Unlike experiments in which buffering chemicals are added to the solution to examine the effects of pH, in these experiments the pH is buffered by radiolysis with very minor changes in the solution chemistry (an increase of about 10 ppm nitrate and 5 ppm nitrite at most).

The effect of radiation from fresh-waste glass was tested at SRL using a 165-frit glass made with actual SRP sludge (Bibler, 1986) and with a SA:V of 100 m^{-1}. This glass contained high levels of cesium-137 and strontium-90, resulting in a dose rate to the surrounding leaching solution (well J-13 water) of about 1400 rads/h. In this experiment, the leachant solution absorbed most of the beta emissions, and there was no substantial dose to air in the vessels. However, the results of the experiment were similar to those just described (Bibler et al., 1984) with solution pH values remaining near neutral during 134 d of leaching. The normalized loss of boron and lithium was similar to experiments conducted with nonradioactive glass. These experiments were conducted in stainless steel vessels and showed that previous work conducted in Teflon™ vessels (Bibler et al., 1984) were compromised by large releases of F^- ion from the Teflon™ into solution, resulting in greatly increased leach rates. One likely source of the acidic buffer in these experiments is the radiolytic production of hydrogen peroxide in solution (Van Konynenberg, 1986).

In an effort to simulate the dissolution of glass in a tuff-dominated environment, the NNWSI Project and SRL have conducted leaching experiments similar to the MCC-1 static leach test but in which the glass sample is held inside a closed cup made of tuff. The glass in the SRL work was 165 frit glass, cast into 304L stainless steel tubes and then cut into wafers for testing. Both simulated and actual-sludge glasses were used (Table 7-25). A similar test was done by Bazan and Rego (1986) using ATM-8, a PNL 76-68 based glass doped with low levels of technetium, neptunium, uranium, and plutonium (Wald, 1985). Results of the SRL experiment are shown in Figures 7-34a and b, (Bibler et al., 1984). In both the SRL and NNWSI Project work, a significant result of the presence of the tuff cups was the control of solution pH by the tuff. The pH remained in the vicinity of 8.3 to 8.7, which is the same as that found when well J-13 water alone is equilibrated with tuff. Glass reaction rates in the presence of rock were reduced relative to those without tuff (Figures 7-34a and b). This may be due to a combination of the lowered pH (≈8.5 with tuff vs. 9.3 without) and the addition of silica to solution by the tuff rock.

The results for plutonium-238, cesium-137, and strontium-90 release from 165-frit glass in the absence of tuff show increases followed by decreases, brought on possibly by the precipitation of colloids. This is typical behavior for these elements in leach testing (Mendel 1984). The NL values for radioactive glass with tuff absent were redetermined by Bibler (1986) using 304L stainless steel vessels, which are unaffected by radiation from the glass that caused fluorine release from the original Teflon™ vessels. Overall degradation of radioactive glass measured by boron and lithium release in the stainless steel vessels was identical within error to the results for nonradioactive glass (Figure 7-34a). In radioactive glass, the results for cesium-137 and strontium-90 were the same within error in the stainless steel and Teflon™ vessels (Figure 7-34b), indicating that they are solubility controlled. Plutonium-238 results were slightly larger in stainless steel (up to NL = 0.69 g/m^2 at 91 d).

Where tuff is present, the solution concentrations may be lowered because of adsorption by the rock. NL values reported for the tuff reaction vessel experiments (Bibler et al., 1984; Bazan and Rego, 1986) do not include amounts that were removed from the glass and then adsorbed onto the tuff. Plutonium, neptunium, uranium, and technetium release from PNL 76-68 based glass were measured in the Bazan and Rego study (1986). Similar results to Figure 7-34a were found, with very low releases of the actinides but no decrease in release rate at long times. Technetium was released at approximately the same rate as the other soluble components of the glass frit (molybdenum and boron). The actinides did not migrate through the tuff vessel in this experiment, but technetium did.

The tuff reaction vessel experiments show that, in the presence of the anticipated repository materials including tuff, tuff-equilibrated well J-13 water, and 304L stainless steel, glass degradation rates and radionuclide release rates are decreased relative to experiments done with either well J-13 water alone or deionized water. The effect is due to pH buffering and silica addition to solution by the tuff. It is not anticipated that glass will contact water under these tuff-dominated conditions in the repository since the pour canister and container will be present and will prevent extensive direct contact of the glass with tuff. However, should these conditions occur, glass degradation would be decreased relative to the expected condition.

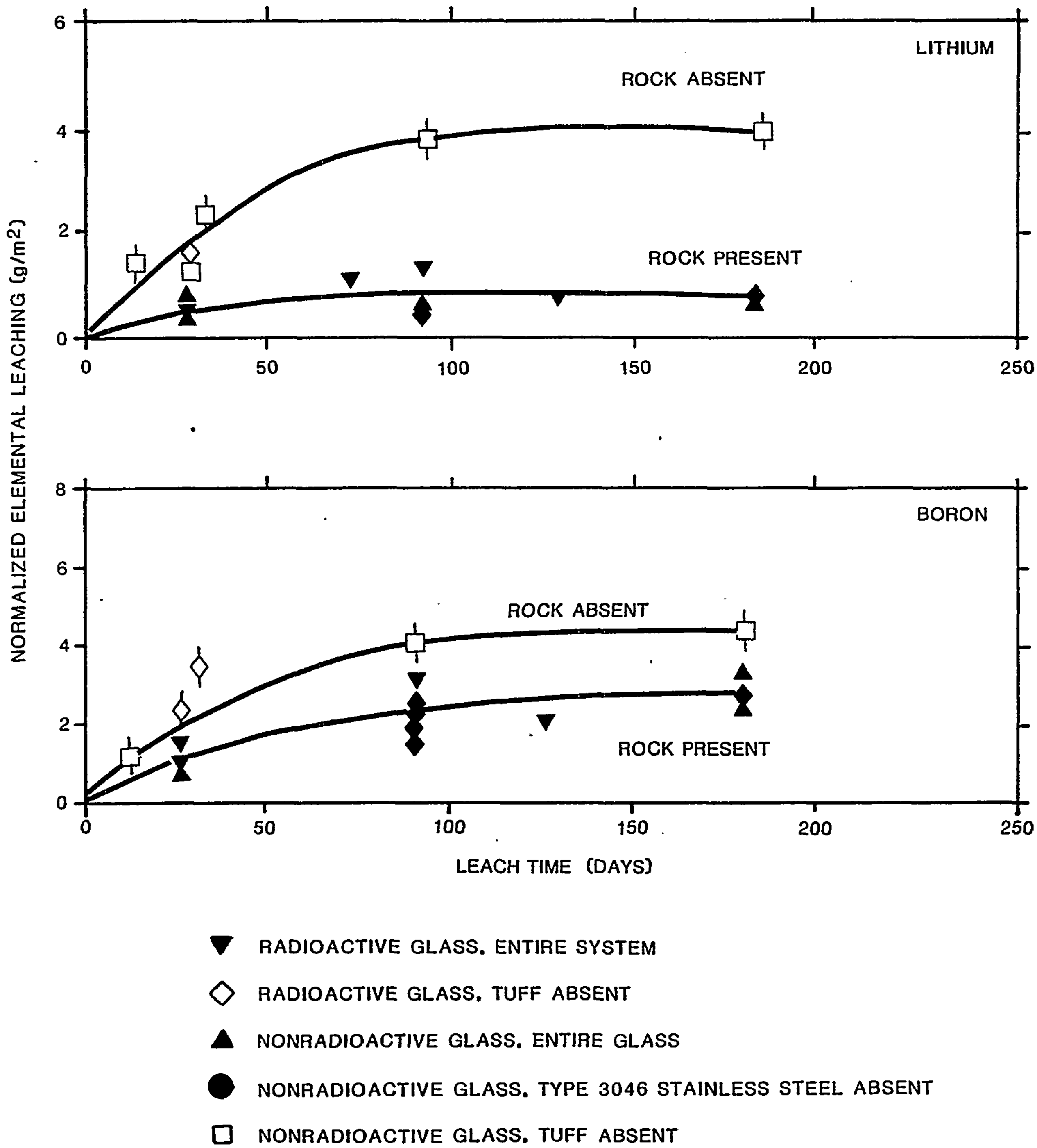


Figure 7-34a. Normalized mass losses (NL) based on lithium and boron for actual and simulated Savannah River Plant waste glass in presence (solid symbols) and absence (open symbols) of tuff leach vessels (Bibler et al., 1984). T = 90°C, SA:V = 100 m^{-1}. The lithium data shown here have been corrected from the original reference according to the new analysis given in Bibler (1986).

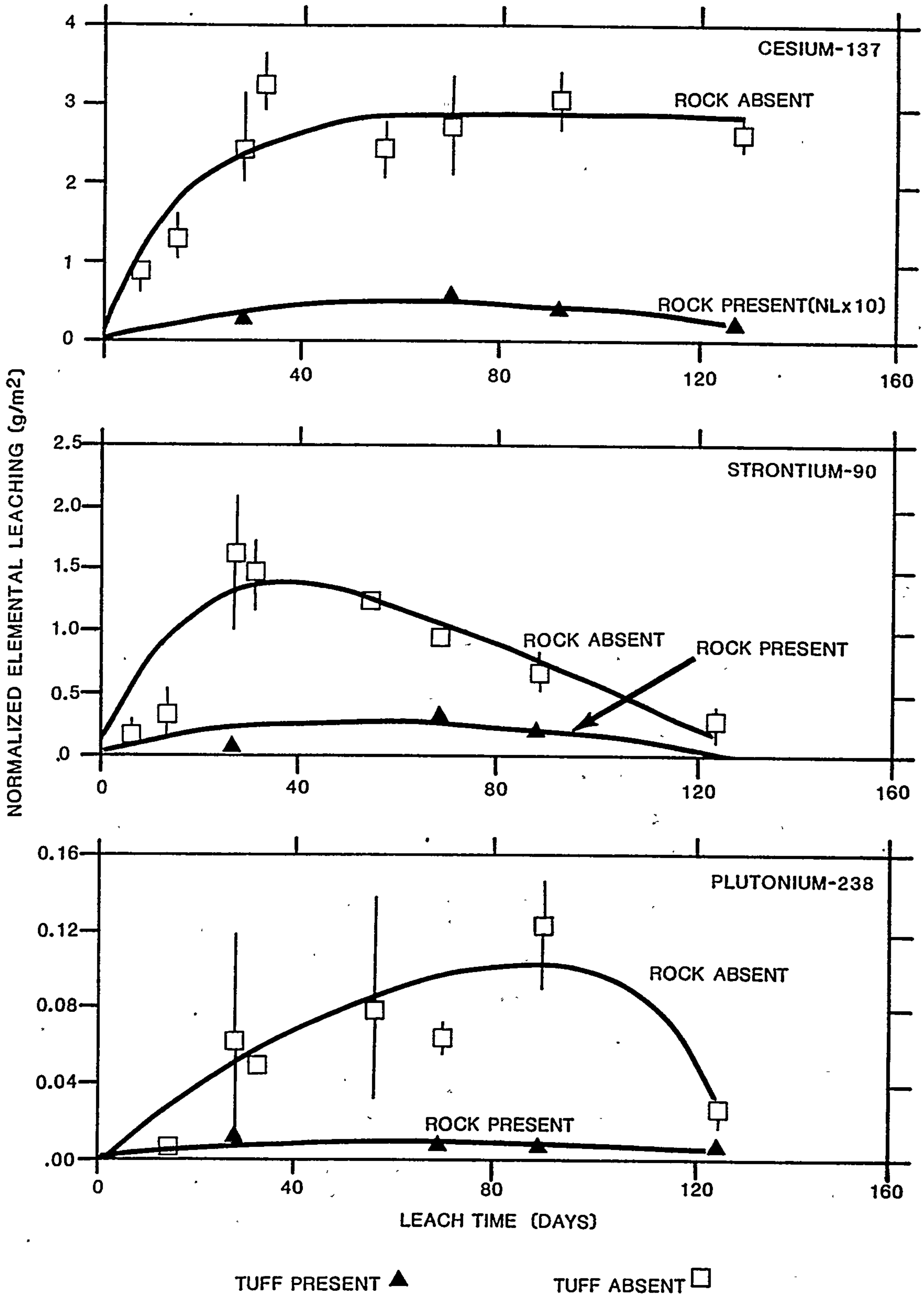


Figure 7-34b. Decrease in normalized mass losses for cesium-137, strontium-90, and plutonium-238 to tuff $T = 90°C$, $SA:V = 100\ m^{-1}$.

Bates and Oversby (1984) compare the quantitative results of three of the static studies discussed here. The results of Bazan and Rego (1985) are clearly lower than those of other experiments (Bates and Oversby, 1984; Bibler et al., 1984). This difference cannot be individually attributed to glass composition (Table 7-25), leaching conditions, or experimental method. All samples had surfaces prepared similarly but not identically, and this is known to affect leach rates (Mendel, 1984). Bates and Oversby (1984) used core-drilled samples, which probably have a slightly rougher average surface than the saw-cut samples used by Bibler et al. (1984) and Bazan and Rego (1985). Although all three experiments used well J-13 water, Bazan and Rego (1985) equilibrated theirs with tuff (at 90°C) for 1 month, Bates and Oversby (1984) for 2 weeks, and Bibler et al. (1984) used unequilibrated well J-13 water.

Tests performed by SRL (Savannah River Plant and Laboratory, 1984) using 165-frit glass and a tuff ground water yielded release rates two to three times higher than those found in NNWSI Project studies. These tests were run at low SA:V ratios of 10 m^{-1} and only for short periods of time. The final pH was not reported. These tests are difficult to compare quantitatively to NNWSI Project tests. A later set of SRL experiments using well J-13 water and SA:V ratios closer to those used by Bazan and Rego (1985) (Bibler, 1986) also yielded about twice as much reaction. It is possible that differences in processing may also affect leachability, despite similar compositions. This amount of variation must be accepted in glass leach testing, and it illustrates the danger in using only one test or test method. Notably, however, the SRL 165 frit glass in all studies released very low levels of all elements. A detailed understanding of solution compositions and glass reaction mechanisms will be required to understand completely the sources of variability in these experiments.

Two tests have been developed by John Bates and coworkers at Argonne National Laboratory to test leaching under water-undersaturated conditions (Bates and Gerding, 1985). In the first test, water is dripped onto a glass-stainless steel assembly designed to model a perforated canister (Figure 7-35(a)) and is known as the NNWSI Project Unsaturated Test (previously the Phase II Materials Interaction Test). In the second test, known as the analog test, water vapor is forced through a tuff cylinder containing a similar waste package (Figure 7-35(b)). The purpose of both tests is to determine what effects are important when leaching occurs in a combination of air, water vapor, and liquid water. These tests examine the effects of one of the important expected environments for glass leaching at Yucca Mountain: one where water contacts the glass and then runs off without accumulating.

Preliminary results for unsaturated and analog testing on 165 frit glass are given in Bates and Gerding (1985), and the results of a complete one year test using SRL-165 frit glass are given in Bates and Gerding (1986). As a result of the experience with that one-year test, some changes were implemented in the test procedure. Preliminary results from a second one year test, as well as parametric studies on the effects of test parameters, are available in Bates et al. (1986a,b). These tests will help determine whether periodic wetting of glass can result in higher release rates than those determined in static tests.

The most important purpose of these tests is to examine interactions between materials. One notable interaction occurs between the perforated 304L stainless steel and the glass it contacts (Figure 7-35). Particularly at welded regions, discoloration of the stainless steel is observed, and rust flakes have been found in the solution (Bates and Gerding, 1986). This interaction has not been found in experiments involving static solutions.

The effect of using deliberately sensitized 304L stainless steel is being investigated currently (Bates et al. 1986b; Bates and Gerding, 1986). This examines the possible importance of the pour (inner) canister in affecting glass release rates. In some instances, the use of sensitized stainless steel has lead to considerable interaction between the glass and the steel (Bates et al., 1986b). In the first year-long unsaturated test (Bates and Gerding, 1986), the assemblies that showed the greatest release from the glass showed extensive discoloration of the stainless steel in contact with the glass. The 26-week test released three times as much boron as the 52-week test, a result that correlates with the observed greater interaction between the glass and the steel in the 26-week test.

Release rates are very low in unsaturated testing, even in the presence of glass-steel interactions. The greatest NL value observed for boron in the first year of testing was 0.8 g/m^2. Some elements (for instance, calcium and silicon) are actually added to the waste form (lost from the well J-13 water) as a result of precipitation reactions on the glass and stainless steel. These tests use the maximum water flow expected in the repository, scaled to glass surface area, and are run at 90°C. Parametric testing has been done to examine the effect of varying the water-to-glass ratio (Bates et al., 1986b). The release of glass components was found to be insensitive to this ratio; as long as there is water on the glass that never dries out completely, the release is approximately constant (within a factor of two). Most of the glass reaction occurs on the bottom of the waste package assembly (Figure 7-35) where water is observed to accumulate before dripping off.

Unsaturated tests are sampled in two ways; some are run to completion, the apparatus is dismantled, and all parts are analyzed. In others, the solutions that have dripped off the waste package are analyzed at 6.5-week intervals, and the waste package is reinserted into a new vessel for further testing. Preliminary comparisons show that similar results are found for each method.

The analog test is intended as a second method of measuring materials interactions and unsaturated leaching. In this test, a waste package similar to that in the unsaturated test is used, but the water contacts it as a result of vapor and unsaturated flow through the tuff rather than dripping onto the waste package from a tube as in the unsaturated test. NL values after 13 weeks of testing (Bates and Gerding, 1986) for radionuclides in the glass were NL (europium-152) = 0.2 g/m^2, NL (barium-133) = 0.2 g/m^2, NL (cesium-137) = 0.3 g/m^2. These are in reasonable agreement with unsaturated test results. The glass in the analog test showed marks indicative of water drying on the surface, and the bottom surface was wet when removed. Some glass-stainless steel interaction was indicated by slight discoloration of the stainless steel. The appearance of the waste package was very similar to waste packages in the unsaturated tests, indicating that the water contact and reaction mechanisms in the unsaturated test are not unique, and the

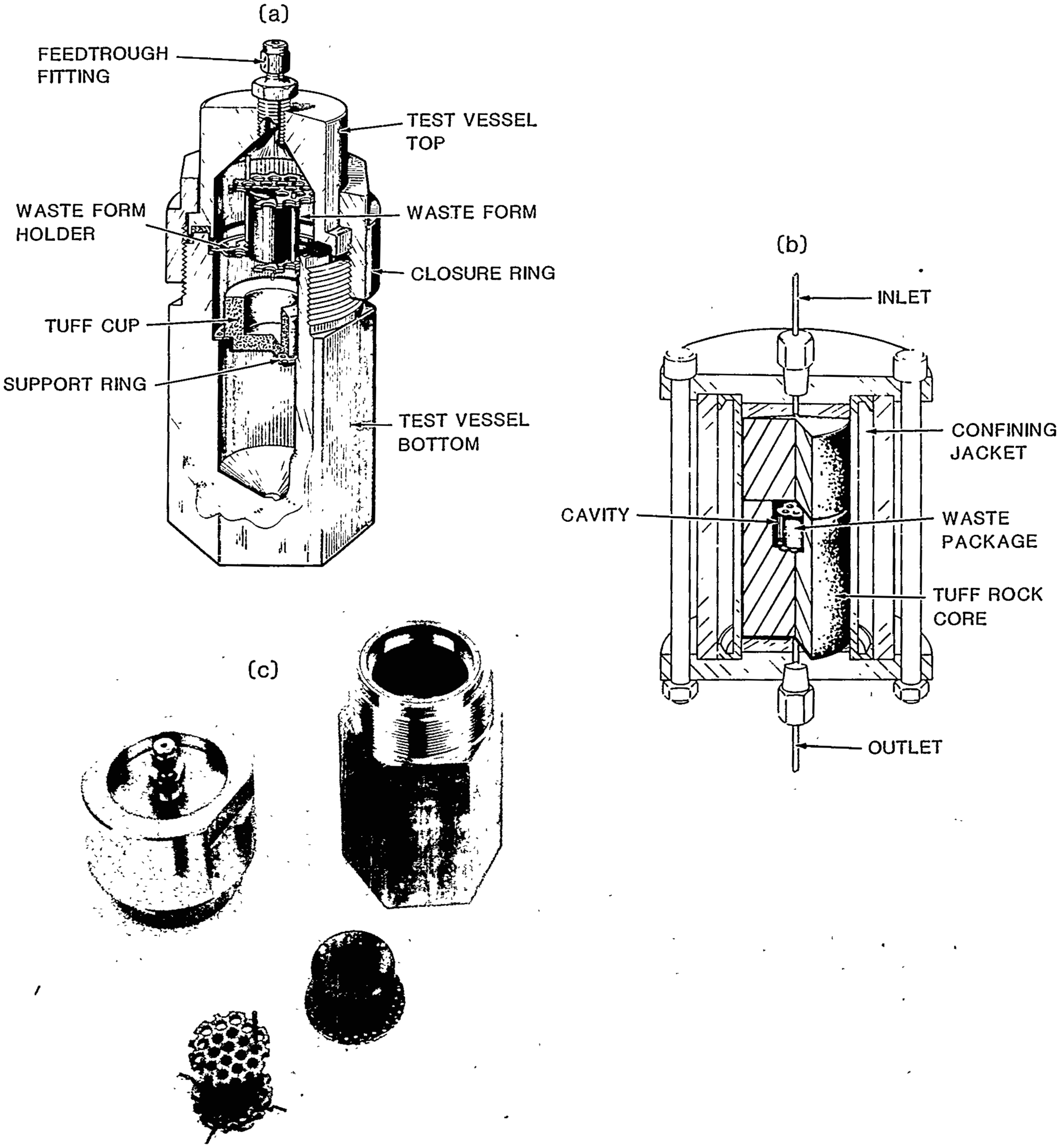


Figure 7-35. Equipment for two tests of leaching under water-unsaturated conditions. Panel (a) shows the unsaturated test vessel, (previously, Phase II materials interaction test) from Bates and Gerding (1985). Vessel is made of 304L stainless steel. Tuff cup is included in some experiments. Water drips from feed-through pipe onto waste package and is then collected from tuff or vessel bottom. Panel (b) shows the analog test apparatus. Water vapor under pressure is forced through the tuff core and collected from bottom. The contact of water with the waste package is uncontrolled. Panel (c) depicts selected parts.

results and interactions observed are applicable to a range of water contact mechanisms.

7.4.3.2.3 Release model for determining the source term for glass waste forms

The results the NNWSI Project and SRL tests cited in Section 7.4.3.2.2 confirm the previously established result that radionuclides are released at (normalized) levels equal to or below that of the soluble matrix elements of the glass (Mendel, 1984). The elements of interest are not left behind in the glass, however. They are reprecipitated onto the surface of the glass or test vessel in amorphous layers or crystalline deposits, or they are found in colloids in the solution that can settle out, attach to test components, or be trapped by filtering before the solution is analyzed. The decrease in plutonium concentration in Figure 7-34 in the absence of tuff was probably the result of colloid formation.

The amorphous layers and crystalline products have been studied by a number of workers, including Bates (Bates et al., 1982); investigators in the defense high-level leaching mechanisms program (Mendel, 1984); and Strachan et al. (1984). These new compounds are more stable forms of the components of the glass, and there are forms even more stable that can be expected to form over longer periods of time. The solubility and solution kinetics of both stable and metastable compounds will control the release of the radionuclides they contain. Only in the event of the fracturing of a glass waste form or of the separation of a surface layer will fresh glass be exposed. Mendel (1984) showed that even in that instance, reformation of a surface layer occurs quickly on a repository time scale. The nature and composition of surface layers can be strongly affected by minor components of the glass.

Prediction of the long-term behavior of waste glasses will require a better understanding of these secondary products. Current evidence (Mendel, 1984) indicates that the secondary products substantially reduce initial leach rates, but in the long-term, the presence of the crystalline products can provide the thermodynamic driving force for glass dissolution to continue even in relatively concentrated solutions (Grambow et al., 1985). The effects of the presence of crystalline materials need to be quantified, and the fate of colloids and flocculent precipitates in the solution needs to be determined, particularly of the hydroxide compounds whose solubilities are very sensitive to pH. The current knowledge of these compounds is not sufficient to rely upon them solely as a control of radionuclide release although, for specific radionuclides, some prediction of solubility behavior and its effect on leaching can now be made. A conservative approach for the present is to assume that some radionuclides will be released at the same rate as are the soluble frit elements and that overall glass degradation rates must be controlled so that these releases meet the regulatory requirements. The existence of the secondary compounds is the primary reason no glass source term can be accurately determined without considering all repository interactions; the glass-canister-container-water-rock system is synergistic, and the behaviors of the individual parts are impossible to separate.

Release limits based on congruent release are straightforward to calculate. The reference DWPF canister weight, 1710 kg (Baxter, 1983), yields a 1-part-in-100,000 release of 16.6 g/yr. Therefore, when scaled for surface area, normalized releases per canister must be less than that value. If the entire surface area of glass in a DWPF canister were exposed, it would have a geometric surface area of 5.0 m^2, yielding an allowed NL (normalized loss for each radionuclide) of 3.3 g/m^2 per year. Because of the expected persistence of container and canister material throughout the isolation period (discussed in Section 7.4.2.), it is unlikely that standing water will be able to react with the entire surface area of the glass. Two more likely scenarios are that the headspace of the canister could fill with water or that water will drip onto and off the waste form. The surface area in the first instance would be that of the top of the glass, which is about 0.275 m^2, yielding normalized loss units of allowed release (NL) of 60 g/m^2 per year. The factor of up to three increase in leach rate due to cracking observed in full-scale tests by Bickford and Pellarin (1986) would make the effective value 30 g/m^2 per year for comparison with laboratory experiments on uncracked glass. Allowed releases for the case where water drips onto and off the waste may be calculated similarly for various contact scenarios.

Current water availability estimates per canister of DWPF waste in the Yucca Mountain repository range from 0.25 L/yr to zero. The value of 0.25 L/yr is calculated using the maximum downward flux of 0.5 mm/yr (Section 7.4.1) and assuming that amount enters a vertical borehole with a diameter of 76 cm. Since there is no mechanism to concentrate water onto the waste packages, this is a reasonable upper estimate of the water flux per container. The unfilled head space of a DWPF canister is about 108 L. If cracks were to occur in the container and pour canister near the top (as emplaced), this space could conceivably fill with water in about 400 yr. The resultant glass surface-area-to-solution-volume ratio (SA:V) would be about 2.5 m^{-1}. The anticipated SA:V for glass in the repository range upward from this value depending on canister perforations, water contact, and cracking of glass.

Aines (1986) presented a calculation of the long-term release rate from glass under the conditions where the head-space fills with water. Larger water fluxes were used in that calculation than are now anticipated (1 mm/yr versus 0.5 mm/yr, respectively), but the calculated releases were well below 1 part in 100,000/yr (1.3 g/yr or 0.08 part in 100,000). These calculations involved extrapolation of the boron and lithium normalized loss rates to long times, using the change in silica concentration as glass dissolves. Because of the controlling effects of silica concentration, the reaction was assumed to reach a steady-state rate when the silica added to water in the headspace was equal to the silica lost by overflow of water.

This type of calculation yields larger releases than those assuming that glass breakdown will follow the same square root of time $(t)^{1/2}$ relationship observed in the laboratory at short times, which predicts that glass degradation effectively stops after several years. However, because of the long times involved in reaching steady state (400 yr just to fill the headspace), it is still difficult to calculate release with high precision. One approach to this problem that is currently being investigated is to use explicit geochemical modeling of aqueous species and glass dissolution kinetics to examine the sensitivity of the analysis to changes in various parameters

(Section 8.3.5.10). Geochemical modeling can also be used to examine whether additional interactions not observed in laboratory experiments may occur at long times and to calculate the composition of fluids that are in thermodynamic equilibrium with the phases precipitated on waste glass and with other repository phases.

Much information regarding conservative estimates of release can be obtained from the solubilities of radionuclides by assuming that the water can, at most, remove from the engineered barrier system an amount equivalent to a saturated solution. This is a conservative approach and is very useful for nuclides whose solubilities are well known under tuff rock conditions. It is not as conservative an approach, however, as using the normalized leach rate (LR_i) for a highly soluble, matrix-forming element such as boron or lithium.[1] Regardless of the fate of a radionuclide after release from the glass (i.e., precipitation, formation of a colloid, adsorption, or removal by remaining in solution), the normalized leach rate will not exceed that of boron or lithium; this is the maximum amount of a radionuclide that could be released from the waste form. However, the amount that can be removed by remaining in solution is limited by the solubility of the element in the form it assumes under the pH and redox conditions of the environment. These effects are being investigated (Section 8.3.5.10), and the coupling of solubility effects (limiting release) with kinetically limited breakdown of the glass matrix (providing maximum release rates) will provide accurate bounds on the release of radionuclides from glass. The performance of glass in the repository will be determined from a combination of these release calculations with water influx and container perforation calculations. These will provide estimates of the number of containers likely to contain standing water and the amount and overflow rate of that water, and the number of canisters subject to water dripping onto and off the glass waste. A description of the research and testing program of the glass degradation and radionuclide release process is given in Section 8.3.5.10 under Issue 1.5.

7.4.4 GEOCHEMICAL MODELING CODES: EQ3/6

Chemical interactions among nuclear waste, waste packaging, ground water, gases, and host rock are critical factors in determining the consequences of geologic waste disposal. The removal of radionuclides from the waste forms and their transport through the geologic environment is controlled by chemical processes. Numerical simulation of the various processes will play an important role in the prediction of long-term performance because of the complexity of these interactions, the physical scale, and the 10,000-yr period being considered for the release limit (Coffman et al., 1984). Two of the four performance objectives detailed by the NRC (10 CFR Part 60) are concerned with the engineered barrier system (EBS). Geochemical modeling is needed to carry out the performance assessment to determine whether these objectives will be met.

The first performance objective for the EBS concerns the 300- to 1,000-yr containment period for the waste package during which there will be a thermal pulse from radioactive decay of high-level waste that will cause an increase of temperature of the surrounding rock, water, and gas in the unsaturated zone. This temperature increase will cause the fluid and rock to react with one another to regain equilibrium, resulting in changes in fluid composition and rock mineralogy. Geochemical modeling will be used to characterize the fluid that may interact with and potentially cause a breach in the waste package. The composition of the water, if present, is important because chemical processes (such as corrosion of the container) depend on the composition of the fluid. Geochemical simulation of the heating of the rock-fluid system will characterize the evolving composition of the fluid as it reacts with the rock.

The second performance objective for the EBS encompasses the radionuclide release rate. Geochemical modeling will be used to predict the release of radionuclides from the EBS following a breach of the container and subsequent interactions involving the waste package, aqueous fluids, gases, and surrounding rock. Furthermore, it will permit evaluation of the potential for migration of radionuclides to an aquifer and the accessible environment.

The geochemical modeling codes that are the most general and have the widest application (Nordstrom and Ball, 1984) are in the EQ3/6 package developed by Wolery (1978) and later improved (Wolery, 1983; Delany and Wolery, 1984; Wolery et al., 1984; Jackson and Wolery, 1985; and Delany et al., 1986). Additional development for both codes and the supporting thermochemical data base is essential to account for all important geochemical processes that potentially could affect the performance of the repository (Jacobs and Whatley, 1985).

EQ3/6 was initially developed by Wolery (1978) to model interactions between sea water and basalt in hydrothermal systems at midocean ridges. It is based on principles elucidated by Helgeson (1968) and Helgeson et al. (1970) who pioneered computerized geochemical modeling. Wolery introduced improved numerical methods for solving the requisite equations, which eliminated computational problems in Helgeson's (1968) code, and reduced computer time for a modeling run by several orders of magnitude. This was an important step because of the large amount of computer time required to run simulation studies for geochemical processes involving complex systems such as for the proposed repository.

Improvements in EQ3/6 since then included adding dissolution kinetics capability and expanding the thermodynamic data base. Simulations with EQ3/6 included modeling of the dissolution of uranium(IV) oxide in granitic ground water (Wolery, 1980). Most of the funding to support EQ3/6 at Lawrence Livermore National Laboratories since 1982 has come from the NNWSI Project, with additional funding from the Salt Repository Project and the Basalt Waste Isolation Project. Recent developments during this later time period are described in the following sections. Plans for future development can be found in Section 8.3.5.10.

7.4.4.1 EQ3NR, a computer program for speciation-solubility calculations

EQ3NR (Wolery, 1983) calculates the thermodynamic state of an aqueous solution, which involves the determination of the distribution of activities of aqueous species. The degree of formation of ion pairs, concentrations of complex aqueous species, ionic strength, Eh, pH, and essentially all the variables needed to fully characterize the solution are evaluated. This allows calculation of a quantitative measure of the thermodynamic state of the solution with respect to solid phases (undersaturated, in equilibrium, or supersaturated). This speciation-solubility calculation is useful as a method to test if a solution is in equilibrium with a particular solid phase. EQ3NR is widely used by experimentalists both as a tool for optimizing experimental conditions and as a method for extracting thermodynamic and kinetic information from experimental results. EQ3NR also initializes the solution composition used as input to reaction-path calculations made by the EQ6 code.

7.4.4.2 EQ6, a computer program for reaction-path modeling

EQ6 (Wolery, 1979) calculates reaction paths and mass transfer in dynamically reacting fluid-rock systems by considering consecutive stages of partial equilibrium as equilibrium is approached and then attained. For example, when a mineral reacts with a fluid with which it is not in equilibrium, a series of solid phases usually precipitate and dissolve before saturation with the initial reactant mineral is reached. EQ6 tracks the changes in mineralogy and corresponding changes in fluid composition as the fluid and the rock react. Changes in pH, oxidation-reduction potential, and other parameters are also output by the code. To calculate precipitation and dissolution events, small amounts of the reactant mineral are dissolved into the solution and the solution is tested for equilibrium. Product phases will precipitate in the case of supersaturation and dissolve in the case of undersaturation. When the solution is in equilibrium with all product solid phases, more of the reactant mineral is allowed to dissolve. The rate at which the reactant mineral dissolves can either be specified arbitrarily by the user or calculated according to one of several kinetic rate laws programmed into EQ6. The reaction path terminates when the solution becomes saturated with respect to the initial reactant mineral, or when the reactant mineral is totally consumed.

7.4.4.3 Thermodynamic data base

The thermodynamic data base is a summary of the available thermochemical data for aqueous species, solids, and gases in tabular form that are necessary to serve as input to the geochemical modeling codes EQ3NR and EQ6. These values have been gathered from the available literature and inserted into the data base format as part of an ongoing effort. Data are made compatible with Committee on Data for Science and Technology (CODATA) task-group-recommended key values and with Nuclear Energy Agency (NEA) data values that are released after peer reviews. The portion of the data file that represents rock-forming minerals is taken from SUPCRT (Helgeson et al., 1978), a data base maintained at the University of California, Berkeley.

Thermodynamic data are processed through a code, MCRT, that checks for thermodynamic consistency, extrapolates heat capacity functions with temperature, and generates data blocks for insertion into the master EQ3/6 thermodynamic data file DATAO. MCRT contains an internal data base of free energies and enthalpies of formation, third law entropies, and heat capacities of specified reactions that are commonly used in geochemical calculations. Thermodynamic data in DATAO are processed through an additional code, EQTL, that checks for mass and charge balance, fits all data to a predetermined temperature grid, and writes the data files that are read directly by the EQ3NR and EQ6 codes.

7.4.4.4 Theory and code development

EQ3/6 simulations of experimental results have revealed chemical processes that should be accounted for in the code to ensure better matches between simulation results and experimental measurements. Credible predictions of geochemical interactions on a scale of up to 10,000 yr require continual updates of EQ3/6 to account for these complex processes. Recent progress in this regard has been made in the ability of the codes to account for solid solutions, precipitation kinetics, and fixed fugacities for gases. Future theory and code development activities will be focused on incorporating a geochemical flow model, studying the effects of nucleation and substances inhibiting precipitation on kinetics, including an option for variable fugacity, extending the solid solutions model to include an option for site mixing, and adding an equilibrium sorption model (Section 8.3.5.10).

The treatment of solid solutions in EQ3/6 has been improved by incorporating a new algorithm for molecular mixing models (Bourcier, 1985). This model is mathematically equivalent to a single-site mixing model. The data base has also been expanded to include calcite-structured solid solutions based on the model of Sverjensky (1984). Solid solutions are common in nature and occur both as primary and secondary phases. At Yucca Mountain, the principal primary minerals present as solid solutions are the feldspars and biotite. Secondary solid solution phases, which result from water-rock interactions, include the clays and zeolites. To compute chemical equilibrium and simulate reaction path models of these interactions, it is necessary to incorporate solid solution models to account for phases with variable composition.

A precipitation-kinetics option has been added to EQ6 that complements a preexisting capability to model dissolution kinetics (Delany et al., 1986). Several precipitation rate laws have been programmed into EQ6 as well as options for activity-term expressions and transition state theory. Transition state theory was reexamined, resulting in new theoretical insights that will provide the foundation for further improvements necessary to model chemical kinetics (Wolery, 1986). Rates of reactions must be taken into account in cases where simulations are made of laboratory experiments in which reaction kinetics are sluggish and equilibrium is not attained. For example, in the hydrothermal tests discussed in Section 7.4.1.8 involving the Topopah Spring tuff and well J-13 water, it is observed that the phase controlling the solubility of silica is cristobalite. However, cristobalite is thermodynamically metastable under the test conditions because quartz is a thermodynamically more stable (less soluble) form of silica. Cristobalite solubility is maintained in the short term only by the slow growth kinetics of quartz, which is present in the tuff. It is also important to be able to model laboratory experiments in real time in order to understand the factors that affect extrapolation to longer time periods. Without geochemical modeling to allow this extrapolation, one would have to assume that short-term conditions observed in experiments apply to the long term. Modeling calculations, on the other hand, allow an evaluation of whether laboratory results do or do not represent thermodynamic equilibrium or may be kinetically controlled.

A fixed-fugacity option has been added to the EQ6 geochemical reaction path code (Delany and Wolery, 1984). Ground-water systems that are either open or closed to the atmosphere as well as buffered laboratory experiments can now be modeled. Permitting the fugacity of any gas to be set at a fixed value allows simulation of the effect of rapid chemical exchange with a large external gas reservoir such as the unsaturated zone at Yucca Mountain, which is open to the atmosphere. In addition, quantities of various gases (e.g., carbon dioxide, methane, and hydrogen sulfide) may be produced by chemical reactions in the repository site. It is therefore imperative that geochemical models have the capability to include gases in the systems under consideration. For example, if a species has a greater proportion of its mass in the gas phase than the aqueous phase, the fugacity of the gas phase can be assumed to control the fugacity of the aqueous phase. The paths and rates of chemical reactions in the aqueous phase can be predicted when so constrained.

7.4.4.5 <u>Thermodynamic data base development</u>

Expansion and refinement of the data base (Section 8.3.5.10) is required to model the complex chemical systems involved in radioactive waste disposal. Sensitivity analysis will be reconducted to (1) identify reactions that can be neglected and (2) recommend additional experiments for improving and obtaining important thermodynamic data. A plan for experimental and theoretical work has been developed to obtain needed thermodynamic data and to resolve inconsistencies in existing data. This plan resulted from a review of the thermodynamic data for the actinides, which uncovered the need to conduct laboratory experiments and to obtain thermochemical data for solubility-limiting solid phases and solution species. An example of how the correspondence, or lack of it, between experimental results and EQ3/6 predictions is being used to identify phases for further thermodynamic study is given below.

The concentration of uranium in solution following spent fuel tests (Oversby and Wilson, 1985) reacting with well J-13 water at 25°C during the first few days is about 5 ppm. The concentration decreases to about 1 ppm after 240 days as equilibrium solubility is approached. If the solubility is controlled by an oxide, shoepite, EQ3NR predicts that 65 ppm uranium should be in solution at equilibrium. On the other hand, if a silicate, soddyite, controls uranium solubility, EQ3NR predicts an equilibrium concentration of 0.2 ppm uranium using provisional thermodynamic data. Clearly, soddyite is a candidate among others (e.g., boltwoodite) for further thermodynamic study.

To adequately model the requisite geochemical processes that are critical in the evaluation of performance objectives, it will be necessary to generate a resource body of data on possible solubility-controlling solid phases and to determine the free energies of solubility reactions needed by EQ3/6. Some redetermination of existing data may be necessary because the solid phases were not adequately characterized with regard to structure and composition. A library of infrared spectra is currently being generated and will be used to identify spectra of unknown solids from spent fuel and waste form dissolution experiments. In addition, this spectroscopic data will be used together with bulk and acoustic properties and structural data to predict thermodynamic functions using an averaging model developed by Kieffer (1985). This model is based on the theory of lattice vibrational properties of minerals and on observed trends in the high-temperature thermodynamic properties of silicates. The code for this model is presently available at Lawrence Livermore National Laboratory. Model values of heat capacities and third-law entropies have been shown to be in good agreement with experimental values to 1,000°C for more than 30 minerals. Heat capacities and third-law entropies are needed to make thermodynamic calculations of free energies of formation of the minerals for which spectra have been collected.

Solubility product constants are not usually sufficient to predict accurate solution concentrations. Hydrolysis products and other complexes can increase solubilities above predicted values based on solubility product alone. Therefore, reliable formation constants for the major solution species that are likely to form under site-specific ground-water conditions are needed to make accurate EQ3/6 calculations. However, gaps and inconsistencies exist in the present thermodynamic data base. One reason is that the indirect methods (e.g., nuclear counting methods) used to determine the amount of the radionuclide in solution do not provide information, critical to geochemical modeling, on the nature of the solution species. Another reason is that some specie's concentrations are at such low levels that conventional spectroscopic techniques (ultraviolet, visible, near infrared) do not have sufficient sensitivity. Attempts are presently under way to apply more sensitive laser techniques to these measurements. The favored approach is to measure directly the deposited energy resulting from the adsorption process rather than the more conventional method of measuring the attenuation in a light beam. This technique, called pulsed photoacoustic spectroscopy, has been shown to be several orders of magnitude more sensitive than the conventional approach. A West German group has demonstrated that this technique can be applied to measurements of actinide speciation at submicromolar concentrations (Stumpe et al., 1984).

A pulsed photoacoustic system is presently being developed using an existing Nd-YAG (neodymium-yttrium/argon gas) pumped dye laser and associated computer-control instrumentation. The capabilities and sensitivity limitations of this system for characterizing waste radionuclide species in aqueous solutions is being evaluated.

7.4.4.6 Applications: water-rock interactions

Although additional geochemical model development is planned (Section 8.3.5.10), several applied results have clearly demonstrated the utility of the approach. These preliminary calculations indicate a good agreement between predicted phase relations and solution composition and those observed in laboratory experiments. It is not possible to conduct all the laboratory tests necessary to cover all possible repository conditions (e.g., varying temperature, solution composition, and rock:water ratios). The size of such a test matrix is prohibitively large with respect to available resources. The close match between geochemical modeling calculations and laboratory tests demonstrates that the code has the capability to quickly and relatively inexpensively provide credible predictions of the results of fluid rock interactions. Simulation predictions are necessary to carry out performance assessments for containment and release rates over time periods of 10,000 yr, which are not attainable in the laboratory.

Experimental results of reacting Topopah Spring tuff with well J-13 water at room and at elevated temperatures were compared with simulation results modeled using EQ3/6 (Knauss et al., 1984; Delany, 1985). The details of this comparison are in Section 7.4.1.8. There was good agreement between the measured and the calculated solution composition. As more complete data on phase composition, dissolution kinetics, and thermodynamic properties become available for primary and secondary phases, it is to be expected that an even closer match between EQ3/6 geochemical modeling results and the observations from laboratory tests will result. This will provide a critical component for making the predictions needed for performance assessment.

In addition to simulating laboratory experiments involving water-rock reactions, EQ3/6 can be used in parametric analyses to determine which factors must be controlled during experimentation and the relative magnitudes of the effects of these factors. The EQ6 code was used to simulate the dissolution of pure albite in distilled water at 25°C for various carbon dioxide pressures (Delany and Wolery, 1984). This was done to investigate the influence of constant fugacities on reacting aqueous geochemical systems. At higher carbon dioxide fugacities, more albite must dissolve to reach saturation. This is expected because, as carbon dioxide is added to maintain the higher fugacities, more acid is created, which in turn hydrolyzes more albite. At lower fugacities of carbon dioxide, the sequence of secondary mineral formation is gibbsite, quartz, and paragonite. With increasing fugacity of carbon dioxide, gibbsite does not appear, and the product assemblage is dominated by quartz and kaolinite. More of the secondary reaction products are formed at higher carbon dioxide fugacities, which is expected because more of the albite dissolves. The secondary phases that formed in these calculations were those that are thermodynamically most stable according to the data base. However, this does not always result in the most realistic model because thermodynamic data is lacking or kinetic effects have not been accounted for. For example, a clay of roughly the composition of sodium montmorillonite, for which thermodynamic data is not available, is a more likely product than paragonite. And a solution may become supersaturated with respect to a phase, such as quartz, rather than precipitating the phase upon reaching saturation. There is a clear need for better kinetic models of mineral precipitation that can describe these phenomena.

7.4.5 WASTE PACKAGE POSTCLOSURE PERFORMANCE ASSESSMENT

7.4.5.1 Introduction

The long-term performance of nuclear waste packages must be assessed to qualify the waste package designs with respect to postclosure performance requirements. The performance requirements are presented in Section 7.2. The processes affecting long-term performance are identified in earlier parts of Section 7.4; knowledge to date, information needs, and investigations planned to satisfy the information needs are presented in Sections 7.4, 7.5, and 8.3. Performance assessment as an activity takes information from the other waste package investigations, constructs coupled models, validates these models, and assesses waste package designs in terms of the performance requirements. The outputs address both performance of an ensemble of waste packages and reliability in the waste packages' performance.

The generic postclosure functional requirements of the waste package were listed in Section 7.2.1.2. The plan for determining the final performance allocation is described in Section 8.3.5.2. Performance criteria addressing the NRC 10 CFR 60.113 are terms of (1) time to loss of substantially complete containment and (2) release rates of radionuclides of concern from the engineered barrier system. The parameters to be calculated in the assessment are as follows: (1) time to failure of individual containers; (2) time to initiation of release for each radionuclide; and (3) release rates and quantities from the waste package of all radionuclides of concern, as a function of time. These parameters will be determined for an ensemble of waste packages. To document regulatory compliance, the aggregate of these measures will be compared with the postclosure performance criteria given in Section 7.2.3.2. Reliability results to be assessed are discussed later in this introduction.

The NRC design criteria in 10 CFR 60.135(a)(1) concerns waste package interactions with its emplacement environment. The waste package performance assessment evaluates impacts of the waste package on its environment through heat, mechanical, radiation, and chemical effects. The effects of the environment on waste package performance are included in the assessment. The magnitudes of heat, mechanical, radiation, and chemical impacts will be available for repository-scale evaluation of the effects of these impacts on repository performance.

Performance assessment as an activity within the waste package investigations creates information needs only with respect to the methodology used for assessment. Basic scientific and engineering studies necessary to predict performance of waste package components and to validate component performance are periodically described earlier in this chapter and in Section 8.3. These studies will satisfy information needs through investigations during site characterization. The information needs of performance assessment are served by those investigations and, therefore, place requirements on those investigations with regard to data type and sensitivity. To understand these data requirements, a brief discussion of performance assessment methodology is useful.

Given the long periods discussed in the criteria and given the general complexity of the problem, long-term assessments will use computational models. It is the task of performance assessment to construct and validate these computational models and then to analyze waste package designs to demonstrate that selected designs perform as required. These analyses will also guide the design process by allowing comparison of alternative waste package designs. Such models can help determine the sensitivity of performance to environmental and design parameters. Further, the integrated performance calculations will aid in determining the adequacy of the current envelope of environmental conditions used in the individual waste package process tests, as determined by one- or two-process detailed models.

Stephens et al. (1986) describes attributes that waste package performance assessment methodologies must have in addition to those previously described. In particular, since the environment and processes affecting performance in a mined repository are subject to uncertainties, computational models that integrate the effects of these processes must also reflect those uncertainties. To allow for these uncertainties in the licensing processes, the NRC intends to use a reasonable assurance standard for determining that regulatory criteria are met (NRC, 1986). The NRC has identified probabilistic reliability analysis as a means to supply reasonable assurance. To date no integrated performance assessments have been performed on NNWSI Project conceptual waste package designs. Calculations of thermal effects and mechanical effects on packages have been made (Hockman and O'Neal, 1984; O'Neal et al., 1984), and the results of those calculations have been considered as part of the conceptual designs that appear in this document. Therefore, the waste package performance assessment methodology intends to include a probabilistic reliability analysis method. This approach will allow waste package performance assessment calculations to provide predictions of time-to-loss-of-containment and release rate as a function of time, as well as the probability distributions of these predicted performance measures. In addition, models must allow consideration of anticipated and unanticipated events. This attribute allows the analysis of performance resulting from extreme event scenarios for use as input to the total repository performance assessment. Finally, these calculations must also incorporate those needed to assess performance with respect to preclosure design criteria.

As stated earlier, methods used for performance assessment will incorporate the results of the studies of physical and chemical processes discussed elsewhere in this chapter. These results, in the form of data or submodels, must be integrated into a single model to represent the interaction or coupling that would exist among processes in a repository setting. The current approach to waste package performance assessment calculations relies on coupling of processes at the systems level; that is, process submodels are joined through an explicit set of data transfers. The data transfers and their timing and logic are represented in a driver model.

The use of parametric submodels for physical and chemical processes adds requirements to the data developed during the previously discussed investigations. First, the detailed studies to be performed on waste package processes must be analyzed for sensitivity of parameters; and second, once the sensitive parameters are identified, descriptive parametric relationships for those processes must be developed. Those relationships may take several forms (e.g., data tables, regressions, and simplified mechanistic models);

however, they must be valid and computationally efficient. Therefore, a third requirement is that submodels that describe processes be validated.

Some indication of the probability of parameter values must also be supplied to perform the probabilistic analysis required for NRC review (NRC, 1983, 1986). At a minimum, sufficient information will be required to perform bounding value calculations of performance so that conservative estimates of performance can be made. Finally, integrated tests of the interaction of several processes will be needed to provide a means for partial validation of the system model.

Clearly, the waste package functions within the larger setting of the total repository. Total system performance assessment is discussed in Chapter 8. Scenarios for anticipated and unanticipated events must be supplied by the total system performance assessment. Although these scenarios are data needs for the waste package performance assessment, they do not generate information needs to be satisfied within waste package activities. This information will be developed in the total system performance activities and will not be discussed in detail in this section. It should be recognized that bounding estimates of performance will often depend on the scenarios for these events.

In summary, the information needs of performance assessment are basically satisfied by the needs of the studies supporting the assessments. The work done to address these needs is discussed in greater detail earlier in this chapter; the plans for collecting additional information is described in Chapter 8. Performance assessment requires that those investigations assess the relative sensitivity of the parameters affecting the individual processes and formulate predictive relationships from these parameters. Further, for those parameters used by the submodels, information describing the probability assigned to given parameter values is also necessary. Finally, data for verification and validation of the predictive methods used must be supplied.

Our approach to developing models for the system performance assessment has several stages. First, a model is being developed that integrates the processes affecting the postclosure performance of a nuclear waste package, with deterministic outcome under a given set of conditions and design parameters. Second, recognition will be given to the distribution of waste heat loadings, emplacement times, and fabricational and environmental parameters in the ensemble of waste packages. Repeated calculations with the model of a single waste package can assess the postclosure performance of the ensemble of waste packages. Third, a probabilistic reliability analysis methodology will be selected and implemented to assess reliability in the waste packages' performance. This methodology will include tools of failure modes and effects analysis, anticipated and unanticipated events scenario analysis, and propagation of input parameter uncertainty, as appropriate. The reliability topics are of two related types: (1) what is the reliability with which the waste package performance values will meet or surpass the postclosure requirements and (2) what is the estimated distribution of the waste package performance values. The products of each of the three stages may be used in sensitivity analysis to quantify the sensitivity of the results of the integrated process to input parameters and to subprocesses.

The deterministic system model of a nuclear waste package performance is being developed in three cycles, to track phases in waste package design, (i.e., conceptual design, advanced conceptual design, and license application design). The purposes of the first-cycle model are to get some informative results from incorporating important processes in the model, to explore for effects of coupling that were not apparent from models of individual processes, and to explore for any information needs not already recognized by the other waste package investigations (Sections 7.5 and 8.3). The conceptual model is described in the following paragraphs to provide the performance modeling perspective for data requirements. Data from site characterization are required for model synthesis, verification, validation, final waste package performance assessment, and final reliability assessment. At each design stage, assessments of waste package performance will serve as a design input.

The following parts of Section 7.4.5 describe the conceptual model development of the first cycle of the deterministic system model. The conceptual model development (O'Connell and Drach, 1986) was based on information in drafts of the SCP, Chapter 7, and on internal reviews by experienced personnel within the NNWSI Project. Section 7.4.5.2 briefly discusses the principal processes that affect waste package performance. This identification is already a modeling step in that it involves selection of the processes to include in the first system model and corresponding computer code. Section 7.4.5.3 is a brief aside to explain that the site-specific processes at the Yucca Mountain unsaturated tuff site are sufficiently different form those included in previous waste package models to require development of a new model. Section 7.4.5.4 describes the conceptual model for the first cycle of the NNWSI deterministic system model.

Included in this background material are summarized results of preliminary performance assessment calculations for some individual physical and chemical processes. Note, however, that these calculations do not represent integrated performance assessment results. The results presented are first estimates of bounding calculations for individual processes, often under different assumed conditions. However, they provide some initial insight into the expected performance of individual waste package components based on work to date. Section 7.4.5.5 identifies some possible techniques for use in probabilistic reliability analysis.

7.4.5.2 Processes affecting waste package performance

This subsection summarizes the important processes affecting the waste package performance. More detail, process information status, and modeling status are provided in later subsections. The summary is based principally on the detailed presentations in Section 7.4.2 on metal barriers and in Section 7.4.3 on waste form performance research and testing. The processes described are assumed to be acting on a waste package. Prospective waste package designs are described in Section 7.3.

The current waste package design provides a sealed metal container barrier. For a single container, a containment failure is assumed to occur when the barrier integrity is lost. This integrity loss can develop by either mechanical means (e.g., rupture due to stress) or chemical means (e.g., uniform corrosion, stress corrosion cracking, or pitting corrosion). Processes leading to containment failure are also influenced by the nearby external environment (e.g., rock failures that can cause localized stress on the waste package, ground-water hydrology, and ground-water chemistry). The processes leading to containment failure are also influenced by the interaction between the internal and external environments. Examples include (1) the heat generated by the radioactive waste and transferred to the external rock, thus establishing a temperature field, and (2) gamma radiation generated by the waste and attenuated by the waste and metal barriers producing a net gamma ray flux at the surfaces of the metal barriers. The gamma ray flux can cause radiolysis in the water and perhaps can increase the corrosion rate. Loss of containment for the ensemble of containers occurs when enough individual containers have experienced failure so that the consequential release of radionuclides exceeds some threshold value.

For spent fuel, the Zircaloy cladding may provide an additional barrier to release of materials from the fuel pellets and, even when partially breached, may help limit the rate of release from the waste. A small part of the spent fuel radioactive waste consists of activated elements in the Zircaloy itself and in the stainless steel framework of the fuel assemblies; for these waste components, the cladding or metal components degradation rate determines the release rate of radionuclides.

Processes influencing or leading to the loss of waste package containment can be listed as follows:

1. Radiation.

   a. Gamma ray source.
   b. Gamma ray attenuation.

2. Thermal.

   a. Heat source from radioactive decay.
   b. Heat transfer, temperature field effects.

3. Mechanical loads.

   a. External (pressure, general, or localized).
   b. Internal (pressure).
   c. Thermal expansion.

4. Yielding.

   a. Ductile rupture.
   b. Crack extension.
   c. Brittle rupture from a crack.

5. Ground-water movement and chemistry.

   a. Flow surrounding engineered barrier system.
   b. Flow mechanisms for water contacting the waste package.

c. Transport in nearby host rock.
d. Water volume available for contact with waste package.

6. Corrosion.

   a. Uniform corrosion.
   b. Stress corrosion cracking.
   c. Pitting and crevice corrosion.
   d. Galvanic corrosion.
   e. Other corrosion modes.

Some of these processes are continually in a quasi-steady state. For example, with some time lag during the first few years after emplacement, the temperature distribution is essentially in equilibrium with the heat generation, which changes very slowly. The stress distribution forms a static equilibrium with the internal and external forces. Other processes cause gradual changes (e.g., uniform corrosion that reduces barrier thickness). Some processes cause abrupt and discrete changes (e.g., perforation of a barrier due to mechanical rupture or due to corrosion penetration reaching the full thickness of the barrier).

The existence and rate of waste release from the waste package are dependent on three processes:

1. Existence of a penetration of the metal container, opening a potential path between the waste form and the external environment.

2. Alteration of the waste form, converting some part of the waste into mobile form.

3. Transport of the mobile form of the waste from the waste form to the external medium (the rock around the borehole).

The major part of the waste is in the spent fuel's uranium-oxide-matrix fuel pellets or in the glass matrix for reprocessed waste. The mobilization of this waste in ground water is assumed to be governed by its solubility in the local ground water. Transport from the engineered barrier system (EBS) is through or with ground water. For the spent fuel, there are a few exceptions: (1) some radioactive elements due to activation of structural materials are in the Zircaloy cladding and in the metal framework of spent fuel assemblies; (2) some elements may be released in gaseous form, principally carbon-14 in the oxidized form of carbon dioxide; and (3) a gap-grain-boundary inventory of the waste is available in a form that can be released rapidly into water after the Zircaloy is breached (Oversby and Wilson, 1985).

The rate of waste form alteration depends on the amount and duration of ground-water contact with the waste form (Section 7.4.3). This contact, in turn, depends on the ground-water flow and on the waste container conditions (e.g., one small breach, several small breaches, many breaches, or essentially dismantled). The rate of waste form alteration may also depend on the temperature, water chemistry, gamma ray flux at the waste form-water interface, and waste form material damage accumulated due to alpha decays and spontaneous fissions within the waste form. The rate of waste form alteration will also depend on the rate of mobile waste transport away from the waste form-water interface.

The mobilized waste may increase the gamma radiation flux near the barrier surface and hence may increase the rate of corrosion of the breached barriers. This is not expected to be an important factor, however, because (1) the radiation source strength will be low by the time the barriers become breached (Croff and Alexander, 1980) and (2) only a small fraction of the waste will be in mobile form in the waste package volume (Oversby and Wilson, 1985).

In summary, the processes affecting the rates of waste form degradation and release from the waste package include most of those listed for barrier failure, plus the following:

1. Radiation.

    a. Alpha radiation (and spontaneous fission) source.
    b. Alpha (and fission) integrated quantities.

2. Waste form alteration.

3. Transport of mobile waste.

The parameters affecting each process and the interactions between processes are described in Section 7.4.5.4.

7.4.5.3 Earlier models of similar scope

Earlier deterministic models of waste package performance are BARIER (Stula et al., 1980) and WAPPA (INTERA, 1983). These waste package system models have relatively simple submodels of each process and treat the interactions of the processes over time, as does our approach.

WAPPA was examined and tested for possible application to the NNWSI Project waste package. That WAPPA provided a good starting list of phenomena and concerns that should be addressed. However, specific aspects of the NNWSI Project candidate repository indicated that additional submodels were needed for waste package design and site-specific conditions. For example, the NNWSI Project waste forms include spent fuel in several geometric arrangements. The heat transfer problem is thus somewhat different. A gamma ray dose attenuation model adequate for sensitivity analysis was needed. WAPPA assumes a location below the water table. The NNWSI Project repository location is above the water table in the unsaturated zone. Hence, the mechanisms of ground-water flow, water contact with the waste package, waste form alteration, and mobilized waste transport differ substantially from those assumed in the WAPPA formulation. Therefore, it was decided to develop a new waste package performance program specific to the NNWSI Project site conditions and design parameters.

7.4.5.4 Nevada Nuclear Waste Storage Investigations Project waste package system model description

This section briefly describes the system model for waste package performance currently under development (O'Connell and Drach, 1986). The system model must treat the waste packages and processes described earlier. The system modeling approach is to model the processes separately as far as possible and then couple these models through an explicit set of data transfers. The process models are simplified ones, combining parametric models and sets of data tables. Detailed models of one or two processes are available or are being developed. These detailed models calibrate the degree of accuracy of the simpler models or provide the data tables. The system model geometry has one dimension of variation, the radial direction in a cylindrical geometry. End effects will be examined in future work. Currently only the more detailed supporting models can treat end effects.

The process models include the following seven models:

1. Radiation model.

2. Thermal model.

3. Mechanical model.

4. Waste package environment model.

5. Corrosion model.

6. Waste form alteration model.

7. Waste transport model (covering transport within the waste package system).

Each model may consist of several interacting or related submodels with diverse time characteristics as noted in Section 7.4.5.2. Functions of the individual models are described and then data interactions and the sequencing of these interactions will be described. The data interactions lead to a grouping of submodels by time characteristics rather than solely by model topic.

7.4.5.4.1 Waste package geometry

The waste package model describes the initial geometry and properties and tracks the current conditions of the waste package in time. A cylindrical geometry is assumed with variability only in the radial direction. Because of the large length-to-radius ratio of the waste packages, an assumption of no axial variation seems suitable over most of the length of the waste package. An examination of processes active in directions other than the radial directions will be conducted to substantiate this assumption. If axial variations are found important, the model will be modified as required. Different conditions at the ends of the waste package will require evaluation in future work. Cylindrical symmetry seems suitable in most instances but its use for some types of spent fuel assemblies (Figure 7-2) will represent a coarser degree of approximation and will require comparison with two-dimensional calculations.

The Zircaloy cladding may serve as an additional release barrier protecting the spent fuel waste form. This barrier does not have the common radial center of the outer barrier(s) but does have the logical relation of one barrier within another. Such barriers, as well as the waste forms, will be tabulated with their logical contained-within-barrier relations and the larger-scale annulus within which they are located.

7.4.5.4.2 Radiation model

The radiation model treats radiation sources and radiation doses in the waste package. The source submodel calculates the heat generation rate, the radiation generation rates, and the radionuclide inventory. The gamma ray dose submodel calculates the gamma ray absorbed dose rate in water at locations in the waste package where corrosion or waste form alteration may take place. The waste form dose model calculates the dose, in terms of atomic displacements per unit volume, due to alpha particles and spontaneous fission. The effects of these dose rates and doses will be considered in the corrosion and waste form alteration models, respectively.

The source submodel interpolates data from tables generated by the ORIGEN2 code (Croff, 1980). ORIGEN2 calculates the burnup of nuclear fuel and the buildup and decay of fission products, activation products, and decay products during reactor operation and after the fuel has been removed from the reactor. ORIGEN2 also accounts for the partitioning of elements if reprocessing is done. The ORIGEN2 tables provide data, as a function of time since removal from the reactor, for a unit quantity of waste derived from nuclear fuel of a specified type and burnup. The unit quantity is the amount of derived waste (or spent fuel) per metric ton of heavy metal (MTHM) originally in the fuel. The source model scales from the tabulated unit quantity to the quantity of waste in the waste package.

The waste form is both a source and an attenuator of gamma rays. The metal barriers are attenuators of gamma rays. If and when water is present at the surface of a barrier or of a waste form, the gamma ray absorbed dose rate in water is significant because of the resulting radiolysis of either liquid or gaseous water. This radiolysis may cause enhanced corrosion or enhanced alteration of the surface of the barrier or waste form.

The attenuation and absorbed dose rate from gamma rays are complex processes. A simplified model cannot be expected to calculate the results from first principals. Rather, reference data from measurements or from detailed calculations is needed. For reference data, the detailed code MORSE-L (Wilcox, 1972), which calculates radiation transport using a Monte Carlo method, will be used. This code has a history of extensive use. Calculations for a spent fuel canister emplacement (Wilcox and Van Konynenburg, 1981) have been validated by measurements (Van Konynenburg, 1984) taken in a granite repository.

The gamma ray dose submodel within the system model will scale or adapt results from a MORSE-L reference calculation to a range of similar waste forms and waste package designs. Two scalings will be performed: the first scales the absorbed dose rate at the waste form surface, and the second scales the attenuation factors between the waste form surface and the barrier surfaces. The waste forms are self-shielding sources, with radii considerably greater than their gamma ray energy absorbtion lengths. This allows a simplified scaling of results for dose rate at the surface. The absorbed dose rate at the waste form surface is scaled linearly with gamma ray source strength in (average MeV x number)/(sec x m^3), and inversely with source mass density. There is to first order no effect from the radius of the waste form, and within a limited range of waste form materials, only a small adjustment for the percentages of atomic composition of the waste form. The absorbed dose rate at the waste form surface times this attenuation factor gives the absorbed dose rate at a barrier surface.

The attenuation factor for absorbed dose rate depends primarily on the mass thickness in kg/m^2 between the waste form surface and the point of observation, with only a small adjustment for the percentages of atomic composition of this intervening material; and it depends on a geometrical factor of one per radius due to the cylindrical geometry. Thus a scaling from a reference calculation can be done between points with the same mass thickness between the waste form surface and the point of observation.

The accuracy of this scaling approach will need to be validated by comparison with a series of MORSE-L calculations, which will determine whether a few or many reference calculations are needed to provide a basis for scaling to a range of waste package designs.

The waste form dose model for atomic displacements per unit volume is simple. Alpha particles and fission fragments have short ranges but cause atomic displacements as a side effect of their slowing-down process. These displacements do not fully anneal at the temperatures encountered in high-level waste disposal. Experimental factors are used for the number of atomic displacements per alpha particle and per spontaneous fission. The model accumulates these displacements over time generated by the respective radiation source rates. For spent fuel, the displacements caused by these processes during emplacement are not expected to be a significant factor with respect to waste package performance since most displacement will have occurred in the reactor environment (Woodley, 1983).

7.4.5.4.3 Thermal model

The thermal model calculates temperature as a function of radial position in the waste package. As input, the model requires (as functions of time) the heat generation rate of the waste form and the boundary temperature at the waste package-borehole wall interface. This boundary temperature will be calculated by a more detailed model that handles dynamic conditions and three-dimensional geometry of the waste containers and repository configurations. The thermal submodel assumes a steady-state condition inside the waste package. The temperature gradient across each component of the waste package is that needed to transport the power generated in the waste form.

The thermal model results will be used as inputs to the mechanical, corrosion, package environment, release, and waste transport models.

A detailed model, TACO2D, a code developed by Burns (1982), has three-dimensional and transient capabilities. The model has been extensively used. The three-dimensional geometry of the waste package and emplacement can be reduced to the two-dimensional calculation in TACO2D by using axisymmetric or plane symmetry and the regular repetition of waste package emplacement locations (Stein et al. 1984). TACO2D handles dynamic conditions, a necessary capability when the repository rock mass is included in the heat transfer calculation. TACO2D does not include hydrothermal effects. WAFE (Travis, 1984), a hydrothermal model currently under testing and development, has the geometric and dynamic capabilities of TACO2D and also considers the heat-transfer effects of vaporizing and recondensing the pore water in an unsaturated medium. This model is also currently being used to characterize the hydrothermal environment surrounding the waste package. TOUGH (Pruess and Wang, 1984) is another hydrothermal model currently under testing.

The steady-state assumption inside the waste package neglects the heat capacity of the waste package. This leads to only a small overestimate of waste form peak temperature.

The steady-state one-dimensional thermal model to be used in the system model is nearly the same as that of WAPPA (INTERA, 1983). Its results agree with those of TACO2D, which was used to calculate thermal histories of reference waste packages (Hockman and O'Neal, 1984) to within 7 percent. The disagreement occurs only at early times near the time of peak temperature, based on rise above ambient temperature.

7.4.5.4.4 Mechanical model

The mechanical model calculates the stress at various locations in the waste package. Contributions will come from external loads, thermal expansion and thermal gradient stresses, volume expansion of corrosion products, and residual stresses from waste package fabrication. The mechanical model will check for mechanical failure modes such as yielding and unstable propagation of a crack (i.e., rupture). Stable crack extension that could result in a physically larger breach in the container will be considered for a future model version.

The surface stress from the mechanical model will be used as input to the corrosion model. The waste form mechanical integrity condition will be used as input to the waste form release model. Changes in barrier integrity may also affect the thermal behavior and thus will be used as input for the thermal model as well.

The NIKE2D code (Hallquist, 1983) has been selected as a detailed model for mechanical stress analysis of the waste package. To further check the mechanical model, the approximate formulas for thin-walled pipes given in the ASME Boiler and Pressure Vessel Code (ASME, 1983) will be used. These formulas may be compared with the mechanical model results when thin-walled metallic barriers in the waste package are modeled.

7.4.5.4.5 Waste package environment model

The waste package environment model will evaluate the flow of water, steam, and air that can affect the waste package barriers and the waste form and assist in waste transport. This model will ultimately include fluid flow rates, paths, and chemical contents for flows to, within, and from the waste package. The waste package environment model should describe the process of water contact with the waste package or waste form. Mechanisms of contact might include unsaturated porous flow, localized dripping, and standing water in a partially degraded waste package.

The mechanisms and quantity of water flow may limit the amounts of corrosion and of waste form alteration. A description of detailed hydrothermal modeling of fluid flow in the host rock surrounding the waste package fluid flow is a future information need (Sections 7.4.1 and 8.3.5.10). For the present system model, a simplified bounding model of the fluid flow is being used. The water flow rate is bounded as described below. The water chemical content is effectively bounded by using experimental corrosion and alteration rates determined with appropriate water samples.

As input, the simplified bounding model takes the time of reestablishment of the liquid water continuum, at the rock boundary of the waste package emplacement borehole, from a more detailed model or evaluation. It also uses the regional ground-water flux value, or a bounding value on that flux. The simplified model assumes that, starting with the time of rewetting, all the regional downward water flux that intersects rock disturbed by repository construction comes into contact with the waste packages. Each package is assigned a catchment area that encompasses an area halfway to neighboring waste packages in each direction.

This modeled water flux is assumed to be an upper bound on the average water flux per waste package. Oversby and Wilson (1985) have calculated the bounding flux to be 40 L/package-yr for spent fuel packages assuming a recharge flux of 1 mm/yr. If no diversion of water toward waste packages were to occur, the flux per package would be approximately 0.5 L per year for a recharge flux of 1 mm/yr passing through a 76-cm diameter vertical borehole.

7.4.5.4.6 Corrosion model

The current corrosion model calculates the corrosion rate and the cumulative thickness of barrier metal altered under two environmental conditions: moist air and water. Each rate may be a function of temperature and gamma ray dose rate; dependence on water chemistry may be added when the water chemistry data become available from the waste form environmental model. The data on corrosion rate in water are input by the program user; they may represent uniform corrosion or uniform and pitting corrosion. The waste package will most likely not be wetted uniformly; the intent is to calculate the corrosion rate at the most critical location.

A future corrosion model will incorporate checks for sensitization history and for environmental conditions that would allow other local modes of corrosion such as intergranular stress corrosion cracking, pitting, and crevice corrosion. Threshold conditions for these modes, or envelopes on such thresholds, and corrosion rates will be established for use in the model. Determination of these conditions will require an assessment of the considerable uncertainties that accompany corrosion models. These uncertainties are an important component of the metal degradation investigations and will be incorporated in the performance assessment model.

Corrosion-induced changes in material thickness will affect the thermal and mechanical behavior of the container. Ultimately, these changes will also affect fluid flow and waste form release behavior. Corrosion model output will be used as input to the corresponding models. The corrosion rate may depend on data from the thermal and gamma ray dose rate models, or a bounding rate may be used.

The corrosion model will find the following in data tables that will be supplied by the metal barriers subtask:

1. Rate of corrosion of the metal (micrometers per year).

2. Percentage of the corrosion product that remains on the surface as a solid layer.

3. Rate of removal of an existing corrosion product layer by water.

Another data item required for transfer to the mechanical model is the change in specific volume between a unit of metal and the corresponding amount of solid corrosion product.

Sections 7.4.2.8 and 7.5.4.6 summarize investigations and data to date on corrosion of the austenitic barrier materials. In water, general corrosion is the most likely corrosion mode, and pitting corrosion may be expected in some locations on the barrier; but the rates are low enough that these degradation modes are not likely failure modes. One corrosion mode that may be a likely failure mode is intergranular stress corrosion cracking. This mode depends on sensitization (Section 7.4.2.5) versus temperature history (see Section 7.4.1.2 and the systems model's thermal model) and on stress (see the system model's mechanical model).

Tests to quantify sensitization and stress corrosion cracking will be conducted during characterization (Section 8.3.5.9, Issue 1.4). When results are available, the process will be added to the corrosion model.

7.4.5.4.7 Waste form alteration model

The waste form alteration model will calculate the annual quantity of each radionuclide converted into mobile forms. The mobile forms are assumed to be solutes in water, carbon-14 in the form of carbon dioxide, and noble gas radioisotopes. The processes and the input values to be developed for both spent fuel and glass waste forms are discussed in Section 7.4.3.

The radionuclide mobilization rate of spent fuel is limited by properties of the waste form (such as the solid matrix form of spent fuel, Zircaloy and steel framework components) and the initially intact cladding of many of the fuel rods.

The radionuclide mobilization term for spent fuel includes five components:

1. Elements whose release is controlled by the spent fuel matrix dissolution rate.

2. Elements present in part in the pellet-cladding gap and available for rapid release when the cladding is breached.

3. Elements contained in stainless steel or Inconel support components.

4. Elements contained in the Zircaloy or stainless steel fuel cladding.

5. Elements present in part in the oxidized layer present on the outer surface of the Zircaloy and available for rapid release when the container is breached (i.e., Carbon-14).

Initially, the model for each component is essentially a table look-up based on data to be determined under relevant conditions of water flow and environment.

Cladding degradation has two functional effects. Cladding corrosion exposes activation products contained in the cladding material for mobilization. Localized degradation leading to cladding breach initiates the exposure of the spent fuel within the fuel rod. The rates for each of these processes will be inputs to the model (Section 7.4.3.1.3). The current model treats a percentage of initially failed fuel rods and degradation of only a single leading fuel rod as representing all intact fuel rods within the package; a future model will treat the ensemble of intact fuel rods within a waste package.

The rapid release fraction for a spent fuel rod will all be assigned to the first year after breaching of the fuel rod. Future model development will account for distribution over time of the time-to-breach for waste containers and time-to-breach for cladding of fuel rods, given a container breach.

The degradation rate of spent fuel depends on the water flux, on the mechanism of water flow in the waste package and on the solubility of uranium. If the barrier container has several perforations and the ground water trickles through, then the matrix dissolution rate will be that determined from measurements under similar conditions, as a function of water flow rate. If the barrier container has one perforation placed so that water can fill the container, then the spent fuel will eventually be exposed continuously to a nearly fixed mass of water. A small rate of inflow of new water will be assumed to displace an equal amount from the standing water. In this instance, the matrix dissolution rate is determined by the matrix dissolution together with the recharge water flow rate. The matrix dissolution rate will depend on uranium solubility.

The approach to waste form alteration modeling for spent fuel is consistent with the methodology presented in Section 7.4.3.1.4. The definitive data for the model are an information need (Sections 7.4.5.7 and 8.3.5.10).

For a glass waste form, the waste form alteration model assumes that release of all elements is controlled by the matrix alteration rate. Similar to the spent fuel case, this rate will be input as determined for two water flow scenarios: the trickle-through scenario and the standing-water scenario. Under very conservative conditions, which differ from those for spent fuel, Aines (1986) estimates that release rates from glass waste forms are approximately 0.08 part in 100,000 per year.

More precise results taking conditions into account are an information need (Sections 7.4.3.2.3, 7.5.4.8, and 8.3.5.10).

Some radioactive elements may be mobilized more slowly than indicated by their congruent release with the spent fuel matrix because of their low solubility. For glass waste forms, surface layers formed with secondary products and additional interactions of the glass-canister-water-rock system may slow down the mobilization of waste elements (Section 7.4.3.2.3). These features are neglected in the first-generation model, but they will be included in later versions.

7.4.5.4.8 Waste transport model

A hydrothermal flow and waste transport model will calculate the flux of each radionuclide at the waste package-repository host-rock interface. The interface is at the borehole wall. Ultimately, this prediction will be a function of the flow, transport, and chemical processes active in the host rock immediately adjacent to the emplacement hole. A waste package near-field environment model will be incorporated to provide a link to near-field processes since package performance both depends on and affects this environment (e.g., the heat flux from the package affects hydrology and geochemistry). This near-field environment model will combine effects from the supporting models for the radiation, thermal, hydrothermal, and fluid chemistry models (see previous subsections of Section 7.4.5.4).

The submodel in the first system model revision will be simplistic and conservative. For water soluble elements, the model considers advection with the water flux. The first-generation model assumes that the water flux out equals the water flux in after any storage volume within the breached container is filled and that any elements mobilized in the water filling the canister are transported out of the waste package volume proportionally with the existing water flux.

Gas phase transport is by diffusion into the host rock. Any gases mobilized are assumed available immediately for transport beyond the boundary of the waste package.

The flux of transported radionuclides will be in units of a fraction per year of the current inventory of a whole waste package. The flux will also be reported in units of grams per year and in units of the fraction per year of the activity of a whole waste package at 1,000 yr after repository closure. The model will require, as inputs, the fluid flow results and the waste mobilization rate within the waste package volume.

7.4.5.4.9 Driver model

The driver model couples all the process models, calculates the time history of the waste package's condition and processes, and from this time history extracts the performance measures.

The specific solution is essentially determined by the initial conditions and the boundary conditions over time. Figure 7-36 shows the overall structure of operations establishing the initial and boundary conditions and calculating the waste package performance. Gane and Sarson's (1979) notation is used for data flow diagrams (Figure 7-37).

Data delivered to the process models must have the current-time values. Data flow diagrams show the logical dependencies of data needs but abstain from specifying program sequence or control. Data flow diagrams are a good starting point for the specification of the necessary properties of the linkages among the process models.

The construction of the data flow diagrams for the driver model is presented in a sequence to express some of the time properties of the data and to clarify parts of the process being modeled. Some data adjust to the boundary conditions with essentially no time delay (e.g., radiation, temperature, and stress). Some data change slowly with time (e.g., intact barrier thickness as reduced by corrosion). Some data have discrete values and change infrequently (e.g., barrier surface environment (dry/wet)).

Table 7-26 identifies the data elements in the data flows that follow. Figure 7-38 shows some data that adjust immediately to the boundary conditions and also shows the associated processes. The radiation model's heat source rate, radiation source rate, and attenuated gamma ray dose rate are immediately dependent on the input value from the radiation source time history. The temperatures adjust to the heat source rate and the input value from the boundary temperature time history. The mechanical stresses immediately find a static equilibrium dependent on the temperatures and the input value from the boundary pressure time history.

The environmental condition and the corrosion rates also depend on the current-time boundary conditions and on the data shown in Figure 7-38 (refer to Figure 7-39). Note that an additional output from the environmental model has a time delay.

Some waste package data change slowly over time. For example, the geometry (intact container wall thicknesses) changes slowly due to general corrosion (Figure 7-40).

Now the sequence of diagramming has reached a closed feedback loop. The processes in the loop are shown in Figure 7-41. (The external data stores of

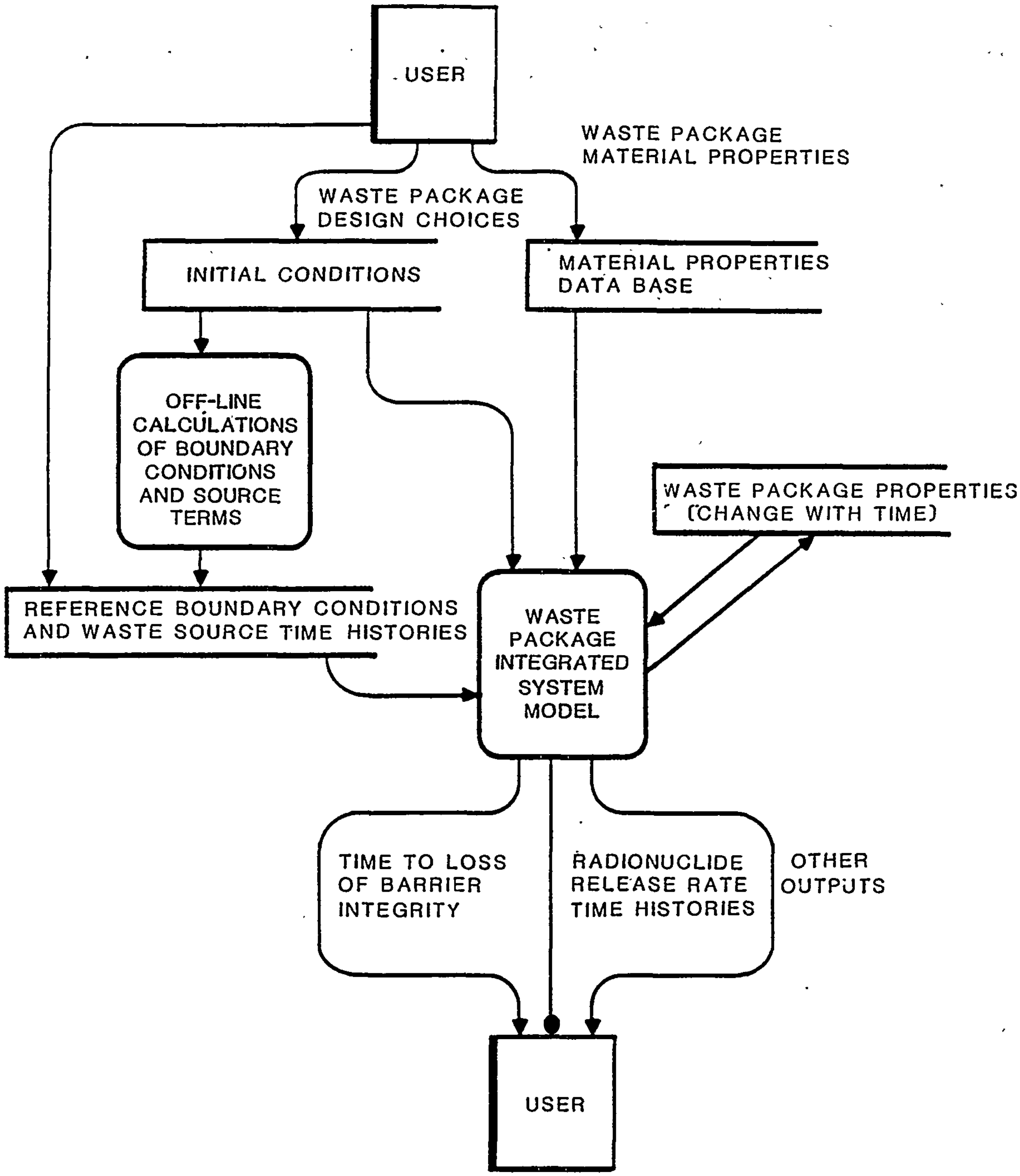


Figure 7-36. The data flows, data stores, and grouped inputs and outputs for the waste package system performance problem. (See Figure 7-37 for notation used in data flow diagrams.)

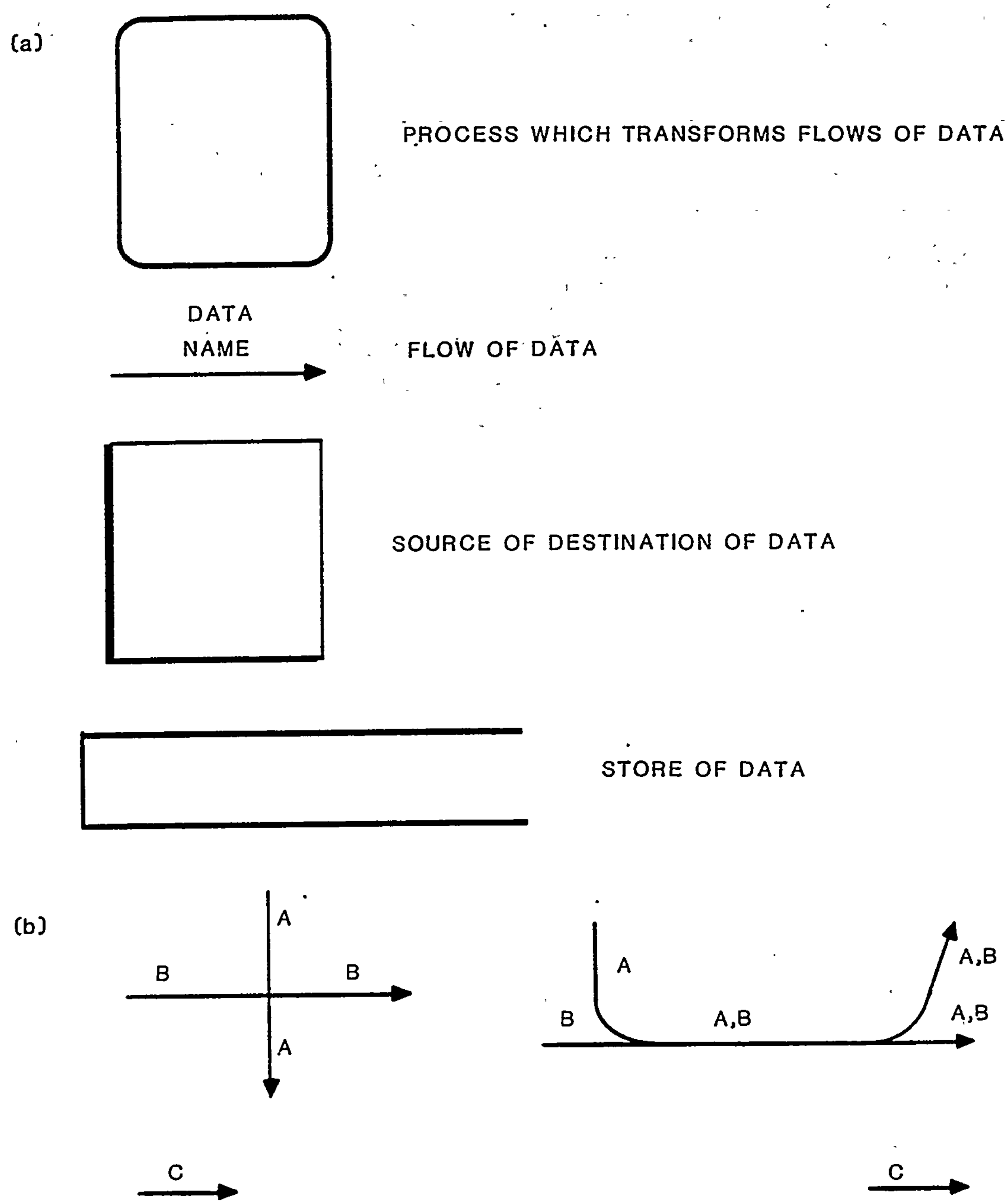


Figure 7-37. Data flow symbols and conventions: (a) shows data flow symbols and (b) conventions. Crossed data flow lines do not join, merging lines do join. To avoid "spaghetti diagrams," connectivity may be indicated by labeling with the data group name, or an entity such as a data store may be drawn in duplicate positions. A data flow indicates data needs and sources; it does not indicate sequence or control.

Table 7-26. Identification of data elements in the data flow diagrams (page 1 of 2)

Label	Data contents

Waste package properties	Geometry
	Status of elements
	    Environment (dry/wet)
	    Strength (intact/yielded)
	    Integrity (intact/breached)
	Material types
	Quantity of waste form

G	Waste package geometry
S	Status of waste package elements
Time	Current time, time of next time step
DB	Data base of material properties
Radiation source t/h[a]	Time history of radionuclide inventories, heat generation rate, radiation generation rate
Boundary temp t/h	Time history of boundary temperature
Boundary pressure t/h	Time history of external pressures in axial and radial directions of waste package
Boundary fluid flow t/h	Time history of rate of water flow into the waste package volume
Heat rate	Heat generation rate of each waste form
R1	Radiation data: Gamma ray absorbed dose rate in water at waste form surface of each barrier element
R2	Radiation data: Alpha particle dose (integrated up to current time) in waste form and alpha particle dose rate in water at waste form surface
R3	Radiation data: Inventory of each radionuclide per unit of waste form

Table 7-26. Identification of data elements in the data flow diagrams (page 2 of 2)

Label	Data contents
Temp	Temperature at inner and outer surfaces of each element of the waste package
Stress	Stress components at surfaces of each barrier element and at surface and interior of waste form
Corrosion rate	General corrosion rate for each barrier
Wet (Y/N)[b]	Environmental status of boundary of waste package
Water flow rate	Water flow rate into the waste package volume
New thickness	New thickness of intact container and of corrosion layers
New barrier status, new dimensions, or no change	Any changes to waste package geometry or status
Time to loss of container integrity	Time of breach of containment function due to loss of container integrity
Standing quantity of water	Quantity of standing water in a partially degraded waste package
Waste mobilization rate, increment	Waste mobilization rate and increment of waste released from the waste form to the waste package volume during a time interval
Waste release rate, increment	Waste release rate and increment of waste released from the waste package to its exterior during a time interval
Mobilized waste quantity in waste package volume	Quantity of waste radionuclides and matrix in mobile form and within the waste package geometric volume
Waste form quantity	Quantity of waste remaining in the unmobilized waste forms

[a]Temperature per hour.
[b](Yes/No).

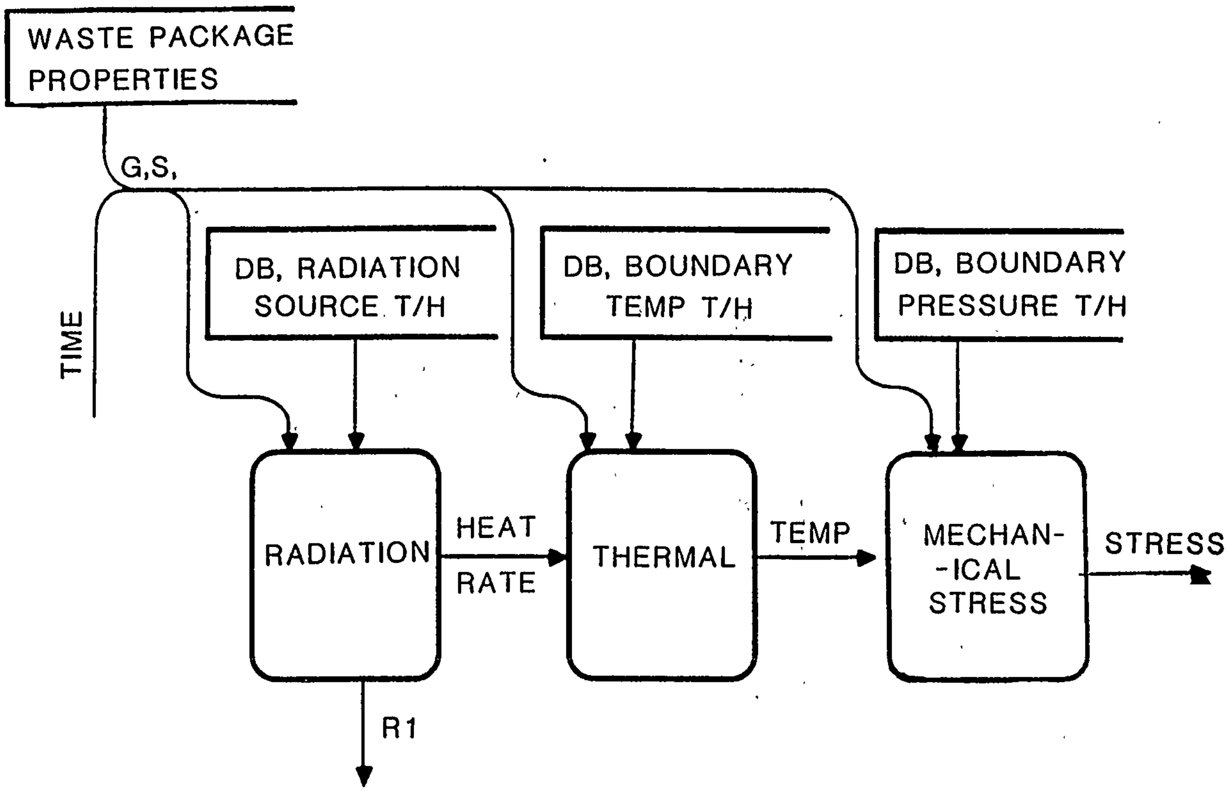


Figure 7-38. Data flows for the radiation, thermal, and mechanical stress processes. The process outputs shown heat rate, temperature, and stress adjust without time delay to the input boundary conditions. (See Figure 7-37 for an explanation of data flow symbols and Table 7-26 for an identification of data elements used in data flow diagrams.)

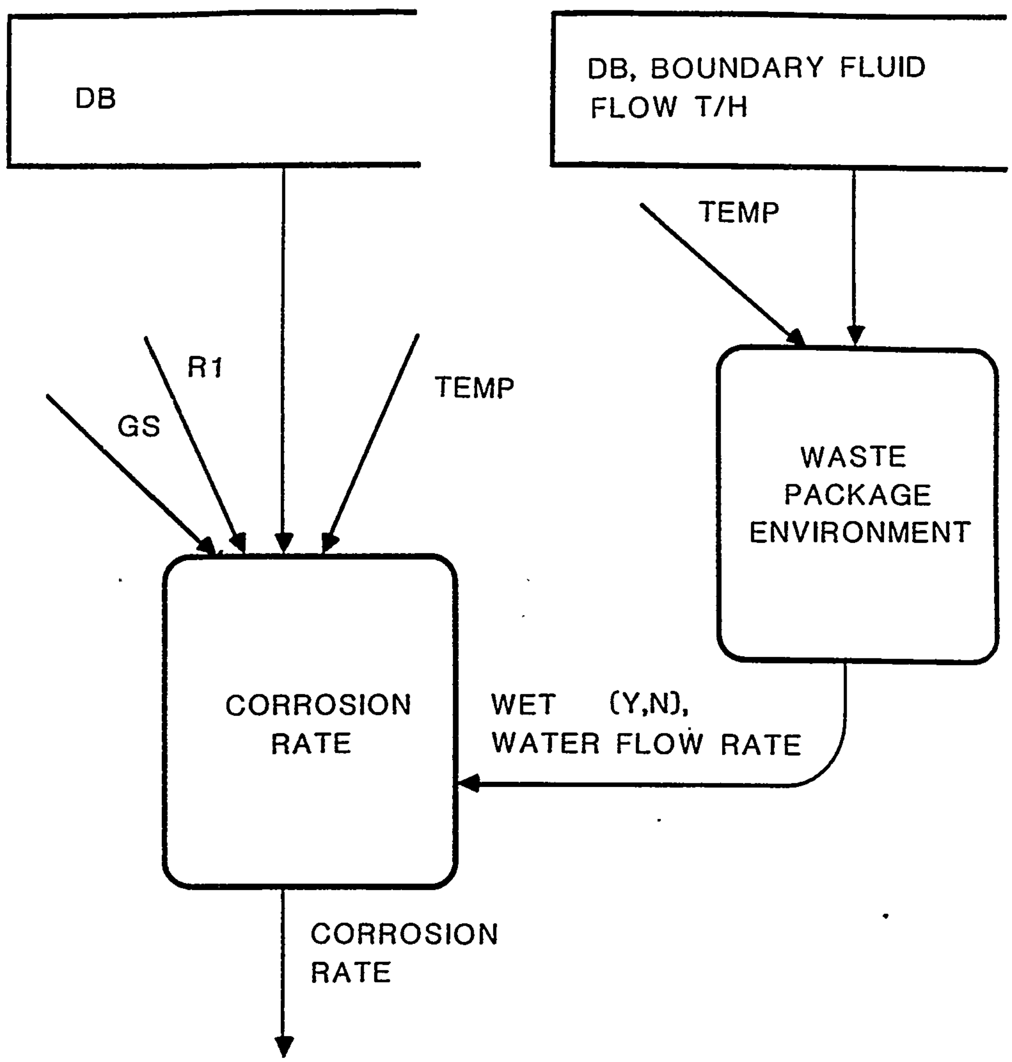


Figure 7-39. Data flows for the waste package environment and corrosion rate processes. The process outputs shown adjust without time delay to the input conditions. (See Figure 7-37 for an explanation of data flow symbols and Table 7-26 for an identification of data elements used in data flow diagrams.)

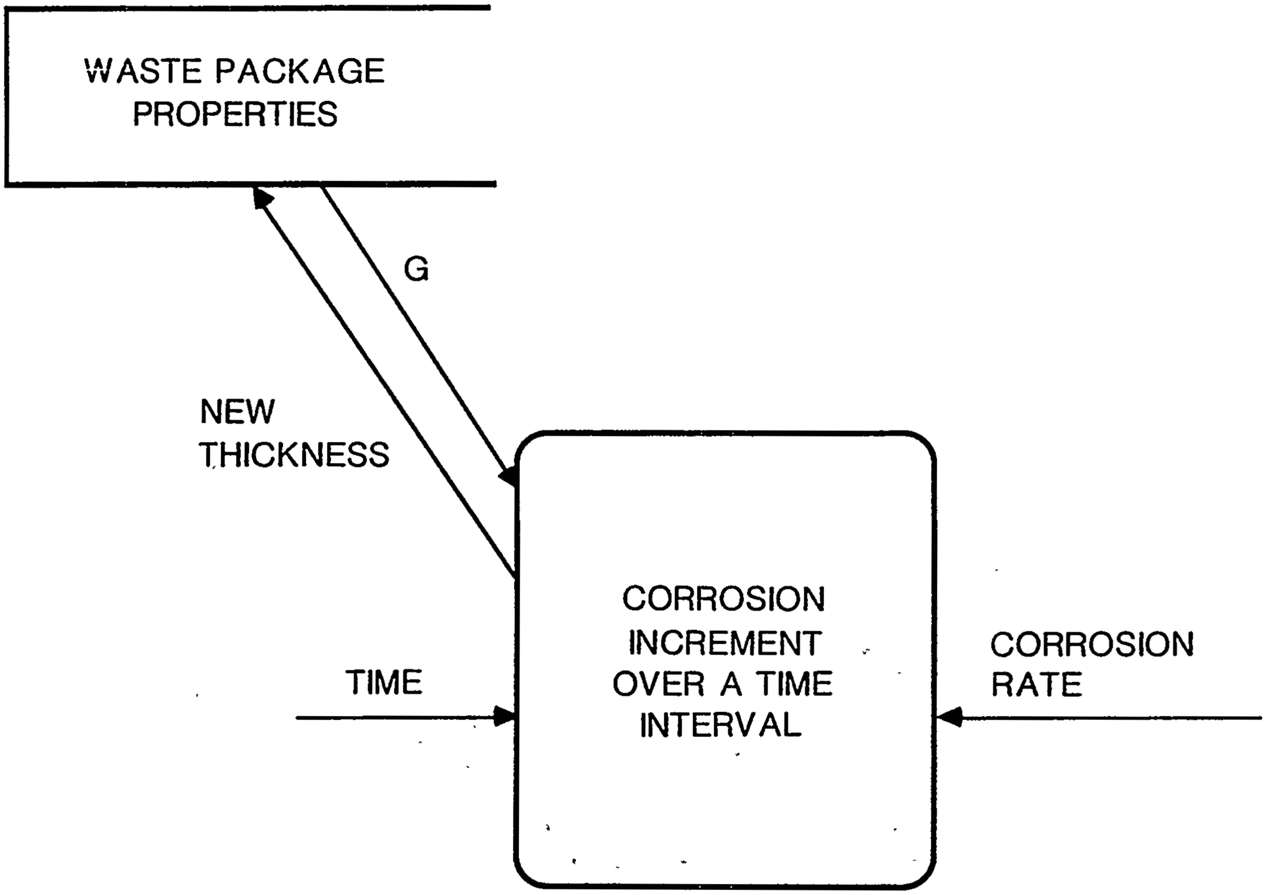


Figure 7-40. Data flows for the corrosion increment process. The output (new thickness) changes slowly; its value depends on past history as well as on current conditions. (See Figure 7-37 for an explanation of data flow symbols and Table 7-26 for an identification of data elements used in data flow diagrams.)

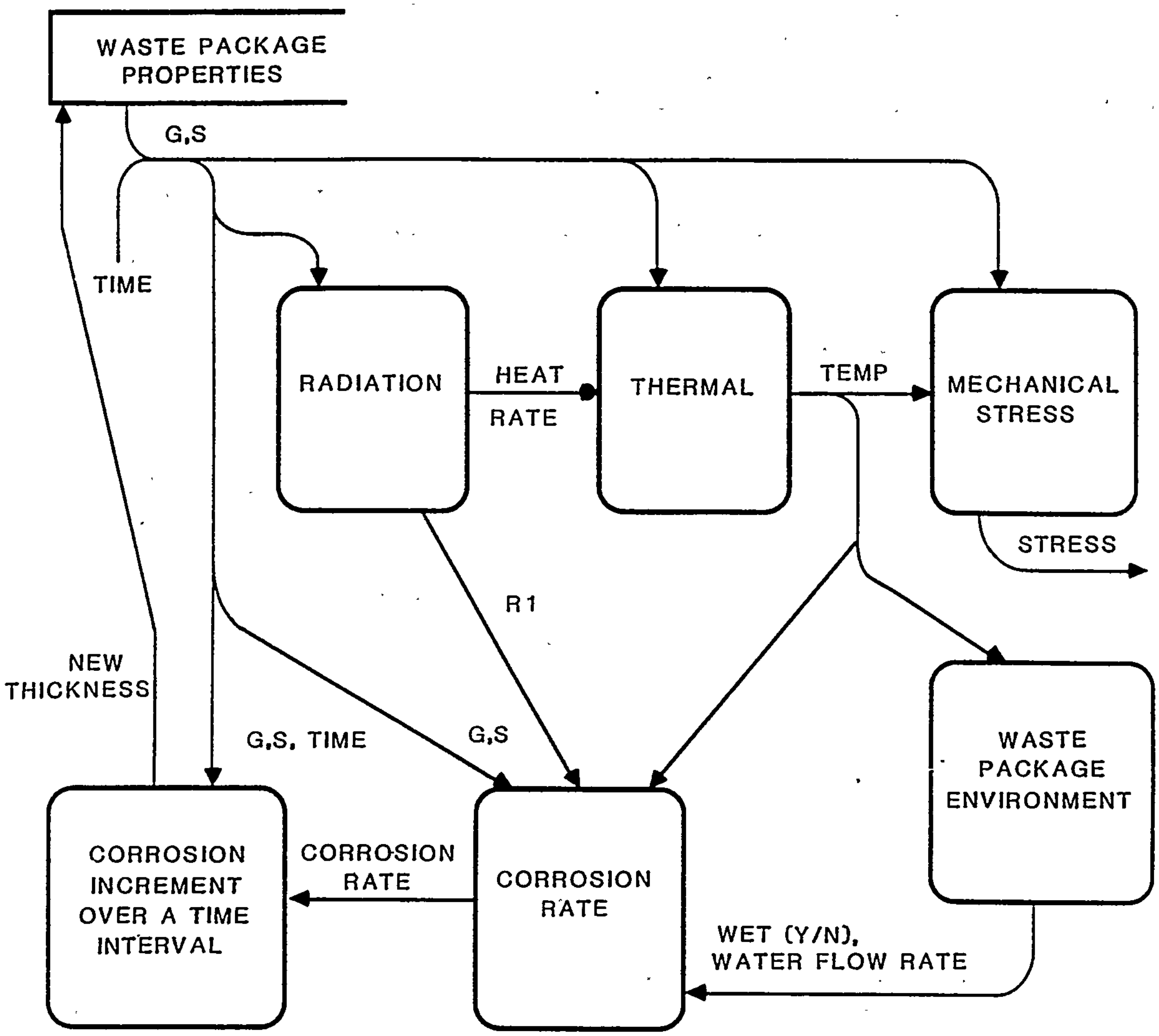


Figure 7-41. Data flow diagram combining the processes shown in Figures 7-38 through 7-40. Some of these processes are coupled in a continuous (over time) feedback loop. (See Figure 7-37 for an explanation of data flow symbols and Table 7-26 for an identification of data elements used in data flow diagrams.)

material properties and boundary time histories are not drawn but are implied.) The progression of corrosion affects barrier thickness. Container wall thickness affects gamma ray attenuation and heat transfer, both of which affect corrosion rate. The corrosion rate affects the further progress of corrosion and of container wall thickness change.

Some waste package data have discrete values and change infrequently. The progression of corrosion implies that eventually the container's containment integrity will be lost. Mechanical yielding or rupture occur rapidly when a threshold stress is reached. Some modes of corrosion may occur and progress rapidly when enabling environmental and stress conditions are reached. A test for conditions for failure modes is included in Figure 7-42. The increment failure modes process requires the time value not for the failure checks but only to report the time of breach occurrence.

The waste form alteration and waste transport models depend on data developed by the other models, including additional data not required for the corrosion model (Figure 7-43). These data include alpha particle and spontaneous fission doses and dose rates, radionuclide inventories per unit of waste form inventory, and quantity of water retained in a partially degraded waste package. Some of these data depend on past history as well as current conditions. The waste form alteration and waste transport models affect one item of the waste package data, the quantity of waste form.

The driver model's solution method calculates the history of the waste package condition and processes in the time domain, and from this time history extracts the performance measures. The algorithm uses large time intervals when conditions and package parameters are changing slowly, and short time intervals when conditions or parameters change with large rates or discretely. The algorithm calculates current-time conditions, then projects corrosion, waste form alteration, and waste release over an interval to the next time, and then calculates next-time conditions and checks for discrete status changes or the exceeding of failure thresholds. If any such change is indicated, the algorithm returns to the current-time and repeats the projection with a smaller time step. If a discrete change occurs during a minimum time interval (specified by the user, down to a minimum of 1 yr), then the waste package status is updated with the change at the current time. If no discrete change occurs during a time interval, then the continuous process results are updated and the next-time becomes the current-time for the start of the next step.

This algorithm models both discrete and continuous changes in waste package condition, identifies the time of breach of containment to within a desired tolerance, and provides radionuclide release rates and release quantities over a time interval. More details of implementation will be developed during the program design stage, but the final algorithm will perform functionally as described in the conceptual model.

7.4.5.5 Reliability analysis

The goal of the reliability analysis is to provide reasonable assurance that the waste packages will meet the postclosure performance criteria. The

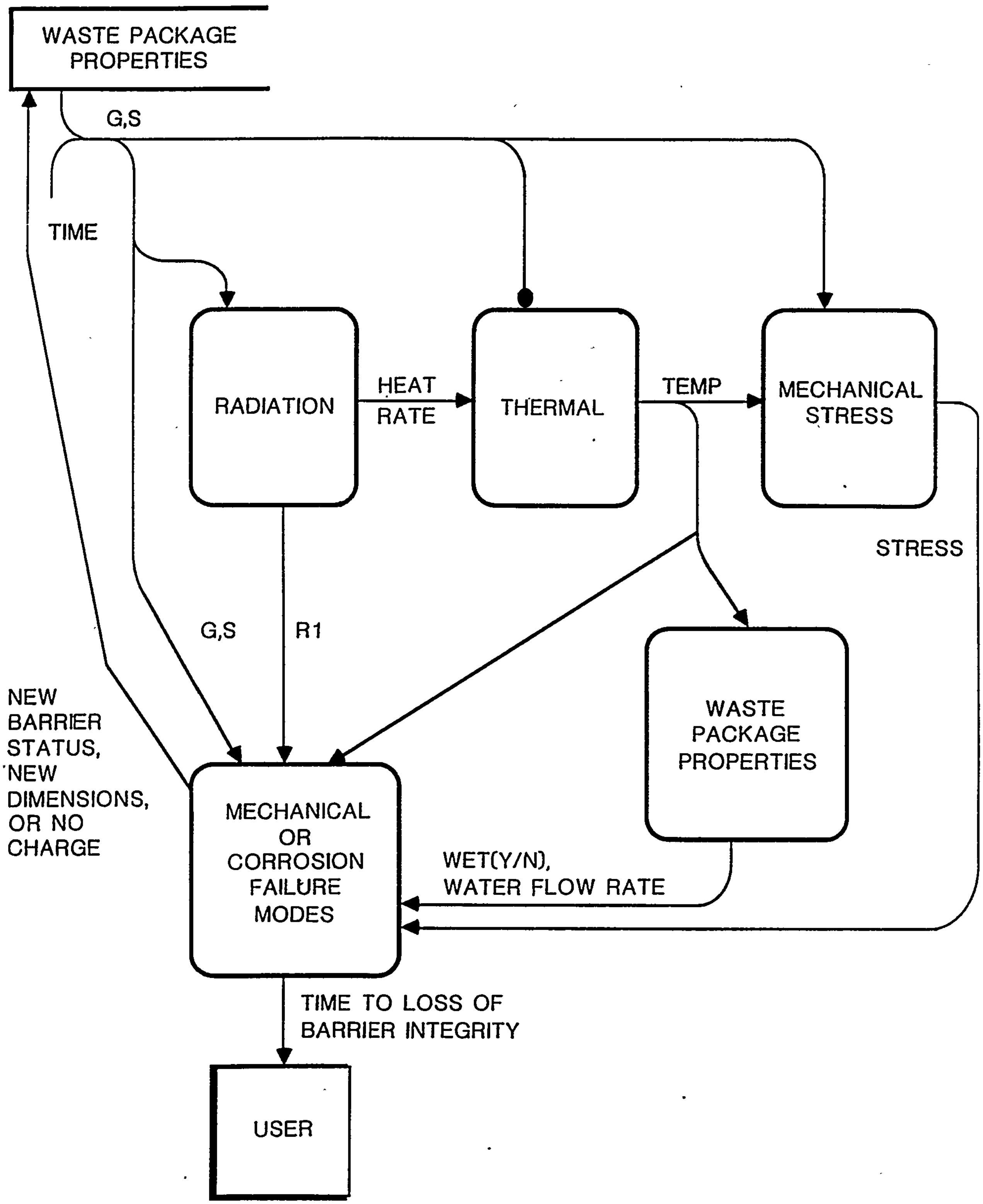


Figure 7-42. Data flow diagram showing the mechanical or corrosion failure modes process and the processes it depends upon for input data. There is a feedback loop present, but the feedback occurs only occasionally and by discrete amounts. (See Figure 7-37 for an explanation of data flow symbols and Table 7-26 for an identification of data elements used in data flow diagrams.)

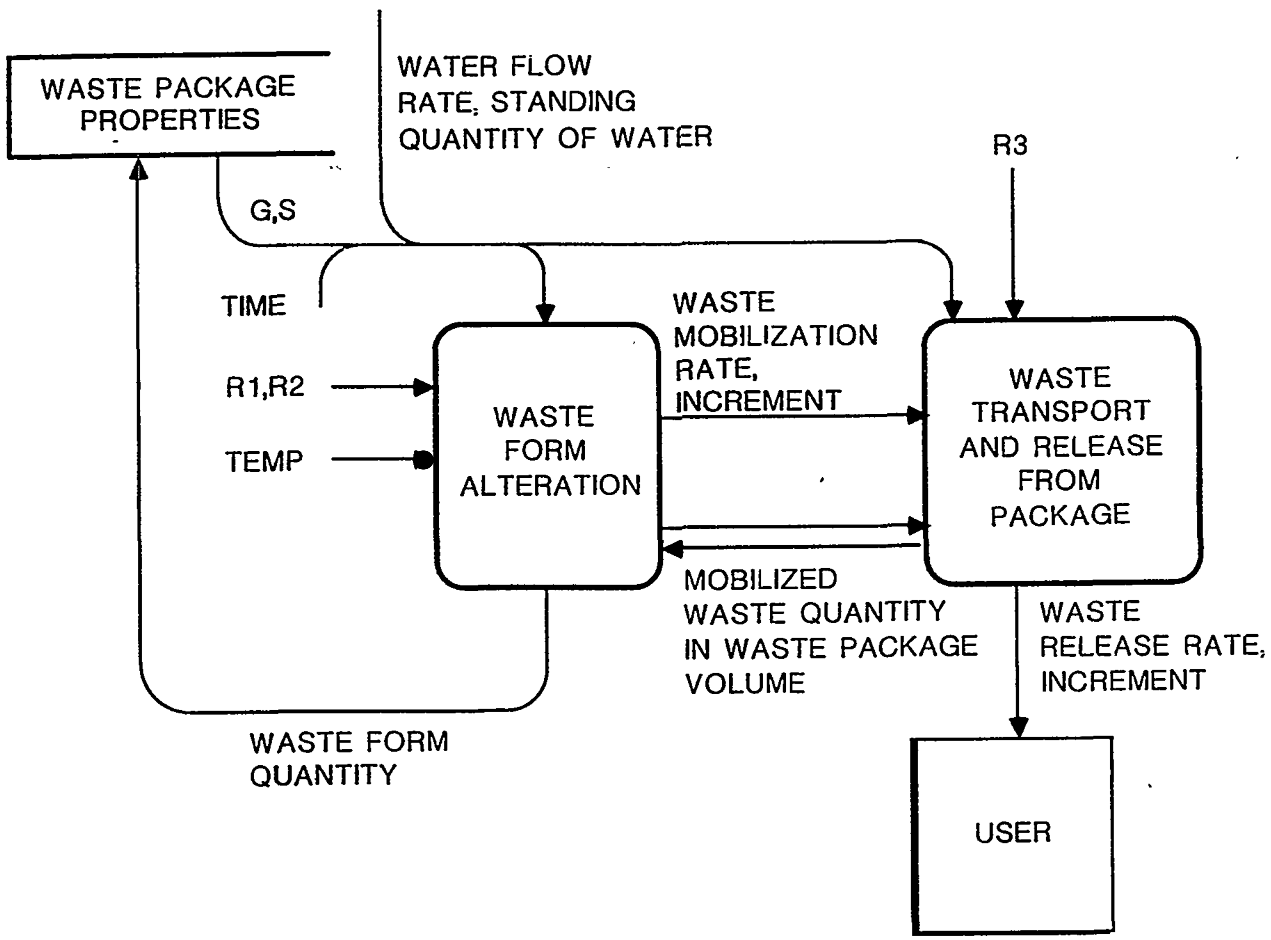


Figure 7-43. Data flow diagram for the waste form alteration and waste transport processes. Inputs from other processes are shown by the data flow names. The outputs and some of the inputs change gradually and depend on past history as well as on current conditions. (See Figure 7-37 for an explanation of data flow symbols and Table 7-26 for an identification of data elements used in data flow diagrams.)

proximate goals of the reliability analysis are (1) to identify the contributing failure modes of the waste packages, (2) to ensure that there are no common mode failures that would lead to an excessive early failure rate, (3) to estimate the reliability with which the waste package times of containment failure and rates of release will be better than values adequate to meet the performance criteria, and (4) to estimate the frequency distribution of the waste package performance values.

A probabilistic reliability analysis methodology will be selected and implemented to assess reliability in the waste package performance. This methodology will include tools of failure modes and effects analysis, anticipated and unanticipated events scenario analysis, and propagation of input parameter uncertainty, as appropriate.

Propagation of input uncertainties will estimate the frequency distributions of probability distributions (cumulative distribution functions (CDF)) of the waste package performance values. A point somewhere on the lower tail of the CDF will correspond to a performance threshold value and the reliability with which the waste packages will exceed this performance threshold.

Propagation of input uncertainties can be evaluated by several methods. One group of methods involves sampling from the probability distribution of the input variables and doing repeated deterministic calculations of the performance using these samples of inputs (NRC, 1986). The deterministic calculations are done using the model for a single waste package. In this way, a sample of output performance values is accumulated, that approximates the CDF of the output. The input sampling may be purely random sampling, stratified sampling such as Latin hypercube sampling (McKay et al., 1979), or stratified selection.

To estimate a point on the lower tail of the CDF to reasonable accuracy with reasonable sample size may require weighted biased sampling emphasizing the region of input corresponding to the threshold output value. Another approach is to determine the threshold surface in the input space and integrate the probability density of the input space below this threshold.

The data requirements are estimates of the variability of package and local environment parameters among the population of waste packages. This variability may be expressed as a probability distribution or a frequency distribution. Since major interest is centered on providing reasonable assurance that most of the performance values are above the threshold criteria, the corresponding interest is on that part of the input space corresponding to or near the output threshold. Bounding values of inputs and statements of the probability that the input values are below or above this boundary may be sufficient for this purpose. Estimates of means and variances of inputs may be needed to estimate the means and variances of the output CDF, but more approximate estimates may suffice as long as the threshold points are separately specified.

To date the project has identified the principles explained above that give some guidance on the types of data required. Selection and implementation of specific methods and acquisition of data are information needs; they are discussed in Sections 8.3.5.9 and 8.3.5.10.

7.4.5.6 Summary

This section emphasizes the waste package system model in order to discuss performance assessment. This model is central to the prediction of long-term performance since it integrates the results of studies described elsewhere in this chapter to make those predictions. The current status of this modeling activity has been described, and its link to the probabilistic analysis needed to satisfy the NRC regulations has been discussed. It has been shown that the information needs of performance assessment that will be satisfied during site characterization will be derived from data to be gathered by the waste package processes investigations. The unique requirements of performance assessment are that those investigations establish the important parameters describing each process, determine predictive relationships for the processes that incorporate those parameters, and provide statements of probability for the occurrence of parameter sets. Therefore, though not directly specifying data needs to be obtained during site characterization, performance assessment does specify data types and analyses that must be performed by the process investigations. Plans for further development of the waste package system model are discussed in Section 8.3.5.10.

7.5 SUMMARY

The summary of Chapter 7 is broken down into the same overall subsections as the chapter itself to facilitate referencing the more detailed discussions.

7.5.1 EMPLACEMENT ENVIRONMENT

Sections 7.1 and 7.4.1 discusses the emplacement environment. One of the candidate sites for disposal of HLW is in the Topopah Spring Member of the Paintbrush Tuff. This member is welded and devitrified and, in the candidate site, is unsaturated. Humid conditions are expected since the degree of saturation averages 65 ± 19 percent.

For the purpose of waste package performance assessment and design it has been assumed that the thermal loading will result in borehole rock temperatures that will reach a maximum of 230°C a few years after emplacement and gradually decrease to about 190°C 10 to 20 years after emplacement. With an expected boiling point of water of 97°C, the elevated rock temperatures will result in a dried-out zone that will extend more than 1 m from the borehole into the rock. This dried-out zone is predicted to remain dry for at least 300 yr. The container design, which is discussed later, takes advantage of the lack of liquid water in the borehole as a result of the drying out. Information that is needed regarding the dried out zone includes a better understanding of the potential heat pipe effect and fracture influence on the drying mechanism. The plans to obtain this information are discussed in Section 8.3.4.2.

Water chemistry has a direct impact on waste package performance. The expected chemistry of the water, as discussed in Section 7.4.1.3, is similar to that obtained from well J-13, which is located in the Topopah Springs welded tuff in an area where the tuff is saturated. Well J-13 water chemistry has been used in the materials testing activities for the waste package. A need to obtain actual samples of the water from the repository area is discussed in Section 8.3.4.2.

7.5.2 DESIGN BASIS

The bases for waste package design are the various regulatory requirements (both direct and indirect waste package requirements) that are placed on the waste package. These are categorized into preclosure and postclosure requirements. The preclosure requirements include consideration of handling, criticality control, identification, and overall repository performance. The postclosure requirements include considerations for substantially complete containment and performance after closure to control release rates of radionuclides. In addition to these requirements, there are general requirements that include consideration of retrieval, the types of waste that will be included (waste forms), the material interactions among components of the waste package, and the limitations on waste form temperature.

In addition, design criteria require that (1) the production and emplacement of the waste package can be demonstrated to be technically feasible using reasonably available technology, (2) the waste package designs shall not make application of reasonably available technology impractical for other portions of the repository system or operations, and (3) the cost effectiveness be considered.

General design requirements for the preclosure period include constraints on explosive, pyrophoric, and chemically reactive materials, and exclusion of free liquids in amounts that could impact long-term performance. For the postclosure period, the waste package design will consider material specification to provide substantially complete containment and to ensure that interactions between the waste package and the environment do not compromise the long-term waste package performance, as discussed in Section 7.4 and Sections 8.3.4 and 8.3.5.

How the performance is allocated (to which system components, etc.) has a direct impact on the waste package design. For preclosure performance, the waste package is designed so that the container will provide containment of the waste forms under normal handling loads. The waste form geometry and internal container environment provide criticality control under anticipated conditions. The identification and exclusion of reactive materials and free liquids is allocated entirely to the waste package. To meet these requirements, additional information will be required regarding the waste forms, as discussed in Sections 8.3.5.10 and 8.3.4.3, respectively.

The allocation of postclosure performance to components of the waste package and the engineered barrier system was not completed at the time that the conceptual designs described in this chapter were developed. The information that will be needed regarding performance assessment of the waste package is discussed in Sections 8.3.5.9 and 8.3.5.10.

7.5.3 WASTE PACKAGE DESIGN DESCRIPTIONS

7.5.3.1 Reference design

The reference design of the waste package specifies a 304L stainless steel metal container and a waste form (either spent fuel or reprocessed waste in a glass poured into a stainless steel canister). Reference designs for spent fuel are shown in Figure 7-2 (Section 7.3.1.3). The container for spent fuel is 66 cm in diameter with 1-cm-thick walls and varies from 3.1 to 4.7m long depending on the waste form dimensions. Container designs have been developed for disposal of repository-consolidated and intact assemblies. There are different designs for pressurized water reactor assemblies and boiling water reactor assemblies. The design provides for different forms of spent fuel with different burnups. The reference design is based on consolidated fuel assemblies. The concept includes the option of disposal of intact assemblies to provide for damaged or failed assemblies that cannot be consolidated as well as to provide for unconsolidated very high burnup and short cooling time fuel disposal. This option is necessary to meet the temperature limitations on the waste form. Thus, there are different designs because of the variety of fuel rod and fuel assembly dimensions and because of the different waste forms. The reference design for reprocessed waste is shown in Figure 7-3, Section 7.3.1.3. The reference design is a container similar to that for spent fuel with a diameter of 66 cm (the 61-cm pour canister is inserted into the 66-cm container) and with a length to accommodate either the West Valley or defense waste processing facility wastes.

Rolled and welded pipe manufacturing processes are representative of the conventional type of fabrication that may be involved in manufacturing the waste containers. Many more advanced techniques are under consideration. The fabrication and closure processes have not been selected at this time. This selection will depend upon the details of design that evolve during advanced conceptual and license application design phases. Plans for evaluation, selection, and development of fabrication, closure, and inspection processes are described in Section 8.3.4.4.1. Welding and nondestructive evaluation of the final closure of the container will be done remotely.

7.5.3.2 Alternative designs

Alternative designs include a conceptual design allowing for disposal of intact PWR and BWR assemblies. The design concept is a modification of the reference design diameter (to 71 cm) and internal spacing geometry to allow packing efficiency of the assemblies and disposing of four PWR and three BWR assemblies(Figure 7-4). Considering the expected mix of PWR and BWR assemblies (40 percent PWR and 60 percent BWR), a hybrid package allowing three PWR and BWR assemblies to be disposed of in the same package leads to a surplus of BWR assemblies. This imbalance is corrected by providing that 7 percent of the packages contain only BWR assemblies in an arrangement with 10 BWR assemblies in a container. Alternative designs also include containers fabricated from a copper or copper-based alloy. The designs for copper containers are similar to the reference designs except a thicker wall may be required to compensate for lower strength of copper, especially at elevated temperatures.

7.5.3.3 Other emplacement hole components

In addition to the waste packages, other components will be present in the reference vertical or alternative horizontal emplacement holes. Because of their proximity to the waste emplacement packages, they have the potential for altering the package environment and impacting the package performance. The materials for these components, borehole liners, emplacement hole shielding plugs, and emplacement dollies will be selected to avoid adverse impact on long-term performance of the waste package.

7.5.4 WASTE PACKAGE RESEARCH AND DEVELOPMENT

7.5.4.1 Radiation field effects

Considerations of the dominant radiation effects indicate that radiolysis products from the interaction of gamma radiation with water, steam, or air are the only effects of significance. More than 99 percent of the gamma radiation will be restricted to the first meter of rock immediately around the emplacement borehole. The radiolysis products will vary with time as the thermal conditions change. During the first 300 yr, the rock in this first meter will essentially be dried out. Therefore, the radiolysis products will be those resulting from gamma radiation in moist air. At some time after 300 yr, when liquid water returns to this portion of the rock, the gamma radiation flux will have decayed by three orders of magnitude. Therefore, the radiolysis of liquid water is not of concern. Additional studies of moist air radiolysis are discussed in Section 8.3.4.2.

7.5.4.2 Water flow

Water transport under ambient conditions is through a combination of vapor transport, water migration through the matrix, and water transport through fracture flow. The influence of the thermal loading on flow under unsaturated conditions is not well understood. Studies of dehydration and rehydration under fully saturated conditions are discussed in Section 7.4.1.5. These studies revealed a basic difference in the response of fractured and intact rock. For intact rock, the permeability was independent of temperature and thermal history. However, there was a significant decrease in permeability of the fractured rock upon successive cycles of dehydration-rehydration. This decrease appears to have been the result of deposition of silica on the fracture walls. The mechanism for the change in hydrologic properties of fractured rock remains to be fully characterized and has not been thoroughly evaluated for unsaturated conditions. Additional work on near-field water flow mechanisms is discussed in Section 8.3.4.2.

7.5.4.3 Numerical modeling

Numerical modeling of water flow under hydrothermal conditions will require an understanding of the fundamental properties of multiphase fluid flow under thermal loading conditions. Some of the properties that will need to be better understood include the characterization of the fractures, characteristic curves for multiphase flow in unsaturated conditions, and actual degree of saturation in the rock immediately adjacent to the containers. These properties will be obtained through studies that are described in Section 8.3. In addition, appropriate models will require better understanding of the fundamental mechanisms governing the hydrology and flow. This will require evaluation of the use of equivalent continuum models, development of means of evaluating gas flow in fractured media where fracture flow can dominate depending on the fracture system, and development of means of evaluating whether water condensing outside of the dried-out zone will allow sufficient saturation to allow water flow through fractures. All these information needs are discussed in Section 8.3.4.2.

7.5.4.4 Rock-water interaction

The corrosion of the waste container and the dissolution and transport of radionuclides are influenced by the chemical composition of the ground water. This in turn is influenced by the mineralogy, temperature, duration of water contact with the rock and ratio of the rock to water volume. Ongoing experiments that have used well J-13 water indicate that there will be a decrease in calcium, magnesium, and carbon dioxide and that there will be early peaks in potassium and aluminum followed by a gradual and moderate decrease. The pH of the solution will decrease from approximately 7.5 to 6.8 in response to precipitation of carbonate from solution. Preliminary conclusions are (1) the fluid in a saturated environment will evolve toward cristobalite and carbonate saturation, (2) the secondary phases resulting from hydrothermal interaction will include sorptive clays and zeolites, and (3) the composition of the fluid phase will remain benign at all temperatures.

It has not been established whether the results from vitric and vitrophyre-rich units are similar to those observed for the devitrified welded tuff. Studies of material from the exploratory shaft to address this question are discussed in Section 8.3.4.2.

7.5.4.5 Modeling rock-water interactions

The EQ3/6 geochemical code has been used to model the results of dissolution studies. The code is not able to assess precipitation as a function of time. Revisions to the code to allow this assessment as well as to address thermodynamics and reaction path are required. Plans for these activities are discussed in Section 8.3.5.10. In addition to model refinements, data will be needed on phase compositions, dissolution kinetics, and thermodynamic properties. This information will be addressed in Section 8.3.5.10.

7.5.4.6 Metal barriers

Six candidate materials from two alloy families (austenitic and copper base) have been evaluated by the Nevada NNWSI Project for use in the waste package container. AISI 304L stainless steel has been selected as the reference material for the NNWSI Project waste package on the basis of its predicted performance in the unsaturated tuff environment at Yucca Mountain. Therefore, the evaluation of performance focused on this alloy. The reference material is a benchmark upon which the performance of the other austenitic materials is compared. The other austenitic materials are AISI 316L and Alloy 825.

Corrosion and oxidation attack of austenitic materials are the most likely degradation modes although the processes of fabrication and closure, as well as stresses, can affect the degradation modes. The most important determinant for which degradation modes are operable is the postclosure anticipated environment.

The expected degradation mode(s) depends on the time frame. During the first 300 yr (substantially complete containment) when the boreholes are anticipated to remain dry, oxidation is the only mode expected to be significant. During the postclosure period from 300 to 1,000 yr, general aqueous corrosion may be possible depending on the time for hydration of the rock around the emplacement boreholes. The rates of general corrosion are sufficiently low (0.07 to 0.15 micrometer per year for dry steam and water immersion, respectively) that the limiting case is rarely general corrosion, but rather the more rapid penetration of localized or stress-assisted corrosion. The NNWSI Project metals testing activities have focused on determining the extent to which the various forms of corrosion would occur during the periods of time outlined above.

The general corrosion rates of the candidate austenitic materials exposed to water of expected chemical makeup are quite small and, thus far, do not seem to be significantly affected by (1) alloy composition, (2) temperature (28 to 150°C), (3) exposure time (up to 11,000 h), (4) irradiation (to more than 10^5 rads/h), or (5) aqueous or steam environment. A rate of approximately 0.2 micrometers per year has been determined, which for the 1-cm-thick reference container would not result in complete container corrosion for well over 10,000 yr. A model is being developed to quantify general corrosion and to allow for extrapolation of the data to long time periods. This development work is discussed in Section 8.3.5.9.

Localized or stress-assisted forms of corrosion, particularly intergranular corrosion (IGA) and intergranular stress corrosion cracking (IGSCC) are potential degradation modes if a sensitized microstructure develops. Difficulties have been encountered in applying American Society for Testing and Materials (ASTM) tests to predict sensitization. Electrochemical potential reactivation (EPR) tests are helpful in screening materials and conditions that could lead to IGA and IGSCC. A series of tests have been undertaken to consider the effects of loading-stresses, low temperature sensitization, environmental factors, residual stresses from fabrication, and alloying factors on IGSCC susceptibility. Analysis shows that if sensitization will occur, it would be evident in a few years at temperatures in the range of 200 to 300°C, which are within laboratory ranges. Plans to determine the time to sensitization are discussed in Section 8.3.5.9.

Experiments have not shown IGSCC of L grades of stainless steels even when stressed beyond yield strength and when heat treated. Severe exposure conditions have also been evaluated. Results indicate that low temperature sensitization could occur in containers that were heavily cold-worked if the temperatures were higher than the reference case. Use of AISI 316 stainless steel or Alloy 825 material would minimize this sensitization. Reduction of the peak surface temperature and reducing the residual stresses in the container would also alleviate intergranular stress corrosion susceptibility. Future testing of these materials is planned as discussed in Section 8.3.5.9.

A model to predict sensitization in austenitic stainless steels and extension of this model to higher nickel content materials is being developed. This model considers temperature, strain, and compositional effects in materials. Plans for further development of this model are discussed in Section 8.3.5.9.

Pitting corrosion, crevice corrosion, and transgranular stress corrosion cracking (TGSCC) are governed principally by the composition of the aqueous environment (ionic species concentration). Both the attack and the causative environment may be localized. In the majority of cases, these degradation modes are dependent on chloride ion concentration in the environment. The candidate austenitic materials appear to be sufficiently resistant to pitting, crevice attack, and TGSCC in unmodified well J-13 water or steam generated from that water to meet the performance objectives. Even if the highest measured localized corrosion rate is added to the general corrosion rate, container service life far exceeds the maximum containment requirement. It is also necessary to consider whether biological degradation is possible, which could enhance the factors leading to pitting or crevice attack. Studies to address these considerations are discussed in Section 8.3.5.9.

Other considerations are related to phase stability of austenite over long periods of time. Phase transformations, particularly in weld and heat-affected zones, might adversely affect long-term fracture toughness and result in a more brittle, possibly less corrosion-resistant container. Some materials, such as 316 stainless steel, will improve the resistance to localized corrosion but will have a greater tendency to form brittle phases over long periods of time. This needs to be addressed in future work as discussed in Section 8.3.5.9.

Hydrogen embrittlement might be of concern after the 300-yr time since the waste package walls would be below a critical temperature (85°C for AISI 304Lss) and water could come in contact with the waste package. Hydrogen could enter the container material from radiolytic decomposition of the water vapor; however, the relatively high temperatures would tend to mitigate against trapping of the hydrogen. Also, the oxidizing nature of the environment will tend to combine oxidizing radicals with atomic hydrogen as it is produced. These effects need to be investigated to determine whether hydrogen is produced and, if so, how much can permeate the metal barrier. Studies to address this question are discussed in Section 8.3.5.9.

Nonuniform corrosion caused by sensitized microstructure during the containment period with subsequent wetting of the affected area that induces IGSCC is one mode of corrosion that might place limitations on containment life. A question remains as to whether lower carbon content in the L grades can confer immunity or at least a high resistance to the phenomenon. Control of the processing of the container can confine this phenomenon to the area around the final closure weld; selection of the alloy can further reduce the susceptibility to allow for substantially complete containment.

Alloy 825 is considered the most resistant of the candidate materials to virtually every form of corrosion that might occur in a geologic repository in tuff. However, some developmental work would be needed to ensure it can be fabricated should this material be selected for the advanced designs. Development work on weld and fabrication technologies is discussed in Section 8.3.4.4.

Although the austenitic materials have thus far exhibited behavior indicative of material that will withstand the long-term environmental conditions in a repository at Yucca Mountain, the possibility exists that further testing may disclose serious degradation process(es) that affect all three alloys. For this reason, copper-base materials are being considered as an alternative alloy system. Three copper-based materials being considered are CDA 102, CDA 613, and CDA 715.

The types of corrosion that may occur for copper-base materials in a tuff repository are uniform corrosion; pitting, crevice, and intergranular corrosion; selective leaching; and stress corrosion cracking. Uniform corrosion is considered the most likely mode; but pitting corrosion may also be likely, especially because of the irradiated environment.

The NNWSI Project has completed a 2-yr feasibility study on the three copper-base materials. At this time, these materials appear feasible alternatives should the austenitic materials be found unacceptable. It is expected that these materials will be tested further as is discussed in Section 8.3.5.9.

7.5.4.7 Spent fuel waste form performance

7.5.4.7.1 Spent fuel dissolution and radionuclide release

Based on a review of the available data, a three-part testing program has been initiated to establish the characteristics of the spent fuel waste form. The first part is devoted to the study of the dissolution and radionuclide release from irradiated UO_2 (Section 7.4.3.1.1), the second part is intended to establish the oxidation rate of spent UO_2 fuel (Section 7.4.3.1.2), and the third part is intended to establish the mechanisms and rate of corrosion of Zircaloy cladding (Section 7.4.3.1.3).

Water contact with the waste itself is the most credible mechanism for removing most radionuclides from the waste package. The contact with the waste is a function of the orientation and type of the waste form as well as the emplacement configuration and failure mechanisms of the container.

Assuming that water can contact spent fuel, americium and plutonium are the most significant elements subject to dissolution. Semistatic leach tests have been conducted on spent pressurized water reactor fuel using well J-13 water. These tests used both bare fuel and rod segments with undefected and intentionally defected cladding. Results are discussed in Section 7.4.3.1.1. Uranium concentrations in solutions over bare fuel peaked within a few tens of days at around 4.5 to 4.9 micrograms per milliliter and then decreased to between 1.2 to 4 micrograms per milliliter. Because of the hydrogen-bicarbonate (HCO_3) ion in well J-13 water, most of the uranium was in true solution. The fractional release of uranium to solution was consistently higher than that of other actinides. Thus, uranium has the potential to be transported preferentially to other actinides.

Several fission products (cesium, iodine, technetium) are not fully dissolved in the UO_2 matrix, but fractions have migrated either to grain boundaries or to the fuel-cladding gap during irradiation. These elements show an initial rapid release upon contact of the fuel with water. Defected cladding did not provide an effective barrier to the early release of these elements. Unlike the other radionuclides of interest, carbon-14 may be released as carbon dioxide gas under oxidizing conditions without the presence of liquid water. Work to evaluate carbon-14 distributions has shown that approximately half of the inventory is associated with the Zircaloy cladding and the other nonfuel assembly parts.

Studies in the literature indicate that the dissolution of UO_2 is a complex, multistage process, the exact nature of which depends strongly on solution chemistry. In all instances however, it appears that the release of uranium occurs from an oxidized surface layer. Further work on radionuclide release is planned and is discussed in Section 8.3.5.10.

7.5.4.7.2 Spent fuel oxidation

Because of the large positive volume change upon oxidation of UO_2 to U_3O_8, an evaluation was made of the potential for cladding rupture by this mechanism. In addition, the dissolution rate of oxidized fuel may differ from that of unoxidized fuel. Differences between the oxidation rates of nonirradiated fuel and irradiated fuel have been reported in the literature; thus, the NNWSI Project work on the oxidation of UO_2 uses spent fuel. Two complementary techniques used to measure the oxidation rate at temperatures relevant to repository conditions are thermogravimetric analysis (TGA) and conventional oven oxidation.

The TGA data obtained to date suggest a two-stage oxidation mechanism: grain boundary diffusion of oxygen and oxidation of the grain surfaces followed by bulk diffusion and oxidation of grain interiors. At high temperature, the first stage is relatively rapid and bulk diffusion is the rate determining step. At lower temperatures, the rate of the first step has a significant effect on the overall oxidation rate on the time scale of the experiments conducted. The activation energy obtained from the bulk diffusion portion of the tests is approximately 107 kJ/mole. This value is within the range of values given in the literature.

In studies conducted by others on the dry storage of spent fuel assemblies, no cladding splits were noted for both undefected and defected nonirradiated fuel stored in an inert atmosphere at temperatures as high as 230°C. Fuel in rods with defected cladding stored in an oxygen atmosphere oxidized causing deformation and rupture of the cladding. Based on these results, a recommendation to use inert atmosphere has been adopted for the atmosphere in disposal containers to help preserve the integrity of the cladding during the early high temperature phase in the repository. Work is continuing on the oxidation rate and mechanism of spent UO_2 fuel and is discussed in Sections 7.4.3.1.2 and in Section 8.3.5.10.

7.5.4.7.3 Zircaloy corrosion

Zircaloy cladding, if it maintains its integrity for long time periods and if the cladding failures that occur are spread out over time, may provide a partial control on the release of the rapidly released, gap and grain boundary inventory of radionuclides. Stress rupture, hydride cracking, and stress corrosion cracking have been investigated as failure mechanisms of the Zircaloy cladding. Stress rupture was concluded not to be a significant failure mode for fuel rods under the NNWSI Project tuff repository disposal conditions at Yucca Mountain. While most data do not support the concept of hydride cracking, there is some concern that hydrides might reorient under storage conditions. Initial studies of Zircaloy-4 together with other components of the waste package in well J-13 water at 90°C did not detect any corrosion after 12 mo at a detection sensitivity of 1 to 2 micrometers per year. General corrosion data in water representative of the repository area is needed. Future studies on Zircaloy corrosion and cladding failure are discussed in Section 8.3.5.10.

7.5.4.8 Glass waste forms

Studies performed for glass waste forms indicate that, at long times, glass may be considered to dissolve congruently with less soluble elements, immediately reprecipitating onto the glass. A very conservative estimate of the leaching rate of any radionuclide is that the maximum rate will not exceed that of very soluble major elements in the glass. Only one possible exception to this was found during the NNWSI Project testing. To confidently predict the long-term behavior of the waste in glass form will require a thorough understanding of the glass leaching mechanisms.

One of the critical factors that influences leach rates is temperature. There is an expected leach rate difference of about two orders of magnitude between glass at temperatures of 40 and 95°C. The NNWSI Project tests for leaching are routinely performed at temperatures of 90°C, which exceeds the anticipated temperature at which leaching would occur in the repository.

An additional important factor is the pH of the leaching solution, which can be controlled to some extent by the overall repository design. When the pH is between 5 and 9, leaching is at a minimum. Since the pH is initially buffered at Yucca Mountain by the presence of dissolved CO_2 and silica as well as by the dissolved glass components, this factor can be controlled. Large excursions in pH, which are feasible, would be more than the buffering capacity of the ground water at Yucca Mountain. Therefore, glass composition must be carefully controlled, which is beyond the NNWSI Project control except for rigid enforcement of acceptance standards or specifications. The use of acceptance standards is discussed in Sections 8.3.5.10 and 8.3.4.3 under Issues 1.5 and 2.6.

Studies of leach rates as a function of the volume of water indicate that volume is an important consideration. Scaling experiments indicate that about a sixfold shortening of experimental time is possible by increasing the surface area to volume by approximately the same factor. However, it is not clear whether the scaling can be applied to predict the result of much larger surface-area-to-volume ratios expected for Yucca Mountain. This, plus the effect of large concentrations of silica in the solution, will be considered in studies discussed in Section 8.3.5.10 under Issue 1.5 and Information Need 2.6.3.

Studies to look at the leach rates given the reference design for the waste packages, the Topopah Springs tuff, and well J-13 water indicate that 304L stainless steel has no substantial effect on leaching and that the presence of tuff slightly decreases the leaching rates. The presence of ductile iron is known to increase leach rates in static tests. Therefore, tests are to be conducted on heat-affected 304L stainless steel, which may be more reactive. This work is discussed in Sections 8.3.5.10 and 8.3.4.3.

Radiolysis studies indicate that, after an early decrease in pH due to radiation exposure (nitric acid generation), increased glass leaching occurred, which in turn buffered the acid production from radiolysis. Thus, for irradiation fluxes above 1×10^3 rads/h, the glass reaction rate becomes insensitive to the flux. Experiments to determine how much radiation is required to maintain the radiolysis buffer are discussed in Section 8.3.5.10. Actinide releases in gamma-irradiation are sensitive to the pH. However, anticipated pH changes that would occur at Yucca Mountain are expected to be small since the rock will be essentially dried out during the period of high gamma radiation flux.

Prediction of the long-term behavior of waste glasses will require a better understanding of secondary products that appear to substantially reduce initial leach rates but, in the long term, these crystalline products may provide the thermodynamic driving force for glass dissolution. The current knowledge of these compounds is not sufficient to rely upon solely as a control of radionuclide release. An approach under development is to use explicit geochemical modeling of aqueous species and glass dissolution kinetics to improve the understanding of this system. This work is discussed in Section 8.3.5.10.

7.5.4.9 EQ3/6 model development

The EQ3/6 geochemical modeling code will supplement field and experimental programs in the NNWSI Project. EQ3/6 will simulate waste package performance over tens of thousands of years and will extrapolate short-term laboratory experiments to much longer time scales. Simulation of laboratory experiments will aid in validating the code. The EQ3/6 package lends itself to modeling the controlled conditions of laboratory experiments. EQ3/6 can then be used to investigate the consequences of variations in the environment on waste package performance.

The EQ3/6 geochemical modeling program focuses on determining

1. The potential degradation of the waste container resulting from interactions between ground water, host rock, and gases in the unsaturated zone during and following the thermal pulse from radioactive decay of high-level waste.

2. The extent and degree of transport of radionuclides into the environment after container breaching and resulting ground-water-host rock interactions.

A three-phase approach is being carried out to address these issues:

1. The EQ3/6 code is being expanded and refined to provide for chemical processes that will control fluid-rock interaction and radionuclide transport in the short and long term.

2. Thermodynamic data are being acquired to describe solid substances and aqueous species.

3. Applications work is under way to

   a. Simulate laboratory results to evaluate code capabilities and interpret laboratory results.

   b. Complement the experimental program by simulating laboratory experiments under a wider variety of conditions than can be carried out in the laboratory.

   c. Extrapolate laboratory results to the long-time periods involved in waste disposal effects.

This work is further discussed in Section 8.3.5.10.

7.5.4.10 Waste package performance assessment

Information needs relative to performance assessment relate to the methodology used for the assessment. Most of the information used in this assessment will be collected outside of the performance assessment itself and is discussed in other portions of Chapter 7 and in Chapter 8. It is the task of performance assessment to construct and validate the computational models necessary and then to analyze waste package designs to demonstrate that they will perform as intended.

The overall approach that will be used is to develop a deterministic model to describe waste package performance under a given set of conditions and design parameters. The most likely approach to uncertainty or reliability analysis will be to use this deterministic model to calculate performance parameters based on input variables selected by a statistically valid method. Using a sufficient number of the calculations, distributions of time to loss of containment and engineering barrier system release rates will be constructed. Site characterization and materials performance data will be required to verify and validate the model and then to make the final waste package performance assessment. How this data will be obtained is discussed in Sections 8.3.4.2, 8.3.5.9, and 8.3.5.10.

A system model for waste package performance that is specific to the NNWSI Project site conditions and design parameters is under development. This model is a compilation of submodels that includes radiation, thermal, mechanical, environment, corrosion, waste form alteration, and waste transport models. Each of these models may consist of several interacting or related submodels. These models are discussed in Section 7.4.5.4, and additional work to further develop and refine the models is discussed in Section 8.3.5.10.

7.5.5 UNCERTAINTIES IN WASTE PACKAGE DEVELOPMENT

Throughout this chapter, references have been made to tests and investigations that are needed to gain additional information. The primary motivation for identification of these activities is reduction of significant uncertainties that are inherent in the project. These will require quantification by expert judgment and bounding calculation or assumptions explicitly stated in the bases for design and assessments of performance.

The following sections briefly discuss the principal sources of uncertainty in the major elements of the waste package development activities and make a preliminary judgment of those factors that are amenable to reduction by additional testing. Those factors that will require quantification by other means are also identified.

7.5.5.1 Waste package design

The waste package design development process provides the mechanism for accommodating the uncertainties in the service environment, the waste forms, and other package components that are discussed in the sections that follow. The design development process is structured to be an iterative, phased approach. The design bases for each phase of the design development will be reevaluated prior to initiation of the design phase to include revisions as appropriate that reflect the current information from the site investigations and other data, such as projections of the characteristics of the waste to be packaged and disposed in the repository. The resulting designs will then be evaluated by comparison of the predicted performance relative to the performance goals and revised as necessary.

Although the design development process introduces few uncertainties, it is structured to assess the effect of uncertainties in the design bases and to accommodate the residual uncertainties in the design itself.

7.5.5.2 Waste package environment

The major uncertainties in the waste package environment arise from inadequate characterization of the repository host rock mass and its response to the loads imposed by the construction of the facility and emplacement of the waste. The level of uncertainty is expected to be substantially reduced by the performance of the site characterization investigations described in Section 8.3. Some residual uncertainty will remain after that work is completed, simply because the natural variability of the host rock will not be completely explored and the response of the coupled geochemical-geohydrologic system for the long time span of required performance cannot be completely tested.

Two major technical uncertainties exist with respect to the near-field environment adjacent to waste packages. The first concerns the composition and form of the chemical products resulting from a system that contains unsaturated tuff, an aqueous fluid (liquid or vapor), and air at elevated temperatures and ionizing radiation levels that vary as a function of time. The second involves the processes that control the flow of vadose water in both liquid and vapor phases under the influence of elevated temperatures, especially those that exceed the unconfined boiling-point, in a fractured medium. Both of these uncertainties will be addressed in the laboratory and field tests that are planned during site characterization. Residual uncertainties will remain because the techniques that may be required to investigate some aspects of these processes involve imposing loads and flow rates that questions remaining on scale effects.

The investigations specifically planned to address these uncertainties, including the development of models for predicting the processes into the far future are described in Section 8.3.4.

7.5.5.3 Metallic containers

The major uncertainties in the metallic components of the waste package are related to the indentification of the degradation modes that will limit the service life and prediction of the rates or frequencies of occurrence of these models. These factors are directly related to the composition of the material and the microstructure and residual stresses resulting from the forming and joining processes used in fabrication. The metal alloy to be used has not been selected, nor have the fabrication processes been determined. Both of these activities have been planned and are described in Sections 8.3.4 and 8.3.5.9. Plans for the development of models to predict the degradation rates and relatively short (few year) duration tests to assist in the validation of these models are also discussed.

The uncertainties in the waste package environment (Section 7.5.5.2) also influence the predictions of the performance of the container. These factors will affect the quantities, speciation and concentration of species in both the vapor and liquid phases present to degrade the container material.

Residual uncertainties will exist due to the lack of relevant experience with currently available metallic materials over the time periods of concern in this project. The service lifetimes are far longer than for any other engineered structures and, therefore, beyond the limits of experience.

7.5.5.4 Waste forms

All the waste forms to be accepted for disposal in a geologic repository are subject to two important factors. First, they are known to possess variable physical and chemical properties at a scale such that detailed individual characterization is not feasible. This implies that the waste forms can only be sampled on a limited basis and the ranges of significant characteristics will require estimation based on the samples. The prediction of the long-term behavior of both waste forms depends upon developing suitably comprehensive models of waste form degradation and geochemical interactions. Uncertainties in this area involve the ability to identify and quantify the most important factors and to obtain an understanding of less important factors sufficient to provide confidence that the long-term models will be accurate to within a known statistical uncertainty.

The second important aspect is that a large fraction of the waste forms to be disposed in the repository do not exist now and will not exist prior to the application for a license for the disposal system. Thus, it will be necessary to develop projections of the important characteristics in a situation where DOE-OCRWM does not have control over the design and operation of the waste generation sources. The importance of these two inherent sources of uncertainty remains to be determined. In addition, there is some uncertainty introduced by regulatory proceedings to revise the definition of high level radioactive waste. The impact of this proposed rulemaking has not been assessed in this document.

7.5.5.4.1 Spent fuel waste forms

Uncertainties in the characteristics of the spent fuel to be received at the repository fall into two broad categories. The first involves the external physical characteristics that affect the design of the waste package and the repository, such as dimensions, configurations, quantities, thermal and radiation outputs. These characteristics, as indicated in Section 7.5.5.4, will remain uncertain to the extent that the waste receipt schedule reflects a projection into the future.

The second category of uncertainty involves those internal characteristics of the spent fuel that may significantly affect the rate of release of radionuclides from the waste packages following the containment period.

Limited testing of irradiated fuel so far has indicated that the release rates may be affected by a number of factors such as oxidation state of the uranium; inventory of nuclides occurring in the fuel-cladding gap and on fuel grain boundaries; magnitude and number of cladding defects; the inventory of nuclides, e.g., carbon-14, that may be released as gases and the sensitivity of this release to environmental conditions such as temperature; and the effect of the composition and quantity of water contacting the waste.

The planned activities described in Section 8.3.5.10 have been designed to address these uncertainties to the extent feasible in the time available for this phase of the project, including the testing of several spent fuels, and development of release models for the time periods of concern.

7.5.5.4.2 Glass waste forms

The high level glass waste forms have less variability in form, dimension, and external characteristics than spent fuel, but share the uncertainty arising from projecting the production of the waste form into the future because none of the vitrification facilities are yet in full scale operation.

The major sources of uncertainty for glass waste forms concern the physical characteristics that may significantly affect the release rate of radionuclides. Testing of the glass waste forms has been limited to surrogate materials compounded to approximate the reference composition of the production glass. Most of the test materials have been produced in very small melters, thus introducing some uncertainty with regard to scale effects. The nature of the wastes and vitrification processes implies that there will be batch compositional variation that will affect release rates. In general, the testing has been performed on glass that is a representative of the average material in a canister. However, the waste most readily available for initial release may be contained in inhomogeneous material at the top of the glass monolith, or splatter or condensate adhering to the canister in the empty space above the glass. Tests have not been performed yet on these types of material, or to determine the relative importance of these factors and glass cracking and canister-glass interactions.

The planned activities described in Section 8.3.5.10 have been designed to address these uncertainties to the extent feasible in the time available for this phase of the project, including the development of release models for the time periods of concern.

7.5.5.5 Waste package performance assessment

The principal uncertainties in the performance assessments arise from the need to propagate all of the uncertainties in the models that represent the process acting on the waste package components and the boundary conditions of the package environment through a coupled deterministic model of the system. This system model will necessarily involve simplifications of the coupled interactions and of the processes themselves to be manageable.

It is also inherently not possible to fully validate the system model over the time periods of concern. Therefore, an additional level of uncertainty will be introduced as to the efficiency of the simplifications to the process models and their interactions.

The methodology for probablistically analyzing the residual uncertainties in the waste package performance has not been selected. This selection will be based on the criterion that it provide a realistic analysis of the uncertainties and, to the extent possible, avoid introducing additional sources of uncertainty.

The plans for phased development of the waste package performance assessment system models are described in Section 8.3.5.10. The planned applications of these models to assess containment and controlled release performance of the waste packages and provide release predictions for the total system assessments are described in Sections 8.3.5.9 and 8.3.5.10.