QA: NA

MOL.20020404.0042

DOE/RW-0539-1

Yucca Mountain Science and Engineering Report

Technical Information Supporting Site Recommendation Consideration

NMSS07 WM-11
Rec'd w/out letter
Scan Paper Copy
FULL CD Distribution

Revision 1

U.S. Department of Energy
Office of Civilian Radioactive Waste Management
February 2002

This publication was produced by the U.S. Department of Energy
Office of Civilian Radioactive Waste Management.

For further information contact:
U.S. Department of Energy
Yucca Mountain Site Characterization Office
P.O. Box 364629
North Las Vegas, Nevada 89036-8629

or call:
Yucca Mountain Information Center
1-800-967-3477

or visit:
Yucca Mountain Site Characterization Project website
http://www.ymp.gov

CONTENTS

CONTENTS (Continued)

Page

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

CONTENTS (Continued)

FIGURES

FIGURES (Continued)

FIGURES (Continued)

FIGURES (Continued)

Page

FIGURES (Continued)

FIGURES (Continued)

Page

FIGURES (Continued)

Page

FIGURES (Continued)

Page

FIGURES (Continued)

Page

FIGURES (Continued)

FIGURES (Continued)

INTENTIONALLY LEFT BLANK

TABLES

TABLES (Continued)

TABLES (Continued)

INTENTIONALLY LEFT BLANK

EXECUTIVE SUMMARY

I. INTRODUCTION

Commercial electric power generation, nuclear weapons production, the operation of naval reactors, and research and development activities produce spent nuclear fuel and high-level radioactive waste. These radioactive materials have accumulated since the mid-1940s at sites now managed by the U.S. Department of Energy (DOE) and since 1957 at commercial reactors and storage facilities across the country. The responsible management and disposal of these materials is a critical part of the DOE mission to dispose of high-level radioactive waste and spent nuclear fuel from federal facilities, including the nuclear weapons program, as well as commercially generated spent nuclear fuel.

The U.S. has evaluated methods for the safe storage and disposal of radioactive waste for more than 40 years. Many organizations and government agencies have participated in these studies. At the request of the U.S. Atomic Energy Commission, the National Academy of Sciences evaluated options for land disposal of radioactive waste in the 1950s. The U.S. Atomic Energy Commission and its successor agencies, the U.S. Energy Research and Development Administration and the DOE, continued to analyze nuclear waste management options throughout the 1960s and 1970s. In 1979, an Interagency Review Group that included representatives of 14 federal government entities provided findings and recommendations to the President. After analyzing a range of options, disposal in mined geologic repositories emerged as the preferred long-term environmental solution for the management of spent nuclear fuel and high-level radioactive waste. This consensus was reflected in the Nuclear Waste Policy Act of 1982 (NWPA), which established the U.S. responsibility and policy for the disposal of spent nuclear fuel and high-level radioactive waste.

Congress established the framework for addressing the issues of nuclear waste disposal in the NWPA and related statutes and designated the roles and responsibilities of the federal government and the owners and generators of the waste. Congress assigned responsibility to:

- The DOE to site, construct, operate, and close a repository for the disposal of spent nuclear fuel and high-level radioactive waste
- The U.S. Environmental Protection Agency (EPA) to set public health and safety standards for releases of radioactive materials from a repository
- The U.S. Nuclear Regulatory Commission (NRC) to promulgate regulations governing the construction, operation, and closure of a repository
- The generators and owners of spent nuclear fuel and high-level radioactive waste to pay the costs of disposal of such radioactive materials.

Congress amended the NWPA in 1987 and directed the DOE to investigate Yucca Mountain, Nevada, exclusively, to determine whether it is a suitable site for the first geologic repository for the nation's spent nuclear fuel and high-level radioactive waste. The DOE has studied Yucca Mountain for more than 20 years to characterize the site and assess the future performance of a potential repository. Preliminary engineering specifications have been developed for surface and subsurface facilities and the waste package. Analyses that integrate design- and site-specific data and models have been conducted to assess how a repository at Yucca Mountain might perform.

Yucca Mountain Science and Engineering Report describes the results of scientific and engineering studies of the Yucca Mountain site, the waste forms to be disposed, the repository and waste package designs, and the results of the most recent assessments of the long-term performance of the potential repository. The scientific investigations include site characterization studies of the geologic, hydrologic, and geochemical environment, and evaluation of how conditions might evolve over time. These analyses considered a

range of processes that would operate in and around the potential repository. Since projections of performance for 10,000 years are inherently uncertain, the uncertainties associated with analyses and models of long-term performance are also described, along with the likely impact of these uncertainties on performance assessment.

The NWPA specifies a process for the recommendation and approval of a site for development of a repository, and it requires that the Secretary of Energy provide a comprehensive statement of the basis for any site recommendation. *Yucca Mountain Science and Engineering Report* contains information that would be included in any site recommendation from the Secretary to the President, consistent with Sections 114(a)(1)(A), (B), and (C) of the NWPA (42 U.S.C. 10134(a)(1)(A), (B), and (C)). The NWPA requires that the DOE hold public hearings in the vicinity of the site before the Secretary makes a decision whether or not to recommend the Yucca Mountain site. The DOE has held numerous public hearings to inform residents of the area, including all 17 counties in Nevada and Inyo County, California, that the site is being considered for possible recommendation and to receive their comments. In May 2001, concurrent with the release of the initial version of this report, the DOE opened a public comment period on the Secretary's consideration of the possible recommendation of the Yucca Mountain site. This initial version and its references provided information for public review and comment in advance of public hearings. These reports also supported the process of informing the public, elected officials, affected units of government, Indian tribes, regulatory agencies, review groups, and other interested parties of the Secretary's consideration of a possible recommendation of the Yucca Mountain site. Since the beginning of the public comment period, the DOE has also made available for public review several supplemental analyses that are being considered as part of the basis for any site recommendation decision.

The DOE's scientific and technical understanding of the Yucca Mountain site continues to evolve and improve as the DOE proceeds through completion of site characterization activities, the site recommendation process, and any license application process. If the site is recommended by the Secretary and the site designation becomes effective, the DOE anticipates that this evolution in understanding of the site and repository design will continue through construction and operation of the repository if the repository is licensed.

II. SOURCES OF MATERIALS CONSIDERED FOR DISPOSAL

By statute, the DOE is responsible for the safe, permanent disposal of spent nuclear fuel from commercial nuclear power plants. The DOE must also dispose of large quantities of DOE-owned high-level radioactive waste from the production of nuclear weapons and smaller quantities of spent nuclear fuel from weapons production reactors, research reactors, and naval reactors.

The NWPA limits the amount of spent nuclear fuel and high-level radioactive waste that can be emplaced in the nation's first geologic repository to 70,000 MTHM until a second repository is in operation. The materials that may be disposed at Yucca Mountain include about 63,000 MTHM of commercial spent nuclear fuel; about 2,333 MTHM of DOE spent nuclear fuel; and about 4,667 MTHM of DOE high-level radioactive waste. All the waste forms transported to and received at a repository would be solid materials. No liquid waste forms would be accepted for disposal. Figure 1 shows the current locations and types of waste that would be emplaced at a repository.

As of December 1999, the United States had generated about 40,000 MTHM of spent nuclear fuel from commercial nuclear power plants. This amount could more than double by 2035 if all currently operating plants complete their initial 40-year license period. By 2035, the United States will also have about 2,500 MTHM of spent nuclear fuel from research reactors, naval reactors, reactor prototypes, and reactors that produced nuclear weapons materials. The majority of this spent nuclear fuel is stored at DOE sites in Idaho, South Carolina, and Washington. In addition, liquid waste from nuclear weapons production programs is stored in underground tanks at the same DOE sites. This high-level radioactive waste will be mixed

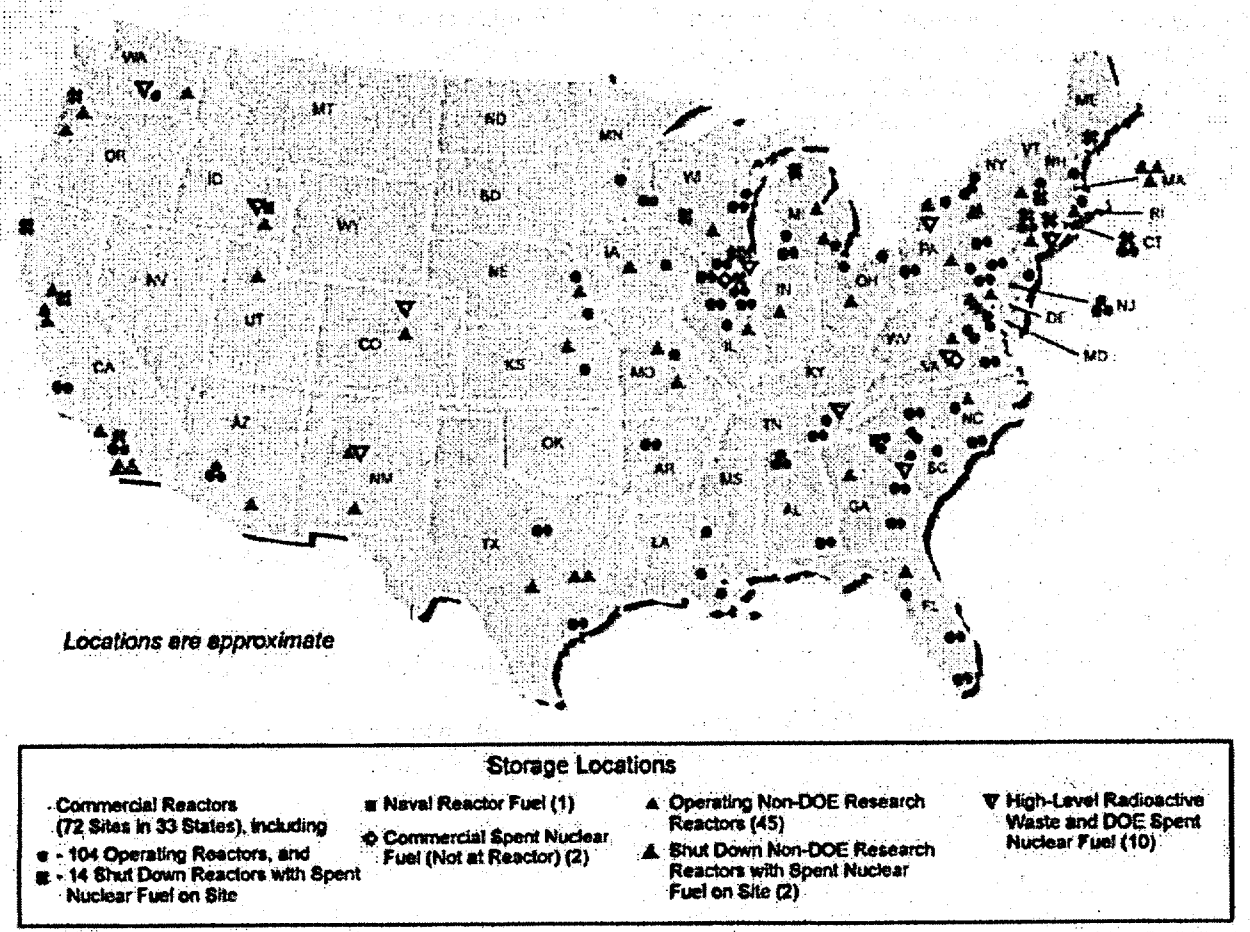


Figure 1. Map Showing Current Locations of Waste Destined for Geologic Disposal
Commercial waste may include West Valley vitrified waste if an agreement is reached between the DOE and the state of New York for disposal of this waste.

with silica sand and other constituents, melted together, and poured into stainless steel canisters. Once the glass solidifies (vitrifies), the canister would be sealed, loaded into a transport cask, and shipped to a repository. Vitrified waste is resistant to dissolution and would stay intact for thousands of years.

Some of the material that would be disposed in a repository would come from surplus plutonium resulting from the production and decommissioning of nuclear weapons. A nominal 50 metric tons of surplus plutonium must be safely dispositioned. Current plans call for some of the surplus plutonium to be combined with uranium to form fuel that would be used in commercial reactors. The resulting spent nuclear fuel would be disposed as commercial spent nuclear fuel. Some of the surplus plutonium would be immobilized in ceramic and placed in canisters for repository disposal. These canisters will be filled with molten high-level radioactive waste glass, which will vitrify into a glass waste form.

III. GEOLOGY OF THE YUCCA MOUNTAIN SITE

The Yucca Mountain site is located on federal land adjacent to the Nevada Test Site in Nye County, Nevada, about 160 km (100 mi) northwest of Las Vegas. The mountain consists of a series of ridges extending 40 km (25 mi) from Timber Mountain in

the north to the Amargosa Desert in the south. The water table at Yucca Mountain is approximately 500 to 800 m (1,600 to 2,600 ft) below the surface of the mountain at the potential repository location. The zone of soil or rock below the ground surface and above the water table is called the unsaturated zone. The underground facility would be located in the unsaturated zone, about 200 to 500 m (660 to 1,600 ft) below the surface and, on average, about 300 m (1,000 ft) above the water table. The deep water table and thick unsaturated zone at Yucca Mountain result from the low infiltration rate of surface water due to low annual rainfall and high rates of evaporation and transpiration (the process by which water vapor passes from soil into plants, then into the air).

The potential repository would be located in volcanic rock, called tuff, that was deposited by a series of eruptions between approximately 11 and 14 million years ago. The characteristics of the volcanic rock have been studied in underground excavations and boreholes, and by geologic mapping of the surface. Mapping and other studies show that faults are present in the vicinity of Yucca Mountain. The location, timing, and amount of movement on these faults have been characterized as part of the DOE's seismic hazard analysis.

The location of the underground facility was identified using several factors, including the thickness of overlying rock and soil, the characteristics of the rock that would host the repository, the location of faults, and the depth to groundwater. The facility would be sited deep enough underground to prevent waste from being exposed to the environment and to discourage human intrusion. The host rock for a geologic repository must be stable enough to sustain excavated openings during repository operations. The rock must also be able to absorb heat generated by the spent nuclear fuel. The Topopah Spring Tuff rock unit, in which the underground facility would be constructed, exhibits these characteristics.

IV. REPOSITORY DESIGN

The DOE has developed a design for a Yucca Mountain disposal system that could give future generations the choice of either closing and sealing the underground facility as early as allowable under NRC regulations or keeping it open and monitoring it for a longer time period. The design for the potential repository would not preclude the option for future generations to make societal decisions to monitor the repository for up to 300 years before making decisions to close the underground facility.

Figure 2 is a conceptual illustration of the proposed facilities and structures that would make up a potential repository. It shows the facilities as they would appear after construction. In general, the operations that would be performed include:

- Receiving spent nuclear fuel and high-level radioactive waste in shipping casks certified by the NRC from rail and truck transporters
- Unloading, handling, and packaging spent nuclear fuel and high-level radioactive waste into waste packages suitable for underground emplacement
- Transporting waste packages from the surface to the underground facility
- Emplacing waste packages in underground drifts
- Monitoring operations and repository system performance to ensure the safety of workers and the public
- Decommissioning and closure.

The design that has been developed is intended to fulfill the functional requirements defined for the facilities while maintaining the flexibility to adapt to various construction and operational conditions and requirements. Four key aspects of design flexibility are:

1. The ability of the repository design to support a range of construction approaches (e.g., change in emplacement drift spacing, modular or sequential construction of surface and subsurface facilities)

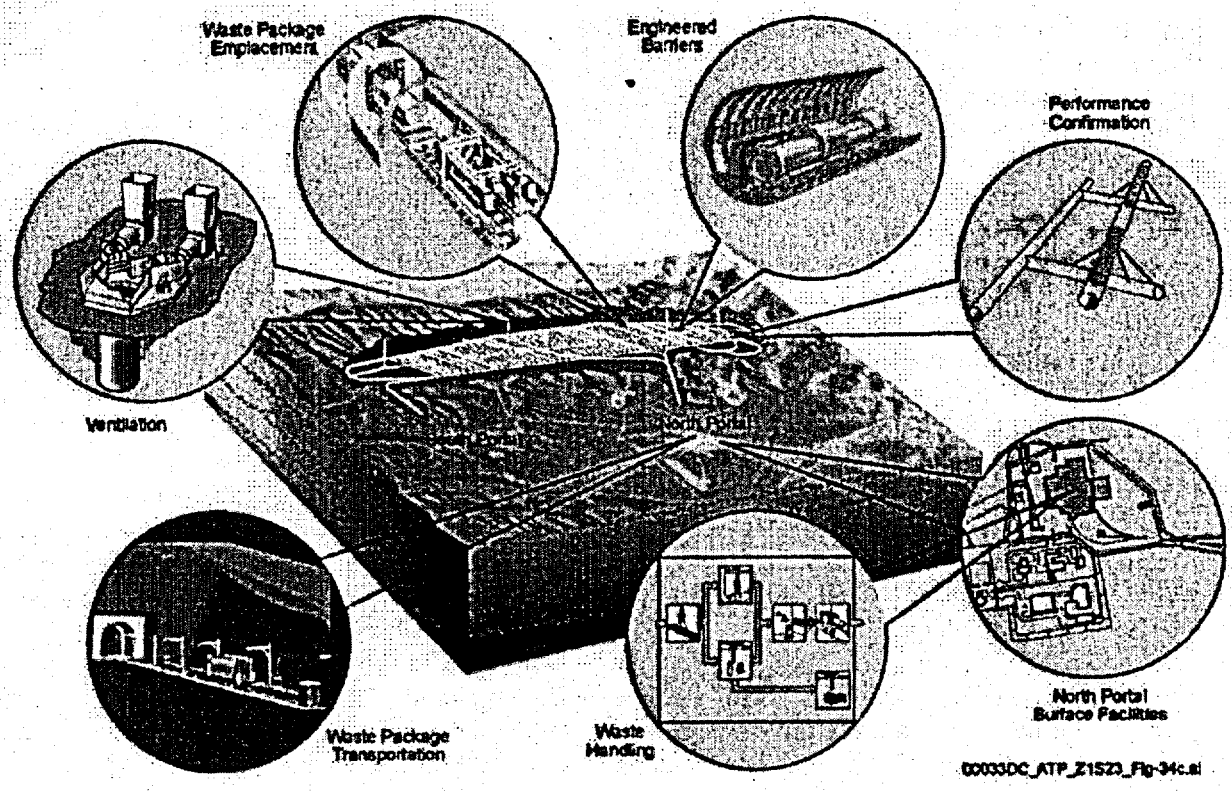


Figure 2. Proposed Repository Facilities

2. The capability to dispose of a wide range of radioactive waste container sizes

3. The ability to support a range of thermal operating modes (e.g., defining a larger waste emplacement area or varying ventilation duration and rates to reduce the temperature in the underground facility)

4. The ability to continue to enhance the design to best achieve performance-related benefits identified through ongoing analyses.

Thermal Management Strategy—Radioactive elements in spent nuclear fuel and high-level radioactive waste become less radioactive over time. As part of this process, energy is released in the form of heat. In the potential repository, this heat could affect thermal, hydrologic, chemical, and mechanical processes in the emplacement drifts and surrounding rock. The DOE plans to manage the thermal environment of the repository to take advantage of potentially beneficial characteristics that might be associated with alternative operational modes. For example, a repository operated at higher temperatures (i.e., above the boiling point of water) would dry out the emplacement drifts, thereby limiting the amount of water available to contact waste packages. A repository operated at lower temperatures would cause less disturbance of the local environment near the emplaced waste and hydrologic and geochemical processes in the rock would be less complex than in a higher-temperature operating mode. For this reason, models of the performance of a repository operated at lower temperature may be less complex and possibly less uncertain than the models of the performance of a higher-temperature repository.

The general repository design concept provides flexibility for operation over a range of thermal operating modes. This range has been and continues to be examined to identify the potential

benefits of different environmental conditions (higher or lower temperatures and associated humidity conditions) in the emplacement drifts. The temperatures at the drift wall and waste package surfaces can be varied, along with the relative humidity, by modifying operational parameters such as the thermal output of the waste packages, the spacing of waste packages in emplacement drifts, and the duration and rate of ventilation (e.g., active ventilation that uses fans or passive ventilation that relies on natural air flow).

Surface Facilities—The potential repository's surface facilities would be located in the North Portal Repository Operations Area, the South Portal Development Area, and the Surface Shaft Areas. All waste receipt and handling operations would be conducted at the North Portal area in the Waste Handling Building.

The Waste Handling Building would receive, prepare, and package the waste for emplacement underground in the repository. All waste handling operations would be conducted using remotely operated equipment. Thick concrete walls, air locks, and controlled area access techniques would be used to protect workers from radiation exposure. The Waste Handling Building and its equipment would also be designed to withstand the effects of ground motion from potential earthquakes on repository operations. The Waste Handling Building would house all systems necessary to prepare waste for emplacement. These include:

- The carrier/cask handling system, which would receive and unload transportation casks from rail and truck carriers

- The assembly transfer system, which would receive transportation casks containing spent nuclear fuel assemblies from commercial reactors, unload the assemblies from the casks, and load the assemblies into disposal containers

- The canister transfer system, which would receive transportation casks containing canisters of DOE high-level radioactive waste or spent nuclear fuel, unload the canisters from the casks, and load the canisters into disposal containers

- The disposal container handling system, which would receive loaded disposal containers from the assembly and canister transfer systems and install and weld closure lids onto the disposal containers (referred to as waste packages after they have been loaded, sealed, and inspected)

- The waste package remediation system, which would receive waste packages that are damaged, have failed the inspection process, or have been selected for retrieval from the repository to examine their performance. Waste packages that are damaged or fail inspection would be repaired or repackaged into another container.

Underground Facility—The potential repository's underground facility would be designed to contribute to the isolation of waste. Waste packages would be disposed in dedicated drifts, supported on emplacement pallets, and aligned end-to-end on the drift floor. For the higher-temperature operating mode, the packages would be spaced about 10 cm (4 in.) apart. The base design includes 58 horizontal emplacement drifts excavated to a 5.5-m (18-ft) diameter at a center-to-center drift spacing of 81 m (266 ft). The total subsurface area required to accommodate 70,000 MTHM is about 1,150 acres. For the lower-temperature operating mode, several alternative waste package and emplacement drift spacings and configurations have been and continue to be evaluated. A final determination of the waste package and drift configuration has not been made. However, a larger area (up to about 2,500 acres) may be required for a lower-temperature operating mode. In the present design, the underground facility would be constructed over a period of about 23 years.

Waste packages would be moved, one at a time, from the surface to the emplacement drifts by way of a connecting rail system. Equipment in the Waste Handling Building would place a waste package into a shielded transporter. Two electric locomotives, one on each end of the transporter,

would move the transporter down the North Ramp, through the repository's main access drift, to an emplacement drift. Once the transporter arrives at the assigned emplacement drift and the drift's isolation doors are opened, the transporter's shielded doors would be opened and the waste package would be moved out of the transporter using a retractable deck. An in-drift gantry would lift the waste package and its supporting pallet off the deck and deposit them in their designated position inside the drift. Before the repository is permanently closed, overlapping and interlocking drip shields would be placed over the waste packages to divert any water that might drip from the top of the emplacement drifts.

Emplacement operations would take place in finished emplacement drifts at the same time as future emplacement drifts are being constructed. During construction, separate ventilation systems operating on the development side and the waste emplacement side would allow separate regulation of airflow to accommodate different needs. During emplacement, ventilation would maintain temperatures within the range for equipment operation. Before closure, ventilation would remove most of the heat generated by the waste packages and keep the relative humidity low.

V. NATURAL BARRIERS

The barriers important to waste isolation are broadly characterized as natural barriers, associated with the geologic and hydrologic setting, and engineered barriers, discussed in the following section. The engineered barriers are designed specifically to complement the natural system in prolonging radionuclide isolation within the disposal system and limiting their potential release. The natural barriers at Yucca Mountain include:

- Surface soils and topography
- Unsaturated rock layers above the repository
- Unsaturated rock layers below the repository
- Volcanic tuff and alluvial deposits below the water table.

Natural barriers would contribute to waste isolation by (1) limiting the amount of water entering emplacement drifts and (2) limiting the transport of radionuclides through the natural system. In addition, the natural system would provide an environment that would contribute to the long lives of the waste packages and drip shields.

The location, elevation, and configuration of the underground facility was based on several factors that take advantage of the natural barriers, including the thickness of overlying rock and soil, the extent and geomechanical characteristics of the host rock, the location of faults, and the depth to groundwater. The host rock for a potential repository should be able to sustain the excavation of stable openings that can be maintained during repository operations and that would isolate the waste for an extended period after closure. In addition, the rock should be able to absorb any heat generated by the waste without undergoing changes that could threaten the site's ability to safely isolate the waste. The host rock should be of sufficient thickness and lateral extent to construct an underground facility large enough to support the design's intended disposal capacity. Moreover, the amount of suitable host rock should provide adequate flexibility in selecting the depth, configuration, and location of the facility.

The Topopah Spring Tuff, which would be the host rock for the potential repository, has a maximum thickness of about 375 m (1,230 ft) near Yucca Mountain. Site characterization studies to date have shown that the Topopah Spring Tuff has the features and characteristics listed above. The results of laboratory and underground testing to date show that the heat added by the emplaced waste would not adversely affect the stability of the geologic repository operations area. Design analyses and experience in the Exploratory Studies Facility indicate that stable openings can be constructed and maintained in the Topopah Spring Tuff.

The distribution and characteristics of fractures in the rock units at Yucca Mountain are important because in many of the hydrogeologic units, particularly the welded tuffs, fractures are the dominant pathways for water flow in both the unsaturated and saturated zones. By controlling where, and at what rates, water is likely to flow under various conditions, the fracture systems are expected to

play a major role in the performance of the disposal system. The underground facility has been designed to take advantage of the free-draining nature of the repository host rock, which would promote the flow of water past the emplaced waste and limit the amount of water available to contact the waste packages.

Fractures are common in the Topopah Spring Tuff. These fractures provide the main pathways for water to flow through the rock unit that would host emplaced waste. The water table below the Yucca Mountain site is located within the rock units called the Calico Hills Formation and the Crater Flat Group which are less fractured, which may result in fewer fracture flow pathways and slower flow through these units. Another important feature of the tuffs of the Calico Hills Formation is the abundance of zeolite minerals in the rock matrix and fractures. Zeolites are silicate minerals that have the ability to sorb (take up on their mineral surface and hold) many types of radionuclides and other ions that might be transported in solution in water.

VI. ENGINEERED BARRIERS

The components of the engineered barrier system are designed to complement the natural barriers in isolating waste from the environment. The repository design includes the following engineered barriers: the waste package, the waste form, the drip shield, and the emplacement drift invert. Figure 3 depicts waste packages within an emplacement drift.

The engineered barriers would contribute to waste isolation by (1) using long-lived waste packages

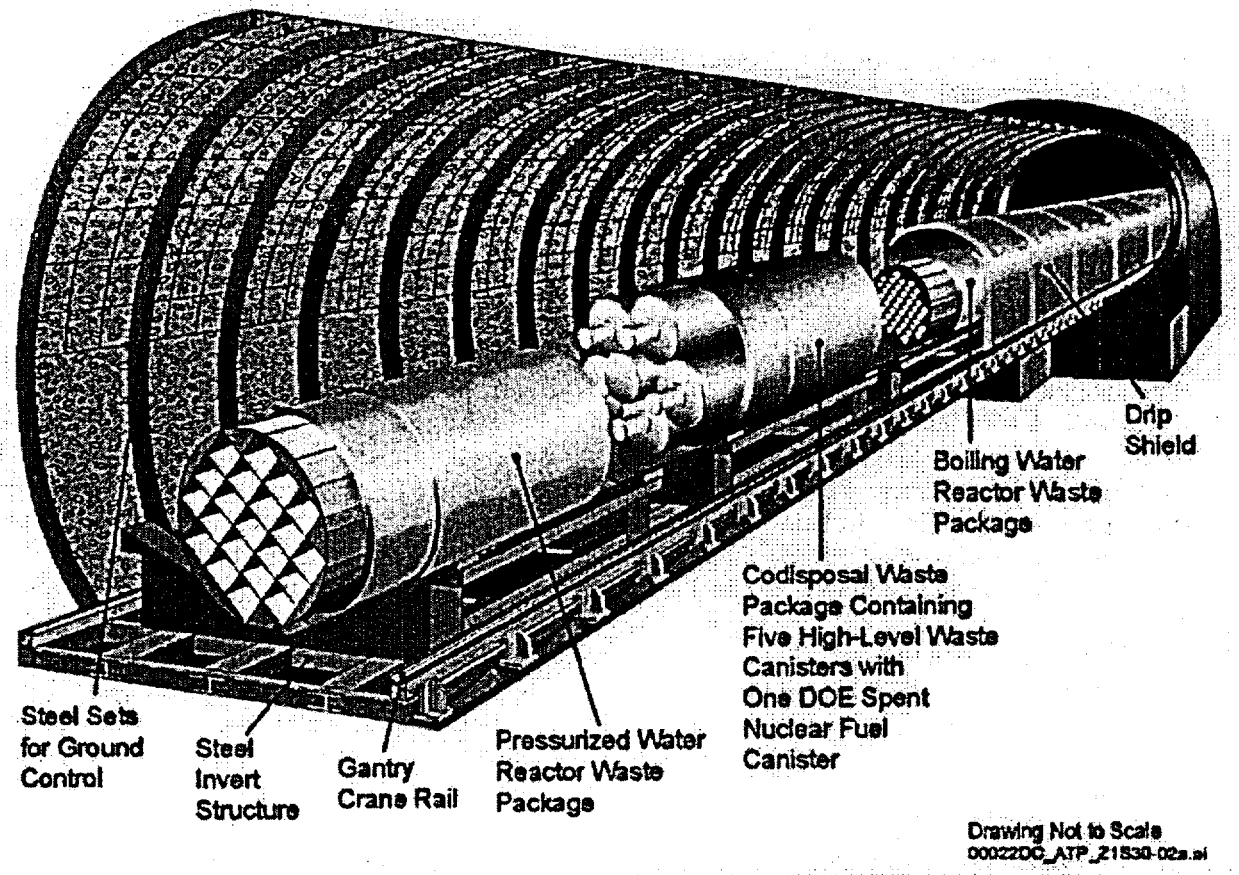


Figure 3. Schematic Illustration of the Emplacement Drift, with Cutaway Views of Different Waste Packages

and drip shields to keep water away from the waste forms and (2) limiting release of radionuclides from the engineered barriers through components engineered for optimum performance in the expected environment.

Waste Packages—Waste packages would have a dual-metal design containing two concentric cylinders. The inner cylinder would be made of Stainless Steel Type 316NG. The outer cylinder would be made of a corrosion-resistant, nickel-based alloy (Alloy 22). Alloy 22 would protect the stainless steel inner cylinder from corrosion, and Stainless Steel Type 316NG would provide structural support for the thinner Alloy 22 cylinder. Corrosion tests have been performed and are continuing in a variety of thermal and chemical environments to provide additional information on the corrosion rate of Alloy 22. Numerous analyses of the expected performance of the waste package and associated uncertainties have been performed. Tests and analyses indicate that Alloy 22 would last considerably longer than 10,000 years in the range of expected repository environments at Yucca Mountain.

Each waste package would have outer and inner lids at each end of the cylinder. The outer (closure) lids would be made of Alloy 22, and the inner lids would be made of Stainless Steel Type 316NG. The loading end of the waste package has a third flat closure lid made of Alloy 22, which would be placed between the inner lid of stainless steel and the outer lid of Alloy 22. The flat closure lid provides an extra barrier against a potential release caused by cracks and corrosion in the closure weld areas.

The basic waste package design is the same for all the waste forms. However, the sizes and internal configurations vary to accommodate the different waste forms. Figure 3 illustrates several common internal designs, including two for different types of commercial spent nuclear fuel and one for high-level radioactive waste and DOE spent nuclear fuel (a codisposal package).

Waste Form—The materials that would be disposed at Yucca Mountain include spent nuclear fuel and vitrified high-level radioactive waste. Both of these waste forms are solid materials that will degrade very slowly in the unsaturated environment of the repository. Spent nuclear fuel primarily consists of heavy metal oxides of uranium, plutonium, and other radionuclides. High-level waste consists of a vitrified borosilicate glass containing radionuclides. The release of radionuclides from the waste form will further be limited by the low solubilities of most radionuclides in the oxidizing environment of the unsaturated repository.

Drip Shields—Drip shields would be installed over the waste packages prior to repository closure. The drip shields would divert any moisture that might drip from the drift walls, as well as condensed water vapor, around the waste packages to the drift floor. All the drip shields would be the same size, so one design could be used with all the waste packages. The drip shields would be made of titanium, which would provide corrosion resistance and structural strength. They are designed to divert moisture around waste packages for thousands of years. Tests continue on drip shield materials to assess how well current data and models can be extrapolated over long periods of time. The drip shields would also maintain their function in the event of expected rockfalls, as the emplacement drifts degrade over time.

Drift Invert—The invert includes the structures and materials that would support the pallet and waste package, the drift rail system, and the drip shield. It is composed of two parts: the steel invert structure and the ballast (or fill) that consists of granular material.

Following closure, one function of the granular material in the invert would be to provide a layer of material below the waste packages that would slow the movement of radionuclides into the host rock. Water is not expected to accumulate and flow beneath the drip shields, so the most likely way radionuclides could move is by diffusion (where dissolved or suspended particles migrate slowly from zones of high concentration to zones of low concentration) through thin films of water on the granular material.

VII. SITE CHARACTERIZATION DATA AND ANALYSES

During the site characterization program, the DOE has performed extensive surface-based tests and investigations, underground tests, laboratory studies, and modeling activities designed to provide the technical information necessary for the evaluation of long-term repository performance. The site characterization program has evolved in response to advancements in scientific understanding, changes in regulatory requirements, and changes in program requirements, such as changes in design requirements for the potential repository.

Yucca Mountain Science and Engineering Report describes the data collected during site characterization and explains the DOE's understanding of the processes that could affect the ability of the potential repository system to isolate waste. Computer models of the hydrologic, geochemical, thermal, and mechanical processes that would operate in the repository system over time have been developed from the data collected. These process models have been used to develop an overall total system performance assessment (TSPA) model that evaluates how the potential repository may behave for 10,000 years. Analyses have also considered uncertainty in model results, and have evaluated alternative conceptual models to assess the extent to which the results of performance assessment depend on the details of the underlying process models.

VIII. PROCESSES IMPORTANT TO LONG-TERM REPOSITORY PERFORMANCE

The processes important to the repository's performance after it is closed (postclosure) include those that control the movement of water through the geologic setting. These processes begin with precipitation, as rain and snow, at the surface, a fraction of which infiltrates into the mountain. This net infiltration would move through the unsaturated zone to the level of the emplacement drifts, then downward through the unsaturated zone to the saturated zone. Within the saturated zone, water would move laterally away, where it could eventually reach the accessible environment. Within the underground facility, water moving past the engineered barriers would be affected by the physical and chemical processes associated with heat from the emplaced waste. These processes could eventually corrode waste packages, degrade the waste form, and dissolve some of the waste. Only after all of these processes have occurred could radionuclides move out of the repository.

To analyze the possible future performance of the repository, the DOE has studied many of the physical and chemical processes that would act on the repository system's barriers. The results of these studies are provided in *Yucca Mountain Science and Engineering Report*, along with an analysis of how the parts of the system would work together to isolate waste. The following subsections present a summary of the interdependent processes that may affect the repository's ability to isolate waste. Figure 4 illustrates the processes that were considered and modeled for the TSPA. Certain disruptive events that could affect these processes are also considered in the following discussion. The treatment of uncertainties associated with these processes is addressed in a later section.

Unsaturated Zone Flow—Because of the present-day arid climate at Yucca Mountain and the surface processes of runoff, evaporation, and transpiration, the amount of water available to contact and transport radionuclides is expected to be small. However, the availability of water may increase as a result of wetter climates that may occur in the future. The higher-temperature operating mode described in this report uses the heat produced by the waste to effectively limit the potential for contact between water and waste packages for hundreds to thousands of years. The pillar area between the emplacement drifts would be maintained below the boiling point of water to promote water drainage through the cooler portions of the rock pillars. Both the higher-temperature and lower-temperature operating modes would also take advantage of natural processes that would divert water around drift openings. Most water moving in the unsaturated zone flows through fractures. Water flowing in narrow fractures will usually remain in them rather than flow into large openings, such as drifts, because of capillary pressure in the fractures. Thus, capillary forces and water flow in unsaturated zone fractures would

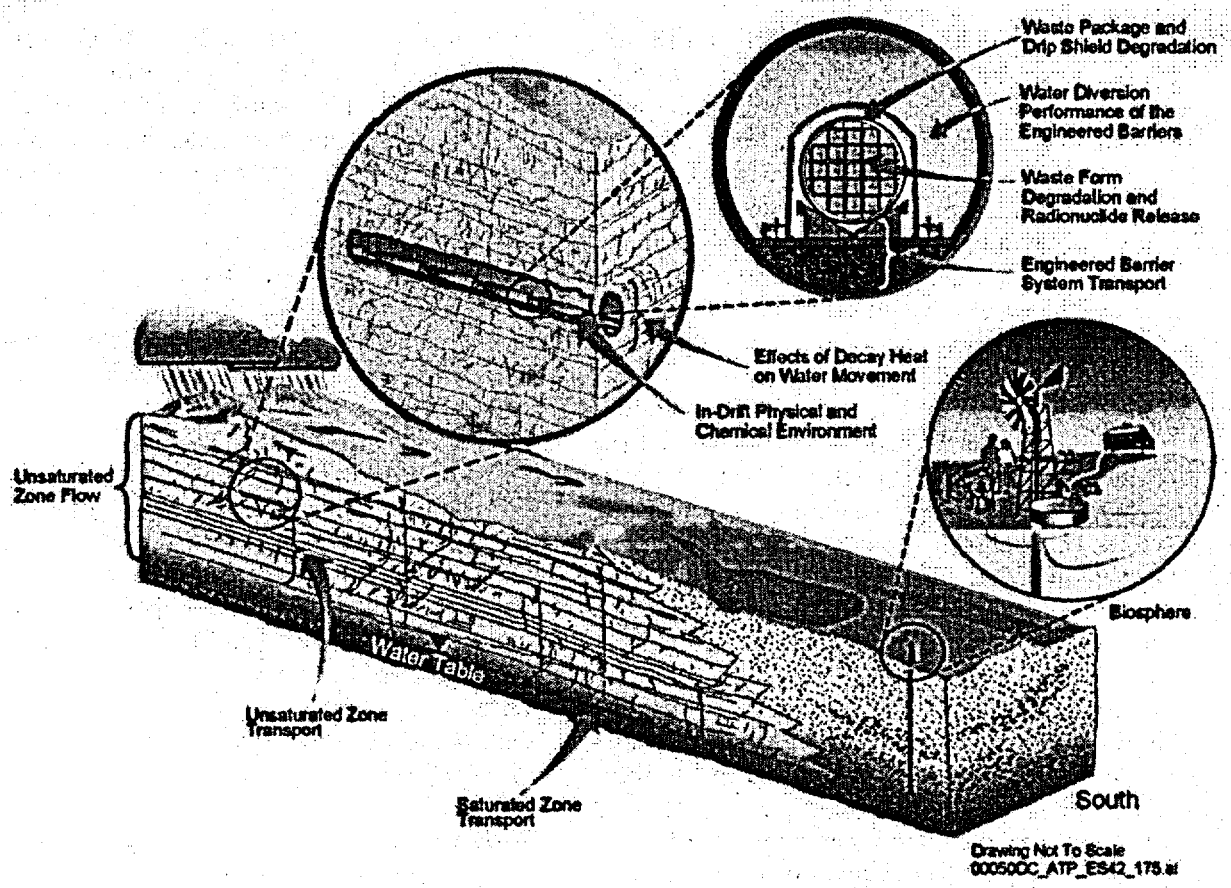


Figure 4. Schematic Illustration of the Processes Modeled for Total System Performance Assessment

limit seepage into openings, allowing most water to move past (not into) the emplacement drifts. If water does seep into an emplacement drift, most of it would flow down the drift wall to the floor and drain without contacting either the drip shields or the waste packages. Models of repository performance have evaluated both higher- and lower-temperature operating modes.

Many processes have been studied during the evaluation of the Yucca Mountain site. Important processes considered in the unsaturated zone flow models include:

- Climate and infiltration
- Fracture flow or matrix flow within major rock units
- Flow above the underground facility
- Flow below the underground facility, including the formation and significance of perched water (a zone of saturation separated from the water table by an intervening unsaturated zone)
- Fracture–matrix interaction
- Effects of major faults
- Seepage into drifts.

Effects of Heat on Water Movement—In the higher-temperature operating mode, no liquid water can remain in the emplacement drifts, and very little can remain in the nearby rock as long as the drift wall remains at temperatures above the boiling point of water. Even after hundreds to a few thousand years, when the waste packages have cooled below the boiling point of water, their

continued, but reduced heat production will still cause evaporation in and near emplacement drifts, thereby limiting the amount of water in the rock near the waste packages. The heat from emplaced waste may change the flow properties of the rock, as well as the chemical composition of the water and minerals in the engineered barrier system and surrounding rock. The nature and extent of these effects, however, would depend on thermal loading, ventilation rates and durations, and thermal operating properties.

Because the repository would be ventilated during operations, the major effects of heat on water movement would take place during the postclosure period. Thermal-hydrologic processes in the repository environment will determine the conditions in the drift, including temperature, relative humidity, and seepage at the drift wall.

In the lower-temperature operating modes that the DOE has evaluated, the heat from emplaced waste would still increase evaporation rates and, to some extent, dry out the rock near the emplacement drifts. Although liquid water could enter drifts, it is likely that only limited amounts of water would be available because of fracture flow and processes related to seepage.

Physical and Chemical Environment—The lifetimes of the drip shields and waste packages would depend on the conditions to which they are exposed: the in-drift physical and chemical environment. Once water enters a degraded waste package, the transport of radionuclides released from the waste form inside also depends on the drift environment. For the higher-temperature operating mode described in this report, the repository environment would be warm, with temperatures at the surface of the waste package initially increasing above the boiling point of water. The expected duration of temperatures above the boiling point of water on the surfaces of the waste packages would be hundreds to thousands of years. The precise time period varies for three main reasons: (1) location within the repository layout, (2) spatial variation in the infiltration of water at the ground surface, and (3) variability in the heat output of individual waste packages. The repository edges would cool first because they would lose heat to the cooler rock outside the perimeter. Water percolating downward through the host rock in response to infiltration at the ground surface would hasten cooling of the repository; locations with greater percolation cool faster. The heat output of individual waste packages will vary, depending on the type and age of the waste they contain; however, this variability can be managed. Except for the first few hundred years, the environment of the repository would be similar for both higher- and lower-temperature operating modes.

The chemical environment is expected to be at near-neutral pH (mildly acidic to mildly alkaline) and mildly oxidizing. Under such conditions, Alloy 22 will form a thin, stable oxide layer that is extremely corrosion resistant. Important processes affecting the chemical environment include the evaporation and condensation of water, the formation of salts, and the effects of gas composition. While the repository environment is warm, relative humidity will probably control the water chemistry.

Waste Package and Drip Shield Degradation—The lifetimes of the drip shields and waste packages will depend on the environment to which they are exposed and the degradation processes that occur. Corrosion is the most important degradation process considered in selecting the materials for the waste package and drip shield. A number of corrosion processes have been investigated in detail. The results have been used to support the selection of materials and the design of components.

Because most corrosion occurs only in the presence of water, and because highly corrosive chemical conditions are not expected in the repository environment, both the titanium drip shield and the Alloy 22 outer layer of the waste package are expected to have long lifetimes. Analyses based on laboratory tests and other evaluations indicate that, in the absence of a disruptive event or human intrusion, no waste packages are expected to be breached by corrosion for more than 10,000 years. The DOE has also evaluated the possibility that waste packages could be breached by processes other than corrosion such as stress corrosion cracking. These analyses indicate that improper

heat treatment of welds could cause cracks in a small number of packages (approximately zero to three) over 10,000 years. The small number of breaches would not have a significant impact on repository performance.

Analyses to date indicate that the drip shields and waste packages will be long-lived for both higher- and lower-temperature operating modes. The DOE has evaluated and will continue to evaluate whether keeping waste package surface temperatures cooler would improve performance or reduce the uncertainty in the models used. Water could contact waste packages sooner in lower-temperature operating modes. However, the physical environment of lower-temperature operating modes may reduce the potential for corrosion susceptibility of Alloy 22. Analyses to date have not demonstrated a significant difference in waste package and drip shield performance between higher- and lower-temperature operating modes. The DOE continues to perform materials tests and evaluate corrosion data to provide a stronger technical basis for the projections of waste package and drip shield lifetimes.

Water Diversion Performance of the Engineered Barriers—The water diversion function of the engineered barrier system is to limit the amount of water contacting waste packages for the 10,000-year performance period and to limit the transport of released radionuclides from breached waste packages to the host rock at the drift wall. The engineered barrier system components that will perform these functions principally include the drip shield and the drift invert (consisting of a steel support structure with crushed rock ballast).

Water that enters the emplacement drifts as seepage can flow along three types of pathways: (1) water flow that bypasses the drip shield and moves down drift walls or drips directly to the invert; (2) water droplets that contact the drip shield but are diverted by it to the invert; and (3) water droplets that contact the drip shield and migrate through breaches to the waste package. Water diversion models used for TSPA are used to estimate the fraction of seepage water that penetrates the drip shield and the fraction of resulting leakage that penetrates the waste package.

Models of the movement of water within emplacement drifts focus on the process of seepage. For water to contact a drip shield or waste package, water droplets must form from seepage above the waste package and fall. Other modes of flow, such as film flow on the drift wall, may divert some or all of the seepage around the drift. Most seepage would probably occur at or near faults or fractures that could focus percolation into a drift. Analyses indicate that the emplacement drifts have enough drainage capacity to ensure that water would not rise above the level of the invert even under extreme seepage conditions.

Supplemental studies of the lower-temperature operating mode suggest that flow processes are similar to those modeled for higher-temperature operating modes, except for the period when temperatures are above the boiling point of water. The composition of seepage at early times (for the lower-temperature operating mode) is within the range of aqueous compositions used in laboratory corrosion testing. For this reason, the modeled performance of the engineered barrier system is similar regardless of the choice of thermal operating modes.

Waste Form Degradation and Radionuclide Release—Because of the characteristics of the natural system and the engineered barriers, the DOE does not expect water to penetrate intact waste packages and contact waste for over 10,000 years. Even if water were to penetrate a breached waste package before 10,000 years, several characteristics of the waste form and the other natural and engineered barriers would limit radionuclide releases. First, because of elevated temperatures associated with both higher- and lower-temperature operating modes, much of the water that penetrates the waste package will evaporate before it can dissolve or transport radionuclides. Both spent nuclear fuel and glass high-level radioactive waste forms will degrade slowly in the waste package environment. Further, data and analyses indicate that most of the radionuclides in the waste are not very soluble in the warm, near-neutral pH conditions that are expected. To dissolve radionuclides that may be soluble (technetium-99, iodine-129, neptunium-237, and all isotopes of uranium), water must also penetrate the metal cladding of the

spent nuclear fuel assemblies. Although the performance of the cladding as a barrier may vary because of possible degradation, it is expected to limit contact between water and the waste.

Release of radionuclides from the waste forms is a three-step process requiring (1) degradation of the waste forms, (2) mobilization of the radionuclides from the degraded waste forms, and (3) transport of the radionuclides away from the waste forms. Radionuclides can be released only after the waste package is breached and air and water begin to enter. The rates of water flow and evaporation will determine when and if water accumulates in a waste package. Even if water is available to dissolve radionuclides, the chemistry inside the waste package would influence the rate at which the waste forms degrade and the mobility of radionuclides from the degraded waste forms.

The waste form solubility model is not significantly affected by the temperature of the operating mode because radionuclides can only dissolve after the drip shield and waste package have degraded and water has reached the waste form inside. The subsequent transport processes could occur only after temperatures have cooled below boiling, thereby allowing water to be available as a transport mechanism.

Engineered Barrier System Transport—The invert below the waste package would contain crushed tuff that would limit the transport of radionuclides from breached waste packages into the unsaturated zone. Transport could occur either through advection, which is the flow of liquid water, or by diffusion. The scarcity of water makes advective transport unlikely, but diffusive transport through thin films on the waste form, on the waste package, and in the invert ballast is possible.

Unsaturated Zone Transport—Eventually, components of the repository's engineered barrier system will degrade and small amounts of water will contact the waste. Even after the engineered barriers have degraded, however, features of the geologic setting and underground facility would limit releases to the accessible environment and slow migration for hundreds to thousands of years. Processes that could be important to the movement of radionuclides include sorption, matrix diffusion, dispersion, and dilution.

Sorption is a process in which minerals in the rock or soil along flow paths attract and hold (adsorb) constituents dissolved in water. Rocks in the unsaturated zone at Yucca Mountain contain minerals that can adsorb many types of radionuclides. Matrix diffusion is a process in which dissolved radionuclides diffuse into and out of the rock pores as water flows in the rock fractures. This process would increase both the time it takes for radionuclides to move out of the repository and the likelihood that they would be exposed to sorbing minerals. Dispersion is a process in which radionuclides contained in water spread out as the water flows, resulting in lower concentrations of contaminants. Dilution occurs naturally as contaminated groundwater flows and mixes with noncontaminated groundwater, which reduces the concentration of contaminants.

Process models of transport in the unsaturated zone have incorporated the processes described above. The results of these models indicate that the movement of radionuclides from a breached waste package down to the water table would require hundreds to thousands of years, depending on the mobility of specific radionuclides. Many radionuclides would not move over much longer time spans because of their particular chemical properties.

Saturated Zone Flow and Transport—The same basic processes that apply to the movement of radionuclides in the unsaturated zone (sorption, matrix diffusion, dispersion, and dilution) also apply to transport in the saturated zone. Flowing groundwater transports radionuclides either in solution (dissolved) or in suspension (bound to very small particles called colloids). Any radionuclides released by water contacting breached waste packages would have to migrate through the unsaturated zone down to the water table and then travel through the saturated zone to reach the accessible environment. Groundwater in the saturated zone below the underground facility generally moves southeast before flowing south out of the volcanic rocks and into the thick alluvium deposits of the Amargosa Desert. Figure 5

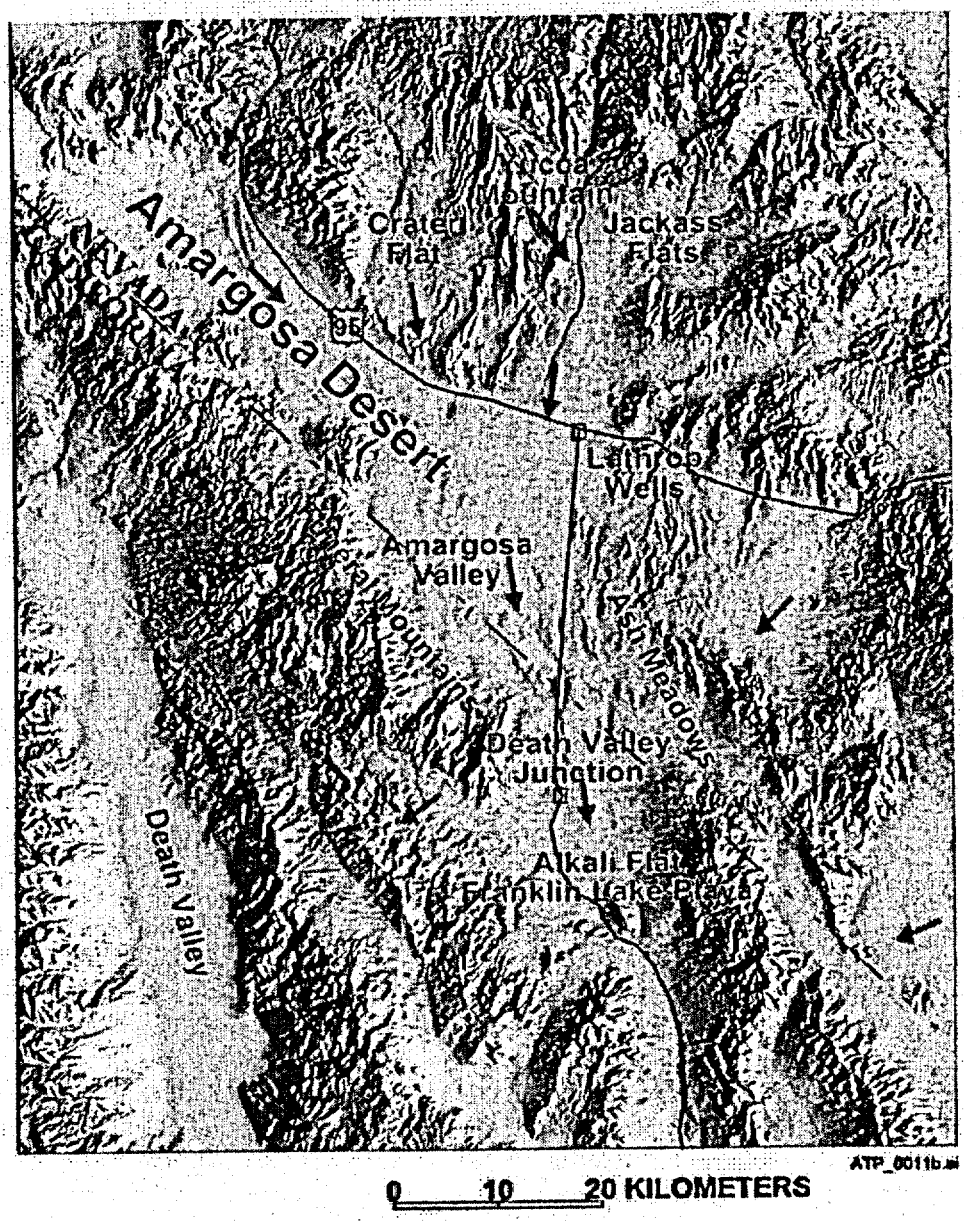


Figure 5. Regional Map of the Saturated Zone Flow System

shows the general directions of groundwater flow in the saturated zone on a regional scale. Analyses show that it would take thousands of years or longer (depending on the mobility of specific radionuclides) for radionuclides to move down through the unsaturated zone, into the saturated zone, and then to the accessible environment.

Biosphere—The biosphere is the ecosystem of the earth and the organisms inhabiting it, including the soil, surface water, air, and all living organisms. Biosphere analyses of the Yucca Mountain site have been performed to develop conversion factors that enable analysts to estimate doses to a receptor from the transport and retention of radionuclides within the biosphere.

The biosphere analyses scrutinize processes and pathways that could either disperse or concentrate radionuclides released from the Yucca Mountain disposal system. In calculating radiation exposure, biosphere analyses consider the environment

around and the lifestyle (including diet and activity) of individuals who would be exposed. The terms "accessible environment" and "reasonably maximally exposed individual" are EPA and NRC regulatory terms that define where radioactive releases from Yucca Mountain must be evaluated and the characteristics of the hypothetical individual who would be exposed to the radiation. All postclosure releases are evaluated within the accessible environment at a point whose latitude is specified and which is above the highest radionuclide concentration in the contaminated groundwater plume. This point lies about 18 km (11 mi) from the southern edge of the currently designed Geologic Repository Operations Area. The reasonably maximally exposed individual who is exposed to releases from the Yucca Mountain disposal system is assumed to have a diet and living style that represents the current residents of Amargosa Valley, Nevada. The radionuclide concentrations that reach this individual are calculated using factors that are unique to the biosphere in which the individual lives.

Disruptive Processes and Events Scenarios—Analyses of disposal system performance must also consider events that could occur in the future that have the potential to compromise the system's ability to protect public health and safety. Analysts have evaluated a wide variety of potentially disruptive processes and events that could affect performance. These range from extremely unlikely events to processes that are likely to occur and that could affect long-term repository performance. The potential for igneous (volcanic) activity in or near the Yucca Mountain site has been specifically included in performance assessments in the disruptive scenario case, and the effects of seismic activity (i.e., vibratory ground motion that might damage the cladding on spent nuclear fuel) have been considered in the nominal scenario case.

A stylized inadvertent human intrusion into the repository was analyzed in a separate performance assessment. NRC licensing regulations at 10 CFR Part 63 would require the DOE to determine the earliest time after disposal when the waste packages would degrade sufficiently that a human intrusion could occur without recognition by the drillers. Analyses indicate that the earliest time after disposal at which a driller would not recognize that a waste package or drip shield has been penetrated would be after about 30,000 years, even if a few waste packages failed (i.e., developed cracks) at earlier times due to improper heat treatment of welds. The DOE also analyzed an inadvertent intrusion occurrence 100 years after closure, based on proposed NRC regulations. This analysis showed that the doses from a 100-year intrusion would be less than about 0.01 mrem/yr.

IX. THE ATTRIBUTES OF SAFE DISPOSAL

The Yucca Mountain disposal system can be described in terms of five key attributes that would be important to long-term performance: (1) limited water entering waste emplacement drifts; (2) long-lived waste package and drip shield; (3) limited release of radionuclides from the engineered barriers; (4) delay and dilution of radionuclide concentrations by the natural barriers; and (5) low mean annual dose considering potentially disruptive events. These attributes are summarized below. The first four reflect the interactions of natural barriers and the engineered barriers in prolonging the containment of radionuclides within the repository and limiting their release. The fifth attribute reflects the likelihood that disruptive events would not affect repository performance over 10,000 years.

Limited Water Entering Emplacement Drifts—The climate at the Yucca Mountain site is dry and arid with precipitation averaging about 190 mm (7.5 in.) per year. Little of this precipitation percolates into the mountain; nearly all of it (above 95 percent) either runs off or is lost to evaporation or transpiration, thereby limiting the amount of water available to seep into the underground facility. For the higher-temperature operating mode described in *Yucca Mountain Science and Engineering Report*, a thermal management strategy was developed that would take advantage of the heat of the emplaced wastes to drive the limited water that exists away from the emplacement drifts. The heat generated by the waste would dry out the rock surrounding the drift and decrease the amount of water available to contact the waste packages until the wastes have cooled substan-

tially. Drainage of water in the rock pillars between drifts would be encouraged by keeping much of the pillar rock between the drifts below the boiling temperature of water. As long as emplacement drift walls remain at temperatures above the boiling point of water, no liquid water can remain in the underground facility, and very little can remain in the underground structure. For the lower-temperature operating modes (and for higher-temperature operating modes after the waste packages have cooled below the boiling point of water), the heat associated with waste will still cause evaporation in and near the emplacement drifts. This process would limit the amount of water in the rock near the waste packages. In lower-temperature operating modes, the waste packages would be exposed to water sooner. Because the rock would eventually cool in any operating mode, there does not appear to be a significant difference in the amount of water to which the waste packages would eventually be exposed.

The repository design also takes advantage of the mechanical and hydrologic processes that divert water around emplacement drift openings in the unsaturated zone. Because of capillary forces, water flowing in narrow fractures tends to remain in the fractures rather than flow into large openings, such as drifts. If any water reached an emplacement drift, it could flow down the drift wall to the floor and drain without contacting the drip shields or waste packages. Thus, the natural and engineered features of a Yucca Mountain disposal system will combine to limit the potential for water to enter the emplacement drifts.

Long-Lived Waste Package and Drip Shield—To further reduce the possibility of water contacting waste, the DOE has designed a robust, dual-wall waste package with an outer cylinder of corrosion-resistant nickel-based metal, Alloy 22. Alloy 22 was selected because it will remain stable in the geochemical environment expected in the potential repository. In the higher-temperature operating mode, the repository environment would be warm, with temperatures at the waste package surface initially rising above the boiling point of water. Waste package surface temperatures are expected to gradually decrease to below boiling after a period of hundreds to thousands of years, depending on the waste package's location in the repository and other factors discussed previously. In lower-temperature operating modes, the waste packages would be exposed to water earlier.

Chemically, the environment is expected to be at near-neutral pH (mildly acidic to mildly alkaline) and mildly oxidizing. Because most corrosion would occur only in the presence of water, and because highly corrosive chemical conditions are not expected, both the titanium drip shield and the Alloy 22 outer barrier of the waste package are expected to have long lifetimes.

Limited Release of Radionuclides from the Engineered Barrier System—Because of the characteristics of the natural system, the drip shields, and the waste packages, the DOE does not expect water to come into contact with the waste forms for more than 10,000 years. Even if water were to penetrate a breached waste package before 10,000 years, several characteristics of the waste form and the repository would limit radionuclide releases. First, because of the warm temperatures, much of the water that penetrates the waste package would evaporate before it could dissolve or transport radionuclides. Neither spent nuclear fuel nor glass waste forms will dissolve rapidly in the expected repository environment. Although the performance of the cladding as a barrier may vary because of possible degradation, it is expected to limit contact between water and the waste. The component of the engineered barrier system below the waste package, called the invert, contains crushed tuff that would also limit the transport of radionuclides into the host rock, as discussed under Engineered Barrier System Transport.

Delay and Dilution of Radionuclide Concentrations by the Natural Barriers—Eventually, components of the engineered barrier system will degrade, and small amounts of water will contact waste. Analyses indicate that this is not likely to occur within the first 10,000 years following repository closure. Even then, features of the geologic environment and the repository would limit radionuclide migration to the accessible environment and slow it by hundreds to thousands of years. Processes that could be important to the movement of radionuclides include sorption, matrix diffusion,

dispersion, and dilution. Rock units in both the unsaturated zone and the saturated zone contain minerals that can adsorb many types of radionuclides (i.e., radionuclides would attach to and collect on the mineral surfaces). As water flows through fractures, dissolved radionuclides can diffuse into and out of the pores of the rock matrix, increasing both the time it takes for radionuclides to move through the geologic setting and the likelihood that radionuclides will be exposed to sorbing minerals. Dispersion and dilution will occur naturally as potentially contaminated groundwater flows and mixes with other groundwater and reduces the concentration of contaminants.

Low Mean Annual Dose Considering Potentially Disruptive Events—Yucca Mountain provides an environment in which hydrologic and geologic conditions important to waste isolation (e.g., a thick unsaturated zone with low rates of water movement) have changed little for millions of years. Analysts have identified and evaluated a wide variety of potentially disruptive processes and events that could affect the performance of the Yucca Mountain disposal system. These range from extremely unlikely events, such as meteor impacts, to events that are likely to occur, such as regional climate change. Although the probability of volcanic activity in or near the Yucca Mountain site is low, volcanic activity was a consideration in TSPA in the disruptive scenario case. Performance assessment results to date show that potentially disruptive events are not likely to compromise the system performance.

X. UNCERTAINTIES IN DATA AND MODELS

Quantitative assessments of the long-term performance of the disposal system consider a comprehensive set of features, events, and processes that may have an effect on that performance. The features and characteristics of the site and geologic setting are incorporated into conceptual and numerical models. The likelihood of occurrence and consequences of processes and events that may affect repository performance are evaluated, then incorporated, as appropriate, into the numerical models. Although the DOE continues to evaluate ways to reduce uncertainties in repository performance models, uncertainties will always remain because of the long time frames over which the system performance must be assessed, the natural variability in features and processes at the site, and limitations on the amount of data that can be collected. Features, events, and processes are generally represented probabilistically in a performance assessment to address this inherent uncertainty and variability.

Numerous analyses have been performed to help the DOE understand the extent to which the results of total system performance assessments are robust (not likely to change significantly as new information is gathered in the future). These include analyses to assess the degree of realism in current process models, to quantify key uncertainties, and to improve the understanding of conservatism in the models and in performance assessment results. The DOE has also evaluated whether uncertainties (especially modeling uncertainties that are not easily quantified) can be reduced further by operating the repository at lower temperatures.

Because uncertainty cannot be eliminated, the DOE's approach to building confidence in analyses of repository performance relies on multiple lines of evidence. Collectively, these multiple lines of evidence are known as the postclosure safety case. Elements of the safety case include:

- Quantitative assessments of long-term performance (i.e., TSPA).
- Selection of a site and design of a repository that provides defense in depth—a system of multiple, independent, and redundant barriers designed to ensure that failure of one barrier does not result in failure of the entire system. The system would also provide a margin of safety against radionuclide releases and a margin of safety compared to applicable radiation protection standards.
- Qualitative insights gained from the study of natural and man-made analogues to the repository or to processes that may affect system performance.

- Long-term management to ensure the integrity and security of the repository and long-term monitoring (a performance confirmation program) to evaluate the behavior of the repository and enable sound scientific and engineering bases for a later repository closure decision.

Quantitative analyses indicate that the repository design and operating modes described in *Yucca Mountain Science and Engineering Report* offer both defense in depth and a significant safety margin. Analogue studies have provided several lines of evidence that suggest most current models are representative and, in some cases, may be overly conservative.

XI. PERFORMANCE ASSESSMENT RESULTS

The numerical (quantitative) evaluation of postclosure repository performance is an important part of demonstrating that a Yucca Mountain disposal system can be constructed that would protect public health and safety. The DOE has completed a TSPA to evaluate the system performance. TSPA is a numerical calculation, based on the process models described above, that forecasts when and to what extent radionuclides released from a Yucca Mountain disposal system might reach the accessible environment and expose human beings. The results are typically displayed as a graph of dose rate plotted in millirem per year against time (10,000 years).

The total system analyses are presented in three scenarios: nominal (the scenario that uses the features, events, and processes expected to occur), disruptive (the scenario that uses possible but unlikely events with potentially harmful consequences), and stylized human intrusion (the scenario proposed by the NRC in which someone drills through a waste package 100 years after the repository had been permanently closed). As noted earlier, the DOE does not expect a human intrusion during the compliance period. The analysis is included, however, to illustrate the disposal system's resilience. Analyses were completed to evaluate the potential importance of each feature, event, or process at the site to system performance and included or excluded each from TSPA analyses, as appropriate.

Performance assessments for the initial 10,000-year period after closure have been completed in a manner consistent with the EPA radiation protection standards and NRC licensing regulations. In addition to these assessments, the peak mean annual dose to the reasonably maximally exposed individual beyond the 10,000-year time period was also calculated to see if dramatic changes in the performance of the disposal system could be anticipated beyond 10,000 years. These results are described in Section 4.4.

Nominal Scenario—The DOE has assessed the repository's performance for the nominal case. The TSPA-SR model projected no dose over a 10,000-year period. A supplemental TSPA model projected a dose to the reasonably maximally exposed individual of 2×10^{-4} mrem/yr, while a revised supplemental TSPA model projected a dose of 1.7×10^{-5} mrem/yr over a 10,000-year period. Both doses were calculated based on the higher-temperature operating mode. The doses calculated by the supplemental and revised supplemental TSPA models would be reduced by approximately one-third if an annual water demand of 3,000 acre-ft was used, consistent with final NRC regulations (10 CFR 63.312).

Disruptive Scenario—The primary disruptive event considered in these analyses is igneous activity. The TSPA models evaluated two igneous disruptions: a volcanic eruption that intersects drifts and brings waste to the surface; and an igneous disruption that damages waste packages and exposes radionuclides for groundwater transport. The probability of igneous disruption is extremely low (the mean annual probability is about one chance in 60 million per year of occurring). The TSPA-SR model projected a probability-weighted mean dose from igneous activity of 0.08 mrem/yr over a 10,000-year period, and both the supplemental TSPA model and the revised supplemental TSPA model projected a dose to the receptor of 0.1 mrem/yr over 10,000 years.

Human Intrusion Scenario—The DOE has assessed the consequences of a stylized human

intrusion into the repository at 100 years after closure, pursuant to the proposed NRC regulation. The results of this analysis show that the peak mean dose is approximately 0.01 mrem/yr over a 10,000-year period. Consistent with final NRC regulations at 10 CFR 63.321, the DOE analyzed the period of time that would be necessary for waste packages to degrade sufficiently that a human intrusion could occur without recognition by a driller. The DOE subsequently found that human intrusion would not occur for more than about 30,000 years. Therefore, no doses related to human intrusion would occur within 10,000 years. The dose from a human intrusion at 30,000 years is analyzed and presented in the final environmental impact statement (EIS).

XII. EVALUATION OF THERMAL OPERATING MODES

The performance assessment presented in *Yucca Mountain Science and Engineering Report* considers both lower- and higher-temperature operating modes. The repository design is flexible: it can be operated in a range of modes that would allow the temperature and humidity in the underground environment to be varied. Analyses of the effects of the higher-temperature operating mode indicate that higher temperatures would effectively limit the potential for contact between water and waste packages for time periods of hundreds to thousands of years. However, during that period, the interaction of rock, water, and heat in and near emplacement drifts may affect rock properties and water chemistry in complex ways that cannot be fully captured in the models.

The complexity of these processes introduces uncertainty into the analyses. For this reason, the DOE has completed extensive analyses to determine whether the complexity of the associated process models could be reduced. As part of these investigations, the DOE has analyzed the performance of the repository over a range of thermal operating modes. The results indicate that the repository's performance is similar over range of operating modes, encompassing above- and below-boiling conditions.

The DOE expects the repository design and operating mode to be refined as the project evolves. This design evolution will be based on a process that includes (1) refining specific design requirements and performance goals to recognize performance-related benefits that could be realized through design and (2) enhancing components of the design to best achieve the performance-related benefits.

XIII. PRECLOSURE SAFETY ASSESSMENT

A potential repository at Yucca Mountain would be designed and operated in a manner that would limit worker and public exposures to radiation. A preclosure safety evaluation has been conducted to evaluate the performance of the Geologic Repository Operations Area during the preclosure period. The safety evaluation plays a key role in identifying design features and controls that are important to safety, and is a primary input to the quality assurance classification process. Results indicate that a repository could be operated such that radiation doses to the public and workers would be below EPA and NRC regulatory limits.

To begin the safety evaluation, the DOE systematically identified and examined a range of potential hazards and the event sequences such hazards could cause, as well as their likelihood and consequences. From this examination, the DOE identified structures, systems, and components important to safety that would be relied upon to protect the public and workers. These structures, systems, and components are those engineered features of the repository whose function is to (1) reasonably ensure that spent nuclear fuel and high-level radioactive waste can be received, handled, staged, emplaced, and retrieved without exceeding regulatory limits; or (2) prevent or reduce the impact of Category 1 and Category 2 event sequences (events the repository would be designed to withstand) that could potentially lead to exposure of individuals to radiation. The structures, systems, and components were then classified in grades according to their importance to safety to ensure that appropriate quality assurance controls are implemented during the repository's operational lifetime. The DOE has developed a preclosure safety test and evaluation

program to verify that structures, systems, and components are designed as specified and perform as required.

XIV. PERFORMANCE CONFIRMATION AND MONITORING

A performance confirmation program established to monitor and confirm that the Yucca Mountain disposal system is performing as expected was initiated during site characterization and will continue to permanent closure. The focus of the performance confirmation program is to gather and analyze data on natural processes and engineered barriers performance that will affect repository performance after closure and to evaluate their impacts. Subsurface facilities, including performance confirmation drifts and alcoves, would be constructed to facilitate monitoring of the underground facility and the performance of the engineered barrier system. Primary testing and monitoring activities will include seepage monitoring to evaluate flow of water into excavations and confirm expected waste package environment; in situ waste package surface temperature monitoring to infer cladding temperatures; rock mass monitoring to confirm the conceptual understandings and numerical simulations of coupled processes considered in performance assessments; and other activities that focus on providing an increased understanding of processes important to repository postclosure safety. Performance confirmation and monitoring activities would continue throughout the preclosure period, which could be extended up to 300 years.

INTENTIONALLY LEFT BLANK

ACRONYMS AND ABBREVIATIONS

Acronyms

AEC	U.S. Atomic Energy Commission
ALARA	as low as is reasonably achievable
CQ	conventional quality
DOE	U.S. Department of Energy
ECRB	Enhanced Characterization of the Repository Block
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
ERDA	U.S. Energy Research and Development Administration
FEP	feature, event, and process
NRC	U.S. Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act of 1982
NWTRB	Nuclear Waste Technical Review Board
QL	quality level
TSPA	total system performance assessment
TSPA–SR	total system performance assessment for site recommendation
USGS	U.S. Geological Survey
VA	Viability Assessment

Abbreviations

atm	atmosphere
BHN	Brinnel hardness number
cm	centimeter
Eh	redox potential
ft	foot
g	gram
gal	gallon
GWd	gigawatt day
hp	horsepower
Hz	Hertz

in.	inch
kg	kilogram
km	kilometer
kPa	kilopascal
kW	kilowatt
MTHM	metric tons of heavy metal
MTU	metric tons of uranium
L	liter
lb	pound
m	meter
M	molar
mg	milligram
mi	mile
min	minute
mL	milliliter
mm	millimeter
MPa	megapascal
mV	millivolt
MW	megawatt
M_w	moment magnitude
µm	micrometer (micron)
nm	nanometer
Pa	pascal
pCi	picocurie
pH	scale of acidity and alkalinity (hydrogen-ion concentration notation)
ppm	parts per million
psi	pounds per square inch
rem	roentgen equivalent man
s	second
W	watt
wt	weight
yr	year

1. INTRODUCTION

Spent nuclear fuel and high-level radioactive wastes are the result of commercial power generation, nuclear weapons production, and other research and development activities. They have accumulated since the mid-1940s at sites now managed by the U.S. Department of Energy (DOE) and since 1957 at commercial reactors and storage facilities across the country. The responsible management and disposal of these materials is a critical part of the DOE mission to dispose of high-level radioactive waste and spent nuclear fuel from federal facilities, including the nuclear weapons program, as well as commercially generated spent nuclear fuel.

The U.S. has evaluated methods for the safe storage and disposal of radioactive waste for more than 40 years. Many organizations and government agencies have participated in these studies: at the request of the U.S. Atomic Energy Commission (AEC), the National Academy of Sciences evaluated options for the disposal of radioactive waste on land in the 1950s (National Academy of Sciences Committee on Waste Disposal 1957). The AEC and its successor agencies, the U.S. Energy Research and Development Administration (ERDA) and the DOE, continued to analyze nuclear waste management options throughout the 1960s and 1970s. In cooperation with the U.S. Geological Survey (USGS), nationwide surveys were performed to identify potentially acceptable locations for disposal sites. During the late 1970s and early 1980s, the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Environmental Protection Agency (EPA) performed numerous studies of disposal options and safety and began developing regulatory standards. An Interagency Review Group that included representatives of 14 federal government entities provided findings and recommendations on nuclear waste management to President Carter in 1979 (Interagency Review Group on Nuclear Waste Management 1979). After analyzing a wide variety of options, disposal in mined geologic repositories emerged as the preferred long-term environmental solution for the management of spent nuclear fuel and high-level radioactive waste. This consensus is reflected in the Nuclear Waste Policy Act of 1982 (NWPA) (42 U.S.C. 10101 et seq.), which established U.S. policy when it was enacted by Congress in 1982 (Public Law No. 97-425) and amended in 1987 (Public Law No. 100-203).

Congressional Findings

Section 111(a) of the NWPA (42 U.S.C. 10131(a)) contains seven Congressional findings for the repository program:

(1) Radioactive waste creates potential risks and requires safe and environmentally acceptable methods of disposal;
(2) A national problem has been created by the accumulation of (A) spent nuclear fuel ...; and (B) radioactive waste ...
(3) Federal efforts during the past 30 years to devise a permanent solution ... have not been adequate;
(4) While the Federal government has the responsibility to provide for ... permanent disposal ... , the costs ... should be the responsibility of the generators and owners...
(5) The generators and owners of high-level radioactive waste and spent nuclear fuel ... have the primary responsibility to provide for, and ... pay the costs of, the interim storage of such waste and spent fuel until ... accepted by the Secretary of Energy ...
(6) State and public participation ... is essential in order to promote public confidence in the safety of disposal ...
(7) High-level radioactive waste and spent nuclear fuel have become major subjects of public concern, and appropriate precautions must be taken to ensure that [they] do not adversely affect public health and safety or the environment for this or future generations.

In the NWPA, Congress described the technical and policy issues that make nuclear waste disposal so difficult. Congress assigned the responsibility for paying the costs of disposal to the generators and owners of high-level radioactive waste and spent nuclear fuel and recognized that a solution could only be achieved through a process that included public and state involvement. The national policy reflected in the NWPA is that geologic disposal should be accomplished by the generation that created the waste, not deferred to future generations.

The NWPA required the EPA to establish standards for the protection of public health and safety; directed the DOE to site, design, construct, operate, and close a repository; and assigned to the NRC the responsibility to regulate and license a repository and ensure compliance with applicable standards. The Nuclear Waste Policy Amendments Act of 1987 (Public Law No. 100-203) directed the DOE to characterize only Yucca Mountain, Nevada, for a repository. It also created the Nuclear Waste Technical Review Board (NWTRB), an independent organization within the executive branch, and charged it with evaluating the technical and scientific validity of the program and reporting its findings, conclusions, and recommendations to Congress and the Secretary not less than twice a year.

The DOE (including the national laboratories) and the USGS began studying Yucca Mountain (Figure 1-1) in the late 1970s. Detailed investigations of the geology, hydrology, geochemistry, and other characteristics of the site have been performed since 1986 to determine whether it is a suitable place to build a geologic repository. As part of this effort, the DOE has developed preliminary designs for surface and subsurface facilities and for waste packages. Analyses that integrate design- and site-specific data and models have been conducted to assess how a repository at Yucca Mountain might perform.

The NWPA specifies a process for the recommendation and approval of a site for development of a repository, and it requires that the Secretary of Energy provide a comprehensive statement of the

ATP_Z161_Fig. 1-1a.ai

Figure 1-1. Aerial View of Yucca Mountain, Looking South, Showing the Desert Environment and the Remote Location

The potential repository would be located about 300 m (1,000 ft) below the eastern slope of the mountain.

basis for a recommendation if the site is recommended. This report is being issued to describe the results of site characterization studies, the waste forms to be disposed, a repository and waste package design, and the updated results of assessments of long-term performance of the potential repository.

Studies of Yucca Mountain will not end if the site is recommended. NRC regulations specify that testing and monitoring of Yucca Mountain would continue during licensing and, if the site is licensed, throughout construction, operation, and after permanent closure of the repository.

1.1 PURPOSE AND SCOPE

If the DOE decides to recommend the development of the Yucca Mountain site to the President, it is required by the NWPA to provide a comprehensive statement of the basis for the recommendation. This report is one part of a comprehensive suite of analyses and documents that the DOE is considering with respect to the possible recommendation. It describes the results of scientific and technical studies that have been performed to determine whether the Yucca Mountain site should be recommended for development as a geologic repository. It contains information that would be included in any site recommendation from the Secretary to the President, consistent with Sections 114(a)(1)(A), (B), and (C) of the NWPA (42 U.S.C. 10134(a)(1)(A), (B), and (C)). Section 1 introduces the report and provides background information on both the Yucca Mountain site and the site recommendation decision process set forth in the NWPA. Section 2 describes a preliminary design for a potential repository at Yucca Mountain, and Section 3 describes the proposed waste package designs and the waste forms to be disposed. Section 4 discusses data related to the safety of the Yucca Mountain site and describes how natural and engineered repository systems would work together to protect public health and limit the release of radionuclides to the environment. It explains the relationship between the waste form, the waste package, and the geologic medium at Yucca Mountain. It also describes analyses of the future long-term performance of a repository at Yucca Mountain. Section 5 describes analyses that evaluate the safety of the potential repository during operation and before final closure.

The DOE has actively sought public involvement and participation in the consideration of the Yucca Mountain site. In May 2001, the DOE released an initial version of this report (DOE 2001a) to facilitate public comments on the consideration of the possible recommendation of Yucca Mountain. It was based on the technical information, and the draft regulations, available at the time of its preparation. Comments regarding the possible recommendation of the site were solicited during initial and supplemental public comment periods announced in the Federal Register, and at numerous public hearings conducted throughout the State of Nevada. Since the beginning of the public comment period, the EPA, NRC, and DOE have also released final versions of regulations relevant to Yucca Mountain. Therefore, this report has been updated to:

1. Identify and incorporate analyses consistent with final regulations, as necessary
2. Reflect public comments received by the DOE
3. Identify and consider the results of supplemental scientific and engineering analyses that have been completed by the DOE and its contractor since May 2001.

This *Yucca Mountain Science and Engineering Report* is based on an extensive foundation of scientific, engineering, regulatory, and programmatic research that analyzed the Yucca Mountain site, the potential repository and waste package designs, and other information. This report is supported by a comprehensive set of scientific and technical integrating documents that explain in detail the basis for the results presented. The major integrating documents include:

- *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a). This document describes in detail the results of the quantitative analysis of the long-term performance of a potential repository at Yucca Mountain.

- *Yucca Mountain Site Description* (CRWMS M&O 2000b). This document contains a thorough description of the natural system at Yucca Mountain, including the geology and hydrology, on both a regional and a site-specific scale.

- *Monitored Geologic Repository Project Description Document* (Curry 2001). This report presents a summary of the design and operational concepts for the potential Yucca Mountain repository, including facilities at the surface for receiving, handling, and packaging waste; facilities underground for waste disposal; and engineered barriers that would contribute to the ability of the repository system to safely isolate waste.

Key technical references include nine process model reports that describe how the natural and engineered systems at a potential repository would perform. In addition, hundreds of reports summarizing scientific and engineering studies on a wide variety of topics have been prepared, including:

- Analysis and model reports describing the detailed scientific models of how the potential repository might behave in the future

- Scientific reports of the results of site characterization investigations, regional studies, and studies of natural analogues that are used to help understand the behavior of a potential repository

- Descriptions of performance objectives, design requirements, design analyses (over a range of thermal operating modes), and design alternative studies for the repository surface and subsurface facilities, engineered barrier systems, and waste packages.

Several supplemental analyses relevant to the site recommendation have been released for public comment since May 2001. These supplemental analyses include:

- *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b). Volume 1 of this report (BSC 2001a) describes (1) supplemental analyses performed to quantify uncertainties, (2) updated process models based on recent scientific information, and (3) analyses of the effects of a lower-temperature repository operating mode on process model results. Volume 2 of this report (BSC 2001b) documents revised performance analyses based on a supplemental TSPA model, incorporating information from Volume 1.

- *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation* (Williams 2001a). This report documents revised TSPA results based on a supplemental TSPA model revised to conform to final 40 CFR Part 197.

- *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b). This report documents additional TSPA sensitivity analyses to address final 10 CFR Part 63 (66 FR 55732) provisions. These analyses are also discussed in the site suitability evaluation.

- *Technical Update Impact Letter Report* (BSC 2001c). This report reviews additional technical information collected since the models and analyses described in this report and *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b) were completed. The report also includes an assessment of the impact of the new information on the TSPA analyses.

1.2 BACKGROUND INFORMATION

This section briefly describes the materials planned for disposal at Yucca Mountain. It also summarizes the evolution of and the rationale for the U.S. nuclear waste disposal program.

All of the waste forms that would be transported to and received at the potential repository would be solid materials that would be stable in a deep, dry

geologic repository. These waste forms fall into two categories: spent nuclear fuel and high-level radioactive waste. Some elements of this waste decay in only a few years; others are radioactive for hundreds of thousands of years (see Section 4.4.1.4). This waste must be safely contained and isolated to protect human health and the environment.

1.2.1 Sources of Materials Considered for Disposal

Commercial nuclear power plants, which supply about 20 percent of the nation's electricity, produce spent nuclear fuel. The DOE manages several facilities that manufactured materials for nuclear weapons and, in doing so, produced spent nuclear fuel and high-level radioactive waste. The DOE also operates research reactors that produce spent nuclear fuel and has operated reprocessing facilities that produce high-level radioactive waste. Most of the waste from reprocessing was generated before 1982. Small quantities of high-level radioactive waste are produced in medical research that uses nuclear materials. In addition, the DOE owns small quantities of spent nuclear fuel from research reactors in foreign countries. This spent nuclear fuel will be returned to the U.S. for disposal to support nonproliferation goals.

The total inventory of spent nuclear fuel and high-level radioactive waste in the U.S. could eventually exceed 100,000 MTHM or MTU, depending on the number of reactors that receive operating license extensions. MTHM and MTU are approximately equivalent terms for the once-through nuclear fuel cycle and can be used interchangeably for most of the material to be emplaced in the repository. The first repository is limited by the NWPA to 70,000 MTHM until a second repository is in operation. Section 3 of this report and Appendix A of the draft environmental impact statement (EIS) (DOE 1999a) describe the materials planned for disposal in a repository. The repository design summarized in Section 2 and analyzed in Section 4 is based on an inventory allocation that includes 63,000 MTHM of commercial spent nuclear fuel; 2,333 MTHM of DOE spent nuclear fuel; and 4,667 MTHM of DOE high-level radioactive waste.

1.2.1.1 Commercial Spent Nuclear Fuel

As of December 1999, the U.S. had accumulated about 40,000 MTHM of spent nuclear fuel from commercial nuclear power plants (CRWMS M&O 1999a, Tables B-11 and B-12) This amount could more than double by 2035 if all currently operating plants complete their initial 40-year license period. Commercial spent nuclear fuel makes up most of the spent nuclear fuel that requires disposal. Spent nuclear fuel contains uranium-235 and uranium-238, relatively short-lived fission products such as strontium-90 and cesium-137, and relatively long-lived transuranic isotopes (i.e., isotopes with atomic numbers greater than 92) such as plutonium-239 and americium-243. Commercial spent nuclear fuel is now stored at 72 commercial nuclear sites in 33 states, where it will remain until a permanent repository, or an NRC-licensed storage facility, is in operation. These sites can only be decommissioned and used for other purposes after the spent nuclear fuel and other radioactive materials have been removed.

1.2.1.2 U.S. Department of Energy Spent Nuclear Fuel

By 2035, the U.S. will have accumulated about 2,500 MTHM of spent nuclear fuel from reactors that produced materials for the nation's nuclear weapons program, from research reactors, from reactors on nuclear-powered naval vessels, and from reactor prototypes. These materials include commercial spent nuclear fuel that has been transferred to the DOE for research, project demonstration, or other purposes. An example is the core debris from the Three Mile Island-2 reactor. All of this spent nuclear fuel is the responsibility of the DOE. The majority is currently stored at sites in Idaho, South Carolina, and Washington (DOE 1999a, Appendix A, Tables A-16 and A-17). Under a negotiated agreement that involved the State of Idaho, the U.S. Navy, and the DOE, the DOE must remove all spent nuclear fuel from Idaho by the year 2035 (*Public Service Co. of Colorado v. Batt*, settlement agreement).

In addition to the sites already mentioned, about 55 other facilities have small quantities of spent nuclear fuel and high-level radioactive waste.

These sites include research reactors at universities, commercial research reactors, DOE laboratories, and commercial fuel fabrication plants. In most cases, DOE spent nuclear fuel and/or DOE waste stored at these facilities will be shipped to one of the larger DOE sites for drying, packaging or other treatment, and storage prior to transport to the repository for disposal.

1.2.1.3 High-Level Radioactive Waste

Large volumes of high-level radioactive waste were created in the past when spent nuclear fuel was treated chemically (i.e., reprocessed) to separate uranium or plutonium isotopes that could be reused from the other elements in the fuel. The high-level radioactive waste left over from this process exists in both liquid and solid form; liquid wastes are stored in underground tanks at DOE sites near Hanford, Washington; Savannah River, South Carolina; Idaho Falls, Idaho; and West Valley, New York (DOE 1997a, p. 34). Liquid high-level radioactive waste will be vitrified (turned into glass) prior to shipment for disposal. In this process, the waste materials are mixed with other components of glass, melted, and poured into stainless steel canisters. The vitrified waste, typically in a form known as borosilicate glass, is leach-resistant and long-lived. Where there are agreements between the DOE and the states where the waste is stored, this high-level radioactive waste will be solidified and placed in about 22,000 canisters for future disposal in any permanent geologic repository for the disposal of high-level radioactive waste (DOE 1997b, Section 1.5.4). No liquid wastes will be received at or disposed in a geologic repository.

1.2.1.4 Surplus Plutonium

The end of the Cold War has reduced the need for nuclear materials for weapons, which has resulted in the closure and cleanup of several weapons plants and the identification of a nominal 50 metric tons of surplus plutonium. The surplus plutonium associated with weapons production are no longer needed and must be safely disposed. Current plans call for some of the surplus plutonium, up to 33 metric tons, to be combined with uranium-238 to form a mixed-oxide fuel that would be used in commercial reactors, with later disposal of the commercial spent nuclear fuel at the potential Yucca Mountain repository. Up to approximately 17 metric tons would be immobilized in ceramic, placed inside stainless steel cans, and placed in canisters. These canisters will be filled with molten high-level radioactive waste glass, which will vitrify into a glass waste form around the stainless steel cans, as described above.

1.2.1.5 Present Location of Spent Nuclear Fuel and High-Level Radioactive Waste

Spent nuclear fuel and high-level radioactive waste are presently stored in 39 states, as shown in Figure 1-2. During repository operations, waste would be consolidated for transport to the repository from 77 storage sites (72 commercial and 5 DOE [DOE 1999a, Section 1]). These storage sites are located in a mixture of urban, suburban, and rural environments.

1.2.2 U.S. Policy: The Rationale for Geologic Disposal

U.S. policy on nuclear waste management has been developed by Congress through legislation and implemented by a succession of agencies, beginning immediately after World War II. The Atomic Energy Act of 1954 (42 U.S.C. 2011 et seq.) placed management of all aspects of the nation's nuclear programs under the jurisdiction of the AEC. Although the AEC initially focused on military applications, Congress soon realized it needed a broader mandate. Congress assigned the AEC major roles, which included continuing the nuclear weapons program, promoting the private use of atomic energy for peaceful applications, and protecting public health and safety from the hazards of nuclear power (Walker 2000).

The AEC recognized that the nation needed a long-term strategy for managing and disposing of radioactive waste. On February 28, 1955, it reached an agreement with the National Academy of Sciences/National Research Council to establish a committee of leading scientists to evaluate methods of disposal and report their findings. The committee was also asked to recommend areas

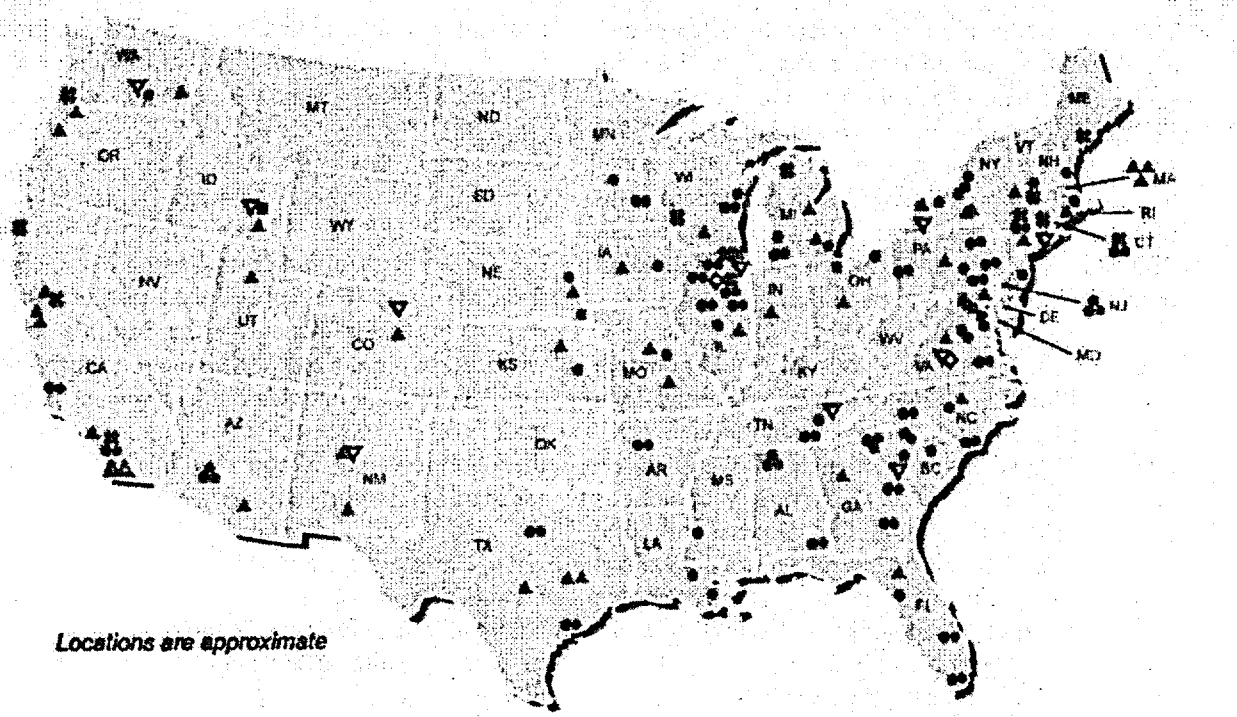


Figure 1-2. Map Showing Locations of Spent Nuclear Fuel and High-Level Radioactive Waste Destined for Geologic Disposal

Commercial waste may include West Valley vitrified waste if an agreement is reached between the DOE and the state of New York for the disposal of this waste. Source: Modified from DOE 1999a, Figure 1-1.

where research was needed (National Academy of Sciences Committee on Waste Disposal 1957, p. 8).

The hazards posed by radioactive waste decline over time because of radioactive decay. Some radionuclides decay quickly and do not present a significant risk after a few decades, whereas others remain radioactive for thousands of years. Therefore, early studies of disposal options sought the most effective way to isolate waste long enough for the hazard to decline to levels that do not pose a significant risk to public health or the environment. The search led to geologic environments that have remained stable and are likely to remain so in the future.

The committee evaluated a wide range of options and reached several conclusions and recommendations (National Academy of Sciences Committee on Waste Disposal 1957). One important conclusion was that a safe disposal facility for radioactive waste could be constructed at many sites in the U.S. However, the committee warned that many areas do not contain any likely disposal sites. The committee specifically noted that favorable conditions for disposal were not found along the Atlantic seaboard.

The committee examined the best means for disposing of the liquid high-level radioactive waste that was already accumulating at sites in Hanford, Washington, and Savannah River, South Carolina.

The scientists explored a variety of options and concluded that storage in tanks was, at the time, the safest short-term method (National Academy of Sciences Committee on Waste Disposal 1957, p. 6). However, tank storage of large volumes of liquid waste was not viewed as a permanent solution; deep disposal in cavities mined in salt deposits was suggested as the most promising possibility for a long-term solution.

The committee noted that disposal would be much simpler if liquid waste could be converted to a solid form of "relatively insoluble character" (National Academy of Sciences Committee on Waste Disposal 1957, p. 1). A variety of options were considered feasible for disposal of solid wastes, including salt or dry mines or excavations in other geologic environments. The committee cautioned that the deep geologic disposal of waste would be a "special problem for each particular installation" (National Academy of Sciences Committee on Waste Disposal 1957, p. 77), and that "the education of a considerable number of geologists and hydrologists ... is going to be necessary" (National Academy of Sciences Committee on Waste Disposal 1957, p. 7).

Between 1957 and the early 1970s, the need to find a safe method for waste disposal grew more urgent as the commercial nuclear utility industry expanded dramatically. However, little progress was made on a long-term solution. In 1970, the AEC tentatively selected a site for a repository in salt deposits near Lyons, Kansas. Under considerable political and technical fire, the Lyons site was abandoned two years later because of concern that nearby salt mine drilling had compromised the integrity of the geologic formation (The League of Women Voters 1993, p. 49).

As the commercial nuclear utility industry continued to grow in the 1960s, the AEC was increasingly criticized because of the conflict between its roles of promoting peaceful uses of atomic energy and regulating the industry to protect public health and safety. Congress passed the Energy Reorganization Act in 1974 (42 U.S.C. 5801 et seq.), which divided the AEC into the ERDA and NRC. The NRC was assigned the regulatory role, with the responsibility to protect public health and safety, both for reactor operations and for the management and disposal of spent nuclear fuel and high-level radioactive waste.

The energy crisis of the 1970s prompted Congress to pass the Department of Energy Organization Act in 1977 (42 U.S.C. 7101 et seq.), which combined the ERDA with other energy-related agencies to form the DOE. The act assigned to the DOE the responsibility for managing the nation's nuclear weapons programs and for managing the program for disposing the nation's nuclear waste. Since the middle of the 1980s, the environmental cleanup of the nuclear weapons complex has been a major focus of DOE efforts nationwide.

After Congress created the DOE, waste management activities accelerated. The DOE initiated the National Waste Terminal Storage Program in 1977 to identify suitable sites and develop the technology to license, construct, operate, and close a repository. Site screening focused on areas with salt deposits and federal lands where radioactive materials were already present, specifically Hanford, Washington, and the Nevada Test Site. The Carter administration initiated an Interagency Review Group in 1978 to review national nuclear waste policy. The group recommended that the U.S. proceed with geologic disposal and proposed that the DOE study geologic settings other than salt deposits. The DOE decided to proceed with a strategy that relied on mined geologic disposal in 1980, as documented in *Final Environmental Impact Statement Management of Commercially Generated Radioactive Waste* (DOE 1980). The report evaluated many options, including disposal in deep boreholes, ice sheets, subseabeds, and space. Geologic disposal in different host rocks was also evaluated. The DOE's decision to move forward with geologic disposal was consistent with the original recommendation of the committee formed by the National Academy of Sciences/ National Research Council and was later affirmed by numerous other evaluations, both in the U.S. and abroad. In a 1984 rulemaking known as the Waste Confidence Decision (49 FR 34658), the NRC found that mined geologic disposal was feasible and that spent nuclear fuel and high-level radioactive waste could be safely managed in a geologic repository. The NRC revisited the deci-

sion in 1990 (55 FR 38474) and again in 1999 (64 FR 68005), with similar conclusions.

Congress enacted the NWPA in 1982 (it became law on January 7, 1983), which established a comprehensive policy for the disposal of the nation's commercial spent nuclear fuel and high-level radioactive waste. The NWPA directed the DOE to develop guidelines for site characterization. The Secretary considered various geologic media in which repositories could be located, including salt, volcanic rock (such as basalt and tuff), and crystalline rock (such as granite). The site selection process developed by the DOE is described in 10 CFR Part 960 (Energy: General Guidelines for the Recommendation of Sites for Nuclear Waste Repositories). In 1983, the DOE selected nine locations in six states to study as potential repository sites and performed preliminary environmental assessments of each. In 1986, the DOE published the results of these assessments, which documented the selection of five candidate sites from the original nine (DOE 1986a). The Secretary then recommended to the President that site characterization programs be undertaken at three sites: Yucca Mountain, Nevada; Hanford, Washington; and Deaf Smith County, Texas (DOE 1986b).

The DOE began developing site characterization plans for each site and was preparing to begin site studies when Congress amended the NWPA in 1987. Concerned with rising cost projections for the simultaneous characterization of three sites, Congress directed the DOE to study only Yucca Mountain, to determine whether it was suitable for a repository, and to discontinue repository-related activities at the other sites and the work on the second repository program unless there is Congressional authorization.

The concept of disposing of waste in the desert regions of the Southwest was first proposed by the USGS in the 1970s. In 1976, the director of the USGS identified a number of positive attributes in and around the Nevada Test Site that would make positive contributions to geologic disposal, including multiple natural barriers, remoteness, and an arid climate (McKelvey 1976). In 1981, a USGS scientist documented that water tables in the desert Southwest are among the deepest in the world, and the geologic setting includes multiple natural barriers that could isolate waste for "tens of thousands to perhaps hundreds of thousands of years" (Winograd 1981). In contrast to the strategy for isolating waste in salt or deep sites below the water table, waste could be disposed near the Nevada Test Site at relatively shallow depths, well above the water table. Following the initial USGS recommendation, the DOE sponsored investigations of the feasibility of disposal above the water table. Formal site characterization at Yucca Mountain began in 1986 and continues today.

Recent experience in the U.S. has demonstrated that it is feasible to site and construct a repository for radioactive waste. In 1975, the ERDA selected a site in salt deposits near Carlsbad, New Mexico, to develop the Waste Isolation Pilot Plant for the disposal of transuranic waste. Transuranic waste contains radionuclides that are heavier than uranium; it emits less radiation and generates less heat than spent nuclear fuel. It is a by-product of the reprocessing of spent nuclear fuel and the use of plutonium in manufacturing nuclear weapons. Like high-level radioactive waste, transuranic waste remains hazardous for centuries and requires long-term isolation. In 1998, after more than 20 years of study, design, and analysis, the Waste Isolation Pilot Plant received the required permits from the EPA to receive and dispose transuranic waste. The plant opened in 1999 and has begun receiving waste from several DOE sites.

With the end of the Cold War, many nuclear weapons have been disassembled and removed from the nation's stockpile, which has resulted in the accumulation of surplus nuclear materials. A repository is part of the strategy for managing and disposing these materials.

Since the first scientific study in 1957, professional organizations that have looked at the nuclear waste problem have agreed that a geologic repository is the best approach for disposal. Scientists have widely agreed that waste encased in robust, long-lived waste packages and placed deep in stable geologic environments could be isolated from the biosphere for the long time periods necessary. As stated in a 1990 report from the National Academy

of Sciences, there is "a worldwide scientific consensus that deep geological disposal, the approach being followed by the U.S., is the best option for disposing of high-level radioactive waste" (National Research Council 1990, p. vii). An international group of scientists issued a similar opinion in 1995 that affirmed the consensus for geologic disposal in a document entitled *The Environmental and Ethical Basis of Geological Disposal of Long-Lived Radioactive Wastes: A Collective Opinion of the Radioactive Waste Management Committee of the OECD Nuclear Energy Agency* (NEA 1995).

The DOE published *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998) in December 1998. This assessment summarized the results of site characterization, described preliminary repository and waste package designs, and presented the DOE analysis of the future performance of a potential repository located at Yucca Mountain. It concluded that Yucca Mountain remained a promising site for development as a repository. The Viability Assessment also described a program of studies to address remaining uncertainties before the Secretary would decide whether to recommend the site.

Although there is a general scientific consensus in favor of geologic disposal, views differ on when to dispose of the waste and whether to dispose of it permanently. Many believe that the recoverable uranium-235 and other fissionable isotopes in spent nuclear fuel could be a future energy resource and should not be irreversibly disposed until their potential economic value is certain. Reprocessing spent nuclear fuel to reclaim the unused material is technically feasible but currently uneconomical in the U.S. Also, U.S. policy is to not encourage the civilian use of plutonium. Accordingly, the U.S. does not engage in plutonium reprocessing. Although reprocessing would reduce the amount of radioactive waste that requires disposal, it would not eliminate the need for disposal because reprocessing would generate additional waste materials that must be treated and disposed.

Some experts advocate alternative technologies that might make geologic disposal easier. For example, accelerator transmutation of waste—bombarding waste with nuclear particles to convert it into material that is less radioactive or shorter-lived—has been proposed as an alternative waste management strategy. A National Research Council study found that transmutation is "technically feasible," but as a method to treat spent nuclear fuel it "would require many tens to hundreds of billions of dollars and require several decades to implement" (National Research Council 1996a, p. 81). Although transmutation would reduce the total radionuclide inventory to be disposed, disposal of large quantities of high-level radioactive waste would still be necessary.

It has been suggested that a monitored retrievable storage facility, constructed at Yucca Mountain or elsewhere, would be preferable to geologic disposal at present. Although a monitored facility could safely isolate wastes for as long as it was properly maintained, this strategy has several disadvantages compared to a repository. It would not close the fuel cycle, and geologic disposal would still be required eventually. If the monitored facility were constructed at a location other than the same location as a repository, waste forms would have to be transported twice before final disposal, increasing the risk to workers and the public. Total program costs would be increased because of the need to construct, operate, and maintain two facilities.

One way to accommodate these different views and still provide a permanent solution is to dispose waste in a manner that permits, but does not require, its retrieval. The NWPA (42 U.S.C. 10101 et seq.) requires that the DOE design a repository so that waste can be retrieved for any reason. NRC licensing regulations require that the DOE design a repository so that waste can be retrieved on a reasonable schedule starting at any time up to 50 years after waste emplacement begins, unless a different period is approved or specified by the NRC (10 CFR 63.111(e) [66 FR 55732]).

Future generations would decide whether to keep the repository open and monitored or close it. To ensure that future decision-makers have some flexibility, the DOE will design the potential repository

so it can be closed as early as 50 years or as late as 300 years after emplacement starts.

1.3 DESCRIPTION OF THE SITE CHARACTERIZATION PROGRAM AND THE YUCCA MOUNTAIN SITE

In 1986, the DOE began a formal program of site characterization, as required by Section 113 of the NWPA (42 U.S.C. 10133). This program is described in *Site Characterization Plan Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1988). The plan established a comprehensive set of studies to gather the information needed to evaluate the suitability of the site against EPA, NRC, and DOE regulations in effect at the time. After releasing the plan in draft form in January 1988 for review by the NRC, the State of Nevada, Congress, and other interested parties, the DOE issued the final plan in December 1988. Progress reports (e.g., DOE 1999b) describing ongoing site characterization activities and plans are issued semiannually.

This report draws on numerous references that describe the investigations and results of the site characterization program in detail. *Yucca Mountain Site Description* (CRWMS M&O 2000b) is the most comprehensive.

1.3.1 Site Characterization Investigations

The site characterization program has performed extensive surface-based tests and investigations, underground tests, laboratory studies, and modeling activities designed to provide the technical basis for an evaluation of repository performance. Figure 1-3 shows the location of the surface-based and underground test facilities at Yucca Mountain, including boreholes and underground excavations. The single largest effort of the site characterization program was the Exploratory Studies Facility, which provided access to the subsurface environment for exploration and testing along the entire north–south extent and at the proposed depth of the potential repository. Work on the Exploratory Studies Facility started in late 1992, and excavation with a tunnel boring machine began in September 1994 (Figure 1-4). The 8-km (5-mi) underground tunnel that is the main part of the Exploratory Studies Facility was completed in April 1997. A drift across the entire planned width of the potential repository, the Enhanced Characterization of the Repository Block (ECRB) Cross-Drift, was completed in October 1998. Additional subsurface niches and alcoves have been excavated to support specific tests.

The suite of site characterization testing and analysis includes the following components:

- Surface-based mapping, sampling, and testing of geologic and hydrologic features and properties
- Surface-based and borehole geophysical testing at both regional and site-specific scales
- Geologic, hydrologic, and geochemical sampling and testing in the Exploratory Studies Facility and the ECRB Cross-Drift
- Studies of hydrologic processes and investigations of coupled thermal-hydrologic-geochemical-mechanical processes in the Exploratory Studies Facility and the ECRB Cross-Drift
- Characterization of geologic and hydrologic properties from borehole studies and long-term borehole monitoring of hydrologic properties
- Surface-based, borehole, and Exploratory Studies Facility studies of hydrologic and geologic properties of faults and fractures, as well as their distribution
- Hydrologic testing in the Calico Hills hydrogeologic unit at the Busted Butte test facility
- Regional geologic studies and trenching for seismic and volcanic hazard studies
- Meteorological monitoring and modeling
- Surface environmental studies, including biological and ecological investigations

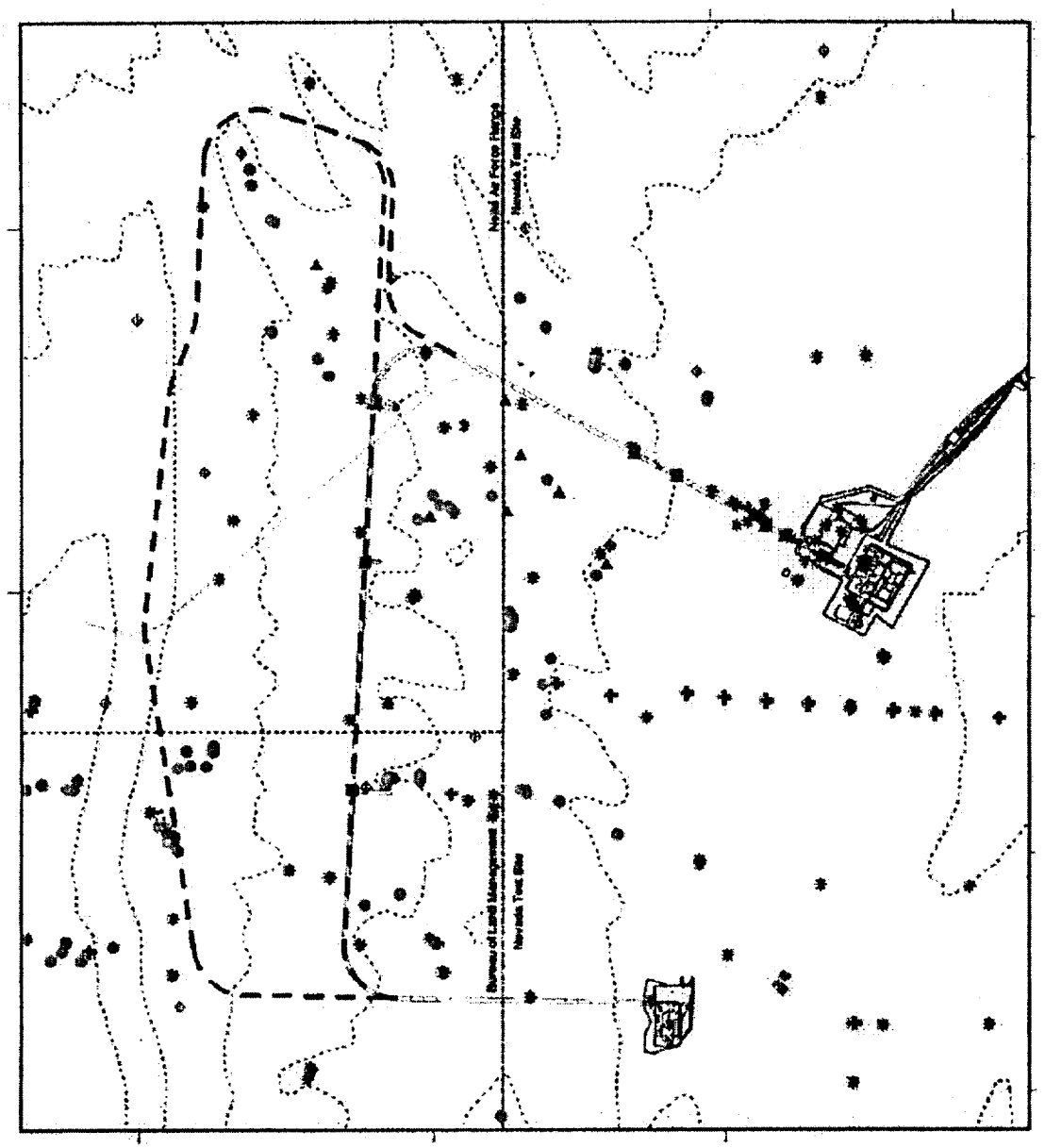
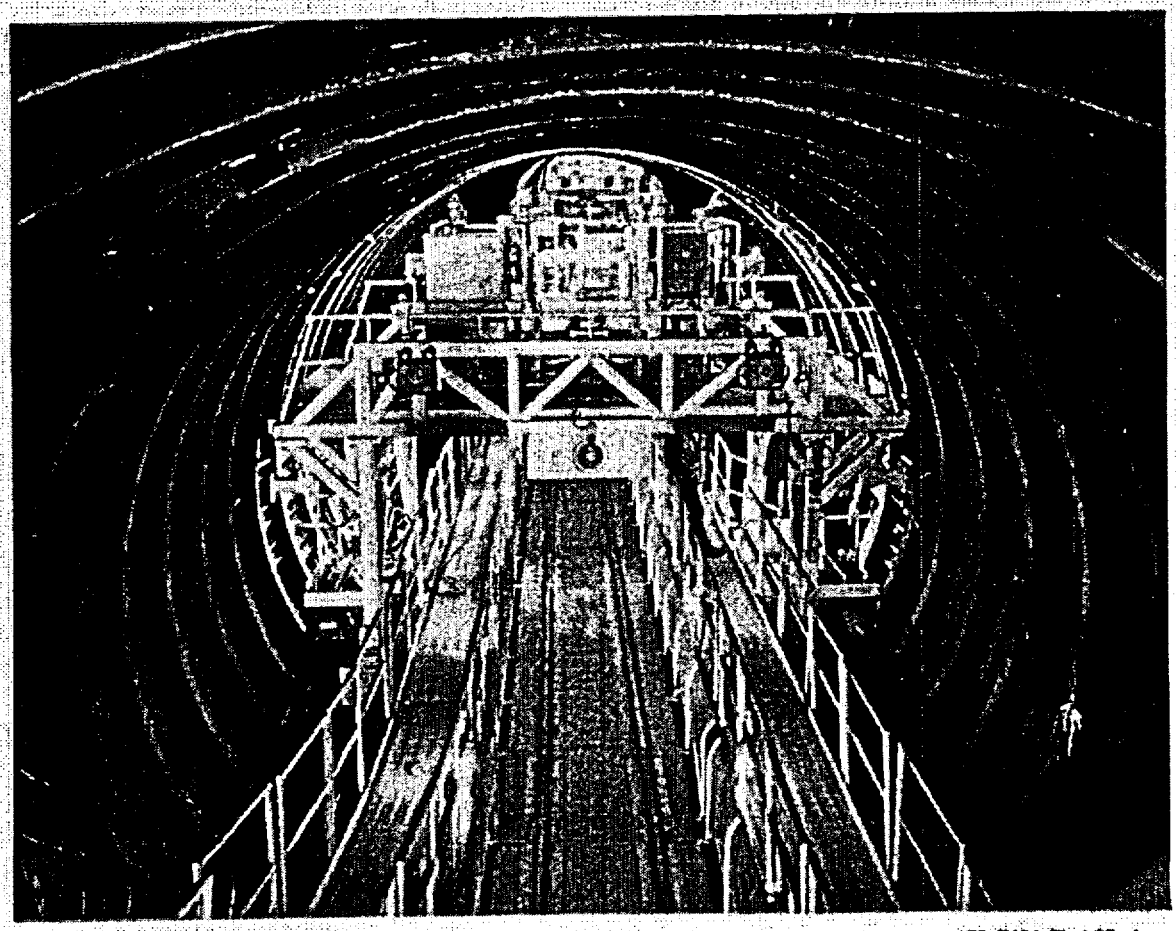


Figure 1-3. Site Investigation Area Showing Location of the Surface-Based and Underground Test Facilities at Yucca Mountain, Including Boreholes and Underground Excavations

Additional investigation locations, including the Nye County Early Warning Program wells, are not shown because they are located outside the boundaries of the area depicted.



ATP_Z1S1_Fig 1-07.cdr

Figure 1-4. Photograph Showing the Tunnel Boring Machine Operating at Yucca Mountain
Work on the Exploratory Studies Facility started in late 1992, tunnel boring machine operations commenced in September 1994, and the Exploratory Studies Facility ramps and main drift were completed in the spring of 1997.

- Geotechnical investigations, including in situ and laboratory testing of soil properties
- Seismic monitoring and seismic hazard studies to address the potential for, and characteristics of, earthquakes that could affect the potential repository
- Laboratory geochemical tests and analyses of the transport characteristics of water and rocks under ambient and potential repository conditions
- Laboratory chemical tests and analyses of the dissolution properties of waste materials under ambient and potential repository conditions
- Laboratory physical tests of the mechanical properties and behavior of rocks under potential repository conditions
- Laboratory testing of materials planned for use in the repository under potential repository conditions
- Development of conceptual and numerical models, and verification and validation of hydrologic, transport, and coupled process models

- Performance assessment modeling of repository behavior
- Analogue studies of hydrologic and geologic processes.

The results of site investigations have been reported regularly since 1990 in semiannual progress reports (e.g., DOE 1999b). The progress reports also demonstrate the evolution of the scientific and engineering program from the *Site Characterization Plan Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1988). The technical program has evolved in response to advancements in scientific understanding and proposed changes in regulatory and program requirements, such as the design of the repository. These changes are summarized in appendices to the 1997 and 1998 progress reports (e.g., DOE 1997c), and have been documented in a separate report since 1998 (e.g., CRWMS M&O 1999b). For example, EPA regulations in place in 1988 would have required the DOE to assess compliance with the radiation protection standards at a point 5 km (3 mi) from the repository boundary. Currently, NRC and EPA regulations would require the DOE to assess potential doses to humans from the transport of radionuclides at the accessible environment specified by 40 CFR Part 197 and 10 CFR Part 63 (66 FR 55732). This regulatory change created a need to understand the hydrologic flow and transport characteristics of the region between 5 km (3 mi) and the accessible environment (approximately 18 km [11 mi] south of the potential repository) from Yucca Mountain. The DOE has addressed this need by supporting and incorporating the results of the Nye County Early Warning Drilling Program and other information into the analyses and models used to assess the performance of the potential repository.

The program has also changed because of information learned during site characterization. One example is the seepage tests in the Exploratory Studies Facility that were added to evaluate how much water flow is likely to be diverted around underground openings.

Scientific and engineering activities are documented in the key technical references supporting this report, which include nine process model reports describing the DOE's understanding of the current and future behavior of a potential repository. These reports address the following topics:

- Integrated site model
- Unsaturated zone flow and transport
- Near-field environment
- Biosphere
- Waste package degradation
- Waste form degradation
- Engineered barrier system degradation, flow and transport
- Saturated zone flow and transport
- Disruptive events (volcanic/seismic hazards).

1.3.2 Description of the Yucca Mountain Site

This section provides an overview of the location, geography, current population, and key geologic characteristics of the Yucca Mountain site. This description is provided as background information for Sections 2 and 3, which present a description of the proposed repository design (including information related to repository performance over a range of operating mode temperatures) and descriptions of the waste forms and waste packages. This section also provides a framework for the descriptions of hydrologic, geochemical, thermal, and other processes related to the performance of the potential repository contained in Sections 4.2, 4.3, and 4.4.

Yucca Mountain is located on federal land in a remote area of Nye County in southern Nevada, approximately 160 km (100 mi) northwest of Las Vegas (Figure 1-5). If the site is recommended for repository development, sufficient land may be withdrawn to consolidate the control of the facilities and land needed to operate the repository. The draft EIS describes the potential withdrawal area (DOE 1999a, Section 3.1.1.3).

1.3.2.1 Geography, Land Use, and Population

Yucca Mountain consists of a series of ridges extending approximately 40 km (25 mi) from Timber Mountain in the north to the Amargosa Desert in the south. The elevation at the crest of the

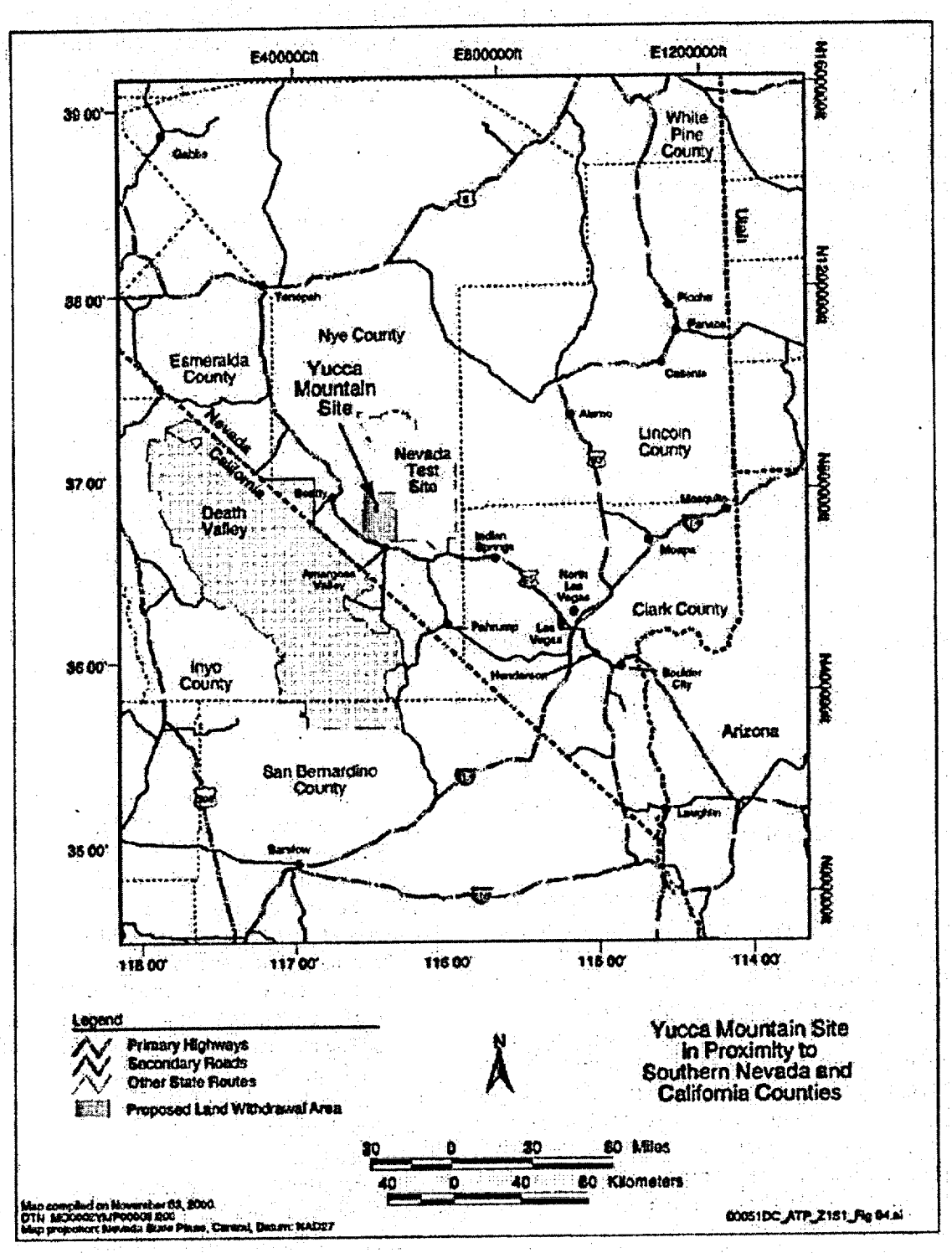


Figure 1-5. Map Showing the Location of Yucca Mountain in Relation to Major Highways; Surrounding Counties, Cities, and Towns in Nevada and California; the Nevada Test Site; and Death Valley National Park

Yucca Mountain is located on federal land in Nye County in southern Nevada, approximately 160 km (100 mi) northwest of Las Vegas.

ridges varies from approximately 1,800 m (5,900 ft) to 900 m (3,000 ft) above sea level. At the potential repository site, the crest of Yucca Mountain is 1,400 to 1,500 m (4,600 to 4,900 ft) above sea level. The western part of the potential repository site is a steep slope that rises approximately 300 m (1,000 ft) above the base of Solitario Canyon. On the eastern side, the mountain slopes gently to the east and is incised by a series of east- to southeast-trending stream channels. The elevation at the base of the eastern slope is approximately 350 to 450 m (1,100 to 1,500 ft) below the ridge crest.

The site is in Nye County, which is bordered by Clark, Lincoln, White Pine, Eureka, Lander, Churchill, Mineral, and Esmeralda counties in Nevada and Inyo County in California. The federal government controls nearly all of the land in the region. The area needed for the potential repository encompasses land controlled by three federal agencies: the U.S. Air Force (Nellis Air Force Range), the DOE (Nevada Test Site), and the U.S. Bureau of Land Management (Figure 1-6). Except for a few scattered facilities constructed on the Nevada Test Site to support former U.S. nuclear propulsion and defense programs, there has been no development at the site. The land around Yucca Mountain will remain federally owned and, should the site be recommended, will be withdrawn from public use.

The tracts of private land nearest to Yucca Mountain are to the south, in the Amargosa Desert. The closest year-round housing is at the intersection of U.S. Highway 95 and Nevada State Route 373, approximately 22 km (14 mi) south of the site. Active agricultural operations can be found approximately 30 km (19 mi) south of Yucca Mountain, in Amargosa Valley. There is one patented mining claim about 16 km (10 mi) south of the potential repository site. Scattered private lands are also present near the town of Beatty, 24 to 32 km (15 to 20 mi) west of the site.

Because groundwater in the Yucca Mountain region flows toward the south, the potential for future exposure to radionuclides would be highest for populations living south of the site. EPA and NRC regulations (40 CFR 197.21(b) and 10 CFR 63.312 [66 FR 55732], respectively) for Yucca Mountain would direct the DOE to assume that the receptor is a person, known as the reasonably maximally exposed individual, with a diet and living style similar to the current residents of the Town of Amargosa Valley.

The population density near Yucca Mountain is very low (CRWMS M&O 2000b, Section 2.3). There are no permanent residents within 22 km (14 mi), and Nye County as a whole averages only 0.6 persons per square kilometer (1.5 persons per square mile). Of the total population of 29,730 in Nye County, 68 percent live in the unincorporated town of Pahrump, 70 to 80 km (43 to 50 mi) south-southeast of Yucca Mountain. The major economic activities in the towns of Amargosa Valley and Beatty include agricultural and mining operations. Most of the other counties surrounding Yucca Mountain, including Lincoln and Esmeralda counties in Nevada and Inyo County in California, also have low population densities. The nearest large populations reside in Clark County, approximately 130 km (80 mi) southeast of Yucca Mountain.

1.3.2.2 Geology

The Yucca Mountain site is located on the western boundary of the Nevada Test Site, where scientists have conducted geologic investigations since the 1950s. Studies related to nuclear waste disposal have focused on Yucca Mountain since the late 1970s and have included careful mapping of the rocks at the surface and in more than 10 km (6 mi) of tunnels below Yucca Mountain, along with the drilling and logging of numerous wells and boreholes. The characterization of the geology of Yucca Mountain provides a framework for understanding the future behavior of the potential repository. Section 4 describes data and analyses related to postclosure safety, including a discussion of processes such as water flow and thermal effects and how they would affect both the natural and engineered parts of the repository system. The following overview of the geology of the site describes the physical environment of the repository. It is based on the comprehensive description contained in *Yucca Mountain Site Description* (CRWMS M&O 2000b).

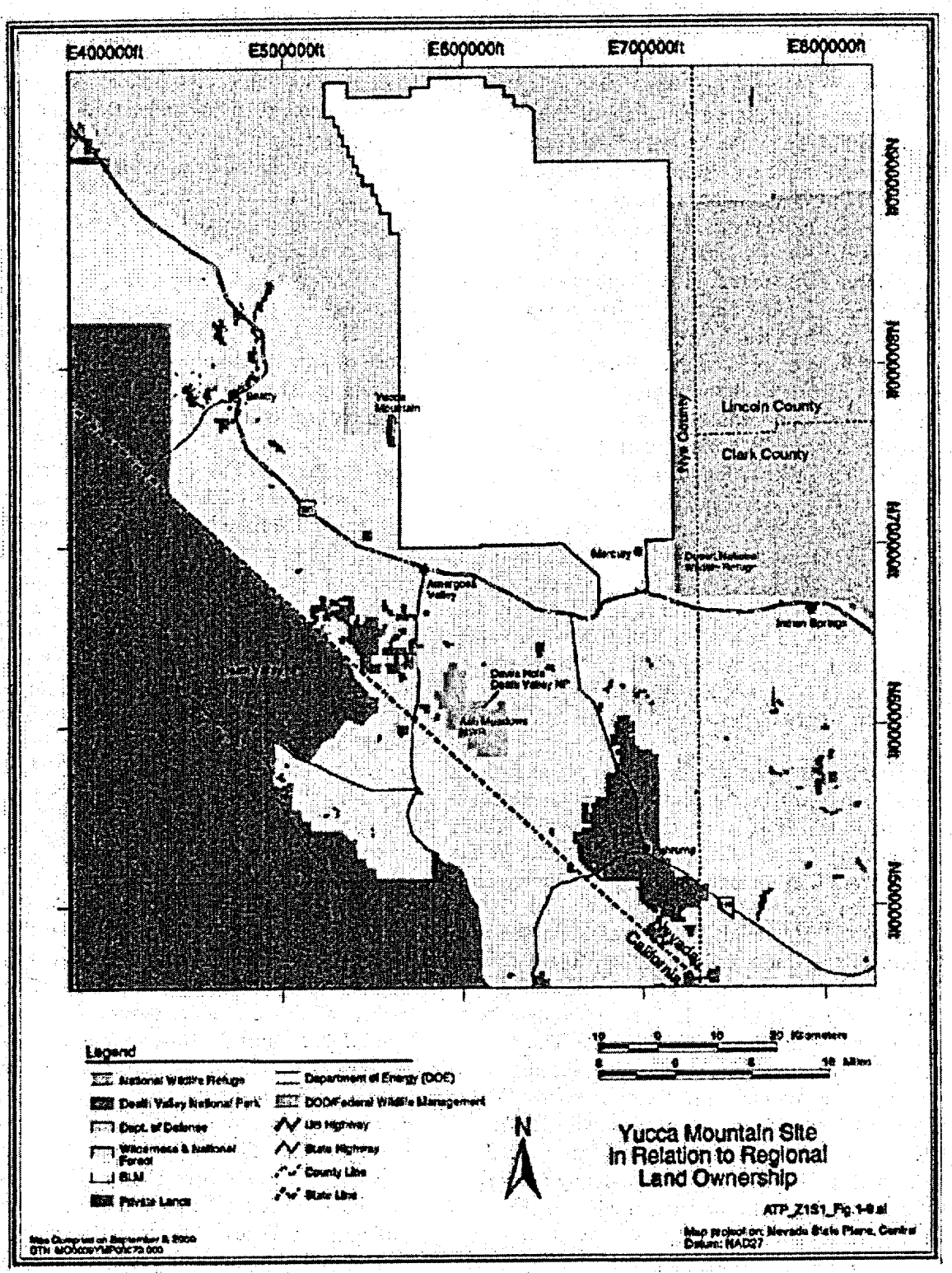


Figure 1-6. Map Showing the Location of Yucca Mountain and Land Status in the Region
The potential repository straddles land controlled by three federal agencies: the U.S. Air Force (Nellis Air Force Range), the DOE (Nevada Test Site), and the U.S. Bureau of Land Management (BLM).

1.3.2.2.1 Regional Geology

As shown in Figure 1-7, Yucca Mountain is located in the Basin and Range province of the western U.S., within the region known as the Great Basin. The Great Basin encompasses nearly all of Nevada and parts of Utah, Idaho, Oregon, and California. The Basin and Range, which extends into northern Mexico, draws its name from its characteristic, generally north–south aligned mountain ranges. These ranges are separated by basins containing thick deposits of sediment (mostly sand and gravel) derived from erosion of the adjacent ranges over millions of years. The structure of the Basin and Range has developed over a period of more than 30 million years. In the region of southern Nevada that includes Yucca Mountain, the pattern of mountains and valleys has formed over the past 15 million years from faults moving on one or both sides of the ranges (Fridrich 1999). The tilted, fault-bounded mountain ranges may extend more than 80 km (50 mi) in length and are generally 8 to 24 km (5 to 15 mi) wide. Relief between valley floors and mountain ridges is typically 300 to 1,500 m (1,000 to 5,000 ft), and valleys occupy between 50 and 60 percent of the total land area.

Most modern tectonic activity (i.e., active faulting and volcanism) in the southwestern Great Basin occurs to the south, west, and northwest of Yucca Mountain. Among the most active areas in the region are the Furnace Creek–Death Valley fault zone, the Sierra Nevada front (i.e., the Owens Valley and Mammoth Lakes area), and the area north of the Garlock fault in the Mojave Desert. This domain includes modern basins and ranges with great structural relief, such as the Death Valley basin and the Panamint Range. The modern faulting and volcanic activity is caused by the continuation of the same tectonic extension that resulted in the formation of the entire Basin and Range. The crust on the western edge of the Great Basin (the Sierra Nevada) is gradually moving to the west relative to the eastern edge of the basin (the Wasatch Front in Utah).

Regional Stratigraphy—Rocks and sedimentary deposits exposed in the region surrounding Yucca Mountain range in age from geologically old (Precambrian, or more than 570 million years old) in the mountains to geologically recent (Holocene, or less than about 10,000 years old) in the valleys. Understanding the distribution of rock types enables geologists to understand the geologic history of the area, which is fundamental to analyses of geologic hazards. Also, the characteristics of rock types below and around Yucca Mountain influence the regional flow of groundwater and would control where any potential releases of radionuclides from the repository system might migrate.

Most of the Precambrian (older than about 570 million years) and Paleozoic (570 to 240 million years old) rocks in the Yucca Mountain region are sedimentary or metamorphic rocks that are not very permeable. The oldest Precambrian "basement" rocks are highly metamorphosed gneisses and schists that have been dated at about 1.7 billion years. These rocks are overlain by less metamorphosed sedimentary layers composed primarily of quartzite, siltstone, shale, and carbonate (CRWMS M&O 2000b, Section 4.2.2). Except for the carbonates described below, all of the Precambrian and Paleozoic units that underlie the volcanic rocks in the Yucca Mountain region are aquitards (i.e., rocks of low permeability that do not readily transmit water). In these units, water flow generally occurs only in strongly fractured zones.

In contrast, there are several Paleozoic carbonate (limestone or dolomite) units that form important aquifers throughout southern Nevada (Winograd and Thordarson 1975). For example, the Spring Mountains, between Yucca Mountain and the Las Vegas Valley, are composed largely of carbonate rocks, which are a major source of recharge (i.e., rainfall that enters the flow system) to the regional groundwater system. Near Yucca Mountain, the most significant groundwater discharge occurs in carbonate rocks throughout the Ash Meadows area, which is about 50 km (30 mi) south-southeast of the potential repository. The aquifer from which this flow originates lies below the aquifers in the tuff units at Yucca Mountain. Knowledge of the location of groundwater recharge and discharge points, the direction of flow, and the relationship between the rock units and the groundwater system is important to analyses of potential future releases of radionuclides to the environment.

Figure 1-7. Map Showing the Location of Yucca Mountain and Major Physiographic Provinces of the Southwest

Yucca Mountain is located in the southern Great Basin, within the Basin and Range province of the western U.S. The Great Basin is bounded on the east by the Rocky Mountains and the Colorado Plateau, and on the west by the Sierra Nevada Range.

Between about 14 and 7.5 million years ago (during the Miocene Epoch of the Cenozoic Era), a series of large-scale volcanic eruptions resulted in the formation of Yucca Mountain and the southwestern Nevada volcanic field (Sawyer et al. 1994), which consists of six major volcanic centers, or calderas (Figure 1-8). The Claim Canyon Caldera, just north of Yucca Mountain, was the eruptive source of the approximately 13-million-year-old rock units, known as the Paintbrush Group, that now form the mountain ridges at the potential repository site. Eruptions from the southwest Nevada volcanic field ended about 7 million years ago. More recently, small volume volcanoes (known as cinder cones) have erupted lava flows and volcanic ash to the west and south of Yucca Mountain (Crowe, Perry et al. 1995). Four cinder cones formed between about 1.3 and 0.7 million years ago in Crater Flat, west of Yucca Mountain. The latest volcanic episode created the Lathrop Wells Cone, about 16 km (10 mi) south of the potential repository site, about 80,000 years ago. This most recent volcanic activity in Crater Flat and at Lathrop Wells is described in *Yucca Mountain Site Description* (CRWMS M&O 2000b, Section 12.2), which provides the basis for the assessment of volcanic hazards in Section 4.3.2.1.

Surficial deposits in the Yucca Mountain region provide a record of the evolution of surface processes and climate conditions over the past several hundred thousand years. Most surficial deposits in the region are composed of sands and gravels that are called alluvium if they are deposited by flowing surface water or colluvium if they originate from hill slopes as flows of debris. Eolian (wind-blown) deposits, such as sand dunes, are generally a minor component of the surficial deposits, except for a massive dune at Big Dune and sand ramps like those that flank Busted Butte southeast of Yucca Mountain. Southwest and south of Yucca Mountain, scientists have mapped minor spring and marsh deposits reflecting past, wetter climates. The ages of surficial deposits range from less than 1,000 years to more than 760,000 years, but most deposits exposed at the surface were deposited during the last 100,000 years. Determining the ages and distributions of these deposits is important to understanding the age and movement of faults in the area. A complete description of the results of studies of surficial deposits can be found in *Yucca Mountain Site Description* (CRWMS M&O 2000b, Section 4.4).

The characteristics of surface deposits indicate that erosion in the Yucca Mountain region generally has proceeded slowly. Volcanic features are well preserved, and basic geologic relationships indicate that modern landforms (e.g., ridges and valleys) were already established by the time of eruption of the Rainier Mesa Tuff, approximately 11.6 million years ago.

Near-surface carbonate deposits (sometimes called caliche or calcrete) occur in soils at shallow depths parallel to the surface and as fracture fillings. Evidence indicates that these deposits are pedogenic (i.e., related to the formation of soil) in origin, supporting the conclusion that past climate in the region has generally been similar to the modern semiarid to arid conditions, although some periods have been wetter. These types of deposits form in arid environments when downward infiltrating rainwater dissolves minerals present at the surface. Calcium carbonate is then precipitated in the soils when the infiltrating water evaporates or is taken up by plant roots.

1.3.2.2.2 Site Bedrock Geology

The rocks that might host the potential repository are important to all aspects of repository design and performance. Figures 1-9 and 1-10 are a simplified geologic map and cross section modified from Day, Dickerson et al. (1998) that show geologic relations at a repository scale. In addition, stratigraphic, structural, and rock property data have been combined to form an integrated site geologic model (CRWMS M&O 1997a). This geologic model provides a common framework for developing the repository design and assessing the performance of the repository system.

Measurements of the water level in boreholes at Yucca Mountain indicate that the water table is approximately 500 to 800 m (1,600 to 2,600 ft) below the ground surface. The potential repository would be located above the water table in the unsaturated zone. The ash-flow tuff layers

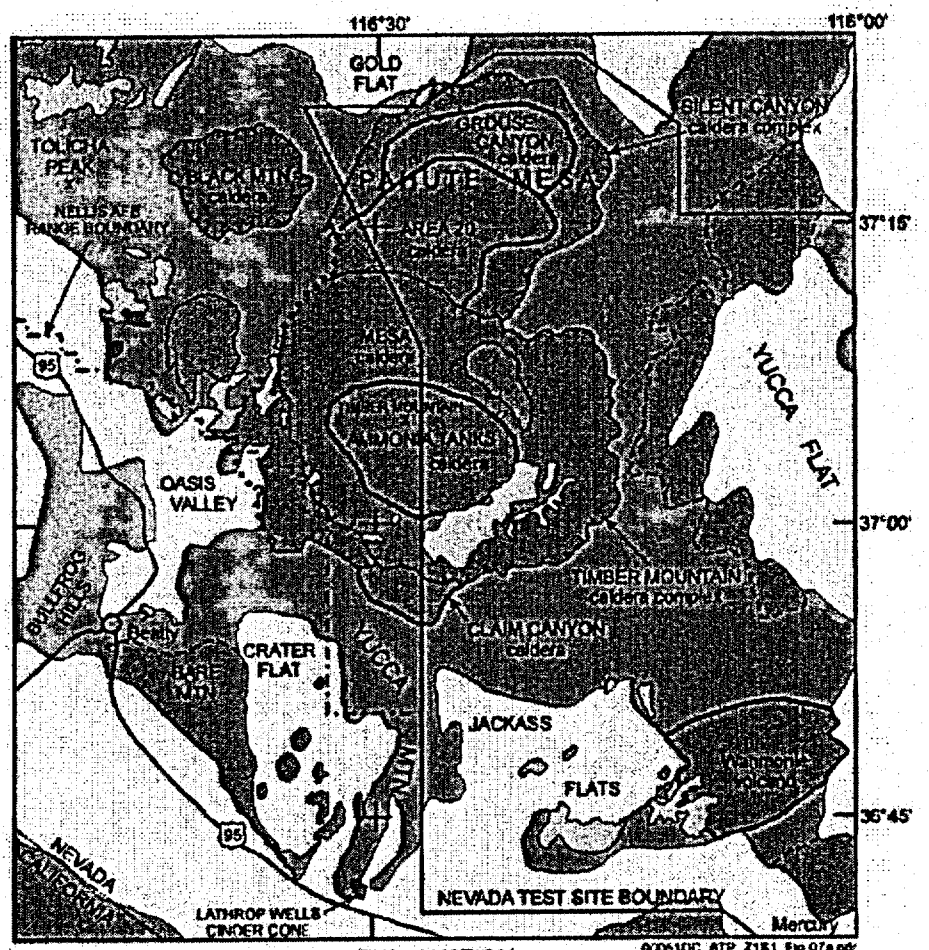


Figure 1-8. Simplified Geologic Map Showing the Location of Yucca Mountain in Relation to the Southwestern Nevada Volcanic Field

The map shows the location of (1) pre-Tertiary basement rocks (mostly Paleozoic sedimentary rocks ranging from 225 to 570 million years in age); (2) Tertiary volcanic rocks of Miocene age (i.e., 15 to 7.5 million years old) deposited during very large, caldera-forming eruptions; (3) Tertiary and Quaternary basaltic rocks of the southwestern Nevada volcanic field formed during much smaller volcanic eruptions, with ages ranging from more than 5 million years to about 77,000 years; and (4) Quaternary deposits of alluvium and other sedimentary deposits less than 2 million years old. Source: modified from Sawyer et al. 1994; Stewart 1997.

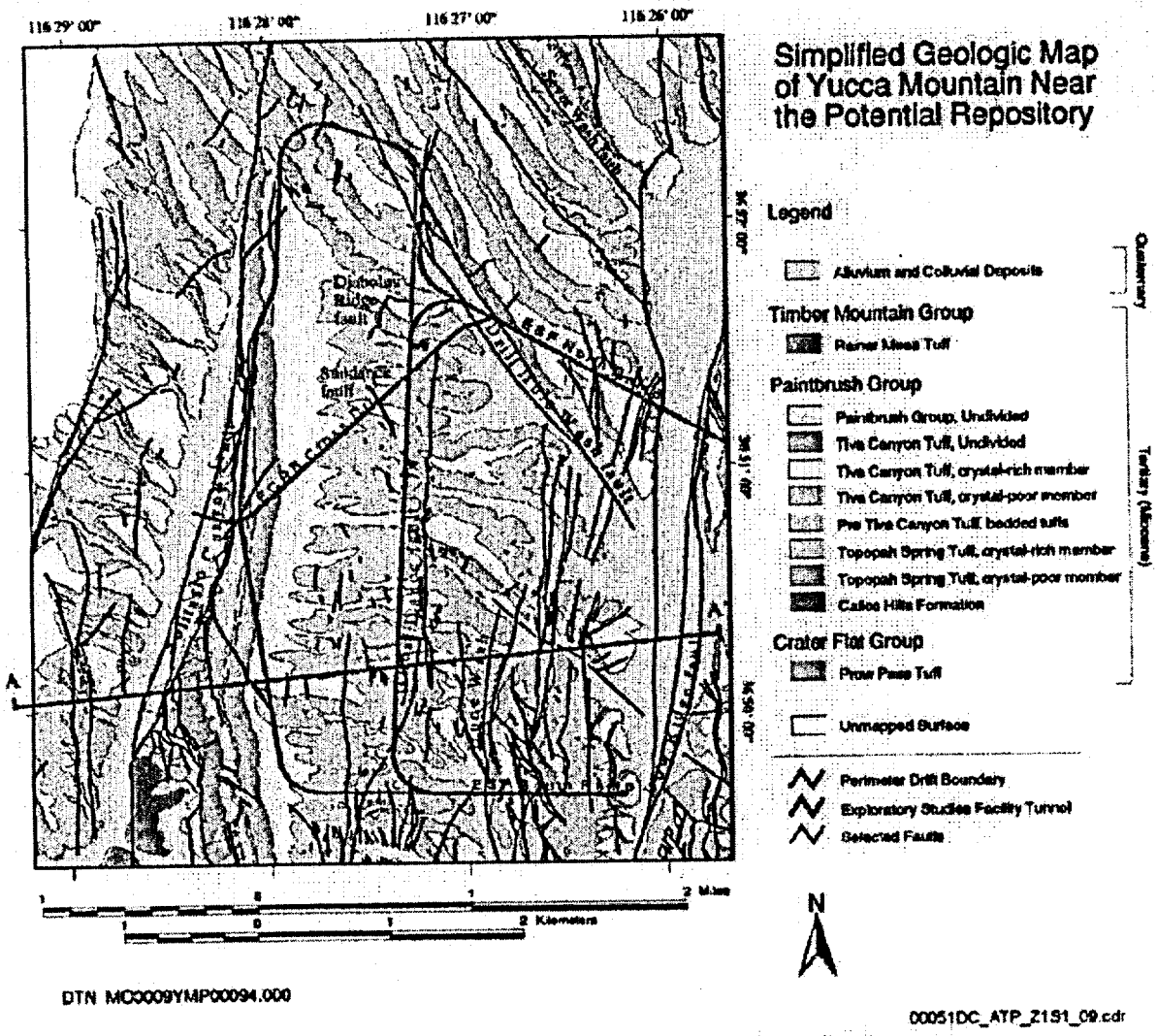


Figure 1-9. Simplified Geologic Map of Yucca Mountain Near the Potential Repository
The geologic map shows the distribution of the eastward dipping volcanic rocks that form Yucca Mountain and the locations of selected faults. The line A–A' indicates the plane of the cross section shown in Figure 1-10. ESF = Exploratory Studies Facility. Source: Modified from Day, Dickerson et al. 1998.

discussed in this section lie mostly within the unsaturated zone.

Site Stratigraphy—Yucca Mountain consists of successive layers of volcanic rocks (called tuffs), approximately 14 to 11.6 million years old, formed by eruptions of volcanic ash from calderas to the north. Individual layers of tuff thin from north to south. Most of these volcanic rocks are ash-flow tuffs of two types, welded and nonwelded, that formed when hot volcanic gas and ash erupted violently and flowed quickly over the landscape. As the ash settled, it was subjected to various degrees of compaction and fusion, depending on temperature and pressure. When the temperature was high enough, the ash was compressed and fused to produce a welded tuff—a hard, brick-like rock with very little open pore space in the rock matrix. Nonwelded tuffs, which occur between welded layers, are compacted and consolidated at lower temperatures, are less dense and brittle, and have a higher porosity (more open pore space in the rock). The composition of the rocks at Yucca Mountain ranges from rhyolite (a volcanic rock

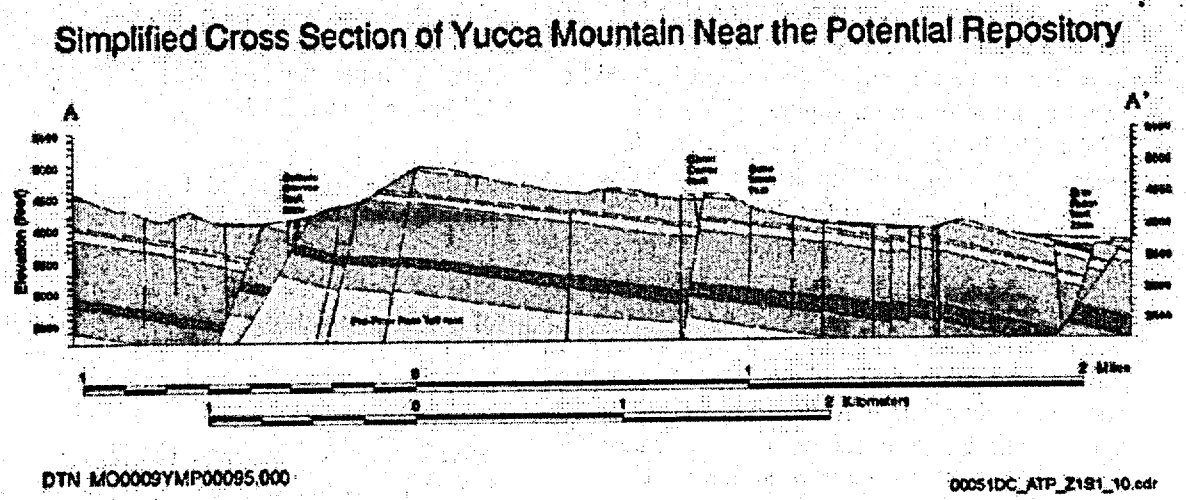


Figure 1-10. Simplified Cross Section of Yucca Mountain Near the Potential Repository
The cross section shows the distribution of the eastward dipping volcanic rocks that form Yucca Mountain and the locations of selected faults. The location of this cross section, A–A', is indicated in Figure 1-9, and the legend in Figure 1-9 corresponds to this geologic cross section. Source: Modified from Day, Dickerson et al. 1998.

type with a chemical composition similar to granite) to dacite or latite. Dacite and latite are volcanic rock types with chemical compositions characterized by silica contents that are intermediate between rhyolite (high silica) and basalt (low silica).

In the immediate vicinity of the potential repository, the stratigraphically highest volcanic unit present is the Rainier Mesa Tuff of the Timber Mountain Group. As shown in Figures 1-9 and 1-10, the Rainier Mesa Tuff, which is approximately 11.6 million years old, is found in only a few locations in the faulted valleys east and west of the crest of Yucca Mountain. It consists of nonwelded to partially welded rhyolitic ash flows that are up to about 30 m (100 ft) thick in this area. Beneath the Rainier Mesa Tuff, other volcanic rocks (known as pre-Rainier Mesa bedded tuffs) are also locally present. These tuffs are also nonwelded ash-flow deposits, and they range in thickness from 0 to approximately 60 m (200 ft).

Most of the surface of Yucca Mountain above the potential repository location is composed of the volcanic rocks of the Paintbrush Group. The Paintbrush Group is composed of three distinct volcanic tuff layers that occur between the surface and the location of the potential repository: the Tiva Canyon welded tuff at the surface, the Topopah Spring welded tuff at the level of the potential repository, and an intervening layer of nonwelded tuffs. As a result of faulting over the last 13 million years, these layers are all tilted to the east about 10° (Figure 1-10). The Tiva Canyon Tuff is a large-volume, regionally extensive ash-flow tuff (Sawyer et al. 1994, Table 1) that has been dated at approximately 12.7 million years old. The thickness of the Tiva Canyon Tuff ranges from 50 to 175 m (165 to 575 ft); it is approximately 100 m (330 ft) thick near the potential repository site.

A layer of nonwelded tuff underlies the Tiva Canyon Tuff near the site of the potential repository. This nonwelded layer includes two separate ash flows, the Yucca Mountain Tuff and the Pah Canyon Tuff. In the vicinity of the potential repository, the total thickness of the nonwelded units ranges from 30 to 50 m (100 to 165 ft). These nonwelded units contain few fractures, so they delay the downward flow of water below the surface.

The lowermost unit in the Paintbrush Group is the Topopah Spring Tuff, which would be the host rock for the potential repository. The Topopah Spring Tuff was formed by an eruption about 12.8 million years ago and has a maximum thickness of about 375 m (1,230 ft) near Yucca Mountain. Based on surface mapping and studies of boreholes and underground exposures, the Topopah Spring Tuff has been subdivided into several layers according to chemical composition, mineral content, the size and abundance of pumice and rock fragments, and other variations in texture and appearance. An important characteristic of the layers is the presence and abundance of lithophysae, which are small, bubble-like holes in the rock caused by volcanic gases that were trapped in the rock matrix as the ash-flow tuff cooled. The average lithophysae range from about 1 to 50 cm (0.3 to 20 in.) in size, with a maximum size of about 1 m (3.3 ft) (CRWMS M&O 2000b, Section 4.5.3.1). Their nature, size, and abundance may affect the tuff's thermal, mechanical, and hydrologic properties.

The lower and middle portions of the Topopah Spring Tuff have been divided into four layers according to the amount of lithophysae they contain. Because these layers are tilted, and the drifts in the potential repository would be approximately horizontal, the potential repository horizon crosses the lithophysal zones. Like the Tiva Canyon Tuff, the Topopah Spring Tuff is fractured throughout; these fractures provide the main pathway for water to flow through the rock unit (see Section 4.2.1).

Beneath the Paintbrush Group, the Calico Hills Formation is a series of mostly nonwelded rhyolite tuffs and lavas that were erupted approximately 12.9 million years ago (Sawyer et al. 1994, Table 1). The formation thins southward, from a total thickness of about 290 m (950 ft) north of the repository block to 40 m (135 ft) south of it. Several characteristics of the Calico Hills Formation are important to repository performance. None of the tuffs of the Calico Hills are densely welded; therefore, they generally have higher matrix porosities than the Topopah Spring Tuff. Because the rock has higher ductility, the fractures that are common in welded tuffs are less common in the Calico Hills Formation. Surface, borehole, and Exploratory Studies Facility observations indicate that the highly fractured Topopah Spring Tuff may overlie tuffs of the Calico Hills Formation that have a lower fracture density (CRWMS M&O 2000b, Section 4.6.6.3).

Analyses of surface and borehole samples (e.g., Bish and Chipera 1986, Table 2; Broxton et al. 1993, p. 1) show another important feature of the tuffs of the Calico Hills Formation: an abundance of zeolite minerals in the rock matrix and fractures. Zeolites are silicate minerals that have the ability to sorb (take up on their mineral surface and hold) radionuclides and other ions that might be transported in solution in water. The DOE's approach to ion exchange sorption is to quantify the extent of radionuclide-sorbent interaction, which does not require identifying the specific underlying processes of sorption (see Section 4.2.8.1). The zeolite minerals may also affect transport properties in another way: tuffs with high zeolite content have reduced matrix permeability, which will tend to focus water flow into any fractures that are present (CRWMS M&O 2000c, Section 3.6.3.1).

The geologic units below the water table contain older volcanic rocks composed mainly of welded and nonwelded ash-flow tuffs. These older units can be up to 1,000 m (3,300 ft) thick below Yucca Mountain (CRWMS M&O 2000b, Section 4.5.4). The volcanic rocks are underlain by the Paleozoic limestones and dolomites described in Section 1.3.2.2.1. Near Yucca Mountain, the older volcanic rocks and the Paleozoic rocks lie deep beneath the surface, but they are found at much shallower depths (and even at the surface) to the south, where they are an important component of the hydrologic flow system.

Selection of the Repository Location and Host Rock—The identification of a subsurface location for a potential repository was based on several factors, including the thickness of overlying rock and soil, the extent and geomechanical characteristics of the host rock, the location of faults, and the depth to groundwater (CRWMS M&O 2000d). Figure 1-11 shows the key features of the site that have controlled the siting of the repository, as described below.

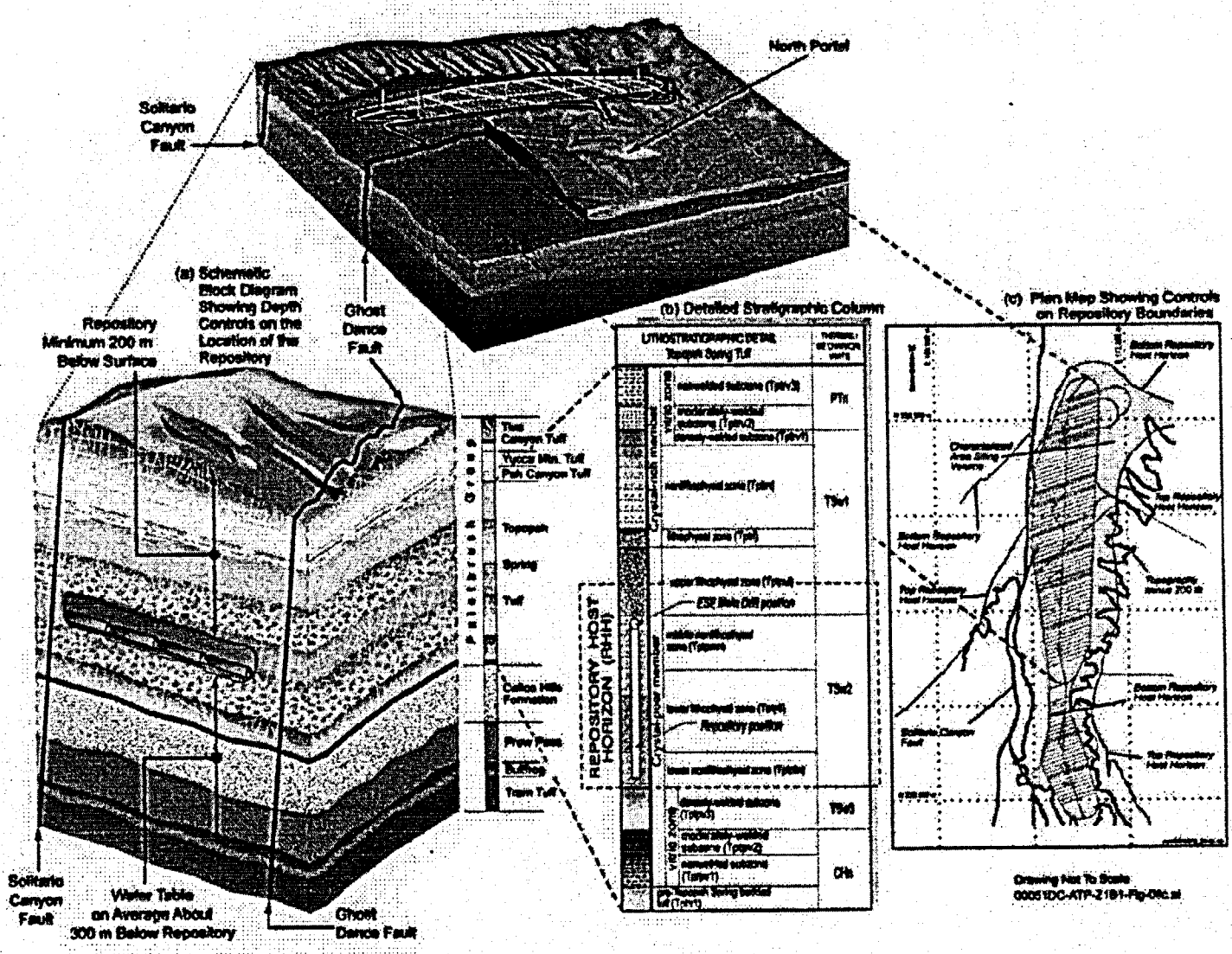
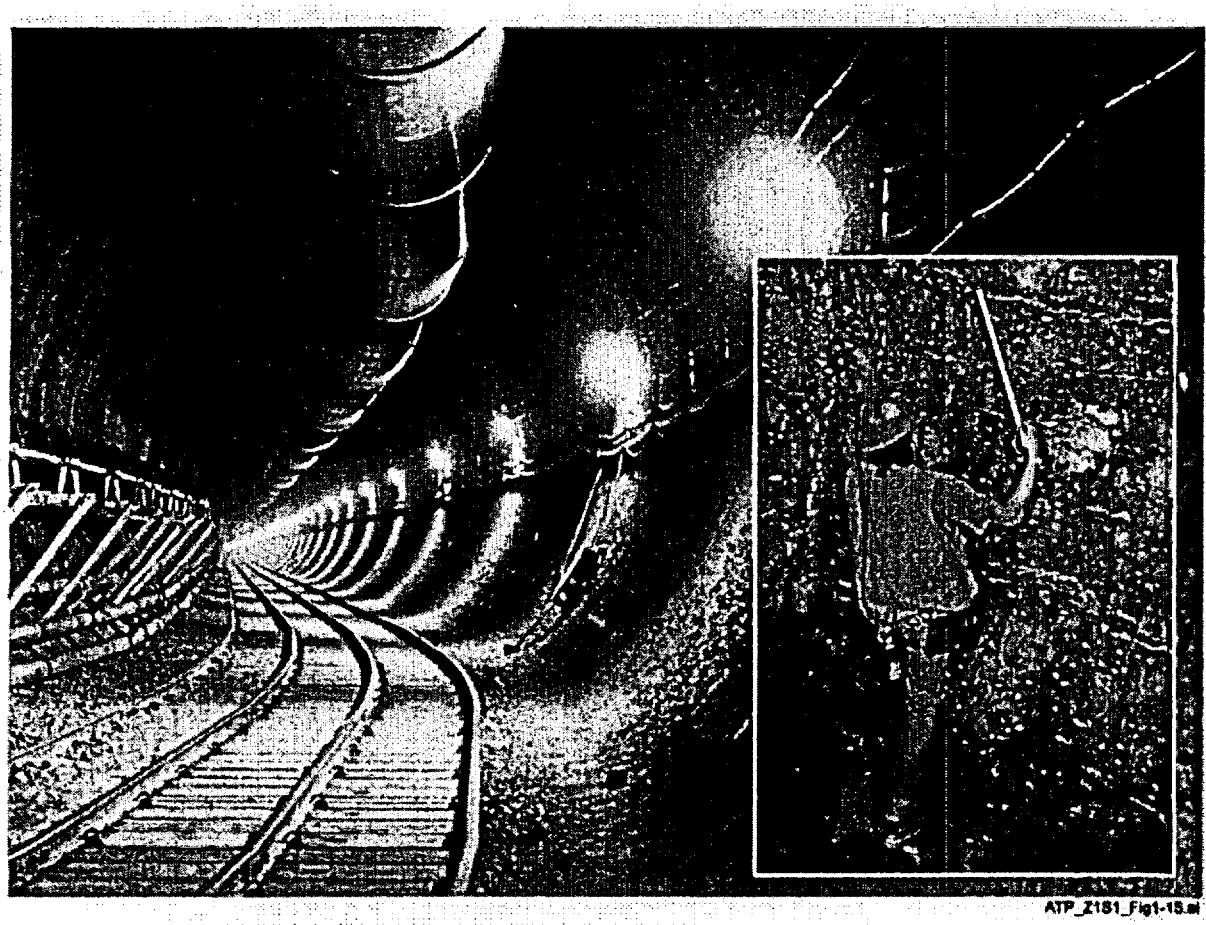


Figure 1-11. Layout and Boundaries of the Potential Repository
(a) Block diagram showing the location of the potential repository with respect to depth below the surface, the water table, and nearby faults. The scale of the emplacement drifts is exaggerated to show schematic waste packages. (b) Stratigraphic section showing the rock units in the Topopah Spring welded tuff that could host emplacement drifts. (c) Plan map showing how the lateral boundaries of the potential repository were identified based on the combined constraints of all of the factors. ESF = Exploratory Studies Facility.

A repository would be sited deep enough to protect waste from exposure to the environment and discourage intentional or inadvertent human intrusion into the facility. Designers have specified a minimum overburden thickness of 200 m (650 ft) to ensure adequate protection from surface events.

The host rock for a repository should be able to sustain the excavation of stable openings that can be maintained during repository operations and that will isolate the waste for an extended period after closure. In addition, the rock should be able to absorb any heat generated without undergoing changes that could threaten the site's ability to safely isolate the waste. The host rock should be of sufficient thickness and lateral extent to construct a repository large enough to support the design's intended disposal capacity. Moreover, the amount of suitable host rock should provide adequate flexibility in selecting the depth, configuration, and location of the repository. Studies to date have shown that the Topopah Spring Tuff has these features and characteristics. Experience gained from excavating the Exploratory Studies Facility demonstrates that openings can be excavated and maintained in the unit (Figure 1-12). The dense welding of the tuff originally occurred at temperatures of approximately 800°C (1,500°F); the results of laboratory and underground testing to date show that the heat added by the emplaced waste would



ATP_Z1S1_Fig1-15.ai

Figure 1-12. View Looking Down Exploratory Studies Facility
The photo of the Exploratory Studies Facility at the potential repository level shows that stable openings can be excavated and maintained in the Topopah Spring welded tuff with standard ground support measures. The inset shows a representative sample of lithophysae in the repository host rock. The lithophysae appear as flattened white voids within the rock. The white color is derived from mineral precipitation (primarily silica polymorphs or calcite) in the lithophysae.

not adversely affect the stability of the underground repository (CRWMS M&O 2000e).

Faults could impact repository performance by affecting the stability of underground openings or by acting as pathways for water flow that could decrease waste package lifetimes and eventually lead to radionuclide release. No faults with significant displacement (i.e., movement of more than a few meters) occur within the area defined for emplacement. Detailed studies of the faults within the emplacement area indicate that they are not active; thus, they are considered to have an extremely low probability of being active in the future (CRWMS M&O 2000f, Section 2.1.1.3). The main potential repository emplacement area is bounded on the west by the Solitario Canyon fault, and on the east by the Ghost Dance fault. To mitigate any possible effects from fractures near faults (e.g., higher potential for water flow in fractures or less stable openings), emplacement drifts will be set back from faults.

Because the potential repository is designed to take advantage of the performance characteristics of the unsaturated zone, separation from the saturated zone is an important component in selecting the repository elevation. The repository would be isolated not only from present-day groundwater levels but also from future fluctuations of the water table. Geologic evidence (CRWMS M&O 2000b, Section 9.4) shows that the water table has not been more than about 120 m (390 ft) higher than its present level over the past several million years, even during cooler and wetter climates. Figure 1-13 illustrates a conceptual repository layout that addresses the design and range of operating modes described in this document, superposed on a contour map of the known water table elevations in the vicinity of the potential repository. Details A through D of the figure indicate the elevations of the northernmost emplacement drifts of the upper and lower blocks. The northernmost emplacement drift in the upper block would be approximately 210 m (690 ft) above the present water table elevation at that location; the northernmost emplacement drift in the upper block is the closest emplacement drift in this layout to the present water table. The water table elevation in Borehole WT-24 can be used as a check on this observation. As indicated in Figure 1-13, WT-24 is located approximately 120 m (390 ft) in a northerly direction from the location of Detail A; the elevation of the water table at WT-24 was reported as 840 m (2,750 ft) (CRWMS M&O 2000g, Table 3). The northernmost emplacement drift in the lower block is approximately 265 m (870 ft) above the present elevation of the water table at that location.

Even at the higher levels associated with a water table rise, the emplacement drifts would still be more than about 90 m (290 ft) above the highest projected water table elevation. Analyses of the potential for variation in the elevation of the water table have also considered the possibility that water table variations could be caused by tectonic, volcanic, or hydrothermal processes. In addition to the DOE (CRWMS M&O 2000b, Section 4.4.5), both the National Academy of Sciences/National Research Council (National Research Council 1992) and the NWTRB (1999a, pp. 19 to 21) have reviewed evidence regarding the hypothesis that tectonic or hydrothermal processes could cause large-scale variations in the water table, possibly compromising the performance of the potential repository. Each review found that the available evidence did not support the hypothesis that large-scale fluctuations in the water table (to the level of the potential repository) had occurred in the past. The National Research Council also evaluated the theoretical possibility that the water table could rise significantly in the future and concluded that large variations were unlikely. More recently, the NWTRB found that a review of information developed and presented after the National Research Council review did not significantly affect their conclusions (NWTRB 1999a, p. 20).

The combination of factors described above resulted in the selection of the middle to lower portion of the Topopah Spring welded tuff as the potential repository horizon (Figure 1-11) (BSC 2001d). This section is densely welded, with variable fracture density and lithophysal content. Experience in the Exploratory Studies Facility (e.g., monitoring of excavation characteristics, rock bolt loads, deformation of portal girders, and strain magnitudes of steel sets) and design analyses indicate that stable openings can be constructed in the Topopah Spring Tuff (Figure 1-12). Also, the

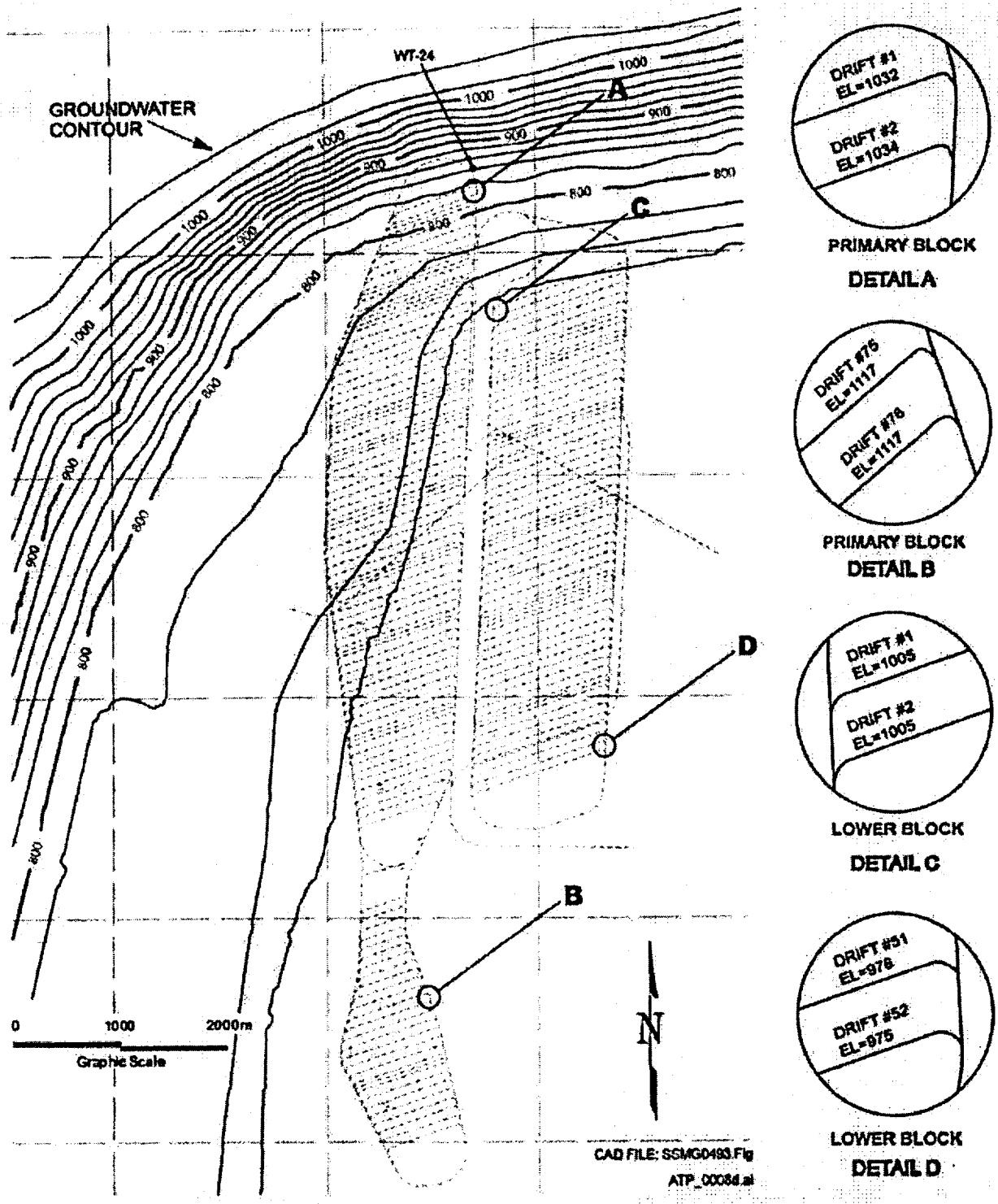


Figure 1-13. Groundwater Elevation Contours, with Their Relationship to a Conceptual Repository Layout
The water table in the vicinity of Yucca Mountain shows a steeper gradient toward the north and northwest. The northernmost emplacement drifts in the primary block area would be closest to the water table, the closest emplacement drift being approximately 210 m (690 ft) above it. The areas where the emplacement drifts in the primary and lower blocks would be closest to (locations A and C, respectively) or farthest from (locations B and D, respectively) the water table have been identified in the figure. EL = elevation. Elevations are given in meters above sea level. Groundwater contours are shown at intervals of 20 m (66 ft). Source: BSC 2001e.

thermal and mechanical properties of the rock should enable it to accommodate the range of temperatures expected during repository construction and operation. The selected horizon is well below the surface and well above the water table. Finally, the potential repository development area is located between major faults, with setbacks to mitigate any potential effects.

Faulting and Local Structural Geology—The distribution and properties of faults and fractures in the volcanic bedrock are important elements of the structural geology of a potential repository at Yucca Mountain. They control the hydrologic and rock-mechanical properties of the system and therefore may affect postclosure performance and design. The distribution and recurrence history of the faults also controls estimates of seismic hazard for the repository. Ground motion from earthquakes is one factor to be considered during the preclosure operation of surface facilities. Studies show that the effects of fault displacement in the repository after closure will not significantly affect performance (see Section 4.3.2.2). The evaluations of seismic hazard, and its potential effects on the preclosure and postclosure performance of the repository, are described in *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada* (Wong and Stepp 1998), Section 3 of the *Disruptive Events Process Model Report* (CRWMS M&O 2000f), and *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f). As described in the *Subsurface Facility System Description Document*, repository emplacement drifts will be set back from known faults (CRWMS M&O 2000h, Section 1.2.2.1.5).

The structural geology of Yucca Mountain is controlled by block-bounding faults spaced 1 to 4 km (0.6 to 2.5 mi) apart. These faults include (from west to east) the Windy Wash, Fatigue Wash, Solitario Canyon, Bow Ridge, and Paintbrush Canyon faults (Figure 1-14). The Dune Wash and Midway Valley faults are also block-bounding faults but differ from the others in that they have no evidence of Quaternary movement (within the past 2 million years). The block-bounding faults commonly dip 50° to 80° to the west, with scattered dips of 40° to 50° and 80° to 90°. Some left-lateral displacement is commonly associated with these faults (Simonds et al. 1995; Day, Dickerson et al. 1998, p. 8).

In some fault zones, several Paintbrush Group rock types have been mixed within the most intensely deformed parts of the fault, indicating that faulting has structurally juxtaposed various subunits as displacement of the bedrock has occurred. This is most apparent in the Solitario Canyon fault system. Individual fault strands within these zones are highly brecciated (i.e., composed of angular, broken fragments of rock).

Displacement between the block-bounding faults occurs along multiple smaller faults, which may intersect block-bounding faults at oblique angles. The Ghost Dance and Sundance faults are examples of smaller "intrablock" faults near the potential repository. The Ghost Dance fault trends in a north–south direction, and can be followed on the surface for 3.7 km (2.3 mi). The fault plane dips steeply to the west (75° to 85°). The displacement and amount of brecciation (the degree to which rocks adjacent to the fault are broken and deformed) varies considerably along its length (Day, Dickerson et al. 1998). The fault zone has a maximum displacement of approximately 27 m (89 ft) down-to-the-west offset. There is no demonstrable Quaternary displacement (movement in the last 2 million years). Mapping in the Exploratory Studies Facility has shown that the zone of brecciation near the fault at the depth of the potential repository is very narrow.

The northwest-trending Sundance fault is the only named fault within the boundaries of the potential repository (Spengler et al. 1994; Potter et al. 1999). It can be traced for approximately 750 m (2,460 ft), and shows no evidence of activity during the past 2 million years. The northeast-side-down vertical displacement across the fault zone does not exceed 11 m (36 ft).

Fracture Characteristics—The distribution and characteristics of fractures at Yucca Mountain are important because in many of the hydrogeologic units at the site, particularly the welded tuffs, fractures are the dominant pathways for water flow in

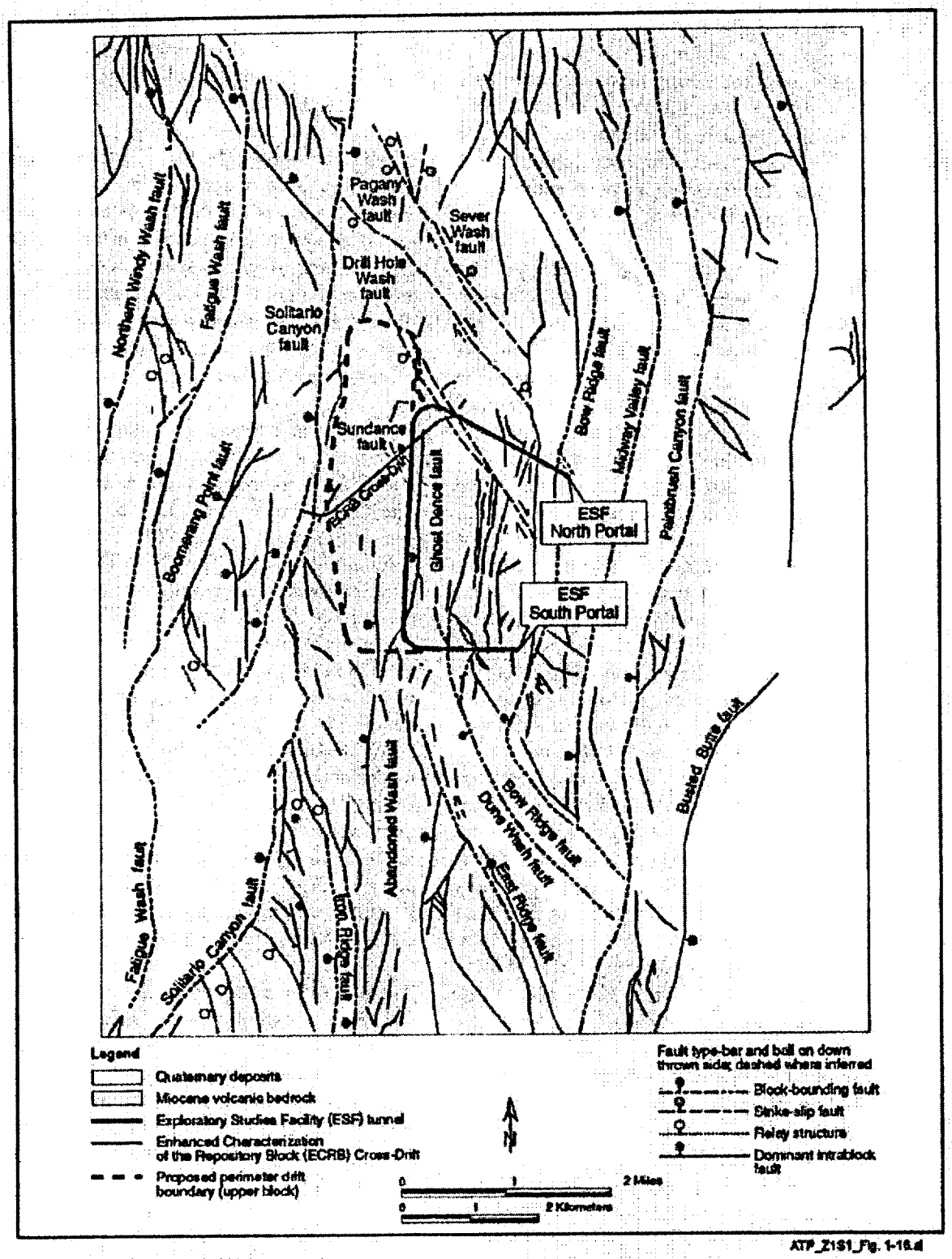


Figure 1-14. Mapped Faults at Yucca Mountain and in the Yucca Mountain Vicinity
This map shows the location of various types of faults, as described in Section 4.6.4 of the *Yucca Mountain Site Description* (CRWMS M&O 2000b).

both the unsaturated and saturated zones. By controlling where, and at what rates, water is likely to flow under various conditions, the fracture systems play a major role in the performance of the repository. The potential repository has been designed to capitalize on the free-draining nature of the repository host rock, which would promote the flow of water past the emplaced waste and limit the amount of possible contact between waste and water. This feature was one of the attributes originally recognized by geologists studying Yucca Mountain (Roseboom 1983).

Fractures at Yucca Mountain are generally of three types: early cooling joints formed during the original cooling of the rock mass, later tectonic joints caused by faulting and rock stress, and joints due to erosional unloading. Cooling and tectonic joints have similar orientations, but cooling joints are smoother. Cooling joints form two orthogonal (at 90° angles to each other) sets of steeply dipping fractures and, in some areas, a set of approximately horizontal fractures. Four steeply dipping sets and one nearly horizontal set of tectonic joints have been identified.

Fracture density (the number of fractures in a given volume of rock), connectivity (the number of fractures that intersect each other), and fracture hydraulic conductivity (the capacity of the rock to transmit water in fractures) are highest in the densely welded tuffs and lowest in the nonwelded tuff units. The Tiva Canyon and Topopah Spring welded units are characterized by well-connected fracture networks, whereas the Paintbrush nonwelded units and the Calico Hills tuffs generally do not exhibit connected fractures. In the lithophysal zones of welded tuffs, the degree of connectivity is intermediate because fractures may end in the void spaces rather than propagate through them. In all the geologic units, fracture density varies both vertically and laterally because of variations in tuff properties.

Fractures related to faults may affect the hydraulic properties near fault zones and provide flow paths through hydrologic units that are otherwise not prone to fracture flow. Even nonwelded units, such as the Pah Canyon and Calico Hills tuffs, may allow water to move in fractured zones adjacent to faults. Based on observations at the surface and in the Exploratory Studies Facility, the zone of influence around faults in which fracture properties are modified may range from approximately 1 to 7 m (3 to 23 ft) (Sweetkind et al. 1997, p. 67). The width of the zone with increased numbers of fractures generally correlates with the amount of movement on the fault (i.e., faults with larger displacements have larger fractured zones). The amount of fracturing also depends on the rock type involved: nonwelded or partially welded tuffs can accommodate a greater amount of fault movement without fracturing than densely welded rocks.

Integrated Site Model—A repository site would support stable excavations that can be safely maintained during the operating life of the facility, and it should have geochemical and hydrologic properties that contribute to waste isolation after closure. The welded tuffs in the unsaturated zone at Yucca Mountain were originally identified as potential candidates for hosting a repository (Roseboom 1983) because of their geologic, hydrologic, and geochemical characteristics. Site characterization has confirmed the basic assumptions about how a repository in the unsaturated zone would likely perform.

The stratigraphic and structural data from the site have been combined with rock property and mineralogical results to build a three-dimensional integrated site model (CRWMS M&O 2000i). Data from boreholes, surface geological mapping, and geophysical surveys form the basis for the conceptual understanding of the geologic framework of the site. This framework was used to develop spatial models of the distribution of geological, geotechnical, hydrologic, mineralogic, and geochemical parameters. The integrated site model thus provides technical input to the design of the potential repository and to the models used to assess its future performance.

1.3.2.2.3 Geomorphology and Erosion

Any potential geologic repository site would be selected and designed so that natural geologic and hydrologic processes do not compromise the integrity of the repository (DOE 1986c, Section 6.3.1.5). Because erosion of the rock overlying the

potential repository could, in theory, threaten waste isolation, an evaluation of erosion rates over time and an assessment of the potential for future erosion have been performed. A wide variety of geologic evidence indicates that erosion at Yucca Mountain has occurred at very slow rates for millions of years and would not adversely affect the waste isolation capability of the site in the future (CRWMS M&O 2000b, Section 7.4).

Geologic evidence indicates that the basic morphology (or shape) of Yucca Mountain was already formed about 10 million years ago. Studies of the tectonic evolution of the area (Day, Dickerson et al. 1998, pp. 17 to 19; CRWMS M&O 2000b, Section 4.6.3.3) demonstrate that most of the faulting occurred shortly before, during, and soon after the eruption of the tuffs that comprise Yucca Mountain. This period of intense tectonic activity began about 16 million years ago with faulting related to the extension of the Basin and Range, followed by eruption of the volcanic units below Yucca Mountain onto the Paleozoic and Precambrian basement. About 12.8 to 12.7 million years ago, the thick tuff units of the Paintbrush Group (including the Topopah Spring and Tiva Canyon tuffs) were erupted from calderas to the north and deposited in approximately horizontal layers. Movement along block-bounding faults then tilted the volcanic rocks 10° to 20° to the east and formed major topographic features, such as Solitario Canyon and Midway Valley. The Rainier Mesa Tuff of the Timber Mountain Group was erupted about 11.6 million years ago onto an irregular land surface that was already similar to modern Yucca Mountain (i.e., major ridges and valleys created by faulting already existed). Faulting continued after deposition of the Rainier Mesa Tuff but at greatly reduced rates: displacement on the block-bounding faults was up to several hundred meters before 11.6 million years but has been only a few meters in the last 10 million years.

Over the past several million years, erosion in the Yucca Mountain region has been slow, with only minor effects on major landforms. Several lines of evidence have been considered in the analysis of potential erosion at or near Yucca Mountain. These include evidence related to:

- The rate of degradation of the slopes and stream channels on Yucca Mountain at the potential repository site
- The rate of erosion in stream channels in the Yucca Mountain region, such as Fortymile Wash (to the west of Yucca Mountain)
- The rate of degradation of landforms in the Yucca Mountain region, such as the Lathrop Wells cinder cone (approximately 16 km [10 mi] south of the potential repository site).

Several techniques have been used to assess the stability of the slopes and stream channels at Yucca Mountain. By studying the geochemical and isotopic characteristics of the surfaces of boulders exposed on hillslopes, it is possible to determine how long they have been exposed to the sun and the atmosphere and to calculate erosion rates. Also, by studying the age and composition of sediments in the stream channels, it is possible to analyze how fast sediments have accumulated or eroded in various streams. The studies indicate that long-term average erosion rates at and near Yucca Mountain are low (CRWMS M&O 2000b, Section 7.4.2.2). Rates of bedrock and hillslope erosion range from less than 0.1 cm to 0.5 cm (0.04 to 0.24 in.) per thousand years in the Yucca Mountain area. These low erosion rates are consistent with the surface exposure ages of hillslope boulder deposits found near Yucca Mountain, which range in age from several hundred thousand to over a million years. The boulders have thick deposits of rock varnish and show no signs of significant movement by hillslope erosion since they were formed.

Stream incision rates have also been calculated for the Yucca Mountain area. In local areas, some downcutting in stream channels can be observed. However, regional studies indicate that Fortymile Wash has apparently reached a state of near equilibrium (CRWMS M&O 2000b, Section 7.4.2.5). The rates of sediment accumulation and erosion are approximately equal, with little aggradation (sediment accumulation) or degradation (erosion) at present and with the entire Fortymile drainage system adjusted to the base level of the main channel. This system-wide, long-term equilibrium

in the Fortymile drainage system indicates that episodic pulses of erosion are unlikely to significantly incise stream channels on Yucca Mountain.

Bedrock channel incision rates have also been evaluated at Yucca Mountain and in Fortymile Canyon. Several small canyons on Yucca Mountain are cut 60 to 100 m (200 to 330 ft) into 12.7-million-year-old volcanic tuff. Thus, the long-term incision rate for the first-order streams is 0.8 cm (0.3 in.) per thousand years or less. The drainage system of Fortymile Wash and its tributaries was established more than 10 million years ago and has changed little in basic plan since then.

Studies of the Lathrop Wells cinder cone at the south end of Yucca Mountain indicate that only minor degradation has occurred over approximately the past 80,000 years (Wells et al. 1990). The lack of degradation indicates that severe erosional processes have not occurred in the Yucca Mountain area, even during a time period that includes substantial climate change from the last glacial period.

1.3.2.2.4 Natural Resource Potential

A repository must isolate the spent nuclear fuel and high-level radioactive waste it contains from both people and the environment. In order to reduce the chance that future individuals or groups might inadvertently encounter waste while searching for other exploitable resources, sites with high potential for natural resources have typically been excluded from consideration. As part of the characterization of the Yucca Mountain site, therefore, the potential for economically valuable resources has been carefully evaluated. This section summarizes the results of that assessment.

Resource potential is difficult to predict because it depends on many factors, including economics (i.e., supply, demand, and cost of production), the potential discovery of new uses for resources, and the discovery of synthetic materials to replace natural resources. Therefore, this evaluation is based on the present-day use and economic value of resources; it does not predict future market trends or undiscovered uses for resources. All common types of natural resources—including metallic minerals, industrial rocks and minerals, hydrocarbons (i.e, petroleum, natural gas, oil shale, tar sands, and coal), and geothermal energy—have been considered.

In a general sense, Nevada contains abundant resources, ranking second in the U.S. in the value of nonfuel (i.e., excluding oil, gas, coal, and geothermal) mineral production in 1996 (Nevada Bureau of Mines and Geology 1997, Summary). Nevada leads the nation in the production of gold, silver, mercury, and barite. Additional metals, including copper, lead, zinc, iron, and such industrial materials as brucite, magnesite, clays, gemstones, gypsum, sand, gravel, and crushed stone are being or have been produced in Nevada. Small but economic oil deposits occur in Railroad Valley in east-central Nevada, and geothermal resources occur in California and northern Nevada within the Great Basin.

Although economic gold mineralization is present in the region (most notably near Beatty), Yucca Mountain contains no identified metallic mineral or uranium resources (CRWMS M&O 2000b, Section 4.9). On the basis of detailed studies of geology, geochemistry, mineralogy, mineral alteration, and geophysical data and remote sensing, the Yucca Mountain site is considered to have little or no potential for deposits of metallic minerals or uranium resources that could be mined economically now or in the foreseeable future. Geological and geochemical comparisons between Yucca Mountain and metal mining districts in the region indicate substantial differences in the geologic and geochemical patterns observed for precious metals, base metals, and pathfinder elements (Castor et al. 1999, Section 6).

Many industrial rock and mineral commodities occur in the Great Basin but not at Yucca Mountain. Although barite, clay minerals, fluorite, limestone, perlite, and zeolites have been identified in samples from Yucca Mountain, these occurrences are minor and at depths too great to be mined economically (Castor and Lock 1995, Section 7).

It is possible that alluvial deposits at the site could be used as concrete aggregate for local construc-

tion and that some of the Tertiary tuff could be used as building stone. However, neither the alluvial deposits nor the tuff have any properties or features that would make them more marketable than other deposits readily available and closer to processing plants and end users (Castor and Lock 1995, Section 6.4.3.1).

Nevada is not a large producer of oil or natural gas, although a few producing fields exist north of the Yucca Mountain region. Some of the conditions of source, reservoir, trap, and seal that characterize petroleum accumulations of the Great Basin are present to some degree in the Yucca Mountain area (French 2000, p. 39). Most evidence, however, indicates that the accumulation of oil or natural gas near the potential repository site is unlikely.

It is extremely unlikely that tar sands, oil shale, or coal occur as economic resources at the Yucca Mountain site. If any of these resources were associated with Paleozoic marine rocks underlying the Tertiary volcanic rocks at Yucca Mountain, they would occur at depths greater than 1,800 m (5,900 ft). Extraction at these depths would not be economically feasible in the foreseeable future.

There are no geothermal discoveries near Yucca Mountain, and there are no potential users located at or near the site. Chemical analyses of fluids throughout the area indicate that most waters are nonthermal in origin. Geophysical data, including gravity, magnetic, seismic, and heat flow data, failed to delineate any systematic structural evidence for a thermal anomaly. Compared with the physical attributes of geothermal systems that have developed in other parts of the Great Basin, no economically viable resources were identified within the Yucca Mountain area (Flynn et al. 1996).

1.4 POSTCLOSURE PERFORMANCE

Since the National Academy of Sciences concluded that geologic disposal was feasible in 1957 (National Academy of Sciences Committee on Waste Disposal 1957), many scientists (e.g., de Marsily et al. 1977; Konikow and Ewing 1999) and reviewers of the U.S. repository program (NWTRB 1999b; NWTRB 2000; Budnitz et al. 1999) have recognized the difficulty associated with assessing repository performance over the long time frames necessary to protect public health and safety.

Developing confidence in the long-term performance of a geologic repository is one of the greatest challenges faced by the DOE. Because of the uncertainty associated with assessments of performance for 10,000 years, there is no simple way to guarantee that the facility will function as modeled throughout the period of performance. In NRC's licensing rule, 10 CFR Part 63 (66 FR 55732), the NRC recognizes this irreducible uncertainty and clearly states that "proof" of performance cannot be produced in the ordinary sense of the word. Rather, the NRC, like the EPA, would require in licensing a "reasonable expectation" that the postclosure performance standards will be met.

The DOE's approach to developing confidence in the safety of geologic disposal, known as the postclosure safety case, relies on multiple, independent lines of evidence. The first element of the safety case is a thorough and quantitative evaluation of the future performance of the repository, based on a comprehensive testing program that has evolved to address identified uncertainties and a repository design developed to complement the natural setting of the site. EPA and NRC regulations specify the method by which the DOE will analyze and demonstrate in licensing that a repository can safely isolate spent nuclear fuel and high-level radioactive waste (i.e., a total system performance assessment [TSPA]). The process used to develop the total system performance assessment for site recommendation (TSPA-SR), which is described briefly in Section 4.4 and in more detail in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a), includes analyses of all the processes expected to operate at the repository that could affect its ability to isolate waste. The supplemental TSPA described in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b) and the revised supplemental TSPA models described in *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final*

Environmental Impact Statement and Site Suitability Evaluation (Williams 2001a) and *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b) all use the same TSPA approach and method. All of these models and analyses also explicitly consider both disruptive events and alternative models that could result in unanticipated behavior (i.e., identify what could go wrong). The evaluation directly addresses uncertainty in both the DOE's knowledge of the site and in future conditions, and it includes numerical sensitivity analyses to test how the repository might perform if current or future conditions differ from those expected.

Because the DOE recognizes that uncertainty about the future performance of the repository cannot be completely quantified or eliminated (i.e., models alone cannot capture all the uncertainties in natural systems), the safety case includes several additional measures designed to provide confidence and assurance that the repository will meet applicable radiation protection standards after it is permanently closed. These measures include:

- Studies of natural and man-made analogues to the repository or to processes that may affect repository performance, which can further the understanding of natural processes related to repository performance that operate over long time frames (thousands of years) or large spatial distances (tens of kilometers) that cannot easily be tested.

- Selection and design of a repository system that provides defense in depth and a margin of safety compared to health and safety requirements. The DOE has implemented this approach through the characterization of the Yucca Mountain site and the development of a repository system with multiple natural and engineered barriers to the migration of radionuclides. The engineered components of the system are designed specifically to complement the natural attributes of the site. This multiple barrier repository system provides defense in depth, so that the safety of the repository does not depend on only one or two barriers. Also, investigations of performance over a range of thermal operating modes provides insight into the best ways to address the treatment of uncertainty inherent in thermally coupled processes.

- Long-term management and monitoring (a performance confirmation program) to ensure the integrity and security of the repository and to ensure that the scientific and engineering bases for the disposal decision are well founded. The NRC regulation, 10 CFR Part 63 (66 FR 55732) establishes a period of 50 years after the start of emplacement in which retrieval must be possible, unless a different period is specified by the NRC. DOE design requirements provide for an extended monitoring period of up to 300 years to provide the ability to retrieve the spent nuclear fuel and high-level radioactive waste for any reason before closure, if future generations decide that doing so is appropriate.

This approach is similar to that recommended by many national and international organizations that have investigated the technical and social problems of nuclear waste disposal. As a panel of the National Academy of Sciences observed, "Confidence in the disposal techniques must come from a combination of remoteness, engineering design, mathematical modeling, performance assessment, natural analogues, and the possibility of remedial action in the event of unforeseen events" (National Research Council 1990, pp. 5 to 6). Such an approach has also been recommended and adopted by most nations with nuclear waste programs (see, for example, *Confidence in the Long-Term Safety of Deep Geological Repositories—Its Development and Communication* [NEA 1999a]). The postclosure safety case is described in more detail in Section 4.1. The rest of this section briefly describes how the DOE has used a performance-assessment-based approach to deal with uncertainty, to guide testing and repository design programs, and to evaluate the likely performance of a repository at the Yucca Mountain site. Taken in total, the approach provides a strong engineering and scientific basis supporting the DOE's evaluation of the performance of the potential Yucca Mountain repository system.

1.4.1 Performance Assessment

As noted above, the regulatory requirements for a potential repository are based on quantitative assessments of the system's performance. For Yucca Mountain, performance assessment provides not only a means for estimating relative performance but also a framework for organizing and describing the site and the repository design. Performance assessment has been used as a management tool to integrate the scientific and engineering programs and to assess the importance and priority of various program activities, consistent with the overall goal of determining whether Yucca Mountain can safely host a repository facility.

Performance assessment is a systematic method for evaluating repository system behavior over an extended time. Analysts build detailed mathematical models of the features, events, and processes that could affect performance. Then they incorporate the results of these detailed models into an overall model of the repository system, called the "total system performance assessment model." This integrated model is used to assess how the natural and engineered elements of a waste disposal system would work together over the long period required to isolate wastes. Sections 4.2, 4.3, and 4.4 present the results of these analyses, which are more fully described in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a). Supplemental analyses are described in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b), and *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation* (Williams 2001a), and *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b).

Performance assessment models are probabilistic (i.e., they consider the likelihood that the system will behave in a certain way) because of the nature of the processes and systems being analyzed. The natural system is heterogeneous both in space and time; the processes simulated in the models are also variable in space and time. For these reasons, performance assessment uses a probabilistic approach that directly incorporates evaluations of the variability of site properties and a range of possible process behaviors that could occur into the estimates of the future performance of the repository. Performance assessments provide one means to identify which uncertainties about the behavior of a disposal system are significant and which are not, and which elements of the repository design are most important to performance. This helps focus efforts to improve the design and the defensibility of performance analyses. Performance assessments are refined iteratively during the course of developing, evaluating, and improving a repository design.

The DOE has also conducted analyses of potential barriers to radionuclide migration to identify and evaluate the performance contribution of natural features of the geologic setting and design features of the engineered barrier system (see Section 4.5). At Yucca Mountain, these multiple barriers may contribute to confidence in the performance of the repository by providing defense in depth: several elements of the natural and engineered systems contribute to the isolation of waste by functioning independently to limit possible releases.

Section 4.4 presents a summary of the results of the performance assessment calculations performed for the Yucca Mountain site and found in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a) and subsequent analyses (BSC 2001a; BSC 2001b; Williams 2001a; Williams 2001b). The TSPA-SR results show that for a 10,000-year period, the calculated dose in the nominal scenario is zero (CRWMS M&O 2000a). A supplemental TSPA model yielded a calculated dose of approximately 2×10^{-4} mrem/yr (BSC 2001b, Section 5.1). A revised supplemental TSPA model calculates a dose of 1.7×10^{-5} mrem/yr (Williams 2001a). The small doses in the supplemental TSPA model result from the inclusion of a few early failures of waste packages due to improper heat treatment. The supplemental TSPA model projected a peak mean dose of 2×10^{-4} mrem/yr for the higher-temperature operating mode and 6×10^{-5} mrem/yr for the lower-temperature operating mode over a 10,000-

year period. Over this same period, the revised supplemental TSPA model projected a peak mean dose to the reasonably maximally exposed individual of 1.7×10^{-5} mrem/yr for the higher-temperature operating mode and 1.1×10^{-5} mrem/yr for the lower-temperature operating mode for the nominal scenario (BSC 2001b, Section 4.1.3; Williams 2001a, Section 6, Table 6).

1.4.2 Importance of Repository System Components to Long-Term Performance

The various components of the repository system contribute to performance in different ways. Certain components of the system are more important to performance than others, as shown by performance assessment sensitivity studies (e.g., Section 4.4.5). Considerations of safety margin, defense in depth, insights from analogues, and expert judgments also help identify the importance of repository components to performance (see Section 4.1). Knowledge of how components of the system affect performance can provide insights on how design or operating mode features could be developed in a manner that could contribute to long-term performance or mitigate potentially adverse conditions. Ongoing evaluations over the range of operating modes could lead to enhanced understanding of how the repository system components could contribute to long-term performance.

1.4.3 Addressing Uncertainty in Total System Performance Assessment

Even though the Yucca Mountain site represents a fairly simple geologic/hydrologic system conceptually—a relatively dry site consisting of fractured volcanic rock hundreds of meters above the water table—it has complexities that are difficult to model but could be important over long periods of time. Uncertainties exist in the understanding of both natural processes (e.g., infiltration of water at the surface, percolation in fractures and rock matrix, disruptive processes of earthquakes and volcanism) and the ways in which the engineered system will perform when exposed to the environment (e.g., seepage into drifts, thermal processes related to the heat generated by the waste, corrosion of waste packages).

Capturing those uncertainties and understanding their impacts is critical to understanding how a repository might behave in the future. Accommodating uncertainties in the assessment of the performance of a potential repository at Yucca Mountain means recognizing that uncertainties exist and explicitly identifying those that may be important to performance. The DOE's approach to dealing with uncertainties is described in Sections 4.1.1.2 and 4.4.1.2. Some, but not all, of those uncertainties can be quantified; that is, scientists can and have collected data that allow them to estimate the probability that a variable will assume different values over the spatial and temporal scales of an operating repository. Those uncertainties that can be quantified can be incorporated directly into performance assessment results. Approaches to address design-related uncertainty concerns through consideration of a range of thermal operating modes and the effects on performance across the range of temperatures are described in Sections 2.1.5 and 4.4.5.1.2.

Studies have been performed to enhance the understanding of the environmental conditions associated with the lower-temperature ranges of the operating modes. Supplemental performance assessment analyses and sensitivity studies have been conducted to evaluate the performance of lower-temperature operating modes. The analyses incorporate the results of other efforts to quantify uncertainties and extend the applicable range of the process models. Of particular interest are analyses that address performance-related responses of the design and operating mode, considering temperature-sensitive parameters and coupled thermal-mechanical-chemical-hydrologic processes. This approach is intended to ensure that the performance evaluations appropriately consider the potentially detrimental and potentially beneficial aspects of the repository's performance over a range of operating modes encompassing temperatures above and below the boiling point of water. Results of these evaluations are described in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b).

There are also uncertainties that cannot readily be quantified (e.g., the possibility that the models used for compliance analyses do not include, or accurately simulate, processes that may be important to performance). The DOE has made a substantial effort to identify, characterize, and mitigate the potential impacts of these unquantified uncertainties. They have been identified and characterized through detailed consideration of the features, events, and processes that might affect repository performance. The principal mechanism for mitigating unquantified uncertainties is the multiple lines of evidence provided by the postclosure safety case, as described in Section 4.1. In particular, design (safety) margin and defense in depth provide a degree of confidence independent of the results of TSPA analyses.

1.4.4 Time Frame for Performance Analyses

EPA standards and NRC licensing regulations relevant to the Yucca Mountain disposal system contain postclosure performance standards that would apply during the first 10,000 years after repository closure.

For this reason, performance assessment results have been presented on plots that extend for 10,000 years, as shown in Section 4.4. The EPA's 40 CFR 197.35 and NRC's 10 CFR 63.341 (66 FR 55732) also specify that the DOE should calculate the peak dose that would occur after 10,000 years but within the period of geologic stability, and present the results in the EIS. Although no regulatory standard applies to the results of this analysis, the EPA and NRC noted that the peak dose calculations would complement the 10,000-year performance assessment results as an indicator of long-term performance. These analyses, which have been performed out to 100,000 years and 1 million years (see Sections 4.4.2.2 and 4.4.2.4, respectively), provide additional confidence in the 10,000-year results.

2. DESCRIPTION OF THE POTENTIAL REPOSITORY

Section 114(a)(1)(A) of the Nuclear Waste Policy Act of 1982 (NWPA), as amended (42 U.S.C. 10134(a)(1)(A)), requires "a description of the proposed repository, including preliminary engineering specifications for the facility." Refining the design and the choice of operating mode for the potential repository is an ongoing, iterative process involving scientists, engineers, and decision-makers. The goal of this process is to develop a design that works with the natural system to enhance containment and isolation of spent nuclear fuel and high-level radioactive waste.

The U.S. Department of Energy (DOE) completed a major milestone in December 1998 with the publication of *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998). That report included a repository design referred to as the Viability Assessment (VA) design. Based on an improved understanding of the interactions of potential repository features with the natural environment and the addition of engineered features for enhanced waste containment and isolation, the VA design has evolved into the design described in this report. As the program moves forward, the design will continue to evolve.

Figure 2-1 depicts a cutaway view of a proposed layout for the repository facilities. Construction of the emplacement drifts and subsurface facilities would be accomplished in phases. In the current plan, about 10 percent of the emplacement drifts

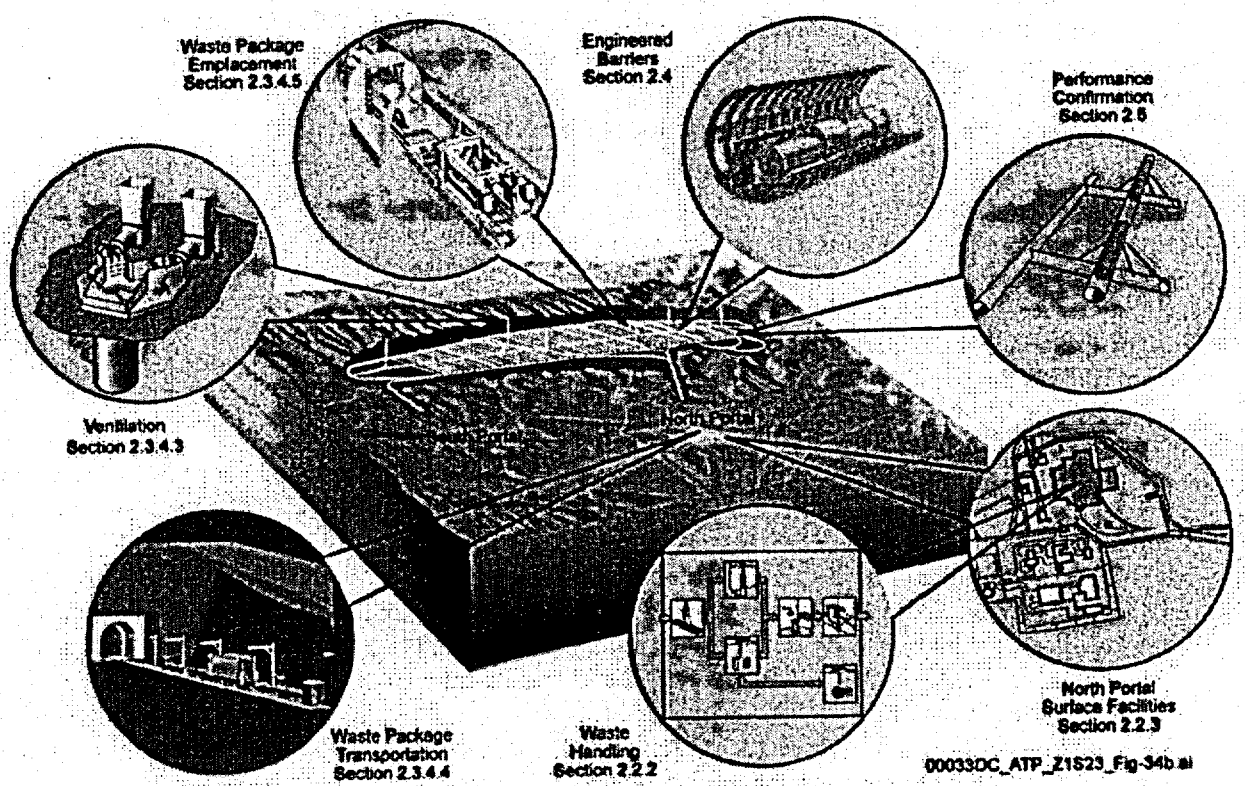


Figure 2-1. Proposed Monitored Geologic Repository Facilities at Yucca Mountain
This figure shows the proposed surface and subsurface facilities for the repository. Waste handling and packaging would take place at the surface facilities, from which the sealed waste packages would be transported to the underground emplacement drifts via the north ramp. The repository subsurface area is divided in two zones, the development side and the emplacement side, each zone having separate ventilation systems. The South Portal would be used mainly for access to the development side of the repository and for transportation of construction materials. The south ramp would also be used as the conveyor route for removal of excavation waste, or muck.

would be completed during the initial construction phase (prior to initiation of waste emplacement), with the remainder of the emplacement drifts being completed during the operation phases. This phased construction would allow the DOE flexibility to develop the repository based on future deliveries of spent nuclear fuel.

The potential repository facilities have been designed to be fully integrated, using a systems engineering approach to identify and then fulfill requirements by providing adequate design solutions. This systems engineering approach to design and development is explained in Section 2.1.1. An important aspect of the design solutions is that they provide the flexibility to accommodate developing operational scenarios, including associated thermal environment characteristics. They provide a basis to refine the design as it evolves in response to increased understanding of the performance of its components. They can also accommodate unanticipated underground conditions that may be encountered during construction with minimal interruptions and no need for expensive retrofits.

The description of the potential geologic repository at Yucca Mountain is organized into five parts. Section 2.1 presents a general overview of the engineering and design process common to all design disciplines in the Yucca Mountain project. It also discusses the evolution of the design, design and operating flexibility, and an assessment of a broader range of thermal operating modes. Design descriptions for the surface and subsurface facilities are covered in Sections 2.2 and 2.3, respectively. Section 2.4 describes the emplacement drift design features that are part of the engineered barriers. Although the definition of engineered barriers includes the waste package, the descriptions of waste package designs and the different waste forms that would be contained within waste packages are provided separately in Section 3. Section 2.5 describes the surface and subsurface facilities that support the performance confirmation program for the potential repository.

The design and operating mode that was analyzed to assess long-term performance of the repository system presented in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a) was based on a higher-temperature operating mode. Subsequent analyses that considered the performance of lower-temperature operating modes are described in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b).

2.1 ENGINEERING AND DESIGN ANALYSIS

Design development for the potential repository follows a structured approach that links statutory, regulatory, and derived requirements to the final design products. Design work has been performed in accordance with a quality assurance program (DOE 2000a). This program has been reviewed and accepted by the U.S. Nuclear Regulatory Commission (NRC). Figures 2-2, 2-3, and 2-4 discussed in the following section, illustrate the steps in the design process.

2.1.1 Design Process

This section describes the design process, including how requirements are identified and passed down to individual systems through an established document hierarchical system and how systems are analyzed and then classified according to their importance to preclosure radiological safety.

2.1.1.1 Allocation of Yucca Mountain Site Characterization Project Requirements

The framework for the design of the major repository structures, systems, and components is consistent with the following:

- The Nuclear Waste Policy Act of 1982 (42 U.S.C. 10101 et seq.)
- 10 CFR Part 63 (Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, NV) (66 FR 55732)

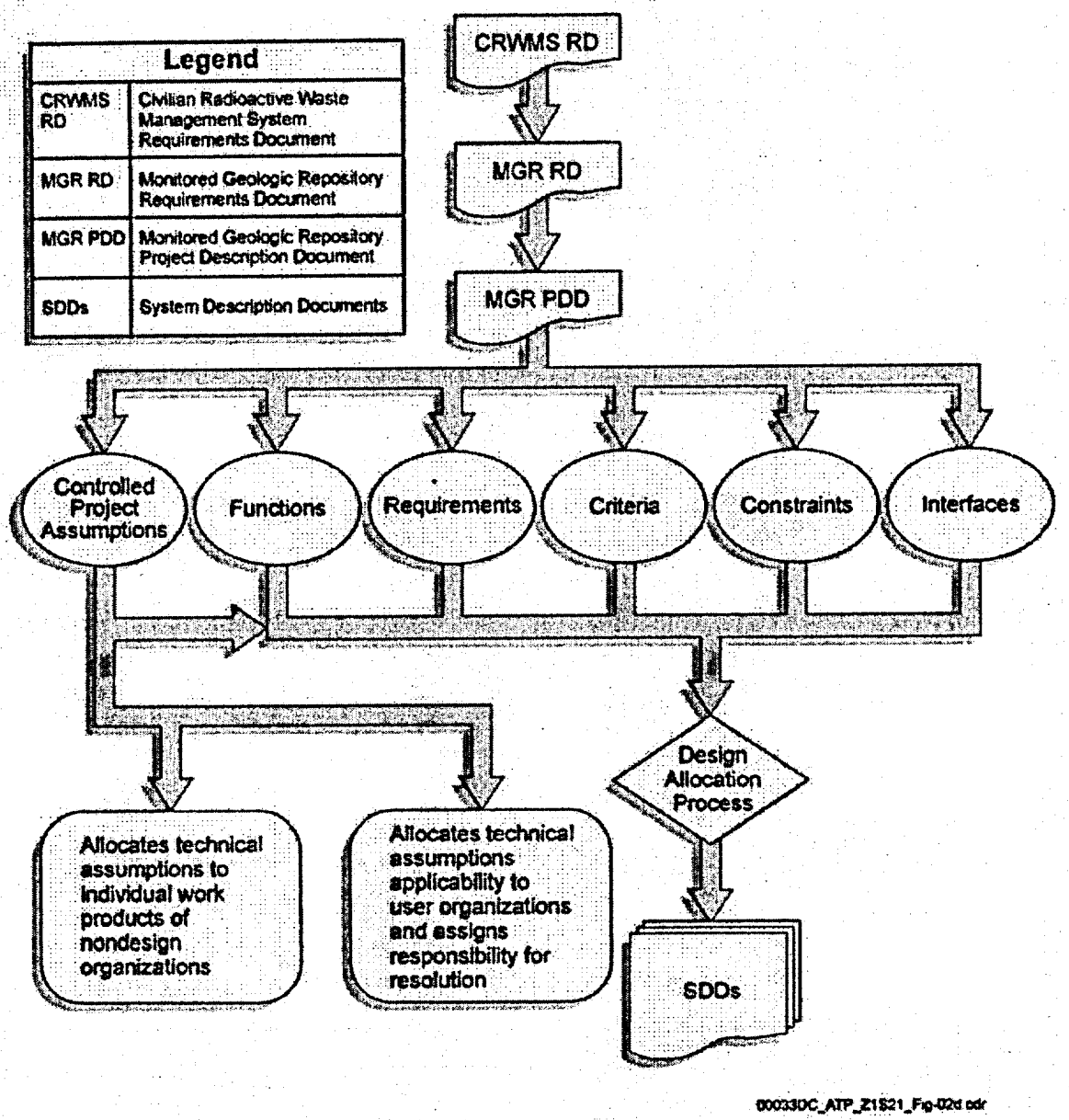


Figure 2-2. Allocation of Functions, Criteria, and Requirements
Design activities for the repository follow a hierarchical system of allocation of functions from the high-level requirements documents to the individual systems. The functions, requirements, and criteria for each individual system are described in the System Description Document for that system. Source: Curry 2001, Section 1.7.

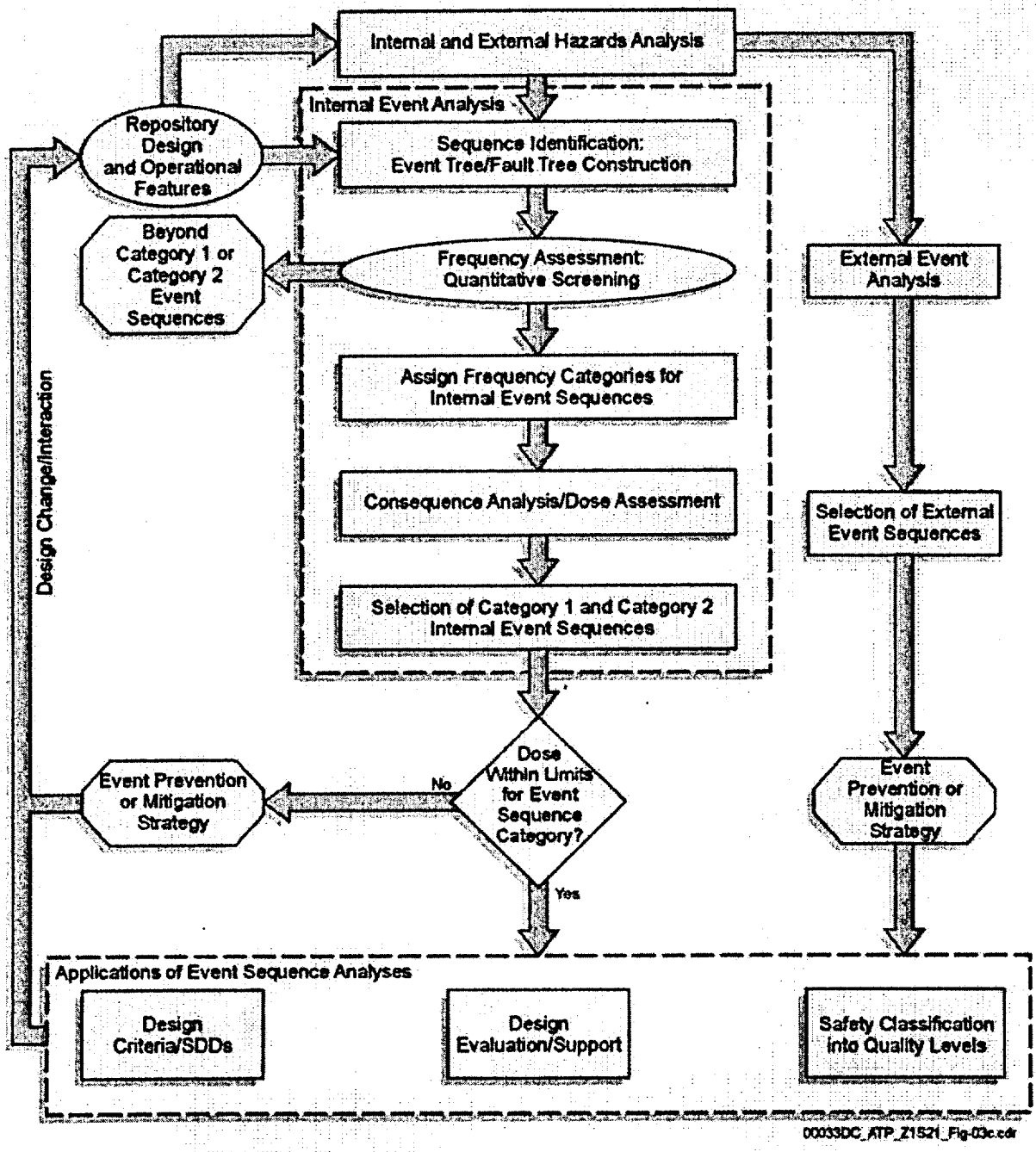


Figure 2-3. Preclosure Safety Analysis Process
Each repository structure, system, and component is subjected to a preclosure safety analysis process that identifies events that are important to radiological safety. The analysis determines the probability of those events, whether they would result in a release of radiation, and the potential consequence of events that would. Structures, systems, and components are then classified into quality levels. SDD = System Description Document. Source: BSC 2001f, Section 5.2.

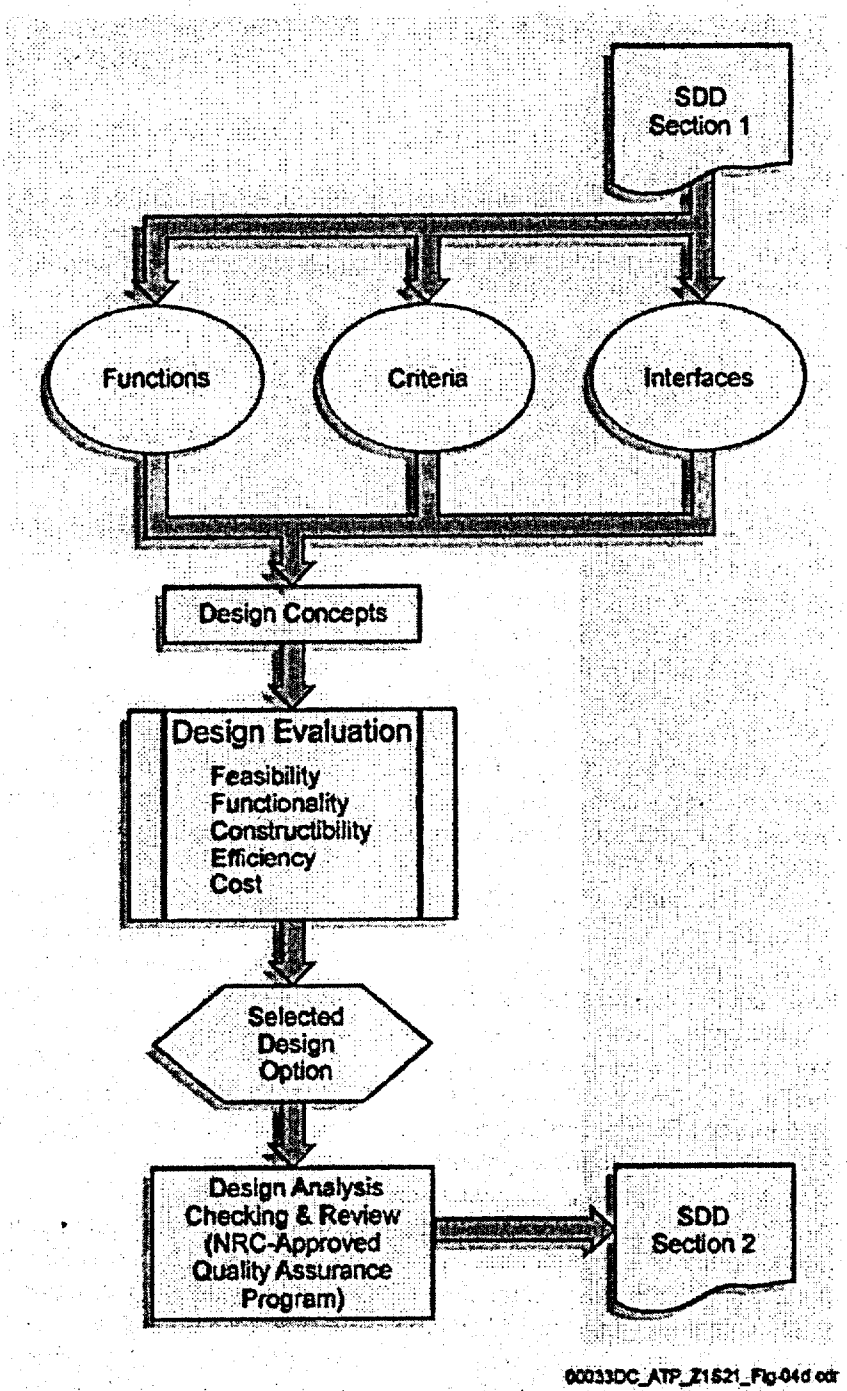


Figure 2-4. Design Documents Development
This figure illustrates the relationship between the system description document and the design analysis, as described in the *Monitored Geologic Repository Project Description Document* (Curry 2001). There is a somewhat parallel development of these two activities, with Section 1 of the System Description Document dictating design requirements to the design analysis. Once the design analysis is completed, the results are rolled into Section 2 of the System Description Document as a design description and requirements compliance section. At the end of the process, the System Description Document contains a complete set of design information for its system. SDD = System Description Document.

- 10 CFR Part 20 (Energy: Standards for Protection Against Radiation)
- 10 CFR Part 73 (Energy: Physical Protection of Plants and Materials)
- 10 CFR Part 71 (Energy: Packaging and Transportation of Radioactive Material)
- 40 CFR Part 197 (Protection of Environment: Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada).

The DOE has published a comprehensive hierarchy of design documents for a monitored geologic repository. The primary document is the *Civilian Radioactive Waste Management System Requirements Document* (DOE 2000b). Requirements in that document applicable to the repository site are allocated to the *Monitored Geologic Repository Requirements Document* (YMP 2000a). The hierarchy then branches into specific areas of scope, becoming more detailed with each level of document, from the *Monitored Geologic Repository Project Description Document* (Curry 2001) to a set of System Description Documents. The *Monitored Geologic Repository Requirements Document* (YMP 2000a) captures top-level functions and requirements. *Monitored Geologic Repository Project Description Document* (Curry 2001) documents the functions, requirements, criteria, and assumptions for a potential repository while allocating each to the appropriate systems, as detailed in the System Description Documents. For instance, the guidance for blending spent nuclear fuel assemblies to achieve a maximum thermal output of 11.8 kW per waste package at the time of emplacement is contained in the *Monitored Geologic Repository Project Description Document* (Curry 2001). This document also includes controlled assumptions and captures performance criteria and design constraints. In this way, it supplements the higher-level approach of the *Monitored Geologic Repository Requirements Document* (YMP 2000a). Figure 2-2 illustrates this transfer of requirements from higher-level to lower-level documents, as well as the allocation of functions, requirements, and criteria by the *Monitored Geologic Repository Project Description Document* (Curry 2001) to individual systems.

The requirements documents also include references to codes and standards applicable to specific structures, systems, and components. These codes and standards are generally developed by professional organizations, such as the American Society of Mechanical Engineers or the American Nuclear Society, in cooperation with the American National Standards Institute.

2.1.1.2 Safety Classification of Structures, Systems, and Components

The design components of the repository system would contribute to performance in varying degrees. Certain design components of the system are more important to safety of preclosure operations, as described in *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f). Other design components are more important to postclosure performance, as shown by performance assessment sensitivity studies (e.g., Section 4.4.5). Considerations of safety margin and defense in depth, insights from analogues, and expert judgments also help identify the importance of repository components to performance (see Section 4.1). Knowledge of how components of the system affect performance can provide insights on how design or operating mode features could be developed in a manner that could contribute to long-term performance or mitigate potentially adverse conditions.

The definitions of design criteria and requirements are influenced by a consideration of the importance of each system, its structures, and its components in the overall safety strategy for the potential repository. That safety strategy has been developed over the years based on determinations of critical factors in design with respect to preclosure safety and postclosure performance. The structures, systems, and components important to preclosure safety are identified through engineering analyses and relate directly to the health and safety of facility workers, the health and safety of the public, and the environment. The factors important to postclosure performance are determined through

evaluations of the importance of the components to overall system performance. These evaluations include total system performance assessment (TSPA), considerations of safety margin and defense in depth, and independent, multiple lines of evidence. These evaluations integrate the performance of natural barriers (i.e., the geologic environment) and man-made barriers (e.g., the drip shield and waste package outer barrier) with respect to their complementary attributes in containing and isolating radioactive waste over long periods of time.

For the preclosure period, the importance of design features is defined in terms of their role in preventing or controlling radiological exposure to repository workers and the public. The design of these features must address the health and safety requirements associated with radiological work. The more important a design feature is to ensuring radiological safety, the more process controls are imposed on that design. To the maximum extent practicable, the design of potential repository structures, systems, and components has been developed from the design of structures, systems, and components already in use at other licensed nuclear facilities. The standards used in the designs of such facilities are well developed.

Following the process summarized in Figure 2-3, structures, systems, and components are classified to define their importance to preclosure safety. Event sequences form the basis of these safety classifications. This process is integrated in the sense that all structures, systems, and components important to safety are analyzed for event sequences that represent a complete set of bounding conditions. This set of events results from scenario analysis and grouping of the internal and external hazards. The preclosure safety assessment integrates safety evaluations through a joint consideration of safety measures that otherwise might conflict, including but not limited to integration of fire protection, radiation safety, criticality safety, and chemical safety measures. Event sequences include natural or human-induced events that are reasonably likely to occur and that could lead to exposure of individuals to radiation. Event sequences also include other natural or human-induced events that are unlikely but sufficiently probable to warrant consideration (i.e., that have at least 1 chance in 10,000 of occurring before permanent closure of the repository).

Event sequences could be internal (e.g., collision, loss of power, or fire) or external (e.g., earthquake, tornado, or flood). A potential bounding event sequence (i.e., the one resulting in the worst failure) is identified for analysis of a particular structure, system, or component. The frequency of occurrence of that event sequence determines its event credibility (probability) and category. An engineering analysis is then performed to determine whether a radiological release could result from the failure of a structure, system, or component because of a credible event. These analyses are performed with mathematical and analytical models of processes and events. The consequences of the release are evaluated to determine whether the resulting doses are within regulatory limits and, if not, what preventive or mitigating measures are required to bring the radiological consequences within compliance limits. Structures, systems, and components required to meet regulatory limits are classified as important to safety. Once the safety classifications are defined, individual systems are ranked based on the importance to safety of the performance of their structures and components. The process depicted in Figure 2-3 is iterative in nature and results in a safety analysis that is integrally tied to the facility design (BSC 2001f, Section 5.2).

This report focuses on the most important systems, those that could adversely affect worker and public safety if they failed. To this end, potential systems have been evaluated for their importance to safety and classified into four groups of quality levels, as defined in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Section 4.4.1):

- **Quality Level (QL)-1:** Those structures, systems, and components whose failure could directly result in a condition adversely affecting public safety. These items have a high safety or waste isolation significance.

- **QL-2:** Those structures, systems, and components whose failure or malfunction could

indirectly result in a condition adversely affecting public safety, or whose direct failure would result in consequences in excess of normal operational limits. These items have a low safety or waste isolation significance.

- **QL-3:** Those structures, systems, and components whose failure or malfunction would not significantly impact public or worker safety, including those defense in depth design features intended to keep possible radiation doses as low as is reasonably achievable (ALARA). These items have a minor impact on public and worker safety and waste isolation.

- **Conventional Quality (CQ):** Those structures, systems, and components not meeting any of the criteria for QL-1, QL-2, or QL-3. CQ items are not subject to the requirements of *Quality Assurance Requirements and Description* (DOE 2000a).

The preclosure safety analysis process is shown in Figure 2-3. Event sequences are classified as Category 1 or Category 2, based on the frequency of the entire event sequence (also known as the scenario frequency). The frequency ranges for each event sequence category, given in Table 2-1, correlate with the probability-based definitions from 10 CFR 63.2 (66 FR 55732), assuming for the purpose of analysis a period of 100 years before repository closure (BSC 2001f, Section 5.2).

2.1.1.3 System Description Documents

Monitored Geologic Repository Project Description Document (Curry 2001, Section 4) captures, by logical groupings, the hierarchical arrangement of the repository design documents. In that hierarchy, the repository is divided into three major systems, each with several secondary systems, as follows:

- Waste Handling System
  - Carrier/cask shipping and receiving systems
  - Waste preparation systems
  - Waste treatment systems
  - Waste emplacement and retrieval systems

- Waste Isolation System
  - Engineered barrier system
  - Natural barrier system
  - Performance confirmation system

- Operational Support System
  - Underground development system
  - Management and administrative systems
  - Safety and security systems
  - Nonradiological waste systems
  - Utility systems
  - Transportation systems (onsite).

The secondary systems are discussed further in this section, since they relate to the surface and subsurface design descriptions. The complete arrangement for all the potential repository systems is described in Section 4 of *Monitored Geologic*

Table 2-1. Event Sequence Frequency Categories

Event Sequence Category	Frequency of Occurrence	10 CFR 63.2 Definition
1	One or more times before permanent closure	A series of actions and/or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation and that are expected to occur one or more times before permanent closure of the geologic repository operations area.
2	At least 1 chance in 10,000 of occurring before permanent closure.	Other event sequences that have at least 1 chance in 10,000 of occurring before permanent closure of the geologic repository.

Source: BSC 2001f, Section 5.2.1.

Repository Project Description Document (Curry 2001).

Each repository system would eventually have a complete System Description Document. Figure 2-4 illustrates the relationship of a System Description Document to the design process. Section 1 of a System Description Document defines the system functions, criteria, requirements, constraints, and interface requirements with other systems, as applicable to that system's structures and components.

Given this set of information, the design engineer develops design concepts that fulfill the system needs; those concepts are subjected to additional evaluations, including technical feasibility, functionality, constructibility, cost, and operations and maintenance efficiency analyses. These analyses identify the selected design option. Selected design options may be subjected to further scrutiny such as ALARA evaluations for radiological exposures.

The design work is documented in calculations, technical reports, and design analyses that are submitted for independent reviewers to check and for impacted organizations to review, following a rigorous NRC-approved quality assurance program. Design documents are revised to incorporate checking and review comments, then rechecked for conformance to checking and design review comment resolutions before approval.

Section 2 of a System Description Document summarizes the design information contained in the approved design documents with a description of the system, its structures, and its components. Section 2 of each System Description Document also includes a demonstration of how the system criteria and requirements are fulfilled by the selected design for systems important to radiological safety.

2.1.2 Design and Operational Mode Evolution

Refining the design for the potential repository and the mode in which the design is operated has been an ongoing, iterative process involving scientists, engineers, and decision-makers. The design and mode of operations of a potential Yucca Mountain repository has evolved as more has been learned about the site and the performance contribution of design attributes and operational objectives.

Previous studies have investigated repository operating modes, layout, and performance considerations for a range of thermal conditions. Mansure and Ortiz (1984) developed evaluations of areas that could be used for expansion of the repository conceptual design footprint based on uncertainty of rock characteristics. This information was used in *Environmental Assessment Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1986c) and the *Site Characterization Plan Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1988). Building on this information, alternative design concepts were presented in *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998, Volume 2, Section 8.3.2). *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998) described the evolution of the repository design from the *Site Characterization Plan Conceptual Design Report* (SNL 1987) to the publication of the Viability Assessment. This section briefly discusses the evolution of the repository design and the range of thermal operations, from the publication of the Viability Assessment in December 1998 to the present. Descriptions of the design and mode of operations used in performance evaluations are covered in more detail in Sections 2.2 through 2.5 and in Section 3 to assist in understanding the design.

The DOE initiated an effort to evaluate a range of alternative design concepts and operational performance objectives during the License Application Design Selection process. This process culminated with the selection of what was referred to as the "license application design" (CRWMS M&O 1999c) and the subsequent analyses and designs for the selected design alternative, Enhanced Design Alternative II (CRWMS M&O 1999d). The Enhanced Design Alternative II concept was modified to remove backfill from the emplacement drifts and add operational performance objectives for thermal loading and ventilation (Wilkins and Heath 1999; Stroupe 2000). This design selection process considered a range of thermal conditions.

The Enhanced Design Alternative II design concept and mode of operations provide a moderate repository environment compared to the design described in the Viability Assessment. The Enhanced Design Alternative II design would keep the spent nuclear fuel cladding temperature below 350°C (662°F), a design requirement established to maintain the integrity of the cladding, and keep the boiling fronts from coalescing in the rock pillars between the emplacement drifts. To achieve the operational objective of keeping the boiling fronts from coalescing in the rock pillars, the ventilation rate in the emplacement drifts was increased, which allowed the spacing between waste packages to be reduced.

The design of structures, systems, and components and the methods used to operate and maintain these structures, systems, and components are distinct but interrelated in their developmental and evolutionary processes. Sensitivity analyses have been performed to identify design features and operational objectives that contribute to the isolation of waste or reduce uncertainty in performance analyses. In the design evaluation process described above, the design features and operational objectives were evaluated together to identify the combination of design features and the range of operational objectives that might enhance the performance of the potential repository system.

The draft environmental impact statement (EIS) for the potential Yucca Mountain repository recognized the potential for operating modes spanning a range of lower temperatures and presented results of analyses of the impacts associated with the range of thermal loadings (DOE 1999a, Appendix I, Section I.4.2). The conceptual layouts analyzed for the draft EIS for the different thermal load scenarios and inventories were developed considering the TSPA model domains and a number of rock areas within which the emplacement drifts would be located. The repository and emplacement areas for the higher- and lower-temperature operating modes with a 70,000-MTHM inventory are illustrated in Figures 2-5 and 2-6. Repository layouts for the draft EIS high, intermediate, and low thermal load scenarios are illustrated in Figure 2-7. The figure also illustrates, for comparative purposes, the layout for the higher-temperature operating mode described in this report.

A more detailed analysis of the lower-temperature operating modes resulted in a repository layout (see Figure 2-10) capable of accommodating the 70,000 MTHM and 97,000 MTHM inventories within the primary and lower blocks (BSC 2001g). This analysis also included parametric studies and layout sensitivity analyses for a wide range of repository thermal operating modes. The postclosure performance assessment for this new layout was discussed in Volume 2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The layout is also described in the final EIS.

The various components of the repository system would contribute to performance in different ways. Certain components of the system are more important to performance than others, as shown by performance assessment sensitivity studies (e.g., Section 4.4.5). Considerations of safety margin and defense in depth, insights from analogues, and expert judgments also help identify the importance of repository components to performance. Knowledge of how components of the system affect performance can provide insights on how design features or the operating mode could be developed in a manner that contributes to long-term performance or could mitigate potentially adverse conditions.

2.1.2.1 Summary of Evolution of Design Features

From the issuance of *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998) to the present, some design features in surface, subsurface, and engineered barrier systems and in the waste package have evolved. This section summarizes the evolution of these features.

Surface Facilities—The design of the surface facilities has evolved in two areas since the publication of the Viability Assessment (DOE 1998). The design of the Waste Handling Building has evolved to include an expanded-capacity spent nuclear fuel blending pool. The number of assembly and canister transfer lines was reduced to two assembly transfer lines and one canister

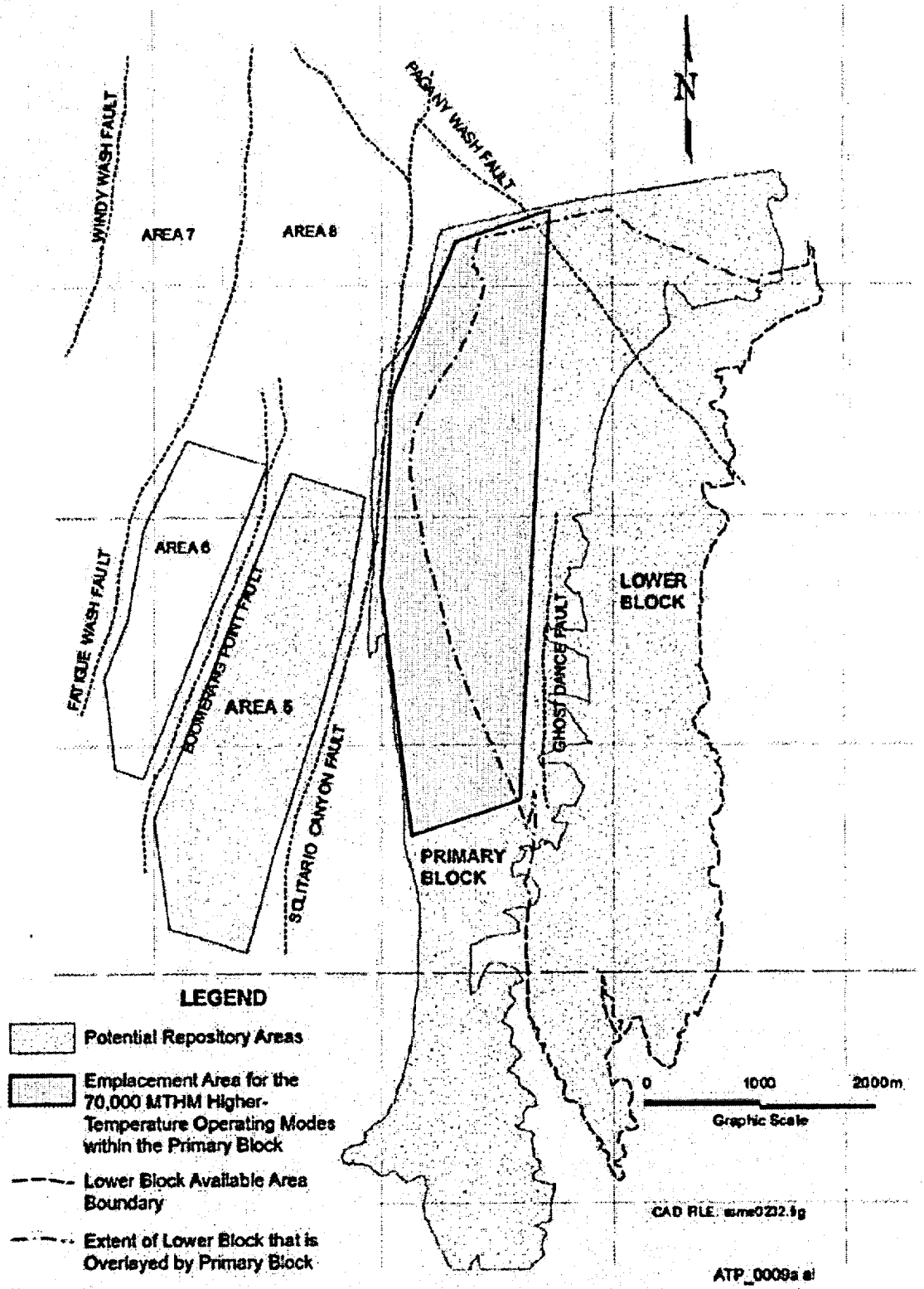


Figure 2-5. Potential Repository Areas and Emplacement Area for the Higher-Temperature Operating Modes

The emplacement area shown within the primary block would accommodate 70,000 MTHM in higher-temperature operating modes. Source: Modified from BSC 2001d; CRWMS M&O 2000d; and CRWMS M&O 2000j.

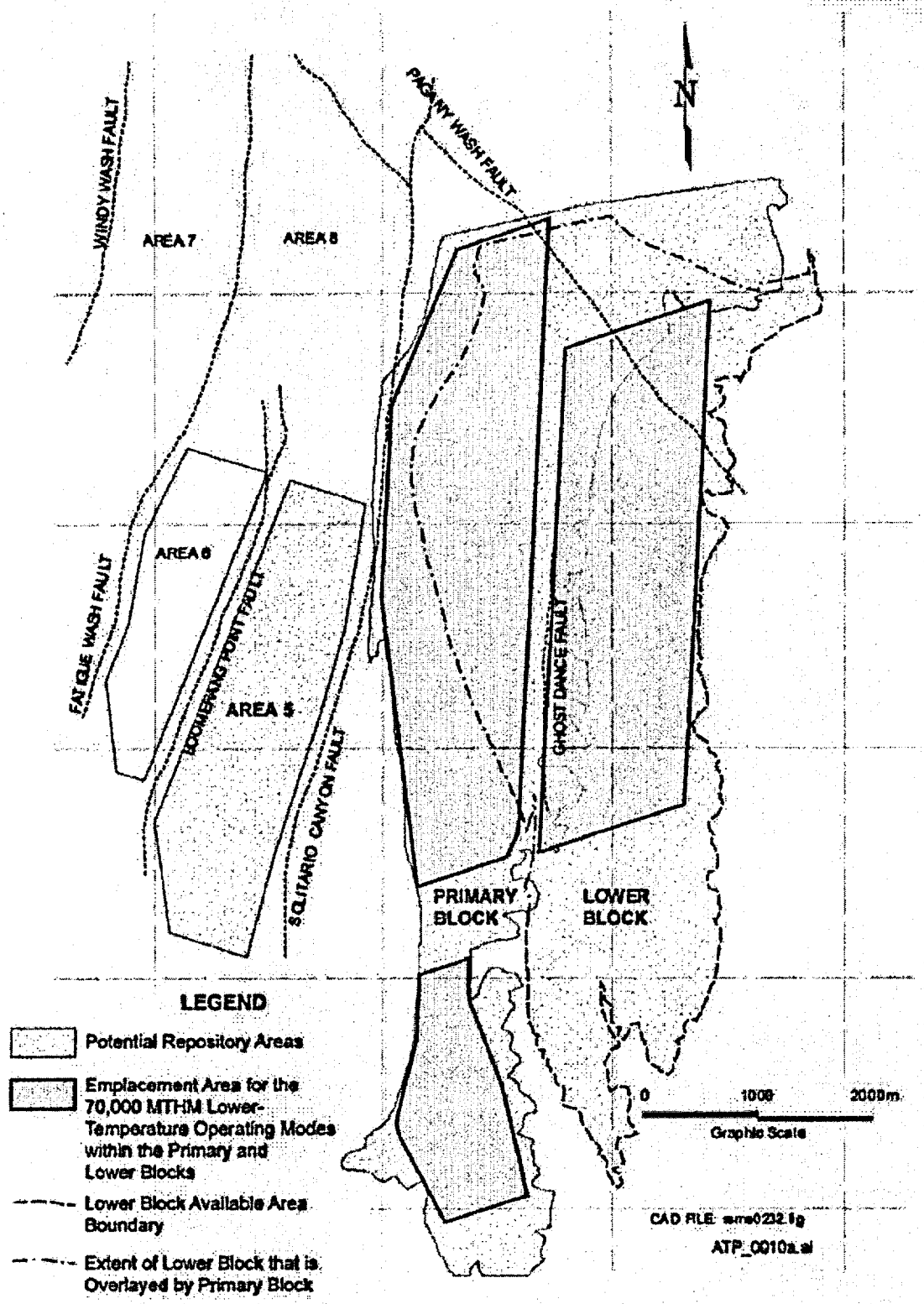


Figure 2-6. Potential Repository Areas and Emplacement Area for the Lower-Temperature Operating Modes

The emplacement areas shown within the primary and lower blocks would accommodate 70,000 MTHM in lower-temperature operating modes. Source: Modified from BSC 2001d; CRWMS M&O 2000d; and CRWMS M&O 2000j.

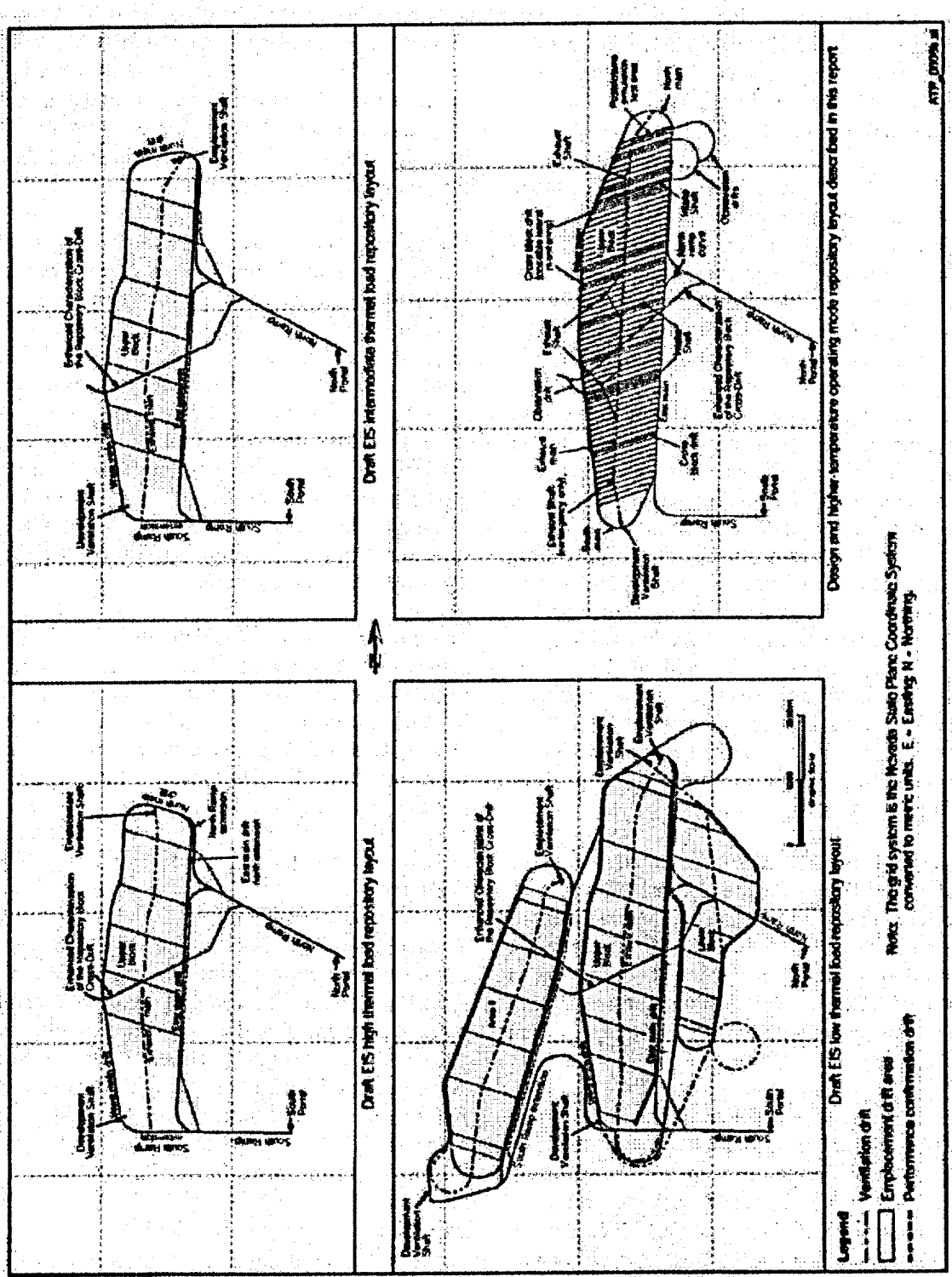


Figure 2-7. Repository Layouts for the Draft Environmental Impact Statement for High, Intermediate, and Low Thermal Load Scenarios and the Design for the Higher-Temperature Operating Mode

Source: High, intermediate, and low thermal load scenario repository layouts for the draft EIS modified from DOE 1999a, Figures 2-14, 2-15, and 2-16.

transfer line; the design described in the Viability Assessment included three assembly transfer lines and two canister transfer lines. The fuel blending strategy is presented in Section 2.2.1. Also, a solar power generating facility was added to supplement power from the site, which gets its power from the southern Nevada grid.

Subsurface Facilities and the Emplacement Drift Portion of the Engineered Barriers—The evolution of the subsurface facilities design has introduced several changes since the VA design, mainly to permit operating the repository to accommodate a range of thermal conditions. The most significant design change relates to thermal loading (the allowable amount of introduced heat per unit of subsurface emplacement area). The higher-temperature operating mode design described in this report uses a spacing between emplacement drifts of approximately 81 m (266 ft), compared to a spacing of approximately 28 m (92 ft) in the VA design (DOE 1998). Operational parameters, such as waste package spacing and ventilation, can be varied to support flexible thermal goals (see Section 2.1.2). In addition, the design and operating mode used in the performance analyses achieves a more uniform distribution of rock temperatures along the drift, limiting potentially complex thermal-mechanical effects resulting from a varying thermal gradient along the drift axis.

The emplacement drifts were reoriented to increase drift stability. Based on the dominant rock mass joint orientation and a minimum offset criterion of 30° between the joint strike and drift orientation, a reorientation of the emplacement drifts to an azimuth of 252° was established for the design described in this report (BSC 2001d, Section 6.2.2.2).

Emplacement drift backfill and drip shields were added to further limit the possibility of water contacting the waste packages. The backfill concept also offers the waste package protection against rockfall (CRWMS M&O 1999c, Section 4.4.2). The backfill concept was later eliminated because of its potential adverse impact on spent nuclear fuel cladding temperatures. The backfill inhibits heat dissipation into the rock, resulting in higher waste package temperatures. Granular backfill placed around the waste packages as a layered system could divert moisture away from the waste package by capillary action (CRWMS M&O 1999c, Section 4.4.2). The drip shield is a metal structure placed over the waste package that serves as a diversion shield for water entering the drift from the rock strata above. The drip shield design concept is described in Section 2.4.

The ground control design concept for emplacement drifts was changed from precast concrete liners with a concrete invert to a combination of steel or ballast invert with steel sets and welded wire fabric, with and without grouted rock bolts. Concerns with the long-term impact of concrete on the alkalinity of the drift environment, along with its implications for corrosion of the engineered barrier and waste package components and for the possible enhancement of colloidal radionuclide transport, motivated this change in the design for ground control in the emplacement drifts. Section 2.3.4.1 describes the ground support system.

The evolution to line-loading of waste packages resulted in modifications in the design and function of the waste package support assembly, from two independent post supports in the VA design to a more substantial emplacement pallet for the entire waste package. This necessitated a change in transporter and emplacement gantry design. The support design simplifies and enhances the reliability of transporting and emplacing waste packages.

Waste Package—Two waste package design attributes have evolved since the publication of the Viability Assessment. The first involves the metallurgy of the inner and outer shells of the waste package. The design described in the Viability Assessment utilized a corrosion-resistant Alloy 22 inner shell and a structurally strong, carbon-steel outer shell. The design described in this report utilizes a corrosion-resistant Alloy 22 outer shell and a structurally strong, stainless steel inner shell. In addition to the enhanced shell design, the waste package has a modified top lid design. A third lid has been added and the lid design has been modified to accommodate stress mitigation techniques in the closure weld area.

2.1.2.2 Summary of Evolution of Operational Parameters

This section presents a summary of the evolution of the operational mode from issuance of *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998) to this report for a range of thermal conditions.

Emplacement Drift Ventilation—The TSPA described in *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998, Volume 3) utilized a mode of operations where the emplacement drift ventilation rate was 0.1 m^3/s (3.5 ft^3/s) after waste package emplacement. This low ventilation rate was not designed to remove heat from the emplacement drifts. This operating mode had temperatures in the host rock throughout the emplacement area above the boiling temperature of water. This zone of above-boiling temperatures extended into the host rock up to approximately 100 m (330 ft) from the emplacement drifts.

To keep the boiling fronts from coalescing in the rock pillars between the emplacement drifts, the ventilation system has to remove approximately 70 percent of the heat generated by the waste packages. Meeting this requirement demands ventilation of the emplacement drifts at an estimated rate of 15 m^3/s (530 ft^3/s) (see Section 2.3.4.3).

Waste Package Spacing—The spacing between the waste packages that was used in the VA evaluations varied depending on both the thermal environment in the emplacement drift and the time-dependent heat output for each individual waste package. The concept behind the spacing was to maintain a fairly uniform linear thermal loading along the length of the emplacement drifts. Since the thermal output of the waste packages varied with the waste form and the age of the waste enclosed, the spacing between the waste packages varied from approximately 1.3 to 9.3 m (4.3 to 30.5 ft). With this spacing, the waste packages could act as point sources for heat in the emplacement drifts.

With the requirement for the ventilation system to remove approximately 70 percent of the heat generated by the waste packages, the concept of uniform linear thermal loading evolved into a line-loading concept for waste package emplacement. Line loading, that is, placing the waste packages end to end, with a separation of 10 cm (4 in.), achieves a more uniform thermal profile along the length of the emplacement drifts with the ventilation conditions described. The concentration of heat sources resulting from line loading required that the emplacement drifts be spaced farther apart to distribute the thermal load over a larger area.

2.1.2.3 Design and Operating Mode Evolution

The repository design concept described in this document is a flexible design that can be operated over a range of thermal conditions. The future evolution of the design and operating mode will follow a process that includes (1) refining specific design requirements and performance goals to recognize performance-related benefits that could be realized through design and (2) enhancing components of the design to best achieve the performance-related benefits. The iterative design process has focused on improving the understanding of the contribution of design features to the performance of a potential repository. The emphasis of the approach using a flexible design and modes of operation for a potential repository at Yucca Mountain is to understand the impact that a design attribute or operational performance objective has on the performance of the site across a range of environmental conditions. This approach examines the sensitivity of a design parameter to a range of operating temperatures and environmental conditions, and it evaluates the performance and uncertainties associated with the temperature variation. If a design attribute is shown to have a significant impact on the performance of the repository, then the attribute undergoes further evaluation to fully develop the positive contribution or minimize the negative contribution to the performance of the repository.

Sensitivity analyses that consider the performance of a design over a range of thermal operating modes provide information that will guide the future development of the repository design. The evolution of the design will take advantage of

insights gained through the performance analyses. Furthermore, if the performance evaluations indicate benefits to be gained by refinement of the basic design concept on which the performance analyses over the range of operating modes were based, the evolution of the design will take advantage of those insights as well. This flexible design, which will continue to evolve for license application if the site is recommended, will complement a set of operational parameters that can be managed to accommodate thermal characteristics of a waste stream with potentially evolving characteristics. Adjustments can continue based on updated information on waste stream and other repository variables during the very long emplacement period.

The DOE has expanded the assessment of performance of the repository design to better understand how to use reductions in uncertainty to improve the design. The DOE is further expanding the range of thermal operating modes by assessing the performance of a potential repository operated to achieve lower temperatures. The TSPA described in Section 4 of this report utilizes an operating mode where the walls of the emplacement drifts are above the boiling temperature of water. Results of preliminary engineering evaluations show that the design can also be operated in a lower-temperature mode (CRWMS M&O 2000k). The results of these preliminary evaluations have been further corroborated with more detailed calculations and analyses presented in the following documents:

- *Calculation of Potential Natural Ventilation Airflows and Pressure Differential* (CRWMS M&O 2000l)
- *ANSYS Calculations in Support of Natural Ventilation Parametric Study for SR* (BSC 2001h)
- *Thermal Management Analysis for Lower-Temperature Designs* (BSC 2001i)
- *Lower-Temperature Subsurface Layout and Ventilation Concepts* (BSC 2001g).
- *Thermal Hydrology EBS Design Sensitivity Analysis* (BSC 2001j).

The effect of heat on the performance of the repository and the associated uncertainties are subjects of ongoing studies. These studies are considering ranges of drift wall temperatures from as high as 200°C (390°F) to below the boiling point of water (96°C [205°F] at the elevation of the emplacement horizon). The performance assessment in the Viability Assessment (DOE 1998, Volume 3) was based on a design that allowed drift wall temperatures to exceed 200°C (390°F). An objective of the operating mode described in this report is to maintain temperatures in a portion of the rock between the emplacement drifts below the boiling point of water. More recent design studies include sensitivity analyses that evaluate repository performance limiting all drift wall temperatures below the boiling point of water (BSC 2001g; BSC 2001h; BSC 2001i; BSC 2001j; CRWMS M&O 2000l). Lower-temperature operating modes to reduce uncertainty about corrosion rates associated with waste package performance have been evaluated in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b). These evaluations have considered waste package temperatures as low as 85°C (185°F).

The evaluation of the performance characteristics of the proposed repository design over a range of operating conditions is an important step in the evolution of the design of a potential repository at Yucca Mountain. While certain details of design are needed to understand the environmental conditions as input to the performance analyses, future evaluations will recognize and be based on the performance implications of the design and accompanying range of operating modes.

2.1.3 Design Flexibility

The design described in this report is part of an overall waste management system that is flexible, continues to evolve, and can adapt to various construction and operational conditions.

In this report, the phrase "flexibility in design" refers to capabilities inherent in the repository design to accommodate changing conditions, such as unanticipated underground conditions and new design requirements. The need for flexibility in the

repository design evolves from operational requirements, such as:

- Disposal of a wide range of radioactive waste forms and container sizes, with the understanding that not all details are yet known on waste receipt and delivery schedules (see Sections 2.4 and 3)
- Maintaining a capability to retrieve one or all of the waste packages (see Section 2.3.4.6)
- Maintaining the ability to monitor the facilities and surrounding environment over many years before committing to closure (see Sections 2.3.4.7, 2.5.2, and 4.6.1)
- Maintaining a fuel blending capability to achieve the selected repository thermal goal (see Section 2.2.1)
- Designing a subsurface ventilation system that can be adapted to achieve the selected repository thermal conditions and to adapt to concurrent and continually advancing drift construction and waste emplacement (see Sections 2.1.5 and 2.3.5)
- Defining a larger emplacement area than is expected to be needed to provide flexibility in achieving the selected repository thermal goals (see Section 2.1.4).

Flexibility in the design allows this evolutionary process to continue and to take advantage of the interrelationship between the design and operating mode. Because the specific thermal criteria (e.g., the waste package and drift wall temperature limits) that will be imposed on future design enhancements have not been selected, the repository design must be sufficiently flexible to allow operation under a wide range of thermal conditions. Assessment of performance over a range of thermal conditions will support the definition of specific thermal criteria upon which future design enhancements will be based. There is a possibility that spent nuclear fuel with higher burnup than presently projected would be received. This could lead to a higher integrated heat over the first 1,000 years of emplacement and could change the predicted temperature profile. The repository would have the flexibility to accept spent nuclear fuel with higher burnup rates.

Key aspects of design flexibility are (1) the ability of the repository design to support a range of construction approaches; (2) the capability to dispose of a wide range of waste container sizes; (3) the ability to support a range of thermal operating modes; and (4) the ability to continue to enhance the design to best achieve performance-related benefits identified through ongoing analyses. The following discussion describes the design's flexibility to accommodate sequential and modular construction of surface and subsurface facilities. Section 2.1.4 describes the design's flexibility to support a range of thermal operating modes.

Sequential and Modular Repository Development—The design of the repository can accommodate modular or sequential construction of surface and subsurface facilities. The phrase "modular or sequential implementation" describes a process in which decisions concerning repository development are made in a stepwise manner: at each step in the process, a decision whether to proceed would be made based on the licensing and regulatory requirements, the funding profile, and operating experience. The next stage of construction would proceed informed by the experience gained from the previous stage.

The DOE has assessed possible benefits of a modular or stepwise approach, which range from the incorporation of lessons learned after each stage of construction to the leveling of annual construction costs (CRWMS M&O 1998a). However, the potential benefits and impacts from modular and sequential construction have not been fully assessed. The DOE has requested that the National Research Council continue the study of possible repository development strategies (Itkin 2000). A report on the results of this study will be provided at a later date, and the DOE will continue to assess this concept for construction activities.

Sequential and modular construction would not be expected to change the design or operational

concepts of a potential repository at Yucca Mountain and would not, therefore, impact an evaluation of repository performance.

2.1.4 Operating Flexibility to Achieve a Range of Thermal Operating Modes

Preliminary engineering evaluations, including *Operating a Below-Boiling Repository: Demonstration of Concept* (CRWMS M&O 2000k) and *Natural Ventilation Study: Demonstration of Concept* (CRWMS M&O 2000m), show that the design described in this report can be operated to support a range of thermal operating modes. For example, the potential repository could be operated in a mode that keeps temperatures below the boiling point of water (96°C [205°F] at the repository elevation); or the potential repository could be operated such that the host rock reaches a temperature that is above the boiling point of water. Drift wall and waste package temperatures and relative humidity can be managed by altering several operational features of the design: (1) varying the thermal load to the repository by managing the thermal output of the waste packages; (2) managing the period and rate of drift ventilation prior to repository closure; and (3) varying the distance between waste packages in emplacement drifts (CRWMS M&O 2000k, Section 3.2). These factors are described in the following paragraphs. Other parameters, such as postemplacement natural ventilation, could also be used to reduce long-term repository temperatures (CRWMS M&O 2000m). Altering design features, such as emplacement drift spacing, could also be used in conjunction with variations in operational parameters to achieve a lower-temperature repository environment.

Recently completed calculations and analyses (BSC 2001j; CRWMS M&O 2000l; BSC 2001h; BSC 2001i; BSC 2001g) support the preliminary findings on the feasibility of operating the repository to meet varying thermal goals by adjusting the three operational features listed above (i.e., controlling the thermal output of the waste packages, adjusting the period of ventilation, or varying the emplacement distance between waste packages).

Since completion of the TSPA-SR analyses, the DOE has performed additional analyses supporting the effects of a range of thermal operating modes on projected system performance. These analyses were described in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b). Evaluations of higher- and lower-temperature operating modes in the supplemental TSPA model have provided insight into system performance at the subsystem level over a range of thermal operating modes. While some subsystem models indicated significant differences, results from changing the thermal operating mode showed only a minor impact on overall repository performance. The supplemental model also showed that the repository could be operated at either a higher- or lower-temperature operating mode with a low probability of the development of aggressive chemistries on the waste package and the subsequent potential for localized corrosion.

If the site is recommended for development of a repository, the DOE will utilize the enhanced understanding of repository performance to further improve the repository design for any license application.

Thermal Output of the Waste Packages—The major contributor of heat in the repository would be commercial spent nuclear fuel, which would likely have a wide range of thermal outputs. The thermal load of a repository is directly related to the amount of thermal energy contained in the fuel emplaced in it. The thermal energy contained in the fuel, in turn, is directly related to its age. The age and burnup rate of spent nuclear fuel received for emplacement in a repository would vary considerably, so the current operational plan for the potential repository specifies that the DOE will manage the fuel inventory by one or more of the following features: (1) fuel blending (i.e., placing low heat output fuel with high heat output fuel within a waste package); (2) de-rating (i.e., limiting the number of spent fuel assemblies to less than the waste package design capacity); (3) placing high heat output fuel in smaller waste packages; or (4) aging (i.e., placing hotter fuel into the fuel blending inventory to be emplaced later). Managing the average thermal output of the waste packages through any of these means can control

drift wall temperatures of the repository (CRWMS M&O 2000k, Section 3.2; BSC 2001a; BSC 2001b).

Duration and Rate of Forced Ventilation—During active repository operations, some of the heat generated by the waste and the moisture in the surrounding rock would be removed from the repository by forced ventilation of the loaded emplacement drifts. The amount of energy transferred from the waste to the host rock can be managed by varying the duration and the rate of emplacement drift ventilation (CRWMS M&O 2000k, Section 3.2.3).

Distance Between Waste Packages—The distance between waste packages in emplacement drifts is another design feature that can be modified to manage the temperature in the potential repository. As waste packages are spaced farther apart, the linear thermal density in the drift (measured in kilowatts of heat output per meter of drift length) decreases, delivering less heat per unit volume of the host rock when the drift-to-drift spacing remains fixed (CRWMS M&O 2000k, Section 3.2.2).

Natural Ventilation—Postemplacement natural ventilation could be employed in several lower-temperature operating scenarios. The subsurface ventilation system described in Section 2.3 could support both forced and natural ventilation. To facilitate natural ventilation, the ventilation system would be enhanced through a combination of air balancing techniques, such as adjusting the size of the openings, location and number of intake/exhaust openings, and flow controls (CRWMS M&O 2000m).

2.1.5 Assessing The Performance of a Lower-Temperature Operating Mode

The basic operational features to achieve a lower-temperature repository were described in *Operating a Below-Boiling Repository: Demonstration of Concept* (CRWMS M&O 2000k). That report outlines the relationships between operational parameters such as ventilation and linear thermal load. In subsequent design evaluations (BSC 2001j; CRWMS M&O 2000l; BSC 2001h; BSC 2001i; BSC 2001g), the DOE has used a range of lower-temperature operating mode characteristics to investigate the performance attributes of possible alternative lower-temperature designs and operating modes.

This section describes features of some of these lower-temperature operating modes. These features have been used to assess the performance of a potential lower-temperature repository and the sensitivities of projected performance to different parameters and design features (BSC 2001b). Evaluations of a range of operating temperatures will support the development of a design and operating mode to support any license application. Features of lower-temperature operating modes have also been assessed to support the development of the ranges of design-related environmental conditions required as input to the TSPA analyses.

Design and performance evaluations could include varying the separation between the emplacement drifts (drift-to-drift spacing) to adjust the areal mass loading; increasing the number and diameter of ventilation shafts to improve efficiency; or zone emplacement, which tailors ventilation and design attributes to the thermal output of waste packages emplaced in specific drifts (or zones). The DOE will also examine strategies to reduce ventilation times and meet thermal goals for a lower-temperature operating mode. This could involve lengthening the time to emplace all of the wastes, further reducing areal thermal densities, rearranging emplaced fuel, or combining the aging of fuel with enhanced fuel blending inventory strategies.

Lower-temperature operating modes are being considered for mitigating some of the potential uncertainties in assessing long-term repository performance. There are many ways of combining operational parameters to achieve lower repository temperatures. Figure 2-8 illustrates the variables affecting the thermal performance of the repository, from waste forms to emplacement drifts. Within the constraints imposed by the physical system, the potential repository can be operated in lower-temperature modes while also meeting other technical, policy, regulatory, schedule, and opera-

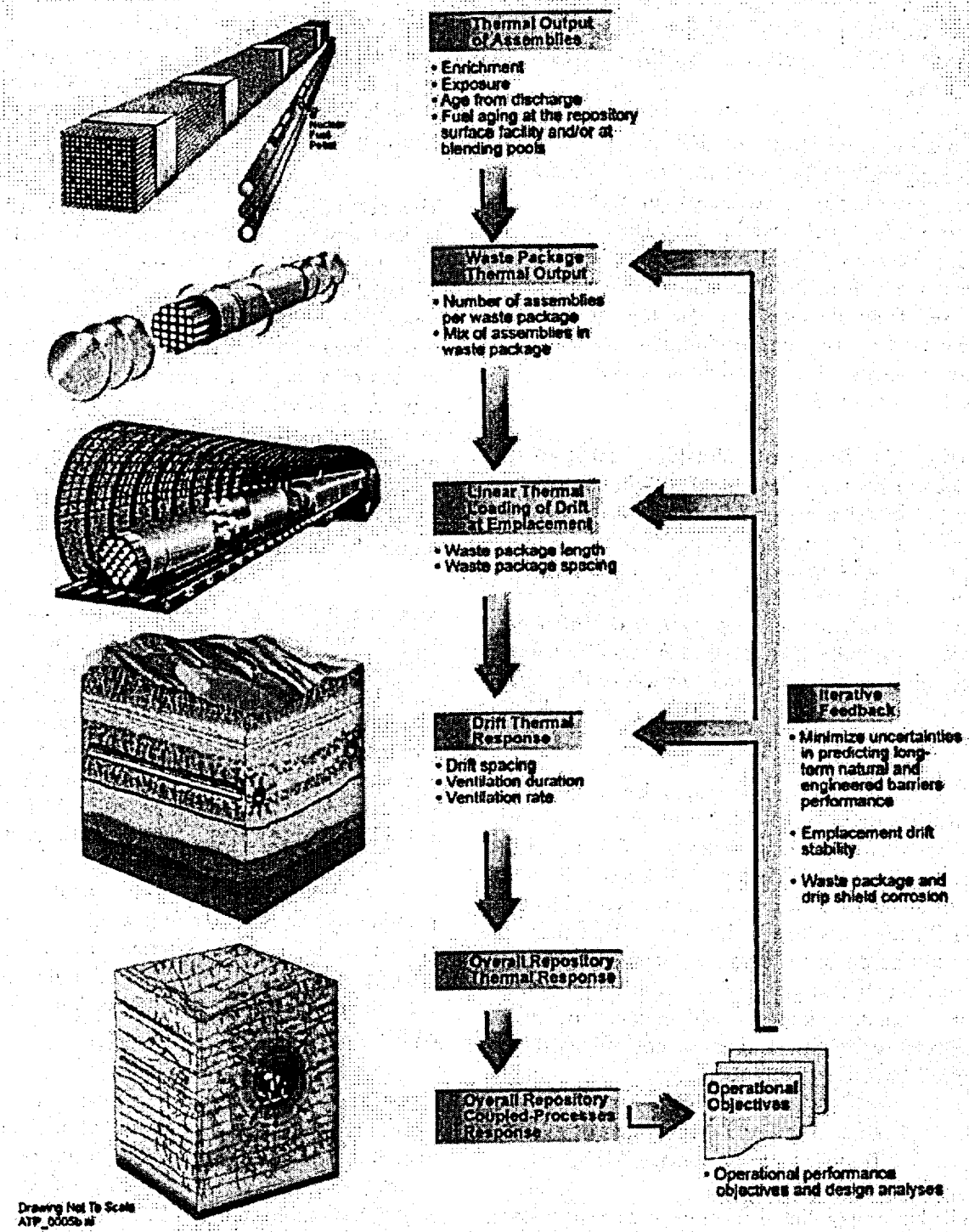


Figure 2-8. Variables Affecting the Thermal Performance of the Repository
This figure illustrates the variables associated with spent nuclear fuel assemblies, waste packages, and in-drift emplacement, and which affect repository thermal performance. Changing these variables through design modifications or operations directly impacts the repository thermal regime and indirectly impacts the host rock chemical, mechanical, and hydrologic regimes. These impacts, in turn, affect performance of the repository engineered barriers. Evolution of optimal designs and operating modes must include assessments of these changes and impacts on an iterative basis for continuous improvement of repository components. This iterative feedback would continue through the licensing process and into the construction and operational phases of the repository.

tional objectives. Some of these objectives are described in the following paragraphs.

Reduced Uncertainty in Corrosion Rates—The corrosion susceptibility of Alloy 22 may be reduced by either keeping waste package surface temperatures at or below 85°C (185°F) or maintaining in-drift relative humidity below 50 percent (see Figure 2-9) (Dunn et al. 1999, p. xvi; CRWMS M&O 2000n, Section 3.1.3.1). Operating the repository such that the combination of in-drift temperature and relative humidity does not enter the window of corrosion susceptibility of Alloy 22 may result in better overall performance.

Keeping Drift Wall Temperatures Below Boiling—In the higher-temperature operating mode, the rock temperatures within the first several meters around the emplacement drifts exceed the boiling point of water. A possible lower-temperature objective would be to keep all the rock in the repository below the boiling point of water to reduce uncertainties associated with thermal-hydrologic and thermal-mechanical processes.

Capacity for Waste Inventory—Potential objectives are (1) to ensure that 70,000 MTHM can be emplaced within the characterized area and (2) to maintain the flexibility to accommodate up to 119,000 MTHM within the repository area.

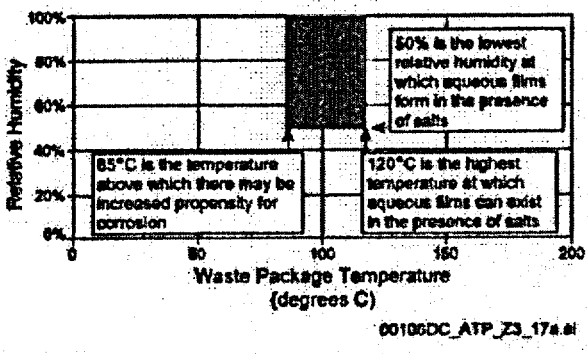


Figure 2-9. Window of Potentially Increased Susceptibility of Localized Corrosion for Alloy 22

The temperature–humidity corrosion susceptibility ranges are preliminary and under evaluation and are subject to the results of ongoing testing. Source: Dunn et al. 1999, p. xvi; CRWMS M&O 2000n, Section 3.1.3.1.

However, the smaller the space that is used for inventory, the more reliance must be placed on other means of meeting lower-temperature goals.

Duration of the Ventilation Period—Potential objectives are (1) to provide for closure and sealing of the repository within a prescribed time limit determined by the Secretary of Energy or (2) to allow for natural ventilation to continue after permanent closure of the repository. Allowing some dependence on extended ventilation reduces the area required for disposal of a given amount of waste. However, a design based on centuries of ventilation through air passages connecting the repository to the surface could involve technical issues that would have to be explored.

Thermal Output of the Waste Package—The thermal output of the waste packages can be managed by combining one or more of the following features: (1) fuel blending to produce a thermal loading at or below 11.8 kW per waste package; (2) de-rating the waste packages to produce a thermal loading below 11.8 kW per waste package; (3) placing high heat output fuel in smaller waste packages to produce a thermal loading below 11.8 kW per waste package; or (4) aging spent nuclear fuel assemblies to reduce their heat output.

2.1.5.1 Lower-Temperature Operating Mode—Coupled Operational Parameters

Figure 2-10 illustrates a layout of potential development areas (or blocks) within the characterized area of Yucca Mountain that may be utilized for the emplacement of waste. The shaded areas denote portions of the upper and lower blocks that could be utilized for the emplacement of waste under different operating modes, ranging from higher-temperature to lower-temperature cases corresponding to the examples given in Table 2-2.

The TSPA-SR evaluated the performance of a higher-temperature operating mode. In this mode, as rock temperatures increase above the boiling point of water, moisture around the emplacement drifts evaporates and is driven away from the drifts as water vapor. As described in Section 4.2.2, this

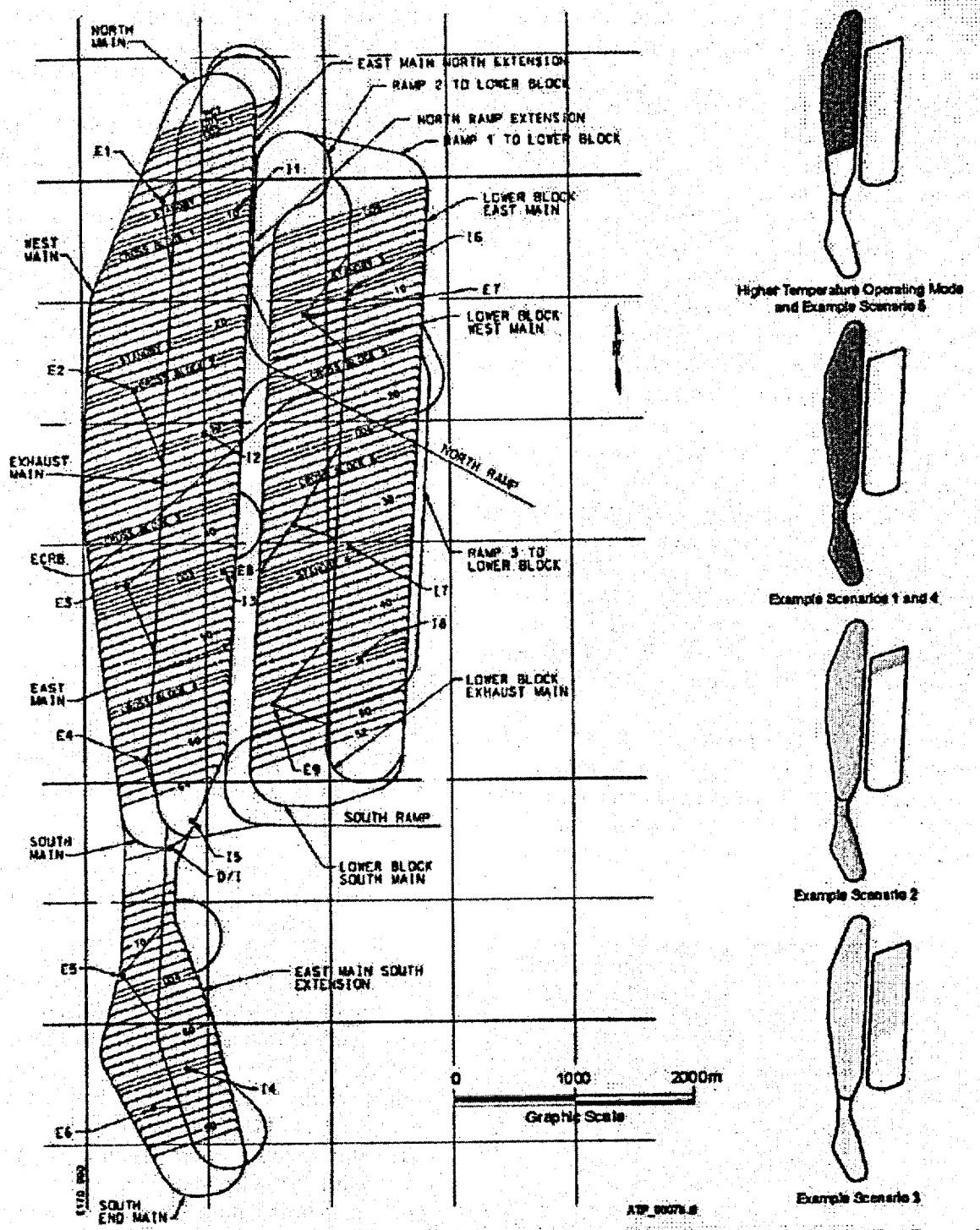


Figure 2-10. Layout of Potential Repository Development Areas, Showing Areas Utilized in Example Operating Mode Scenarios for a Repository Capacity of 70,000 MTHM

The shaded areas on the right denote the portions of the upper and lower blocks that would be utilized for emplacement of waste under different operating modes, ranging from higher-temperature to lower-temperature cases. These five cases are the example scenarios defined in Table 2-2. Dimensions, coordinates, and elevations are given in meters. PC = postclosure test drift; OD = observation drift; I1 = intake shaft; E1 = exhaust shaft; D/I = development/intake shaft; ECRB = Enhanced Characterization of the Repository Block Cross-Drift. Source: Modified from BSC 2001d.

Table 2-2. Comparison of Estimates of Operational Parameters for Example Lower-Temperature Operating Modes

Parameters	TSPA-SR Higher-Temperature Operating Mode	Example Scenarios				
		1	2	3	4	5
		Increased Waste Package Spacing and Extended Ventilation	De-Rated or Smaller Waste Packages	Increased Spacing and Duration of Forced Ventilation	Extended Surface Aging with Forced Ventilation	Extended Natural Ventilation
Variable Parameters						
Waste package spacing (m)	0.1	2	0.1	6	2	0.1
Maximum waste package thermal loading	11.8 kW	11.8 kW	<11.8 kW	11.8 kW	<11.8 kW	11.8 kW
Linear thermal loading objective (kW/m) at emplacement	1.45	1	1	0.7	0.5	1.45
Years of forced ventilation after start of the emplacement	50	75	75	125	125	75
Years of natural ventilation after forced ventilation period	0	250	250	0	0	>300
Dependent Parameters						
Size of pressurized water reactor waste packages	21 PWR	21 PWR	<21 PWR	21 PWR	21 PWR	21 PWR
Total excavated drift length (km)	~60	~80	~90	~130	~80	~60
Required emplacement area (acres)	~1,150	~1,600	~1,800	~2,500	~1,600	~1,150
Average waste package maximum temperature (°C)	>96	<85	<85	<85	<85	<96

NOTE: PWR = pressurized water reactor.

may have performance benefits because it would delay the time at which liquid water could begin to seep into the emplacement drifts. However, there are heat-related uncertainties about the long-term impact of thermal, chemical, and mechanical effects on the hydrologic properties of the potential repository host rock. Maintaining the repository at lower temperatures (below the boiling point of water) may reduce these uncertainties. Additional analyses of lower-temperature operating modes were performed and are presented in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b).

To demonstrate how the DOE can vary operational parameters of the design to manage the thermal load of the repository, a set of preliminary calculations was performed to assess the effects on repository temperatures of the parameters described in the previous section: (1) the age of the spent fuel emplaced; (2) the duration of ventilation; and (3) the distance between waste packages (CRWMS M&O 2000k; CRWMS M&O 2000m).

The family of operational parameters that would produce average maximum drift wall temperatures below the boiling temperature of water (96°C [205°F]) is shown in Figure 2-11. This figure plots these parametric curves using postloading ventilation duration as the horizontal axis and distance between waste packages as the vertical axis. Each of the labeled curves in this plot represents a different duration of additional aging beyond the average age of the waste stream in the year it is received. The curve with 0 years of aging corresponds to the case where the fuel is emplaced at the rate it is received, without additional aging. The curve for 5 years of aging corresponds to the case where the waste is allowed, on average, an additional 5 years of aging.

An example of how these curves are used can be shown by examining the point on the 10-year aging

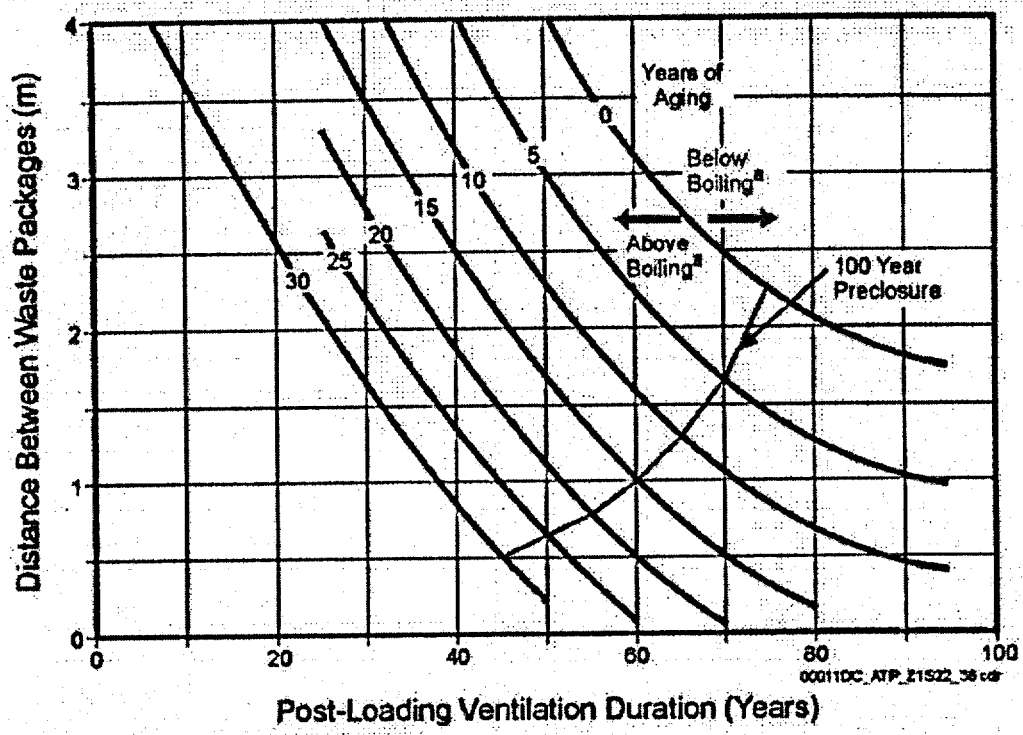


Figure 2-11. Below-Boiling Repository Operating Curves

*The above-boiling and below-boiling arrows are associated with the particular curve indicated and reflect conditions that lie to the left and right of that curve. The arrows may be reproduced for each curve in the figure to reflect where the associated above and below-boiling conditions lie.

This figure illustrates the relationship between two design variables: (1) end-to-end spacing between waste packages in the emplacement drift and (2) average years of aging of the waste prior to emplacement—and how these two variables relate to the duration in years of forced (mechanical) ventilation required to maintain the average maximum host rock temperature below the boiling point of water during preclosure and after closure of the repository. The "years of aging" (bold curved lines) mark the boundary at which the average maximum host rock temperature would exceed the boiling temperature of water at a given increment of waste package spacing, for a given duration of ventilation. For example, for a 1-m (3.3-ft) spacing between waste packages whose waste has been aged for 15 additional years, 60 years of forced ventilation would be required to maintain average maximum rock temperatures below 96°C (205°F) during preclosure and after closure. This figure uses a 100-year preclosure duration as an example to illustrate the combinations of aging and ventilation. Source: CRWMS M&O 2000k, Figure 3-4.

curve for 50 years of ventilation and a waste package spacing of about 2.3 m (7.5 ft). For a drift operated this way, the postclosure emplacement drift wall would reach a peak temperature of just at or below 96°C (205°F). If the drift were to receive fewer years of ventilation, keeping the age of the fuel loaded in the drift and the waste package spacing constant (to the left of the point in question), then the average maximum temperature of the host rock walls would increase above the boiling point of water during the postclosure period. On the other hand, if an emplacement drift were ventilated longer (to the right of the point in question), then the peak temperatures in the emplacement drift rock would be farther below the boiling point of water.

The trade-offs between the operational parameters that can be changed to achieve a below-boiling repository operating mode can be determined from Figure 2-11. For example, if an emplacement drift is loaded with waste packages spaced around 2.3 m (7.5 ft) apart without fuel aging, 75 years of post-loading ventilation is required. If the age of the waste stream is increased by 10 years, a waste package spacing of 2.3 m (7.5 ft) can be used with only 50 years of ventilation. For this case, aging the waste for 10 years saves 25 years of ventilation.

Increasing the duration of aging, however, yields diminishing benefits. As the waste ages appreciably, the rate of decay in heat generation diminishes greatly. Consequently, whether the waste is aged further on the surface or underground in the potential repository makes less and less difference in keeping the drift wall below boiling. The flattening shape of the curves in Figure 2-11 beyond 75 years of ventilation also shows that increasingly longer ventilation periods would be required to remove the same quantity of decay heat from the repository.

2.1.5.2 Example Lower-Temperature Operating Scenarios

Lower-temperature operating modes have been developed to provide options for mitigating some of the potential uncertainties in assessing long-term repository performance. Analyses of the potential performance of lower-temperature operating modes are described in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b).

This section describes several possible operating scenarios, each of which meets the primary goals for lower-temperature operating modes described in Section 2.1.5 while achieving one set of possible operating objectives. Table 2-2 compares operational parameters used in the TSPA-SR with operational parameters that could be used to achieve a lower-temperature operating mode.

The following sections compare five examples of lower-temperature operating scenarios. These examples provide a basis for understanding the technical issues associated with developing a lower-temperature operating mode.

2.1.5.2.1 Lower Waste Package Temperature Achieved through Extended Ventilation and Minimal Increase in Disposal Area

By extending the time during which loaded emplacement drifts are ventilated, the repository could be operated at lower temperatures with minimal increase in the disposal area. This section describes two example lower-temperature operating mode scenarios that are likely to satisfy the following three objectives:

1. Maintain average waste package surface temperatures below 85°C (185°F)
2. Ensure that 70,000 MTHM of waste fits mostly within the upper block, as shown in Figure 2-10
3. Close and seal the repository within approximately 300 years.

Example scenarios 1 and 2 could achieve a lower-temperature operating mode through extended forced and natural ventilation. Example scenario 2 also includes a de-rated waste package.

Example Scenario 1: Increased Waste Package Spacing and Extended Ventilation—In this example, loaded waste packages would be emplaced an average of about 2 m (6.6 ft) apart to create a 1-kW/m drift thermal load at emplacement. The drift-to-drift spacing would remain at 81 m (266 ft), as specified in the design described in this report. For the first 75 years after the start of emplacement, fans would actively ventilate the drifts with an airflow rate of 15 m^3/s per drift. Because of the time required for emplacement, the drifts loaded last would be actively ventilated for 50 years. The repository would be allowed to ventilate naturally for 250 years. Other operational parameters would be unchanged.

Figure 2-12 contains plots of waste package surface and drift wall temperatures over time for the drifts loaded in the final year of emplacement operations under this scenario, as projected by two-dimensional models (BSC 2001h). In the plots shown in Figure 2-12 (also in Figures 2-13, 2-14, and 2-15), the time axis begins after the last emplacement drift is loaded; this results in bounding projections of the maximum waste package and drift wall temperatures in drifts that are loaded earlier in the emplacement period. Descriptions of the repository layout and the changes to the ventilation system that would be required to implement this scenario are discussed later in this section.

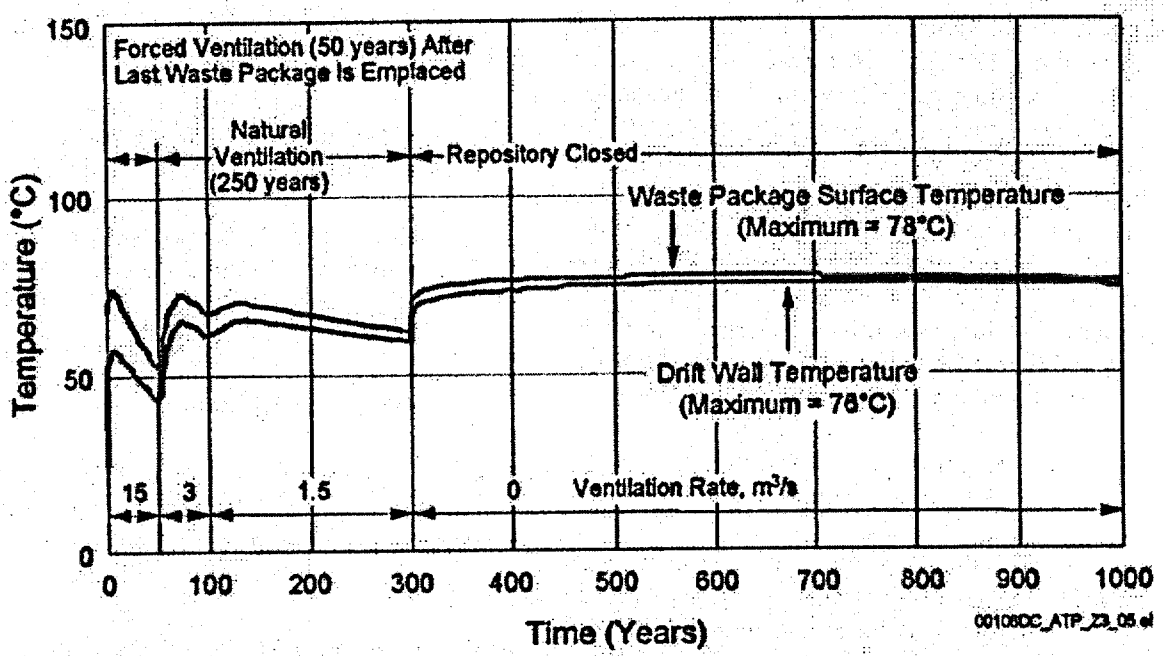


Figure 2-12. Waste Package Surface and Drift Wall Temperatures for an Operational Scenario in Which Drifts Loaded at 1 kW/m in the Last Year of Emplacement Operations are Actively Ventilated for 50 Years and Naturally Ventilated for Another 250 Years

Source: Modified from BSC 2001h, Figure XVII-4.

Some of the implications of this scenario are (1) the flexibility to readily adjust to a higher-temperature operating mode in drifts loaded later by moving waste packages closer together; (2) a requirement for additional drift excavation to accommodate more widely spaced waste packages; (3) increased complexities in projecting the thermal-hydrologic response of the repository because the widely spaced waste packages would act more like point heat sources within drifts; and (4) the programmatic uncertainty associated with the protracted period of natural ventilation.

Example Scenario 2: De-Rated or Smaller Waste Packages—In this scenario, the thermal output of the waste packages is reduced by limiting waste package loading. This can be achieved by limiting the number of spent nuclear fuel assemblies to less than the waste package design capacity (de-rating) or replacing the large waste packages (e.g., ones containing 21 pressurized water reactor fuel assemblies) with smaller waste packages (e.g., ones that would contain 12 pressurized water reactor fuel assemblies) that have a lower thermal output.

All waste packages would be placed approximately end-to-end within the drifts to create a 1-kW/m linear thermal load at emplacement. Other operational parameters of this scenario are identical to those in Scenario 1. Two-dimensional projections of waste package surface and drift wall temperatures over time are the same as those shown in Figure 2-12.

Figures 2-12 through 2-15 show variations in the waste package and drift wall temperature responses with time. Figure 2-12 illustrates a representative process, as follows.

1. During the forced ventilation period (0 to 50 years), the heating effect of the waste is steadily reduced until the fans are turned off in year 50.

2. From that point, natural ventilation flow continues to remove heat; however, the reduced flow rate (3 m^3/s) takes several

years to turn the temperature response to a downward trend (approximately years 70 to 100).

3. The natural ventilation flow rate gradually decreases over time as the waste decays, since the heat from the waste is the main driver in inducing convective currents in airflow. The simplified calculation methods used could not simulate a steady decline in the natural ventilation rate, so this decline was simulated by stepping down the rate in years 100 to 300 from 3 m^3/s to 1.5 m^3/s.

4. The temperature rises from years 100 to approximately 130 due to this abrupt reduction in the natural ventilation flow rate. After year 130, the reduced flow rate starts reducing temperatures until year 300, when the repository would be sealed and closed.

5. At that point, the lack of ventilation induces a steep rise in the drift temperatures until the host rock's capacity to transfer heat away from the waste packages starts to have a regulating effect on drift temperatures and establishes a very slow downward trend in thermal response over a period of 1,000 years or longer.

Note that in the more natural situation of a steady decrease in the natural ventilation flow rate, the calculated thermal response curves would actually show a steady decrease in drift temperatures across most of the natural ventilation period, rather than the variations shown around the 100-year period.

The primary difference between this scenario and Scenario 1 is a potential reduction in the complexity of modeling the thermal-hydrologic response of the repository because the thermal loading would more closely resemble a line load. A second difference is the increase in the number of waste packages needed, which could result in an increase in the total excavated drift length and the total area required for emplacement.

2.1.5.2.2 Lower Waste Package Temperature Achieved through Increased Disposal Area and Limited Ventilation Period

Lower-temperature operating goals can also be achieved with a limited increase in the ventilation period by increasing the area used for emplacement. The three objectives for this set of examples are:

1. Maintain average waste package surface temperatures below 85°C (185°F)

2. Ensure that 70,000 MTHM of waste fits within the upper and lower blocks (Figure 2-10)

3. Close and seal the repository within approximately 125 years.

Example scenarios 3 and 4 could achieve lower temperatures through extended forced ventilation. Example scenario 4 also includes extended surface aging of spent nuclear fuel.

Example Scenario 3: Increased Spacing and Duration of Forced Ventilation—Lower temperatures could be achieved with a limited increase in the preclosure period by emplacing waste packages an average of about 6 m (20 ft) apart. This would create a drift thermal load at emplacement of approximately 0.7 kW/m. The drift-to-drift spacing would remain at 81 m (266 ft). The loaded drifts would be actively ventilated for 125 years from the start of waste emplacement, with the drifts that are loaded in the last year of emplacement operations receiving 100 years of forced ventilation. Figure 2-13 displays preliminary projections of waste package surface and drift wall temperatures over time for this scenario.

The implications of this approach include: (1) a preclosure period comparable to the current preclosure period of approximately 100 years; (2) an increase in waste package spacing, which would likely result in point thermal loading of drifts and may give rise to increased complexities in modeling the thermal-hydrologic response of the host rock; (3) increases in the total excavated drift

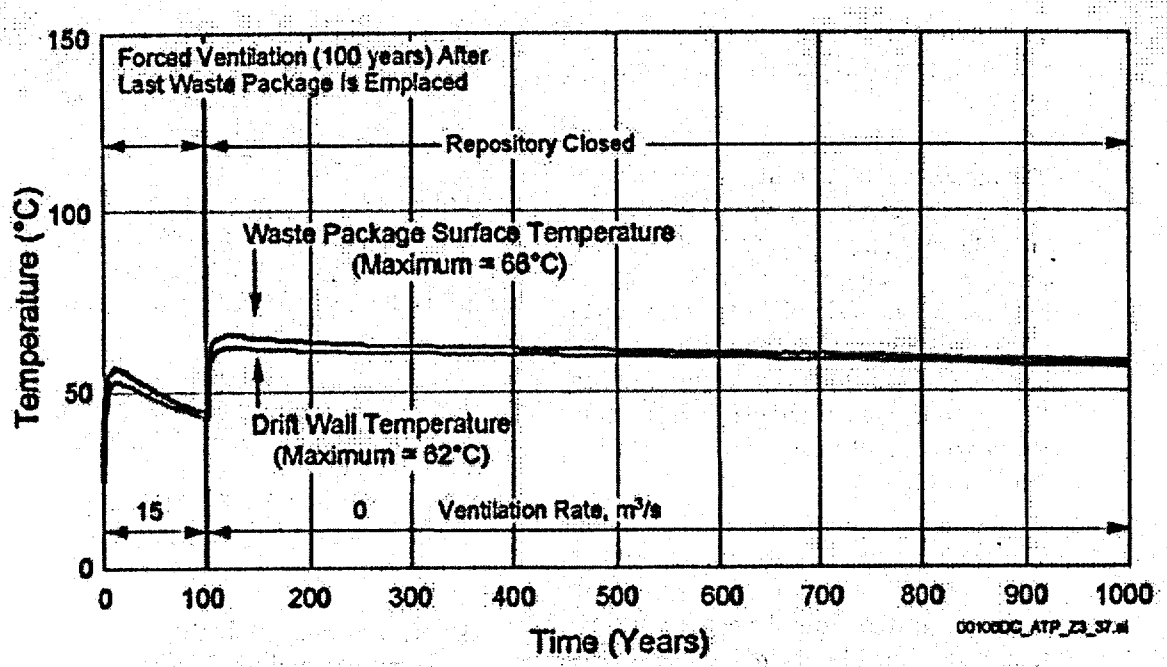


Figure 2-13. Waste Package Surface and Drift Wall Temperatures for an Operational Scenario in Which Drifts Loaded at 0.7 kW/m in the Last Year of Emplacement Operations are Actively Ventilated for 100 Years

Source: CRWMS M&O 2000o, Figure 3.

length and the total area required for emplacement; and (4) requirements for additional years of forced ventilation, maintenance, and other site and operations support.

Example Scenario 4: Extended Surface Aging with Forced Ventilation—In this example, surface aging of the hotter portion of the commercial spent nuclear fuel inventory, combined with the spacing of waste packages approximately 2 m (6.6 ft) apart within the drifts, reduces the linear thermal load to about 0.5 kW/m at emplacement. Surface aging of the hottest wastes would extend the total emplacement period from approximately 24 years to 50. However, initiation of repository operations would not be delayed because the cooler commercial spent nuclear fuel, along with the generally cooler DOE waste forms, could be emplaced immediately while the hotter commercial spent nuclear fuel cools through aging.

To meet the goal of a maximum waste package surface temperature of 85°C (185°F), forced drift ventilation would continue for approximately 125 years from the start of waste emplacement, with the last drifts loaded receiving 75 years of forced ventilation. At the end of the operating period, the repository would be closed and sealed, with no provision for extended natural ventilation. Figure 2-14 shows preliminary projections of waste package surface and drift wall temperatures over time for this scenario.

The implications of this scenario include: (1) the ability to accommodate the waste packages, with drift-to-drift spacing of 81 m (266 ft), in the areas currently characterized for a repository; (2) a preclosure period comparable to that of the higher-temperature operating mode; (3) a smaller increase in total drift length and disposal area than in Scenario 3; (4) an increase in the spacing of waste packages, which may result in thermal point loading that introduces additional complexities in modeling the thermal-hydrologic response of the potential repository; and (5) a longer emplacement period and additional fuel handling activities, which could increase the preclosure safety risk.

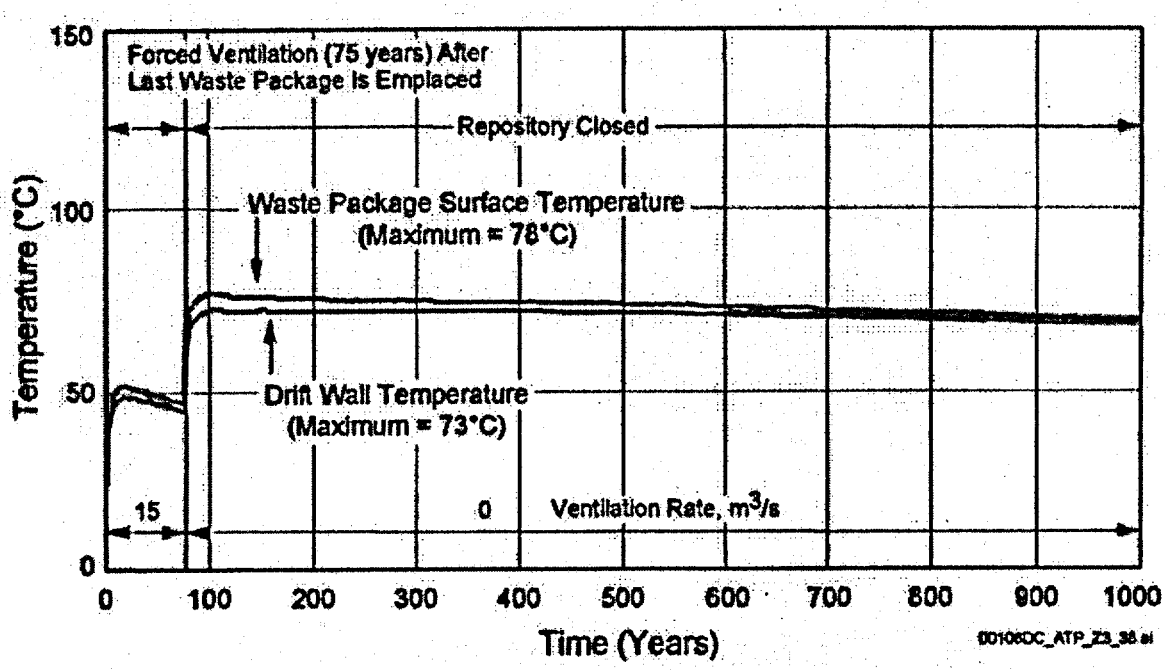


Figure 2-14. Waste Package Surface and Drift Wall Temperatures for an Operational Scenario in which Drifts Loaded at 0.6 kW/m in the Last Year of Emplacement Operations are Actively Ventilated for 75 Years

Source: CRWMS M&O 2000o, Figure 2.

2.1.5.2.3 Lower Rock Temperature and In-Drift Relative Humidity through Indefinite Natural (Passive) Ventilation

A third approach for meeting the goals of lower-temperature operating modes is to keep the temperature of the host rock below the boiling point of water and maintain in-drift relative humidity below 50 percent by incorporating passive natural ventilation into repository operations. A repository that maintains a low relative humidity is possible because of the natural characteristics of the Yucca Mountain site, including the arid environment and thick unsaturated zone.

An example lower-temperature scenario using this approach is framed around the following objectives:

1. Maintain in-drift relative humidity below 50 percent
2. Maintain the host rock temperature below the boiling point of water
3. Ensure that 70,000 MTHM of waste fits within the upper block (Figure 2-10).

Example Scenario 5: Extended Natural Ventilation—One concept for creating a lower-temperature, dry repository is to increase the ventilation duration of the current operation to approximately 75 years after the start of emplacement or 50 years after the last drift is loaded, followed by an indefinite period of natural ventilation. At the end of the forced ventilation period, the repository would stay open to allow natural ventilation to circulate cooler air for an extended period of time. Figure 2-15 shows projections of waste package surface temperatures, drift wall temperatures, and in-drift relative humidity over time (BSC 2001h).

Implications of this approach include: (1) a total excavated drift length and emplacement area comparable to the high-temperature operating mode layout; (2) the same thermal line loading,

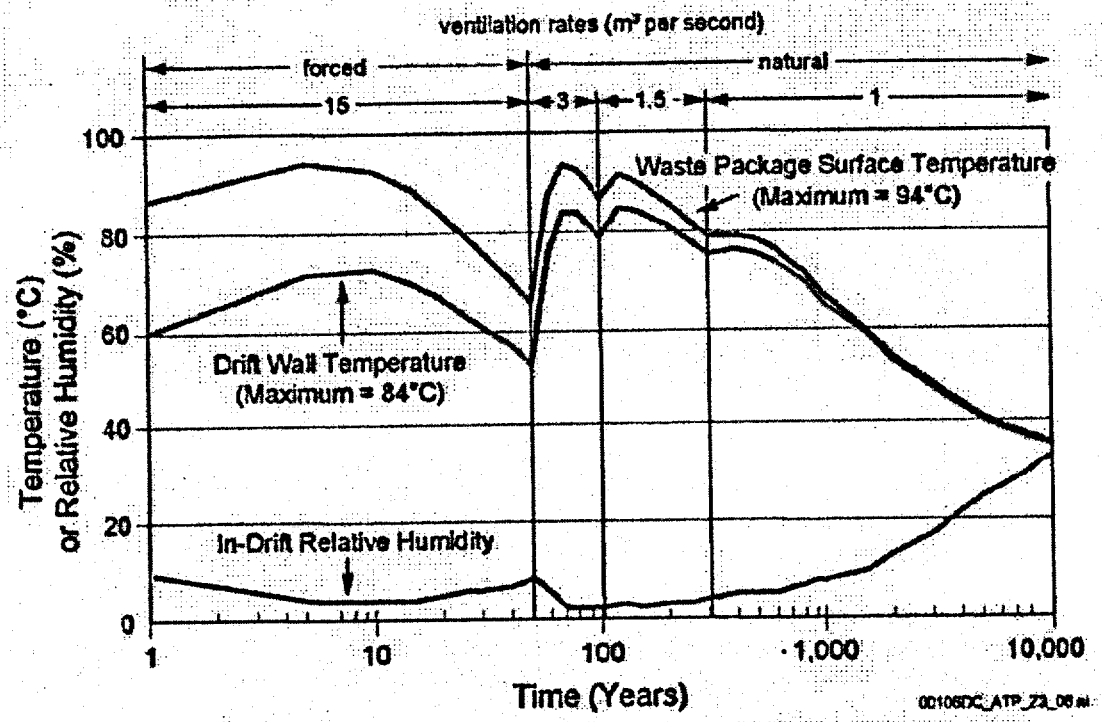


Figure 2-15. Waste Package Surface Temperature, Drift Wall Temperature, and In-Drift Relative Humidity for an Operational Scenario in Which Drifts Loaded at the End of Emplacement Operations at 1.45 kW/m are Actively Ventilated for 50 Years and then Naturally Ventilated Indefinitely

After 100 years, the relative humidity increases as the temperature declines, and moisture returns to the host rock around the drift. Source: Modified from BSC 2001k.

waste package spacing, and drift-to-drift spacing as the higher-temperature operating mode; and (3) the programmatic uncertainty associated with the indefinite period of natural ventilation.

2.1.5.3 Comparative Analysis of Alternative Lower-Temperature Operating Scenarios

Thermal management involves the consideration of complex, nonlinear relationships among many parameters of the repository system. The major determinants of peak temperatures are the age of the fuel at emplacement, the linear heat load along each drift, and the ventilation period after emplacement. Figure 2-11 shows the relationships among preemplacement aging, ventilation period, and waste package spacing for an average thermal goal of 96°C (205°F) at the drift wall (BSC 2001h). For this case, the thermal goal can be achieved with a total preclosure period of about 100 years. The results change as the temperature goal is lowered to 85°C (185°F) or below on the waste package surface; in particular, the preclosure period might be extended. With longer periods of ventilation after waste emplacement, aging spent nuclear fuel before emplacement has less impact, and the key parameters become the ventilation period and the overall areal mass loading. The areal mass loading can be varied by changing the linear thermal load in each drift, the drift spacing, or both.

The examples in Section 2.1.5.2 were selected for discussion because they illustrate the effects that varying parameters can have in achieving the objectives of lower temperature and humidity. This section summarizes the main issues associated with the principal trade-offs in the example scenarios.

2.1.5.3.1 Reducing Peak Temperatures through Extended Ventilation Period or Reduced Linear Thermal Load

Analyses of ways to achieve a below-boiling repository indicate that reducing peak temperatures even further, so that waste package surface temperature does not exceed 85°C (185°F), would require a significant reduction in the linear thermal load in each drift when the total preclosure period is about 100 years, which is the period now contemplated for open operation of a repository. Scenarios 1, 2, 3, and 4 would require a substantially larger repository area. If ventilation can be extended well beyond 100 years, meeting the goal of temperatures at or below 85°C (185°F) would not require as much expansion of the disposal area. This extended ventilation period also adds flexibility in choosing between reduced waste package loading or increased waste package spacing. The added flexibility results because, with extended ventilation, the most important factor is total areal mass loading. Scenarios 1 and 2 in Section 2.1.5.2 assume that the ventilation period is extended to approximately 325 years after the start of emplacement. Scenarios 3 and 4 assume that forced ventilation continues for about 125 years after the start of emplacement; they rely primarily on reducing the linear thermal load. These two cases represent different ways of reducing the linear load at emplacement: increasing waste package spacing and aging waste. The following paragraphs focus on the broad trade-offs between extended ventilation and reduced linear thermal load.

2.1.5.3.2 Reducing Linear Heat Load by Using Widely Spaced Large Waste Packages or Closely Spaced Small Waste Packages

The total heat load over the entire emplacement area determines the repository temperature in the long term. However, in the short term, the amount of heat per unit length of drift is more important in determining the peak temperatures of both the rock and the waste package surface. Heat per unit length of drift is the dominant factor before the heat from each drift has time to move through the rock and influence temperatures in the adjacent drifts, and before the heat-producing radionuclides with a short half-life (e.g., strontium-90 and cesium-137) have had time to decay to comparatively low levels. As illustrated by Scenarios 1 and 2 in Section 2.1.5.2, there are two ways of reducing the linear thermal load at emplacement: moving larger waste packages farther apart or using de-rated or smaller waste packages without increasing the spacing.

2.1.5.3.3 Aging Hotter Waste or Reducing the Linear Heat Load to Allow a Short Ventilation Period

To achieve a maximum waste package temperature of 85°C (185°F) with ventilation and closure periods of 100 years or less, the linear thermal load at emplacement must be reduced substantially, requiring an increase in drift length. Preemplacement aging can have a significant effect on the magnitude of the increase needed by allowing the heat-generating radionuclides with an intermediate half-life to decay before emplacement. Scenarios 3 and 4 in Section 2.1.5.2, which use roughly the same preclosure ventilation period, show the effects of using aging to reduce drift length requirements.

2.1.5.4 Other Considerations of Lower-Temperature and Lower-Humidity Operating Modes

Additional considerations associated with the various operating modes that reduce drift temperatures and relative humidity are discussed in this section.

Drip Shields—The drip shield is intended to protect the waste package from seeping water. This aspect of corrosion protection is effective early in the postclosure period and is a component of the DOE defense-in-depth strategy. In a lower-temperature design specifically engineered to remain outside the window of corrosion susceptibility, such protection may be less important or unnecessary. The benefits of drip shields in deflecting seepage and the time of their emplacement will be evaluated in light of the environments associated with the range of thermal operating modes.

Another related design issue is that some of the lower-temperature options include waste package spacing of several meters. In such cases, it may be more cost effective to have individually placed drip shields for each waste package rather than a continuously interlocked set of drip shields (BSC 2001l). The impact of these individually placed drip shields on repository performance will be the subject of future evaluations.

Repository Layout—Changes in such variables as waste package spacing or emplacement drift spacing to achieve lower-temperature operating conditions may alter the layout of the potential repository, including the total size of the repository footprint, as identified in Table 2-2. Layouts have been developed for both the primary (upper) and lower blocks characterized as areas for a potential higher-temperature operating mode, as described in *Site Recommendation Subsurface Layout* (BSC 2001d), and for a lower-temperature operating mode as presented in *Lower-Temperature Subsurface Layout and Ventilation Concepts* (BSC 2001g). The latter analysis developed layout configurations that support lower-temperature operating modes for the 70,000 MTHM and 97,000 MTHM cases, utilizing the primary and lower blocks. The analysis also presented results of parametric studies to evaluate the sensitivity of the layout configurations against operational and design options including waste package spacing, smaller waste package sizes, waste package heat output, and emplacement drift spacing.

Nonuniform Temperatures within the Drift—When the thermal loading within drifts is adjusted by increasing the distance between waste packages, the thermal load to the rock environment becomes more discretized along the drift axis, more like point loads than a continuous line load. This could result in a nonuniform temperature distribution along the axis of the drift after the repository is sealed. A significant nonuniform temperature distribution in the drift walls or within the emplacement drift could cause water or steam to move from the high heat region toward the low heat region, resulting in water dripping from the walls onto the drip shields, or water condensing on drip shields or low heat output waste packages. The rate of moisture transport decreases with a decrease in the thermal gradient.

Natural Ventilation—Another possibility for mitigating uncertainties associated with corrosion is to keep in-drift relative humidity below 50 percent by allowing longer-term natural ventilation of the repository. By taking advantage of natural ventilation, a repository can be operated so the temperature of the host rock remains below the boiling point of water while relative humidity stays below 50 percent, the value bounding the conservatively defined window of susceptibility for localized corrosion of Alloy 22.

It is illustrative to examine a scenario in which the period of natural ventilation is extended indefinitely. In example scenario 5, the repository is operated through the periods of emplacement and for 75 years after emplacement, during which the emplacement drifts are actively ventilated, after which the repository is allowed to ventilate naturally for an extended period of time. Figure 2-15 depicts nominal waste package surface temperature, emplacement drift wall temperature, and in-drift humidity as a function of time for this scenario, as predicted by two-dimensional models (BSC 2001h). In this case, the natural ventilation period was allowed to extend indefinitely. Natural ventilation flow rates were modeled at 3 m^3/s per drift for the first 50 years, at 1.5 m^3/s per drift for the next 200 years, and at 1 m^3/s per drift thereafter. In calculating relative humidity, an infiltration flux of 60 mm/yr (2.4 in./yr) was assumed over the emplacement drifts. (This infiltration flux is ten times greater than the present-day mean infiltration rate over the repository footprint.) As Figure 2-15 shows, the peak drift wall temperature for this scenario is projected to stay below 90°C (194°F). Because of the ample carrying capacity of the dry desert air, the average relative humidity in the drifts does not exceed 30 percent until waste package temperatures drop below 40°C (104°F).

Natural Analogues—Natural analogues support the concept that a deep geologic repository in the unsaturated zone would keep waste dry and isolated, particularly through natural ventilation. Dry environments, in which fragile cultural arti-

facts have been preserved for millennia, could provide a hospitable environment for more robust materials. Prehistoric paintings, for example, survive in numerous caves that are analogous to a deep geologic repository. Water tends to flow around caves and tunnels in the unsaturated zone, depending partly on the size of the openings: the smaller the size of the cave, the more likely water will flow around it. Most unsaturated zone caves with highly preserved artwork have openings larger than the 5.5-m (18-ft) diameter proposed for emplacement drifts. This suggests the drifts may remain even drier than caves, which have stayed dry enough to preserve ancient paintings (Stuckless 2000, pp. 2 to 4). A cooler repository would be more closely related to these natural analogues than a higher-temperature repository.

The oldest authenticated examples of Paleolithic (older than 10,000 years) human cave paintings are the Chauvet paintings in southeastern France. These recently discovered cave paintings include images of the mammoth, long extinct, and the rhinoceros, long absent from Europe. The intact Chauvet paintings are more than 30,000 years old and still survive in a subhumid climate with annual precipitation ranging up to roughly 80 cm (31 in.); average annual precipitation at Yucca Mountain is less than 20 cm (8 in.). On the cave's ceiling, even the soot from later human use of oil lamps has remained, despite having been deposited some 26,000 years ago (Stuckless 2000, pp. 2 to 4).

Spirit Cave, about 75 miles east of Reno, Nevada, is another example of a natural analogue to a repository. Several small burial pits were discovered in this cave; in one of these pits, archaeologists found an almost completely intact human body, mummified by the dryness of the climate. Subsequent analyses, including carbon-14 testing, have shown the body to be that of a 45- to 55-year-old male who died around 9,400 years ago. His scalp was completely intact, including a small tuft of hair. He wore a breechcloth of fiber and was wrapped in mats woven of fibers from tule leaves, all highly preserved (Barker et al. 2000, Section 4).

2.2 REPOSITORY SURFACE FACILITIES

The design described in this report focuses on protecting the health and safety of both repository workers and the public. To meet these requirements, an integrated set of surface facilities is being designed. Operation of a repository would involve a number of distinct but interrelated waste activities and functions; the major ones include receiving, handling, and packaging. The following design description discusses the surface facilities for a potential repository for spent nuclear fuel and high-level radioactive waste, based on current concepts documented in *Engineering Files for Site Recommendation* (CRWMS M&O 2000p) and *WHB/WTB Space Program Analysis for Site Recommendation* (CRWMS M&O 2000q).

The surface facilities would be located in the North Portal Repository Operations Area, the South Portal Development Area, and the Surface Shaft Areas. Figure 2-16 shows the general surface layout of the North Portal Repository Operations Area and South Portal Development Area. The Surface Shaft Areas, where the ventilation shafts and fans would be located, would be on the crest of Yucca Mountain. Together (but excluding the area for the solar power arrays), these areas would cover more than 105 acres of land, on which at least 30 structures would be built to house the operations and services needed for safe and effective repository operations (CRWMS M&O 2000p, Attachment II, Section 2.11.1).

The North Portal Repository Operations Area is logically segregated into the Radiologically Controlled Area, the balance-of-plant area, and the site services area. The Radiologically Controlled Area comprises all facilities necessary to receive, package, and emplace waste in the repository. The balance-of-plant area comprises general infrastructure facilities, such as administration, emergency management (medical and fire), and motor pool and fleet services. The site services area comprises general parking and the visitor center.

The South Portal Development Area would support continuing construction of a repository, even as the North Portal Repository Operations Area accepts and prepares waste for underground emplacement

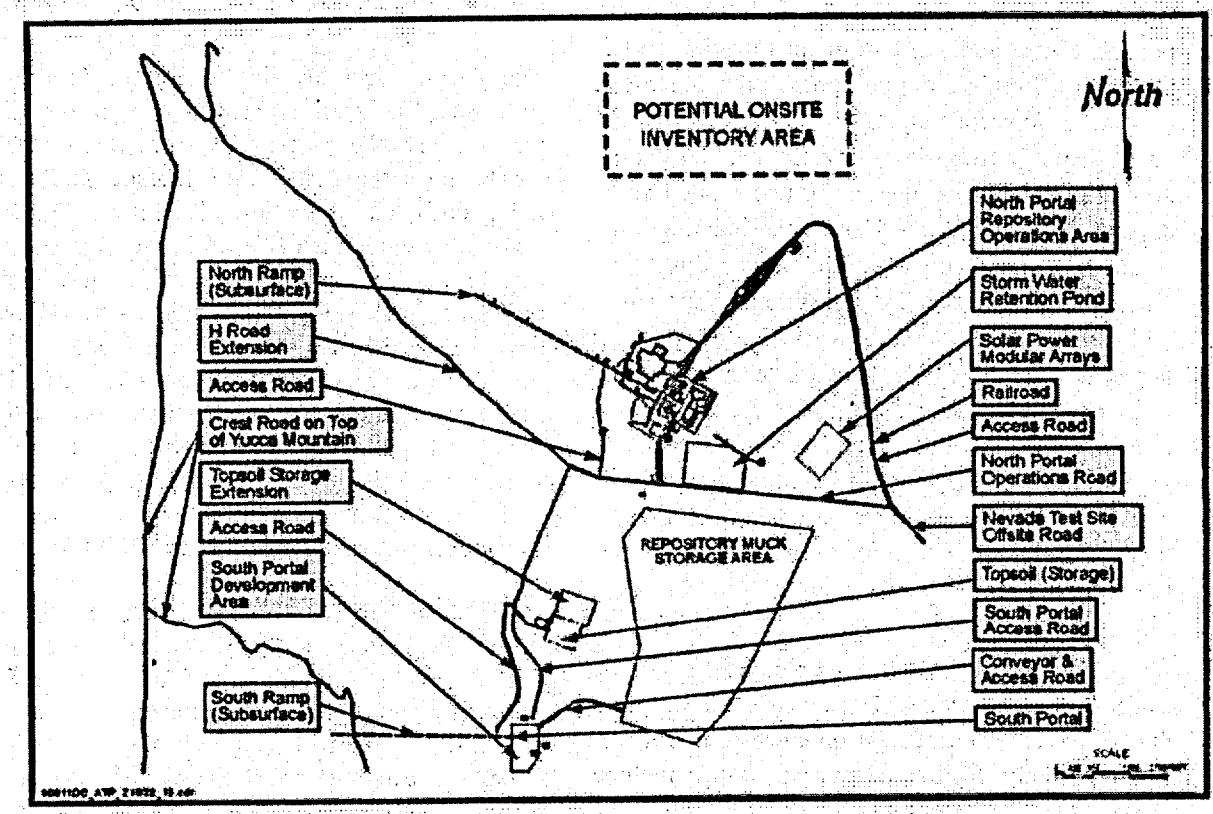


Figure 2-18. Repository Overall Site Plan
Surface facilities would be developed to support operations and construction activities at the North and South Portals, respectively. Roadways would be constructed to link these areas with supporting surface areas. Source: Modified from CRWMS M&O 2000p, Figure I-19.

in the first emplacement drifts. This section focuses on the proposed design and development of three primary structures at the North Portal surface facility. Section 2.3 addresses other areas more closely associated with subsurface development and emplacement activities.

Consistent with 10 CFR Part 63 (66 FR 55732), a set of monuments extending at least 6.2 m (20 ft) above the surface would be constructed prior to the beginning of postclosure to identify the geologic repository area.

In addition to the design requirements driven by the fuel blending inventory strategy (described in Section 2.2.1), continual emphasis on environmental, industrial, and radiological health and safety criteria will be designed into all of the surface facilities and systems. The ALARA radiological health and safety design and operational criteria are discussed in Section 2.2.5. Operations at the North Portal Repository Operations Area are described in Section 2.2.2. Details of these surface facilities and systems, and the designers' emphasis on health and safety, are presented in Section 2.2.4.

2.2.1 Fuel Blending Inventory Strategy

Spent nuclear fuel and high-level radioactive waste arriving at the repository would be in solid form but in a variety of types and sizes. Hence, the materials would arrive in a variety of transportation casks, all certified for use by the NRC.

Commercial spent nuclear fuel would arrive as individual fuel assemblies placed directly into transportation casks, or in dual-purpose canisters, which would have to be opened to remove the fuel

assemblies. DOE spent nuclear fuel would arrive in disposable canisters. Because of the variety of waste forms, a number of different designs for disposal containers (called waste packages once they are loaded, sealed, and certified) would be needed. Figure 2-17 depicts the anticipated types of transportation casks, waste forms, and disposal containers (CRWMS M&O 2000p, Attachment II, Section 1.1.3.1).

The radioactive decay process generates heat. The concentration of the particular isotopes present vary among the different waste forms, so different waste forms generate different amounts of heat. The design and operating mode described in this report were developed to meet established temperature limits for the higher-temperature operating mode (i.e., a maximum heat output of 11.8 kW for all waste packages) (CRWMS M&O 2000p, Attachment II, Section 1.1.1.4).

The limit on heat output from individual waste packages imposes special considerations for operations. The strategy for controlling heat output for the waste packages is to load low heat output waste with high heat output waste to balance total waste package heat output. This process, called "fuel blending," applies only to commercial spent nuclear fuel.

Some fuel assemblies must be placed into fuel blending inventory until they generate less heat from radioactive decay, or until additional low heat output fuel assemblies arrive for blending. Fuel assemblies would stay in inventory until they are selected for blending. By carefully planning and implementing a fuel-blending procedure, the heat

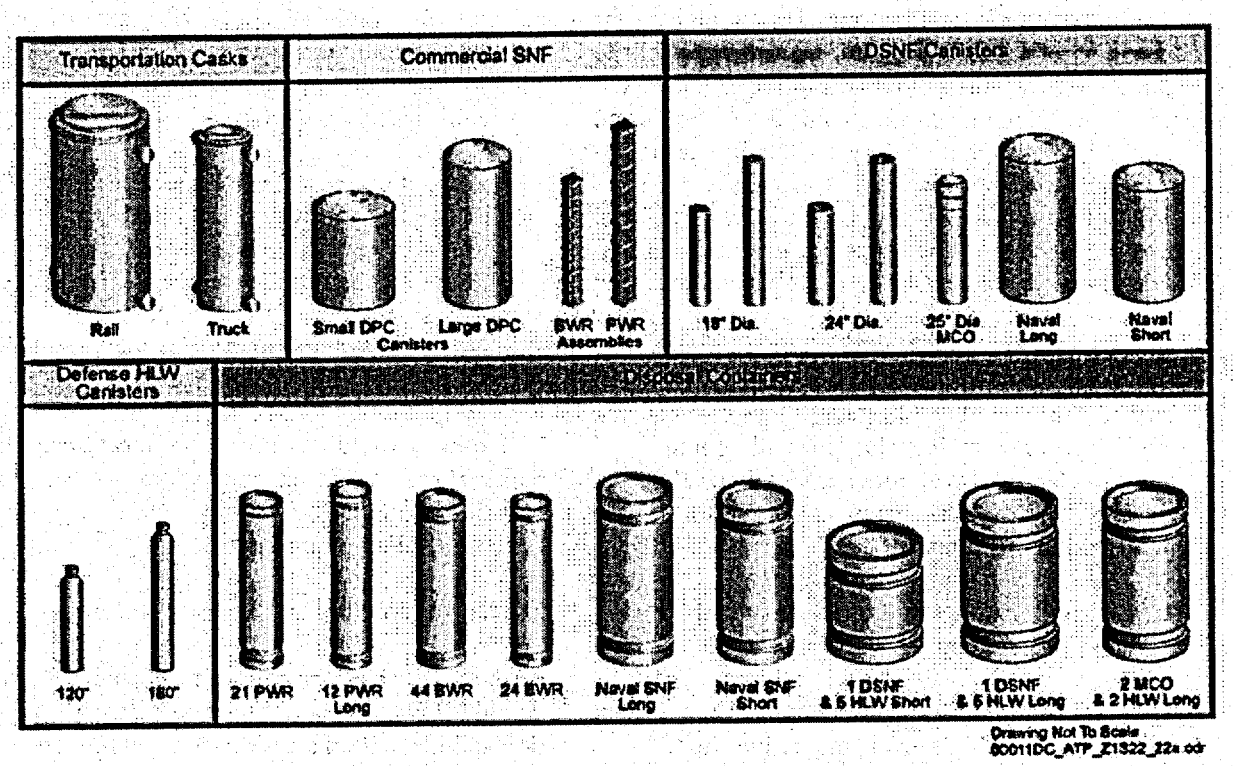


Figure 2-17. Casks, Containers, and Waste Forms Handled at the Surface Facility
Items handled at the surface facility include large and small transportation casks containing the waste forms and disposal containers. Dimensions for all waste forms and containers (except the transportation casks) are provided in Section 3. SNF = spent nuclear fuel; DSNF = DOE spent nuclear fuel; DPC = dual-purpose canister; HLW = high-level radioactive waste; PWR = pressurized water reactor; BWR = boiling water reactor; MCO = multi-canister overpack. Source: Modified from CRWMS M&O 2000p, Figure I-2.

output of the waste packages can be limited and optimized without increasing the number of waste packages and the amount of fuel blending inventory unnecessarily. The tracking, recording, and archiving of waste forms would be handled by the operations monitoring and control system (Section 2.2.4.2.12).

Engineers developed a computer model to estimate the design amount of fuel blending inventory space needed. Computer model simulations helped form the present configuration of the Waste Handling Building (CRWMS M&O 2000p, Attachment II, Section 1.1.1.3). Results indicate that, for the 11.8-kW heat output limit for the waste package, a fuel blending inventory capacity of approximately 5,000 MTHM, or 12,000 spent nuclear fuel assemblies, will be needed. This simulation also showed the surface facilities will need an inventory of approximately 40 small canisters (containing DOE spent nuclear fuel or high-level radioactive waste) for the canister transfer system. This small canister inventory is needed to ensure that a waste package containing five high-level radioactive canisters and one DOE spent nuclear fuel canister can be loaded using the canister inventory on hand. The inventory size accounts for the fact that these canisters will not always arrive at a ratio of five high-level waste canisters to one spent nuclear fuel canister. Additionally, the simulations established the need for two assembly transfer lines and one canister transfer line (CRWMS M&O 2000p, Attachment II, Section 1.1.4.1). The use of fuel blending inventory in the assembly transfer system ensures that any disposal container loaded with commercial spent nuclear fuel will not exceed a thermal output of 11.8 kW for the design and operating mode described in this report.

2.2.2 Operations in the North Portal Repository Operations Area

This section describes the movement of spent nuclear fuel and high-level radioactive waste forms through the North Portal Repository Operations Area. It presents the interrelationship of the systems, equipment, and facilities for receiving, preparing, packaging, and transporting these waste forms. It also provides information on how the waste would arrive at the repository, how the systems for handling the different waste forms would operate, and how secondary low-level radioactive waste would be handled.

2.2.2.1 Waste Receiving Operations

Spent nuclear fuel and vitrified high-level radioactive waste would be transported to the repository in NRC-certified transportation casks by U.S. Department of Transportation-licensed cask transportation contractors. The waste would be transported by rail and/or road to the North Portal security station, where personnel will verify the shipping manifests, then inspect and survey the cask and its carrier. After the cask and its carrier enter the Radiologically Controlled Area, they would be stationed in parking areas designated for either truck carriers or rail carriers. When the cask is scheduled for processing, a site prime mover (transporter) would move the cask and carrier to the Carrier Preparation Building. Figure 2-18 presents a simplified flow diagram of waste receiving operations.

Inside this building, workers:

- Retract or remove personnel barriers
- Retract or remove impact limiters
- Survey the cask surface for radiation
- Decontaminate cask surfaces (if necessary)
- Measure the cask's temperature
- Install any required special tools or devices (e.g., cask-lifting attachments).

The casks would be taken on the carrier to a parking area to await movement to the Waste Handling Building carrier bay according to operations scheduling requirements.

2.2.2.2 Waste Handling Operations

At the carrier bay, the carrier/cask handling system would lift the transportation cask to a vertical position and place it on a cask transfer cart. Depending on the cask's contents, the cart would move to one of two transfer systems. Casks that contain disposable containers (e.g., DOE canisters that would not be opened but transferred, as is, directly into a disposal container) would go to the canister transfer system. Casks that contain commercial

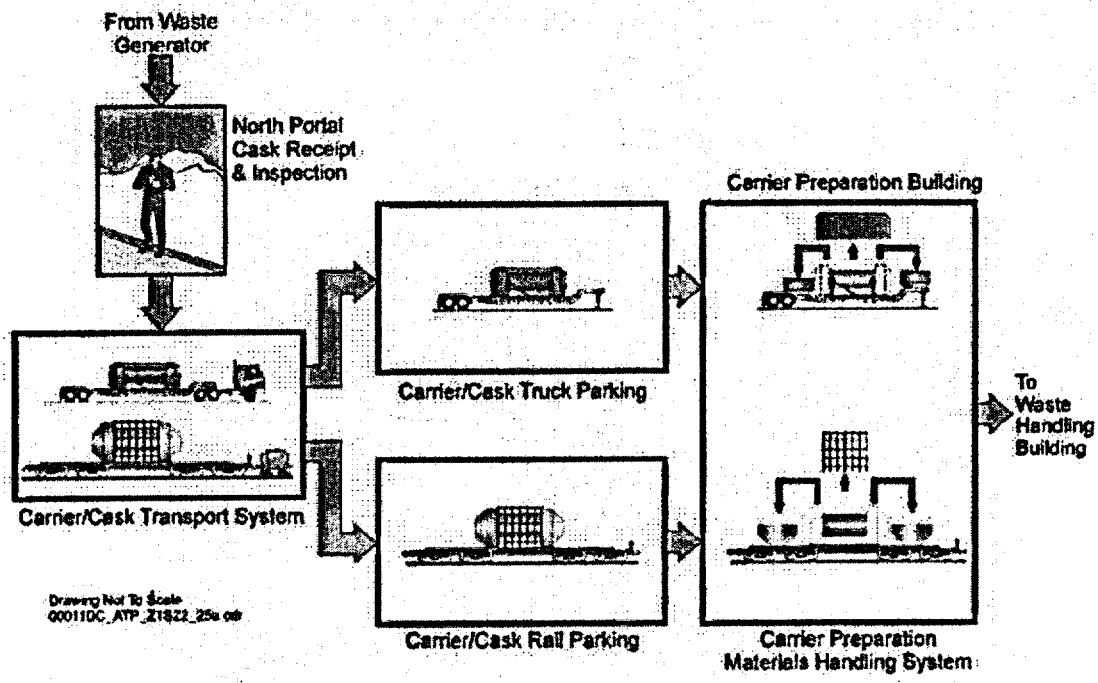


Figure 2-18. Waste Receiving Operations
Waste shipments by road and rail would be stopped at the security gate for inspection. Offsite prime movers would be decoupled from the carriers, and site prime movers would be attached. Carriers may be staged temporarily within the controlled security area until ready for processing in the Carrier Preparation Building, where the transportation casks would be made ready for transfer to the Waste Handling Building. Source: Modified from CRWMS M&O 2000p, Figure I-18.

spent nuclear fuel in dual-purpose canisters or individual fuel assemblies would go to the assembly transfer system. Figure 2-19 is a flow diagram of Waste Handling Building operations (CRWMS M&O 2000p, Attachment II, Sections 1.1.1 through 1.1.5).

The Waste Handling Building would have one canister transfer line that moves the disposable fuel canisters through the building to prepare the waste for emplacement in the repository. The system would move arriving casks through an air lock on a transfer cart into a cask preparation area. Once a cask arrives inside the cask preparation area, workers would use remotely operated equipment to vent and sample gases from the cask, remove the lid bolts, and open the cask. An overhead crane would move the cask to a transfer cart, which would take the cask to a shielded transfer area. Inside the transfer area, machines would remove the canister from the cask. The canister may go directly into a disposal container for repository emplacement or to a holding rack for later placement in a disposal container. Another transfer cart would move loaded disposal containers to the disposal container handling system. A transfer cart would move the empty transportation casks back to the cask decontamination area, where they would be surveyed and decontaminated, if required, before return shipment. From the decontamination area, casks would be moved to the carrier/cask handling system, which would place them back on a transporter. The empty cask and cask transporter would return to the Carrier Preparation Building to be readied for offsite shipment.

The Waste Handling Building would have two assembly transfer lines. Each line would operate independently to handle waste throughput and support maintenance operations. The assembly transfer process begins by moving the cask on a transfer cart through the air lock into the cask preparation area. Once inside the cask preparation area, workers would use remotely operated equipment to inspect, vent, and cool the cask and remove the cask lid bolts. A large overhead crane would lift the

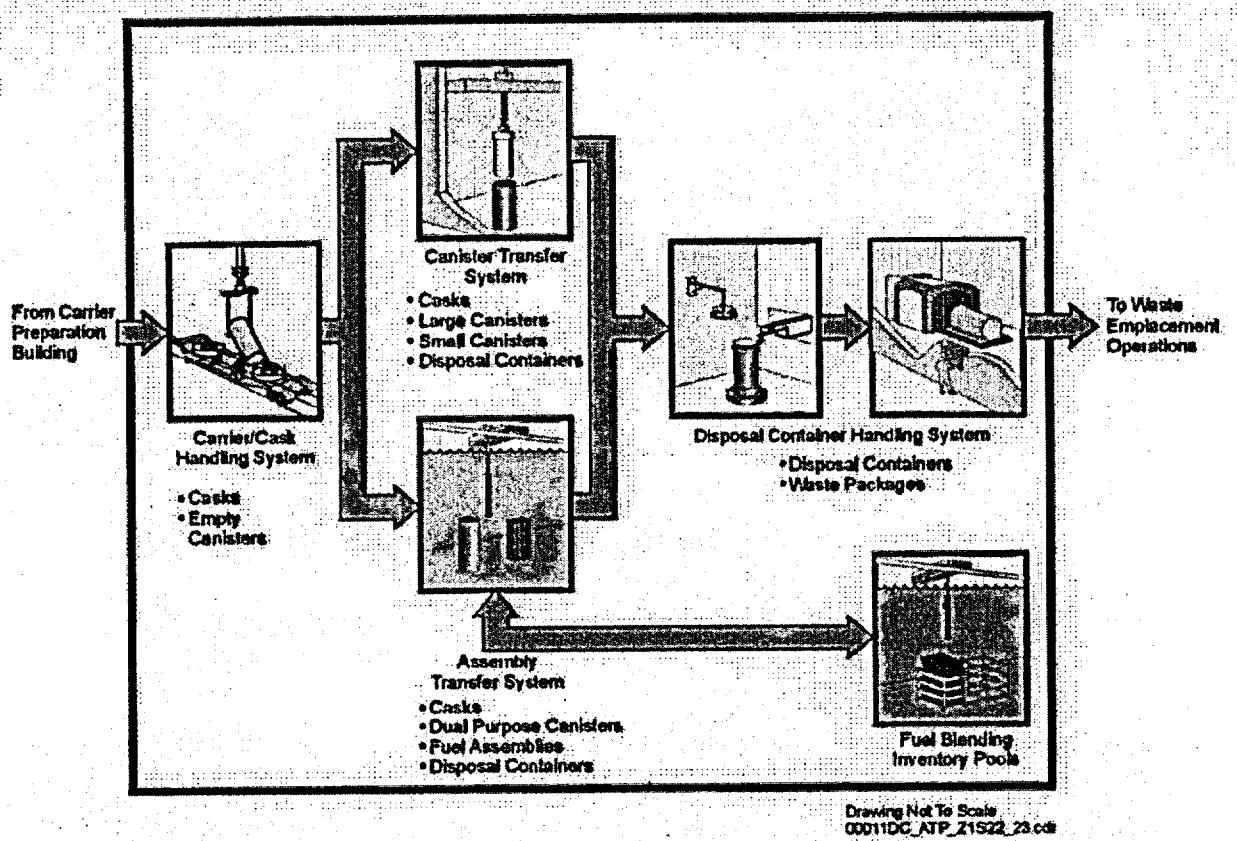


Figure 2-19. Waste Handling Operations
In the Waste Handling Building, transportation casks would be transferred from their carriers to one of two transfer systems: the canister transfer system for spent nuclear fuel and high-level radioactive waste already packaged in canisters suitable for direct insertion into disposal containers; or the assembly transfer system for spent fuel not packaged in containers suitable for direct insertion into disposal containers. After loading, the disposal containers would be welded closed and the welds inspected. After acceptance of the welds, the disposal container is referred to as a waste package. Source: CRWMS M&O 2000p, Figure I-1.

casks and place them in a cask unloading pool, where fuel-handling machines would open the casks and unload the fuel assemblies. If the cask contains dual-purpose canisters, they would be removed and placed in an overpack, where the top of the canister would be cut off. The system would move the empty casks and dual-purpose containers back out through the cask decontamination area. The fuel-handling machines would transfer the fuel assemblies, one at a time, to a holding pool, where they would be placed in assembly baskets. A transfer cart would move the baskets containing the fuel assemblies underwater from the assembly holding pool through a transfer canal to a fuel-blending inventory pool. When a fuel assembly is selected from the fuel inventory pool for packaging, a transfer cart would move it underwater back through the fuel blending pool to an inclined transfer canal and onto a cart that connects to the assembly drying area.

After fuel assemblies arrive at the assembly drying area, a fuel-handling machine would transfer them into one of two drying vessels. After drying, the system would retrieve the assemblies and transfer them, one at a time, to a disposal container. The empty assembly baskets would be returned to the pool area for reuse. After loading, workers would purge the disposal container with inert gas and temporarily seal it for transfer to the disposal container handling system.

The disposal container handling system would receive loaded disposal containers from both the

canister transfer system and the assembly transfer system. Each disposal container would again be purged with inert gas, after which the container's lids would be welded and the welds inspected. If the welds meet inspection criteria, the sealed disposal container would be reclassified as a waste package. A crane would transfer the waste package to the transporter loading area, where it would be decontaminated and placed on a pallet, then on a transporter for emplacement in the subsurface repository. Table 2-3 gives the preliminary performance specifications for this equipment.

2.2.2.3 Treatment of Low-Level Radioactive Waste from Repository Operations

Operations at a repository—receiving, handling, and repackaging commercial and DOE spent nuclear fuel and high-level radioactive waste—would generate secondary low-level radioactive waste. Some of this waste may be defense-related transuranic waste. In this report, any transuranic and low-level waste is referred to collectively as low-level waste. The majority would be generated in the Waste Handling Building; smaller quantities may be produced in the Waste Treatment Building, where secondary low-level waste is processed. Operations in the Carrier Preparation Building are not expected to generate any significant waste quantity. All secondary waste would be monitored at the point of generation, and administrative controls would direct its disposition. The design of the repository would include a processing system, which would minimize the volume of liquid and solid waste. To the extent applicable, it would be properly treated and packaged for shipping and disposal off the repository site.

Some hazardous wastes may also be generated during repository operations. The use of hazardous constituents will be reviewed and controlled so that the generation of hazardous wastes is minimized. When hazardous wastes are commingled with

Table 2-3. Preliminary Crane and Lifting Machine Performance Specifications

Equipment Description	Quantity	Capacity	Span meters (ft)
Carrier Preparation Building Bridge Crane	2	10 tons	17.68 (58)
Carrier Bay Bridge Crane	1	160 tons	23.77 (78)
Cask Unloading Area Bridge Crane	2	160 tons Aux. Hoist: 25 tons	13.11 (43)
Assembly Handling Cell Bridge Crane	2	15 tons Aux. Hoist: 5 tons	10.85 (35.6)
Nonstandard Fuel Pool Overhead Crane	1	30 tons	10.97 (36)
Fuel Basket Storage Pool Overhead Service Crane	4	20 tons	15.24 (50)
Canister Transfer Cell Bridge Crane	1	65 tons	10.67 (35)
Offnormal Canister Handling Cell Bridge Crane	1	15 tons	9.14 (30)
Disposal Container Handling Cell Bridge Crane	2	150 tons Aux. Hoist: 10 tons	22.56 (74)
Empty Disposal Container Preparation Bridge Crane	1	60 tons Aux. Hoist: 10 tons	25.6 (84)
Welder Maintenance Bridge Crane	1	20 tons	10.97 (36)
Waste Package Horizontal Lifting Machine	1	100 tons	6.71 (22)
Waste Package Transporter Load Cell Bridge Crane	1	10 tons	10.36 (34)
Waste Package Remediation System Cutting Machine	1	4 tons	10.67 (35)
Waste Package Remediation System Manipulator and Hoist	1	4 tons	10.67 (35)
Assembly Transfer System and Canister Transfer System Equipment Transfer Corridor Bridge Crane	1	50 tons	14.63 (48)
Disposal Container Handling and Waste Package Remediation System Equipment Transfer Corridor Bridge Crane	1	50 tons	17.37 (57)

Source: CRWMS M&O 2000r.

radioactive wastes, the result is mixed waste. The generation of hazardous and mixed wastes will be minimized at the repository. However, if it is generated, it would be collected and repackaged for shipment to an approved offsite location for treatment and disposal. The packaged mixed waste would then be staged in the Waste Treatment Building for transport offsite to an approved facility. No low-level or hazardous waste would be disposed in the potential repository. Hazardous wastes that are not mixed wastes would also be packaged for shipment offsite to an approved treatment disposal facility. Wet solid low-level waste (e.g., spent ion exchange resins and filtration materials) generated in the Waste Handling Building would be collected and packaged for disposal. Then it would be transferred in containers to the Waste Handling Building for shipment off the site for disposal. Any other solid low-level waste generated in the Waste Handling Building that does not exceed the radioactivity limit for the Waste Treatment Building would be collected at its point of origin and transferred to the Waste Treatment Building to be processed and packaged for shipping and disposal at an approved low-level radioactive waste facility. Solid waste that exceeds Waste Treatment Building administrative activity limits would be packaged at the source of generation for shipment and disposal off the repository site. Spent dual-purpose containers would be volume-reduced and disposed at a suitable low-level radioactive waste disposal site. Recycling of spent disposal containers for recovery of metal content will be examined in future design activities. The Waste Treatment Building would be large enough to hold all the necessary equipment for processing the proposed maximum annual secondary-waste generation rate on a regular operating schedule (CRWMS M&O 2000p, Attachment II, Section 1.2).

2.2.3 North Portal Repository Operations Area Layout

This section describes the overall orientation, configuration, and general construction features of the repository surface facilities.

The North Portal Repository Operations Area would include a Radiologically Controlled Area and a balance-of-plant area (Figure 2-20). The Radiologically Controlled Area, also known as the protected area, is where the waste would be received from offsite transportation carriers and placed in waste packages for disposal. The balance-of-plant area includes all structures and systems supporting repository operations that are not encompassed by the Radiologically Controlled Area. An additional area of 350 acres is available to stage retrieved waste, should the need arise; this area should be sufficient to stage all the waste that may be emplaced in the potential repository (CRWMS M&O 2000p, Attachment II, Section 2.11.3.3).

A buried storm drainage collection system would contain water runoff from the Radiologically Controlled Area. The drainage system would also prevent spillage over the fill slopes and runoff from the balance-of-plant area. A retention pond would be built to prevent storm water pollution (CRWMS M&O 2000p, Attachment II, Section 2.11.3.1).

Except for its north edge, the North Portal pad would be above the flood-prone area of the probable maximum flood. Two open channels around the perimeter of the pad would protect the North Portal from water flow. The operating floor of the Waste Handling Building is 0.5 m (1.5 ft) above the maximum elevation of the flood stage that intersects the building to allow for freeboard (CRWMS M&O 2000p, Attachment II, Section 2.11.3.1).

2.2.4 Surface Systems and Structures

This section includes detailed descriptions of the layout, support structures, systems, and utilities of each major surface facility. Design objectives and criteria are derived from NRC regulations, industry standards for nuclear facilities, and DOE policy. For example, NRC regulatory guides were used in preparing the System Description Document design requirements. NRC guides used in developing the System Description Document for the Waste Handling Building included:

- Regulatory Guide 3.49, *Design of an Independent Spent Fuel Storage Installation (Water-Basin Type)*

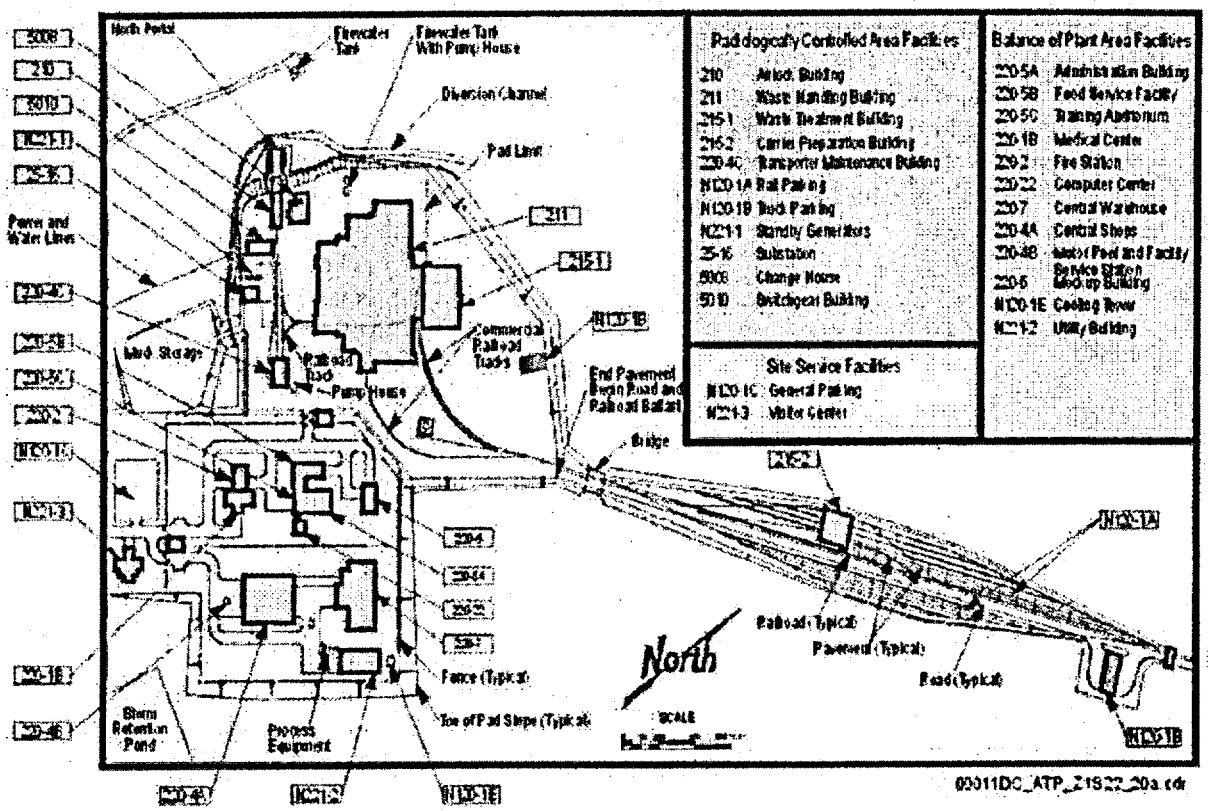


Figure 2-20. North Portal Repository Operations Area Site Plan
The North Portal Repository Operations Area is defined by three distinct use areas: the Radiologically Controlled Area, the balance-of-plant area, and site service facilities. All radiologic materials would be handled only in the Radiologically Controlled Area. RCA = Radiologically Controlled Area; BOP = balance-of-plant. Source: Modified from CRWMS M&O 2000p, Figure I-20.

- Regulatory Guide 1.13, *Spent Fuel Storage Facility Design Basis*
- Regulatory Guide 1.102, *Flood Protection for Nuclear Power Plants*
- Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable*
- *Human–System Interface Design Review Guideline* (NUREG-0700) (NRC 1996).

System Description Documents for the surface facilities would be maintained and updated as required over the life of the facility or system. Codes and standards were chosen based on a review of relevant federal laws and regulations, as well as applicable industry codes, standards, and good engineering practices. Three structures and systems in the surface facilities have been classified as QL-1 (YMP 2000b):

- The Waste Handling Building structure
- The assembly transfer system
- The canister transfer system.

QL-1 systems include those structures, systems, and components whose failure could directly result in a condition adversely affecting public safety. These items have a high safety or waste isolation significance. For this reason, QL-1 systems are discussed in greater detail than systems that are not classified as QL-1. Systems that are not classified as QL-1 are defined in Section 5.2.5, where their functions important to safety are likewise described.

The Carrier Preparation Building, Waste Handling Building, and Waste Treatment Building are described in this section, as are the processes that would occur within each and the equipment that would support such activities. Some of the operations discussed, along with the equipment needed to perform them, are:

- Removing the waste forms from their transportation casks
- Cleaning and decontaminating the casks for reuse
- Loading the waste forms into disposal containers
- Sealing the disposal containers
- Repairing defective disposal containers and waste packages
- Controlling radiation exposures to personnel
- Processing, cleaning, recycling, and solidifying low-level radioactive waste for shipment and disposal off the repository site.

2.2.4.1 Carrier Preparation Building

The Carrier Preparation Building, to be located at the North Portal pad, would support preparation of the waste transportation casks before they enter the Waste Handling Building. Planned as a steel-framed structure, the building would be approximately 58 m (190 ft) long, 37 m (120 ft) wide, and 14 m (46 ft) high. The operations area, divided into two identical carrier operations bays, would accommodate four parallel rail tracks/roadways for passage of both rail and truck carriers. Each bay would have two rail/truck lines, separated by a dual-function work platform and equipment laydown area, a bridge crane, and a bridge-mounted manipulator. The transportation carriers would enter and exit the building through one of eight remotely operated roll-up doors (CRWMS M&O 2000p, Attachment II, Section 1.3).

2.2.4.1.1 Carrier Preparation Building: Architectural and Structural Features

The Carrier Preparation Building would be an on-grade, one-story, high-bay, steel-framed structure, enclosed with insulated steel roof and wall panels. The interior framing would be of light-gauge steel and easily decontaminated panels. The foundations would consist of reinforced concrete spread footings, to support the building's columns, and continuous reinforced concrete mat foundations, to support the railroad tracks. To mitigate vibrations from carrier movement, the spread footings would be separated from the mat foundations. The building's columns would support two bridge cranes running the length of the building, each of which would span a gantry crane for servicing the tracks (CRWMS M&O 2000p, Attachment II, Section 1.3.1.3.1).

2.2.4.1.2 Carrier Preparation Building: Material Handling System

The material handling system in the Carrier Preparation Building would receive and inspect shipping casks from the carrier/cask transport system, then prepare the casks for unloading in the Waste Handling Building (Figure 2-21). Four parallel tracks/roadways would permit the passage of both truck and rail carriers. The two outer tracks/roadways would serve incoming carriers from the rail yard or truck-parking area, and the two inner tracks/roadways would serve outgoing carriers (CRWMS M&O 2000p, Attachment II, Sections 1.3.2.1, 1.3.1.3.1).

Receiving operations include:

- Performing a radiation survey of the carrier and the transportation cask
- Removing or retracting the personnel barrier(s)
- Sampling the cask exterior for contamination
- Measuring the cask's temperature
- Removing or retracting the cask impact limiters
- Installing the cask's lifting attachments (if any).

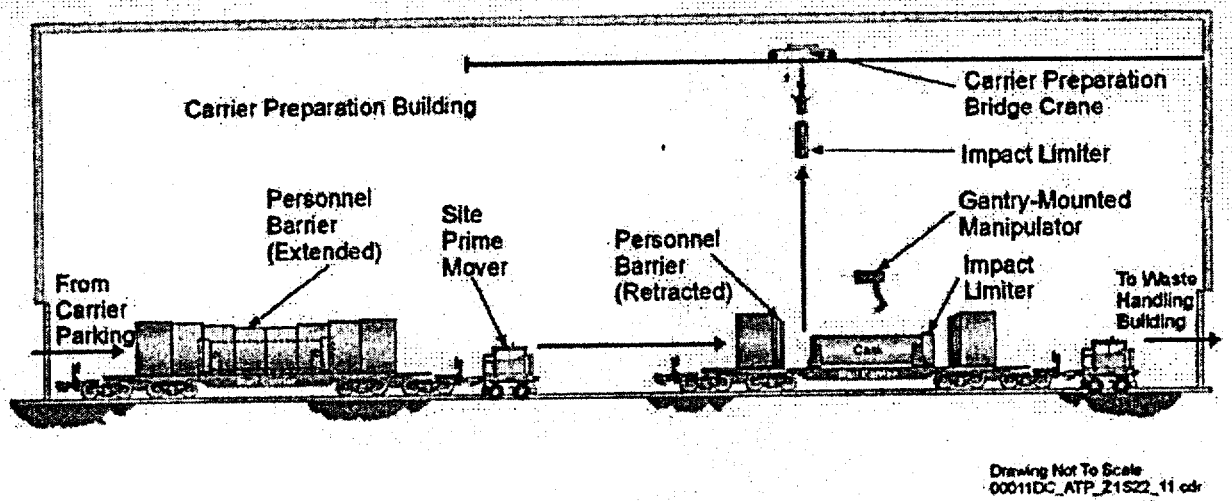


Figure 2-21. Carrier Preparation Building Materials Handling System
The materials handling system uses manual and remote equipment to prepare incoming road and rail transportation casks for offloading in the Waste Handling Building. The system removes/retracts personnel barriers which prevent inadvertent touching of the thermally hot casks, removes the impact limiters from both ends of the casks, and adjusts any tie-down bolts for easy removal so the casks can be offloaded in the Waste Handling Building. Source: Modified from CRWMS M&O 2000p, Figure I-16.

Shipping operations for carriers/casks leaving the repository would include:

- Removing the cask's trunnions (if required)
- Checking the cask's tie-downs
- Installing the cask's impact limiters
- Performing another radiation survey of the cask
- Installing the personnel barriers.

Unloaded casks would undergo the same series of operations as loaded casks, except in reverse (CRWMS M&O 2000p, Attachment II, Section 1.3.2.1).

One 10-ton capacity, remotely operated overhead bridge crane spanning 17.7 m (58 ft) and one remotely operated manipulator would serve each pair of preparation lines. Operations would support both manual and remote handling of carrier/cask materials. Having both manual and remote handling options would improve the maintenance of facilities and equipment, permit the replacement of interchangeable components, and lower radiation doses to workers. The building's support equipment would include tools and fixtures for removing and installing personnel barriers, impact limiters, cask lifting attachments, and cask tie-downs (CRWMS M&O 2000p, Attachment II, Section 1.3).

The material handling system would interface with the cask/carrier transport system to move carriers to and from the building. The Carrier Preparation Building would house all necessary equipment and systems (e.g., facility, utility, safety, auxiliary) to support its operations and protect personnel (CRWMS M&O 2000p, Attachment II, Section 1.3.2.1). No spent nuclear fuel or high-level radioactive waste casks would be lifted in the Carrier Preparation Building. The waste forms would be protected by the transportation casks.

2.2.4.2 Waste Handling Building

The Waste Handling Building would provide the space, layout, structures, and built-in systems to support waste handling operations, loading and holding of waste packages, and inventory of unused disposal containers (CRWMS M&O 2000s). This complex would also provide a safe environment for personnel and equipment involved in waste handling operations. Figures 2-22 and 2-23 show the Waste Handling Building in plan

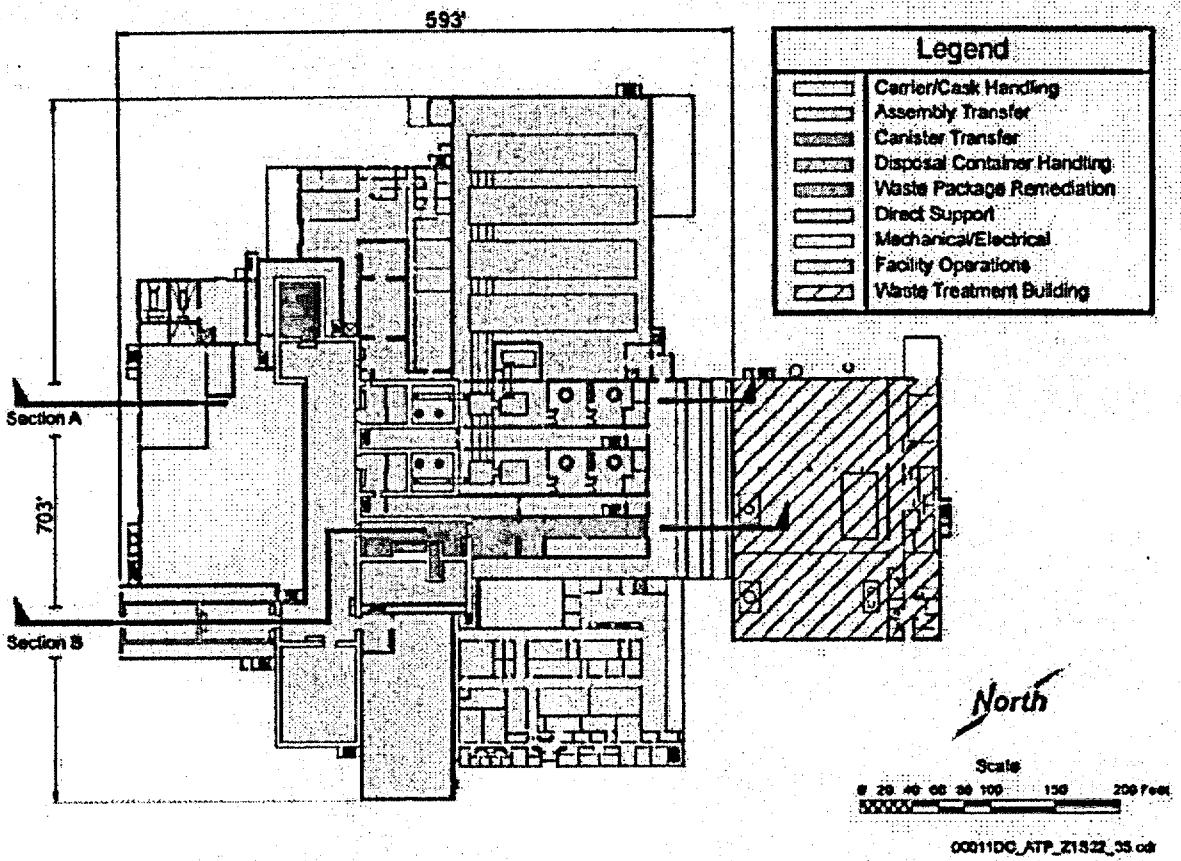


Figure 2-22. Waste Handling Building Systems Layout
Spent nuclear fuel and high-level radioactive waste in transportation casks would be removed from their carriers in the carrier/cask handling area and sent to one of two systems, depending upon how they are packaged, for further processing: either the assembly transfer system or the canister transfer system. After the transportation contents have been transferred into a disposal container, the disposal container would be moved to the disposal container handling system, where the lids would be welded onto the container and then inspected. A waste package remediation area would be provided to address any mechanical problems with the waste package. The Waste Treatment Building would collect and process solid, liquid, and gaseous low-level radioactive waste. Solid waste would be packaged for shipment and offsite disposal. Sections A and B are shown in Figure 2-23. Elevation: 100 ft. Source: Modified from CRWMS M&O 2000q, Figures I-1 and I-4.

view and sections, respectively. A 30.5-m (100-ft) elevation designation has been assigned to the finished grade elevation of the Waste Treatment Building; all other references to building elevation are measured from this assigned designation. Table 2-4 gives the preliminary performance specifications for the Waste Handling Building.

The Waste Handling Building would contain different systems to:

- Confine potential contaminants
- Enhance industrial safety
- Control and monitor operations
- Provide general safeguards and security
- Supply necessary fire protection, ventilation, and utilities.

The building would also provide the needed space and layout for maintenance, administration, and other support operations associated with waste handling activities. It would have designed-in means of protecting its systems and subsystems from the adverse effects of any natural or man-made environment. The complex will also be designed to ensure that radiation exposure levels to workers are kept ALARA.

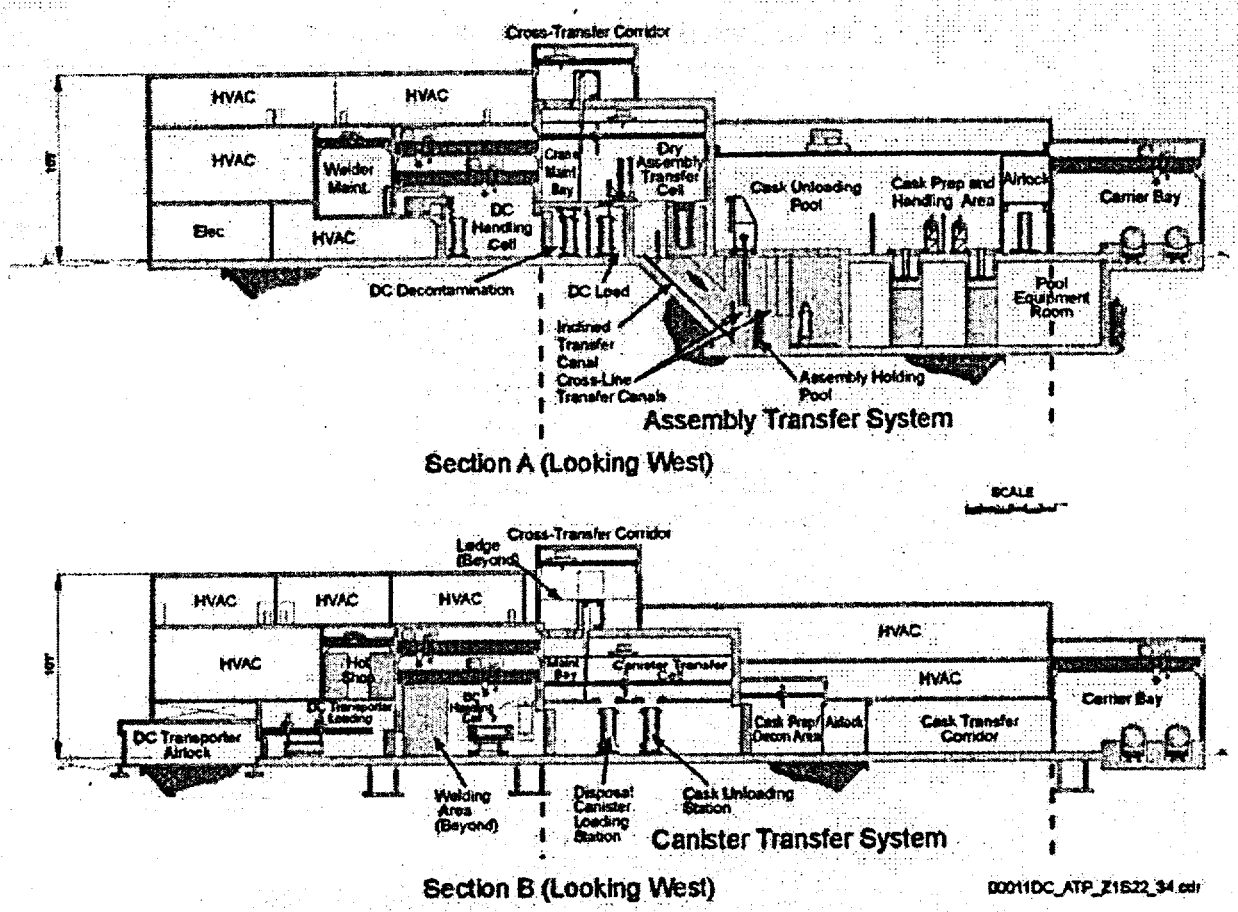


Figure 2-23. Waste Handling Building Sections
The top figure, Section A, depicts the assembly transfer system (two lines), where individual spent nuclear fuel assemblies would be loaded into disposal containers. The bottom figure, Section B, depicts the canister transfer system (one line), where canisters already loaded with spent nuclear fuel or high-level radioactive waste and suitable for direct insertion would be transferred into disposal containers. Various electrical and heating, ventilation, and air conditioning equipment support waste handling operations. Locations of these two sections are shown in the plan view in Figure 2-22. More detailed figures follow. HVAC = heating, ventilation, and air conditioning; DC = disposal container. Source: Modified from CRWMS M&O 2000q, Figure I-9.

2.2.4.2.1 Waste Handling Building: Architectural Features

The Waste Handling Building would be located in the North Portal area. Built adjacent to the south wall of the Waste Treatment Building, the Waste Handling Building would be a multilevel, concrete and steel structure made of noncombustible materials. The exterior walls would be mainly concrete; walls that do not provide shielding for radiation protection would be constructed of metal siding panels with insulation. The building would be approximately 180 m (600 ft) wide by 210 m (700 ft) long (CRWMS M&O 2000q, Figure I-4). Figure 2-23 shows section views of the building. All personnel would enter the building through a security portal. Staff who work in contaminated or potentially contaminated areas would change into protective clothing in the change rooms before proceeding to workstations through entrance/exit corridors. All operations levels would be accessible by corridors and stairwells. To meet function and safety requirements, the operating galleries would be located outside the transfer cells. Operating galleries are shielded areas where operators can safely and remotely observe and control operations. Shielding walls, windows, and doors would

Table 2-4. Preliminary Waste Handling Building Performance Specifications

Source of Requirement	Performance Specification
Performance Criteria	Support and optimize dry handling of high-level radioactive waste for the expected waste throughput
	Support and optimize wet handling of high-level radioactive waste for the expected waste throughput
	Provide space for staging, layout, and storage of equipment and tools
	Provide space for preparation of canisters, waste packages, and empty disposal containers
	Facilitate the collection and transfer of solid and liquid low-level waste (no mixing with noncontaminated systems) for processing in the Waste Treatment Building
Nuclear Safety Criteria	Structure shall be designed to withstand a Frequency Category 2 design basis earthquake
	Provide suitable barriers to impede the spread of fires to systems and equipment that are important to confinement
	System design shall provide that during and after a design basis events and off-normal environmental conditions, operation of QL-1 systems important to safety is not affected by failure of other structures, systems, and components
	Building shall maintain control of radioactive waste and radioactive effluents
	Building shall limit radioactive contamination by creating confinement zones based on isolating activities that have a potential for emissions
	Building shall be designed for the design basis flood in accordance with guidelines in Regulatory Guide 1.102, *Flood Protection for Nuclear Power Plants*
	Building design shall limit the accumulation of radioactive contamination and facilitate decontamination of components and surfaces
Nonnuclear Safety Criteria	Floor surfaces upon which transportation casks are loaded and unloaded shall be flat and at the same level as the top of any railroad track rails
	Building shall be designed to permit prompt termination of operations and evacuation of personnel during an emergency
Environmental Criteria	Building shall be designed such that components susceptible to radiation will not deteriorate excessively when exposed to the radiation environment in which it is located
	Building shall be designed for an outside temperature of 5° to 117°F (-15° to 47°C)
	Building shall be designed for a maximum daily snowfall of 10 in. (25.4 cm) and a maximum snowfall accumulation of 17 in. (43.18 cm)
	Building shall be designed for a maximum annual precipitation of 10 in. (25.4 cm) and a maximum daily precipitation of 5 in. (12.7 cm)
	Building shall be designed to withstand a frost line depth of 15 in. (38.1 cm) below the undisturbed ground surface.

Source: CRWMS M&O 2000s.

protect staff operating and maintaining the five primary waste handling systems.

The Waste Handling Building would integrate the five primary systems (CRWMS M&O 2000q, Section 6.2.1) that receive, lift, unload, handle, reload, package, and deliver high-level radioactive waste to subsurface waste handling systems. Table 2-5 summarizes these preliminary engineering specifications. The primary systems in the building would be:

- Carrier/cask handling system
- Assembly transfer system
- Canister transfer system
- Disposal container handling system
- Waste package remediation system.

Carriers would move transportation casks into and out of the Waste Handling Building through vertical lift doors at the carrier bay. The casks would then be transferred through air locks to one of two assembly transfer system lines or to the one canister transfer system line. Each assembly transfer line would contain:

- A cask preparation and decontamination area
- A pool area for holding and unloading the cask
- An inclined transfer canal to an assembly handling cell
- A disposal container loading cell
- A disposal container decontamination cell.

Table 2-5. Preliminary Facility Space Specifications for the Waste Handling Building

Facility Space/System	Floor Area m^2 (ft^2)
Cask/carrier handling system	1,456 (15,680)
Assembly transfer system	7,645 (82,300)
Canister transfer system	1,374 (14,800)
Disposal container system	4,751 (51,140)
Waste package remediation system	185 (2,000)
Primary support areas	6,791 (73,100)
Heating, ventilation, and air conditioning equipment areas	22,413 (241,260)
Other building areas not listed above	15,358 (165,320)
TOTAL	59,978 (645,600)

Source: CRWMS M&O 2000q, Table 6-2.

The assembly transfer pools would be connected, via fuel basket transfer canals, to four compartmentalized fuel blending inventory pools. The canister transfer line would consist of a cask preparation and decontamination area, a cask unloading area, and a station for loading the disposal containers with the waste. After being loaded, the disposal containers from the three transfer lines would be staged in the disposal container handling cell, where their lids would be robotically welded on. Finally, the loaded, sealed, inspected, and accepted disposal containers, now referred to as waste packages, would be transferred into the waste package transporter cell and loaded onto a transporter for subsurface emplacement.

A number of systems and structural features would support these waste handling operations. An area would be designated for preparing empty disposal containers, and a holding area would provide room for loaded and sealed waste packages waiting for emplacement. A maintenance bay would be available to maintain the handling cranes. The building would also have shops to repair and maintain instruments, robotic welders, and other equipment, along with storage areas for all necessary tools, maintenance materials, high-efficiency air particulate filters, and gas bottles.

2.2.4.2.2 Waste Handling Building: Structural System

The building's foundation would be a reinforced concrete mat (CRWMS M&O 2000p, Attachment II, Section 1.1.6.4). Before construction of the foundation, the undocumented fill of the existing North Portal pad would be improved by the DOE. The building would be designed to withstand (CRWMS M&O 1999e):

- Tornado winds of up to 302 km/hr (189 mi/hr)
- Tornado-generated missiles
- A pressure drop of 0.81 psi
- A rate of pressure drop of 0.3 psi/s.

Although unlikely, ground motion from earthquakes of significant magnitude may occur at the site of the Waste Handling Building (for a discussion of site earthquake ground motion, see Section 4.3.2.2). The best way to provide confidence that the Waste Handling Building can be safely constructed and operated is to determine the response of the building to ground motion. To this end, the DOE has performed a preliminary soil–structure interaction analysis, using a simplified conceptual design of the Waste Handling Building. This analysis demonstrates that a Waste Handling Building can be designed so that its response to large earthquakes would be within acceptable levels and its structural members would not be overstressed (CRWMS M&O 2000t). The final detailed design of the Waste Handling Building will be in accordance with NRC regulations, which require the building to safely survive a site-specific earthquake with a probability of occurring once in 10,000 years.

The design of the Waste Handling Building would include features to limit worker radiation exposure to levels that are ALARA (CRWMS M&O 2000p, Attachment II, Section 1.1.6.3.4). Various areas in the Waste Handling Building have been designated to have radiation levels that either preclude human occupancy or in which occupancy would be controlled. These areas are designated as radiation access zones, which are defined as areas with radiation levels that potentially fall within boundaries that correlate to the limits in 10 CFR Part 20 and

10 CFR Part 835 if the facility were to be licensed. The object of the radiation access zones is to provide a design framework that realistically limits radiation exposures to the lowest levels that are reasonable, given the state of technology, economics, and benefits to public health and safety. Figure 2-24 shows the radiation access zones that are currently designated in the Waste Handling Building. The slab thicknesses for missile protection will be validated during detailed design. Radiological areas would have 1.5-m (5-ft) thick concrete floors that can support loads of up to 126 metric tons (140 tons) of heavy equipment that would handle casks and waste packages (CRWMS M&O 2000p, Attachment II, Section 1.1.6.4). The walls would also be concrete, with stepped decreases in thickness at the ceiling. The roof would be a concrete slab supported by steel beams and concrete walls (CRWMS M&O 2000q, Section 6.2.7). The roof structure would also be designed to withstand tornado-force winds and wind-generated missiles.

Underwater waste handling operations would be conducted in the assembly transfer system pool area, which would consist of several pools for holding and unloading casks and one nonstandard fuel pool. Fuel would be stored in the Pool Fuel Blending Inventory Building, near the assembly transfer system line unloading pools. The Pool Fuel Blending Inventory Building would be a steel-framed structure measuring approximately 85 m

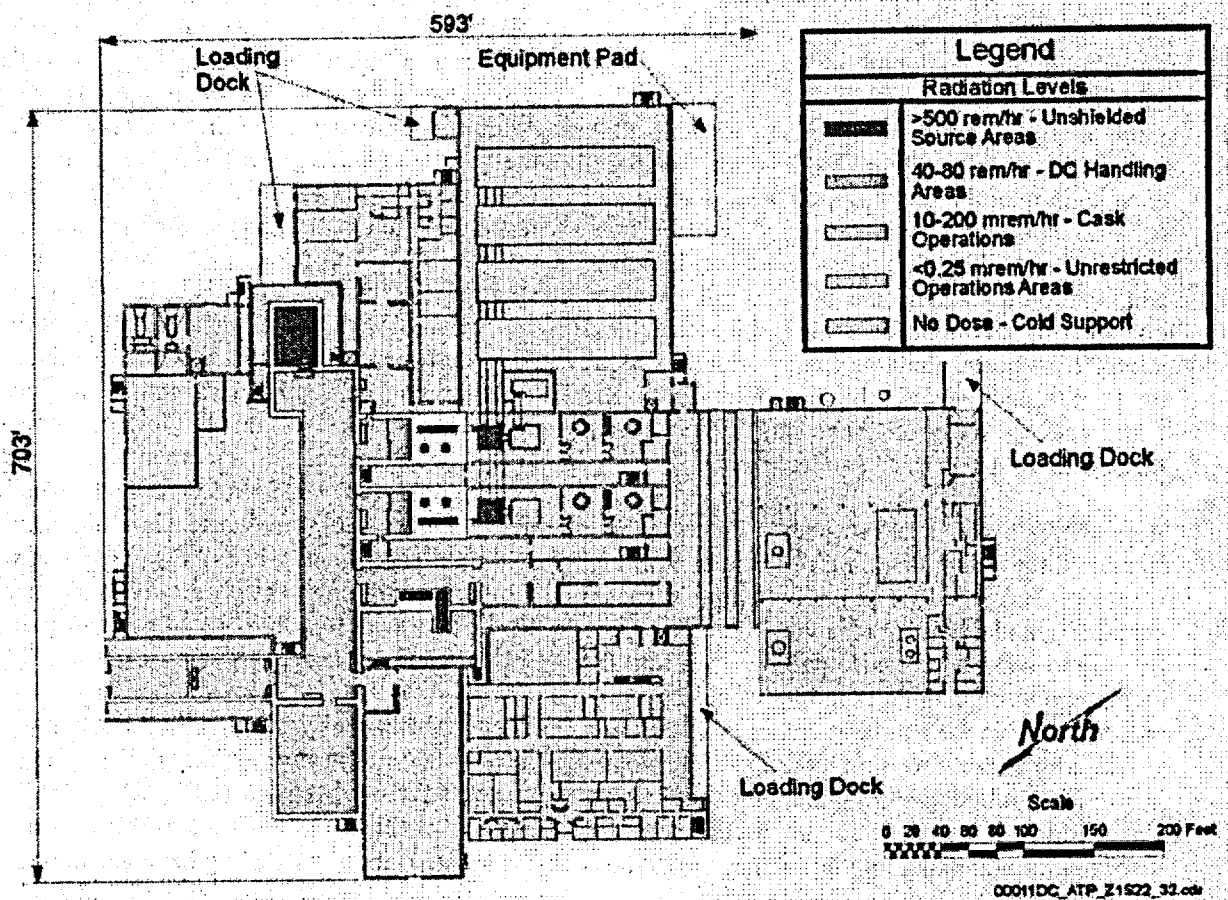


Figure 2-24. Waste Handling Building Radiation Levels
All waste handling operations would be conducted behind shielded concrete walls using remotely operated equipment. Personnel access to such areas would be prohibited during operations, and the doors to the cells would be locked to prevent accidental entry. Radiation areas would be marked in accordance with the requirements of the NRC, which places an upper limit on annual worker radiation exposures of 5,000 mrem. Elevation: 100 ft. DC = disposal container. Source: Modified from CRWMS M&O 2000p, Figure I-28.

(280 ft) long, 59 m (194 ft) wide, and 21 m (70 ft) high. It would house four inventory pools measuring 48 m (160 ft) long, 11 m (37 ft) wide, and 15 m (50 ft) deep. The inventory pools would be connected to the assembly transfer system unloading pools by two transfer canals. The pools would be designed to withstand earthquakes and other anticipated design basis events. They would meet radiation shielding requirements through construction of reinforced concrete floors and walls lined with stainless steel plate, which would also keep pool water from coming into contact with the concrete pool, floor, and walls, and leaking out (CRWMS M&O 2000p, Section II, Attachment 1.1.6.4). The stainless steel liners would contain leak detection systems.

The Waste Handling Building complex would contain several nonradiological facilities that would be separated from the main building to avoid possible interactions during an earthquake (CRWMS M&O 2000p, Attachment II, Section 1.1.6). The facility support area would include administrative offices and laboratories. It would be a two-story, steel-framed structure with sheet metal siding and a metal deck roof. The structure would consist of steel beams, columns, and bracing, with metal-clad siding and insulated roofing (CRWMS M&O 2000p, Attachment II, Section 1.1.6.4). The first floor would be concrete slab on grade; the second floor would be concrete slab on metal decking. The columns would have combined spread footings to avoid settlement.

The other nonradiological facilities would be used for cleaning transport equipment and storing and preparing new waste packages. The structure for these facilities would be light steel framing with sheet metal siding. The floors would support the loads from rail and truck transporters bringing waste packages into the buildings and from storing prepared, empty disposal containers.

2.2.4.2.3 Waste Handling Building: Carrier/Cask Handling System

The Waste Handling Building carrier/cask handling system would receive rail and truck carriers containing transportation casks from the carrier/cask transport system, unload the casks, and reload the empty casks back onto the carriers for shipment off the repository site. Loaded casks would be transferred to the assembly transfer system or the canister transfer system.

The carrier/cask handling system would also receive dual-purpose canister overpacks from the carrier/cask transport system, unload them, transfer them to the assembly transfer system, receive the overpacks with the empty dual-purpose canisters back from the assembly transfer system, and reload the empty dual-purpose canisters onto carriers for shipment off the repository site. The carrier/cask handling system would be configured to accommodate the waste transportation and processing schedules established for the repository (CRWMS M&O 2000p, Attachment II, Section 1.1.3.1).

The carrier/cask handling system would be housed in the carrier bay of the Waste Handling Building. Figure 2-25 provides a mechanical flow diagram that is based on the waste handling system operations documented in the *Engineering Files for Site Recommendation* (CRWMS M&O 2000p, Attachment II, Section 1.1.3.1).

A site prime mover would tow the truck carrier or railcar into the carrier bay's loading area. After removal of the cask tie-downs, a 160-ton capacity bridge crane would lift the cask off the carrier and place it onto a cask transfer cart for delivery to the assembly transfer system or the canister transfer system, as appropriate. After the cask is unloaded and decontaminated, it would be returned to the carrier/cask handling system for shipment offsite.

The Waste Handling Building, which would house the carrier/cask handling system, would provide the necessary utility and safety systems (CRWMS M&O 2000p, Attachment II, Section 1.1.3.1). No spent nuclear fuel or high-level radioactive waste would be directly handled by the carrier/cask handling system. The waste forms would be protected by the transportation casks.

2.2.4.2.4 Waste Handling Building: Canister Transfer System

The Waste Handling Building would house the canister transfer system, which would receive rail

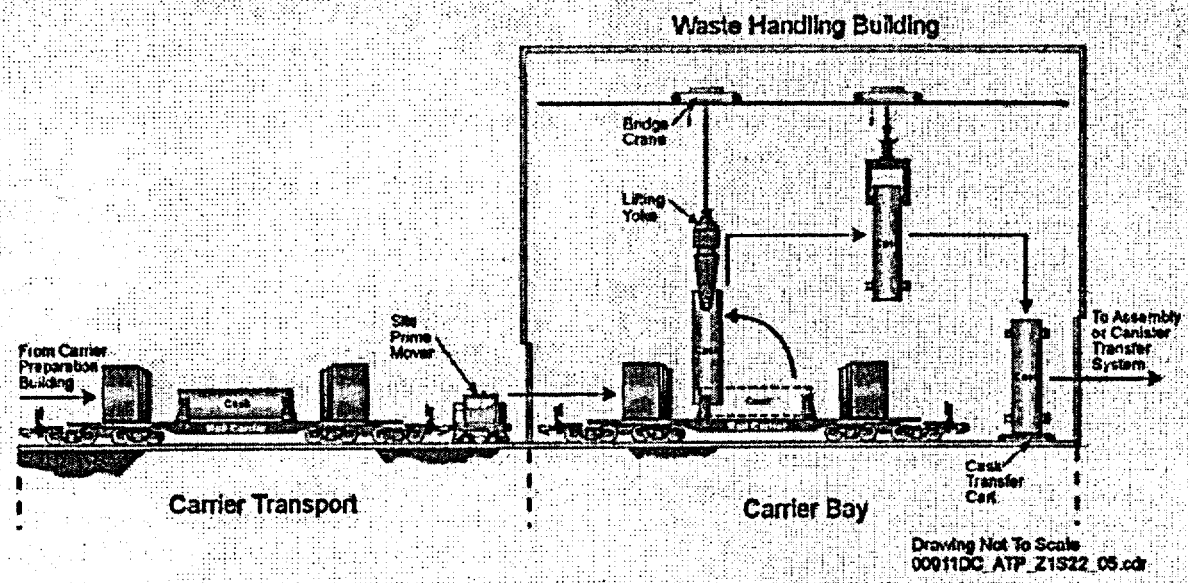


Figure 2-25. Carrier/Cask Handling System
Transportation casks containing spent nuclear fuel and high-level radioactive waste would be offloaded from road/rail carriers in the carrier bay of the Waste Handling Building. The cask would be placed on a seismically restrained cask transfer cart that sits on rails and is powered by electric motors. The cart is the means by which loaded and emptied transportation casks would be moved about within the Waste Handling Building. The transportation casks would remain closed while in the carrier bay. Source: CRWMS M&O 2000p, Figure I-11.

and truck transportation casks from the carrier/cask handling system and empty disposal containers from the disposal container handling system. Figure 2-26 provides a mechanical flow diagram for the operations of the canister transfer system. One canister transfer line would handle canister waste and support maintenance of the system. The line would be configured to handle disposable canisters of DOE high-level radioactive waste or spent nuclear fuel, ultimately loading them into disposal containers. The canister transfer line would also have:

- An air lock
- A cask preparation and decontamination area
- A canister transfer cell
- An off-normal canister handling cell
- A transfer tunnel connecting the canister transfer and off-normal canister handling cells.

WHB/WTB Space Program Analysis for Site Recommendation (CRWMS M&O 2000q, Section 6.2.1.3) documents the physical arrangement of the canister transfer system. Preliminary engineering specifications are given in Table 2-6.

A transportation cask containing canisters of DOE high-level radioactive waste or spent nuclear fuel would be unloaded in the Waste Handling Building's carrier bay, then transferred to a cask transfer cart and secured against overturning. The cask transfer cart would move through a transfer corridor into a canister transfer system air lock, which would have isolation doors at both ends to maintain a lower air pressure in the canister transfer work areas than in the carrier bay. The cart would take the cask through the air lock to the cask preparation area (CRWMS M&O 2000p, Attachment II, Section I.1.2.1).

The cask preparation area would include a preparation station and a decontamination station. Remote handling equipment would consist of a cask transfer cart, cask preparation manipulator, and the tools required to perform cask unbolting, venting,

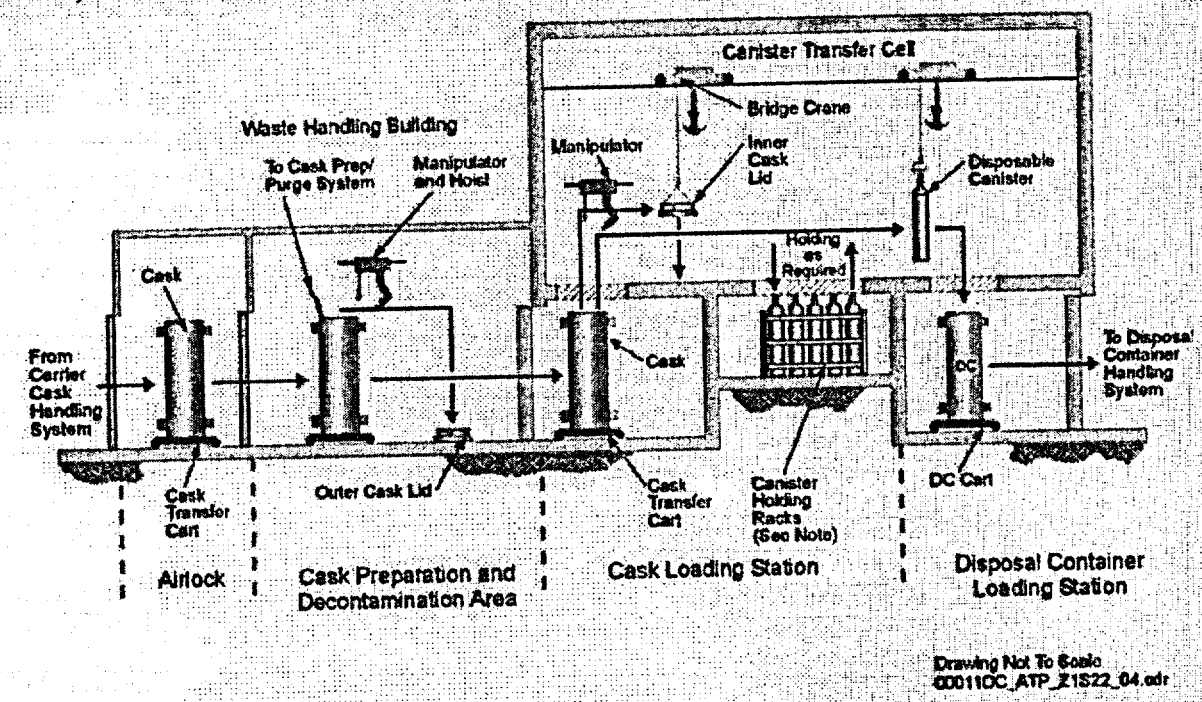


Figure 2-26. Canister Transfer System
The transportation cask outer lid would be removed and the internal gas sampled using a purge system and remote manipulator. If necessary, the gas would be filtered prior to release. The cask transfer cart would move the cask into the canister transfer cell, where the inner cask lid would be removed and the canister removed from the transportation cask and inserted directly into a disposal container. Canister holding racks are not in the direct transfer path from the loading station to the disposal station; they are provided to facilitate efficient operations. Such inventory allows the creation of precise thermal loads for each waste package. The disposal container would be moved and positioned using a seismically restrained transfer cart similar to the one used for moving the transportation cask. DC = disposal container. Source: Modified from CRWMS M&O 2000p, Figure I-6.

lid removal, and decontamination. Workers preparing a cask would:

- Sample the cask's vent ports
- Vent the cask and purge the cavity gas, if required, to the atmosphere through a high-efficiency particulate air filtration system
- Loosen the outer lid bolts
- Secure a lifting fixture to the outer lid
- Remove the outer lid and stage it in the cask preparation area.

Casks containing disposable canisters are not required to have outer and inner lids; for example, the naval spent nuclear fuel canister would have only one closure lid. A special lifting fixture would be installed on naval spent nuclear fuel canisters, using the hoist and the manipulator. The cask transfer cart would then move the cask to the canister transfer cell, where the canister would be loaded directly into a disposal container (CRWMS M&O 2000p, Attachment II, Section 1.1.2.1).

Once the canisters are removed from the transportation cask, the empty cask would be prepared for shipment back to a transportation contractor for reuse. The cask transfer cart would move the cask to the decontamination area, where workers would:

- Remove the inner lid lifting fixture
- Install and tighten the inner lid bolts
- Install and tighten the outer lid bolts
- Remove the outer lid lifting fixture
- Perform a radiation survey on the empty cask
- Decontaminate the cask (if required).

Table 2-6. Preliminary Canister Transfer System Performance Specifications

Source of Requirement	Performance Specification
Performance Criteria	System shall transfer required canister waste types
	System shall support required average transportation cask turnaround times to the Regional Servicing Contractor
	System shall provide for canister inventory
	Provide features to sample, measure, and monitor the transportation cask parameters prior to opening
	System shall limit canister temperatures while in inventory racks
	Provide capability for off-normal canister handling
Nuclear Safety Criteria	System shall be designed to ensure that canisters cannot be breached from any credible drop, considering each type of canister handled[a]
	Canister racks shall be designed to maintain their geometry during and following a Category 2 design basis earthquake
	System design shall prevent the drop of suspended canisters during and following a Category 1 design basis earthquake
	Cranes and hoists shall be designed to remain on their rails during and following a Category 2 design basis earthquake
	System shall be designed to retain suspended loads during and after a loss of electrical power
	System shall be designed for criticality safety under normal and accident conditions
	Maintain control of canistered waste and permit prompt termination of operations during an emergency
	System design shall define safe load paths for the movement of heavy loads to minimize the potential for drops on canister waste
Nonnuclear Safety Criteria	System shall be designed to incorporate the use of noncombustible and heat resistant materials to the extent practicable
	System design shall include provisions for decommissioning and decontamination

[a]The exception is the naval canister, which is not certified to withstand all credible handling impacts.
Source: CRWMS M&O 2000u; BSC 2001f, Section 5.4.

After the cask has been prepared, it would be moved to the carrier bay for loading onto a rail or truck carrier (CRWMS M&O 2000p, Attachment II, Section 1.1.2.1).

All canister transfer operations would be performed remotely in shielded canister transfer or off-normal canister handling cells. The canister transfer cell would consist of:

- Upper and lower transfer rooms
- A cask unloading port
- A disposal container loading port where canisters would be loaded
- An off-normal canister transfer port
- A small canister holding area
- A crane maintenance area.

Small canisters would either be loaded directly into a disposal container or staged in the canister transfer cell until enough canisters are available to fill a disposal container. Large canisters of naval spent nuclear fuel would be loaded directly into a disposal container. The canister transfer system would then deliver the loaded disposal containers to the disposal container handling system. Any canisters that are damaged during handling or received in a condition that does not meet acceptance criteria would be considered off-normal. Off-normal canisters would be transferred to the off-normal canister handling cell for corrective action (CRWMS M&O 2000p, Attachment II, Section 1.1.2.1).

The canister transfer cell would be divided into lower and upper rooms, as previously described, with transfer ports between the rooms to allow vertical lifting of a canister from its transportation cask to its disposal container, to the holding area, or to the off-normal transfer tunnel. The upper

room would include a maintenance bay and ports for:

- Unloading transportation casks
- Loading disposal containers
- Transferring off-normal canisters
- Moving the canisters to the holding area.

The lower room would include:

- A station for unloading casks
- A station for loading disposal containers
- The canister holding area
- A transfer tunnel and cart for off-normal canisters.

A rack would accommodate temporary holding of 20 small canisters in a shielded area. This arrangement would reduce the height of any potential canister drop during transfers (CRWMS M&O 2000p, Attachment II, Section 1.1.2.3).

The canisters would be removed from a transportation cask one at a time by remote equipment and placed in a disposal container, taken to the holding area, or moved through the port for off-normal canisters to the off-normal canister handling cell. Remote handling equipment in the transfer cell would include a 65-ton overhead bridge crane (sized to handle large naval canisters), an electromechanical manipulator, and a suite of small canister-lifting fixtures. The remote equipment would be designed to facilitate in-cell operations, maintenance, and recovery from off-normal events. A maintenance bay inside the cell would facilitate in-cell maintenance. Interchangeable components would facilitate maintenance, repair, and replacement of equipment. Lay-down areas would be provided, as required, for fixtures, tooling, and canister grapples. If in-cell equipment fails, the crane and manipulator can be remotely withdrawn to the maintenance bay by using off-normal and recovery operations. Once a disposal container has been loaded, it would be moved to the disposal container handling system (CRWMS M&O 2000p, Attachment II, Section 1.1.2.1).

A separate off-normal handling cell would be located next to the canister transfer cell, connected by the off-normal canister transfer tunnel. Special equipment would receive, handle and, if necessary, repackage off-normal canisters before final disposal in the repository. The cell's equipment would include a small overhead crane, a bridge-mounted electromechanical manipulator, and two overpack loading and welding stations (for canisters with different diameters and heights). The loading and welding stations would be located in a pit to reduce the height above the floor that a canister must be lifted for placement into an overpack. At both the loading and welding stations, special fixtures would be used to properly position, load, and weld the overpacks. A robotic welding machine, positioned between the pits, would remotely weld a loaded overpack in either station. The off-normal canister handling cell would also contain the following (CRWMS M&O 2000p, Attachment II, Section 1.1.2.3):

- A canister transfer cart
- Racks for holding 20 small canisters
- A canister repair station
- Canister overpacks
- Remote-handling fixtures
- A decontamination station
- Closed-circuit television systems
- Shield windows.

2.2.4.2.5 Waste Handling Building: Assembly Transfer System

The assembly transfer system would include the equipment, facilities, workers, and processes for preparing individual spent nuclear fuel assemblies for disposal in the potential repository. Preliminary engineering specifications are given in Table 2-7.

Two assembly transfer system lines would be housed in the Waste Handling Building. Each would operate independently to handle waste throughput and support maintenance operations. Each would include a cask unloading area and a transfer cell area. The cask unloading area would contain an air lock, a cask preparation and decontamination area, and a pool area. The pool area would contain a cask unloading pool and an assembly holding pool. A single transfer canal would connect the two pools. An inclined transfer canal would be used for moving the spent nuclear fuel assemblies from the holding pool to the

Table 2-7. Preliminary Assembly Transfer System Performance Specifications

Source of Requirement	Performance Specification
Performance Criteria	Transfer intact fuel assemblies from specified assembly classes
	Remove spent nuclear fuel assemblies from the dual-purpose canisters
	System shall support average specified transportation times to the regional servicing contractor
	Provide for inventory of 5,000 MTHM of spent nuclear fuel assemblies
	Provide temporary seals, evacuate gases, and backfill disposal containers with inert gas to preclude oxidation of the spent nuclear fuel assemblies
	Provide capability for nonstandard canister handling
Nuclear Safety Criteria	System design shall reduce the probability of a spent fuel assembly drop onto another spent fuel basket during dry handling to less than a Category 1 design basis event
	Spent fuel assembly transfer baskets and basket staging racks shall be designed for a Category 2 design basis event
	Overhead cranes and fuel transfer machines shall be designed for a Category 2 design basis earthquake and not be dislodged from their rails
	System shall be designed to retain suspended loads during and after a loss of electrical power
	System shall be designed for criticality safety under normal and accident conditions
	System design shall define safe load paths for the movement of heavy loads to minimize the potential for drops on spent nuclear fuel
	System shall provide overhead limit sensing and alarming capabilities to automatically stop handling operations and warn operators of unsafe conditions
Nonnuclear Safety Criteria	System shall be designed to incorporate the use of noncombustible and heat resistant materials to the extent practicable
	System design shall include provisions for decommissioning and decontamination

Source: CRWMS M&O 2000v.

assembly handling cell. The assembly transfer system would include an assembly handling cell, a disposal container loading cell, and a disposal container decontamination cell. The assembly transfer system would also include fuel blending inventory pools and a special pool for nonstandard fuel, which would be located in an annex to the Waste Handling Building. The physical arrangement of the assembly transfer system is documented in the *WHB/WTB Space Program Analysis for Site Recommendation* (CRWMS M&O 2000q, Section 6.2.1.2).

Figures 2-27, 2-28, and 2-29 depict mechanical flow diagrams that illustrate assembly transfer system operations. The process begins with the receipt of transportation casks from the carrier/cask handling system and the receipt of empty disposal containers from the disposal container handling system. The casks are prepared for unloading, cooled, and filled with water. The system would then transfer the rail and truck transportation casks to the cask unloading pools. In the pool, the system would remove the inner cask lid and unload individual spent nuclear fuel assemblies, single-element canisters, and dual-purpose canisters from the transportation casks. Dual-purpose canisters would be opened in the pool using remote underwater tools. Other assembly transfer system operations include (see also Table 2-7):

- Holding assemblies in pools
- Transferring assemblies to transfer cells
- Drying assemblies
- Loading assemblies into disposal containers
- Filling the containers with inert gas
- Installing lid sealing devices
- Decontaminating the lid areas
- Transferring loaded disposal containers from the cell area to the disposal container handling system

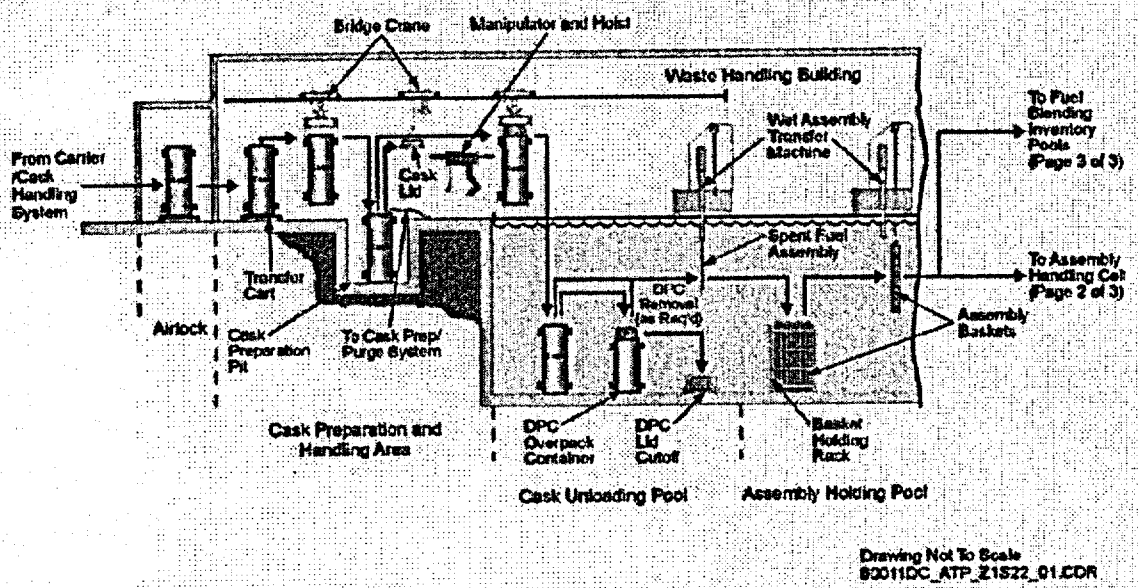


Figure 2-27. Assembly Transfer System (1 of 3)
An overhead building crane would lift the cask from the cask transfer cart and place it in one of two cask preparation/purge pits, where the outer lid would be removed and the interior gas sampled and, if necessary, filtered prior to release. The crane would then move the cask from the pit into the pool, where the inner lid would be removed. Individual spent nuclear fuel assemblies would be removed from the cask one at a time and placed into basket staging racks for transfer to the assembly handling cell or the fuel pool blending inventory area. Spent fuel assemblies contained in dual-purpose canisters, which are not approved for insertion into disposal containers, would be removed after the lid has been cut off. Use of a fuel blending inventory allows precise thermal loads to be created for each waste package. DPC = dual purpose canister. Source: CRWMS M&O 2000p, Figure 1-3.

- Repackaging nonstandard fuel assemblies to meet waste package criteria
- Preparing empty transportation casks and dual-purpose canister overpacks for offsite shipment.

Cask Preparation and Handling—A transportation cask would enter a cask preparation area on a cask transfer cart (CRWMS M&O 2000p). The cask preparation and decontamination area would include two cask preparation and decontamination rooms. Each room would contain a station for unloading and loading transportation casks from the cask transfer cart to a cask preparation pit. These stations would also be used to transfer empty transportation casks and dual-purpose container overpacks (on transfer carts) to the decontamination area. Each cask preparation pit would contain access platforms that can be adjusted for various cask sizes. The pit would also include a cask preparation manipulator and hoist that would be operated remotely. The manipulator and hoist would mount on a gantry and straddle the pit and access platforms. The system would contain a variety of remotely operated tools and accessories for preparing and decontaminating casks using the cask preparation manipulator and hoist. Each assembly transfer system line would include a large overhead bridge crane. The cask preparation area would also include a maintenance bay where workers can perform maintenance on the bridge crane.

The cask preparation and handling area (Figure 2-27) equipment would include:

- A cask transfer cart
- A bridge crane that serves the cask unloading area
- Two cask preparation manipulators with hoists mounted on gantries
- Yokes for lifting casks and dual-purpose canister overpacks

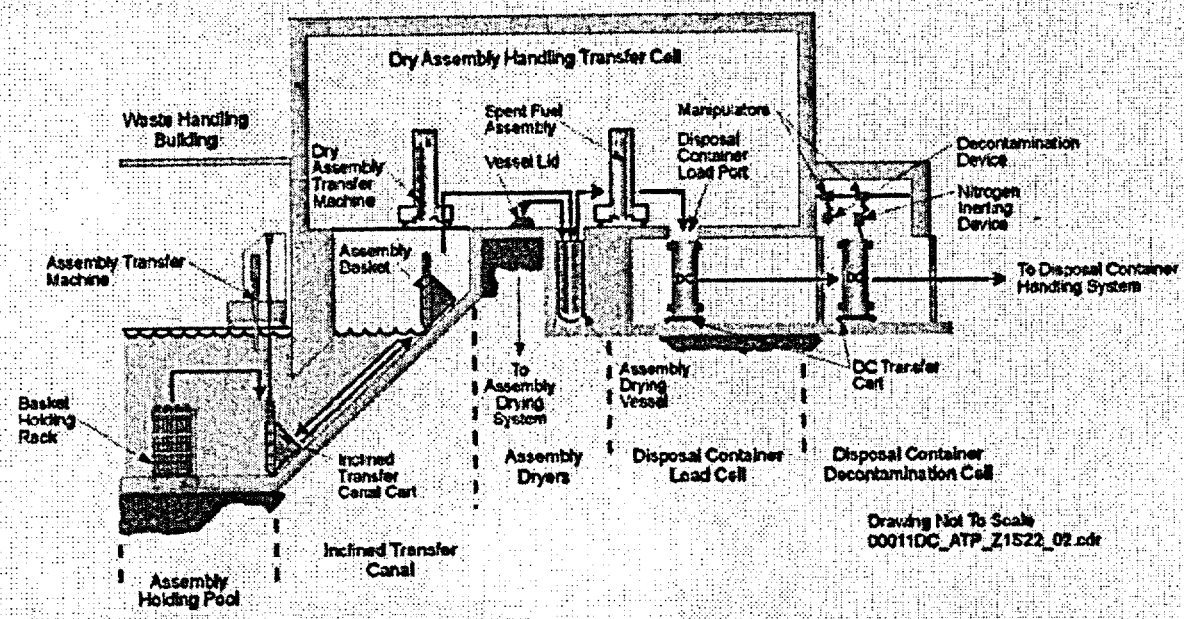


Figure 2-28. Assembly Transfer System (2 of 3)
Individual baskets containing four to nine spent nuclear fuel assemblies would be removed from the staging racks and transferred to an assembly handling cell. Baskets would be transferred from the pool to the assembly handling cell via the inclined transfer canal cart. The baskets would then be placed into one of two drying vessels, where water would be removed to ensure that the interior of the disposal container is exposed to little moisture. After drying, the assemblies would be removed from the dryer one at a time and loaded into the disposal container through the load port in the floor of the cell. After loading, the disposal container would be temporarily sealed and inerted using nitrogen in preparation for later inerting with helium and permanent closure in the disposal container handling system. The disposal container would be moved and positioned using a seismically restrained transfer cart similar to the one used for moving the transportation cask. DC = disposal container. Source: CRWMS M&O 2000p, Figure I-4.

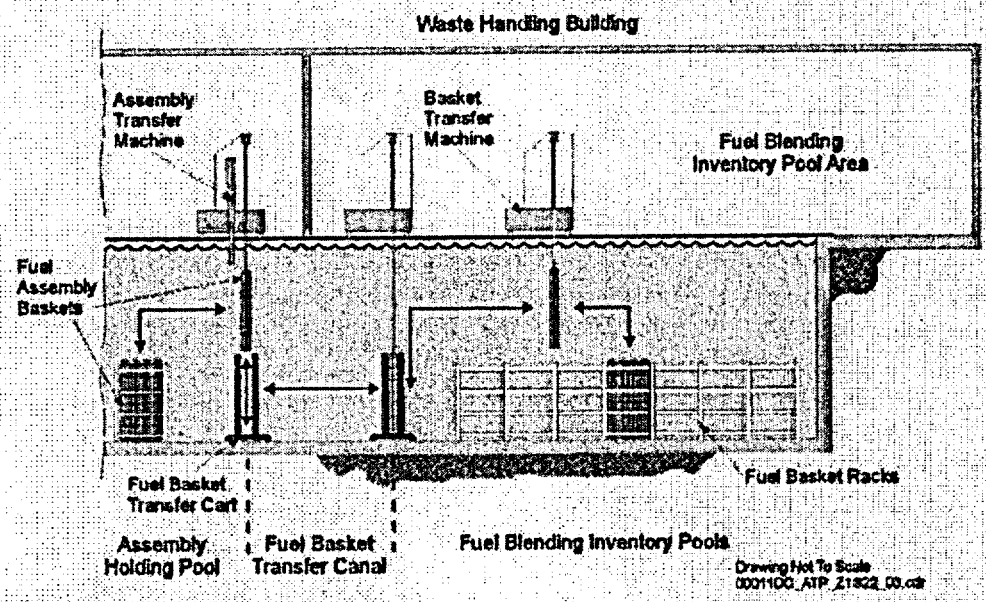


Figure 2-29. Assembly Transfer System (3 of 3)
Each waste package must meet precise thermal load limits. To facilitate meeting such limits, an inventory is established. In that inventory would be older, and therefore cooler, spent nuclear fuel, as well as younger, and therefore hotter, fuel. With an adequate inventory, specific thermal loads can be created for each waste package. Source: CRWMS M&O 2000p, Figure I-5.

- Handling fixtures
- Remotely operated tools and accessories.

The cask unloading and holding pools would be equipped with:

- Remotely operated assembly transfer machines mounted on the pool deck
- Grapples for lifting fuel assemblies
- Cutting tools for removing lids from dual-purpose canisters
- Dual-purpose canister overpacks
- Assembly baskets
- Basket holding racks
- Transfer carts.

The assembly transfer process begins when the carrier/cask handling system unloads a transportation cask from a truck or rail carrier. A crane would lift the cask vertically and transfer it to a cart at the entrance to an assembly transfer system line. The cask would be secured against overturning. The transfer cart would move the cask into an air lock, which would control air movement between the carrier bay and the cask preparation and decontamination area for contamination protection. The air lock would have isolation doors at both ends to maintain a slightly lower air pressure in the radiation work areas than in the carrier bay. The transfer cart would then move the cask to one of two rooms for preparing and decontaminating casks.

The cask preparation and decontamination rooms would have equipment for both remote and manual operations. The large bridge crane in the unloading area and a dry cask lifting yoke would be used to lift the cask from the transfer cart, move it, and lower it into a cask preparation pit. Access platforms would adjust to the size of the cask.

Cask preparation activities would require both remote and manual operations using the crane, manipulator, and associated tools. All cask preparation activities would be performed in a dry environment, but the casks must be prepared for direct transfer to the cask unloading pools. Cask preparation would involve casks with and without dual-purpose canisters. Casks without canisters would have an outer lid or lids, and an inner lid with a built-in shield plug to provide radiation protection for manual operations. Casks with dual-purpose canisters would have a shield plug built inside the top of the canister.

Remote or manual cask preparation operations consist of:

- Sampling and venting gas from the cask
- Unbolting and removing the cask lid
- Cooling the cask with gas and/or water
- Unbolting the shield plug
- Attaching the shield-plug lifting fixture.

Gas sampling and venting operations would be performed by the cask prep/purge system, which vents to the atmosphere through a high-efficiency particulate air filtration system. If the cask contains individual spent nuclear fuel assemblies with no dual-purpose canister, it would be filled with water in the preparation pit and transferred to the cask unloading pool.

If the cask contains a dual-purpose canister, workers would remove the outer lid while the cask is in the preparation pit. Using remotely operated and manual tools, workers would then open the vent valves on the dual purpose canister; sample, vent, and cool the interior cavity; attach a lifting fixture to the canister; and fill the canister with water. The bridge crane and lifting yoke would transfer the cask containing the dual-purpose canister to the cask unloading pool.

Cask Unloading Pool—The pool area would contain a cask unloading pool and an assembly holding pool (see Figure 2-27), connected by a transfer canal. As shown in Figure 2-28, another inclined transfer canal would connect the assembly holding pool to a handling area. Transfer canals that contain transfer carts for fuel baskets would connect both holding pools to fuel blending inventory pools. Another transfer canal would connect the cask unloading pool and the nonstandard fuel pool.

If a cask contains individual spent nuclear fuel assemblies, then the bridge crane, cask shield plug fixture, and wet lifting yoke would be used to remove its shield plug underwater in the cask unloading pool. If the cask contains a dual-purpose

canister, then the bridge crane, canister lifting fixture, and wet lifting yoke would be used to lift the canister from the cask and place it in a dual-purpose canister overpack. Two 160-ton bridge cranes, each with a 25-ton auxiliary hoist, would be provided to facilitate these operations. Using remote cutting tools, workers would then sever and remove the dual-purpose canister lid. All of these activities would take place underwater in the cask unloading pool (CRWMS M&O 2000r).

A wet assembly transfer machine would remove the individual fuel assemblies from the opened shipping casks and dual-purpose canisters, then load them into assembly baskets in the holding pool. The fuel would remain in these baskets until it is dried and placed in inspected and approved disposal containers. The fuel baskets would contain either four fuel assemblies from pressurized water reactors or eight fuel assemblies from boiling water reactors. The holding pool can hold a maximum of 16 fuel baskets at any one time. When the assembly baskets in the holding pool are full, the wet assembly transfer machine would move the baskets to a transfer cart, which, in turn, would move the loaded fuel baskets to a fuel blending inventory pool or the assembly handling transfer cell for disposal container loading.

The pool area bridge crane and wet handling tools would return the empty transportation casks, and the canister overpacks containing empty dual-purpose canisters and lids, to the cask preparation and decontamination area. The cask preparation and decontamination process would include replacing and bolting down the lids on the empty transportation casks and dual-purpose container overpacks. Workers would decontaminate and dry the casks and containers and then survey them for remaining contamination. After decontamination, workers would transfer empty casks and containers to the carrier/cask handling system for shipment off the repository site and reuse or disposal.

Fuel Blending Inventory—The fuel blending inventory area (shown in Figure 2-29), located in an annex to the Waste Handling Building, would contain four modular inventory pools for spent nuclear fuel and one pool for nonstandard fuel. Each modular pool would have a maximum inventory of 750 fuel baskets loaded with spent nuclear fuel and would be equipped with a 20-ton capacity overhead crane. Transfer canals from the assembly holding pool in each assembly transfer line would connect the modular fuel-blending inventory pools. The pools would have isolation gates so that, if necessary, one pool can be isolated from the other pools. The fuel inventory area would also have a separate pool for handling nonstandard and defective fuel in single-element canisters. All spent fuel and basket-handling operations would be conducted underwater.

The fuel assemblies would stay in the inventory pools until they are selected, according to their heat output, for placement in a disposal container. For the design and operating mode described in this report, the maximum heat generation limit for a waste package is 11.8 kW (CRWMS M&O 2000p, Attachment II, Section 1.1.1). Any higher heat-generating fuel loaded into the disposal container must be thermally blended with lower-heat-generating fuel to meet this limit, a procedure called "fuel blending." Some fuel assemblies would remain in the inventory pool until they generate less heat from radioactive decay, or until lower-heat fuel assemblies become available for blending. Approximately 12,000 spent fuel assemblies in 2,800 assembly baskets would accumulate in the inventory pools during the emplacement period to satisfy the 11.8-kW waste package limit. The inventory pools would be large enough to accommodate 5,000 MTHM of spent nuclear fuel; each pool would have a capacity of 1,250 MTHM, or 750 fuel baskets.

Dry Assembly Handling—A fuel assembly is selected for a disposal container according to the heat generation of the assemblies planned for placement in the disposal container. Assembly baskets and fuel would be transferred from the fuel blending inventory pools. The basket transfer machine would lift and place the fuel basket on a transfer cart, which would take the basket back to the assembly holding pool. The wet assembly transfer machine would move the assembly basket to another transfer cart for the inclined transfer canal. This cart would transport the assembly basket up the inclined canal, out of the pool water, and into the dry assembly handling transfer cell.

The dry assembly handling transfer cell (shown in Figure 2-28) would contain:

- A disposal container loading port
- An assembly transfer machine
- An in-cell manipulator
- Two drying vessels
- An in-cell service crane
- A maintenance bay.

A dry assembly transfer machine would move the assembly basket into one of two drying vessels. It will be necessary to dry the fuel assemblies to meet repository waste package performance criteria. After drying the assemblies, the machine would remove them from the drying vessel and load them into a disposal container. The disposal container would be joined to the disposal container loading port below the assembly transfer cell. The dry assembly transfer machine would install the sealing device and the disposal container's inner lid. The transfer cart would then transfer the disposal container to the decontamination cell, where the top lid area and the inner-lid sealing device would be decontaminated. The system would then evacuate the disposal container internal cavity and fill it with nitrogen gas to exclude oxygen and prevent oxidation. Finally, the transfer cart would transfer the disposal container to the disposal container handling system for lid welding and inspection.

Disposal Container Loading—An empty disposal container equipped with a lifting collar, a base collar, and an inner-lid sealing device would be transferred (Figure 2-28) into the disposal container loading cell and mated with the disposal container loading port. The dry assembly transfer machine would remove the disposal container loading port lid and the inner-lid sealing device. After the fuel assemblies are dry, the dry assembly transfer machine would remove fuel assemblies, one at a time, from the baskets in the drying vessel and load them into the disposal container, positioned below the disposal container loading port. When the disposal container is loaded, the inner-lid sealing device and the loading port lid would be reinstalled using the assembly transfer machine. The disposal container would be disengaged from the loading port and transferred to the decontamination cell, using the transfer cart. The empty assembly baskets would be returned to the pools.

Disposal Container Decontamination and Inerting—In the decontamination cell, the lid of the disposal container and the inner-lid sealing device would be decontaminated. The disposal container would be evacuated and filled with nitrogen gas using an inerting manipulator. The disposal container would then be transferred to the disposal container handling system on the transfer cart for lid welding, inspection, and emplacement in the repository.

A transfer cart would transfer disposal containers between the disposal container handling cell, the decontamination cell, and the loading cell. An isolation door would separate the loading cell and the decontamination cell, and a shield door would separate the decontamination cell and the handling cell. A loading port mating device in the loading cell would provide a contamination barrier between the assembly handling cell, the disposal container loading port, and the disposal container during transfer of spent nuclear fuel. The decontamination cell (Figure 2-28) would be equipped with:

- A bridge-mounted inerting manipulator
- A bridge-mounted decontamination manipulator
- A decontamination tool
- A contamination sample pass-through glove box (this would be used to transfer contamination survey samples into an adjacent operating gallery for counting).

All assembly transfer system remote operations would be controlled from operating galleries next to each assembly transfer cell. Strategically located closed-circuit television systems and shield windows would monitor remote operations. All transfer cell area equipment would be designed to facilitate remote operations and removal for contact decontamination and maintenance. Interchangeable components would be provided where appropriate. The assembly transfer system would

also be designed to provide safe and efficient recovery from equipment failures and malfunctions.

Handling Nonstandard Fuel—One assembly transfer system line would be specifically designed and equipped for processing spent nuclear fuel that does not meet the standard loading criteria for disposal containers. This nonstandard fuel might not meet the loading criteria because, for example, it is in an irregular size single or multiple-element canister, it is not intact, or it contains failed fuel (for a definition of nonstandard fuel, see the glossary). The Waste Handling Building would include a handling room for nonstandard fuel in the fuel blending inventory pool annex. The assembly transfer system line cask unloading pool would connect to the fuel handling room through an underwater transfer canal equipped with isolation gates and a transfer cart. To meet disposal container loading criteria, the nonstandard fuel canister may undergo cutting, unloading, and repackaging operations. These operations would take place underwater in the nonstandard fuel pool.

Any cask containing nonstandard spent nuclear fuel would be directed to the appropriate transfer line. After the cask has been prepared, it would be placed in the cask unloading pool. The cask would be opened and the isolation gates between the cask unloading pool and the nonstandard fuel pool opened. The wet assembly transfer machine would unload the fuel assemblies or canisters from the cask and place them in assembly baskets in the nonstandard assembly basket transfer cart. The transfer cart would then be moved to the nonstandard fuel pool. Once the fuel has been unloaded and transferred, the isolation gates between the two pools would be closed. Using a 30-ton capacity overhead bridge crane, the assembly baskets would be removed from the transfer cart and placed into the nonstandard fuel pool basket holding rack. After the fuel has been repackaged, it would be loaded into the assembly basket again and sent back to the cask unloading pool by reversing the above steps. Once in the cask unloading pool, the loaded fuel baskets would be directed either to the fuel blending inventory pools or to the assembly handling transfer cell.

2.2.4.2.6 Waste Handling Building: Disposal Container Handling System

The disposal container would consist of two concentric metal cylinders: an inner cylinder made from stainless steel and an outer cylinder made of a high-nickel alloy (Alloy 22). The bottom end of each cylinder would be closed with inner and outer lids made of the same materials used to make the cylinder itself. The upper end of each cylinder would be used for final closure lid welds, one for the inner cylinder and two for the outer cylinder. Refer to Section 3 for information on the design, fabrication, and welding of the disposal container. The disposal container would arrive without any waste forms inside. After the radioactive waste is loaded, the disposal container design provides some radiation shielding for operating personnel but not enough to keep occupational exposures within allowable limits. Therefore, all loaded disposal container operations would be done remotely in hot cells. Operations on disposal containers and waste packages containing waste would be performed in dedicated, shielded cells.

The lids for the upper end of the disposal container would be fabricated, but they would not be welded and inspected until after waste is loaded into the container. The three top lids would be installed and welded inside the Waste Handling Building. Once the disposal container is loaded, and its inner and outer top lids are welded, inspected, and accepted, the disposal container is called a waste package.

Figure 2-30 provides a mechanical flow diagram for the operations of the disposal container handling system. The system would receive loaded disposal containers from the canister and assembly transfer systems (Sections 2.2.4.2.4 and 2.2.4.2.5), as well as empty disposal containers from the carrier/cask transport system (Section 2.2.4.2.3). The system, located in the Waste Handling Building, would include areas for:

- Preparing empty disposal containers
- Welding disposal container lids
- Holding loaded disposal containers
- Docking and loading the waste package transporter

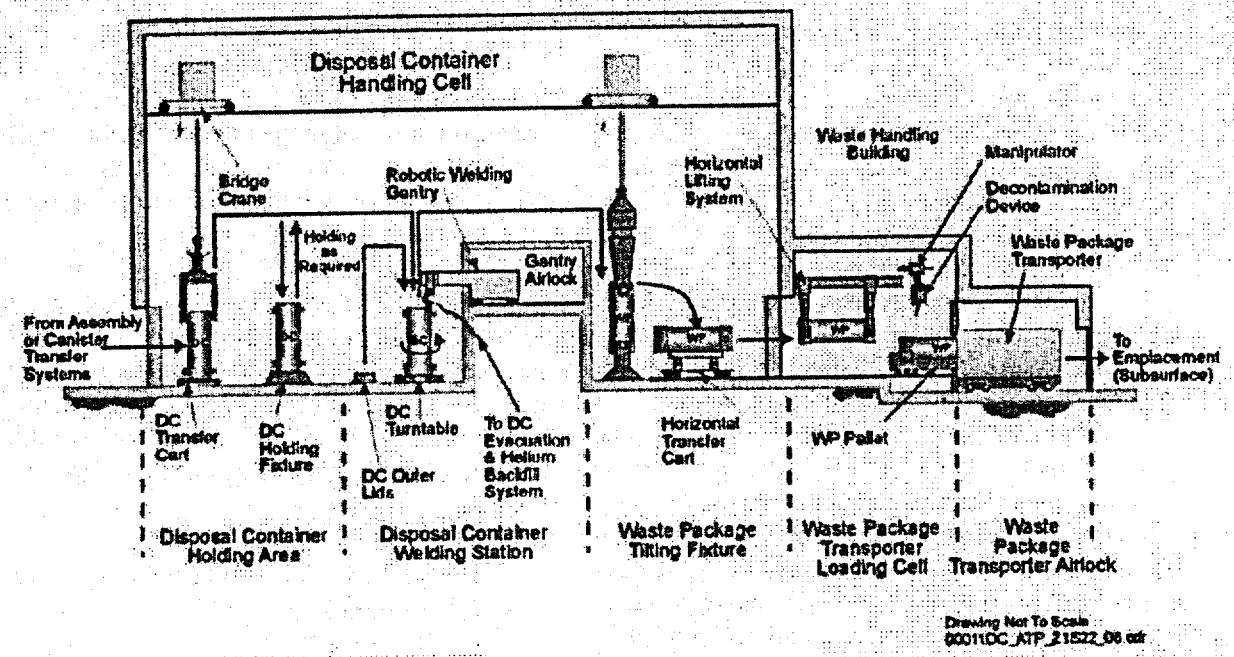


Figure 2-30. Disposal Container Handling System
Loaded disposal containers would be received from the assembly or canister transfer systems. An overhead bridge crane would lift the disposal container from the transfer cart and place it onto the disposal container turntable, where the disposal container is permanently sealed (inner lid is welded) and the interior backfilled with helium to promote efficient heat transfer from the inner portion of the container to the outer walls. Next, the two outer lids would each be welded on. The welds of each of the three welded lids would be checked after each weld. After remote inspection and acceptance of the welds, the disposal container is termed a waste package. The waste package would be moved to a horizontal position in preparation for loading into the waste package transporter, which would be used in moving the waste package underground to the emplacement drifts. DC = disposal container; WP = waste package. Source: CRWMS M&O 2000p, Figure I-13.

- Maintaining equipment used in handling the disposal containers/waste packages.

To adequately accommodate the throughput rates for disposal containers/waste packages, these areas would operate concurrently for 120 hours per week, 50 weeks per year (CRWMS M&O 2000p, Attachment II, Section 1.1.5). The Waste Handling Building's physical arrangement for the disposal container handling system is documented in the *WHB/WTB Space Program Analysis for Site Recommendation* (CRWMS M&O 2000q, Section 6.2.1.4).

The empty disposal container would be fabricated at a commercial supplier's facility and shipped, with the inner and outer closure lids, to the Waste Handling Building for loading. The disposal container handling system would receive and prepare the empty container for loading, then deliver it to either the assembly or canister transfer systems for loading.

Once loaded, the container would be returned to the disposal container handling cell for welding. A number of welding stations would be provided to receive loaded containers from the assembly or canister transfer system lines. The welding operations include:

- Mounting the container on a turntable
- Removing temporary lid sealing devices
- Installing and welding the lids
- Conducting nondestructive examination
- Performing in-process weld stress relief.

Following nondestructive examination and acceptance of the weld, the container would be certified as a waste package and either staged or transferred to a tilting station for transport to the underground

repository. Any disposal container that does not meet the weld examination criteria would be transferred to the waste package remediation system for repair or corrective action. A suite of handling fixtures, including yokes, lift beams, collars, grapples, and attachments, would support the operations of the disposal container handling system. The remote equipment will be designed to facilitate decontamination, maintenance, and use of interchangeable components, where appropriate. Set-aside areas would be included, as required, for fixtures and tooling to support off-normal and recovery operations. Semiautomatic, remote, manual, and backup control methods would be used to support normal, maintenance, and recovery operations. The interfaces of the Waste Handling Building would provide the facility, utility, maintenance, safety, and auxiliary systems required to support operations and radiation protection activities.

Disposal Container Handling—The disposal container handling cell would be a large, shielded structure containing areas for several welding and inspection stations, holding of loaded containers, transfer cart operations, tilting of the waste package to a horizontal position, and maintenance of the overhead cranes. Handling operations for disposal containers would involve two remotely operated, 150-ton capacity bridge cranes (spanning 22.6 m [74 ft]) and hoists, as well as peripheral equipment. An empty disposal container would be lifted by one of the cranes. The container would either be staged or directly transferred to a transfer cart servicing one of the two assembly transfer system lines or the canister transfer system line for loading. The outer lids for the disposal container would be staged near the welding stations for sealing after the container is loaded. The empty container would be taken into either the assembly transfer system or the canister transfer system for loading. When loaded, the disposal container would be returned to either the holding area or to one of eight welding stations. Each welding station would be equipped with a robotic gantry, a turntable, and multiple sealing tools.

Following examination and certification of the welds, the waste package would be prepared for transport underground to the repository. A completed waste package would be moved to the holding area for loaded disposal containers or to the waste package tilting area, where the waste package would be rotated to a horizontal position resting on a horizontal transfer cart. This cart would transfer the waste package to the transporter loading cell.

Equipment for the disposal container handling system (shown in Figure 2-30) is designed to facilitate remote retrieval for manual decontamination, maintenance, and component replacement, as required. All handling operations would be supported by a variety of remote handling fixtures, including:

- Disposal container lifting and base collars
- Lifting trunnions
- Lifting yokes
- Lifting beams
- Tilting fixtures
- Holding fixtures
- Lid sealing devices.

A crane maintenance bay at the far end of the handling cell would allow for contact maintenance and testing of the cranes in the cell.

Disposal Container Sealing and Closure—The disposal container handling system shown in Figure 2-30 would receive a loaded and temporarily sealed disposal container from the assembly transfer system or the canister transfer system, then transfer it to a holding area or a welding station. Sealing and closure would include a number of steps and remote equipment operations. Additional steps and remote equipment would also be required to conduct weld inspections and postweld heat-treatment operations. Following weld inspection and weld certification, the container would either be staged or prepared for transfer to the subsurface of the repository.

The cranes in the disposal container handling cell would be used to lift and transfer a loaded container to one of several independent lid-welding stations. A remotely controlled robotic gantry would set up, prepare, weld, and backfill the container with inert gas. The gantry would also

serve as the remote handling platform to inspect the sealing operations, which would include:

- Securing the disposal container to the welding station's turntable
- Removing temporary sealing devices
- Purging the lid with inert gases for welding
- Backfilling the container with helium prior to closure
- Turning the container
- Welding the inner lid
- Installing the outer lids
- Welding the outer lids.

Welding would be performed using automatic welders deployed from the robotic gantry platform such that they can be remotely removed from the cell for retooling, testing, adjustments, and maintenance. This feature eliminates the need for personnel to enter the radiation environment in the handling cell. Laser peening and annealing processes may be used to relieve stresses in the weld area.

One air lock would be provided for each of the eight welders. The welder air locks would provide access to the robotic gantry, remote welder, nondestructive examining equipment, and postweld heat-treating equipment (CRWMS M&O 2000p, Attachment II, Section 1.1.5.4). Access and service work on the equipment would be possible in these rooms without exposing the workers to the atmosphere and radiation sources in the disposal container handling cell.

Waste Package Holding—The holding area for loaded disposal containers would be used to stow loaded disposal containers or waste packages awaiting transfer to the waste package transporter loading cell. Waste handling simulations have shown that holding 20 loaded disposal containers in the disposal container handling cell can accommodate a 2-week interruption in repository emplacement operations (CRWMS M&O 2000p, Attachment II, Section 1.1.5.4).

To reduce radiation levels in the crane maintenance bay, loaded disposal containers would be staged in a separate area inside the disposal container handling cell. This area would have partial walls and an access door to facilitate transfers of disposal containers to and from holding locations. The partial walls would provide shielding for the main portions of the cell and the maintenance bay, where protection of equipment and personnel is required. The design configuration incorporates both distance and shielding by isolating radiation sources to one area of the transfer cell and adding a wall separating the staged disposal containers from the welding, handling, and crane maintenance areas. This would significantly reduce radiation doses to equipment during normal operations, while also reducing radiation levels during manned entry into the cell for periodic maintenance and test operations.

Waste Package Transporter Loading—As shown in Figure 2-30, the final handling sequence for the surface facilities involves:

- Repositioning the waste package to a horizontal position
- Transferring the waste package to a decontamination and transporter loading cell
- Loading the emplacement pallet onto the waste package transporter and the waste package onto the emplacement pallet
- Decontaminating the waste packages (final)
- Inspecting the waste packages (final)
- Certifying and tagging waste packages.

These operations would be performed using:

- A remotely operated horizontal transfer cart
- A waste package horizontal lifting machine
- Decontamination and inspection manipulators
- The waste package transporter.

Only one transporter loading line would be available for the final decontamination, inspection, transfer, and loading of waste packages onto a transporter. The waste package, once it is moved into the transporter loading cell from the disposal container handling cell, would be lifted off the

horizontal transfer cart by the lifting collar, the base collar, and the horizontal lifting machine. While suspended, the waste package would be decontaminated, inspected, and certified. Important data needed for repository record keeping would be recorded. The mobile pallet of the transporter would then move into the cell, and the waste package would be lowered onto the pallet. The handling collars would be remotely removed and taken out of the waste package transporter loading cell for reuse. Any contamination picked up during disposal container sealing would be manually removed in rooms for equipment contamination before the collars are transferred to the empty disposal container preparation area for reuse.

A transporter air lock would be provided at the Waste Handling Building exit of the transporter loading line so the waste package transporter vehicle may enter and be docked for loading. The air lock would prevent movement of air between the transporter loading cell and the outside atmosphere. In the final surface waste handling steps, the waste package pallet and rolling bed plate would be pulled into the shielded waste package transporter, the transporter shield doors would be closed, and the waste package transporter would be disengaged from the loading cell dock. Then the waste package would be transported to the subsurface of the repository.

2.2.4.2.7 Waste Package Remediation System

A waste package found to be out of specification would be transferred from the disposal container handling system to the waste package remediation system. This system would be housed in a multipurpose cell inside the Waste Handling Building. Figure 2-31 provides a mechanical flow diagram for the system's operations. The diagram is based on the waste handling system operations documented in the *Engineering Files for Site Recommendation* (CRWMS M&O 2000p, Attachment II, Section 1.1.4.1).

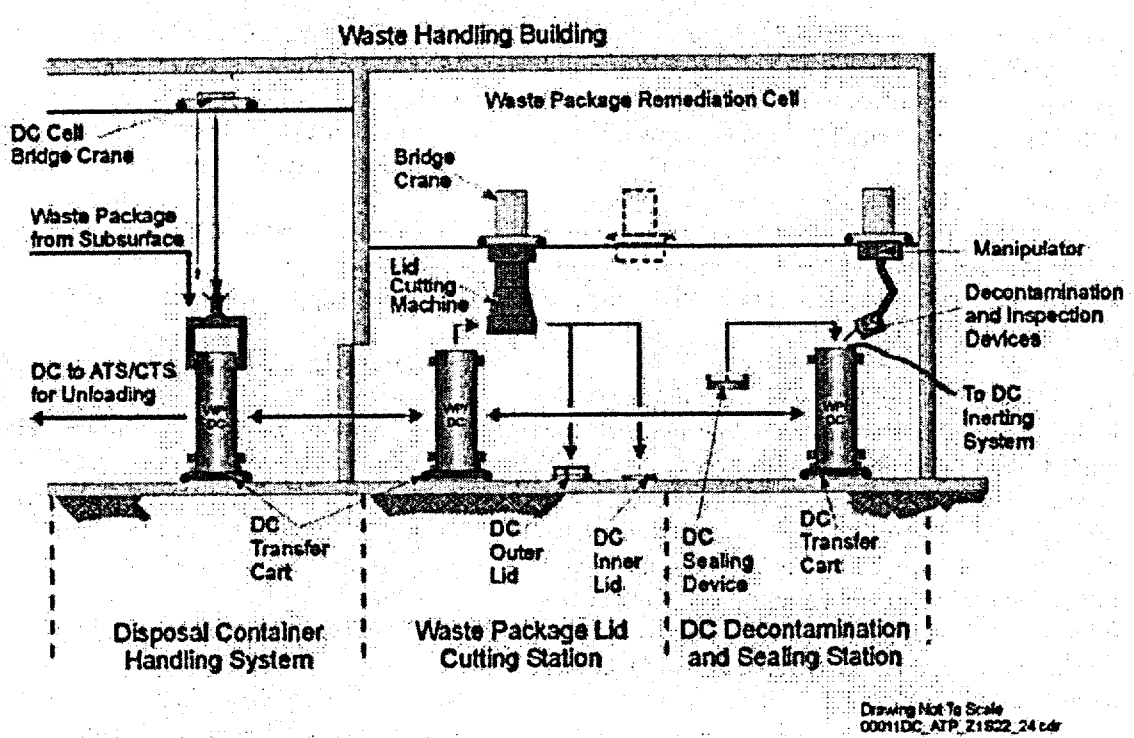


Figure 2-31. Waste Package Remediation System
The waste package remediation system can be used to repair minor defects in disposal containers and waste packages. The system will be used to remedy weld defects and unload waste packages. A lid cutting machine would be used to open the waste packages. DC = disposal container; WP = waste package; ATS = assembly transfer system; CTS = canister transfer system. Source: CRWMS M&O 2000p, Figure I-12.

The waste package remediation system would receive disposal containers and waste packages that:

- Are damaged out of specification
- Have failed the weld inspection processes
- Have been selected for retrieval from the repository for examinations.

The out-of-specification waste packages would be repaired. After the waste packages have been examined, repaired or, if necessary, unsealed and repackaged, the remediation system would deliver them back to the disposal container handling system (CRWMS M&O 2000p, Attachment II, Section 1.1.4.1).

If inspections of the closure weld reveal an unacceptable but repairable welding defect, the disposal container would be prepared for rewelding, which may include partial or complete weld removal. Correction of rejected closure welds would require removal of the defect in such a way that the disposal container can be returned to the disposal container handling system to complete the closure welding process. If examination of the closure weld shows the defect or damage to be irreparable, the container would be opened. If a waste package is retrieved from the repository for any reason—suspected damage, known failure, or planned performance confirmation examinations—it would be opened in the waste package remediation system (CRWMS M&O 2000p, Attachment II, Section 1.1.4.1).

The processes for opening a waste package or disposal container would include remotely cutting the closure weld, collecting and processing the cutting fines, removing and disposing of the cutting waste, and installing a temporary seal to confine contamination to the inside of the container.

All remediation operations on radioactive waste packages or disposal containers would be performed remotely in a dedicated, shielded cell accessible directly from the large handling cell inside the disposal container handling system. The remediation cell would accommodate one waste package or disposal container at a time. A shield door would open to allow the transfer cart to enter. After the transfer cart enters the remediation cell, the damaged container would be positioned at one of two workstations in the remediation cell and would exit the cell without being removed from the cart. The two remotely operated workstations would accommodate different repair tasks. One workstation would facilitate the cutting, removal, and holding of the container lids. The other would allow remote inspection, examination, and purging of the container, as well as backfilling it with inert gas, temporarily sealing it, and decontaminating it (CRWMS M&O 2000p, Attachment II, Section 1.1.4.2.4).

The remediation system would use a variety of remotely operated equipment, including an overhead bridge crane, an in-cell multipurpose manipulator, a lid-cutting machine, and closed-circuit television viewing systems. System operations would all be performed remotely, using equipment designed to facilitate decontamination, maintenance, and replacement of interchangeable components, as required (CRWMS M&O 2000p, Attachment II, Section 1.1.4.2.4).

If examination of the closure weld indicates an irreparable welding defect, or if a waste package has been retrieved because of suspected failure or damage, the package would be opened. Opening waste packages and disposal containers should be infrequent but would require the capability to unseal the container, vent it, measure the temperature and pressure, and sample the composition of the gas inside. Opening a sealed container would require:

- Cutting the closure welds of the outer and inner lids
- Removing and holding the lids
- Collecting and processing cutting fines
- Removing and disposing of cutting waste
- Installing a temporary seal to confine contamination to the inside of the container.

Following remediation, the container would be inspected for contamination and remotely decontaminated, as required. The container would then

be returned to the disposal container handling system for rewelding, transferred to the assembly transfer system for unloading of fuel assemblies, or transferred to the canister transfer system for unloading of canisters (CRWMS M&O 2000p, Attachment II, Section 1.1.4.2.4).

The remediation system would also interface with the performance confirmation data acquisition/ monitoring system to gather data needed to support the performance confirmation program.

2.2.4.2.8 Waste Handling Building: Ventilation System

The ventilation system for the Waste Handling Building would provide heating, ventilation, and air conditioning for worker health, safety, and comfort and would maintain the appropriate environmental conditions for waste handling operations. It would protect the environment, the public, and workers by preventing airborne emissions of radioactive particulates (CRWMS M&O 2000p, Attachment II, Section 1.1.7.3.1); mechanical equipment selection will consider the application of alternative technology to minimize the use of toxic refrigerants and lubricants.

The Waste Handling Building would have uncontaminated and potentially radiologically contaminated areas. Separate ventilation systems would be installed in these areas to limit the spread of airborne particulates by controlling the air pressure between the areas and directing the airflow from the uncontaminated areas to the potentially contaminated areas (Figure 2-32) (CRWMS M&O 2000p, Attachment II, Section 1.1.7.3.1).

The potentially contaminated area would be divided into three confinement zones: primary, secondary, and tertiary (Figure 2-33). These zones would be classified according to the type, quantity, and packaging of waste to be handled and the potential for contamination from airborne radioactive material or other hazardous materials. The primary zone would normally have some contamination, the secondary zone would have a high potential for contamination, and the tertiary zone would have a low potential for contamination. Each zone would have an independent ventilation system to prevent cross-contamination. The primary and secondary zones would have once-through ventilation; the tertiary zone would use some recycled air. A secondary contamination confinement zone designation is assigned to the various pool water areas and their associated pool water cooling and treatment systems. Ventilation confinement in pool water areas is assigned a tertiary ventilation contamination confinement zone designation. The ventilation system in the confinement zones would remain operational during normal and off-normal operating modes and during and after any design basis event (CRWMS M&O 2000p, Attachment II, Section 1.1.7.3.1).

The air supply equipment for each ventilation system would contain multiple air-handling units, supply-air fans, and associated distribution ductwork. Each air-handling unit would contain a prefilter, final filters, heating coils, cooling coils, and a humidifier. The exhaust air equipment would contain high-efficiency particulate air filtration units, exhaust-air fans, and associated ductwork. Each filtration unit would contain prefilters (moisture eliminators) and high-efficiency particulate air filters. Final exhaust air (after filtration) from all the confinement zones would discharge to the outside environment through a common exhaust air stack with contamination monitoring sensors. The exhaust air stack is located on the south side of the Waste Handling Building; it is designed to be taller than the adjacent structure and is located away from all outside air intakes to preclude reentrainment of any potentially emitted radiological contaminant (CRWMS M&O 2000q, Section 6.2).

The uncontaminated areas of the Waste Handling Building would have a separate ventilation system, configured to maintain proper air quality standards (CRWMS M&O 2000p, Attachment II, Section 1.1.7.3).

The ventilation equipment for the uncontaminated area would consist of air-handling units, supply-air and exhaust-air fans, and associated distribution ductwork (CRWMS M&O 2000q, Section 6.2.5). This separate ventilation system would maintain higher air pressure than the potentially contaminated area to prevent airborne contamination from

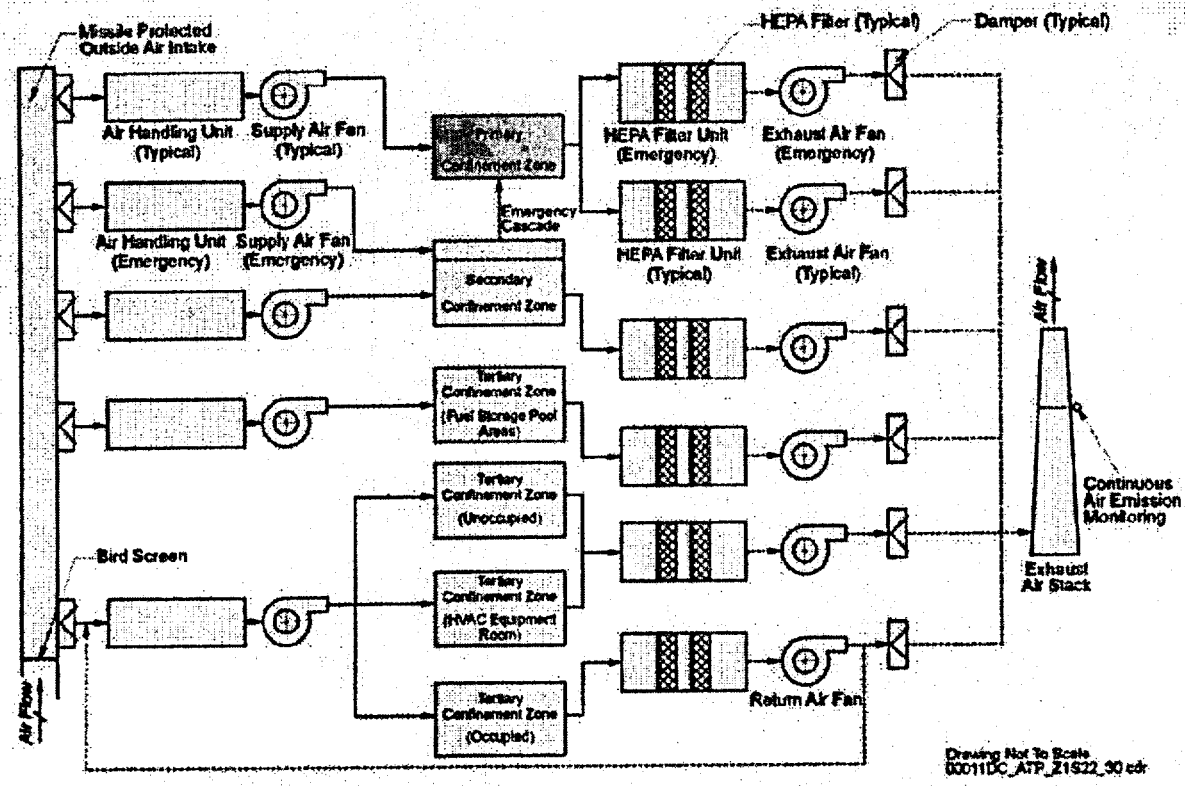


Figure 2-32. Waste Handling Building Heating, Ventilation, and Air Conditioning Confinement Flow Diagram

The heating, ventilating, and air conditioning systems are segmented to control air flow. High-efficiency particulate air filters would be used to trap particulates. The pressure drop across each filter would be measured and the filters would be replaced, as necessary. Tornado dampers would be provided to isolate the systems in case of a tornado and prevent depressurization of the confinement zones. All air flows would be radiologically monitored prior to release from the exhaust air stack, and the systems would shut down automatically in case of an alarm. HEPA = high-efficiency particulate air; HVAC = heating, ventilation, and air conditioning. Source: CRWMS M&O 2000p, Figure I-25.

entering (CRWMS M&O 2000p, Attachment II, Section 1.1.7.3).

2.2.4.2.9 Waste Handling Building: Treatment and Cooling System for Pool Water

The Waste Handling Building would contain nine pools and several transfer canals for underwater waste handling operations. The nine pools consist of two cask unloading pools, two holding pools, one nonstandard pool, and four blending inventory pools. The water treatment and cooling system would:

- Control water temperatures
- Control water quality and radioactive contamination
- Operate the pool's leak detection systems
- Control pool water levels.

Water Cooling System—Spent nuclear fuel assemblies would emit heat, which, if not properly removed, could cause pool water temperatures to increase to unacceptable levels. Therefore, each assembly transfer pool, fuel inventory pool, and associated transfer canal would be connected to a water cooling system to maintain the water temperature at approximately 15°C (60°F). This low temperature would reduce water evaporation and retard the growth of algae, which would improve water clarity (CRWMS M&O 2000p, Attachment II, Section 1.1.8).

The water cooling systems would continually monitor and control the temperature of water in the

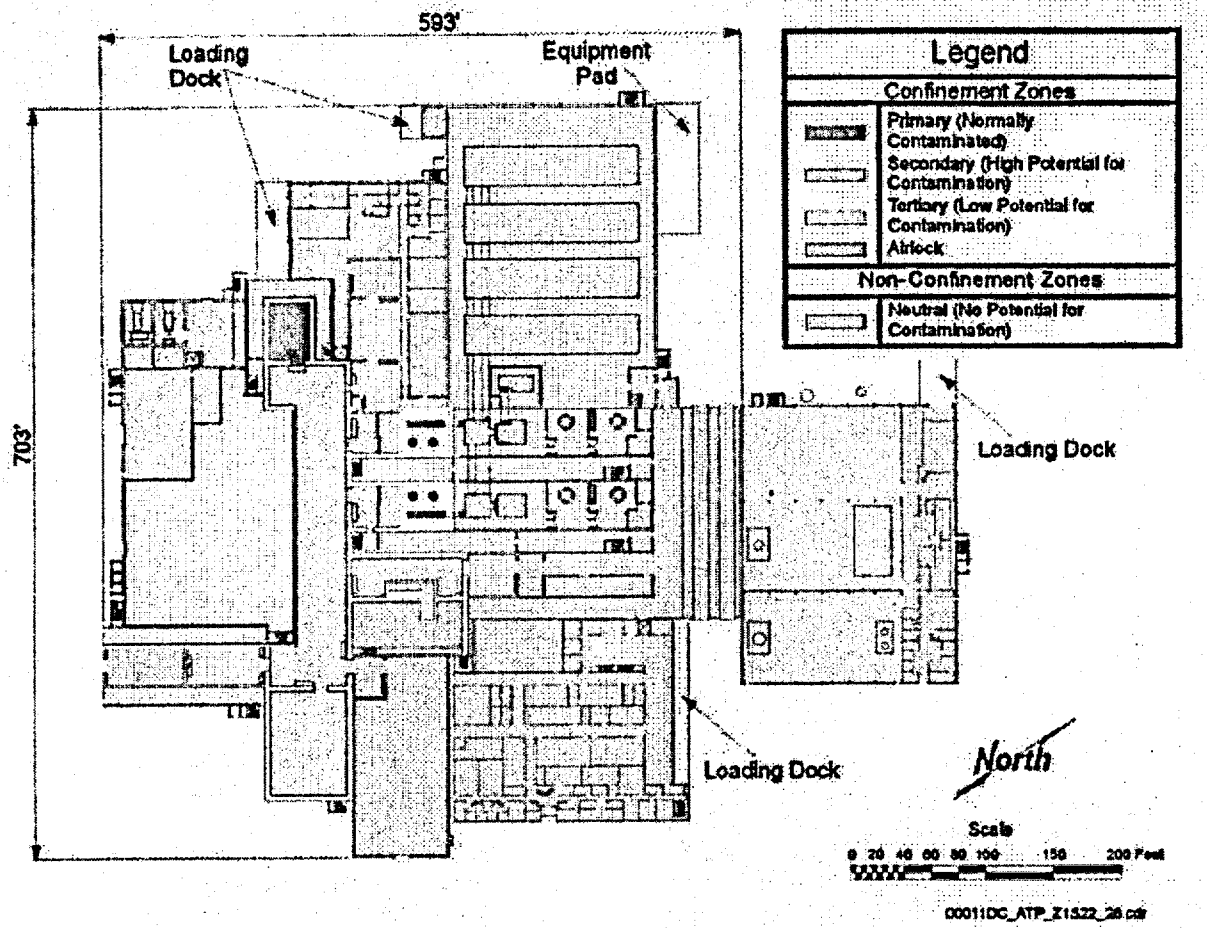


Figure 2-33. Waste Handling Building Confinement Zone Configuration
Multiple heating, ventilating, and air conditioning systems would be used to control air flow from areas of lower potential contamination to areas of greater potential contamination. Elevation: 100 ft. Source: Modified from CRWMS M&O 2000p, Figure I-23.

pool and transfer canal. When the temperature rises to a set point, the system would pump water from the pool through a heat exchanger to cool the pool water and return it to the source. The heat exchanger would transfer heat from the water to an independent chilled-water loop. The chilled-water loop, in turn, would transfer the heat to a refrigerant system, which would discharge it to the environment through air-cooled heat exchangers. The refrigerant system would use an environmentally acceptable refrigeration agent. The water cooling systems would maintain higher pressure on the chilled-water side of the heat exchangers than on the pool side to prevent the chilled water from becoming contaminated if a heat exchanger fails (CRWMS M&O 2000p, Attachment II, Section 1.1.8).

Water Treatment System—The water in the waste handling pools and transfer canals would be subject to continuous radioactive contamination from underwater waste handling operations. Therefore, all assembly transfer pools, assembly holding pools, and associated transfer canals would be connected to a water treatment system that would continuously filter radioactive material and purify the water to keep worker radiation exposures ALARA and maintain pool water clarity (CRWMS M&O 2000p, Attachment II, Section 1.1.8.2.2.1).

The treatment system would continually pump the water in the pools and treatment canals through the system. It would process all water in the pools at least once every 72 hours. The water in the treatment system would first pass through filters to

remove particulates. After filtration, the water would pass through a mixed-bed ion-exchange system to remove residual radionuclides. The deionized water would pass through an organic-material-capturing filter to remove trace organic materials and any ion-exchange resin carried out of the mixed-bed system. Finally, the water would pass through an ultraviolet sterilization system to destroy algae. The system would then return the water back to the pool system (CRWMS M&O 2000p, Attachment II, Section 1.1.8.2.2.1).

The treatment system would also remove floating debris from the surfaces of the pools. Workers would use vacuums to remove particles from pool walls and floors. Water from these cleaning operations would pass through roughing filters to remove large particles before passing through the water treatment system. Workers would process and package the radioactive materials collected in the water-treatment process (the source of generation) and transfer them to the Waste Treatment Building for shipping and disposal off the repository site (CRWMS M&O 2000p, Attachment II, Section 1.1.8.2.2.1).

Leak Detection System—Each waste handling pool would have a leak detection system to identify, locate, contain, and quantify any leakage between pool liners and the surrounding concrete walls and alert personnel if a leak occurs. Any leaking water would collect in sumps below the pools. The sumps would be able to pump accumulated water back to the pools. Personnel would analyze this water and take the appropriate remedial actions (CRWMS M&O 2000p, Attachment II, Section 1.1.8.3).

Water Level Management System—Each waste handling pool would have a water level management system to maintain water levels within design parameters. This would ensure that the pool water provides sufficient shielding to keep worker radiation exposures ALARA (CRWMS M&O 2000p, Attachment II, Section 1.1.8.4). Return water lines would be designed such that a break in the line would not allow the pool to siphon below an acceptable level above the assemblies.

Assembly transfer pools would be equipped with overflow weirs to control the maximum water level during cask handling and unloading operations. Overflow water from the weirs would collect in a sump. The system would reuse this water to supplement the water level when it is too low. If the water level drops below a set point, an automatic system would pump more water into the pools from the sump or from a supplemental water system (CRWMS M&O 2000p, Attachment II, Section 1.1.8.4).

Supplemental Water System—If necessary, a supplemental water system would provide water from the site's deionized water system to increase the water level in a waste handling pool. This supplemental system can provide enough water to compensate for evaporation and other water loss. The supplemental water system would connect to backup power, or a gravity feed system would be used to provide make-up water to the pools in the event of a power outage (CRWMS M&O 2000p, Attachment II, Section 1.1.8.5).

2.2.4.2.10 Waste Handling Building: Fire Protection System

The fire protection system in the Waste Handling Building would provide active and passive features to protect life and property against fire (CRWMS M&O 2000p, Attachment II, Section 2.14.1).

Automatic fire suppression systems would be used in selected areas of the Waste Handling Building. These system would provide the capability to extinguish potential fires in the building (CRWMS M&O 2000p, Attachment II, Section 1.1.9.1).

Sensors would monitor the building for fires and early combustion byproducts. The fire protection system would warn occupants of a fire, alert fire response personnel, and transmit an alarm when the automatic sprinkler system is activated. It would include manual pull stations and detection equipment to initiate alarms. In the event of a fire, the system would automatically transmit an alarm to the Yucca Mountain fire station via radio or dedicated telecommunications lines (CRWMS M&O 2000p, Attachment II, Section 1.1.9.1). Where required for control of fire suppression

system runoff or accumulation, features would be provided to prevent the spread of radiological contamination and criticality.

2.2.4.2.11 Waste Handling Building: Electrical Power System

The Waste Handling Building electrical power system would receive electrical power from the site electrical power system. If the site electrical power supply becomes unavailable, the Waste Handling Building's emergency power supply system would provide electricity to equipment and systems that are important to radiological safety. This system would include a single diesel generator and battery system with enough capacity to supply necessary electrical loads (CRWMS M&O 2000p, Attachment II, Section 1.1.10.1). Workers can test the system during facility operations or while the facility is shut down.

The electrical system for supporting equipment important to safety would be separate from the electrical system supporting equipment not important to safety, and would be able to function during and after any unexpected adverse event (CRWMS M&O 2000p, Attachment II, Section 1.1.10.1). The electrical system important to safety would have redundant load groups, so each group can receive power either from the normal electrical system or from the emergency system.

The DOE would install a grounding system for the Waste Handling Building that would connect to the switchyard substation ground. All major electrical equipment in the Waste Handling Building, including power panel boards, lighting panel boards, and motors, would connect to the grounding system. The building's lightning arresters (i.e., lightning rods) would also connect to the grounding system (CRWMS M&O 2000p, Attachment II, Section 1.1.10.3).

2.2.4.2.12 Repository Operations Monitoring and Control System

Two central control centers would be established in the Geologic Repository Operations Area to monitor and control repository surface operations and perform emergency command functions. The control centers would be located in the Waste Handling Building and the Administration Building.

Operators in the Waste Handling Building control center would monitor and control specific operations in the Waste Handling Building and throughout the plant: for example, track all radioactive material in the building, operate building utilities, and control radiation containment doors. The control system would have local consoles at various locations, including the operating galleries, where workers would control remote waste handling operations while watching the mechanical operations. Operators at the main control center would monitor gallery console operations and respond to emergency and off-normal events. If a gallery console malfunctions or a gallery becomes uninhabitable, the main control center would perform any necessary emergency response functions. Gallery console operators would communicate with main control center operators via secured communication lines (CRWMS M&O 2000p, Attachment II, Section 1.1.11.3). A separate emergency control panel would serve as a backup to the main control center system. If the main system fails, this panel would be able to bring waste handling systems and other systems that are important to safety to a safe operating or shutdown condition (CRWMS M&O 2000p, Attachment II, Section 1.1.11.3).

2.2.4.3 Waste Treatment Building

The Waste Treatment Building would be located on the North Portal pad, north of the carrier bay in the Waste Handling Building. It would house structures, systems, and components that support the collection, segregation, and disposal of both liquid and solid low-level radioactive waste generated during operations. Table 2-8 summarizes these preliminary engineering specifications. It would contain areas for processing nonrecyclable low-level waste, solid low-level waste, and recyclable liquid low-level waste. The building would also contain the equipment, tanks, and piping needed to recycle water in waste handling and waste preparation pools (CRWMS M&O 2000p, Attachment II, Section 1.2.1.4).

Table 2-8. Preliminary Facility Space Specifications for the Waste Treatment Building

Facility Space	Floor Area m^2 (ft^2)
Process area (low-level waste)	3,244 (34,920)
Mixed and hazardous waste	351 (3,780)
Other areas not listed above	3,363 (36,200)
TOTAL	6,958 (74,900)

Source: CRWMS M&O 2000q, Table 6-4.

The activities conducted in the Waste Treatment Building would include:

- Sorting dry waste resulting from repository operations
- Super-compacting and grouting waste resulting from repository operations
- Cleaning liquid waste resulting from repository operations.

The systems for performing these tasks would be located in a limited access area and would be designed to limit worker radiation doses to ALARA levels (CRWMS M&O 2000p, Attachment II, Section 1.2.1.1).

2.2.4.3.1 Waste Treatment Building: Architectural and Structural Features

The Waste Treatment Building would be one story with a mezzanine, approximately 60 m (200 ft) wide and 80 m (260 ft) long. It would be an open, high-bay industrial structure. The main operating floor would consist of a reinforced concrete slab at grade level. The superstructure would be a braced frame of structured steel, with metal siding and a metal deck roof. The Waste Treatment Building would be a lightly loaded building. It would not contain overhead cranes, and the major equipment would be anchored to the ground floor slab. The foundation would consist of individual spread footings of reinforced concrete, which would support the building columns (CRWMS M&O 2000p, Attachment II, Section 1.2.1.5). The Waste Treatment Building would be separated from the Waste Handling Building by a seismic joint to prevent structural interaction during an earthquake (CRWMS M&O 2000p, Attachment II, Section 1.2.1.4). As a safety-related nuclear facility, the Waste Handling Building would be designed to withstand a potential collapse of the Waste Treatment Building.

2.2.4.3.2 Management System for Liquid Low-Level Radioactive Waste

A number of operations at the repository would produce liquid low-level radioactive waste, that is, liquids containing low-level radioactive materials. These liquids would be generated in the processes of handling the transportation casks, the canisters containing DOE high-level radioactive waste or spent nuclear fuel, commercial spent nuclear fuel assemblies, and the waste packages. The main source of this low-level waste would come from decontamination of casks placed into the wet unloading pools, where these casks would be exposed to the contamination that has washed off of the individual spent nuclear fuel assemblies. Depending on the characteristics of the waste and the materials used in processing it, the low-level radioactive liquid waste generated may or may not be recyclable. Therefore, surface facilities would be available to prepare liquid low-level radioactive waste for recycling and reuse, if possible, or for disposal off the repository site in a dry form. There would be two separate systems in the Waste Treatment Building for collecting and treating recyclable and nonrecyclable liquids (CRWMS M&O 2000p, Attachment II, Section 1.5.2).

After decommissioning, or if a pool has to be emptied for pool liner repairs or other reasons, there are several options for treatment of the water in the fuel pools prior to disposal. The water in the pools would be circulated through the pool water treatment system to reduce the radionuclide content to the maximum extent possible. This might involve multiple change-outs of the ion-exchange resin beds in the treatment system. Following this treatment, the water could be routed to the recyclable water treatment system, which uses ion-exchange beds and an evaporator/condenser to treat recyclable water for reuse. The evaporator could be used to evaporate excess water to the atmosphere using the high-efficiency particulate air filtration system and discharge stack.

If ion-exchange and evaporation prove to be unsuitable, the water from the nonrecyclable low-level radioactive waste treatment system could be mixed with grout and disposed at a suitable low-level waste disposal site.

2.2.4.3.3 Management System for Recyclable Liquid Low-Level Radioactive Waste

Recyclable liquid low-level radioactive waste from all repository operations would be collected and treated. Treating and recycling conserves water and reduces the volume of waste to be disposed. Treated liquids would also be recycled to provide water for decontamination activities in radiologically controlled areas of the repository. Any residual liquid from the recycling treatment process would be routed to the system for managing nonrecyclable liquid low-level radioactive waste, solidified into waste materials, packaged, and shipped off the repository site for disposal. There would be no liquid effluents from the recyclable liquid treatment system (CRWMS M&O 2000p, Attachment II, Section 1.5.2).

Recyclable liquid low-level radioactive waste would be segregated from nonrecyclable liquid waste at the point of generation. The recyclable liquid would then be pumped through a piping system to the Waste Treatment Building for processing.

Once in the Waste Treatment Building, the recyclable liquids would be filtered to remove particulates. After filtration, the liquid waste would be transferred to a batch evaporator, where water would be evaporated. The evaporated water would then be condensed and passed through a mixed-bed ion-exchange unit, which would remove any residual radionuclides. The deionized water would pass through an organic-material-capturing filter to remove trace organic materials and any ion-exchange resin carried out of the mixed-bed system. The final recycled water would then be transferred to a recycled-water storage tank, where it would be pumped to the Waste Handling and Waste Treatment Buildings for reuse. During the entire process, specially implemented administrative controls would limit and direct all activities involving liquid waste streams. Figure 2-34 depicts the flow of recyclable liquid low-level radioactive waste through the treatment system (CRWMS M&O 2000p, Attachment II, Section 1.5.2.1.3).

2.2.4.3.4 Management System for Nonrecyclable Liquid Low-Level Radioactive Waste

The system for managing nonrecyclable liquid low-level radioactive waste has two functions: (1) collecting nonrecyclable liquids from repository operations and (2) processing them into a solid form suitable for disposal off the repository site. As a result, there would be no liquid low-level radioactive waste effluents from the repository (CRWMS M&O 2000p, Attachment II, Section 1.5.3).

Nonrecyclable liquid low-level radioactive waste would be segregated from recyclable liquid waste at the point of generation. The nonrecyclable waste would then be pumped through a piping system to the Waste Treatment Building for processing (CRWMS M&O 2000p, Attachment II, Section 1.5.3).

Once in the Waste Treatment Building, the pipes would convey the nonrecyclable liquid radioactive waste into a surge tank. Residual liquid from the evaporation step in the recyclable liquid waste processing system would also be collected in this tank. The contents of the tank would then be mixed, sampled, and analyzed to determine the pH of the liquid. If necessary, acid or caustic solution would be added to neutralize the tank's contents (CRWMS M&O 2000p, Attachment II, Section 1.5.3).

The nonrecyclable liquid low-level radioactive waste would then be mixed, in drums, with Portland cement or another solidifying agent to take out all the liquid. The finished drums would be moved to a curing area and later shipped off the repository site for disposal. Figure 2-35 depicts the flow of nonrecyclable liquid low-level radioactive waste through the treatment system (CRWMS M&O 2000p, Attachment II, Section 1.5.3). The design of the repository would include a processing

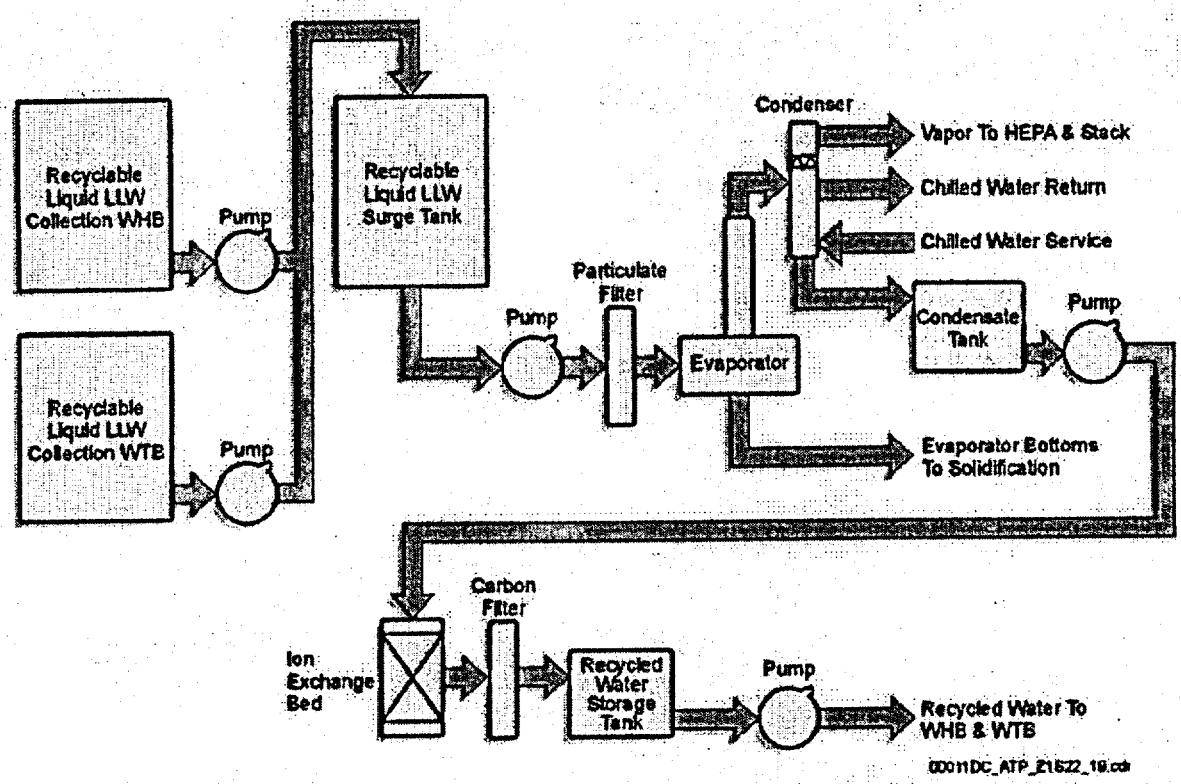


Figure 2-34. Recyclable Liquid Low-Level Radioactive Waste Collection System Diagram
As part of a resources conservation program, water that becomes contaminated via facility operations will be treated with state-of-the-art technology to recover such water for reuse in facility operations. Precipitated particulates (evaporator bottoms) would be transferred to the nonrecyclable liquid low-level waste treatment system. Water would be treated by particulate filter, ion exchange bed, and carbon filter. LLW = low-level radioactive waste; WHB = Waste Handling Building; WTB = Waste Treatment Building; HEPA = high-efficiency particulate air. Source: Modified from CRWMS M&O 2000p, Figure II-3.

system that would minimize the volume of liquid and solid waste.

2.2.4.3.5 Management System for Solid Low-Level Radioactive Waste

A number of operational processes at the repository would produce what is called "solid low-level waste," meaning solids containing low-level radioactive materials. These materials would be generated in the process of handling the transportation casks, the canisters containing DOE high-level radioactive waste and other vitrified waste, the spent nuclear fuel assemblies, and the waste packages that would be emplaced in the subsurface of the repository.

The solid waste would be collected at its point of generation and then treated, usually in the Waste Treatment Building (CRWMS M&O 2000p, Attachment II, Section 1.5.4). Surface facilities would be able to process this waste into a form suitable for disposal off the repository site. Solid waste is composed of two categories, dry solid waste and wet solid waste, discussed in the following paragraphs.

Dry Solid Waste—The system for managing dry solid waste has several functions:

- Collect the waste
- Segregate the waste according to type
- Reduce the volume of the waste
- Package the waste for shipping and disposal off the repository site.

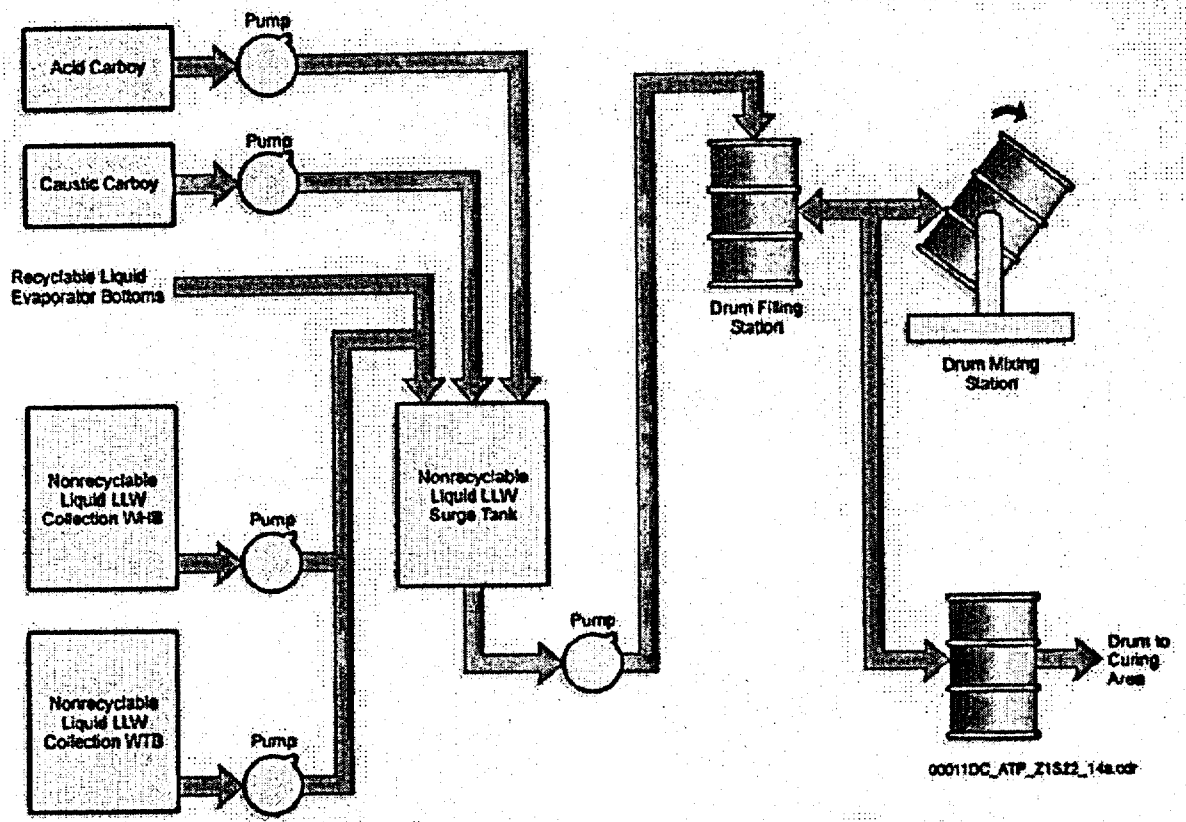


Figure 2-35. Nonrecyclable Liquid Low-Level Radioactive Waste Treatment System Diagram
Materials that cannot be recycled would be mixed in drums with cement or other solidifying agents to create a solid low-level radioactive waste. Such waste would be staged and subsequently transported offsite for disposal. LLW = low-level waste; WHB = Waste Handling Building; WTB = Waste Treatment Building. Source: Modified from CRWMS M&O 2000p, Figure II-4.

Dry solid waste would be collected at its point of generation and transferred to a special area of the Waste Treatment Building for processing. Once inside the Waste Treatment Building, the dry solid waste would first be sorted to separate noncontaminated waste from contaminated waste. The waste would then be sorted to separate noncompactible waste from compactible waste. Noncompactible waste would be reduced in size by mechanical methods, such as shearing or disassembly, before being packaged for disposal. Compactible waste would be fed into a shredder for size reduction. The shredded dry waste would then be compacted into drums for disposal off the repository site. Figure 2-36 presents a simplified flow sketch of the system in the Waste Treatment Building for processing dry solid low-level radioactive waste (CRWMS M&O 2000p, Attachment II, Section 1.5.4).

Wet Solid Waste—The system for managing wet solid waste also has several functions:

- Collect the waste
- Dewater the waste
- Package the waste for shipping and disposal off the repository site.

Materials generated within the Waste Handling Building may be sufficiently contaminated to be classified as "noncontact handled," which means they should be placed into radiation-shielded containers immediately at their point of origin. Noncontact handling is not anticipated for wet solid wastes produced in the Waste Treatment Building (CRWMS M&O 2000p, Attachment II, Section 1.5.4).

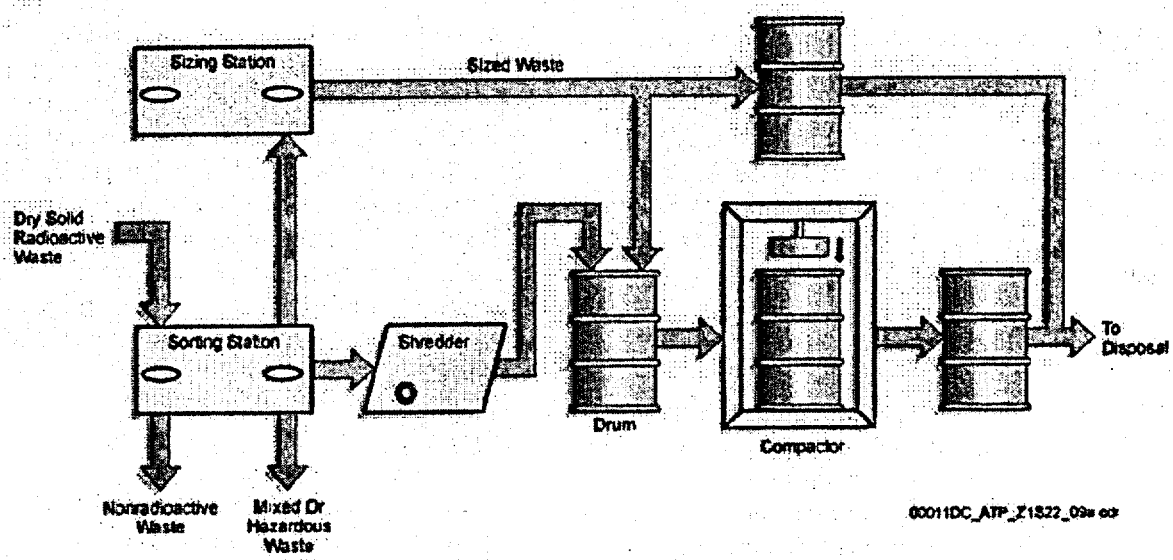


Figure 2-36. Dry Solid Low-Level Radioactive Waste Processing System
Dry waste is of two types. Sized waste consists of such items as contaminated tools and equipment. Such items are not suitable for compaction. The second type of dry waste consists of contaminated clothing, such as overalls and rubber gloves. Such materials are suitable for shredding and subsequent compaction, which reduces the overall volume of such wastes. Drummed wastes would be staged until they are transported offsite for disposal. Source: Modified from CRWMS M&O 2000p, Figure II-5.

Spent Ion-Exchange Resins—Most of the spent ion-exchange resins would be generated in the Waste Handling Building, though a much smaller quantity would be generated in the Waste Treatment Building. Spent ion-exchange resins would be processed at the point of generation. In the Waste Handling Building, ion-exchange resins become waste when they are saturated with radionuclides; likewise, when resin in the Waste Treatment Building's system for processing recyclable liquid low-level radioactive waste becomes saturated with radionuclides, it must be replaced, producing wet solid low-level waste. In both these locations, the waste resins would be transferred directly, via piping systems, to high-integrity disposal containers. Air drying would remove free water from the containers. When the contents are dry, the containers would be loaded into transportation casks for shipping and disposal off the repository site (CRWMS M&O 2000p, Attachment II, Section 1.5.4).

Filter Cartridges—Special filter cartridges would be used in the pool water treatment system of the Waste Handling Building and in the system for treating recyclable liquid low-level waste in the Waste Treatment Building. These cartridges would be changed when they become fully loaded with debris. The cartridge unit would be removed from the system and placed in a 55-gallon drum. The drum's lid would be placed onto the drum remotely. Used filters from the Waste Handling Building that emit more radiation than allowed under the administrative controls for the Waste Treatment Building would be processed at the source of generation. If the filters do not emit more radiation than allowed in the Waste Treatment Building, however, drums containing the used filters would be transferred there for processing. At either location, the drum would be moved to a grouting station, where premixed grout would be added to thoroughly encapsulate the waste filter media. The encapsulated waste would then be shipped for disposal off the repository site (CRWMS M&O 2000p, Attachment II, Section 1.5.4).

2.2.4.3.6 Mixed Waste Management System

Since administrative controls designed into a monitored geologic repository would generally restrict the use of hazardous materials, the DOE does not anticipate that operating a repository would generate a significant amount of mixed waste. If

mixed waste is generated, however, it would be collected and repackaged for disposal at the point of generation. During packaging, samples would be collected for analysis. The packaged mixed waste would then be transferred to the Waste Treatment Building for holding before being transported to a suitable facility (CRWMS M&O 2000p, Attachment II, Section 1.5.5).

2.2.4.3.7 Radiological Safety System

The radiological safety system would ensure that activities associated with operating, monitoring, and closing a repository would limit the exposure of workers and the public to radiation doses. Moreover, the DOE will implement its policy of keeping both occupational doses and possible doses to the public ALARA (CRWMS M&O 2000p, Attachment II, Section 2.7).

The radiological safety system would establish the controls for ensuring that the repository and its operations would have appropriate and sufficient radiation protection features. The system would cover such aspects as facility designs (i.e., shielding and facility layout), operational activities, and facility policies to assure radiation safety. The radiological safety system is part of the overall formal radiation protection program for controlling radiological areas, approving radiological work, and monitoring worker exposures (CRWMS M&O 2000p, Attachment II, Section 2.7).

2.2.4.3.8 Process-Oriented Radiological Monitoring, Sampling, and Analysis Systems

Radiological data is collected and analyzed to:

- Control processes
- Keep doses to workers and the public within applicable regulatory limits and ALARA
- Characterize and classify waste
- Certify compliance with regulations.

Thus, the repository would have a designed-in, process-oriented system for monitoring, sampling, and analyzing radiological data (CRWMS M&O 2000p, Attachment II, Section 1.5.6).

Both online radiological monitoring and direct sample/analysis monitoring would be used in the systems for processing waste generated at the repository. Use of both methods would ensure proper process control and effective quality control of the waste forms. Sampling would be performed with automatic sampling equipment under human control. The analytical equipment would be both computer-controlled and human-controlled (CRWMS M&O 2000p, Attachment II, Section 1.5.6).

Surface dose rates at the ion-exchange beds and filter modules would be continuously monitored to ensure that dose limits are not exceeded. The dose limits would be set to ensure that the radiation protection limits provided by the shielding designs of both the facility and the cask would not be exceeded. Thus, doses to personnel during standard operations would remain within limits, including ALARA goals. Ion-exchange resin beds and filter cartridges would be changed out before their surface dose rates exceed established limits (CRWMS M&O 2000p, Attachment II, Section 1.5.6).

Samples would be analyzed in laboratory areas in the Waste Handling and Waste Treatment Buildings. For quality assurance, samples may also be analyzed at offsite commercial laboratories (CRWMS M&O 2000p, Attachment II, Section 1.5.6). In addition, computer-controlled instruments would perform continuous online monitoring and notify operators if radiation set points are approached (CRWMS M&O 2000p, Attachment II, Section 1.5.6).

2.2.4.3.9 Waste Treatment Building: Ventilation System

The ventilation system would protect the environment and public by preventing airborne effluents and emissions of radioactive or other hazardous contaminants from the Waste Treatment Building. The Waste Treatment Building would have designated uncontaminated and potentially radiologically contaminated areas. Separate ventilation systems would be used in these areas to limit the spread of airborne contaminants by controlling the air pressure between the areas and directing airflow

from the uncontaminated areas to the potentially contaminated areas (CRWMS M&O 2000p, Attachment II, Sections 1.2.5.1 and 1.2.5.3).

The potentially contaminated area may receive airborne contamination from low-level radioactive waste or other hazardous materials during or after an off-normal operating event. Therefore, it would have an independent ventilation system, preventing cross-contamination with the uncontaminated area. The air pressure in the potentially contaminated area would be maintained at a lower pressure than the outside environment to prevent leakage to the outside. The ventilation system in the potentially contaminated area would remain operational at all times (CRWMS M&O 2000p, Attachment II, Section 1.2.5.3.1).

2.2.5 Surface Facilities Radiological Control and Management Systems

This section describes the process involved in designing the repository surface facilities to comply with the site's radiation protection criteria. As for other nuclear facilities, a repository must be designed so that the protected (i.e., radiologically controlled) spaces within its operation areas maintain radiation levels and doses, as well as concentrations of radioactive material in the air, to within specified limits. Nuclear licensees make every reasonable effort to maintain radiation exposures even lower—as far below the dose limits as is reasonably achievable. This goal of ALARA means the DOE will make every reasonable effort to maintain exposures as far below the established dose limits as is practical (CRWMS M&O 2000p, Attachment II, Section 1.9.1).

The DOE plans to continue development of a radiation protection program to ensure consistency with 10 CFR Part 20 criteria and to incorporate ALARA guidelines and training into all phases of the design and operations of the repository (CRWMS M&O 2000p, Attachment II, Section 1.9.1.1).

Radiation protection would be planned and implemented in a multifaceted manner. A sound program of designing for radiation protection must have the following essential elements (CRWMS M&O 2000p, Attachment II, Section 1.9.1):

- Management commitment to the program
- Training in the ALARA approach
- Oversight of the design program by radiation protection specialists
- Radiation protection analysis
- An effective means for monitoring radiation in facilities and during operations
- A plan for implementing the radiation protection program during actual operations.

During design, engineers identify key structures, systems, and components. Radiation protection personnel perform occupational dose assessments based on this identification. Finally, system designs are evaluated against performance criteria to assure compliance. Compliance criteria can include (CRWMS M&O 2000p, Attachment II, Sections 1.9.1.1 and 1.9.1.2):

- Total cumulative occupational dose
- Maximum individual dose
- Average individual dose
- Person-rem reductions.

2.2.5.1 Dose Assessment and Designing for ALARA Goals

Dose assessments combine estimated dose rates in radiologically controlled work areas, the time needed to perform a task, the number of personnel performing a task, the type of personnel performing a task, and how often a task is performed each year. These factors provide the average exposure to an individual in a work group and the total group exposure. These values can then be evaluated to determine how effectively the design protects workers from radiation exposure (CRWMS M&O 2000p, Attachment II, Sections 1.9.1.1 and 1.9.1.2).

The design group would use the dose assessment information to evaluate where the design can be changed to reduce projected doses cost-effectively. As the design progresses, the dose assessments may be modified to incorporate design changes, as well as more complete data on labor, layout, and

equipment requirements (CRWMS M&O 2000p, Attachment II, Sections 1.9.1.1 and 1.9.1.2).

Dose assessments would be performed on all significant work in radiation areas, including normal, abnormal, accident, and recovery events and activities. The dose assessment can also be useful in evaluating decommissioning options (CRWMS M&O 2000p, Attachment II, Sections 1.9.1.1 and 1.9.1.2).

The health safety system would monitor repository workers' exposures to radiation and prevent personnel from receiving doses above applicable regulations and DOE ALARA limits for radiological workers (10 CFR Part 20). The system would include radiation monitors in repository facilities, personal dosimetry equipment, and decontamination facilities and equipment.

The health safety records of all personnel performing waste handling and other radiological work at the repository would be monitored and tracked (CRWMS M&O 2000p, Attachment II, Section 2.12). To enter the radiologically controlled areas, workers would read, sign, and perform work according to the instructions and conditions stated on the radiation work permits or on special work permits (CRWMS M&O 2000p, Attachment II, Section 2.12.1). Monitors would control access to radiologically controlled areas according to each worker's permit status and radiation exposure history. Personnel entering or leaving controlled facilities would pass through radiation detection portals that can detect very low levels of radioactive contamination. Any workers who have detectable contamination on them would undergo a decontamination procedure (CRWMS M&O 2000p, Attachment II, Section 2.12.1).

2.2.5.2 Radiological and Emergency Response Systems

The purpose of the site radiological monitoring system area-radiation and continuous-air monitors would be to keep radiation doses to repository workers ALARA. The site radiological monitoring system would continually monitor the repository surface facilities for general area radiation levels and concentrations of airborne radioactive particulates. Monitors would also test air in the exhaust stacks for radiological effluents and emissions into the ventilation exhausts of the Waste Handling Building and the Waste Treatment Building. If radiation in the exhaust air from these buildings exceeds preset threshold levels, the monitors would trigger alarms (CRWMS M&O 2000p, Attachment II, Sections 1.1.13 and 2.7).

The system would provide continuous local and central displays of all radiation levels. It would audibly warn workers of unsafe radiation levels and trends and would communicate with the central control and radiation protection systems. The range of these monitoring systems would be specified to ensure that the instruments are on scale during both normal operations and potential off-normal occurrences. The site radiological monitoring system as a whole, and its component instruments individually, would be designed to self-test their operating status and calibration, recording results and reporting anomalies and failures (CRWMS M&O 2000p, Attachment II, Section 1.1.13.3).

2.2.6 Site-Wide Support Systems

This section describes the structures, systems, and components that provide common or site-wide support necessary for safe waste handling operations. These systems include emergency response and management, safeguards and security, maintenance and supply, and site power.

2.2.6.1 Emergency Response System

The emergency response system monitors the status of top-level systems such as the operations monitoring and control, security, and emergency response systems. It would provide emergency analysis and dispatch personnel to respond to site off-normal conditions (CRWMS M&O 2000p, Attachment II, Section 2.14). The following provide data to the emergency response system:

- Fire station
- Safeguards and security
- Health and safety
- Radiological monitoring
- Site environmental monitoring

- Meteorological monitoring and forecasts
- Emergency communications
- Site utility systems.

The emergency response system is composed of facilities, equipment, personnel, and instruments that support emergency response, medical emergencies, and underground response (CRWMS M&O 2000p, Attachment II, Section 2.14).

2.2.6.2 Site Fire Protection

The fire protection system for the surface facilities would include alarm systems for detecting explosions and fires and include the appropriate facilities, staff, and equipment for extinguishing fires and responding to emergencies. The design for the fire protection system will comply with applicable DOE requirements (CRWMS M&O 2000p, Attachment II, Section 2.6.3).

Each surface facility would provide fire alarms to warn occupants of a fire, signal activation of automatic fire protection systems, and alert fire response personnel. Surface facilities would connect to the site fire alarm system, so if a fire alarm sounds in a surface facility, a signal would transmit to the site fire station (CRWMS M&O 2000p, Attachment II, Section 2.6.3). The site fire protection system would also include a firewater system with the appropriate flow and pressure for automatic fire sprinklers and hoses to suppress a fire. Water storage and pumping systems would be provided by the DOE to satisfy this requirement.

A central fire annunciator in the fire station would monitor fire alarms throughout the site. This system would also monitor sprinkler systems, deluge fire systems, firewater pumps and tanks, and special fire protection systems (CRWMS M&O 2000p, Attachment II, Section 2.6.4).

2.2.6.3 Surface Environmental Monitoring System

The surface environmental monitoring system would monitor weather, seismic, air quality, and radiological data throughout the plant. The monitoring system would interface with the central command control system by providing the collected data. The data would be analyzed and appropriate action initiated in the central command control room. The system would help protect the public, the environment, repository personnel, and property from potential radioactive and hazardous material emissions (CRWMS M&O 2000p, Attachment II, Section 2.8.3).

The surface environmental monitoring system would use monitoring and testing instruments located throughout the surface facilities. A complete data management and distribution system would store and distribute the environmental information acquired by the monitors and instruments, as well as the analyses and reports necessary to meet regulatory compliance requirements (CRWMS M&O 2000p, Attachment II, Section 2.8.3).

An additional laboratory-based information management system (CRWMS M&O 2000p, Attachment II, Section 2.8.3) would allow further data acquisition, real-time process control, and analyses of the acquired data to:

- Provide information for managing hazardous materials and reporting hazardous and toxic chemical inventories, as required by the Emergency Planning and Community Right-to-Know Act of 1986 (42 U.S.C. 11001 et seq.)

- Monitor the overall environmental situation at the repository site (for example, the meteorological monitoring system would collect meteorological information about the site and provide weather forecasting and climatological data)

- Support shallow subsurface exploration and analysis of general geologic conditions, including collecting vegetation and soil samples and analyzing their physical and chemical properties

- Provide seismic monitoring and analysis by collecting data for ongoing seismic surveys of the region.

2.2.6.4 Safeguards and Security System

The safeguards and security system would protect the public by safekeeping the physical inventory of radioactive waste at the repository against radiological sabotage. Under the safeguards and security system, workers would record the receipt of all spent nuclear fuel and high-level radioactive waste as it arrives and track it as it is prepared for emplacement. The waste packages would be welded closed and provided with proper identifiers before emplacement. Identifiers would be securely attached on the outside of each waste package.

This record would contain specific information for each waste package, including:

- Identification numbers
- A description of the material in the package
- Shipping manifests and other information on the origin of the materials
- Documentation on any removal of a waste package from the repository.

In addition, the DOE would periodically inventory the radioactive waste onsite and generate nuclear material transfer and other status reports. Physical inventory of nuclear waste would be performed by reviewing emplacement records and, if applicable, removal records (CRWMS M&O 2000p, Attachment II, Section 1.7.1).

The safeguards and security system would provide protection against loss of physical control of radioactive wastes through physical barriers, access controls, continuous surveillance, intrusion detection, and an alarm system. The system would also respond to barrier intrusions and threat assessments, communicate reportable events to required agencies, and provide for evacuation of facility personnel. The safeguards and security system would comply with applicable regulatory requirements (CRWMS M&O 2000p, Attachment II, Section 1.7).

2.2.6.5 Maintenance and Supply System

The primary waste handling systems would be operational about 70 percent of the time (6,000 hours per year) but unavailable for the remaining 30 percent for regular maintenance. Workers would perform scheduled preventive maintenance and periodic repair, replacement, and testing of equipment (CRWMS M&O 2000p, Attachment II, Section 2.10.3).

Equipment failures may cause a system or area to be taken down for corrective maintenance or repair. Preventive maintenance would be scheduled based on the likely failure rate of a piece of equipment, as determined by mean-time-between-failure criteria. These rates would be predicted from models of waste handling systems. Thus, both scheduled and unscheduled maintenance downtimes would be incorporated into the repository design (CRWMS M&O 2000p, Attachment II, Section 2.10.3).

The maintenance and supply system would have a structured maintenance program. The appropriate quantity and type of equipment, spare parts, components, and supplies needed for both scheduled and unscheduled maintenance to support repository operations would be procured, stored, and distributed. The system would use maintenance schedules developed for corrective, preventive, and periodic repair and replacement of surface facility components (CRWMS M&O 2000p, Attachment II, Section 2.10.3).

2.2.6.6 Site Electrical Power

The site electrical power system would receive and distribute utility power to all repository facilities that require electrical service. The electrical system, remotely monitored and controlled from the surface operations monitoring and control system, would be functional during all active phases of the repository's lifetime: construction, operation, monitoring, closure, and decontamination and decommissioning (CRWMS M&O 2000p, Attachment II, Section 2.2.1).

A transmission line from offsite would terminate in the main substation, where it would connect to transformers through substation power circuit breakers and disconnect switches. From the main substation, the system would branch out to four primary electrical distribution points: a North Portal substation, a South Portal substation, and

two shaft substations (CRWMS M&O 2000p, Attachment II, Section 2.2.3.3). Figure 2-37 shows the arrangement of the site electrical power system.

Standby power would be provided for personnel safety items and the subsurface ventilation system, if needed (CRWMS M&O 2000p, Attachment II, Section 2.2.1).

2.2.6.7 Site Solar Power System

Electricity from renewable energy sources would also be used at the repository (Griffith 2000). Solar power would be used in conjunction with commercially available power to meet power requirements. Solar power panels would be added in modular fashion up to 3 MW.

2.2.7 Operational Maintenance

The repository operational maintenance program would keep all facilities and systems important to repository operations and safety in working order. The maintenance program would be based on federal regulations and nuclear industry standards for maintenance on similar facilities and systems at other nuclear sites. It would include the necessary work processes, training, and equipment for inspecting, testing, and maintaining structures, systems, and components important to safety. The system would also maintain operational readiness (CRWMS M&O 2000p, Attachment II, Section 3).

The repository maintenance program would ensure that all maintenance personnel have the necessary qualifications, experience, licenses, and training for specific maintenance tasks (CRWMS M&O 2000p, Attachment II, Section 3). It would require

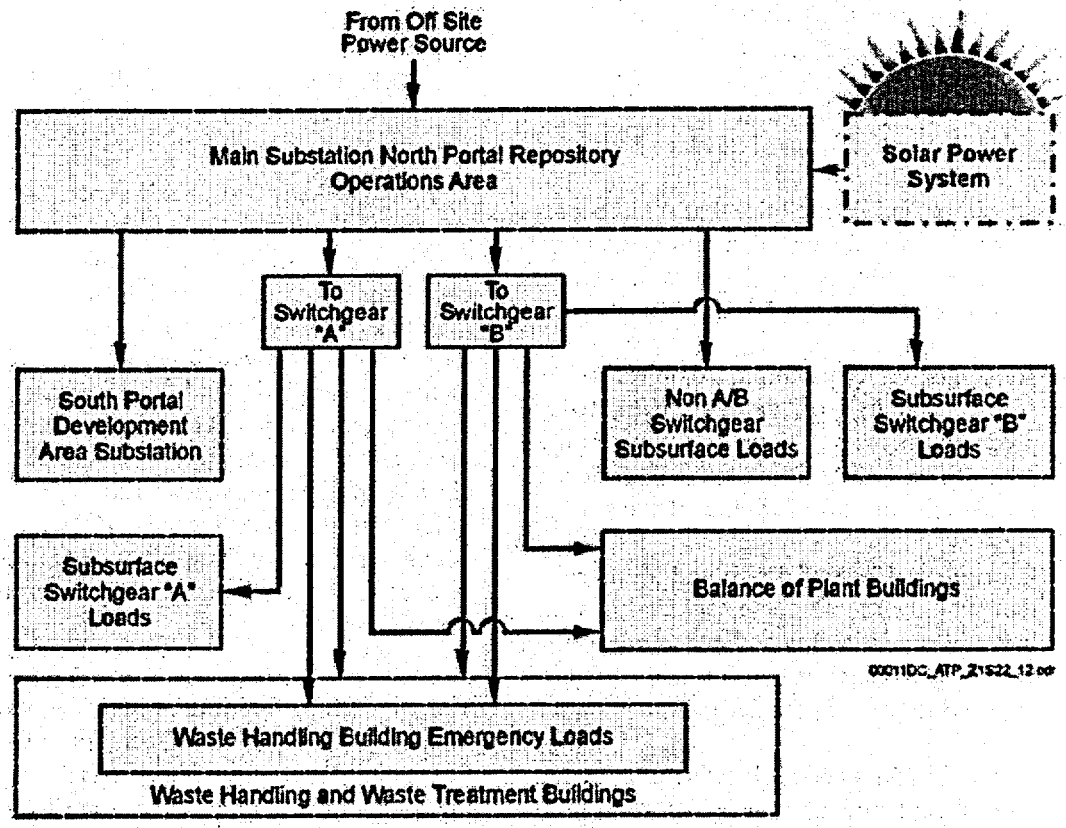


Figure 2-37. Monitored Geologic Repository Site Electrical Power Distribution Diagram
The facility would rely on a combination of electricity from commercial generating sources, as well as the sun.
Source: CRWMS M&O 2000p, Figure I-32.

personnel to perform all routine and nonroutine maintenance activities under administrative controls and according to established procedures and job-specific safety criteria. Health physics personnel would assist maintenance staff with all work in radiologically controlled areas (CRWMS M&O 2000p, Attachment II, Section 3). The maintenance program would use a maintenance activity database to track information on maintenance actions, system failures, and corrective actions. The database would include detailed information, such as the time and location of every failure, its type and cause, the repairs needed, and descriptions of any parts needed to make the repairs. This information would be valuable for controlling material inventories and establishing trends in equipment failures.

Maintenance program personnel would review all NRC reports on nuclear industry activities and incidents, then incorporate lessons-learned information from other sites into the repository maintenance program (CRWMS M&O 2000p, Attachment II, Section 3).

2.2.8 Decontamination and Decommissioning of Surface Facilities

Any repository surface facilities that could become contaminated are designed to simplify decontamination. Thus, the design of the surface facilities would support three general DOE goals:

- Heath and safety performance objectives that would limit loss of, or damage to, federal property, including losses resulting from an inability to readily decontaminate or decommission facilities for subsequent uses
- Interior corridors of a size and arrangement that would accommodate the facility's ultimate decontamination and decommissioning, including enough room for the equipment required to perform the decontamination
- Dispersion limits on radioactive or other hazardous contaminating materials to simplify periodic decontamination and the ultimate decontamination and decommissioning of the facility for disposal or reuse.

To more directly support these goals, surface facilities would include specific features for future decontamination and decommissioning. For example, items like service piping, conduits, and ductwork would be kept to a minimum in areas of potential contamination; where they are necessary, they would be arranged to ease the job of decontamination. Ventilation system filters would be located to minimize potential contamination of ductwork. Walls, ceilings, and floors would be finished with washable or strippable coverings. Metal liners would be used where needed to ensure a means of decontamination (e.g., pool and transfer cell areas).

Taken together, these and other design features would serve the goals of decontamination and decommissioning of the surface facilities, while permitting reuse or proper disposal of federal property after the repository is sealed and closed (CRWMS M&O 2000p, Attachment II, Section 1.10.3).

2.3 REPOSITORY SUBSURFACE FACILITIES

The design described in this report is focused on protecting the health and safety of both repository workers and the public. To meet these requirements, an integrated set of surface and subsurface facilities is being designed. Operation of a repository would involve a number of distinct but interrelated activities and functions; the major ones include repository development, waste emplacement, retrieval capability, and eventual sealing and closure of the repository. The following design description discusses the subsurface facilities for a potential repository for spent nuclear fuel and high-level radioactive waste, based on current operating concepts.

The subsurface facilities would be located in the South Portal Repository Operations Area and the Subsurface Repository Operations Area. The South Portal Repository Operations Area would support continuing construction of the repository as the North Portal Repository Operations Area accepts and prepares waste for underground emplacement.

This section focuses on the proposed design and development of the potential subsurface repository. A continual emphasis on environmental, industrial, and radiological health and safety criteria would be designed into all of the subsurface facilities and systems.

This section contains design descriptions for the major subsurface facilities, including:

- Access ramps
- Main drifts
- Exhaust mains
- Turnouts
- Emplacement drifts
- Ventilation shafts and ancillary subsurface and surface facilities
- Mechanical and structural support systems
- Underground utilities
- Waste emplacement and retrieval equipment
- Surface-based controls for subsurface operations.

The information provided in this section on the proposed design and development of the potential subsurface repository is summarized from a large set of design descriptions and analyses, including:

- *Site Recommendation Subsurface Layout* (BSC 2001d)
- *Ground Control for Emplacement Drifts for SR* (CRWMS M&O 2000w)
- *Overall Subsurface Ventilation System* (CRWMS M&O 2000x)
- *Subsurface Transporter Safety Systems Analysis* (CRWMS M&O 2000y)
- *Bottom/Side Lift Gantry Conceptual Design* (CRWMS M&O 2000z)
- *Retrieval Equipment and Strategy for WP on Pallet* (CRWMS M&O 2000aa).

2.3.1 Repository Design Capacity

The subsurface repository design considers two cases. The base case, also referred to as the "statutory case," addresses a nuclear waste storage capacity of 70,000 MTHM (63,000 MTHM of commercial spent nuclear fuel and 7,000 MTHM of DOE waste). The NWPA limits the first repository capacity to no more than 70,000 MTHM until such time as a second repository is in operation. A second case, referred to as the "full inventory" case, would have a capacity of 97,000 MTHM, while not precluding 119,000 MTHM (referred to as "additional repository capacity") for bounding purposes (see Stroupe 2000, Attachment 1).

2.3.1.1 The Base Case Repository Layout

The 70,000-MTHM subsurface repository layout (Figure 2-38) consists of 58 emplacement drifts excavated to a 5.5-m (18-ft) diameter at a center-to-center drift spacing of 81 m (266 ft). The emplacement drifts would be excavated between the east and west mains with an azimuth of 252 degrees. Emplacement drifts 1 through 51 are required for emplacement of the waste package inventory. Emplacement drift 52 would be excavated as an allowance for possible variances in the waste stream. Emplacement drifts 53 through 58 are contingency drifts that would only be excavated in the event that conditions during development of the repository are found to be such that the intended emplacement area cannot be used because of unexpected adverse ground conditions. The required total emplacement drift length to accommodate the 70,000 MTHM is 56,222 m (184,455 ft), which requires an area of approximately 1,150 acres and results in an areal mass loading of approximately 56 MTHM/acre (this rate is calculated based on the tonnage of commercial spent nuclear fuel only, that is, 63,000 MTHM). The repository emplacement drift average thermal load, based on line-loading with 10 cm (4 in.) end-to-end spacing between waste packages, is 1.42 kW/m for the base case layout (CRWMS M&O 2000h, Section 2.3.1; BSC 2001d, Section 6.3.1). Table 2-9 summarizes the preliminary subsurface excavation dimensions for the base case repository layout.

Within the emplacement area, six standby and cross-block drifts with a diameter of 5.5 m (18 ft) would be excavated within the pillars of adjacent emplacement drifts (see Figure 2-38). The two standby drifts would be located within the first half of the emplacement drifts, to be available early for possible relocation of waste packages. The four cross-block drifts would perform possible ventilation, monitoring, emergency egress, and

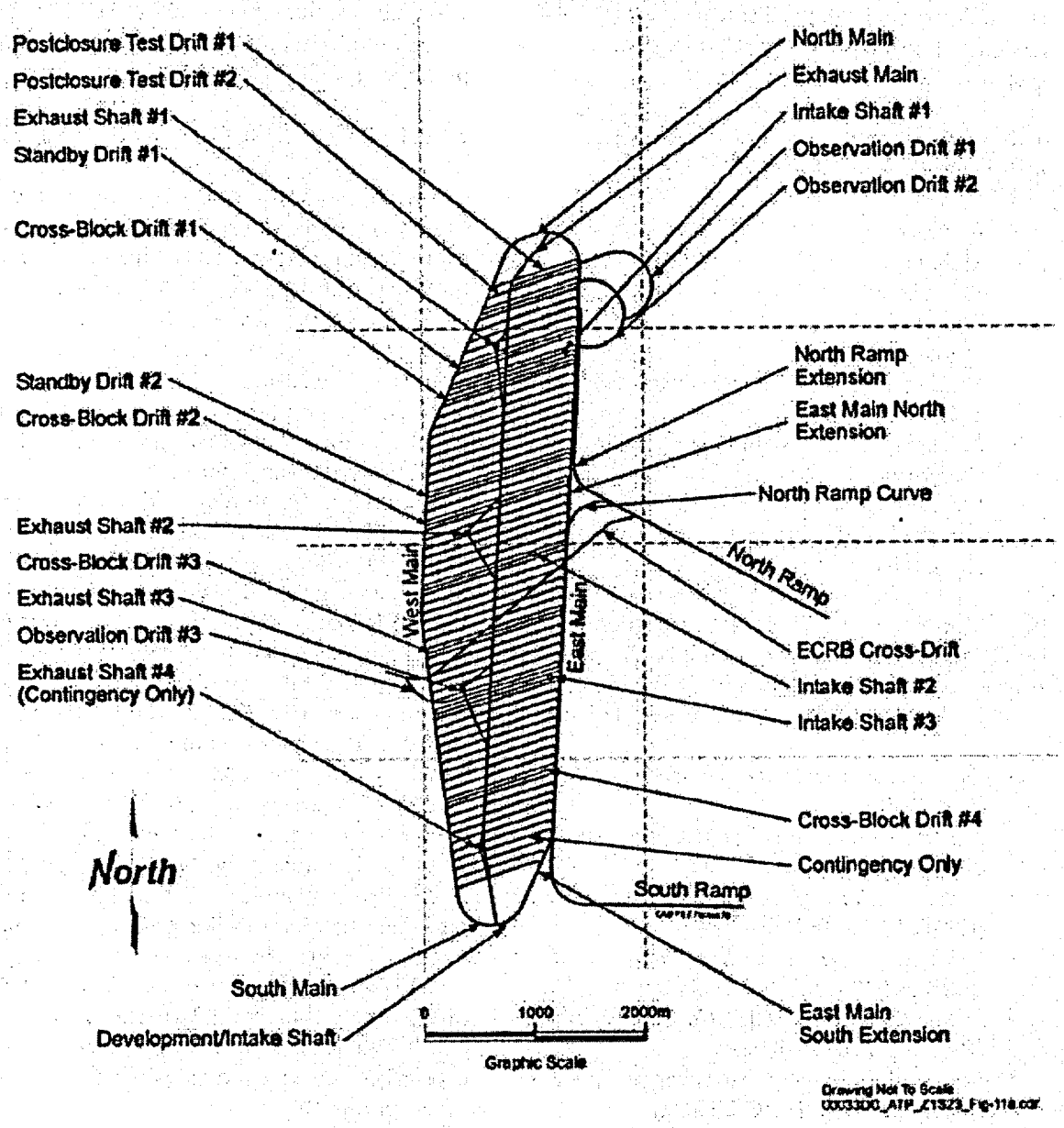


Figure 2-38. Repository Layout for the 70,000-MTHM Case
The repository layout for the subsurface facilities illustrated in this figure will accommodate 70,000 MTHM of spent nuclear fuel and high-level radioactive waste. This quantity represents the statutory capacity for the repository as mandated by the NWPA and is referred to in this report as the base case. Source: BSC 2001d, Section 6.3.

Table 2-9. Preliminary Subsurface Excavation Dimensions for the Base Case Repository Layout

Designation	Excavated Diameter or Size meters (ft)	Length meters (ft)	Notes
Access Ramps • North Ramp • South Ramp	 7.62 (25.0) 7.62 (25.0)	 2,804 (9,199) 2,223 (7,293)	The access ramps were constructed as part of the Exploratory Studies Facility and will be upgraded for use by the repository
Main Drifts • East Main • North Main • South Main • West Main • North Ramp Extension • North Ramp Curve	7.62 (25.0)	12,988 (42,612)	Typical 300 mm (1 ft) concrete layer over concrete invert segment
Exhaust Main Drift	7.62 (25.0)	6,542 (21,463)	Located below the emplacement horizon
Turnouts	7.0 × 8.0 (23.0 × 26.2)	Varies	7 m (23 ft) high with 4 m (13.1 ft) radius arch; 24 m (78.7 ft) straight section and variable curve lengths
Emplacement Drifts	5.50 (18.0)	56,222 (184,455)	52 each (for base case) on 81-m (266 ft) center-to-center spacings. Typical 908-mm (3-ft) thick circular segment invert fill.
Enhanced Characterization of the Repository Block Cross-Drift	5.0 (16.4)	2,681 (8,796)	This facility is already constructed; measured length from as-built surveys.
Observation Drifts • Observation Drift #1 • Observation Drift #2 • Observation Drift #3	 5.50 (18.0) 5.50 (18.0) 5.50 (18.0)	 1,931 (6,335) 2,103 (6,900) 1,568 (5,144)	
Ventilation Shafts	8.0 (26.2)	Varies	Three intake, three exhaust, one development; variable lengths because of fluctuations in surface terrain
Ventilation Raises	2.0 (6.6)	15 (49.2)	
Alcoves	Varies	Varies	Various configurations, minor excavations

NOTES: Dimensions are approximate unless already constructed. Sources: BSC 2001d, Sections 6.2 and 6.3; for Enhanced Characterization of the Repository Block Cross-Drift dimensions, CRWMS M&O 1999f, Section 4.2.

performance confirmation functions. The 70,000-MTHM layout includes other drifts, such as the postclosure test drifts and the observation drifts, which are used for the performance confirmation program discussed in Sections 2.5 and 4.6.

The access mains to the emplacement area for the 70,000-MTHM layout include the north main, west main, north ramp extension, east main north extension, east main south extension, and south main. These access mains account for approximately 12,988 m (42,612 ft) of excavation. The access mains would be excavated to a diameter of 7.62 m (25 ft). The east and west mains, including the east main north and south extensions, would be used primarily for access to the emplacement drifts during waste emplacement operations. The north and south mains have been configured between the east and west mains to provide access from one side of the repository block to the other. The north ramp extension is provided to the north end of the repository block as a bypass from the north ramp to allow waste emplacement operations concurrent with subsurface facilities development.

The exhaust main, excavated to a diameter of 7.62 m (25 ft), is approximately 6,542 m (21,463 ft) in length and is situated a minimum of 10 m (33 ft) below the emplacement horizon, extending from the north end of the repository block to the extreme south end. The exhaust main is graded upwards in the vicinity of the north and south mains to allow interception for access into the exhaust main from either end of the repository block. The ventilation system uses three intake shafts, excavated to a diameter of 8 m (26.2 ft), in addition to the north and south ramps, for air intake. The development shaft in the south end,

also excavated to a diameter of 8 m (26.2 ft), provides air intake for the construction and development operations for the subsurface facility. Three exhaust shafts, excavated to a diameter of 8 m (26.2 ft), are required to exhaust air from the subsurface facility. Exhaust Shaft #4 would only be required if excavation within the emplacement contingency area is necessary.

2.3.1.2 The Full Inventory Repository Layout

A 97,000-MTHM layout (Figure 2-39) requires 90 emplacement drifts. Drifts 1 through 78 are required for emplacement of the waste package inventory. Drifts 79 and 80 would be excavated as an allowance for possible variances in the waste stream. Drifts 81 through 90 are contingency drifts that would only be excavated in the event that conditions during development of the repository are found to be such that the intended emplacement area cannot be used because of unexpected adverse ground conditions. Emplacement drift orientation is the same as that for the base case layout.

The required emplacement length to accommodate a 97,000-MTHM layout is approximately 74,214 m (243,484 ft), which requires an area of approximately 1,485 acres. This layout results in an areal mass loading of approximately 57 MTHM/acre (this rate is calculated based on the tonnage of commercial spent nuclear fuel only, that is, 83,800 MTHM) and an emplacement drift average thermal load of 1.43 kW/m for the same waste package line-loading condition as the base case (CRWMS M&O 2000h, Section 2.3.2; BSC 2001d, Section 6.4.1).

2.3.1.3 Additional Repository Capacities

In addition to the full inventory design for the repository, the design does not preclude the flexibility to be expanded to accommodate greater quantities (up to 119,000 MTHM). This amount of waste could be generated only if numerous commercial nuclear reactors receive extensions to their initial operating lives from the NRC. This expanded repository capacity could be used to dispose of both additional commercial spent nuclear fuel and DOE high-level radioactive waste and spent nuclear fuel. As much as 42,000 MTHM of commercial spent nuclear fuel, in addition to 14,250 canisters of DOE high-level radioactive waste and spent nuclear fuel, may require disposal. These amounts are over and above the 70,000 MTHM base case repository layout.

2.3.2 Functional Requirements

The repository facilities have been designed to be fully integrated, using the systems engineering approach to identify functional requirements based on performance objectives and then fulfill these requirements by providing design solutions. This section describes the functional attributes of the systems considered important to preclosure radiological safety.

2.3.2.1 Subsurface Systems Functions

All 17 subsurface systems were evaluated and classified according to their importance to preclosure radiological safety (see Section 2.1). Three systems were classified as QL-1, two as QL-2, one as QL-3, and eleven as CQ. The QL-1 systems are those considered most critical to radiological safety; the CQ systems are those considered to have no impact on radiological safety. Table 2-10 provides the subsurface system safety classifications.

Six principal systems (shaded in Table 2-10) from this set of 17 were prioritized for development of detailed System Description Documents. These systems comprise components that are key to understanding the performance of the potential repository at Yucca Mountain. Preliminary engineering specifications, as required by the NWPA (42 U.S.C. 10101 et seq.), were developed for the selected systems. The six System Description Documents produced were:

- *Subsurface Facility System Description Document* (CRWMS M&O 2000h)
- *Emplacement Drift System Description Document* (CRWMS M&O 2000ab)

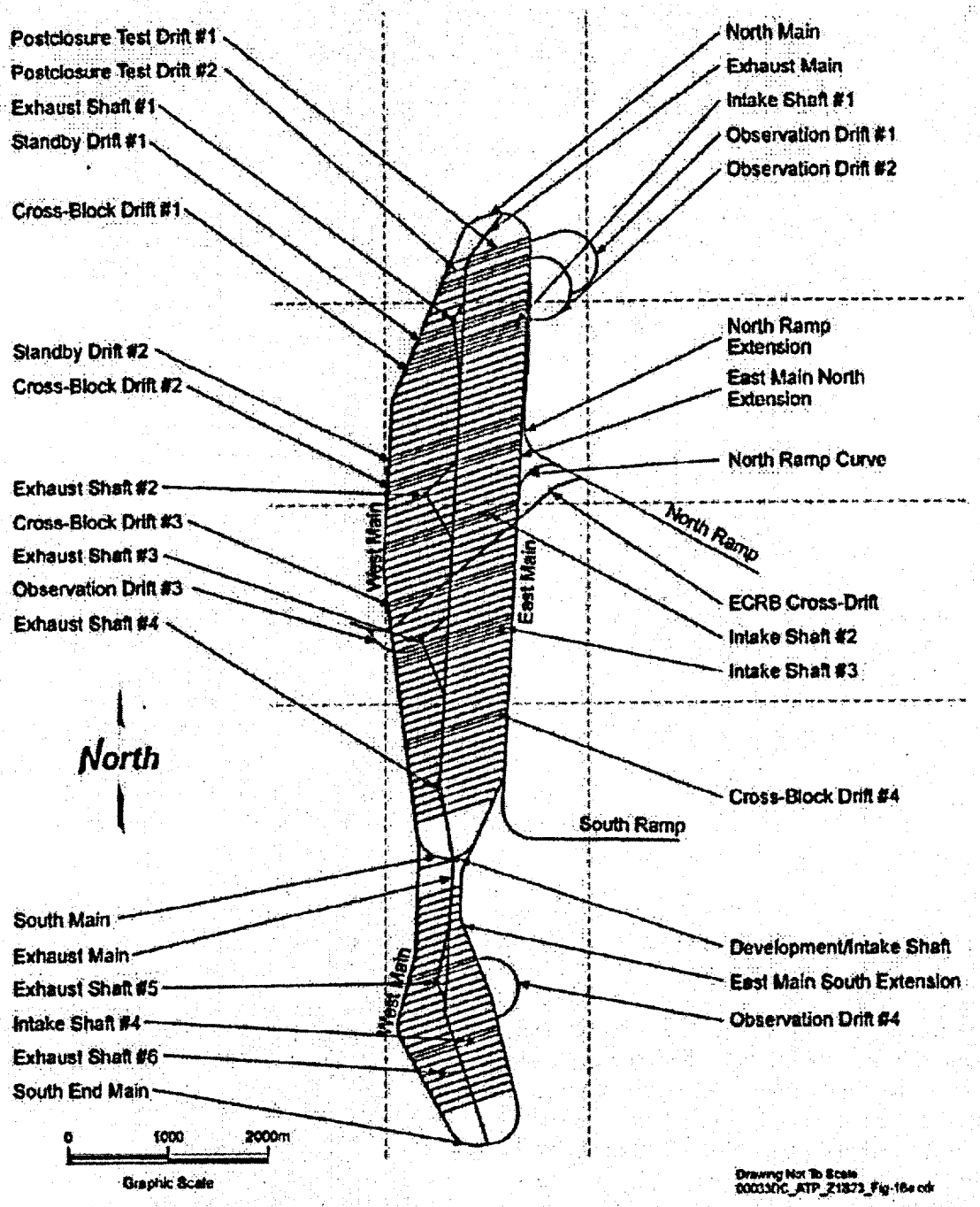


Figure 2-39. Repository Layout for the 97,000-MTHM Case
The subsurface facilities illustrated in this layout represent enough capacity for the storage of 97,000 MTHM of spent nuclear fuel and high-level radioactive waste. This amount is designated in this report as the full inventory case, and it accommodates additional waste resulting from potential extended operation (relicensing) of existing nuclear power plants. Source: BSC 2001d, Section 6.4.

Table 2-10. Safety Classifications for Repository Subsurface Systems

System Designation	System Description	QL Classification
SFS	Subsurface Facility	1
EDS	Emplacement Drift	1
WES	Waste Emplacement/Retrieval	1
GCS	Ground Control	2
BES	Backfill Emplacement	2
PCM	Performance Confirmation Emplacement Drift Monitoring	3
SVS	Subsurface Ventilation	CQ
SED	Subsurface Electrical Distribution	CQ
SCA	Subsurface Compressed Air	CQ
SWD	Subsurface Water Distribution	CQ
MHS	Muck Handling	CQ
SDT	Subsurface Development Transportation	CQ
SCS	Subsurface Closure and Seals	CQ
SWC	Subsurface Water Collection/Removal	CQ
SET	Subsurface Emplacement Transportation	CQ
SES	Subsurface Excavation	CQ
SFR	Subsurface Fire Protection	CQ

Source: YMP 2000b.

- *Waste Emplacement/Retrieval System Description Document* (CRWMS M&O 2000ac)
- *Ground Control System Description Document* (CRWMS M&O 2000ad)
- *Backfill Emplacement System Description Document* (CRWMS M&O 2000ae)
- *Subsurface Ventilation System Description Document* (CRWMS M&O 2000af).

The selected group includes all the QL-1 and QL-2 systems and one CQ system. The subsurface ventilation system, although classified as a CQ system, was included in the selection because of the importance of ventilation to the overall repository design concept. Design descriptions and other information about the remaining 11 systems in Table 2-10 were obtained from a variety of sources. Information on the performance confirmation emplacement drift monitoring system was obtained from the *Performance Confirmation Plan* (CRWMS M&O 2000ag) and from a design analysis, *Instrumentation and Controls for Waste Emplacement* (CRWMS M&O 2000ah). Three separate analysis reports (CRWMS M&O 2000ai; CRWMS M&O 2000aj; CRWMS M&O 2000ak) provided information on the subsurface closure and seals system. Design descriptions provided in subsequent sections explain how the design of subsurface facility components fulfills the functions assigned to them. Functions assigned to other project systems are discussed in Sections 2.2 and 3. The functions of the six principal subsurface systems are summarized below.

Subsurface Facility System—The functions assigned to the subsurface facility system are to:

- Accommodate safe disposal and retrieval of waste packages
- Provide layout, location, and size of underground openings
- Allow personnel and equipment access throughout the repository
- Provide openings for subsurface ventilation
- Provide subsurface refuge areas for subsurface emergencies and exits for emergency evacuation

- Control subsurface flood events and limit subsurface inundation from the surface
- Protect subsurface personnel and equipment from radiation emanating from emplacement drifts
- Locate the repository in an area that would limit radioactive emissions.

Emplacement Drift System—The functions assigned to the emplacement drift system are to:

- Isolate spent nuclear fuel and high-level radioactive waste
- Limit the possibility of a self-sustaining fission reaction in both the near and far fields
- Limit the effect of rockfall on waste packages
- Provide physical support for waste packages
- Influence the environment within the emplacement drift to protect the waste packages and the natural barrier
- Limit the movement of radionuclides to the natural barrier in the event of waste package breach
- Limit microbial activity
- Allow periodic inspection, testing, and maintenance of structures, systems, and components before permanent closure.

Waste Emplacement/Retrieval System—The functions assigned to the waste emplacement/retrieval system are to:

- Receive waste packages at the Waste Handling Building
- Transfer waste packages to the subsurface repository
- Emplace waste packages in their final location within drifts
- Retrieve to the surface some or all emplaced waste packages under normal conditions
- Recover individually selected emplaced waste packages to the surface under normal and off-normal conditions
- Remove rock, ground support, failed equipment, and debris impeding retrieval and recovery operations
- Install ground support to permit safe conduct of retrieval and recovery operations
- Limit or prevent the spread of radioactive contamination
- Support the collection of material control and accounting data
- Operate within the natural and human-induced environmental conditions expected at the surface and subsurface of the repository
- Provide features to minimize radiation exposure to workers
- Provide features and equipment for reducing the risk of, responding to, and recovering from off-normal events and credible design basis events
- Provide features for the inspection, testing, and maintenance of system equipment
- Remediate off-normal events involving the portions of the system supporting waste emplacement, retrieval, and recovery
- Mitigate the effects of a radioactive release in the subsurface repository
- Emplace and recover surrogate (empty) waste packages to support performance confirmation investigations.

Ground Control System—The design functions for the ground control system are intended for the preclosure period; however, they may provide

some benefit during the early part of postclosure. The functions assigned to the ground control system are to:

- Provide structural support for the subsurface repository openings
- Protect waste packages against rockfall, loosening of rock blocks, and fracturing and surface deterioration of the rock mass surrounding each opening
- Maintain adequate subsurface operating envelopes (dimensional clearances)
- Provide for monitoring of ground control performance.

Backfill Emplacement System—The design described in this report does not use backfill in the emplacement drifts; however, there are potential design advantages that warrant maintaining the capability to use it. Design documents include references to emplacement drift backfill and assessments of the effect of backfill on emplacement drift features. Moreover, maintaining this design option provides the flexibility to accommodate future changes, as necessary. Backfill emplacement is required for repository closure operations, which include backfilling ramps, mains, and shafts; however, these closure functions fall under the subsurface closure and seals system. The functions assigned to the backfill emplacement system are to:

- Retrieve and prepare stockpiled material to be used as backfill
- Transport prepared backfill material from the surface to the subsurface
- Install prepared backfill material in the emplacement drifts
- Provide mechanisms and safety features for the protection of operating personnel and equipment.

Subsurface Ventilation System—The functions assigned to the subsurface ventilation system are to:

- Generate and control subsurface airflow
- Maintain air quality standards within established limits
- Contribute to the control of subsurface air temperatures and the rate of air temperature change by removing heat
- Control personnel access to the emplacement drifts
- Monitor the status of system and equipment operation
- Provide operating parameters and air-related environmental data
- Mitigate the spread of a subsurface fire.

2.3.2.2 Containment and Isolation

The subsurface facilities design features contribute to waste containment and radiological shielding during the preclosure period and to the containment and isolation of the waste from the environment during the preclosure and postclosure periods.

The repository design described in this report includes a project design requirement for removal of 70 percent of the heat generated by the waste packages during the preclosure period. Meeting this requirement demands ventilation of the emplacement drifts at an estimated rate of 15 m^3/s (530 ft^3/s) (see Section 2.3.4.3). To accomplish this ventilation rate, the emplacement drift isolation doors would have louvers to permit airflow through the drifts while the doors are closed. Exhaust shafts would draw the drift airflow through the exhaust raise and the exhaust main to the surface. This high ventilation rate would keep emplacement drift wall temperatures below boiling during preclosure and would also contribute to the removal of moisture naturally present in the rock,

as well as water introduced by construction activities.

The continuous preclosure ventilation design contributes to waste isolation by preventing a portion of the rock in the pillars between emplacement drifts from reaching boiling temperatures after closure. Above-boiling temperatures near the emplacement drift walls drive moisture away from the drifts; water will drain through the below-boiling zones of the pillars, rather than into the emplacement drifts. Field tests show that moisture condenses in the pillar area and percolates downward, away from the repository horizon (CRWMS M&O 2000al, Section 3.6.1.3.1). This will reduce the availability of fluid in the matrix of the rock surrounding the emplacement drifts. Lower-temperature operating modes result in lower rock and waste package temperatures. While more water would be available to contact the waste packages, the lower potential for corrosion contributes to containment.

During the preclosure period, engineered subsurface design features (other than movable radiation shields) that can contribute to radiation protection for workers are the emplacement drift isolation doors and the turnouts. The isolation doors, which provide ventilation and access control for the emplacement drift, also contribute to radiation attenuation as well as isolation of potential airborne contaminants. The curved geometry of the turnouts provide protection to workers in the main drifts; direct radiation from the emplacement drifts would be scattered and attenuated by the turnout walls so that only limited radiation reaches the main drifts. After a drift is loaded, the isolation doors would remain closed. The doors cannot be manually operated because of their weight and control mechanisms. Their operation (opening, closing, and louver regulation) would be continuously controlled and monitored by the repository monitoring and control system (see Section 2.3.3). When the doors are open during a waste package transfer operation, the docking area of the turnouts would be under video surveillance by the remote control operators, and personnel access to the area would be denied. Entering the emplacement drifts through a ventilation raise would be difficult, and this mode of entry would not be considered accidental. Other repository design features for prevention of human access during preclosure are described in Section 2.2. The repository preclosure safety test and evaluation program (see Section 5.4) describes the monitoring, testing, and evaluations of repository structures, systems, and components. This program would help ensure safe operation of waste emplacement and other repository functions.

Corrosion-resistant materials have been chosen for waste package supports to aid in maintaining the integrity and stability of waste packages during preclosure and postclosure (see Sections 2.3.4.4.2 and 2.4.3.1).

The emplacement drift engineered barriers (see Section 2.4) and the structures designed for repository closure and sealing (see Section 2.3.4.8) are the two subsurface design features that contribute to waste package containment and isolation during the postclosure period.

2.3.2.3 Thermal Management

The higher-temperature operating mode described in this report ensures that there will be portions of the pillars with temperatures below the boiling point of water (96°C [205°F] at the elevation of the emplacement horizon). This will allow the mobilized water to drain through the pillars, rather than into the emplacement drifts, when the temperature of the adjacent rock is above boiling (see Section 4.2.2). The development of localized boiling fronts around each emplacement drift, rather than a single boiling front encompassing all of the emplacement drifts, ensures that a much smaller amount of water is allowed to build up above any emplacement drift. This substantially decreases the likelihood of any liquid water entering the emplacement drifts. These evaluations resulted in new thermal management criteria for the operating mode analyses presented in this report that seek to keep boiling temperatures from spreading all the way through the rock between drifts during postclosure. Those criteria also introduced the concept of line-loading the waste packages, that is, placing the packages with minimal end-to-end separations to achieve a more uniform thermal profile along the length of the emplacement drifts. The concentration of heat

sources resulting from line-loading required that the emplacement drifts be spaced farther apart to distribute the thermal load over a larger area. Operating modes that result in lower temperatures on the waste package surface and in the rock mass are also relevant to the DOE's thermal management strategies. As discussed in Section 2.1, recent evaluations have investigated performance-related attributes of operating modes of the design described in this report that result in lower waste package surface temperatures, as well as lower rock temperatures. In particular, below-boiling conditions associated with lower-temperature operating conditions could reduce uncertainties in the understanding of corrosion processes for the waste packages.

Given a projected thermal-loading schedule from waste package emplacement, projected nuclear fuel decay, and various rates of ventilation, computer simulations helped define ventilation design concepts that could satisfy the performance objectives related to a range of operating modes for a lower-temperature repository.

2.3.3 Concept of Operations and Maintenance

Current planning calls for the potential repository to be developed and operated in six major phases: (1) site characterization, (2) construction, (3) operations, (4) monitoring, (5) closure, and (6) postclosure (CRWMS M&O 1999g, Section 3.1). This section focuses on two of these phases: construction and operations.

All waste handling activities would be conducted pursuant to approved operations and maintenance procedures to assure safe operations. The project would also have a site-wide centralized instrumentation and controls system to assist in subsurface operations.

2.3.3.1 Subsurface Facilities Construction

The subsurface repository facilities would have concurrent construction and waste emplacement activities; as described in Section 2.3.5, some differentiation is made between initial construction and construction during emplacement. The construction activities during the first 5 years are called initial construction, and the construction activities that are simultaneous with emplacement are called development..

Examples of the major activities that would take place during the initial construction phase are:

- Transitioning the existing Exploratory Studies Facility to repository facilities
- Refurbishing Exploratory Studies Facility drifts
- Constructing access shafts and ventilation systems
- Excavating at least one exhaust shaft and one intake shaft
- Excavating and finishing portions of the main drifts and exhaust main
- Installing the muck handling system
- Installing utilities, controls, and communication systems
- Constructing the initial set of isolation air locks
- Excavating performance confirmation facilities and installing testing and monitoring equipment.

Development activities would be scheduled so panels of emplacement drifts are completed and commissioned for emplacement at different times throughout the development phase (see Section 2.3.5). These activities would include such work as:

- Continuing excavation, finishing, and equipping of the subsurface facilities, including main and exhaust drift extensions, emplacement drifts, performance confirmation drifts, intake and exhaust shafts, access drifts, and ventilation raises

- Constructing additional air locks and installing additional ventilation equipment
- Commissioning panels of emplacement drifts
- Installing additional monitoring, controls, and communications systems.

In addition to the construction and development activities described above, major subsurface system operations would be performed, including:

- Ground control
- Ventilation
- Waste emplacement.

These major systems would be supported by other subsurface systems (e.g., electrical distribution, compressed air, water distribution, and fire protection) on a regular basis. Routine inspections and maintenance operations activities would be performed as required for each system.

The *Monitored Geologic Repository Concept of Operations* (CRWMS M&O 1999g), as well as applicable NRC nuclear safety requirements, national safety requirements, and DOE Orders (as applicable), would provide the basic guidelines and requirements for operations and maintenance of the subsurface systems.

2.3.3.2 Operations Support Facilities

The repository subsurface facilities would include (1) access facilities, (2) the subsurface Geologic Repository Operations Area, and (3) support areas.

The main access facilities are the north and south ramps, with the development/intake shaft providing additional access throughout the development phase. The north ramp would be the access route for waste emplacement operations; however, during initial construction, it would also be used as a major ventilation air intake. The south ramp would provide primary access for hauling mined rock (muck), personnel, equipment, and construction materials to support repository development operations. The south ramp would also serve as a main ventilation intake and exhaust airway throughout the development phase. The development/intake shaft would be available during the development phase for personnel access, muck removal, and introduction of equipment and construction materials, as needed.

The subsurface Geologic Repository Operations Area would consist of all the completed and commissioned subsurface facilities supporting waste emplacement, containment, and isolation. These include the emplacement drift panels, mains, performance confirmation drifts and associated facilities, and subsurface ventilation facilities. This operations area would grow until it reached fully developed dimensions about 25 years after initial construction.

The support facilities for the subsurface operations would include support systems at the surface facilities area, such as:

- Equipment maintenance and repair
- Personnel access control
- Decontamination facilities
- Control and communications centers
- Performance confirmation data processing center
- Security and emergency response systems.

The support facilities at the South Portal would include staging areas for construction equipment and materials, areas for equipment assemblage, muck stockpiles, and construction personnel access and control facilities.

2.3.3.3 Operational Phase

The subsurface operational phase would include the following major activities:

- Transport and emplacement of loaded waste packages
- Performance confirmation
- Completion of drift excavation
- Placement of engineered barriers
- Monitoring and maintenance of structures, systems, and components.

Physical and functional separation of development and emplacement operations would be maintained throughout the operational phase by using engi-

neering controls (e.g., isolation air locks) and administrative controls (e.g., separate access controls). When a predetermined number of newly excavated drifts are ready for waste emplacement, the controls would be moved to include the newly commissioned drifts in the emplacement area.

Waste emplacement operations would include transfer of the waste packages underground, emplacement drift operations, and waste package management. Sections 2.3.4.4 and 2.3.4.5 describe the first two operations. Waste package management operations would include such activities as:

- Monitoring the integrity of the waste packages after emplacement
- Thermal management of the repository block
- Monitoring drift conditions that may affect waste package integrity and implementing mitigation measures
- Periodic removal or relocation of waste packages for testing or mitigation measures, as needed.

Performance confirmation operations include maintaining facilities, equipment, instrumentation, and data communication systems, and constructing new facilities, such as alcoves and boreholes, in support of performance confirmation monitoring. Section 2.5 describes performance confirmation subsurface facilities, and Section 4.6 explains the performance confirmation program.

Operations related to completion of emplacement drifts include calibration of ventilation flow regulators, installation of shielding and local ventilation controls (e.g., turnout air locks), and operational testing of isolation doors.

Section 2.4 discusses the in-drift engineered barriers. The general operational activity in this area is the preparation of the emplacement drifts for installation of the drip shields over the waste packages. Support operations related to this activity would include staging and transportation of the drip shields to the subsurface. Drip shields would be installed during the closure phase.

The repository would be monitored and maintained continually during the emplacement and monitoring phases. Permanently installed sensors would monitor waste packages, drifts, and the surrounding rock from accessible locations. These sensors would provide the data required for performance confirmation activities and for day-to-day monitoring of operational and environmental conditions. System tests and evaluations would also be conducted to ensure functionality and confirm expected performance for the various subsurface systems. Section 5.4 explains the repository test and evaluation program. The following section discusses the repository maintenance approach.

2.3.3.4 Maintenance

To the extent practical, maintenance of equipment and facilities would be conducted by removing and replacing faulty or suspect components. Faulty components would be returned to the shop facilities for repair or for processing for offsite repair or salvaging. Maintenance for nuclear operations requires a specialized workforce certified in the operation of remote handling equipment, containment equipment, and hardware decontamination. A formal maintenance and training program would be implemented, and specialized maintenance procedures would be developed.

In addition to stationary and mobile equipment, the subsurface facilities would have a wide range of monitoring and control systems. To the extent practical, these systems would be designed in a modular fashion. This modular design would allow maintenance personnel to unplug failing units, such as radiation or thermal sensors, programmable logic controllers, input/output circuit boards, and video cameras, and quickly replace them with functioning units. The emplacement drift design philosophy is to minimize in-drift maintenance after the first waste package has been emplaced.

To minimize downtime, a comprehensive scheduled maintenance program would be implemented to ensure that appropriate systems and equipment are given adequate preventive maintenance. Periodic, planned maintenance would preserve the

integrity of the shafts, ramps, and main and access drifts.

Maintenance in response to major system malfunctions would be supported by the surface maintenance facilities at both the North and South Portals. A staff of mechanics, electricians, and supervisors would respond to and repair unplanned equipment outages. Equipment that can be moved would be taken to the surface maintenance facility for evaluation and repair. Some equipment, such as large excavation machinery (e.g., roadheaders or tunnel boring machines), would most likely be repaired in place.

An underground maintenance shop may be added to facilitate repairs and support ongoing drilling or boring operations (CRWMS M&O 1999g, Section 4.6.4).

2.3.4 Design Descriptions and Systems Operations

The following sections describe the design characteristics and concept of operations for the most important subsurface systems of a potential repository at Yucca Mountain.

2.3.4.1 Developing and Maintaining Stable Excavations

2.3.4.1.1 Underground Openings and Excavation Methods

The potential repository excavations were divided into two groups for ground control design analyses: emplacement drifts and nonemplacement areas. The reason for this differentiation is that the emplacement drifts are exposed to higher temperatures than nonemplacement openings (CRWMS M&O 2000w, Sections 5.2 and 6.3.3).

The emplacement drifts would be 5.5-m (18-ft) diameter drifts excavated by a tunnel boring machine, which is a mechanical excavator equipped with a rotating cutting head that chips the rock with hardened disc cutters. Temporary ground control would be installed immediately behind the excavator to provide structural support and worker protection (CRWMS M&O 2000w, Section 6.2.2).

The repository would have several types of nonemplacement excavations (BSC 2001d, Section 6.2):

- Main drifts: The main drifts are large circular tunnels with a diameter of 7.62 m (25 ft), excavated with tunnel boring machines. Examples of main drifts are the existing north ramp and the repository perimeter mains.

- Exhaust main: The exhaust main, also to be excavated with a tunnel boring machine, has a diameter of 7.62 m (25 ft).

- Turnouts: The turnouts are curved sections of tunnel connecting the emplacement drifts to the main drifts. They would be excavated with roadheaders, which are mechanical excavators equipped with a rotating grinding drum mounted at the end of an excavating arm. Drill-and-blast techniques are an alternative option for excavating the turnouts. The turnouts have vertical walls and an arched ceiling. The excavated turnouts are 8 m (26.2 ft) wide, with the ceiling arch reaching a maximum height of 7 m (23 ft) (BSC 2001d, Section 6.2.1.2).

- Performance confirmation test drifts: The performance confirmation test drifts are the same as the emplacement drifts in configuration and construction.

- Performance confirmation observation drifts: These drifts are similar in cross-sectional area to the emplacement drifts and provide access to the monitoring instrumentation stations and alcoves. The observation drifts may be constructed by a combination of tunnel boring machines and roadheaders or by conventional drill-and-blast techniques.

- Alcoves: The alcoves are areas of limited length that would be excavated with roadheaders or by conventional drill-and-blast techniques. Alcoves would serve different purposes, such as refuge chambers and equipment or instrumentation chambers. Mechanical excavation techniques would be used in areas where rock properties being

monitored might be adversely affected by blasting.

- Ventilation shafts and raises: These vertical openings, typically circular, would be excavated by mechanical means (e.g., vertical mole, drilling, and raise boring) or by drill-and-blast techniques. The repository ventilation shafts are large, 8 m (26.2 ft) in diameter. The raises are much smaller, with a diameter of 2 m (6.6 ft). The use of drill-and-blast techniques may be limited near the emplacement horizon because of the potential for disturbing the natural rock conditions.

The Exploratory Studies Facility is a drift 7.62 m (25 ft) in diameter that was excavated with a tunnel boring machine. This facility would become the north ramp–north ramp curve–east main–south ramp loop in the potential repository (see Figure 2-38). The Exploratory Studies Facility uses several types of ground control supports (CRWMS M&O 2000am), including:

- Shotcrete
- Welded wire mesh with rock bolts
- Steel sets with welded wire mesh, steel lagging, or rock bolts, or without additional support.

The Exploratory Studies Facility ground support would be upgraded to cast-in-place concrete lining during the initial construction phase.

Other existing excavations in the repository area significant to ground control are the Enhanced Characterization of the Repository Block Cross-Drift (Figure 2-38) and the Drift Scale Test. The ECRB Cross-Drift is a drift 5 m (16.4 ft) in diameter, excavated with a tunnel boring machine. The drift is equipped with ground control supports similar to those installed in the Exploratory Studies Facility drifts (CRWMS M&O 1998b, Section 7). The ECRB Cross-Drift traverses the repository horizon in a southwesterly direction and intercepts a major fault zone outside the potential repository emplacement area, providing opportunities to characterize the rock environment at the repository level and in areas of major geological disturbance. Using the ECRB Cross-Drift as a performance confirmation facility is also being proposed.

The Drift Scale Test is located off the north ramp in an alcove loop (CRWMS M&O 1998c). It is a prototype emplacement drift equipped with high-powered electric heater canisters that simulate waste packages. The rock environment and its response to the heat load is being meticulously monitored and documented as part of the characterization phase of the repository program. A portion of the Drift Scale Test facility uses a cast-in-place concrete ground support system. The remainder of the facility uses rock bolts and wire mesh for ground support.

This diversity of excavations in the potential repository area has given engineers firsthand experience with the different levels of support needed and with the effectiveness and reliability of different ground control techniques. This knowledge base is supplemented with rock mechanics information collected from the Drift Scale Test, where the repository rock formation is subjected to thermal stresses equal to or greater than those expected from the thermal load of a higher-temperature operating mode (SNL 1998).

2.3.4.1.2 Proposed Ground Control Structures

Ground control structures are required to ensure that underground openings are stable. The design and construction of an underground high-level radioactive waste repository introduce challenges not ordinarily found at other subsurface facilities. The presence of heat from the spent nuclear fuel and the resulting thermal-mechanical stresses introduce a series of requirements for the overall design and construction of the facility to conform to project performance objectives. In situ loads, construction loads, potential loads from repository operations, and loads from seismic occurrences must also be addressed in the design.

2.3.4.1.2.1 Emplacement Drifts

The initial ground support system for the emplacement drifts also serves as the final support system;

it consists of steel sets with welded-wire fabric and fully grouted rock bolts. Figure 2-40 shows a cross section of a typical emplacement drift with the proposed ground support and a perspective view of the drift with the steel sets and welded-wire fabric detail. This ground support system minimizes routine maintenance, which is not practical once the emplacement drifts are loaded. Project design requirements (CRWMS M&O 2000ad, Section 1.2.2.2.2) state that the ground control system shall also include provisions that support a deferral of closure for up to 300 years. This design satisfies a project design requirement for a service life of not less than 175 years (CRWMS M&O 2000ad, Section 1.2.2.2.4). The steel sets and rock bolt systems are well suited for quick installation with the tunnel boring machine excavation, thereby providing personnel safety during construction. These systems also allow geologic mapping activities in support of the performance confirmation program, since the rock wall remains mostly visible through the support structures.

An all-steel ground support system for emplacement drift support is installed in a single-pass operation. This system consists of W6 × 20 rolled steel ring beams, called steel sets, bolted together to form a full circle, along with welded-wire fabric. Following immediately behind the tunnel boring machine, steel set sections would be bolted together around the drift circumference between the temporary invert sections that support the tunnel boring machine rail. The steel sets would be set in place at 1.5-m (5-ft) intervals along the length of the drift, and tie rods would be inserted between the steel sets. A partial shield, extending from the rear of the tunnel boring machine, would protect the steel set assembly area from rockfall. The welded-wire fabric would be installed behind the shield between the steel set and the rock, with steel pins holding it against the exposed rock to prevent movement of the rock blocks into the drift. Using hydraulic jacks, the steel set would be expanded against the fabric as the tunnel boring machine moves forward (CRWMS M&O 2000w, Sections 6.2.1.1 and 6.5).

A rock bolt system consists of a pattern of steel rock bolts installed through the welded-wire fabric and grouted with cementitious material to hold them in place. The material used for grouting the rock bolts does not present a problem for long-term

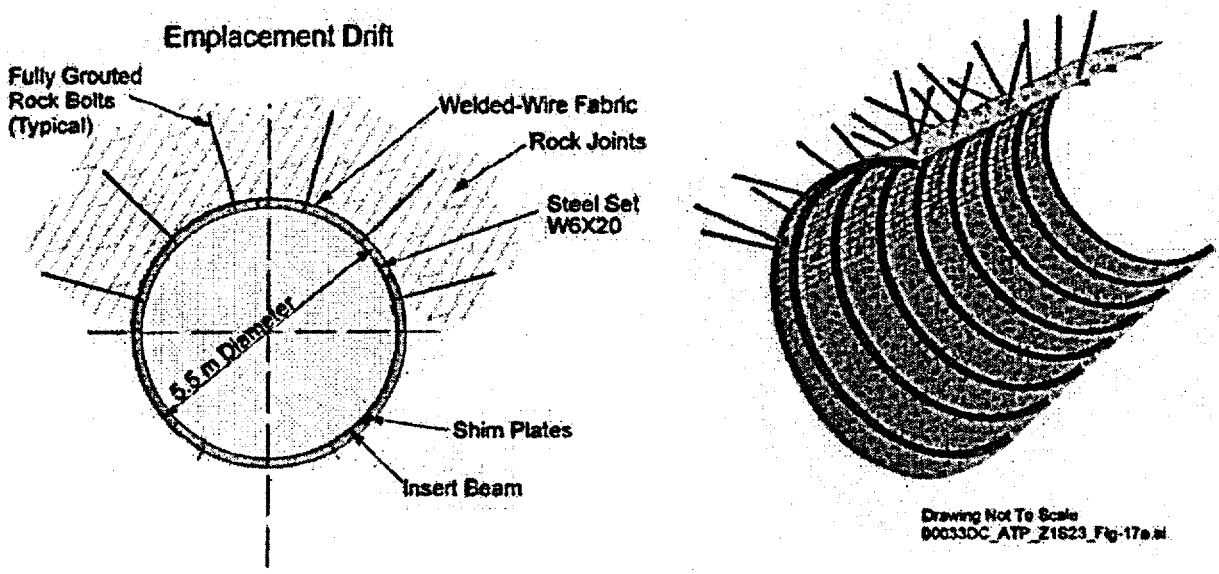


Figure 2-40. Emplacement Drift Ground Support
Ground support systems for the emplacement drifts consist of a combination of steel sets, rock bolts, and welded-wire fabric. The steel sets would be bolted together to form individual "rings" spaced at 1.5-m centers along the length of the drift. Rock bolts and fabric would be installed in the upper half of the drift. Density of rock bolt installations would vary with degree of fracturing in the rock. The wire fabric would prevent loose blocks of rock from becoming detached from the drift wall. Source: CRWMS M&O 2000w, Section 6.

waste isolation (e.g., the alkalinity issue related to elimination of precast concrete liners in emplacement drifts, as discussed in Section 2.1.2.1) because (1) the amount used for grouting is relatively small compared to the total volume of concrete that it would take to line the emplacement drifts and (2) the high-strength cementitious grout is likely to remain intact and attached to the rock even if the rock collapses. A typical rock bolt for this application is 3 m (10 ft) long. Rock bolts prevent key blocks from loosening. In areas with massively jointed rock, or where key blocks are mobilized by excavation and can fall, rock bolts would be installed in a radial pattern above the drift springline (the level of maximum horizontal width of the drift). These conditions are anticipated primarily in nonlithophysal rock areas, which comprise about 30 percent of the emplacement area (CRWMS M&O 2000w, Sections 6.2 and 6.5).

2.3.4.1.2.2 Nonemplacement Excavations

The main drifts, turnouts, exhaust main, and ventilation shafts will have initial and final ground support systems developed. Initial ground control methods will vary depending on ground conditions and would include a combination of steel sets, welded-wire fabric, rock bolts, and shotcrete (concrete sprayed onto the surface at high pressure). Steel sets would be needed to control more difficult, localized geologic conditions. The prevalent initial ground control is expected to be a combination of welded-wire fabric and rock bolts. Occasional use of shotcrete is expected to stabilize areas with extensive fracturing. The final ground support system for these nonemplacement excavation areas would be cast-in-place concrete liners with a nominal thickness of 0.3 m (1 ft). Figure 2-41 shows typical cross sections for a main drift and a turnout (CRWMS M&O 2000w, Sections 6.1, 6.2, and 6.6).

The observation drifts, which support the performance confirmation program, would have a ground support system similar to that for the emplacement drifts if they are excavated with a tunnel boring machine. Otherwise, they would have a combination of support systems, including steel sets, welded-wire fabric, rock bolts, and shotcrete, depending on ground conditions (CRWMS M&O 2000w, Section 6.2.2).

2.3.4.1.3 Determination of Ground Control Design Loads

In designing the repository excavations and their ground support structures, stresses resulting from four sources were considered: (1) in situ stresses, including excavation effects; (2) stresses from construction and operation activities; (3) thermal stresses from the waste packages; and (4) seismic stresses. In situ stresses are present before drift

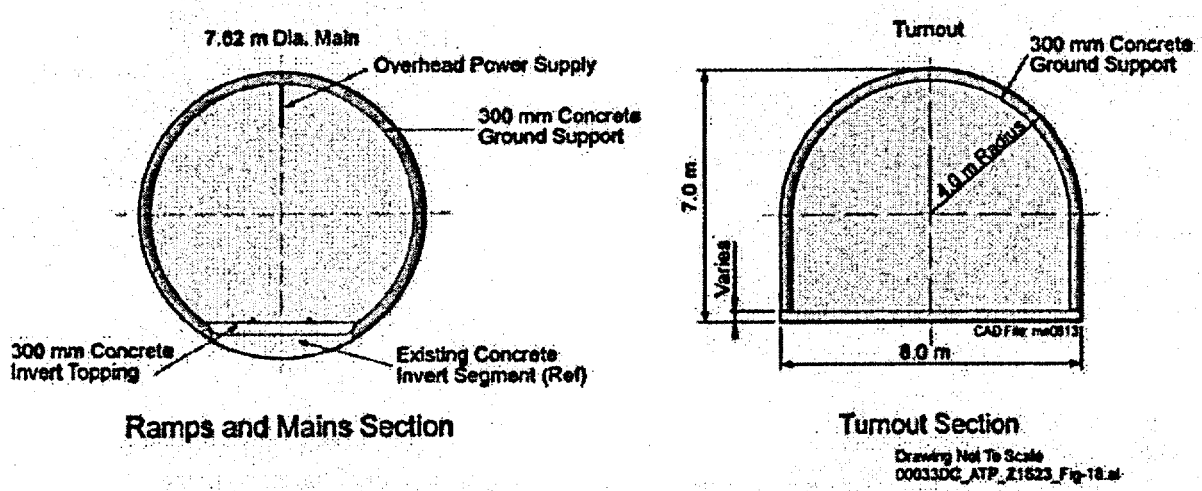


Figure 2-41. Typical Final Ground Support System for Nonemplacement Excavations
Nonemplacement drifts and turnouts use a cast-in-place concrete liner, approximately 300 mm (12 in.) thick, as the final ground support system. Source: CRWMS M&O 2000w, Section 6; BSC 2001d, Section 6.2.

excavation and will be altered during repository excavation. The stresses during construction, including stresses due to installation activities (e.g., jacking process) and equipment movement (e.g., the weight of the tunnel boring machine during excavation), are considered in the design of the ground support systems. The stresses due to repository operations, such as loads caused by gantry or waste package weight, are also considered in the design. Thermal stresses will occur after waste emplacement, and the magnitude of these temperature-induced loads in the rock will depend on the location of the rock relative to the emplaced waste packages, the repository thermal loading, and the time that has passed since emplacement. The magnitude and duration of earthquake-induced stresses are a function of the intensity and duration of the earthquake, the distance of the event from the repository, and the direction and size of the seismic wave relative to the opening (CRWMS M&O 2000w, Section 6.3).

Different techniques were used to estimate stresses and design loads. Analytical techniques used and results obtained are documented in a design analysis report, *Ground Control for Emplacement Drifts for SR* (CRWMS M&O 2000w, Sections 6.3 through 6.6).

Table 2-11 describes the relationship of the lithostratigraphic units in the Topopah Spring Tuff to the thermal-mechanical units within the potential repository emplacement horizon. Table 2-12 summarizes the ground control component preliminary specifications for the subsurface excavations.

Table 2-11. Lithostratigraphic Units and Relationship to Thermal-Mechanical Units of the Topopah Spring Tuff Within the Repository Emplacement Horizon

Formation	Thermal-Mechanical Units	Lithostratigraphic Units
Topopah Spring Tuff	TSw1	Lower part of upper lithophysal zone
	TSw2	Middle nonlithophysal zone
		Lower lithophysal zone
		Lower nonlithophysal zone

Source: CRWMS M&O 1998d, Figure C-2.

Table 2-12. Subsurface Ground Control Components

Excavations	Ground Control Components
Access main drifts, exhaust main drift, turnouts, ventilation shafts	Initial ground control—shotcrete, rock bolts, welded-wire fabric Final ground control—300-mm (12-in.) thick cast-in-place concrete liner
Emplacement drifts	152-mm (6-in.) wide flange beam steel sets, grouted rock bolts, welded-wire fabric
Other subsurface excavations	Various combinations of steel sets, welded-wire fabric, rock bolts, and shotcrete as needed

The emplacement drifts would be essentially contained within the TSw2 unit. This unit includes both lithophysal and nonlithophysal rock, sloping slightly, so that 70 percent of the emplacement drifts would be in the former and 30 percent in the latter. Because the nonlithophysal rock is more jointed and has lower strength and stiffness properties, the emplacement drift ground support design focused on this type of rock. This results in a conservative design. Analyses were also performed on the lithophysal rock for sensitivity effects and completeness. Some initial ground supports for the nonemplacement drifts, such as the east main, were installed during Exploratory Studies Facility construction (CRWMS M&O 2000am). Knowledge acquired and lessons learned during construction of the Exploratory Studies Facility and the ECRB Cross-Drift (CRWMS M&O 1998b) were used in identifying effective initial ground support systems for the proposed main drift extensions.

Results of the analyses for unsupported excavations indicate that the proposed drifts will be inherently stable for the design loads analyzed, including the seismic load. The thermal loads induce minor deformations to which the ground support systems must adapt. Analyses showed that the selected ground support structures for the emplacement drifts could safely handle these anticipated deformations (CRWMS M&O 2000w, Section 7).

Analysis results for the nonemplacement drifts concluded that the maximum combined stress in the concrete lining will be about 50 percent of the

concrete allowable strength. Therefore, the cast-in-place concrete lining would be suitable as the final ground support system for the nonemplacement drifts (CRWMS M&O 2000w, Section 7).

2.3.4.2 Maintaining a Safe Working Environment

The subsurface facilities will be maintained as a safe working environment managed under the *Integrated Safety Management Plan* (DOE 2000c). The facilities have been designed with worker safety as a priority in all phases of development, operations, and closure. The applicable nuclear safety requirements of the Code of Federal Regulations and DOE Orders will be followed. The repository underground environment will be continuously monitored for key environmental quality parameters to alert workers of potential hazards, and dosimetry records for individual workers will be monitored to ensure compliance with radiation dose limits. The facilities would be provided with refuge chambers, evacuation facilities, and emergency response equipment and capabilities. Subsurface communication systems would be available throughout the subsurface facilities for communicating with surface control centers and for broadcasting hazard warnings or emergency communications.

The major potential environmental hazards at the repository, besides waste package radiation, are expected to be: (1) heat; (2) dust from construction activities; (3) dust from other activities containing harmful contaminants or materials, such as silica dust or airborne radioactive particulates; (4) radon progeny (radon gas and its radioactive decay daughter products); (5) fire; and (6) rockfall. Engineering controls are incorporated into the repository design of features and components where these hazards may be encountered. Ventilation and ground control are key subsurface facilities systems where these engineering controls are incorporated into the design.

These engineering controls will be supplemented with frequent monitoring of repository conditions, an aggressive maintenance program and, as needed, administrative controls.

2.3.4.2.1 Ventilation System Monitoring and Operational Safety Requirements

2.3.4.2.1.1 Design Requirements

Ventilation system analyses (CRWMS M&O 2000x, Section 4.2; CRWMS M&O 2000an, Section 4.2) identify criteria for limits on subsurface air temperature, prevention of cross-contamination, radon concentrations, and safety systems.

Temperature Limits—The ventilation system would be designed to limit the maximum dry bulb temperature to 48°C (118°F) in subsurface areas requiring human access (CRWMS M&O 2000af, Section 1.2.1.3). For subsurface areas requiring human access for a full shift (i.e., eight hours or more) without personnel heat stress protection, the ventilation system shall limit the maximum effective temperature to 25°C (77°F) (CRWMS M&O 2000af, Section 1.2.1.3).

Temperature limits would also be imposed on the ventilation system for protection of instrumentation, monitoring equipment, and remote access equipment. For those areas and activities where remote access is required (i.e., activities in emplacement drifts like emplacement, retrieval, recovery, and off-normal modes of operation), the dry bulb temperature must be maintained at or below 50°C (122°F).

Prevention of Cross-Contamination—Engineering controls for prevention of cross-contamination of air supplies are imposed by the requirement that the ventilation systems for the development areas and the emplacement areas be separate and independent systems, with the emplacement system operating under negative pressure and the development system operating under positive pressure. A separate requirement states that the emplacement drift ventilation system be designed to prevent reverse airflow (i.e., airflow from the emplacement drifts into the turnouts).

Radon—The ventilation system must control concentrations of radon progeny in potentially occupied areas to levels that will not result in worker exposure exceeding regulatory limits. The

ventilation system has been designed to satisfy Section 14.5 of ANSI N13.8-1973, *American National Standard Radiation Protection in Uranium Mines*, to control concentrations of radon progeny in potentially occupied areas of the repository.

Preventing the Spread of Surface and Airborne Contamination—The ventilation system will be designed to ensure that occupational doses are ALARA, in accordance with repository program goals, applicable NRC regulations, and NRC Regulatory Guide 8.8.

Safety Systems—The ventilation system has to provide for the following alarm status indicators, as a minimum, from the subsurface safety and monitoring systems: (1) subsurface fire detection, (2) subsurface radiological conditions, and (3) subsurface air quality.

2.3.4.2.1.2 Operational Safety Requirements

The emplacement area ventilation system would be separated from the construction system with isolation barriers, or air locks. The use of these air lock systems is discussed in Section 2.3.5. Operating the emplacement surface fans in an exhaust mode would prevent airflow reversal from the emplacement side to the construction side. Section 2.3.4.3 describes how the surface fans draw air through the exhaust mains at a rate that ensures that air always circulates into emplacement drifts from the main drifts and never allows air to recirculate back to the main drifts (CRWMS M&O 2000x, Section 6.1). Section 2.3.4.3 also discusses the regulation of air temperatures in the emplacement drifts.

Human access to the exhaust main is made possible by dividing the main in halves along its length with an airtight partition. For the operating mode described in this report, rock temperatures around the observation drifts have been estimated to peak at about 46°C (115°F) during the preclosure period. Therefore, during the preclosure period the observation drifts would be exposed to temperatures within the design conditions for human access, even without accounting for any cooling from airflow. Airflow in the observation drifts will vary and can be adjusted to maintain temperatures as needed for human comfort during the preclosure period. The observation drifts and emplacement drifts will be under the same negative pressure system. The airflow in each drift would be controlled by regulators at the drift entrances or the exhaust raise outlets. The intake air for each drift is supplied via the east or west mains. Airflow and pressures would be monitored electronically to ensure that proper airflow directions and quantities are maintained. Maintaining the airflow direction from the mains into the drifts will prevent airborne radiological contamination from migrating into the mains and observation drifts. The air exiting these drifts would be carried in an air stream, separate from the emplacement air stream, into one half of the exhaust main, designated the "service side." That half of the main would be considered a normal workplace. If maintenance work is needed in the emplacement exhaust side of the exhaust main, temporary openings through the partition could redirect cooler air to the warmer air stream. The exhaust shafts would not be partitioned, so if maintenance or repair work is required in those areas, the emplacement air volume would be temporarily reduced so personnel can work in the shafts (CRWMS M&O 2000x, Section 6.1).

Section 2.3.5 discusses ventilation techniques used during construction and dust control techniques used in the active excavation fronts. The ventilation analysis discusses dust control techniques in use at the site. Those dust control guidelines are expected to be followed during the repository construction and operation phases (CRWMS M&O 2000x, Section 6.4.1).

Radon gas emissions are common in any excavation in igneous rock. Radon gas is liberated in proportion to the exposed area of rock, but dilution can control it. Using blowers in development headings (at the excavation front), filtering the air, and spraying lining materials on the rock surface could also control radon exposure. However, use of spraying foam may not be acceptable at the repository because of the prohibition against the use of organic materials in the subsurface environment. Ongoing characterization work at the Exploratory Studies Facility will determine radon emission rates; once these have been determined, airflow volumes required for dilution can be estimated and

ventilation rates established to limit radon exposures (CRWMS M&O 2000x, Section 6.4.2).

2.3.4.2.1.3 Ventilation and Radiation Monitoring

The subsurface ventilation monitoring system is one of the components of the repository operations monitoring and control system. Figures 2-42 and 2-43 show conceptual locations and monitoring parameters for the ventilation system components (CRWMS M&O 2000x, Section 6.5).

2.3.4.2.2 Fire Protection

A subsurface fire hazards analysis reported in the subsurface ventilation analysis (CRWMS M&O 2000x, Section 6.6.2) concluded that the potential for a toxic, biological, or radiological incident due to a fire is not considered significant for the following reasons:

- Most materials selected for construction are essentially inert.

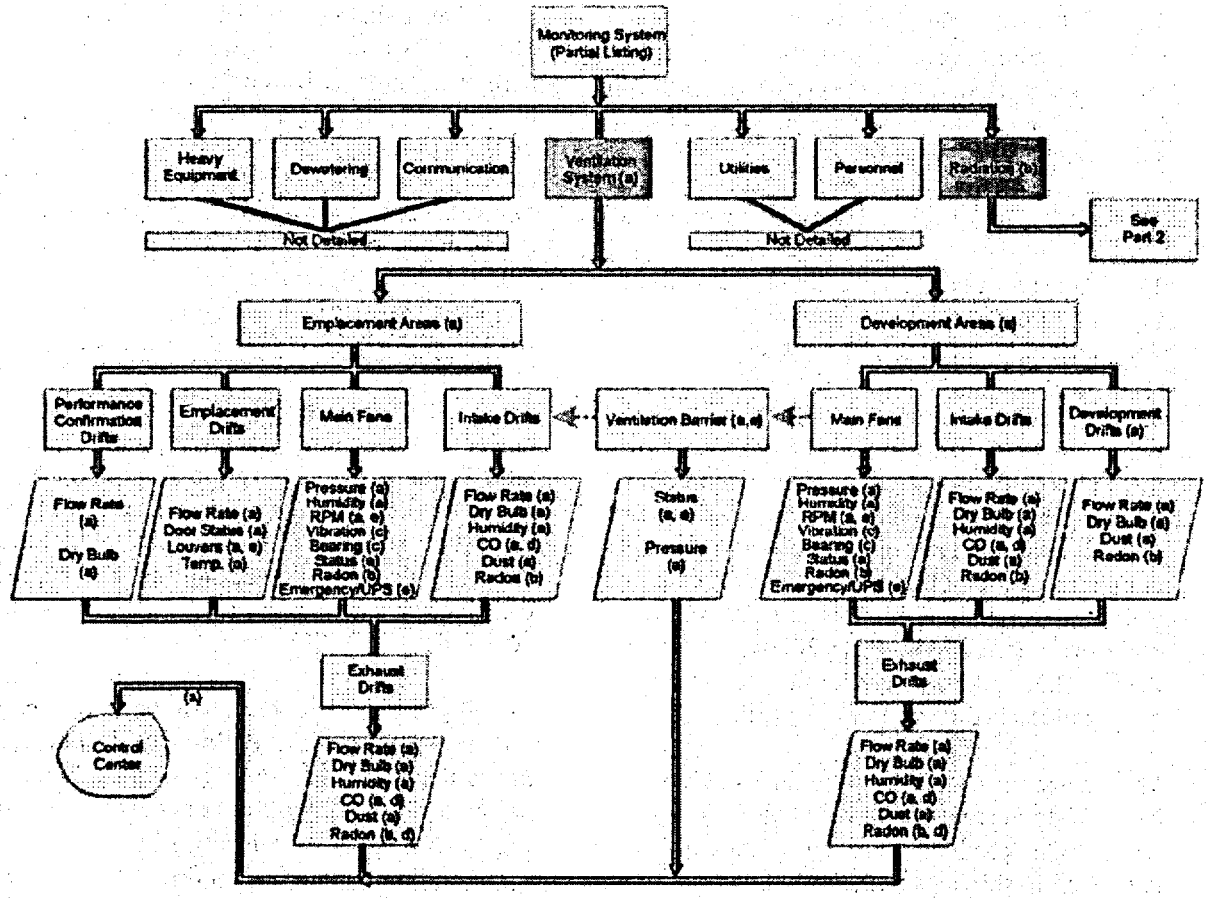


Figure 2-42. Ventilation and Radiation Monitoring Conceptual Diagram, Part 1
(a) Monitoring operations interface/parameters; (b) Site radiological interface; (c) Motor parameters; (d) Alarm status for fire, radiation, and air quality; (e) Operator control of fans, dampers, and doors. The ventilation system would be continuously monitored for critical component function and performance and for key parameters of airflow at many locations throughout the repository. Only ventilation and monitoring system parameters are sketched; the monitoring system list is partial. Nonradioactive particulate is referred to as "dust" in these diagrams. RPM = revolutions per minute; UPS = uninterruptible power supply. Source: CRWMS M&O 2000x, Section 6.5.

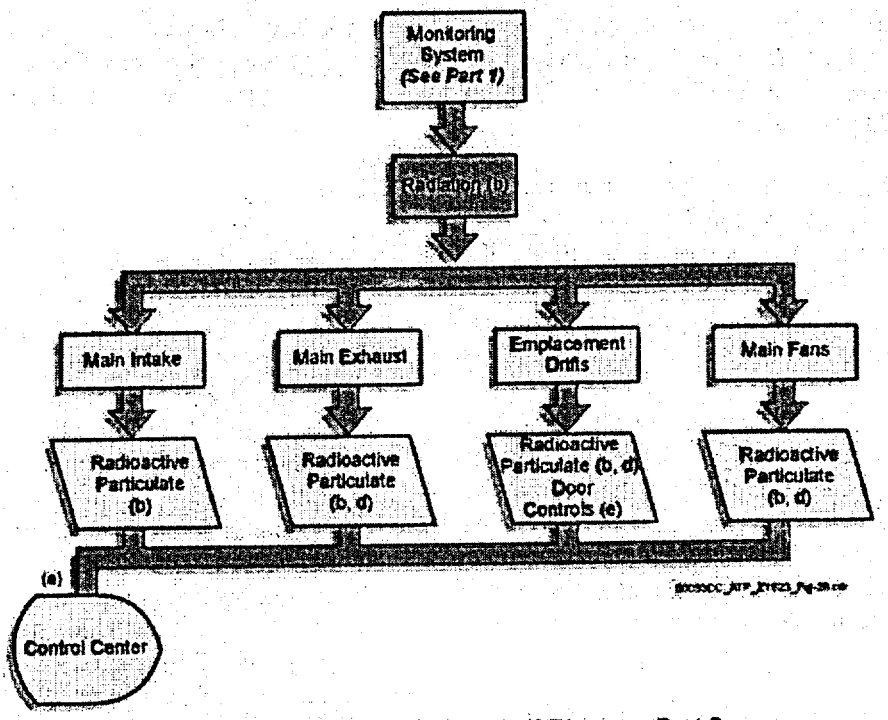


Figure 2-43. Ventilation and Radiation Monitoring Conceptual Diagram, Part 2
Notes (a) through (e) are given in Part 1 of the conceptual diagram (Figure 2-42). Monitoring of repository ventilation includes measurements of airborne radioactivity at all intake and exhaust points. Nonradioactive particulate is referred to as "dust" in these diagrams. Source: CRWMS M&O 2000x, Section 6.5.

- The risk of combustion due to burning electrical cables is low because of the expected use of special high-temperature and fire-rated cables, and because power cables will be separated from any instrument cables that are not fire-rated.

- Spent nuclear fuel and high-level radioactive waste will not be present in the areas being constructed. During the emplacement phase, radioactive materials, although present, will be contained in the sealed waste packages. A fire in the subsurface large enough to produce heat exceeding the waste package cladding peak temperature tolerance of 570°C (1,060°F) is considered a beyond Category 1 and Category 2 event sequence.

Remotely controlled equipment that may be susceptible to electrical fires would have the same fire detection and suppression systems as the waste package emplacement gantry, as described in Section 2.3.4.5.1. The potential for fire would also be minimized by administratively controlling the types and amounts of combustible materials brought underground.

2.3.4.2.3 Ground Control Monitoring and Maintenance

Design criteria for emplacement and nonemplacement drifts ground control (CRWMS M&O 2000w, Section 4.2) includes the following requirements directly related to worker safety:

1. The ground support systems must be designed to prevent rockfall that could result in personnel injury.

2. The ground control system must be designed to withstand a Category 1 or Category 2 earthquake, as appropriate to the seismic frequency classification assigned to each particular structure, system, and component.

3. The ground control system must be designed to include provisions which support a deferral of closure for up to 300 years (CRWMS M&O 2000w).

4. The ground support systems must be designed to function without planned maintenance during the operational life of the repository while providing the option to perform unplanned maintenance as needed.

Meeting these design criteria will create solid bases for a safe underground working environment as related to ground control. Even though the necessary maintenance for ground support systems is minimal, the rock face and ground supports would be inspected regularly. Additional supports would be installed as needed, and existing supports would be replaced if found to be inadequate or defective.

The presence of personnel in the emplacement drifts after waste emplacement is not expected unless repair or remedial work is absolutely necessary because of an off-normal event. Nonetheless, the emplacement drifts would be periodically inspected using remotely controlled inspection gantries. These remote inspections would document the conditions of the ground support systems inside the emplacement drifts. If conditions develop that could result in rock or ground support failure, appropriate maintenance and repairs would be scheduled. Preventive maintenance is intended to provide stable and safe subsurface facilities for personnel during preclosure.

2.3.4.3 Thermal Load Requirements

Results from computer simulations were used to analyze the effects of preclosure continuous ventilation in the emplacement drifts (CRWMS M&O 2000x, Section 4.1.1) and to define bounding conditions for the ventilation system design. Estimated temperatures in the emplacement drift exhaust air that would result from waste package linear heat loads of different magnitudes over a period of up to 200 years were considered for various ventilation rates. The results obtained for a ventilation rate of 15 m^3/s (530 ft^3/s) and for initial linear heat loads from waste packages of 1.4 and 1.6 kW/m indicated that peak air temperatures in the emplacement drifts of 58°C (136°F) and 63°C (145°F), respectively, would be reached 10 years after emplacement. The calculations also showed that, at heat loads of 1.4 kW/m and 1.6 kW/m and a ventilation rate of 15 m^3/s (530 ft^3/s), 71 percent of the heat generated would be removed in 50 years. The achieved heat removal percentage agrees favorably with the design requirement for the ventilation system to remove 70 percent of the heat generated by the waste packages during preclosure (CRWMS M&O 2000af, Section 1.2.1.9). The 1.4-kW/m and 1.6-kW/m linear loads provide a lower and upper bound for the average 1.45-kW/m linear heat load. Stroupe (2000, Attachment 1) imposed a maximum linear heat load of 1.5 kW/m on the operating mode described in this report. These linear loads are average loads for an emplacement drift. The waste emplacement requirements document (CRWMS M&O 2000ac, Section 1.2.4.16), however, allows emplacement of individual waste packages with a thermal output as high as 11.8 kW.

To achieve the 71 percent heat removal rate, an emplacement drift has to receive continuous ventilation for 50 years from the start of emplacement with the operating mode described in this report. If an emplacement drift is ventilated for more or less than 50 years, the amount of heat removal will be different (CRWMS M&O 2000x, Section 4.1.1).

For an average line load of 1.45 kW/m, an exhaust air temperature of approximately 60°C (140°F) was calculated as the expected peak (CRWMS M&O 2000x, Section 4.1).

Computer modeling using the program NUFT 3.0 shows that the temperature at the center portion of the rock pillar between emplacement drifts, defined as the quarter-pillar area, remains below 96°C (205°F) for the 1.45-kW/m thermal load. This condition prevails during the preclosure and postclosure periods for the ventilation rate specified for the higher-temperature operating mode (BSC 2001d, Section 6.2.3.7).

The emplacement drift ventilation flow can be varied to allow limited-time personnel or remote equipment access for evaluation and remediation

work for off-normal operation events. Since the emplacement drift exhaust air temperature peaks at 60°C (140°F), additional airflow will be required to lower the temperature below 50°C (120°F). An off-normal ventilation rate of 47 m^3/s (1,660 ft^3/s) per emplacement drift split, referred to as "blast cooling," will rapidly decrease drift temperatures. After 50 years of emplacement, the drift outlet temperature would fall below 50°C (120°F), and blast cooling would no longer be needed (CRWMS M&O 2000x, Section 6.1.4). Tables 2-13 and 2-14 summarize the characteristics of the ventilation system for the base case repository design and the allowable subsurface working temperatures.

Table 2-13. Design Basis of Ventilation System for Base Case Repository Layout

Item	Detail
Number of exhaust ventilation shafts	3
Number of intake ventilation shafts	3
Number of ventilation fans per exhaust shaft	2
Total electrical power for exhaust fans	10,000 hp
Airflow Quantities • Exhaust shaft airflow rate • Emplacement drift (70% heat removal) • Emplacement drift (blast cooling)	 800 to 850 m^3/s 15 m^3/s 47 m^3/s
Airflow Velocities • Minimum • Haulage mains and ramps • Exhaust main • Intake and exhaust shaft accesses • Intake and exhaust shafts	 1 m/s 6 m/s 8 m/s 8 m/s 20 m/s
Emplacement drift average heat load (based on 10 cm end-to-end waste package spacing)	1.42 kW/m

Table 2-14. Allowable Subsurface Working Temperatures

Item	Temperature °C (°F)
Human access maximum temperature	48 (118)
Human full shift occupation (8 hours +)	25 (77)
Instruments, monitoring equipment, and remote access equipment limit	50 (122)

2.3.4.3.1 Ventilation System Design

The ventilation system design covers four phases of the repository: (1) construction; (2) waste emplacement; (3) monitoring; and (4) closure. The emphasis of the system description in this section is thermal load management. Other functions of the ventilation system related to personnel safety are discussed in Section 2.3.4.2. All four phases are relevant to management of thermal loads.

The ventilation design is flexible in terms of repository capacity expansion, if needed. Ventilation designs for two repository sizes were analyzed: the first for the base case, or "statutory case," of a 70,000-MTHM repository, and the second for a hypothetical "full inventory" case of 97,000 MTHM.

2.3.4.3.1.1 Repository Ventilation System Concept

Figure 2-44 shows a general airflow pattern for ventilation of the emplacement drifts, using a representative section of the fully developed repository. Figure 2-45 shows a flow process diagram for the emplacement ventilation. These two figures illustrate the terms and processes described in this section. In the basic ventilation design, fresh air enters through intake shafts and the ramps and is distributed to the east and west mains. From the mains, air enters the emplacement, performance confirmation, or reserve drifts and flows to exhaust raises located near the center of each drift. The exhaust raises direct the airflow down to the exhaust main, where it continues to an exhaust shaft and then to the surface.

Fans located at the surface at the ends of the exhaust shafts would provide the moving force for subsurface repository airflow. The fans are designed with enough power to exhaust the maximum amount of air required during the emplacement, monitoring, and closure phases. The airflow volume that is moved by the fans would be variable, so as the thermal requirements of the repository change with time, the air volume can be adjusted. Air distribution within the repository would be controlled by flow regulators (valves or louvers) at each emplacement drift and in the exhaust main. The surface fan installation would create a higher negative air pressure on the exhaust main level than on the emplacement level, preventing recirculation from the emplacement drifts out to the main drifts. In other words, the surface fans would draw air through the exhaust

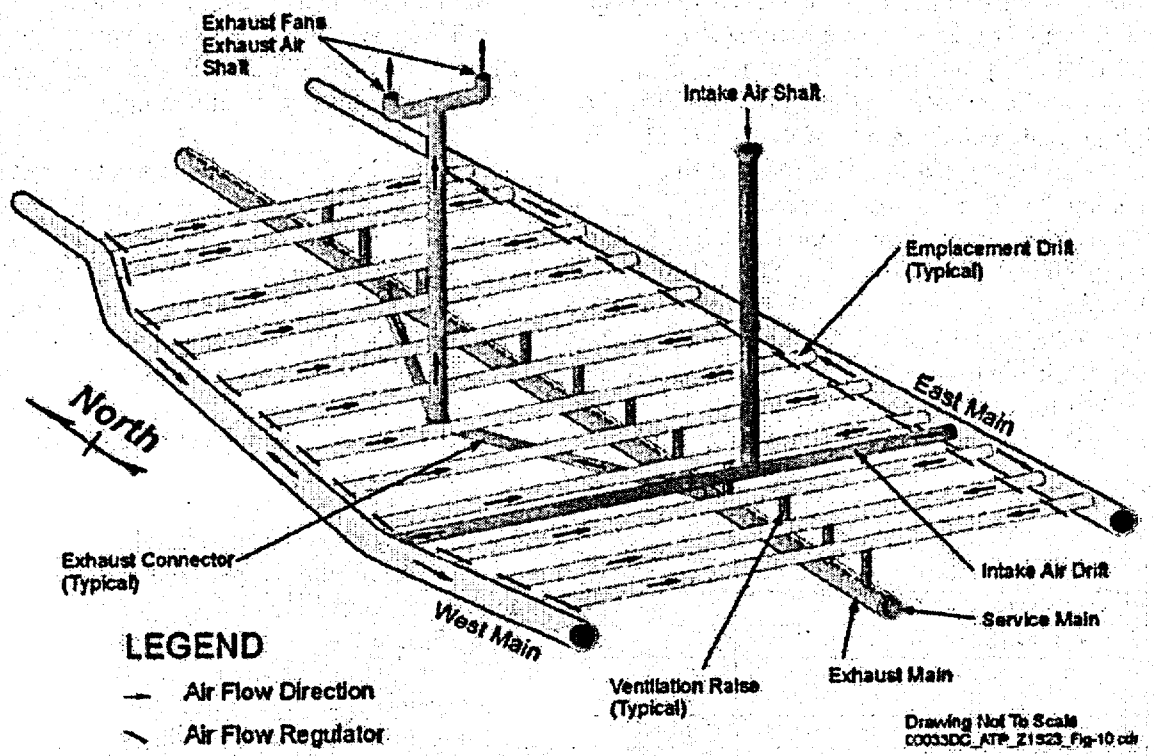


Figure 2-44. Repository Emplacement Area General Airflow Pattern
This simplified representation of the repository illustrates the emplacement side ventilation airflow patterns. Fresh air would be introduced through the intake air shafts and diverted to the east and west mains via the intake air drifts. Airflow from the mains would be available at both entrances to the emplacement drifts; the airflow rate into each "half" of the emplacement drifts would be controlled by airflow regulators. Heated air exits the emplacement drifts via the ventilation raises and enters the exhaust side of the exhaust main. Air from the exhaust main would be moved through the exhaust connectors to the exhaust air shafts, where it would exit to the surface. Dual exhaust fans at the surface end of each exhaust shaft would provide the moving force for the emplacement side ventilation system, such that the entire emplacement side is always under negative (vacuum) pressure. Source: CRWMS M&O 2000x, Section 6.3.3.

mains at a rate that ensures that air always circulates into the emplacement drifts from the main drifts and that never allows air to recirculate back to the main drifts.

The cross-block and performance confirmation drifts are designed to carry a variable airflow to react to off-normal events, maintenance requirements, waste package retrieval, drip shield installation, or drift repair. These drifts can carry additional airflow to the east or west mains or down the exhaust raises to vary or redirect the airflow on either the emplacement level or the exhaust level (CRWMS M&O 2000x, Section 6.1).

2.3.4.3.1.2 Emplacement Drift Ventilation

Ventilation requirements for emplacement drifts will vary according to the activities conducted in the emplacement drifts. Prior to emplacement, ventilation provides fresh air and controls dust levels to provide an acceptable environment for construction personnel. During emplacement, ventilation maintains drift temperatures within an acceptable range for equipment operation. After emplacement, ventilation will remove 71 percent of the heat generated by the waste packages. Ventilation rates in the emplacement drifts are adjustable to react to other requirements, such as mitigation of off-normal events in emplacement drifts (e.g., repair of collapsed ground supports). These various

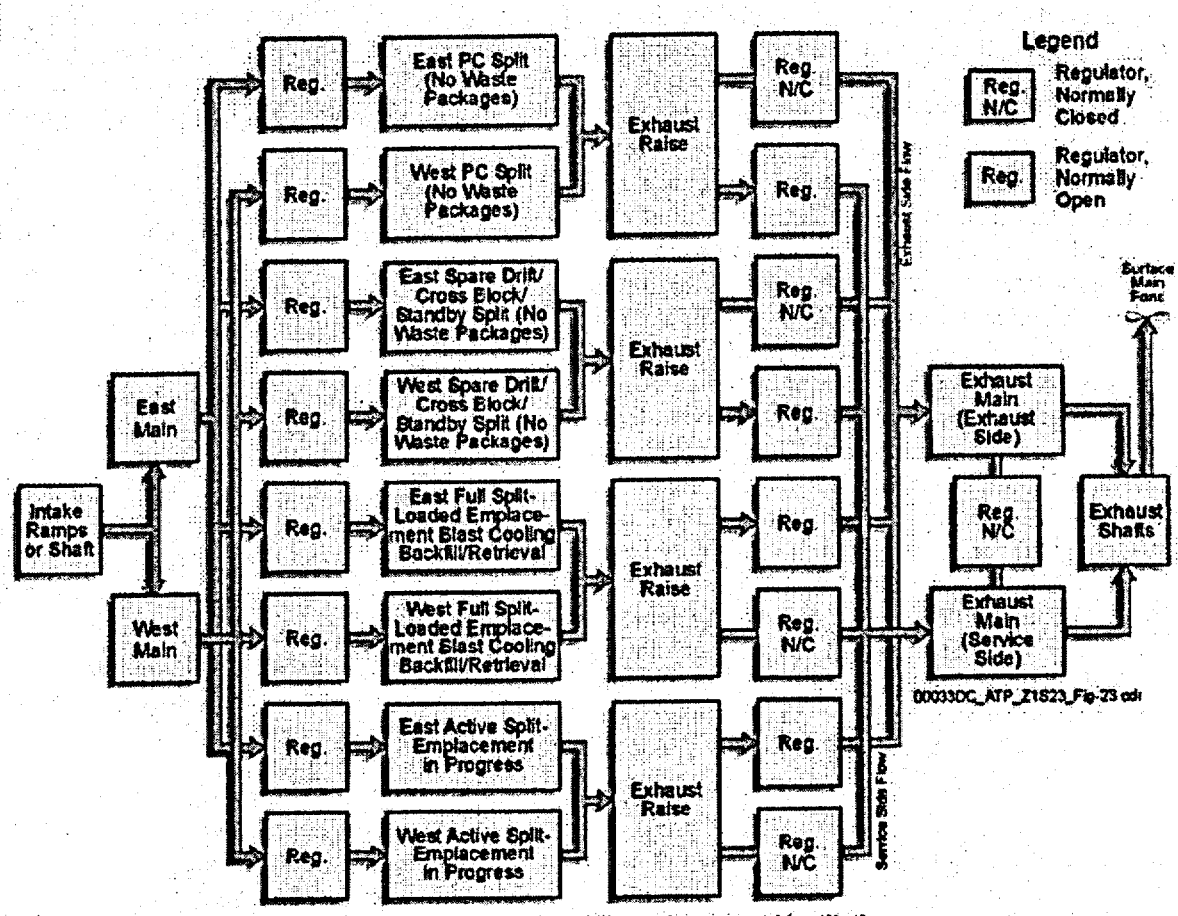


Figure 2-45. Flow Process Diagram for the Repository Emplacement Ventilation
This flow process diagram shows how the ventilation airflow is distributed and regulated throughout the emplacement side of the repository. Loaded emplacement drifts and drifts with emplacement in process would normally exhaust into the exhaust side. Cross-block and performance confirmation drifts would normally exhaust into the service side. PC = performance confirmation. Source: CRWMS M&O 2000x, Section 6.1.

requirements would be met by providing airflow regulators at the raises and doors of the emplacement drifts, and varying the total air volume at the surface fans.

The emplacement drift design includes two emplacement doors (isolation doors), one in the east turnout and one in the west turnout. Louvers built into these doors would serve as the inlets for the ventilation air, and a gate at the exhaust raise would serve as the outlet. These devices serve as the airflow regulators for the emplacement drifts. Both regulators are needed to control the airflow in emplacement drifts, since emplacement doors would periodically be opened to receive the waste packages. When the doors are opened, the valve at the exhaust raise would be the sole means of controlling the airflow (CRWMS M&O 2000an, Section 6.3.1).

2.3.4.3.1.3 Flow Rate Requirements for Heat Removal

The calculations described in Section 2.3.4.3 concluded that, with a ventilation flow rate of 15 m^3/s (530 ft^3/s), 71 percent of the heat would be removed after an emplacement drift had been ventilated for 50 years. This meets the requirement for 70 percent heat removal during the preclosure period (CRWMS M&O 2000af, Section 1.2.1.9).

Since each emplacement drift is divided in half (each half is called a "split"), an emplacement drift ventilation rate of 15 m^3/s (530 ft^3/s) per split was used in the ventilation analysis (CRWMS M&O 2000x, Sections 6.1 and 6.2). As air passes over the waste packages, it is heated, and it expands as a result of the heat. The equivalent ventilation rate, adjusted to account for the expansion of the air, is then calculated to be 17 m^3/s (600 ft^3/s) for an exhaust temperature of 60°C (140°F). This expanded ventilation rate is multiplied by two (two splits per emplacement drift) and by the number of emplacement drifts to obtain the total ventilation rate required for emplacement heat removal. The calculated emplacement ventilation rates are 1,870 m^3/s (66,000 ft^3/s) and 2,584 m^3/s (91,250 ft^3/s) for the base and full inventory cases. After adding incremental ventilation rates for nonemplacement drifts and rapid cooling demands for off-normal cases (such as for off-normal retrieval, as described in Section 2.3.4.6), the total repository ventilation rates for the two design cases are estimated as 2,225 m^3/s (78,580 ft^3/s) and 2,901 m^3/s (102,450 ft^3/s), respectively.

In case of ventilation system failure after the waste is emplaced, it would take a period of approximately 2 to 3 weeks for the drift wall maximum temperature limit (96°C [205°F]) to be exceeded. To maintain thermal goals, any required repairs would have to be completed and the ventilation system restarted during this period.

2.3.4.3.1.4 Sizing of Ventilation Facilities

The minimum number of ventilation shafts and fans required to provide these total ventilation rate requirements have been determined in the ventilation analysis (CRWMS M&O 2000x, Sections 6.2 and 6.7). Using two fans per shaft operating in parallel, as shown in Figure 2-46, with a combined exhaust shaft ventilation rate of 800 to 850 m^3/s (28,250 to 30,000 ft^3/s), it is estimated that at least three exhaust shafts are needed to support the 70,000-MTHM repository design. The 97,000-MTHM repository design would require at least four exhaust shafts. Each exhaust shaft is paired with an equivalent intake shaft. Therefore, the two repository design cases would require at least six and eight ventilation shafts, respectively. The combined electric motor capacity required to operate these ventilation systems has been calculated as at least 10,000 hp and 13,000 hp, respectively.

As explained in Section 2.3.5, when the repository is undergoing concurrent development and emplacement operations, two separate ventilation systems would have to be maintained for each operational area. This separation is accomplished by placing air locks in the main drifts to physically separate the air space between the two areas. On the development side, the ventilation system works under positive pressure, with air being forced in through the development/intake shaft or the south ramp through a duct and exhausted through the south ramp. Exhausting air through the exhaust main is also possible (through the service or "cool" side of the exhaust main) if the required facilities are available. In the early stages of repository construction, the exhaust shafts on the development side are being developed, so they would not be available for exhausting air. On the emplacement side, all the required ventilation facilities for the commissioned emplacement panels are available and operational in their final configuration; the ventilation system works under negative pressure by drawing air out through the exhaust main (through the exhaust or "hot" side of the exhaust main) and from there through the exhaust shafts. This is accomplished by operating the ventilation fans installed at the exhaust shaft surface openings in an exhaust mode. Air in the emplacement side is drawn in through the intake shafts and the north ramp. The air locks prevent air from the emplacement side, which could be contaminated, from migrating to the development side. If the air locks are not completely airtight, airflow is induced from the development side to the emplacement side because of the pressure differential across the air locks (i.e., the negative pressure on the emplacement side acts as a vacuum at this interface).

The overall repository design layouts for the 70,000- and 97,000-MTHM case, as illustrated in Figures 2-38 and 2-39, show proposed locations for the intake and exhaust shafts, emplacement and nonemplacement drifts, mains, and ramps.

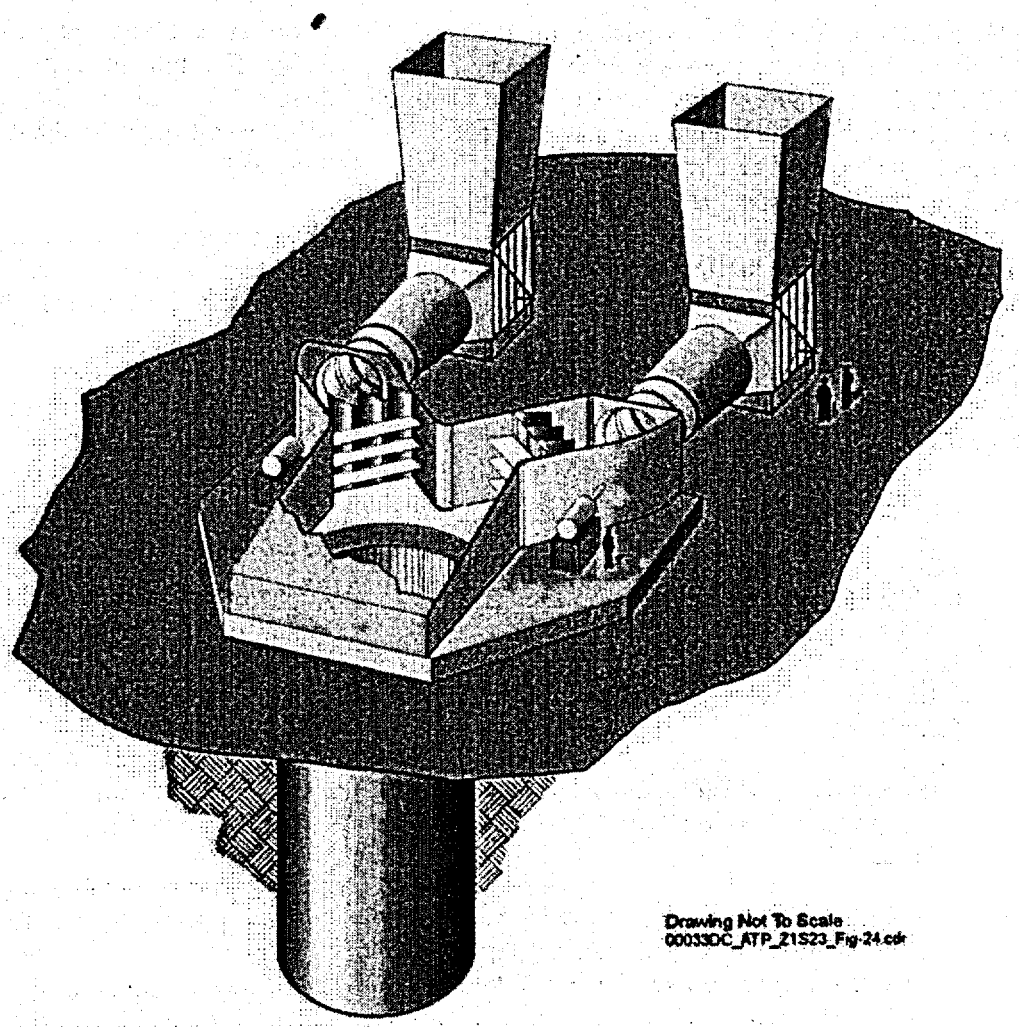


Figure 2-46. Repository Exhaust Shaft Conceptual Dual Fan Installation
Each repository exhaust shaft would be equipped with dual fans with a combined capacity to remove up to 850 m^3/s of heated air. Source: CRWMS M&O 2000x, Section 6.2.4.

Tables 2-13 and 2-14 summarize the preliminary engineering specifications for the subsurface ventilation system.

2.3.4.4 Waste Transfer and Transport

This section describes the equipment and methods proposed for transferring and transporting waste packages from the repository surface facilities to the emplacement drifts. The equipment and methods for emplacing waste packages into emplacement drifts are discussed in Section 2.3.4.5.

2.3.4.4.1 Onsite Transportation Routes and Waste Package Handling Sequence

The onsite transportation routes from the surface facilities to the underground emplacement area will vary, depending on the emplacement drift selected as the destination for a particular waste package. Selection of the emplacement drift is based on a predefined emplacement plan that considers waste package thermal properties and other operational considerations. Generally, however, waste emplacement will occur from north to south, lagging behind emplacement drift construction, with waste packages emplaced in recently

completed drifts as they become available. Figure 2-47 shows the overall repository subsurface layout (BSC 2001d, Figure 10), to which a portion of the surface facilities and the rail line connecting the surface and subsurface facilities have been added. The repository features pertinent to the transfer of waste package operations and shown in the figure (where they are labeled by the following numbers) are:

1. The Waste Handling Building, which is part of the surface facilities and is where the waste packages would be transferred to the train that transports them underground

2. The rail line from the Waste Handling Building to the North Portal

3. The North Portal, which is one of two entrances to the repository subsurface facilities

4. The north ramp, which is an underground tunnel leading from the North Portal to the east main

5. The north ramp extension, which is the tunnel that extends from the north ramp (above the curve) toward the north to meet the east main

6. The north ramp curve, which is the section of the north ramp that turns toward the south to meet the east main drift

7. The north main, which is the loop that connects the east main to the west main at the north end of the repository block

8. The east and west mains, which are the tunnels that provide access to the ends of the emplacement drifts

9. The south main, which is the loop that connects the east main to the west main at the south end of the repository block

10. Emplacement drift turnouts, which are the curved sections of drift connecting each end of an emplacement drift to the main drifts. The turnout approach from the main drifts is always from the south, at both the east and west sides of the repository block.

The subsequent sections describe the subsurface system components and equipment. Refer to the following figures for additional illustration of components:

- Locomotives and the transporter: Figures 2-48, 2-49, and 2-50
- Emplacement drift turnouts: Figure 2-49
- Pallets: Figure 2-51
- Waste packages: Figure 2-52
- Emplacement gantries: Figures 2-55 and 2-56.

The subsurface facilities listed above would be used at different times during the life of the repository for waste package transfer operations. In the early years of emplacement, a typical route and sequence of events would be:

- The waste package will be loaded onto the transporter at the Waste Handling Building.

- The locomotives will move the loaded transporter from the Waste Handling Building to the North Portal, down the north ramp to the north ramp extension, and from there to the preselected emplacement drift turnout. The train (transporter and locomotives, as shown in Figure 2-48) will generally follow the shortest route to the emplacement drift; however, orientation of the transporter is important, depending on whether the emplacement drift is going to be reached from the east or west main. The open deck of the transporter must face the drift docking area for transfer of the waste package. The transporter and locomotives, once they enter the subsurface area in these early emplacement years, cannot rotate or change the orientation of the transporter. Therefore, the transporter must be oriented in the proper direction at the surface facilities using railroad turnouts.