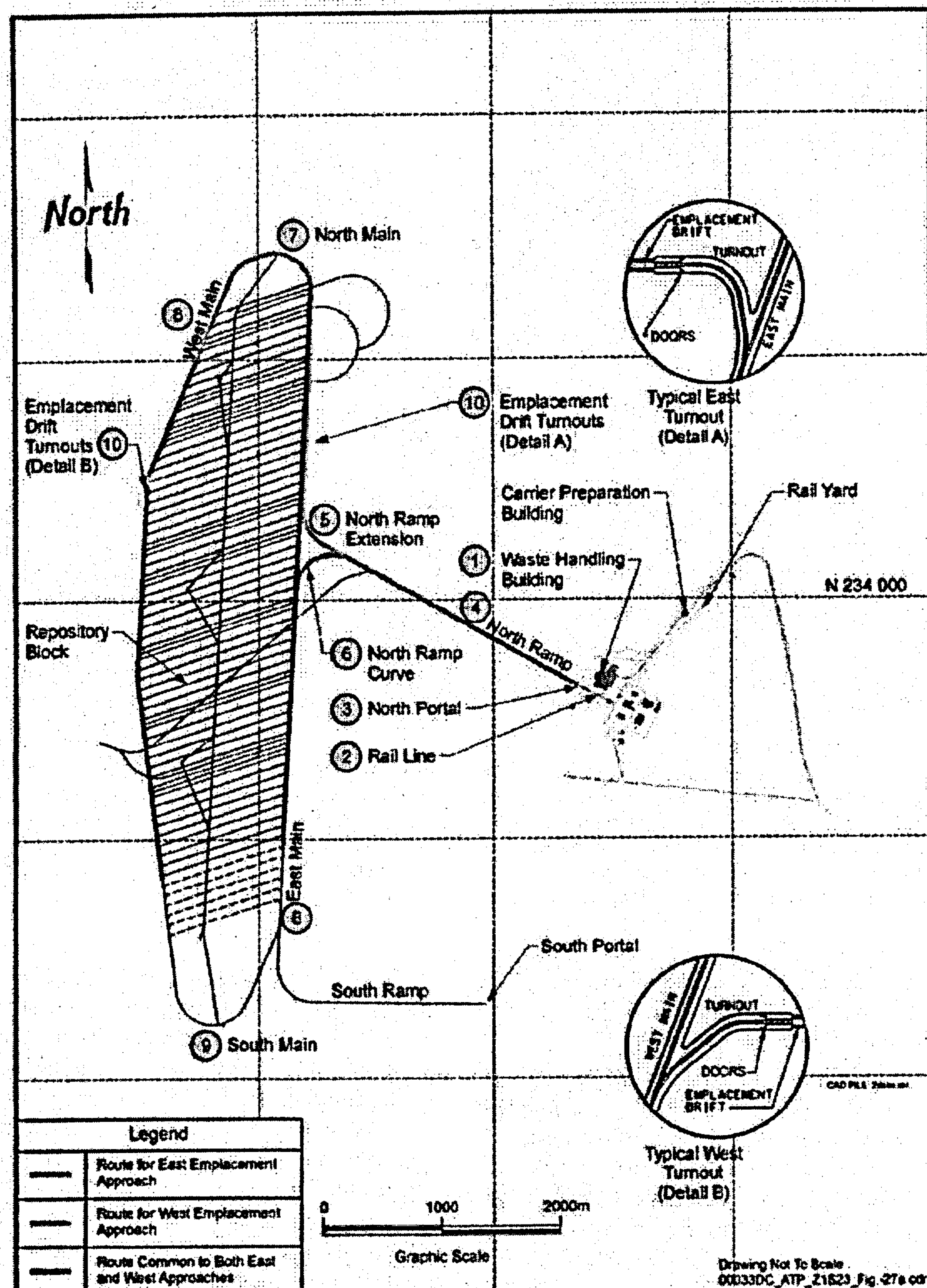

Figure 2-47. Waste Package Emplacement Route—Key Locations

The waste package transportation train would follow a predetermined route selected according to the destination of the waste package in the underground drifts. The emplacement location of each waste package would be determined based on thermal management procedures and operational considerations, such as emplacement sequencing.

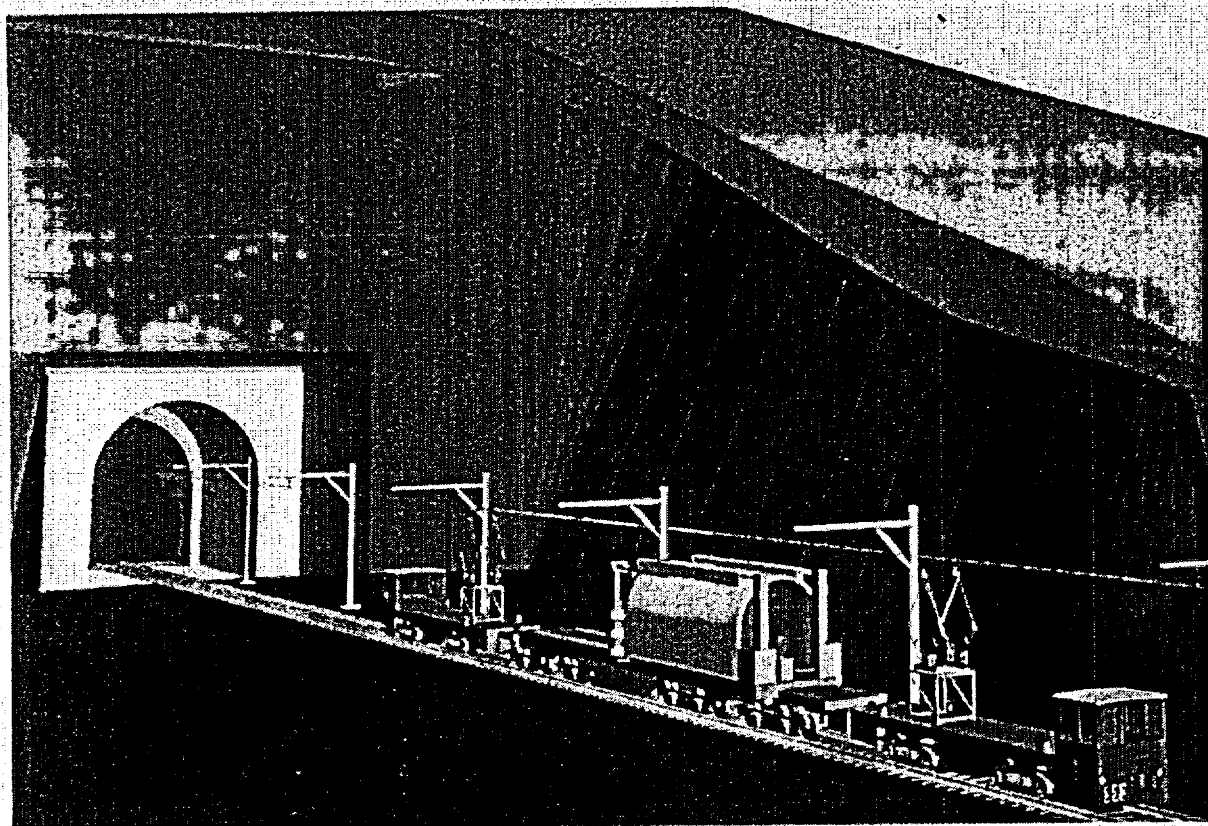

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Figure 2-48. Locomotives and Waste Package Transporter Approaching the North Portal
Two electric locomotives, one on each end of the waste package transporter, would be used to move the transporter containing the waste package. This train would travel from the Waste Handling Building at the surface to the emplacement drift entrance. Travel down the gently sloping north ramp would be made safe by using dual operators and locomotives and a transporter equipped with redundant and independent brake systems. Automatic brake controls would monitor the speed of travel to minimize dependency on operators.

- If an emplacement drift will be reached from the west main, the train will get there by going around the north end.

- The return trip to the surface will follow the same route in the reverse sequence. The transporter's open deck must face the Waste Handling Building to accept another waste package. Therefore, the transporter must be properly oriented before docking at the Waste Handling Building.

A typical route from the Waste Handling Building to an emplacement drift would change as construction of the repository and emplacement activities progress. The open deck of the transporter must face the emplacement drift docking area for transfer of the waste package, and it must face the Waste Handling Building to accept another waste package. Therefore, routing takes into account the orientation of the waste package transporter. The proper orientation of the transporter is achieved by making a circular route all the way around the repository perimeter mains, or by switching the direction of the train at the surface turnouts or at locations where the north ramp and the north ramp extension meet the east main.

The waste package handling sequence is simple and would not change throughout the waste emplacement period. Waste package transportation for emplacement is always by rail. The sequence can be summarized as follows (CRWMS M&O 2000ao, Section 6; CRWMS M&O 2000y, Section 6):

1. The waste package, loaded on an emplacement pallet, is placed on the deck of the transporter waiting at the receiving dock of the Waste Handling Building.

2. A semirigid chain mechanism pulls the pallet and waste package into the shielded enclosure of the transporter via the bed plate, which is supported on rollers.

3. The shielded enclosure doors are closed to protect the operators, and a primary locomotive pulls the loaded transporter away from the Waste Handling Building docking area.

4. During a stop at a track turnout outside the Waste Handling Building, a secondary locomotive joins the train by coupling itself to the transporter. Both locomotives are driven by operators.

5. Both locomotives, one in front and one behind, move the loaded transporter into the subsurface facilities.

6. The train stops at the main drift, near the predetermined emplacement drift turnout.

7. The locomotive at the rear of the transporter is decoupled from the transporter. The locomotive operators leave the locomotives and move to a designated location to protect themselves from radiation while the emplacement drift doors are open.

8. The locomotive controls are turned over to remote control operators in a control center at the surface, and the locomotive in front of the transporter moves the transporter into the turnout and stops before reaching the emplacement drift docking area.

9. The transporter doors are fully opened, and the emplacement drift isolation doors are also fully opened from the surface control center.

10. The locomotive docks the transporter (see Figures 2-49 and 2-50), pushing the open deck section of the transporter completely inside the emplacement drift.

11. The semirigid chain mechanism pushes the pallet and waste package from inside the shielded transporter enclosure to the open deck area of the transporter via the bed plate, which is supported on rollers.

12. The waste package emplacement gantry, also riding on rails and remotely operated, moves from inside the emplacement drift completely over the waste package and pallet, straddling the transporter's open deck. The gantry lifts the waste package by its pallet and moves back into the drift to the waste package emplacement location.

13. The locomotive moves the transporter away from the emplacement drift docking area and stops. The transporter doors and the drift doors are completely closed.

14. The locomotive moves the transporter from the turnout to the main drift and stops to allow the second locomotive to couple to the train.

15. The operators board the locomotives, the controls are turned back to manual operation, and the train proceeds to the surface for another cycle.

16. Before docking at the Waste Handling Building, the train stops, the locomotive in front of the transporter is decoupled and moved away to a standby location, and the locomotive behind the transporter moves the transporter to the building dock.

Some of the steps in this sequence are illustrated in Figures 2-48, 2-49, and 2-50.

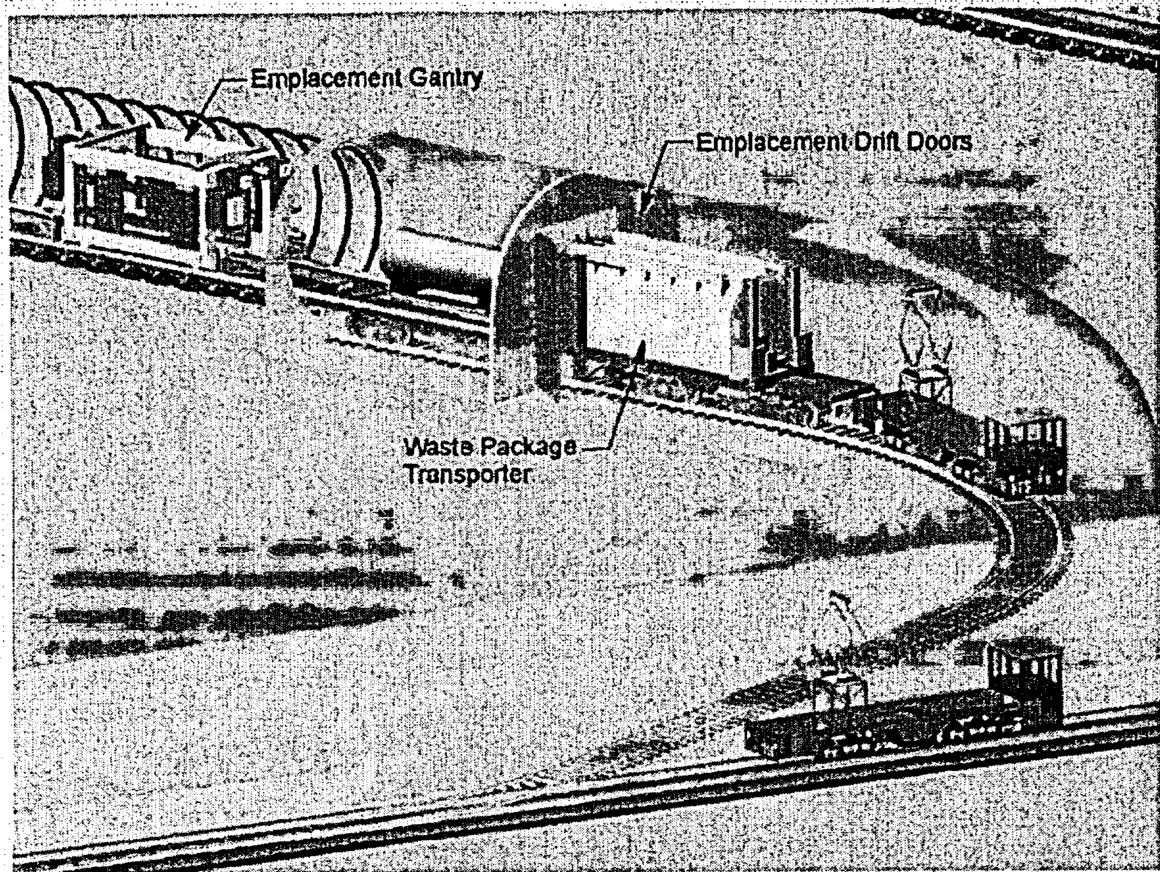

Figure 2-49. Locomotive Operations at Emplacement Drift Turnout
This figure has been designed with see-through drift rock walls so the maneuvering of the locomotives and transporter at the emplacement drift turnouts can be better appreciated. One locomotive is decoupled from the transporter at the main drift, the operators leave the locomotives and take shelter while the drift doors open, and the other locomotive is then remotely operated for the travel into the turnout area and delivery of the waste package. This figure shows one locomotive on standby while the other locomotive moves the transporter into the turnout.

2.3.4.4.2 Waste Packages and Pallets

Section 3 provides specific information about waste package types and sizes, their characteristics, and the waste forms they contain.

Figure 2-51 provides an isometric view of an emplacement pallet. There would be two sizes of pallets: one that holds most of the waste packages, and a second, shorter version used for the 5-DHLW/DOE SNF waste package (CRWMS M&O 2000ap, Section 6.2). The emplacement pallets would be fabricated from Alloy 22 plates welded together to form the waste package supports. Two supports would be connected by square stainless steel tubing to form the completed emplacement pallet. The supports would have a V-groove top surface to accept all waste package diameters. Emplacement pallet surfaces that contact the waste package would be Alloy 22, the same material used for the package's outer shell. The pallet would be shorter than the waste packages, so the waste package is supported on the outer package shell between the trunnion collar sleeves (see Figure 2-52). The waste package would not be mechanically attached to the pallet; it would rest on the V-groove surfaces of the pallet. The ends of the waste package extend past the ends

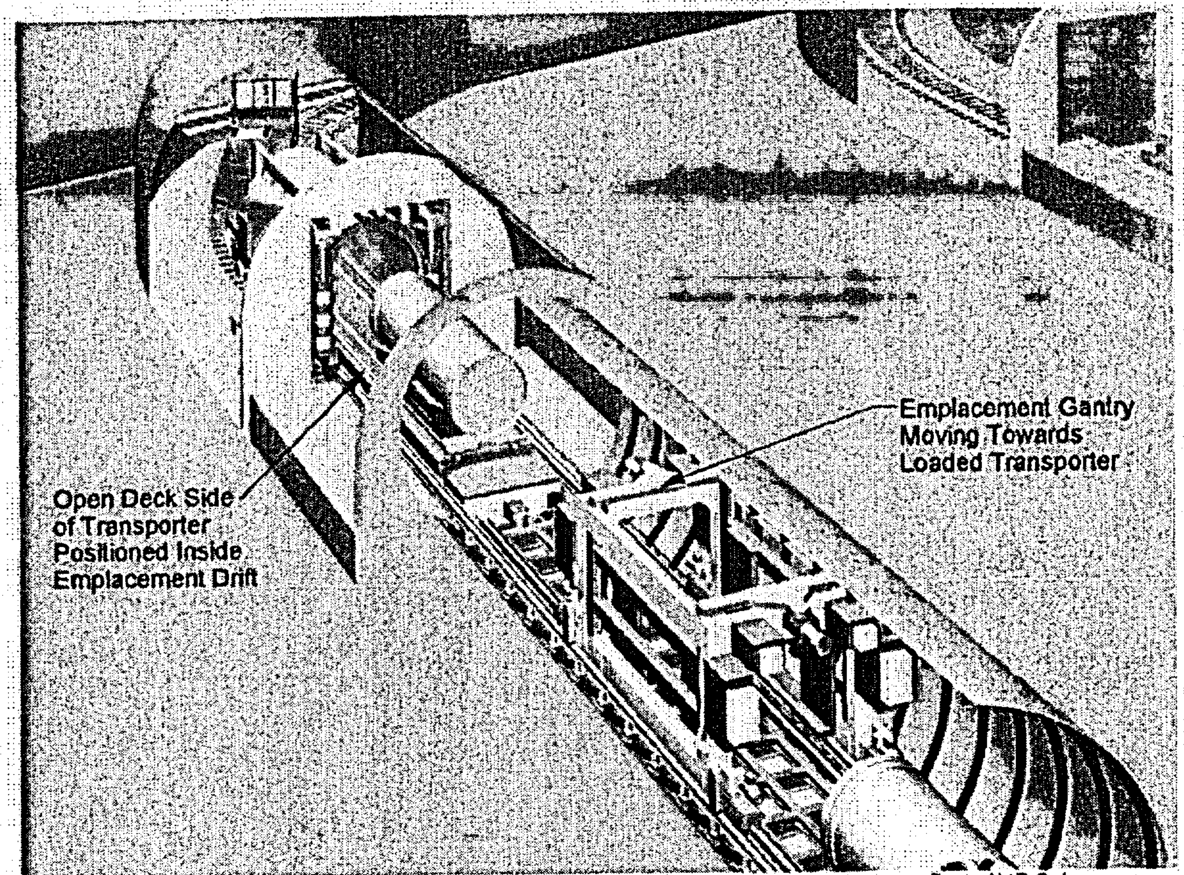

Figure 2-50. Docked Transporter with Pallet and Waste Package on Transporter's Open Deck and Emplacement Gantry Approaching the Docking Area for Pickup

As the transporter is docked into the emplacement drift entrance, the waste package resting on the pallet would be rolled out of the transporter shielded enclosure via a bed plate mounted on rollers. The emplacement gantry, waiting inside the emplacement drift, would be moved over the waste package to straddle and pick up the pallet and the waste package combination. The loaded gantry would then be moved into the drift for emplacement of the waste package and pallet.

of the emplacement pallet, which would allow the waste packages to be placed end to end, within 10 cm (4 in.) of each other, without interference from the pallets.

The emplacement pallet would be moved as one unit with the waste package from the Waste Handling Building to the emplacement drift, and would support the waste package in the drift permanently. While loaded with a waste package, the pallet would be lifted by lifting points at the support, directly under the upper stainless steel tubes, as Figure 2-52 illustrates. The pallet design meets the design requirements for structural strength (CRWMS M&O 2000ap, Section 6.2) during lifting under the weight of the heaviest waste package, as documented in *Design Analysis for the Ex-Container Components* (CRWMS M&O 2000ap, Section 6.4). Dimensions and dimensional clearances for the pallet/waste package combination are provided in Section 2.3.4.5.

The *Emplacement Drift System Description Document* (CRWMS M&O 2000ab, Section 1.2.1.20) requires that the waste packages be retrievable for a period of up to 300 years. Therefore, the emplacement pallet must remain in a condition that can be lifted even at the end of this period. Neces-

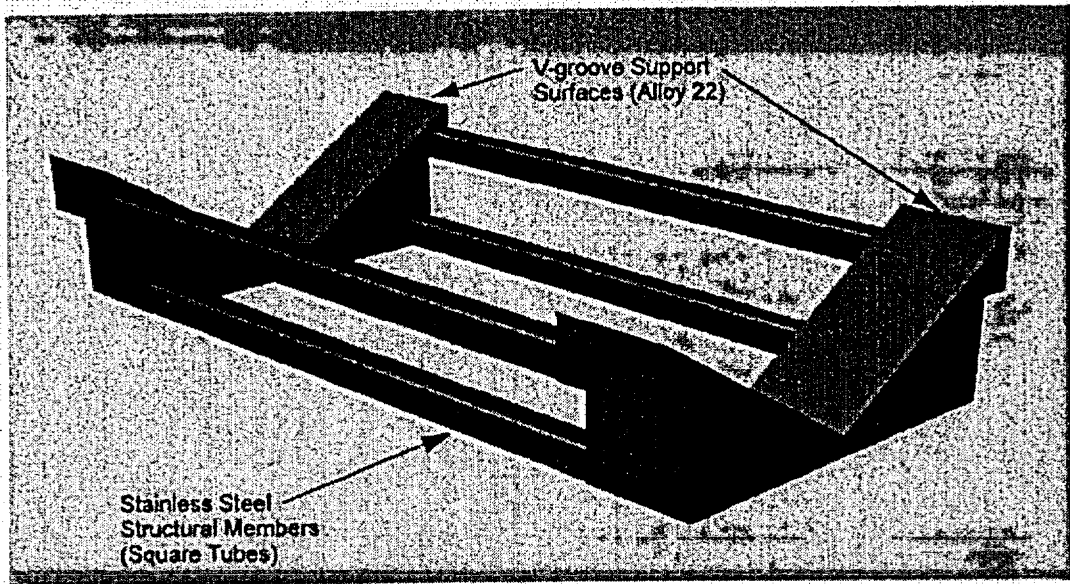

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Figure 2-51. Emplacement Pallet Isometric View
This figure illustrates the configuration of the pallet. There would be two sizes, one for long packages and another for short packages. Source: CRWMS M&O 2000ap, Section 6.2.

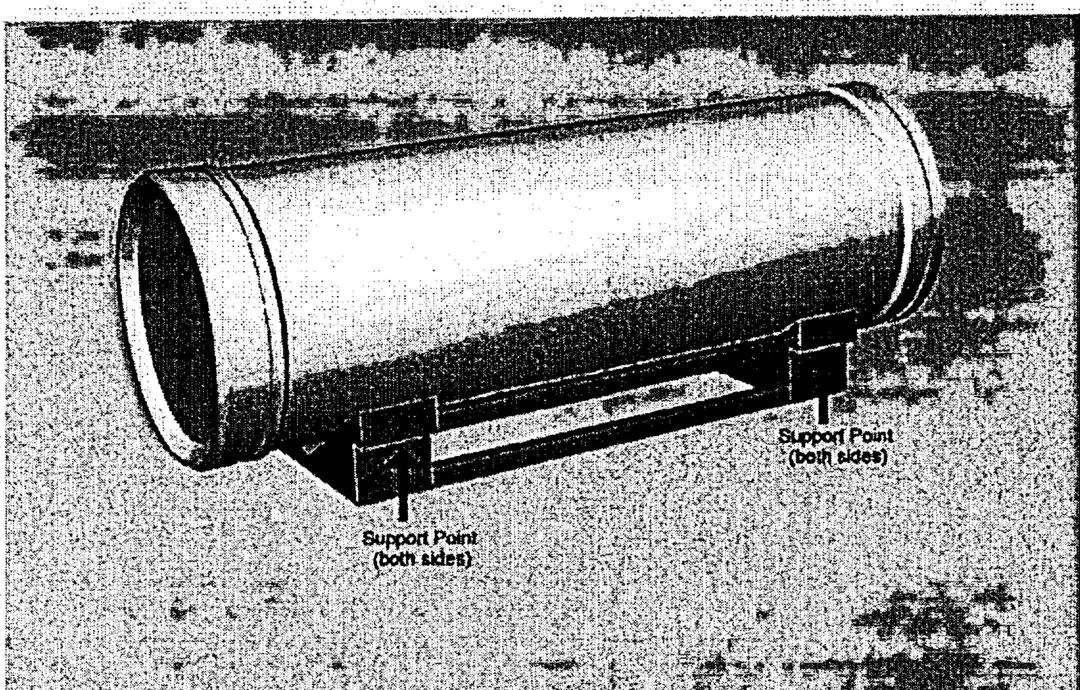

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Figure 2-52. Emplacement Pallet Loaded with Waste Package
The emplacement gantry lifting arms engage the pallet at the support points illustrated in the figure, on both sides of the pallet. The gantry during the lifting or emplacement operations does not contact the waste package. The shorter length of the pallet, with respect to the length of the waste package, allows the gantry to emplace one waste package close to another without interference by the pallets. Source: CRWMS M&O 2000ap, Section 6.2.

sary calculations have been done to support this design requirement (CRWMS M&O 2000ap, Section 6.4). Seismic activity could also affect the ability of the pallet to maintain the normal waste package position. Seismic calculations would be incorporated in support of license application (CRWMS M&O 2000ap, Section 6.2).

2.3.4.4.3 Locomotives and Waste Package Transporter

2.3.4.4.3.1 Rail System

The locomotives and waste package transporter use a track gauge of 1.44 m (56.7 in.). An American Institute of Steel Construction standard crane rail of 66.9 kg/m (135 lb/yd) would be used in the north ramp, in all the main drifts and turnouts, and for the rail line leading to the Waste Handling Building (CRWMS M&O 2000ao, Sections 6 and 7). The rail system is designed to support the transportation of waste packages over the life of the system, including retrieval, for the design loads imposed on the rail by the locomotives and transporter. The bounding weight used in the calculations for a loaded transporter is 400 metric tons (CRWMS M&O 2000ao, Section 6.4.3.4). This bounding weight includes component weights listed below, in addition to an adequate design margin:

- The heaviest loaded waste package: 85 metric tons (includes 12.5-metric ton design margin)
- The pallet: 3 metric tons
- The bed plate that rolls in and out of the shielded enclosure: 9 metric tons
- Transporter shielding and fixed equipment: 164 metric tons
- The transporter flat deck car: 136 metric tons.

The trucks and wheels for the locomotives and transporter are designed to negotiate the 20-m (65.6-ft) curve radius that is the design basis of the track system. The curvature of the emplacement drift turnouts imposes this limitation. The north ramp and other main drifts will not have curvatures with less than a 305-m (1,000-ft) radius (CRWMS M&O 2000ao, Section 4.2.3.4). This limitation is not driven by the rail alignment but by the allowable curvature of the muck conveyor system used during excavation to avoid transfer stations.

The steepest railroad grade for the entire waste emplacement and transportation system is found in the north ramp. The grade is only 2.15 percent, which is well within the operational range of the proposed locomotives (CRWMS M&O 2000y, Sections 4 and 5). Rail grades in the south ramp are steeper but are not intended for waste package transportation. The south ramp would be used to support repository development activities, such as transportation of construction materials.

2.3.4.4.3.2 Locomotives

The two waste package transport locomotives would be 50-ton, 4-axle, 8-wheel units, as Figures 2-48 through 2-50 illustrate. The locomotives use electrical power (650-volt nominal direct current) supplied through an overhead catenary wire and a pantograph (a wide contactor supported by a hinged, diamond-shaped structure mounted on the roof). The rails act as the ground, completing the electrical circuit. The overhead catenary wires would be installed in all drifts and turnouts where the locomotives would operate. Each locomotive has dual direct-current electric motors rated at 170 hp each (CRWMS M&O 1998e, Section 7.3). Other functions performed by the waste transport locomotives include:

- Transporting waste packages to the surface in support of retrieval operations, if necessary
- Transporting waste packages between drifts, as needed, to support repository operations and maintenance activities
- Transporting the emplacement gantry carrier from the surface to the emplacement drift turnouts and from drift to drift
- Transporting the inspection gantry carrier from the surface to the emplacement drift turnouts and from drift to drift

- Transporting the drip shields and their emplacement gantry from the surface to the emplacement drifts.

For redundancy and added safety, two locomotives would be used any time the transporter is mobilized (except during docking operations at the Waste Handling Building and at the emplacement drift turnouts, where only one locomotive is used). One locomotive would be in front of and the other behind the transporter. This train configuration provides assurance against the possibility of a runaway transporter or adverse situations created by operator error. Section 2.3.4.4.4 discusses safety analyses of the locomotives and waste package transport operations.

The locomotives and transporter would operate within the space clearances in the north ramp, north ramp extension, and main drifts. The main drifts would be excavated to the same diameter as the ramps, 7.62 m (25 ft). Figure 2-53 is a perspective view of the locomotives and transporter in transit along one of the main drifts. This figure illustrates the size of the equipment relative to the main drift diameter.

The maximum operating speed for the locomotives and transporter has been defined as 8 km/hr (5 mph) (CRWMS M&O 2000ac, Section 1.2.2.1.2). Standard underground industry practice defines 16 km/hr (10 mph) as the maximum standard operating speed for locomotive travel within a mine environment (CRWMS M&O 2000y, Section 6.6). Therefore, the selected train operating speed for the repository is well within safety limits established by the industry.

2.3.4.4.3.3 Waste Package Transporter

The waste package transporter consists of a flat railcar 22 m (72.4 ft) in length, with a waste package shielding structure at one end and an open deck at the other end. Ancillary equipment installed on the transporter includes mechanisms for opening and closing the shielded doors; a bed plate supported on rollers; a semirigid chain mechanism; and tracks for rolling the bed plate in and out of the shielded enclosure. Figure 2-54 illustrates the transporter and its components.

This waste package transporter design with an integrated transfer deck eliminates the complexities of aligning several separate components at the waste

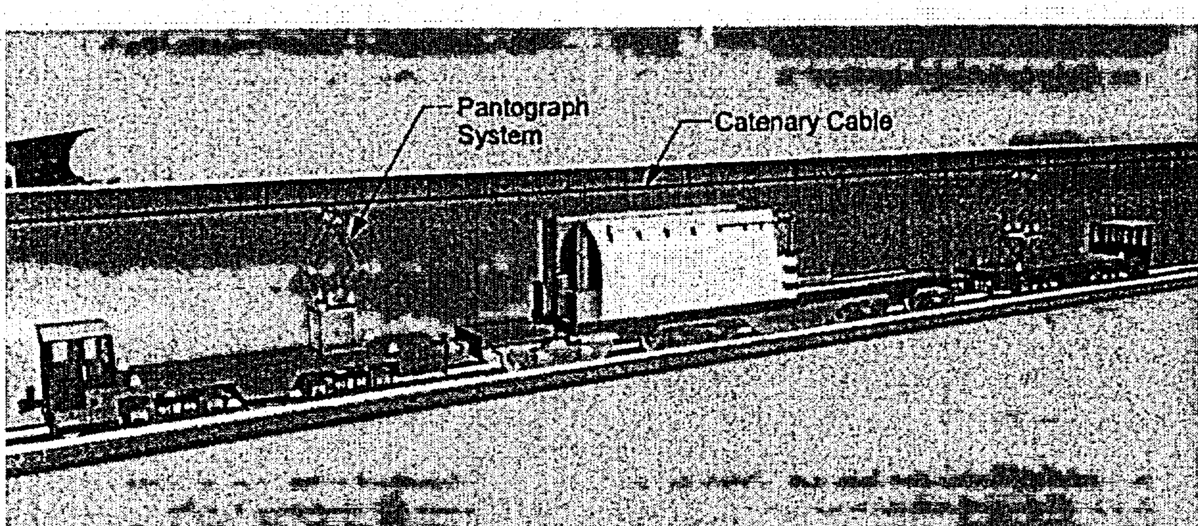


Figure 2-53. Waste Package Transportation Equipment Traveling Along Main Drift
This figure illustrates the size of the waste package transportation equipment relative to the size of the main drift. It also illustrates the catenary cable and pantograph system for continuous power supply to the locomotives.

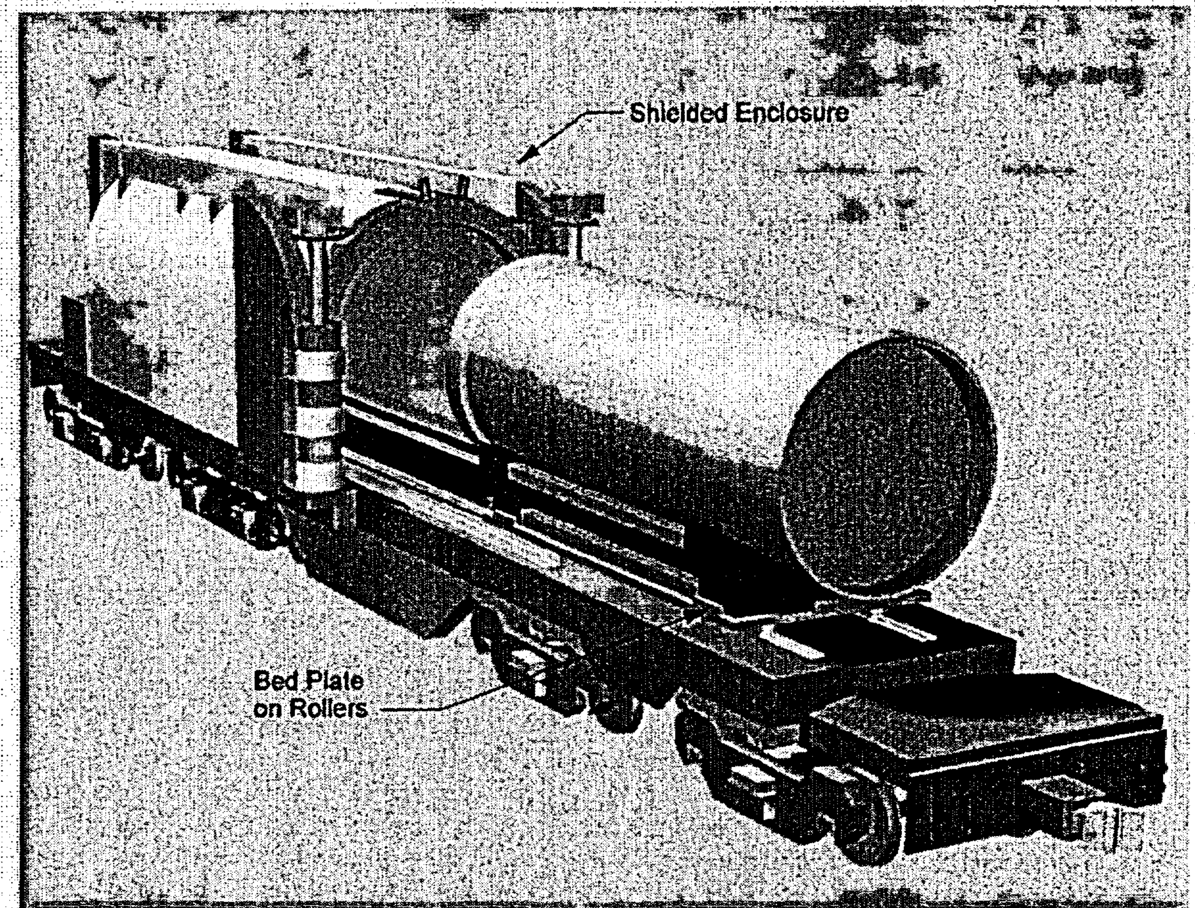


Figure 2-54. Waste Package Transporter and Its Components
The waste package transporter would be equipped with a radiation shield enclosure and self-contained mechanical systems for opening the enclosure doors and for rolling the bed plate in and out of the enclosure. The open deck portion of the transporter extends into the emplacement drift when docked at the drift entrance. This allows the emplacement gantry to straddle the transporter and be positioned right over the loaded pallet for its pick up. The emplacement drift rail gauge would be wide enough to accommodate the transporter between rails.

package transfer locations. The integrated transporter design, with the shielded enclosure located toward one end of the flat deck car, allows the open deck area to extend adequately beyond the shielding structure, which permits the waste package and pallet to be moved together out of the enclosure and positioned for access by the emplacement gantry. The waste package transporter would be docked between the rails of the emplacement gantry in the emplacement drift, enabling the emplacement gantry to straddle the transfer deck to pick up the waste package and pallet.

A bed plate sized to contain and restrain the pallet has four low-profile, heavy-duty rollers affixed to its underside to aid movement of the waste package out of the shielded enclosure. After the emplacement gantry engages the pallet and raises the loaded pallet high enough to clear the bed plate, the bed plate would be retracted back into the shielded enclosure. This transporter design eliminates the need for precise alignment of tracks between the transporter and the emplacement drift, and it diminishes the possibilities for waste package mishandling and dropping incidents (CRWMS M&O 2000ao, Section 6.4). The gantry lifting screw limits the maximum height the waste package can be lifted to less than 1 m above the deck of the transporter (CRWMS M&O 2000z, Section 6.3). This maximum lifting height limit is much less than the 2-m (6.6-ft) vertical lift limit specified for protection of the waste package (CRWMS M&O 2000ac, Section 1.2.2.1.3). To eliminate the risk of dropping a waste package in the docking area, the transporter remains in the area until the loaded gantry moves back into the emplacement drift.

Design calculations for the transporter (CRWMS M&O 2000ao, Section 6.4) were done for various static and dynamic loading conditions, which resulted in the selection of industry standard 320 BHN (Brinnel hardness number) wheels and rails. The wheels have a diameter of 762 mm (30 in.). The 320 BHN wheels can be used by equipping the transporter with dual 3-axle trucks for the rear and dual 2-axle trucks for the front.

The bed plate consists of a thick steel plate that distributes the waste package and pallet load through rollers (attached to the bottom of the bed plate) to the transporter. The design uses reliable, low-profile unitary rollers. The roller assembly runs on an inverted channel installed on the deck of the transporter. The channel safely and reliably guides the bed plate rollers to a position centered with the emplacement gantry for pickup. The bed plate would be attached to the ends of two gear-driven semirigid chains that run in floor-mounted guides on each side of and parallel to the bed plate as it rests on the transporter. The chain guides extend from inside the shielded enclosure along the transfer deck. The semirigid chain attached to the bed plate pushes the bed plate out of, and draws it back into, the shielded transporter enclosure. The bed plate incorporates restraints and spacers to immobilize the waste package, pallet, and bed plate inside the transporter for safe transit to the emplacement drift (CRWMS M&O 2000ao, Section 6.4).

The shielded enclosure is designed to meet the radiation dose rate limit of 100 mrem/hr on the transporter surface (CRWMS M&O 2000ao, Section 6.7.3), which is accomplished with a very thick A 516 steel enclosure wall. The steel wall provides sufficient shielding for nonaccessible areas (i.e., the deck floor below the shielded enclosure) during normal operations. In addition to the A 516 steel, the shielded enclosure has a borated polyethylene layer for neutron shielding (CRWMS M&O 2000ao, Section 6.7).

2.3.4.4.4 Waste Transfer and Transport Safety Analyses

A safety analysis (CRWMS M&O 2000y) was performed for the waste package transport system. The analysis addressed scenarios for a runaway train and derailment, tipover determinations, effectiveness of impact limiters, and mitigation of uncontrolled descents. During the development of the analysis, several safety features were identified and evaluated, resulting in the system components described in the discussions that follow.

2.3.4.4.4.1 Transporter Safety Features

The primary brake system for the transporter consists of an automatic, fail-safe tread-brake system activated by a decrease in air pressure. The application of a brake shoe against the tread of a rail wheel is called tread braking. Although most commercial railcar brake systems are activated by a decrease in air pressure, the actual tread brake application is performed with compressed air. A pneumatic control valve responds to signals sent by the locomotive operator; these signals are in the form of changes in air pressure within the main air line, or trainline. The trainline is the physical air connection from the locomotive to each railcar air brake system. Compressed air is supplied by the air compressor on the locomotive and stored in air reservoirs on each railcar. When the air pressure in the trainline is reduced, the control valve on each railcar mechanically senses the pressure drop and delivers compressed air from the railcar reservoir to the brake cylinder(s). The amount of air sent to the brakes is proportional to the drop in brake pipe pressure.

Although commercial railcar brake systems are fail-safe and proven in the railroad industry, a simpler brake system is envisioned for the transporter. This brake system uses a spring-acting/air-release brake cylinder to provide the stopping force. Powerful compression springs within the cylinder apply the required brake shoe force against the wheel, rather than using air pressure from the reservoir. With this system, compressed air is used to collapse the springs, thereby releasing the brake shoe from the transporter wheels. As air pressure decreases in the trainline, brake shoe force is proportionally applied by the springs. This brake system eliminates the need for the waste package transporter to have a compressed air reservoir and control valve, resulting in a simpler fail-safe brake system (CRWMS M&O 2000y, Section 6.1).

The secondary, or redundant, brake system for the transporter is a hydraulically applied disk brake system. This brake system comprises a disk and a caliper mounted on each axle of the transporter trucks. The calipers are hydraulically applied and spring released, which is not a fail-safe configuration. Pressure is applied to the brake calipers through a hydraulic connection to the locomotive hydraulic system.

2.3.4.4.4.2 Locomotive Safety Systems

The locomotive braking system consists of four brake systems: tread, disk, parking, and dynamic brakes. The tread and disk brakes can each stop the train independently. The parking brakes hold the train in position once stopped. The dynamic braking systems regulate the train speed as it descends the north ramp but would not be used for a complete stop.

The locomotive tread brakes are automatic, fail-safe air brakes, similar to the transporter tread brakes. Like the transporter brakes, the stopping force of the locomotive tread brakes is provided by powerful compression springs. Air pressure is used to collapse the springs, thereby releasing the brake shoes from the locomotive wheels.

The disk brakes are the backup, redundant brake system installed within the transmissions of the locomotive. The disk brake calipers are hydraulically applied and spring released, which is not a fail-safe configuration.

The locomotive parking brake is a spring-applied, manual-release disk brake. The brake consists of an independent caliper mounted to operate on the same disk as the redundant disk brake system.

Dynamic braking uses the locomotive direct-current drive motors to typically control train speeds on steep grades. The locomotive wheels are allowed to transfer mechanical power opposite the moving direction by back-driving the transmissions and drive motors. Extended-range dynamic braking may be included on the locomotive, allowing it to be used at speeds as low as 4.8 km/hr (3 mph).

2.3.4.4.4.3 Summary of Results of Safety Evaluations

From the maximum operating speed of 8 km/hr (5 mph), the transporter and the two locomotives will take 13.5 m (44.3 ft) to stop at a 13 percent brake ratio. This is a ratio, expressed as a percentage, of the total net brake shoe force applied at the wheel tread to the rated gross weight of the railcar. From the same maximum operating speed of 8 km/hr (5 mph), a full emergency brake application (60 percent brake ratio) will stop the transporter and the two locomotives in 1.58 m (5.18 ft) (CRWMS M&O 2000y, Section 6.3.3).

The analysis also showed that the impact force resulting from a collision at a maximum transporter operating speed of 8 km/hr (5 mph) is less severe than the impact from a 2-m (6.6-ft) design basis waste package drop.

Calculations of a hypothetical runaway scenario down the north ramp, with the runaway train (locomotives and transporter) reaching a maximum velocity of 31.9 m/s (71.4 mph), concluded that the equivalent stopping distances are 2,518 m (8,260 ft) at a 13 percent brake ratio, and 321 m (1,054 ft) with the emergency brake application (60 percent brake ratio) (CRWMS M&O 2000y, Section 6.3.3).

Calculations of various runaway scenarios show that the runaway velocity is above the tipover speed. Therefore, it is possible for the loaded waste package transporter to partially tip over during the defined runaway scenario (CRWMS M&O 2000y, Section 6.4.3). Impact limiters are devices attached to the waste package transporter that would help absorb impact energy in the event of a collision. In a worst-case runaway condition, the transporter would most likely tip over on the north ramp curve. Because of the size of the transporter relative to the drift diameter, the transporter would not tip over completely; the transporter would impact the wall above the deck level. Therefore, impact limiters installed at the transporter deck level and on the ends would provide negligible impact protection for the tipover condition (CRWMS M&O 2000y, Section 6.5). The analysis suggests that additional impact protection could be provided by incorporating an energy-absorbing layer on the inside of the radiological shield of the transporter (CRWMS M&O 2000y, Section 6.5), although it would increase the overall size of the shielding. Such an energy-absorbing layer, which would present design and operational complexities, will be evaluated further.

An analysis for derailment showed that a wheel-climb derailment was only possible with severely worn rails under full runaway conditions (CRWMS M&O 2000y, Section 7). This condition would be prevented by inspection and replacement of the rails before they become severely worn. However, other types of derailment, such as those that could result from rail failure, would still be possible. Periodic inspection and maintenance would help prevent rail failure occurrences. Following derailment, the transporter would hit the wall at the north ramp curve on a course almost parallel to the wall. For this lateral impact, the impact protection from an impact limiter installed on the front of the transporter deck would be negligible because the impact limiter would not make contact with the drift wall. The top of the transporter would make contact with the drift wall instead (CRWMS M&O 2000y, Section 6.5).

To limit the potential for a runaway condition, additional controls (i.e., magnetic track brakes and car retarders) are being evaluated to mitigate an uncontrolled descent. In addition, automatic controls are being evaluated. The magnetic track brakes can provide railcar deceleration rates on the order of 2.46 to 3.58 m/s^2 (8.1 to 11.7 ft/s^2). Car retarders, when used, are incorporated into the rail system. The major types of retarders used are the wheel clamp and the hydraulic piston-type retarders. The former produces friction by clamping to both sides of each rail wheel, while the latter are passive energy absorption systems similar in function to a shock absorber.

For the safety classification of structures, systems, and components, an event sequence impact is considered in the design if its estimated frequency of occurrence is 1×10^{-6}/yr or greater. The safety analysis concluded that the occurrence of a runaway transporter event could be reduced to a frequency less than 1×10^{-6}/yr. Therefore, a runaway event could be screened out as a beyond Category 1 or Category 2 event sequence. The event frequency is reduced to less than 1×10^{-6}/yr by enhancing the design features of the brake control and communications system.

A primary reason for enhancing the onboard systems is to reduce the reliance on human operators to respond properly to a runaway event. These design features include an electronic interlock that prevents the operator from starting the train down the north ramp without having dynamic brakes engaged; an alarm to alert the operators when the speed of descent is too high; and an automatic application of tread brakes to maintain speed within the normal operating range (CRWMS M&O 2000y, Section 7.1). These design features imply greater reliance on the brake control and communications system, thus creating the need for redundancy and diversity among brake systems. Additionally, a reliability analysis was performed on a conceptual rail-mounted speed retarder system. Such a system is based on proven technology and appears to be promising as a different means of controlling the train's speed of descent that is redundant with the onboard brake systems (CRWMS M&O 2000y, Sections 6.7 and 6.8.3).

2.3.4.5 Waste Package Emplacement

As described in the previous section, waste package emplacement is a remotely controlled operation directed from a control center at the surface facilities. The environment inside the emplacement drifts, due to high temperatures and radiation, makes it unsafe for a person to enter the drift when waste packages are present. If equipment recovery or drift maintenance becomes necessary, human entry would be possible only after all the waste packages have been temporarily relocated to a different drift and after increased ventilation has lowered the ambient temperature of the drift to below 50°C (122°F).

Emplacement of waste packages for the 70,000-MTHM base case has been estimated to take a total of 24 years (YMP 2000a, Table 3-2). The period of time for emplacement of the 97,000-MTHM full inventory case has not been estimated, and there are different emplacement options. The ultimate emplacement duration will depend on what strategy is adopted for codisposal of DOE spent nuclear fuel and high-level radioactive waste. The maximum emplacement rate is expected to be 605 waste packages per year (CRWMS M&O 2000ac, Section 2.2.2.4); therefore, the capability is available to emplace 97,000 MTHM of full inventory waste in less than 30 years.

Tables 2-15 through 2-19 summarize preliminary engineering specifications for waste package transportation and emplacement system components.

Table 2-15. Summary of Waste Emplacement Track Specifications

Item	Detail
Rail type	AISC Standard Crane Rail of 320 standard BHN
Rail weight	66.9 kg/m (135 lb/yd)
Track gauge	2.95 m
Rail support beam	Two W8x67
Transverse rail and pallet support beam	One W12x65 per 1,500 mm spacing
Longitudinal pallet support beam	Three W12x65
Guide beam	Two W6
Stiffener bracket	Not specified
Design load	400 metric tons
Turn radius of curvature • Emplacement drift turnouts • North ramp and main drifts	 20 m 305 m
Maximum track grade	2.15 percent

NOTE: AISC = American Institute of Steel Construction.

Table 2-16. Summary Description of Waste Package Transporter Components

Item	Detail
Structure	Flat railcar with partial overhead enclosure
Undercarriage	3-axle rear trucks, 2-axle front trucks, 762-mm diameter wheels of 320 standard BHN
Dimensions • Height • Width • Length • Weight	 4.3 m 2.9 m 22 m 400 metric tons maximum when loaded
Waste package pallet	Alloy 22 welded plates for waste package support, Stainless Steel Type 316L stainless steel square tube supports
Shielding	Standard Steel Type A 516 steel enclosure, 171.5 mm radial wall thickness, 196.9 mm axial wall thickness; borated polyethylene layer

Table 2-17. Summary of Waste Emplacement Locomotive Specifications

Item	Detail
Unit weight	50 tons
Maximum speed	8 km/hr (5 mph)
Undercarriage	4 axle, 8 wheel
Drive	Dual electric 170 hp motors
Power	Electric 650 volt direct current
Power supply	Overhead catenary wire, pantograph, and ground rails

Table 2-18. Summary of Bounding Weights of Waste Package Transporter Components

Component	Mass (metric tons)
Loaded transporter maximum weight	400
Transporter shielding	164
Flat car	136
Largest waste package (includes design margin)	85
Bed plate	9
Waste package pallet	3

Table 2-19. Design Basis Summary of Waste Package Transporter Performance

Item	Detail
Waste emplacement period	24 years
Operational period during which retrieval could be initiated	Any time from the start of emplacement up to approximately 100 years, with the potential for retrieval to start during an extended operational period of 300 years
Duration of retrieval planning and operations	Up to 34 years
Number of waste packages	11,750
Maximum emplacement rate of waste packages	605 per year
Maximum gantry speed	2.7 km/hr
Transporter braking • Deceleration rate • Stopping distance - 8 km/hr speed at 13% braking - 8 km/hr speed at 60% braking	 2.46 to 3.58 m/s^2 13.2 m 1.58 m
Radiation dose limit on transporter surface	100 mrem/hr
Maximum lifting height of waste package	1 m

NOTE: All numbers are for the base case waste package inventory (CRWMS M&O 2000h, Table 1).

2.3.4.5.1 Emplacement Gantry

The gantry designed to handle the waste package and pallet is called the bottom/side lift gantry (CRWMS M&O 2000z). The gantry illustrated in Figure 2-55 rides on 135 lb/yd rails with a rail centerline to rail centerline distance of 2.95 m (8.2 ft). The gantry has been designed so it will not drop a waste package or become inoperable as a result of a Category 1 earthquake event.

The gantry is designed to operate in the high-temperature, high-radiation environment inside the emplacement drifts. Its operation only involves moving forward and backward on the rails and moving the lifting arms up and down. This simple operation reduces the complexity of the gantry's mechanical, electrical, and control systems, which makes the gantry inherently reliable. The incorporation of high-quality hardware and software components further enhances control system reliability. These components mainly include redundant programmable control computers, instruments, and communications equipment. Fault-tolerant operation is ensured by physically separating the redundant components, providing backup electrical power and data communication systems, and employing diverse technologies that will not be susceptible to similar failures from a single cause. Shielded and insulated cabinets protect the heat- and radiation-sensitive instruments, and solid-state air conditioning units regulate the temperature. Built-in fire detection would automatically activate fire suppression systems if an onboard fire is detected (CRWMS M&O 2000z, Section 6.6.6). At this point, the gantry would be retrieved from the drift for repairs.

The primary source of electrical power for the gantry is an electrified third rail (conductor bar) system. The vehicle would have redundant power pickup mechanisms to ensure a reliable and continuous connection to the source of power. The gantry would have an emergency backup power system (onboard rechargeable storage batteries) with enough power to lower and release the load and return to the drift entrance. The locomotion system would have four independent direct-current drive motors, with one motor at each of the wheel assemblies. The maximum operating speed that the gantry can travel when carrying an 88-metric ton load waste package and pallet is limited to 2.7 km/hr (150 ft/min or 1.7 mph). The gantry would have independent fail-safe braking systems (a primary system and an emergency system). In the event of a power or communication loss, or of a vehicle control system malfunction, the braking systems would engage and bring the gantry to a stop (CRWMS M&O 2000z, Section 6.6.2). At this point, the gantry would be retrieved from the drift for repair.

The video system of the gantry would provide operators at the remote control center with real-time visual information about the operating environment and vehicle performance. This system would consist of several onboard high-resolution, articulated, closed-circuit television cameras and a series of high-intensity lights. Thermal and radiological sensing instruments would provide the remote control center with real-time status on these two environmental conditions in the emplacement drift (CRWMS M&O 2000z, Section 6.6.5).

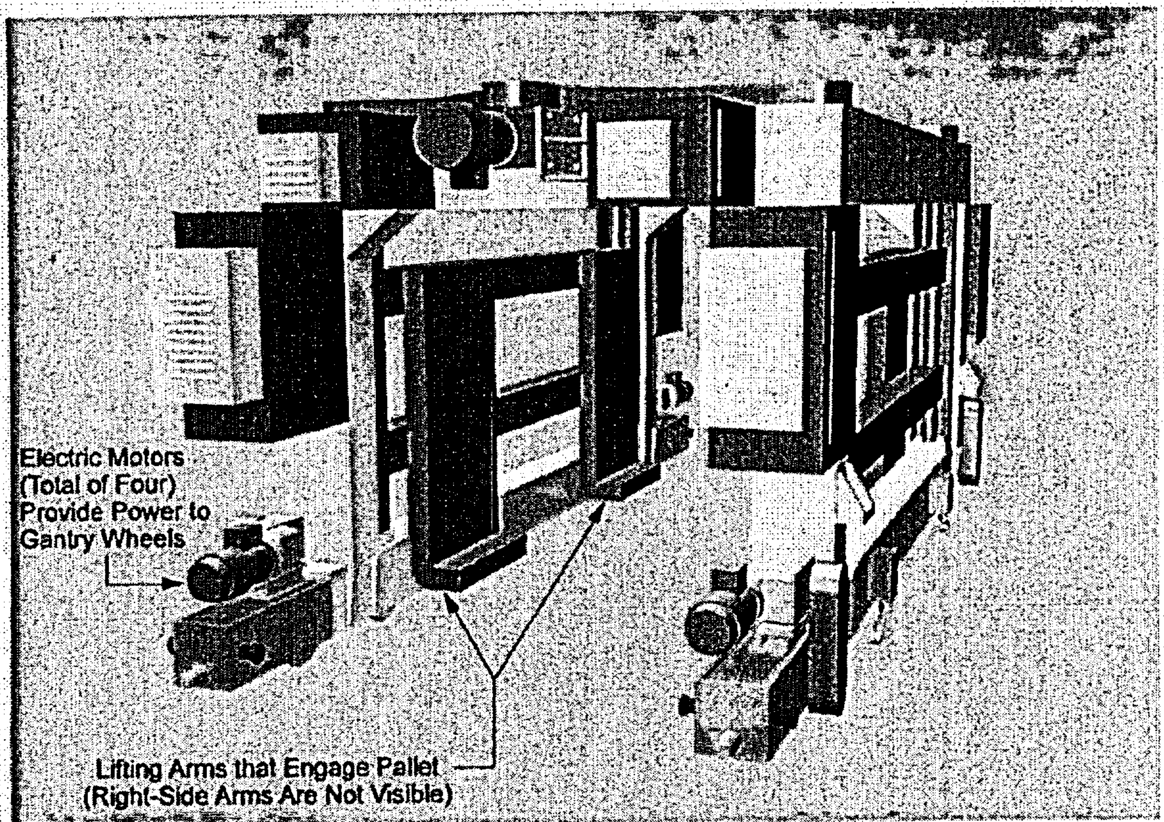


Figure 2-55. Bottom/Side Lift Emplacement Gantry—Perspective View
The emplacement gantry has four lifting arms that travel only in the vertical directions. Their vertical displacement is limited to one meter. Their sole function is to pick up the loaded pallet and gently place it on the drift floor. The gantry would be electric and remotely operated and equipped with data gathering and transmitting instrumentation, control computers, high-resolution television cameras and lights, and fire detection and suppression systems. Source: Modified from CRWMS M&O 2000z, Figures 7 through 9.

2.3.4.5.2 Gantry Operation

As discussed in Section 2.3.4.4, the emplacement gantry would be inside the drift when the loaded transporter is docked at the drift entrance for delivery of a waste package.

The rail in the emplacement drifts upon which the gantry would travel is not continuous from the east to the west end. A ventilation access (raise), which connects the drift to the exhaust main below, interrupts the gantry rail. The raise is located approximately at the midpoint of most drifts (Figure 2-44). Because the gantry cannot lift a pallet and waste package over a previously emplaced waste package, emplacement in each half of the drift starts at the raise, where the rail ends. Emplacement progresses from the raise toward the drift entrance until that half of the drift is full.

Before moving the gantry, the vertical positions of the lifting arms (Figure 2-55) would be adjusted so that their horizontal extensions can slip underneath the projecting parts of the pallet structure as the gantry moves over the unit. The gantry would then raise its arms to engage the pallet structure such that the pallet and its accompanying waste package are lifted vertically off the transporter (or off the drift invert). During normal operations, the gantry would lift a waste package and pallet unit approximately 20 cm (8 in.) off the floor. The gantry would keep the waste package and pallet at that elevation when moving. After raising the loaded pallet to the desired elevation, the gantry would move the waste package to its emplacement location in the drift. It would lower the waste package and pallet until the pallet rests directly on the drift invert. The gantry would then move slowly away from the emplaced waste package until the arms are clear. Returning to a normal speed, the gantry would proceed to the drift entrance to await the arrival of another loaded transporter (CRWMS M&O 2000z, Section 6.1).

The bottom/side lift gantry is designed to handle a wide range of waste package sizes without having to adjust the horizontal spacing of the arms. Lack of a need for such adjustments reduces the complexity of mechanical components, thereby increasing their reliability. The lifting arms never contact the waste package, thus eliminating the possibility of waste package damage by contact. This gantry concept also allows waste packages to be emplaced 10 cm (4 in.) apart, end to end (CRWMS M&O 2000z, Sections 4.2.8 and 6.1).

The emplacement gantry is designed to operate within the clearances inside the 5.5-m (18-ft) diameter emplacement drifts. Figure 2-56 shows a perspective view of the gantry inside the emplacement drift.

The emplacement gantry would not be left inside an emplacement drift for extended idle periods because of the potential detrimental effects of heat and radiation on the gantry sensors and instruments. An emplacement gantry carrier, which is a

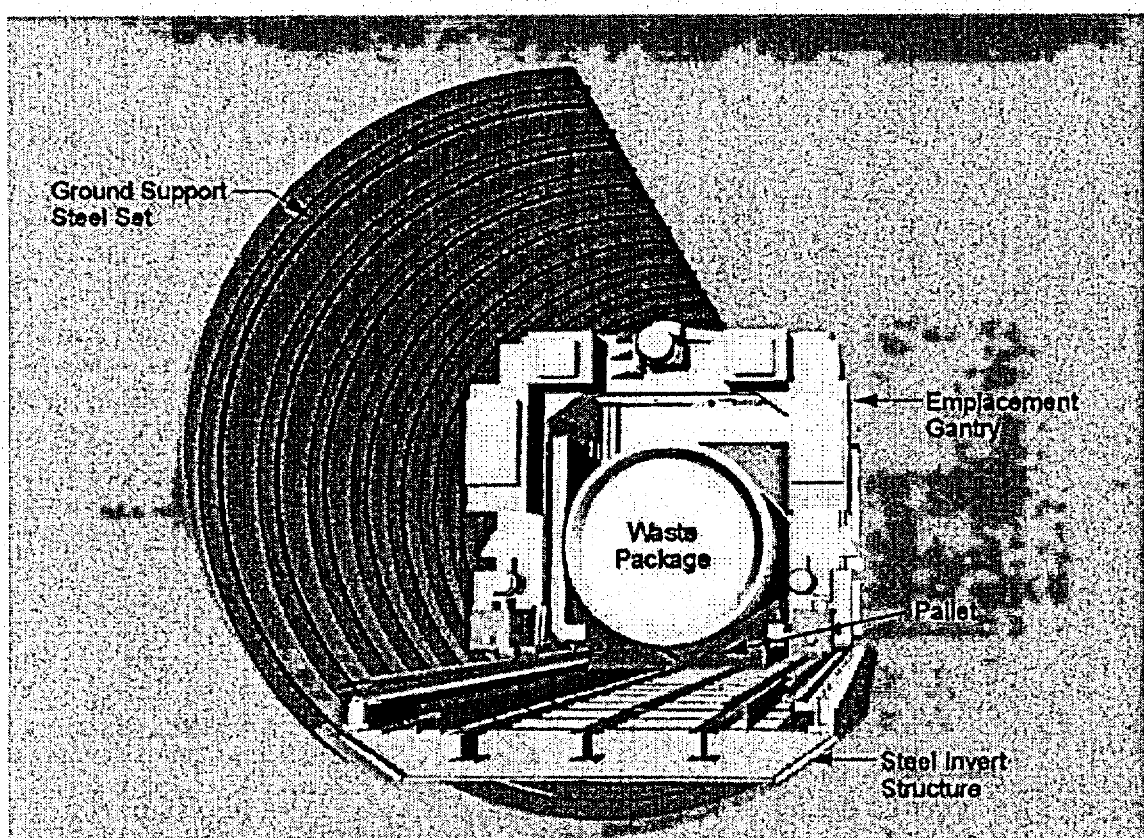


Figure 2-56. Bottom/Side Lift Emplacement Gantry—End View within Emplacement Drift
This figure illustrates the emplacement gantry transporting a loaded pallet inside an emplacement drift. Also illustrated is the steel invert section that constitutes the "floor" or emplacement surface of the drift. Relative size of the gantry with respect to the emplacement drift diameter can also be appreciated in this figure. Source: Adapted from CRWMS M&O 2000z, Figure 10.

flatbed railcar with on-deck rail tracks that mate the drift tracks, moves the gantry from drift to drift, or to a staging area.

2.3.4.6 Retrieval

2.3.4.6.1 Requirements for Retrievability

Retrievability at a high-level radioactive waste repository is a requirement mandated by the NWPA (42 U.S.C. 10101 et seq.). The NWPA, as amended in 1987, provides the reasons for which retrievability may be exercised. Section 122 of the NWPA (42 U.S.C. 10142) states that any repository to be approved as a result of the NWPA shall be designed and constructed to permit the retrieval of any or all spent nuclear fuel placed in such a repository. It also states that such retrieval should take place during an appropriate period of operation of the facility. Retrievability is justified under the NWPA for reasons that include public health and safety, environmental concerns, and recovery of the economically valuable contents of spent nuclear fuel.

The NRC regulation, 10 CFR Part 63 (66 FR 55732), establishes a minimum period during which retrieval must be possible. Waste must be retrievable on a reasonable schedule, starting anytime up to 50 years after the start of emplacement. The DOE has developed a schedule that would permit retrieval in about the same time taken for construction of the Geologic Repository Operations Area and the emplacement of wastes. The design described in this report allows for a preemplacement construction period of 5 years and a period of repository operations and performance confirmation testing of 50 years, including emplacement. It should be noted that the performance confirmation testing period started during the site characterization and would continue until closure of the repository. Retrieval, if needed, would require up to 34 additional years, which includes 10 years for planning, engineering, and procurement and 24 years for retrieval (CRWMS M&O 1998f).

The regulatory requirements for retrievability help define three basic criteria for the design of the subsurface facilities.

Robust Infrastructure—The preclosure period for the repository would vary, depending on decisions and licensing issues that would evolve with time. Current estimates for closure include scenarios for no retrieval action, exercise of the retrieval option, and an extended preclosure monitoring period.

- If no retrieval is deemed necessary, closure activities could be initiated as early as 60 years after waste emplacement begins. This period includes 50 years for emplacement and monitoring, and 10 years for decommissioning, sealing, and closure.

- If the retrieval option is exercised, and assuming that all waste packages are retrieved, the preclosure period increases to 99 years (rounded to 100). This includes preemplacement construction, emplacement, monitoring, planning, retrieval, decommissioning, sealing, and closure.

- The requirements documents reflect criteria for an extended monitoring of the repository that would extend the preclosure period to as long as 300 years after the last waste package is emplaced. The decision to close the repository would be made by future generations, based on considerations that cannot be anticipated at this time, such as the availability of future treatment technology, the reuse of the nuclear fuel, or the realization of a better disposal alternative.

Because the potential repository may remain open for as long as 300 years, its structures must last that long with manageable maintenance in an environment with high radiation and high temperature. This requirement applies to excavations, ground support, permanent mechanical components, transportation, ventilation, and other utilities.

Selective Retrieval—Maintaining the ability to selectively retrieve emplaced waste packages affects the basic repository layout, mechanical equipment, and infrastructure (to allow performance of emplacement functions in reverse order), as well as the flexibility of access to individual waste packages. Section 2.3.4.6.3 discusses the operation of retrieval equipment to selectively retrieve a single waste package.

Alternate Ingress and Egress Routes— Emplacement drift ground control deterioration and rockfall, progressive or sudden, could block access and ventilation to a particular location of the repository. To maintain the capability to retrieve any spent nuclear fuel from the repository, dual or redundant drift access is important. The design and operating mode described in this report has the ventilation raise coming up in the center of each drift, so the emplacement rails are not continuous through the drift. This does not allow full access to all parts of the drift from either end without temporary modifications. Design improvements are being considered to maintain rail access across the raise, such as a lower-profile raise collar or a ventilation exhaust main above the plane of the repository emplacement drifts, which would allow a different raise configuration. Temporary capping or other means to span the raise if access is needed across it are also being considered.

2.3.4.6.2 Drift Conditions during Retrieval and Impact on Retrievability

After the emplacement drifts have been loaded with waste packages, ventilation would continue throughout the preclosure period. With a sustained ventilation flow rate of 15 m^3/s (530 ft^3/s), air temperatures inside the drift would remain at or below a peak temperature of approximately 60°C (140°F) during the first 100 years (CRWMS M&O 2000x). An increase in the ventilation rate would lower the drift air temperatures below 50°C (120°F), low enough for the monitoring and retrieval gantries to operate reliably in the drift environment. Radiation levels would be too high for unprotected personnel access (CRWMS M&O 2000ao, Section 6.7). In cases where human incursion may be necessary, some or all of the waste packages would probably have to be removed, and extensive preparations would have to be made for radiation control.

Other potential drift conditions would affect retrievability, including:

- The structural integrity of the support pallets
- The conditions of the invert structure and railing
- A large block rockfall
- Conditions related to the containment integrity of the waste packages
- Any major deterioration and collapse of the ground support and drift walls.

The unlikely presence of a potentially breached waste package would require case-specific information and planning before any retrieval attempt, as would any detected adverse condition in the emplacement drift that would force a deviation from normal operating conditions.

2.3.4.6.3 Normal Retrieval Procedure and Equipment

The emplacement drifts would be monitored periodically, with an expected inspection frequency of 10 years. A remotely operated inspection gantry would be used during those inspections. Such monitoring would provide data for a database on drift conditions that would be used to plan maintenance activities. If it were decided that emplaced waste would be retrieved from the repository, this database and additional case-specific monitoring would be available for planning of retrieval activities. If drift conditions were normal, retrieval operations would be executed using the same waste package emplacement equipment in reverse sequence as that used for emplacement. To maintain air temperatures below 50°C (120°F) during retrieval, the ventilation flow rates may have to be adjusted several weeks before the drift incursion.

The normal retrieval sequence of operations is illustrated in Figure 2-57 (CRWMS M&O 2000aa). This sequence has been simplified to show key steps in the process. A more detailed description of the normal retrieval sequence is:

1. The locomotive and gantry carrier travel to the emplacement drift.

2. The emplacement drift isolation doors are opened and the gantry carrier engages the emplacement drift dock.

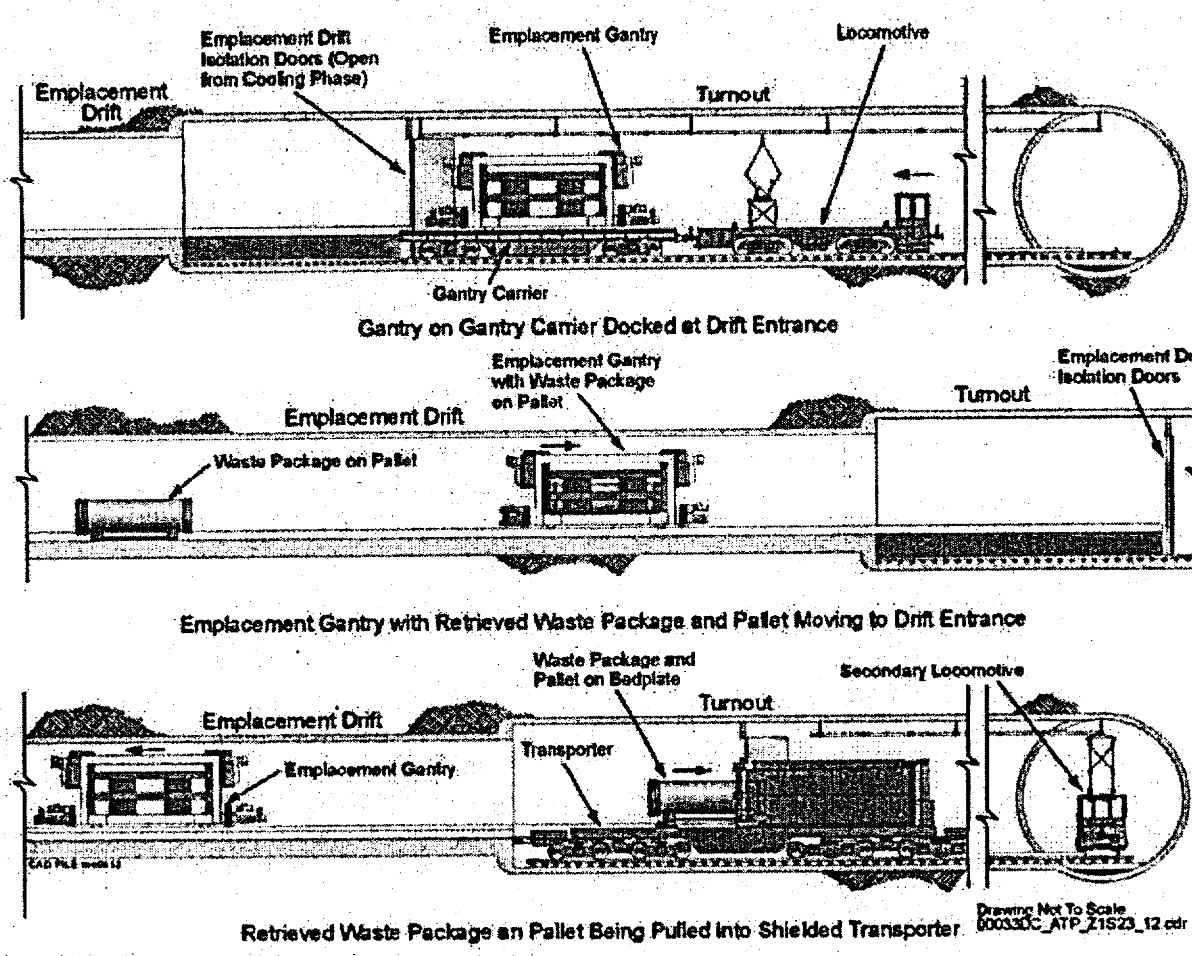


Figure 2-57. Equipment and Sequence of Operations for Normal Retrieval
The normal retrieval process for waste packages uses the same equipment and techniques utilized for emplacement but in the reverse order. The emplacement gantry would be mobilized to the drift where the waste package to be retrieved is located. The gantry would be maneuvered by remote controls over the waste package; the gantry would pick up the pallet with the waste package on it and deliver it to the transporter for transportation to the surface. Source: CRWMS M&O 2000aa, Section 6.

3. The gantry moves off the gantry carrier and the carrier is removed from the drift turnout.

4. The drift isolation doors are closed, and the gantry moves to the location of the first waste package.

5. The gantry picks up the pallet and waste package and moves back to the emplacement drift dock.

6. The transport locomotives and empty waste package transporter move to the main drift adjacent to the emplacement drift. The controls are turned over to remote operation, and the operators leave the locomotives and move to a designated protected area.

7. A secondary locomotive decouples ahead of the drift entrance. The primary locomotive and waste package transporter move from the main drift into the drift turnout.

8. The drift isolation doors open, the waste package transporter doors open, the waste package transporter docks, and the bed plate is extended onto the open deck.

9. The loaded gantry moves over the open deck of the transporter, deposits the pallet and waste package on the transporter bed plate, and moves back into the drift.

10. The pallet and waste package are pulled into the shielded compartment of the transporter via the bed plate, which is supported on rollers.

11. The primary locomotive moves the waste package transporter away from the edge of the emplacement drift, and the transporter doors and drift doors close.

12. The primary locomotive and waste package transporter move from the turnout into the main drift.

13. The secondary locomotive couples to the rear of the waste package transporter, the operators board the locomotives, and the train moves to the surface. (Steps 5 through 13 are repeated for each waste package to be retrieved from the drift.)

14. The locomotive moves the gantry carrier from the main drift to the emplacement drift turnout.

15. The drift doors are opened and the gantry carrier engages the emplacement drift dock.

16. The gantry is moved onto the gantry carrier, the locomotive pulls the carrier away from the emplacement drift dock, and the drift isolation doors close.

17. The locomotive and the gantry carrier move from the turnout to the main drift and return to a standby location.

If the waste package to be retrieved were not readily available from the drift entrance, all waste packages in front of it would have to be temporarily relocated to a standby drift or to another emplacement drift, following a sequence of events similar to that described above. If a condition preventing normal retrieval is encountered, then the sequence is interrupted, and a contingency plan would be initiated to mitigate the problem. Retrieved waste packages that are in good condition would be taken to the surface and staged in a dedicated area within the surface facilities complex (see the Potential Onsite Inventory Area in Figure 2-16). In the unlikely event that waste packages were damaged, or if waste packages were suspected of being damaged, they would be taken to the surface facilities for detailed inspection, repairs, or repackaging.

The locomotives, gantry, gantry carrier, and waste package transporter used for normal retrieval are the same as those described in Section 2.3.4.4. Therefore, their characteristics and preliminary engineering specifications are not repeated in this section.

2.3.4.6.4 Off-Normal Retrieval Procedures and Equipment

When the equipment and operating sequence described above cannot be used, retrieval conditions are considered off-normal. Off-normal retrieval conditions, most likely due to drift wall deterioration and the resulting blockage of the railing system, prevent the use of the waste package gantry. Under off-normal retrieval conditions, several additional operations would be added to the retrieval sequence. These additional steps would most likely include the following:

- Monitoring and detailed characterization of damage
- Development of a case-specific contingency plan
- Cleanup and removal of debris
- Stabilization of the drift (including ground support repairs or replacement)
- Restoration of the tracks and other damaged structures and utilities
- Repositioning of the pallet and waste package, if necessary

- Establishment of radiation controls and other administrative controls for retrieval, as needed.

Off-normal retrieval procedures and equipment would be used commensurate with the situation. A series of event sequences that could affect normal retrieval conditions were analyzed (CRWMS M&O 2000aa, Section 6.2.2) to determine the operational procedure and type of equipment that could be used to retrieve waste packages. The two event sequences analyzed and of particular relevance to off-normal retrieval conditions in the emplacement drifts were (1) rockfall or ground support collapse onto a waste package from causes other than seismic events and (2) rockfall or ground support collapse onto a waste package caused by a seismic event (a beyond Category 1 or Category 2 event sequence earthquake). Other event sequences analyzed with respect to off-normal retrieval pertained to the mechanical failure of the gantry and the derailment of the gantry at normal speed.

The two pieces of equipment designed for removal of the gantry, the emplacement drift gantry carrier and the multipurpose hauler (Figure 2-58), operate on rollers. Steel plate would have to be installed over the drift invert to deploy this equipment. A multipurpose vehicle (Figure 2-59) can be operated from the multipurpose hauler to clear debris, emplace steel plates, and cut and remove damaged structures. This equipment would be remotely operated.

When a derailed or damaged gantry has to be removed from the drift, the emplacement drift gantry carrier can load and carry the gantry away from the emplacement drift (Figure 2-60). The multipurpose hauler can be used to pull and load the pallet and waste package onto the deck of the hauler to proceed with retrieval operations (Figure 2-58).

2.3.4.6.5 Summary of Retrievability

Waste package retrieval under normal conditions uses the same subsurface equipment and facilities as emplacement, but in reverse order. This provides a built-in capability for retrieval that can be readily implemented. Individual waste package removal for inspection, testing, and maintenance reasons is not considered retrieval; however, waste package removal for these purposes, if needed, would involve the same equipment and operational steps.

Alternative waste package retrieval equipment has been identified for off-normal conditions when normal retrieval procedures may be difficult or impossible to execute. Additionally, support equipment (i.e., equipment to remove obstacles, prepare surfaces, or install temporary ground supports) that can be used in retrieval operations under off-normal conditions has been identified. Various scenarios of off-normal retrieval have been analyzed, and conceptual use of such equipment has been demonstrated (CRWMS M&O 2000aa, Sections 6.2 and 6.3). Proof-of-principle demon-stra-tions of waste package retrieval may be conducted following the license application as another step in the series of requirements for successful construction and operation of the repository.

2.3.4.7 Decommissioning

The primary objective of the decommissioning of the subsurface facilities is the removal of any material or equipment that is not part of the permanent repository installation. Decommissioning will precede closure activities, or in some cases will be concurrent with them.

Subsurface decommissioning activities would include:

1. Dismantling structures and equipment for removal of components from the underground facilities, decontamination if necessary, and transport to the surface for additional decontamination, release, or disposal.

2. Demolition of reinforced concrete structures and steel support structures after the equipment and fixtures have been removed.

3. Transporting the removed materials and equipment to onsite or offsite disposal

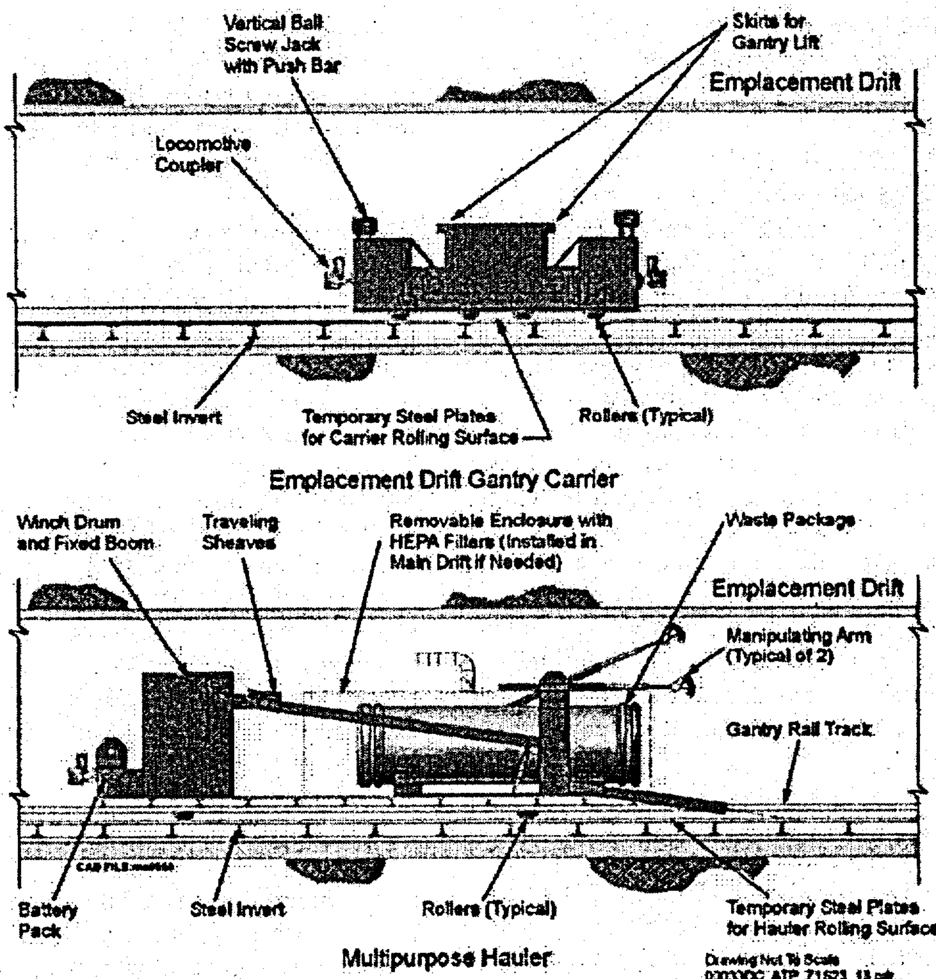


Figure 2-58. Emplacement Drift Gantry Carrier and Multipurpose Hauler
During off-normal retrieval, when emplacement drift conditions do not allow implementation of normal retrieval procedures, alternative retrieval equipment can be utilized. These consist of an assortment of equipment designed for removal of disabled gantries, removal of debris, construction of temporary structures to support retrieval, and retrieval of damaged or undamaged waste packages. HEPA = high-efficiency particulate air. Source: CRWMS M&O 2000aa, Section 6.2.

facilities, depending on type, quantity, and waste characterization, in accordance with a preapproved DOE waste management plan. These materials and equipment will include ventilation system components, railroad track, rolling stock, utilities, concrete and construction materials, instrumentation and communication equipment, pumps, pipelines, and any other materials that may be considered detrimental to repository performance after closure.

4. Identifying and removing equipment and materials for salvage. Some equipment and materials may be targeted for salvage because of their value or because their reuse is anticipated (e.g., for site restoration or other future activities at the repository surface facilities). Such equipment and materials will be protected as needed and staged at the surface facilities.

5. Placement of underground markers, monuments, or any other permanent features deemed necessary before closure activities.

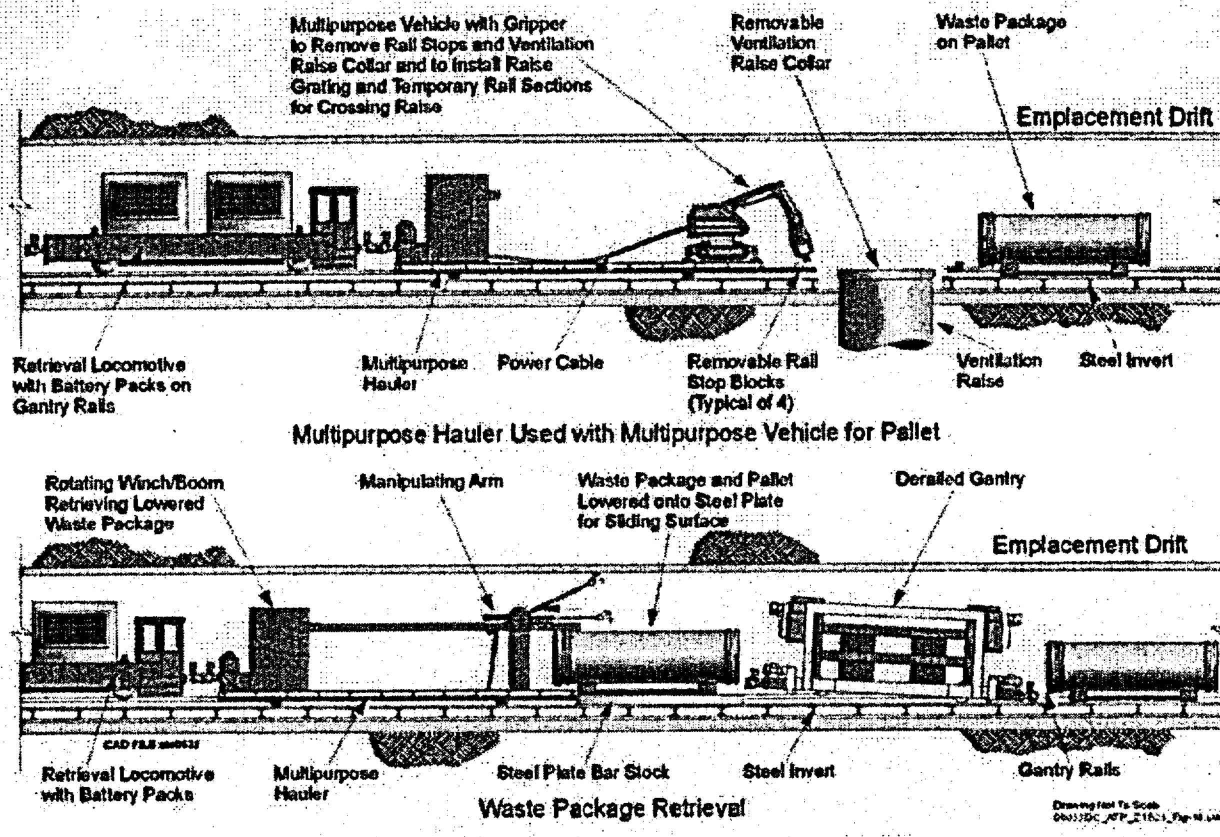


Figure 2-59. Multipurpose Hauler Used with Multipurpose Vehicle for Pallet and Waste Package Retrieval
Remotely controlled equipment can be utilized for different functions during retrieval, thus preventing personnel exposure. Source: CRWMS M&O 2000aa, Section 6.2.

6. Completion of as-built surveys of the subsurface facilities before closure.

7. General cleanup in preparation for closure activities.

It is likely that some of these activities can be performed gradually during the operational phase of the repository.

2.3.4.8 Closure and Sealing Structures

Closure of the repository subsurface facilities requires closing and sealing all openings from the surface to the underground facilities. These openings consist of the north and south access ramps, ventilation shafts, and exploratory boreholes within the repository footprint (the waste emplacement area projected vertically to the surface) and in an area extending 400 m (1,300 ft) from the footprint boundaries. If left unsealed, these openings could enhance the movement of moisture from the surface into the waste emplacement area, and the larger openings, such as the ramps and shafts, could allow unauthorized human intrusion. Openings connecting the waste emplacement area to the surface could also serve as conduits for airborne radioactive contamination to migrate into the atmosphere.

Regulatory requirements imposed on the repository design to provide protection against human intrusion, as well as requirements to prevent or minimize release of contamination, create the necessity that these openings be sealed during the closure phase of the repository.

Tables 2-20 and 2-21 summarize the preliminary engineering specifications for closure and sealing components.

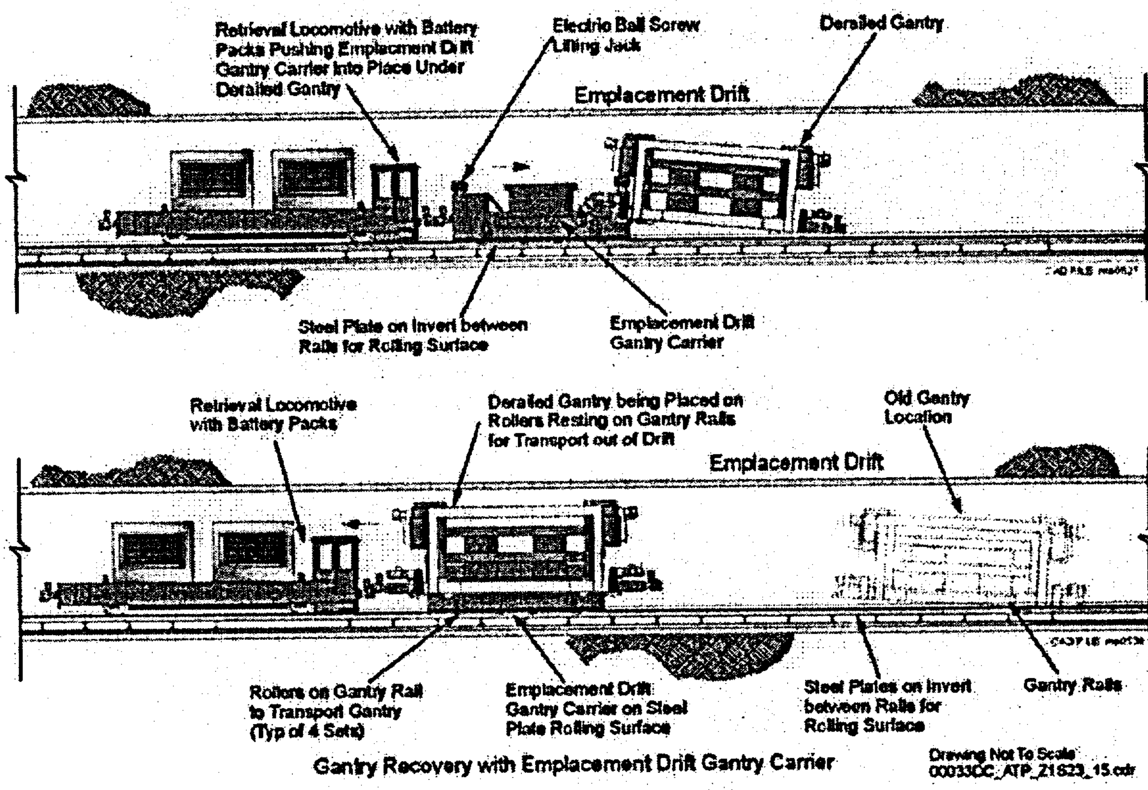


Figure 2-60. Gantry Recovery with Emplacement Drift Gantry Carrier
This figure illustrates the operation of a gantry carrier used for retrieval gantry recovery, in case of a disabled gantry. Source: CRWMS M&O 2000aa, Section 6.2.

Table 2-20. Summary of Design Basis of Closure and Sealing Components

Item	Property
Favorable material characteristics	• Low permeability • Chemical stability in thermal-hydrologic environment • Material longevity • Availability • Similar hydraulic conductivity and permeability to host rock mass
Concrete structural loading factors	• Seismic activity • Host rock stress relief • Hydraulic pressure • Backfill material lateral pressure

Table 2-21. Summary of Closure and Sealing Component Materials

Component	Material
Ramp and shaft plugs and bulkheads	Concrete
Seals and backfill	Concrete, grout, crushed tuff, bentonite, bentonite–sand mixture
Boreholes	Bentonitic grout

2.3.4.8.1 Repository Openings

Section 2.3.1 describes the locations of the north and south ramps, and Section 2.3.4.3 describes the ventilation shafts. The specific locations of the boreholes within the repository footprint, including the 400-m (1,300-ft) buffer zone around the perimeter, are not addressed in this report but are well documented in the DOE's scientific database (BSC

2001d, Attachment III). There are approximately 16 deep boreholes in the potential repository area, including a 400-m (1,300-ft) buffer zone around the potential repository perimeter.

2.3.4.8.2 Criteria for Selection of Seal Locations

The geological environment and the rock structure are central considerations in determining the specific locations where seals would be most effective. The constructibility advantages offered by one location over another are also considerations in determining seal locations. For example, seal construction in large openings is preferable in the straightaway rather than in curves or intersections because of such issues as simplicity of design, construction, equipment access, and materials placement.

Two stratigraphic nomenclature systems are used for the classification of the host rock at Yucca Mountain: a thermomechanical nomenclature and a lithostratigraphic nomenclature. The thermomechanical nomenclature is based solely on the thermal and mechanical properties of the rock. The lithostratigraphic nomenclature is based on primary geologic processes (e.g., the depositional character and assemblage of the rock) and secondary geologic processes (e.g., the degree of welding, devitrification, and vapor-phase crystallization). A thermomechanical unit typically contains several lithostratigraphic units. The thermomechanical units are defined by the thermal and mechanical characteristics of the natural rock, and they do not necessarily correspond to lithostratigraphic formation boundaries. The PTn and the TCw thermomechanical units (see Sections 1.3.2.2.2 and 2.3.4.1.3 for geologic unit descriptions) are important in defining the location of closure seals above the emplacement block. The TCw unit has a well-connected fracture system and high permeability. The PTn unit has very low permeability. These contrasting attributes can be used advantageously by locating a seal in the more permeable unit (the TCw) near its boundary with the less permeable unit (the PTn). At this location, the seal would help disperse sporadic infiltration fluxes into the more permeable matrix of the TCw formation, which otherwise would tend to create a moisture buildup in the less permeable PTn unit.

The condition of the rock immediately surrounding a potential seal location is the most important factor in deciding the locations of closure seals in the ramps and shafts, mainly for reasons of seal structural integrity and stability. Closure seal locations should be selected to avoid ground conditions resulting from geologic disturbances, such as faults and fracture zones. Therefore, developing detailed characterizations of the rock structure at the ramps and ventilation shaft locations is essential. The structural geology of the north and south ramps is well characterized, and several candidate seal locations have been identified (CRWMS M&O 2000aj, Appendices A and B). Detailed characterization of the ventilation shafts, however, would be possible only after their excavation. Therefore, detailed seal design must also be performed after shaft excavation.

Faults create instability in the rock and present potential pathways for inflow of moisture. Major faults along the north and south access ramps are already documented (CRWMS M&O 2000aj, pp. 22 and 23), and they will be avoided in selecting closure seal locations. Similar identification and evaluation of major faults will be performed for the ventilation shafts after their excavation; however, the locations of the shafts have been selected partly by avoiding documented geologic hazards, such as faults or excessively fractured rock zones.

Fractures (structural discontinuities in the rock) and joint sets (groups of fractures with similar orientation) also provide an indication of rock mass stability. Fractures directly affect the strength of the rock. The frequency and orientation of the fractures can affect the short- and long-term stability of the excavation. Five joint sets have been identified in the Exploratory Studies Facility, which includes the north and south ramps (CRWMS M&O 2000aj, Table 6). Within the repository block, the Undifferentiated Overburden and PTn thermomechanical units have the lowest density of fracturing, while the nonlithophysal portions of the other three thermomechanical units (TCw, TSw1, and TSw2) have the highest fracture density (CRWMS M&O 2000aj, p. 24).

2.3.4.8.3 Selection of Closure Seal Materials

Selection of seal materials must consider material strength and the ability to slow moisture and airflow through the repository block. Seals would be composed of different materials, individually or in combination, that will effectively discourage human intrusion and control moisture inflow.

Seal and closure system components must be mechanically, chemically, geologically, and thermally compatible with the subsurface environment. Materials that can meet these compatibility requirements include concrete, grout, crushed tuff, bentonite, and bentonite–sand mixtures.

Earthen materials have been recommended for use as backfill in ramps and shafts to prevent preferential pathways for moisture movement and discourage human intrusion. Properties considered desirable for backfill materials include low permeability, chemical stability in the thermal and hydrologic environment in the repository, material longevity, and availability. The selected backfill material should have the approximate hydraulic conductivity and permeability of the surrounding rock mass. Earthen materials analyzed for use as closure backfill include crushed tuff, clays (such as bentonite), and combinations of bentonite and sand (CRWMS M&O 2000ak). Crushed tuff would be readily available as a by-product of tunnel excavation; this excavated material is often called muck. The muck material would be screened and crushed to enhance the characteristics useful for closure backfill. It has been determined that a bentonite–sand mixture with 5 to 10 percent bentonite would result in an acceptable hydraulic conductivity range (10^{-4} to 10^{-7} cm/s) for use in closure backfill or seal designs (CRWMS M&O 2000ak, Figure 1).

Several grout types were analyzed as sealing media for rock fractures and for sealing between the rock face and the structural components of the main closure seals. These grouts included chemical, cementitious, and bentonite-based materials. Their advantages and disadvantages are listed in *Closure Seal Materials and Configuration* (CRWMS M&O 2000ak, pp. 11 to 13).

Concrete is recommended for ramp and shaft plugs and bulkheads. Concrete structures can be designed to withstand loading from seismic activity, host rock stress relief, and hydraulic pressure, and to retain backfill material.

2.3.4.8.4 Closure and Sealing Structures

Conceptual seal designs and closure structure designs have been developed from the recommended materials described in Section 2.3.4.8.3. These concepts may be expanded into preliminary designs during the license application phase; however, detailed designs of the shaft seals may not be feasible until complete geologic mapping of the excavated shafts is finished.

Repository closure operations would include placement of backfill throughout the ramps, shafts, and main drifts. Figures 2-61 and 2-62 illustrate typical backfill placement methods for the ramps and main drifts and for the shafts, respectively. The method illustrated in Figure 2-61 consists of a pneumatic conveyance system. However, more conventional placement methods are available that use standard underground equipment (e.g., load-haul-dump underground haulers). Pneumatic systems provide good placement control, but they require extensive dust control to maintain acceptable working conditions. The shaft backfill placement method illustrated in Figure 2-62 is a conventional and industry-proven practice that produces satisfactory results.

Deep boreholes would be sealed by injecting a bentonite grout from the surface opening. Concrete caps would be installed near the surface, closing all openings, to prevent or discourage human intrusion. These caps would be covered with backfill as part of the surface restoration work. Figure 2-63 illustrates a typical installation for a shaft concrete cap, as well as the water dispersion structure placed in the TCw thermomechanical unit (see Section 2.3.4.8.3).

Seals would be placed in shafts and ramps at several locations, depending on more detailed

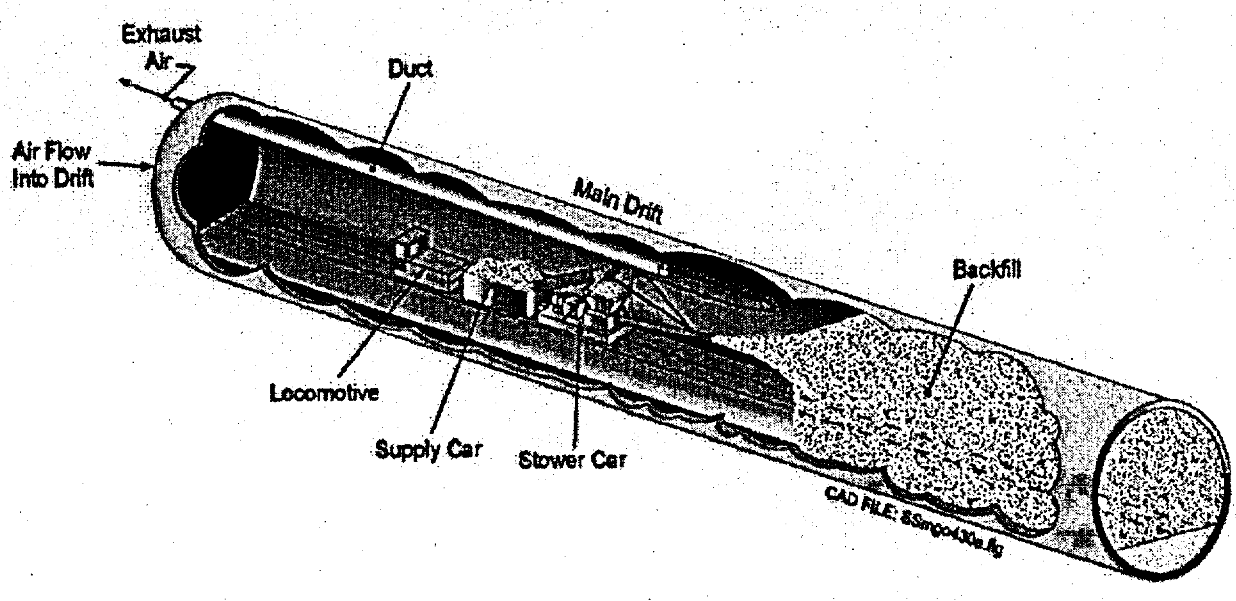


Figure 2-61. Conceptual Arrangement for Placement of Backfill in Ramps and Main Drifts
Repository closure activities include backfilling of the ramps and main drifts with granular material. This figure portrays a method by which such operation could be accomplished. Source: CRWMS M&O 2000ai, Section 6.1.2.

designs for control of moisture infiltration into the emplacement area. Seals would be located in relation to major faults and highly fractured rock zones to improve performance by attenuating sudden inflows of water and promoting slow percolation back into the rock matrix. Conceptual seal plug designs have been developed for areas of high, moderate, and low water inflow. Figure 2-64 illustrates the seal plug design concept for a high water inflow application. This design concept consists of a combination of dual concrete plugs that "sandwich" a bentonite–sand plug. The concrete plugs would provide structural support, while the bentonite–sand mixture fill would expand, filling voids and fractures. Conceptual designs for moderate and low water inflow applications would be simpler, consisting of single concrete plugs "sandwiched" by backfill material. In all these seal applications, construction of the concrete plugs would be preceded by interface grouting of the rock formation around the plug perimeter to enhance the watertightness of the seals.

2.3.5 Phased Construction—Development and Emplacement Sequence

The subsurface components of the repository would be built in phases to support waste emplacement. The current design assumes that waste packages would be delivered over a period of 24 years (YMP 2000a, Table 3-2), which allows a phased approach to completion of the emplacement drifts. This phased construction approach is described in the discussions that follow.

The construction phases for the subsurface facilities will be (1) initial construction and (2) development. Subsurface systems operations will vary during these construction phases as emplacement activities begin and continue over a period of several decades.

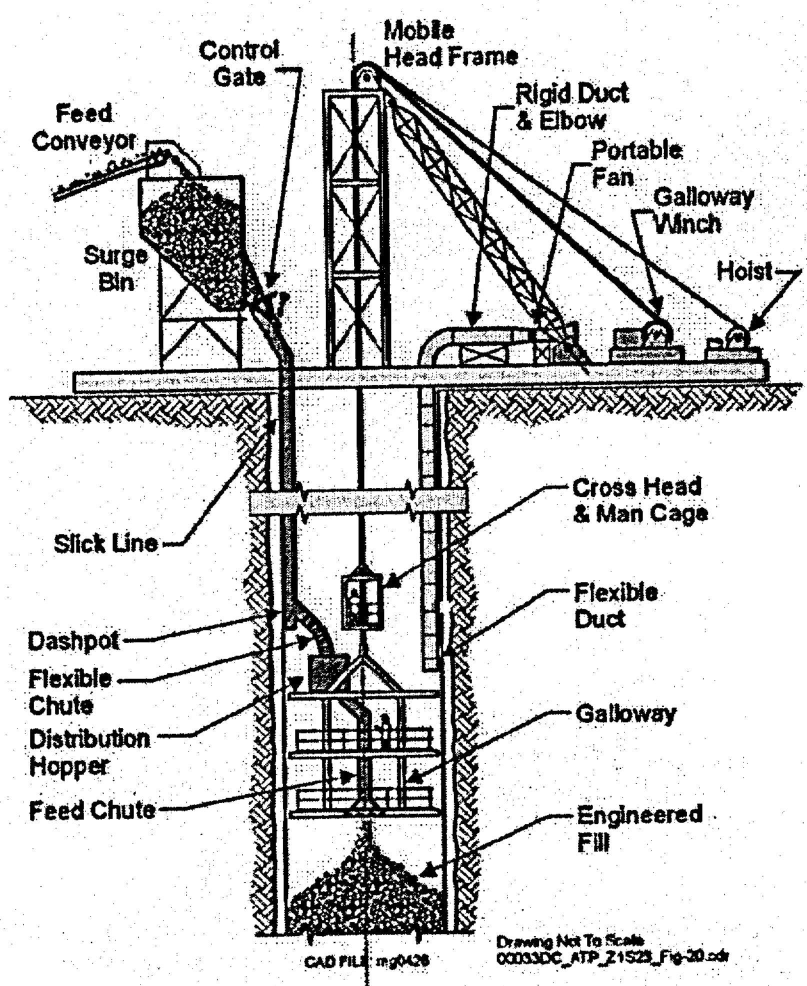


Figure 2-62. Conceptual Arrangement for Placement of Backfill in Shafts
Intake and exhaust shafts would also be backfilled with granular material as part of the repository closure activities.
Source: CRWMS M&O 2000al, Section 6.1.2.

2.3.5.1 Initial Construction

2.3.5.1.1 Facilities

In the initial construction phase, activities in the north side of the emplacement block would include excavation of the north ramp extension and excavation of the east main, north of the north ramp curve (Figure 2-65). These two drifts would be 7.62 m (25 ft) in diameter, excavated with tunnel boring machines. After excavation of the north ramp extension, one of the tunnel boring machines would continue with excavation of the north main and part of the west main. The other machine would start excavating the exhaust main from the north, heading south.

The first two drifts in the emplacement block would be excavated with another tunnel boring machine, 5.5 m (18 ft) in diameter. These two drifts, Postclosure Test Drifts #1 and #2, would be dedicated to performance confirmation testing; their construction would be identical to the construction of emplacement drifts. The performance confirmation function of these postclosure test drifts is explained in Section 2.5. After excavating these first two drifts, the smaller-diameter tunnel boring machine would excavate the first set of ten emplacement drifts, progressing from north to south.

A second 5.5-m (18-ft) diameter tunnel boring machine would start excavation of Observation

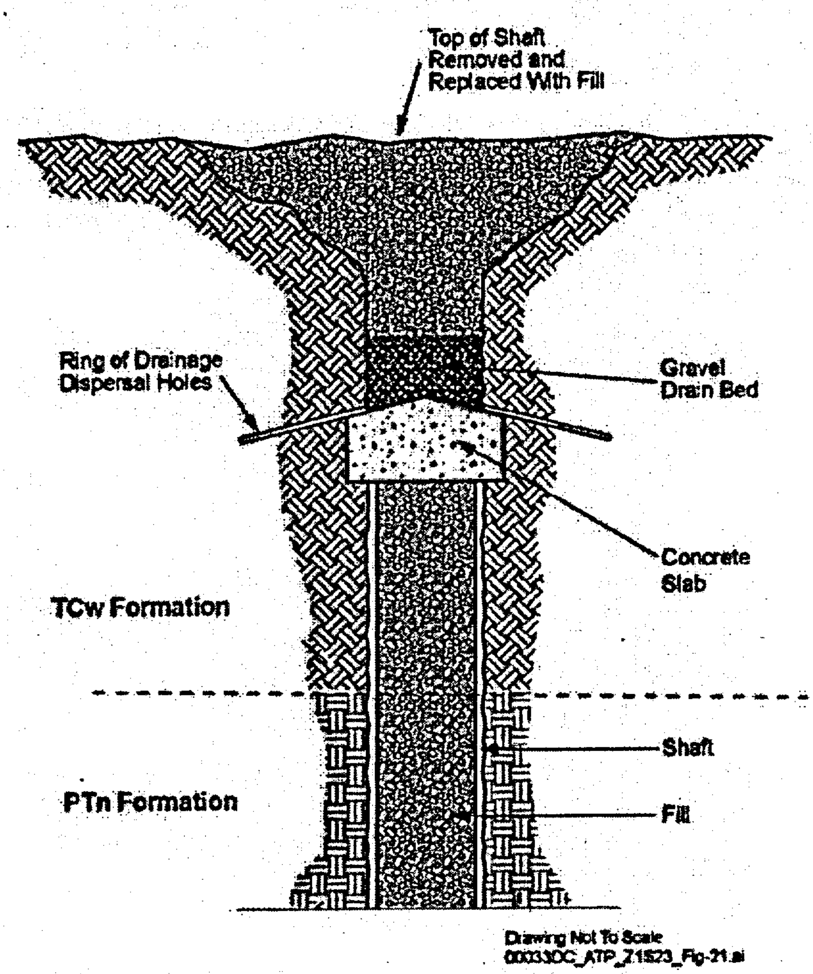


Figure 2-63. Conceptual Arrangement of Shaft Plug
Location of shaft plugs, another repository closure feature, would depend on characteristics of the geologic strata. These seals would be located in fractured rock strata with a higher permeability than the underlying strata, to promote dispersion of surface water inflows into the rock formation. Source: CRWMS M&O 2000ai, Section 6.1.1.

Drifts #1 and #2, in sequence. These two drifts are also part of the performance confirmation facilities; the functions of these drifts are explained in Section 2.5. After completing these two drifts, this tunnel boring machine would be relocated ahead of the other 5.5-m (18-ft) diameter tunnel boring machine. Then it would excavate a cross-drift for Intake Shaft #1 and another standard emplacement drift connecting the southernmost extension of the west main to the east main.

Other subsurface facilities excavated during this phase would include Intake Shaft #1, Exhaust Shaft #1, and two 7.62-m (25-ft) diameter access drifts connecting the exhaust shaft to the exhaust main. Observation Drift #3 (Figure 2-65) would also be excavated during this period by the second 5.5-m (18-ft) diameter tunnel boring machine, starting from the ECRB Cross-Drift. The intake and exhaust shafts are 8 m (26 ft) in diameter, excavated from the surface down by drill-and-blast techniques (BSC 2001d, Sections 5.2.7.2, 6.3.2.3, and 6.3.2.5).

2.3.5.1.2 Ventilation and Controls

During the initial construction period, air would be taken in from the surface through the north ramp and exhausted out to the surface through the south ramp. This section defines the global ventilation system for this phase of construction. Within the global system, local ventilation systems support

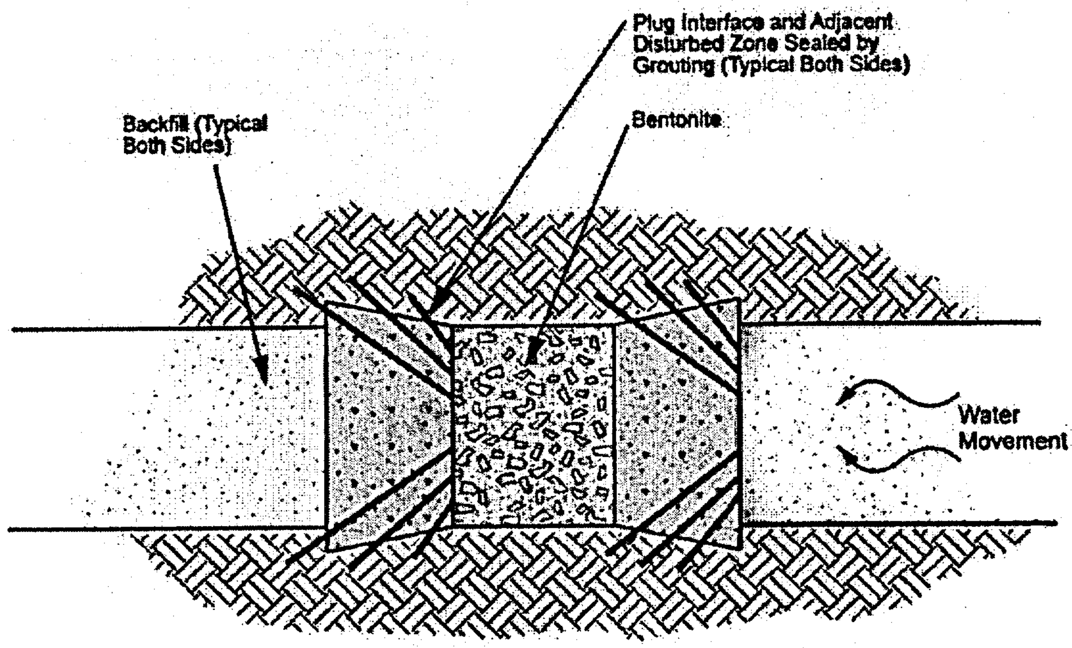


Figure 2-64. Dual Concrete Seal Plug Design Concept
Dual concrete seal plugs are one of the proposed closure and sealing features for the repository ramps. Source: CRWMS M&O 2000ak, Section 5.

each excavation deadheading. These local systems provide ventilation from the main drift through to the working face, where a mechanical excavator extends the length of a drift that is a dead-end heading. Ventilation at the excavation headings is achieved in three ways: (1) intake through a duct, exhaust through a drift; (2) intake through a drift, exhaust through a duct; and (3) intake through a duct, exhaust through another duct. Figure 2-66 illustrates the global and local ventilation design concepts as they apply to the initial construction phase. For local ventilation, the figure only illustrates the concept described in item 1 above. Local dust scrubbers remove dust from exhausted air before the air is returned to the main drift and exhausted from the underground to the surface. Each shaft excavation would be ventilated by local systems that would take fresh air in from the surface (CRWMS M&O 2000x, Section 6.3.1).

2.3.5.2 Development and Emplacement Phase I

2.3.5.2.1 Facilities Construction

As the first emplacement drifts from the initial construction phase become available for waste emplacement, transportation of waste packages to the subsurface would begin. This event marks the beginning of Phase I of development and emplacement. During Phase 1, concurrent with waste emplacement in the first drifts, construction of new emplacement drifts would progress in a southerly direction. Figure 2-67 illustrates the subsurface facilities constructed during Phase I of development and emplacement.

During Phase I, eight emplacement drifts would be excavated. Excavation of the exhaust main and the west main would continue in a southerly direction. After completing the excavation of the west main,

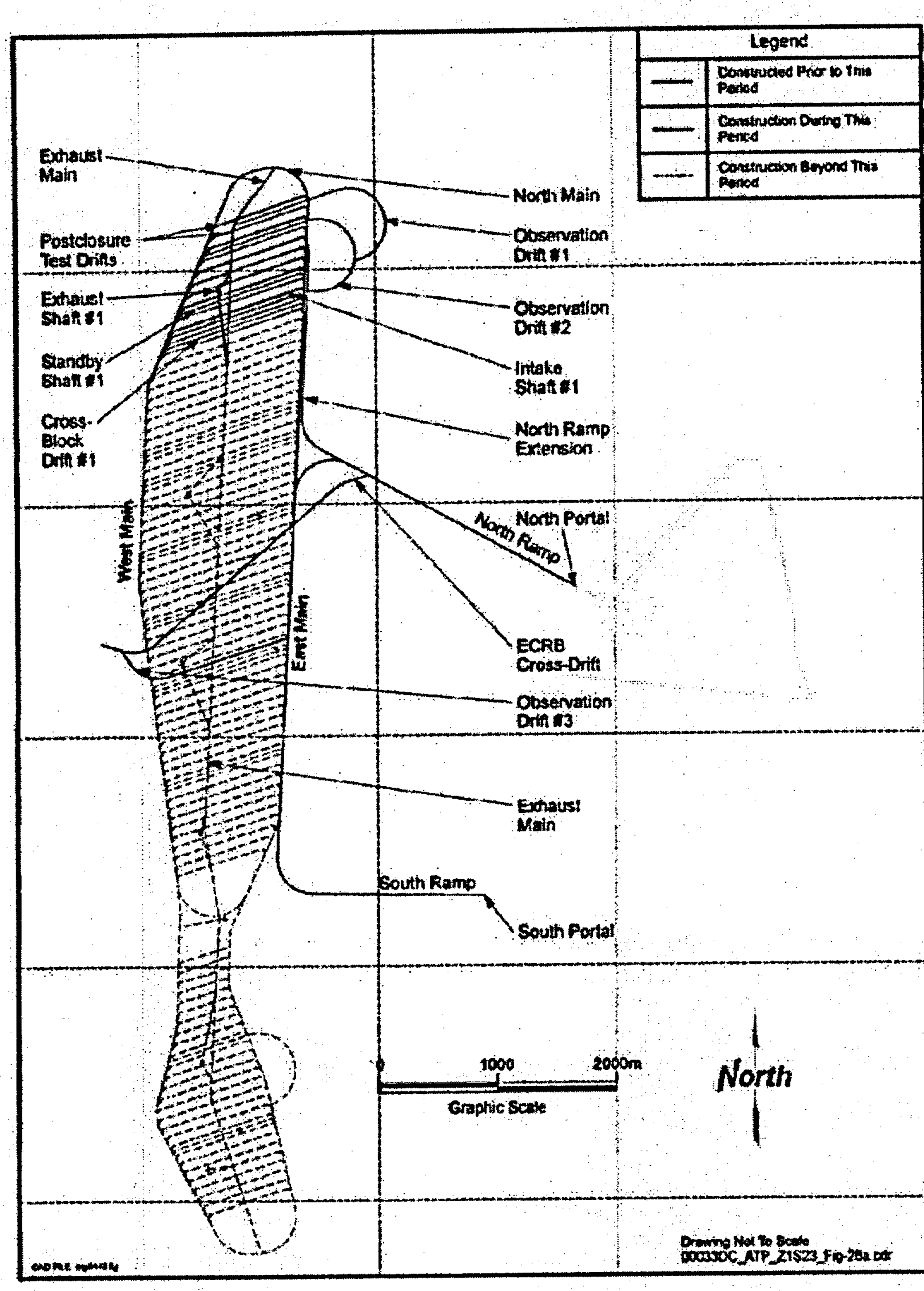


Figure 2-65. Repository Subsurface Layout—Initial Construction
Initial repository construction includes three observation drifts and ten emplacement drifts.

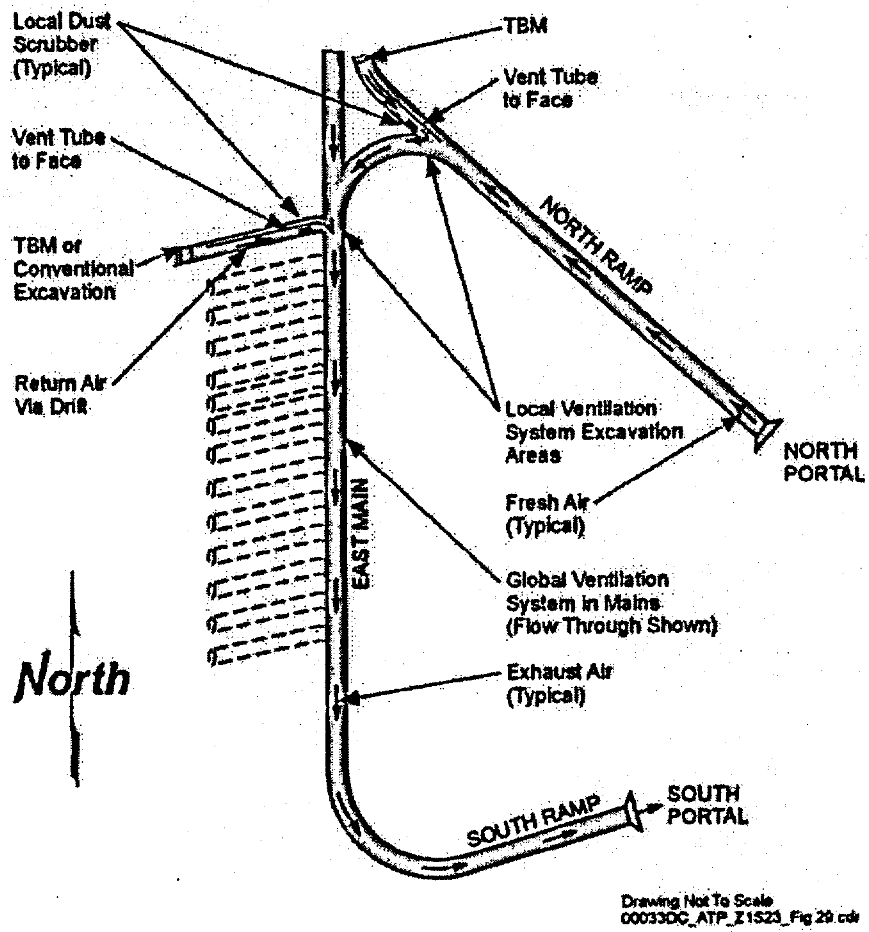


Figure 2-66. Examples of Global and Local Ventilation
Dust scrubbers would be used at excavation headings to control airborne particulates before air is returned to the main airways. TBM = tunnel boring machine. Source: Modified from CRWMS M&O 2000x, Section 6.3.

the 7.62-m (25-ft) diameter tunnel boring machine would continue excavating to open the south main to its intersection point with the existing east main. During this phase, excavation of the exhaust main by the other 7.62-m (25-ft) tunnel boring machine would continue to the south main. Other excavations during this phase include Exhaust Shaft #2 and the access drifts connecting it to the exhaust main; Intake Shaft #2 and the cross-drift connecting it to the east and west mains; and the development intake shaft.

2.3.5.2.2 Ventilation and Controls

The first ten drifts have a separate ventilation system to support waste emplacement. Air locks in the exhaust main, in the east and west mains north of the tenth drift turnouts, and on the north ramp curve would isolate the initial panel (the first ten drifts) from the new development area that starts with construction of the thirteenth drift. The emplacement area air intake would be drawn through the north ramp and exhausted through Exhaust Shaft #1. On the development side, air would be blown in through Intake Shaft #1 and exhausted through the south ramp.

2.3.5.3 Development and Emplacement Phase II

2.3.5.3.1 Facilities Construction

Figure 2-68 illustrates Phase II development and emplacement work. Facilities constructed during

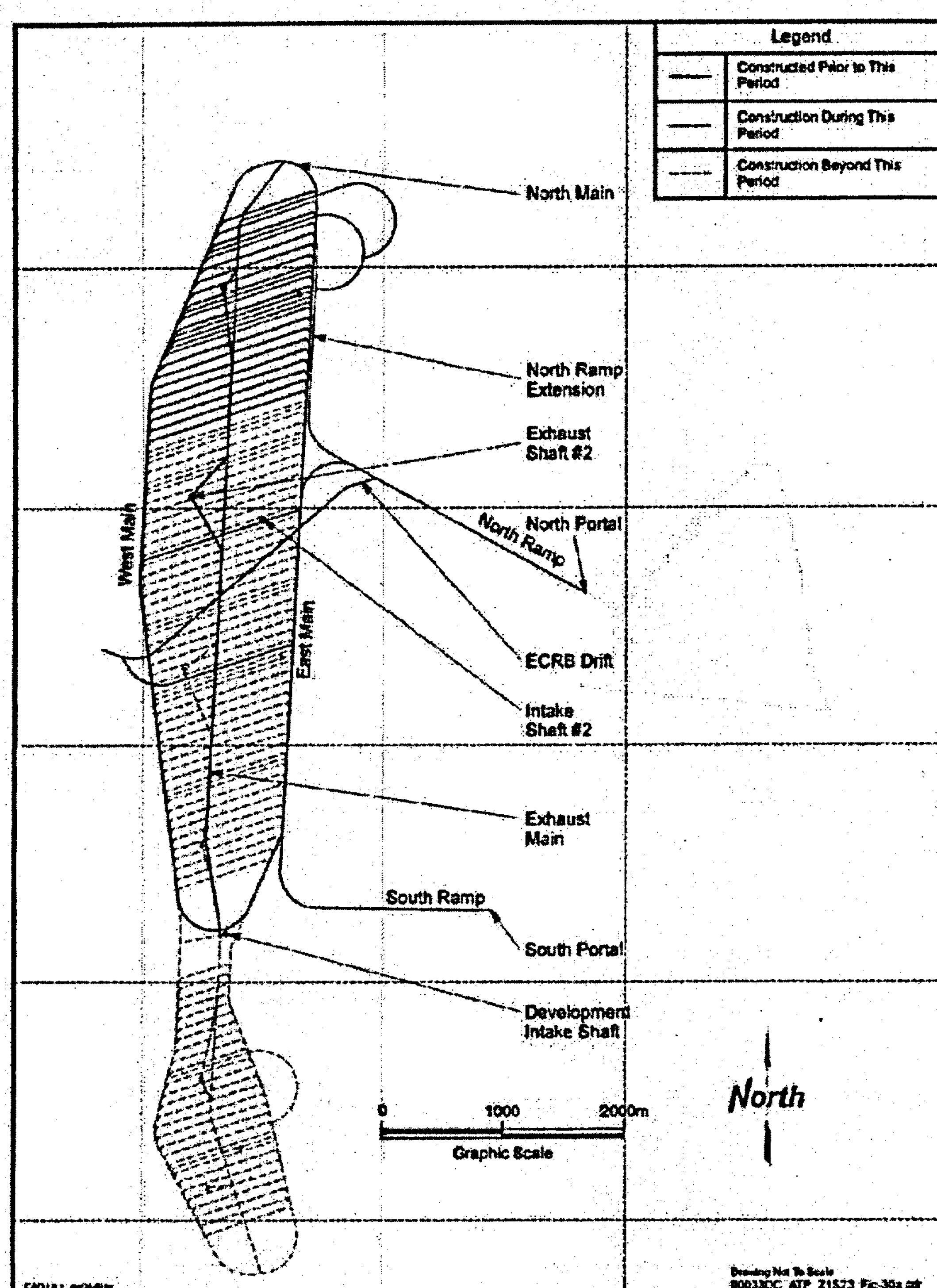


Figure 2-67. Repository Subsurface Layout—Development and Emplacement for Phase I
Phase I development activities include construction of eight emplacement drifts while waste emplacement in the first ten drifts is initiated.

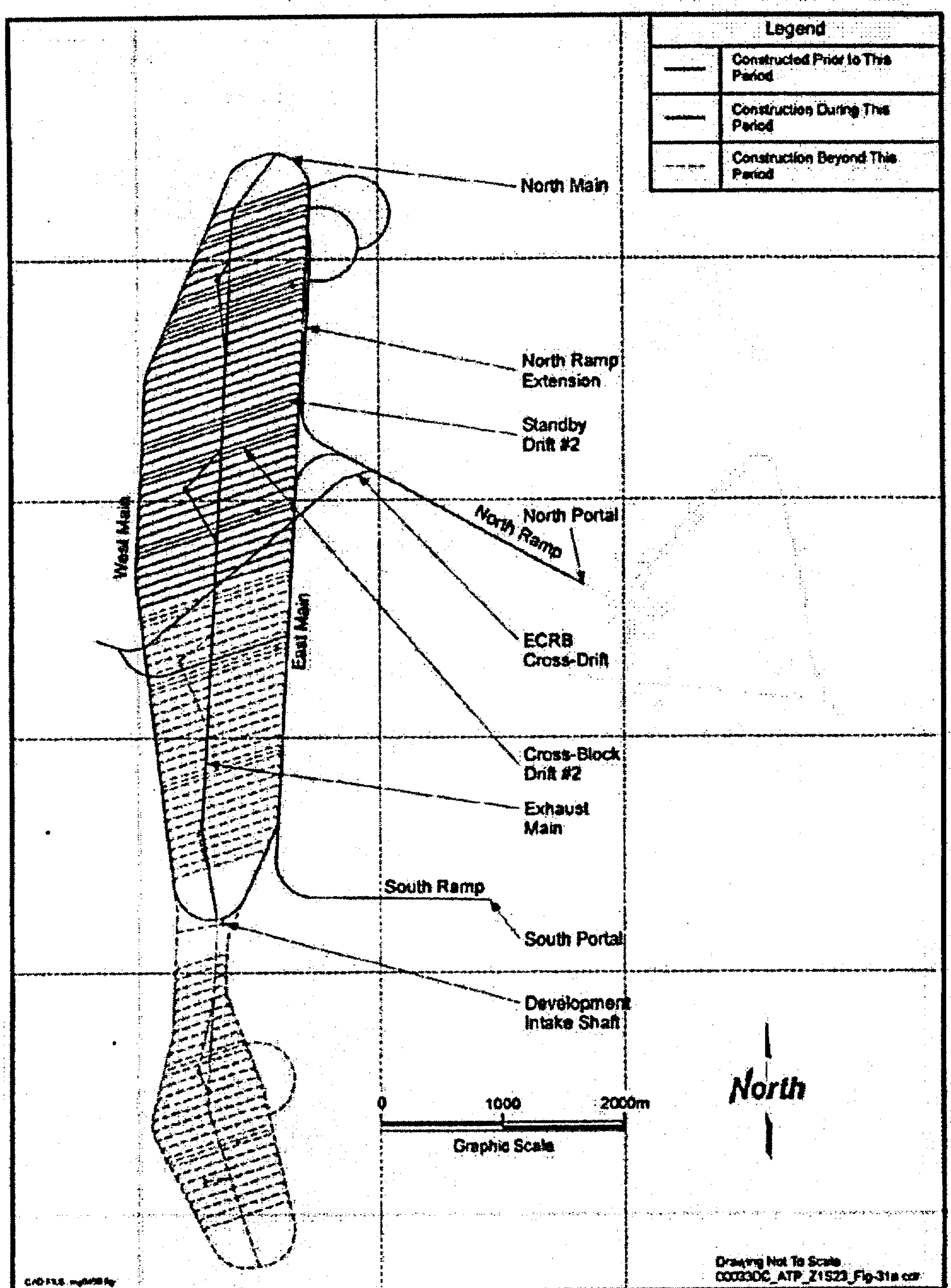


Figure 2-68. Repository Subsurface Layout—Development and Emplacement for Phase II
The second phase of development and emplacement includes the construction of 17 emplacement drifts while waste emplacement operations go on in the previously built 18 drifts.

this phase consist of Standby Drift #2, Cross-Block Drift #2, and 17 emplacement drifts. Emplacement operations would continue in the emplacement drifts completed during Phase I.

2.3.5.3.2 Ventilation and Controls

Air locks used to separate the emplacement area from the development area would be relocated southward along the exhaust main and the east and west mains. During Phase II, intake air for the emplacement area would be drawn in through the north ramp and Intake Shaft #1 and exhausted through Exhaust Shaft #1. Fresh air for the development side would be blown in through the intake development shaft and Intake Shaft #2 and exhausted through Exhaust Shaft #2 and the south ramp.

2.3.5.4 Development and Emplacement Phase III

2.3.5.4.1 Facilities Construction

Figure 2-69 illustrates Phase III of development and emplacement work. Development activities during this phase consist of the following excavations:

- Fourteen emplacement drifts
- Intake Shaft #3 and the cross-drift connecting it to the east and west mains
- Exhaust Shaft #3 and access drifts connecting it to the exhaust main
- Cross-Block Drift #3.

In Phase III, emplacement activities would continue in emplacement drifts completed during Phase II.

2.3.5.4.2 Ventilation and Controls

The ventilation air locks would be relocated along the exhaust main and the east and west mains to isolate the development ventilation system from the emplacement area. Intake air for the emplacement area would be drawn through Intake Shafts #1 and #2 and the north ramp and exhausted through Exhaust Shafts #1 and #2. On the development side, fresh air would be blown in through Intake Shaft #3 and the intake development shaft and exhausted through Exhaust Shaft #3 and the south ramp.

2.3.5.5 Development and Emplacement Phase IV

2.3.5.5.1 Facilities Construction

The final development and emplacement phase for the 70,000-MTHM repository design, Phase IV, consists of construction of the last three emplacement drifts and one cross-block drift (Cross-Block Drift #4). Emplacement operations during this phase would continue in emplacement drifts excavated during Phase III. Figure 2-70 illustrates the construction activities during Phase IV.

2.3.5.5.2 Ventilation and Controls

The development side ventilation system would be separated from the emplacement side by relocating air locks along the exhaust main and the east and west mains to points immediately north of the construction activity. The air for the emplacement area during this phase would be drawn through Intake Shafts #1, #2, and #3 and the north ramp. Emplacement air would be exhausted through Exhaust Shafts #1, #2, and #3. Air for the development side would be blown in through the intake development shaft and exhausted through the south main.

2.4 ENGINEERED BARRIERS

The engineered barriers include those components within the emplacement drift that contribute to waste containment and isolation. The design and operating mode described in this report includes the following components as engineered barriers: (1) waste package, (2) emplacement drift invert, and (3) drip shield.

Previous designs (DOE 1998, Volume 2; CRWMS M&O 1999c; CRWMS M&O 1999d) included other engineered barriers that were subsequently eliminated. These include emplacement drift back-

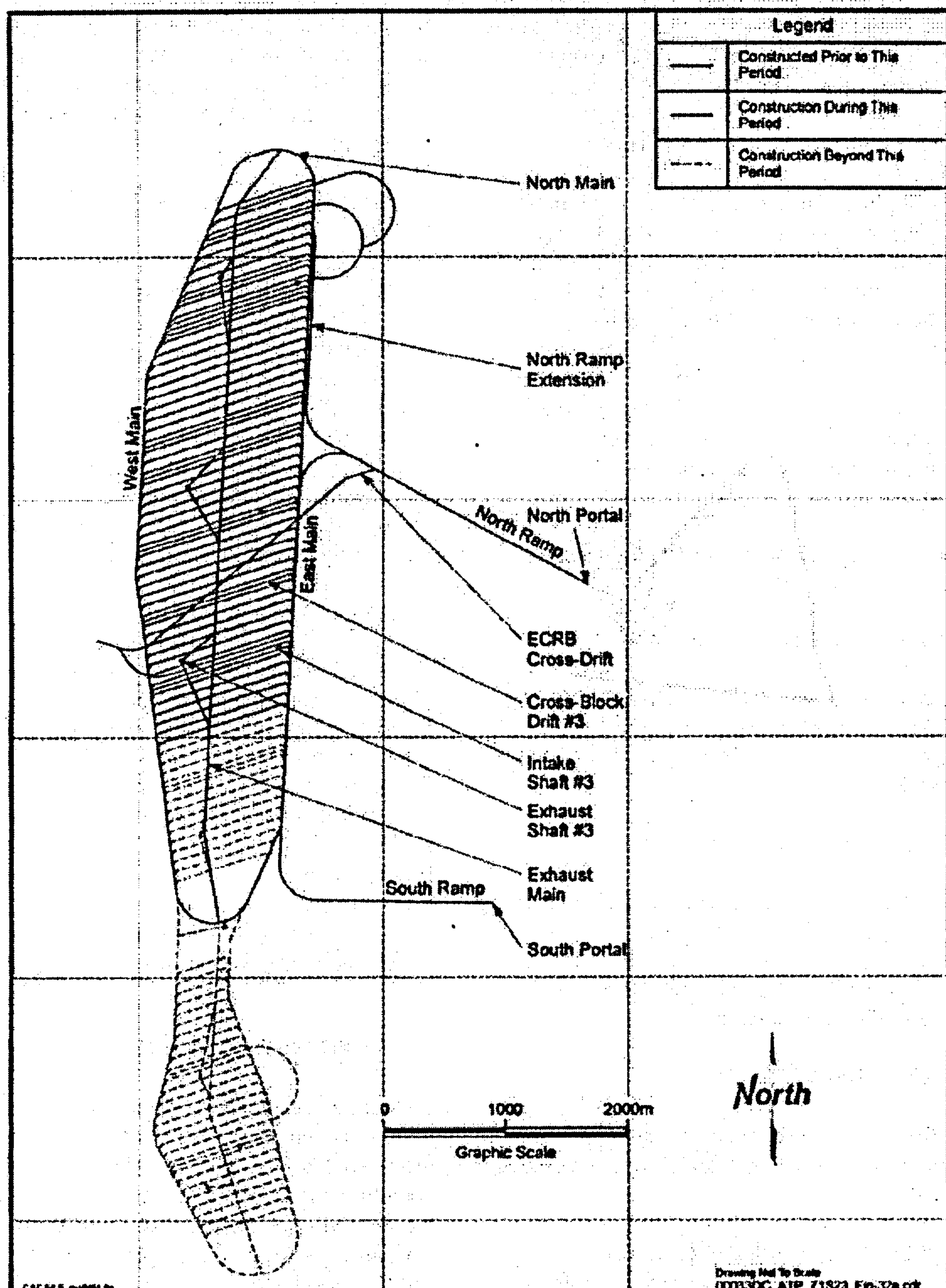


Figure 2-69. Repository Subsurface Layout—Development and Emplacement for Phase III
Construction of emplacement drifts continues during the third phase of development, with completion of 14 more emplacement drifts. Waste emplacement operations continue in previously commissioned drifts.

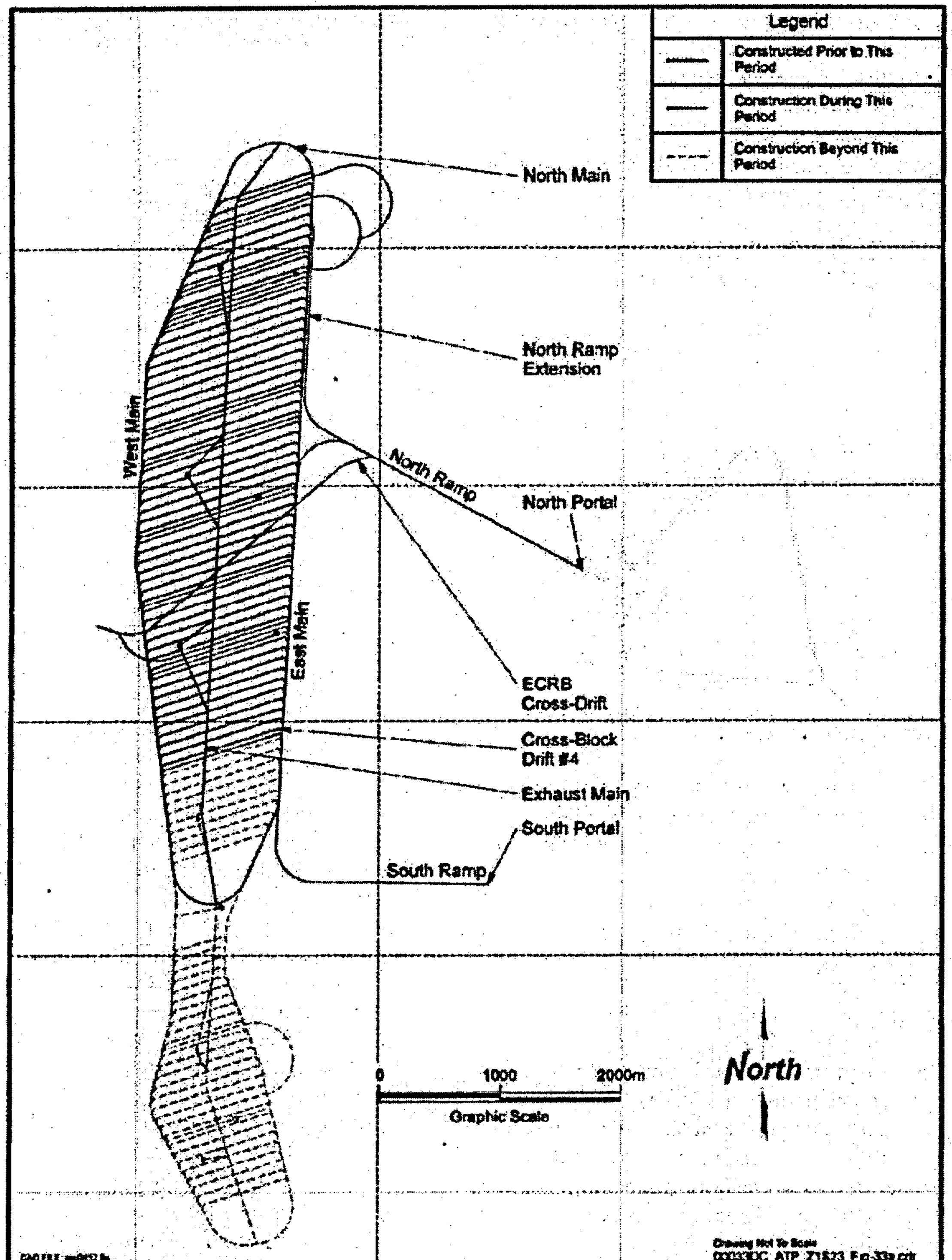


Figure 2-70. Repository Subsurface Layout—Development and Emplacement for Phase IV
Construction activities are limited in Phase IV of development, with the last three emplacement drifts for the base case layout being completed.

fill material, placed either as a homogeneous fill or as a layered system with flow capillary barrier capabilities; chemical retardation and sorption materials in the invert; and precast concrete emplacement drift lining (a ground support system with limited barrier attributes). Discussion of the design evolution and reasons for the elimination of these other engineered barriers are provided in Section 2.1.2 and documented in design assessments (CRWMS M&O 1999c; CRWMS M&O 1999d; Wilkins and Heath 1999; Dyer 2000). The emplacement drift backfill has been retained as an engineered barrier option (Dyer 2000).

The description of engineered barriers provided in this section is summarized from a large set of design descriptions and analyses, including:

- *Invert Configuration and Drip Shield Interface* (CRWMS M&O 2000aq)
- *Design Analysis for the Ex-Container Components* (CRWMS M&O 2000ap)
- *Drip Shield Emplacement Gantry Concept* (CRWMS M&O 2000ar).

2.4.1 Drift Invert

The term "invert" in this design includes the structures and materials that form a platform that supports the pallet and waste package, the drift rail system, and the drip shield. The invert is composed of two parts: the steel invert structure and the ballast (fill), which is composed of granular material.

2.4.1.1 Steel Invert Structure

The design of the steel invert structure must consider construction loads, waste package emplacement and retrieval loads, drip shield loads, and thermal and seismic loads. In case the backfill design option is used in the future, the invert design also considered any backfill loads and related backfill emplacement loads. The invert configuration has to accommodate the other structures inside the emplacement drifts, namely the ground support structures, waste package pallet, waste package emplacement gantry, inspection gantry, drip shield emplacement gantry, and backfill emplacement equipment (if backfill is used). The invert structures would support repository preclosure operations for up to 300 years with limited maintenance in the emplacement drift environment. Additionally, the invert design will be coordinated with the pallet design to fulfill the requirement for maintaining the waste packages in a horizontal emplacement position for 10,000 years after closure (CRWMS M&O 2000aq, Section 4.2.1.7).

Steel materials were selected for the inverts to avoid uncertainties associated with the use of cementitious materials in the emplacement drifts (CRWMS M&O 2000aq, Section 6.2). There is some concern with the use of steel products in the emplacement drifts because of potential impacts on radionuclide transport after waste packages have degraded. The concern is rooted in the fact that ferric-based iron colloids can transport radionuclide materials. Although the concern is applicable only after the waste packages have failed, it may affect the peak dose rate at the time beyond 10,000 years, which is the design life for the waste packages. The ferric-based iron colloids will mainly affect solubility-limited radionuclides, such as plutonium and americium, which have a high affinity with iron-oxyhydroxide colloids (CRWMS M&O 2000as, Section 3.1.2). This potential impact has been considered in TSPA evaluations. Figure 2-71 illustrates the proposed steel invert structure in place in the drift. It shows the relationship of the steel invert structure with the invert ballast, drift wall, ground support structure, pallet and waste package, and drip shield. Figure 2-72 provides a perspective view of a section of an emplacement drift, illustrating the location of the invert components with respect to the ground support steel sets. The steel drift invert is not part of the ground control system. The transverse support beams for the invert would be installed between the structural steel ground control components, as shown in Figure 2-72. To maintain the structural independence of the invert, the ground control components would not contact the steel invert components.

The transverse support beams, which are part of the structural steel invert frame, would rest on the drift wall in such a way that they transfer the loads directly to the rock. Installation of the steel invert structure would include shimming, aligning, and

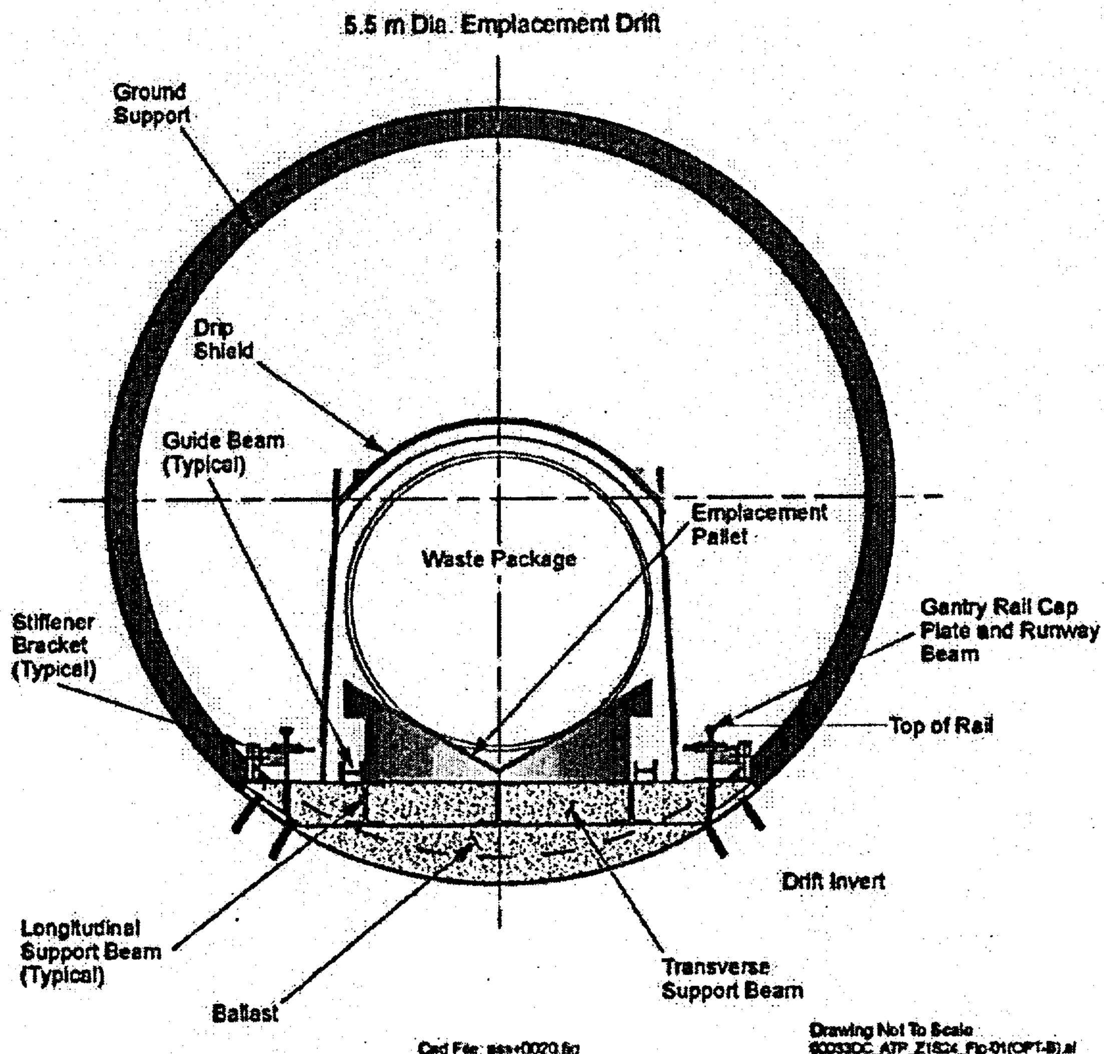


Figure 2-71. Emplacement Drift Cross Section with Invert Structure in Place
This cross section of an emplacement drift shows the engineered barriers and the invert section structures and components. This figure also illustrates how the invert structure is supported by and attached to the drift wall. Source: CRWMS M&O 2000aq, Figure 5.

anchoring the attached base plates to the drift rock wall. The pallet, loaded with a waste package, would rest directly on the structural steel invert frame. The steel invert frame would also support the drip shield and any backfill material, if used, placed directly above the frame and directly above the drip shield, along with the backfill emplacement equipment, the inspection gantry, and the drip shield emplacement gantry. The drip shield would transmit the weight of backfill placed directly above it to the emplacement drift invert structure.

The emplacement drift rail, as shown in Figures 2-71 and 2-72, is mounted on the steel invert. Therefore, the invert structure would also support the transportation loads from the waste package gantry and the remote inspection gantry (CRWMS M&O 2000aq, Section 6.4). The inspection gantry is described in Section 2.5.

The invert structure has three longitudinal support beams that provide continuous support for the emplacement pallets (Figure 2-72). The spacing of the gantry rail centerlines accommodates the drip shield gantry, used for shield emplacement opera-

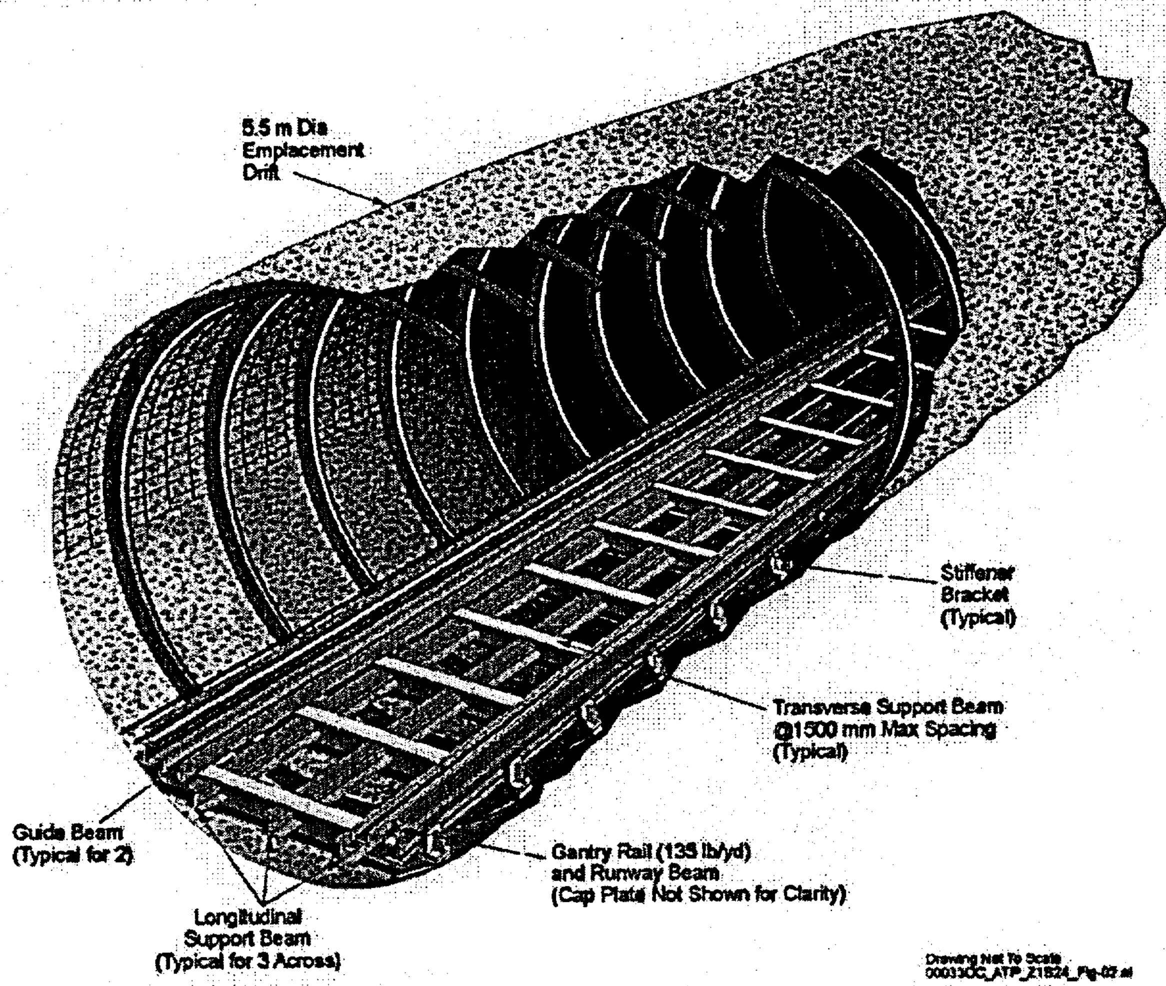


Figure 2-72. Emplacement Drift Perspective View with Steel Invert Structures In Place
This figure shows the transverse and longitudinal beams for the invert structure and how they are positioned with respect to the ground support steel sets. Source: CRWMS M&O 2000aq, Figure 6.

tions. After all the assembly details are specified, the invert design will be checked against dynamic seismic loads to ensure structural integrity and to ensure that the loaded pallet and drip shield do not move significantly from their original locations because of seismic events (CRWMS M&O 2000aq, Section 6.4). Movement of the loaded pallet and drip shield can be mitigated, if necessary, by the installation of guide beams made of structural steel, as shown in Figures 2-71 and 2-72. The emplacement drift invert can be periodically monitored and repaired or replaced as necessary to ensure that the intended emplacement position of each waste package is maintained for a preclosure period of up to 300 years (CRWMS M&O 2000aq, Section 6.4).

Selection of the invert structure materials was based on structural strength properties, compatibility with the emplacement drift environment, and expected longevity. These requirements are contained in the *Emplacement Drift System Description Document* (CRWMS M&O 2000ab, Section 1.2.1). The structural members of the emplacement drift invert would be made of ASTM A 572/A 572M steel. A crane rail of ASTM A 759 carbon steel would be used for the gantry rail. Compatibility of the carbon steel invert materials with the materials forming the drip shield base is being evaluated to determine the potential effects of corrosion. The shield plates for the drip shields would be made with Titanium Grade 7, and the drip shield structural members would be made with

Titanium Grade 24 (CRWMS M&O 2000ap, Section 4.1). A structural angle of Alloy 22 would be used as a base (at the bottom of the side plates of the drip shield) to separate the titanium materials from the carbon steel invert materials (CRWMS M&O 2000aq, Section 6.5).

2.4.1.2 Invert Ballast

The repository subsurface layout is configured so moisture entering the emplacement drifts during preclosure and postclosure periods can drain directly into the surrounding host rock without draining along the drift. A design criterion requires that the invert ballast material be crushed tuff (CRWMS M&O 2000aq, Section 4.3.3). Crushed tuff would be produced by crushing the material removed from Yucca Mountain in the process of excavating the emplacement drifts. Because of the inherent resistance of crushed tuff and carbon steel to the thermal loads expected, these materials will not be affected by the temperatures expected in the emplacement drifts (CRWMS M&O 2000aq, Section 6.5). The principal function for the invert ballast material as an engineered barrier is to provide a layer of material below the waste packages in which the transport of radionuclides from the emplacement drifts into the host rock is dominated by diffusive transport, which is slower than the kind of flow expected to occur in rock fractures. Radionuclides would only be released into the emplacement drifts after deterioration of the waste packages results in the release of radionuclides to the environment. It is expected that crushed tuff ballast materials will provide such a barrier (CRWMS M&O 2000aq, Section 6.4). Materials and their properties are being tested and investigated; the results of those investigations will be incorporated into future design enhancements.

Ballast material would be placed in and around the steel members of the invert structure to an elevation just below the top of the longitudinal and transverse support beams (Figures 2-71 and 2-72). This level of fill will ensure that the pallets and drip shields do not rest on the ballast material, but are fully supported by the steel invert beams. The ballast material would be sufficiently compacted to consolidate it to the point where settlement of the ballast over time would not be significant. The gap resulting from consolidation would be small enough that the ballast would prevent the backfill material, if used, from migrating under the bottom of the drip shield during placement.

2.4.2 Ground Support

The primary purpose of ground support systems is to maintain the stability and geometry of the emplacement drifts during preclosure. However, some ground support systems (see Section 2.3.4.1 for a description) offer secondary benefits as engineered barriers.

The VA design (DOE 1998, Volume 2) used a precast concrete liner as the ground support for the emplacement drifts. This type of ground support system would offer some additional value as an engineered barrier by serving as a barrier to flow into the drift. The precast concrete liner would not prevent water from entering the drift, but would serve as a diversion barrier by forcing water to follow the path of least resistance. This engineered barrier function, however, would last only while the concrete liner remained intact, and concrete may raise the alkalinity in the drift, adversely impacting the containment of radioactive materials. Consequently, the concrete liner in the emplacement drift was eliminated from the design described in this report.

The concrete liner design for the ground support system for the emplacement drifts was replaced with steel sets with welded-wire fabric and rock bolts. Like the concrete liner, these types of structures support waste isolation by preventing rockfall during preclosure. This report and its supporting documents currently support the use of steel sets, in addition to rock bolts and steel wire fabric, to enhance the reliability of the no-maintenance or low-maintenance design concept for the emplacement drifts. However, concerns about introduction of man-made materials into the emplacement drifts may cause future changes in the design. Deterioration of steel products over time results in iron-oxide by-products that may potentially impact the longevity of waste package materials and the transport of radionuclides away from the drift. This concern, if proven valid and significant, may result in a requirement for minimization of steel products in the emplacement drifts, which may shift the emphasis to a ground support design that relies mainly on rock bolts and steel wire fabric or other attractive design options.

2.4.3 Support Assembly for the Waste Package

In accordance with project design criteria, the invert and the emplacement pallet must maintain the waste package in its nominal emplacement position for 300 years. This requirement satisfies waste package retrieval needs, if that option is exercised. Another design criterion for the invert and the emplacement pallet is that they shall maintain the nominal horizontal emplacement position of the waste package for 10,000 years after closure (CRWMS M&O 2000ap, Section 4.2). This criterion supports waste isolation by keeping the waste package under the drip shield, where it is less vulnerable to rockfall or degradation from water-induced corrosion.

The design of the emplacement pallet fulfills these requirements by using construction materials that will last 10,000 years, and by providing structurally sound and stable invert and pallet designs. Horizontal movement of the waste package over 10,000 years may be caused by support beam corrosion, ballast settlement, and seismic activity (CRWMS M&O 2000ap, Section 6.2.2). Support beam corrosion and ballast settlement will be evaluated during later design phases, as will the seismic response of the emplacement pallet loaded with a waste package.

2.4.3.1 Pallet Design

The pallet design supports line-loading of waste packages by allowing placement of waste packages end to end within 10 cm (4 in.) of each other. The pallet is shorter than the waste package (Figure 2-52), so it does not interfere with the close placement requirement.

The emplacement pallets would be fabricated from Alloy 22 plates welded together to form the waste package supports. Alloy 22 is highly resistant to corrosion. Two supports, one at each end of the pallet, would be connected by four square stainless steel tubes to form the completed emplacement pallet assembly (Figure 2-51). These pallet tubes would be fabricated from Stainless Steel Type 316L for corrosion survivability (CRWMS M&O 2000ap, Section 4.1). The Alloy 22 supports have a V-shaped top surface to accept all waste package diameters (CRWMS M&O 2000ap, Section 6.2.1). All surfaces of the emplacement pallet that contact a waste package outer barrier would be Alloy 22, the same material as the outer barrier (CRWMS M&O 2000ap, Section 6.2.2).

Calculations determined the dimensions required of emplacement pallet components to keep stresses within the allowable limits while under static loading. These calculations were performed for an emplacement pallet loaded with the heaviest waste package and resting on the invert (CRWMS M&O 2000ap, Section 6.5). Stress corrosion cracking is not considered an issue for the emplacement pallet because the waste package loads acting on the supporting surfaces are compressive loads. Thus, the emplacement pallet plates would not be susceptible to stress corrosion cracking. Corrosion during the first 300 years would be negligible.

In summary, the prevention of stress corrosion cracking, in conjunction with the corrosion resistance of Alloy 22 and Stainless Steel Type 316L, will contribute to long-term pallet integrity. This, in turn, will ensure that the pallet will maintain the waste package in its normal emplacement position for a preclosure period of up to 300 years (CRWMS M&O 2000ap, Section 6.5). The same factors will contribute to the ability of the pallet to maintain the waste package in its nominal horizontal position for 10,000 years after closure.

2.4.3.2 Pallet Interface with Invert

As explained in Section 2.4.1.1, the steel invert has transverse support beams that rest on the rock wall. Attached to these transverse support beams are three longitudinal support beams that provide continuous support for the emplacement pallets (Figures 2-71 and 2-72). The transverse support beams and longitudinal support beams thus transfer the waste package load to the rock at the points where the transverse beams rest on the rock wall. The steel invert materials would be fabricated in panels consisting of longitudinal and transverse support beams with attached base plates. Adjacent panels would be attached by bolting or welding.

2.4.4 Drip Shield

Design requirements for the emplacement drift system (CRWMS M&O 2000ab) state that drip shields be installed over the waste packages before closure. After closure, and after the heat produced by the waste packages has dissipated, moisture may enter the emplacement drifts in liquid form or as water vapor. The function of the drip shields is to divert the liquid moisture that drips from the drift walls, as well as the water vapor that condenses on the drip shield surface, around the waste packages and to the drift floor, prolonging the longevity and structural integrity of the waste packages. The drip shields are designed to link together, forming a single, continuous barrier for the entire length of the emplacement drift. Table 2-22 summarizes the preliminary engineering specifications for the drip shield design.

Table 2-22. Drip Shield Design Detail

Item	Detail
Material	
• 15-mm water diversion surfaces	Titanium Grade 7
• Structural members	Titanium Grade 24
• Stands (feet)	Alloy 22
Dimensions	
• Height	2,586 mm (overlap end) 2,521 mm (butt end)
• Width	2,512 mm (overlap end) 2,505 mm (butt end)
• Length	6,105 mm
• Weight	3.087 metric tons

Source: CRWMS M&O 2000at, Attachment II.

The design requirements for drip shields include corrosion resistance as well as structural strength. Corrosion resistance is required so the drip shields can perform their moisture diversion function with high reliability for 10,000 years. Structural strength is required so the drip shield can protect the waste package against damage by rockfalls resulting from degradation of the drift walls, withstanding damage from rocks weighing up to 6 metric tons (CRWMS M&O 2000ab, Sections 1.2.1.14 and 1.2.1.15).

The drip shields must also withstand the static loads from backfill (if used) and rock from the design basis rockfall without exceeding 20 percent of the yield strength of the drip shield materials. This requirement is needed to prevent stress corrosion cracking (CRWMS M&O 2000ap, Section 4.2.3).

2.4.4.1 Drip Shield Design

Figure 2-73 illustrates the drip shield design. The drip shield would be fabricated from Titanium Grade 7 plates for the water diversion surfaces, Titanium Grade 24 for the structural members, and Alloy 22 for the feet. The Alloy 22 feet would be mechanically attached to the titanium drip shield side plates, since the two materials cannot be welded together. The Alloy 22 feet prevent direct contact between the titanium and the carbon steel members in the invert, which could result in hydrogen embrittlement of the titanium (CRWMS M&O 2000ap, Section 6.1.1). All the drip shields would be uniformly sized, so one design will suffice for any waste package size. As Figure 2-74 illustrates, the drip shield sections interlock to prevent separation between sections. The interlocking is accomplished by using pins and holes and also by using an overlapping section with connector guides.

To provide long-term protection against corrosion, the drip shield would be manufactured entirely of Titanium (Grades 7 and 24) and Alloy 22. Calculation results have verified that the long-term static stresses that could lead to stress corrosion cracking will remain below 20 percent of the yield strength of the materials in the drip shield components (CRWMS M&O 2000ap, Section 6.1.2.7). A Titanium Grade 7 thickness of 15 mm (0.6 in.) has been selected for long-term corrosion resistance. Calculations have verified the adequacy of this thickness and material grade. Titanium Grade 24 was selected for the structural components because of its superior strength in comparison with Titanium Grade 7. The required dimensions for the Titanium Grade 24 structural components, shown in Figure 2-75, have been defined by structural calculations (CRWMS M&O 2000ap, Section 6.1.2).

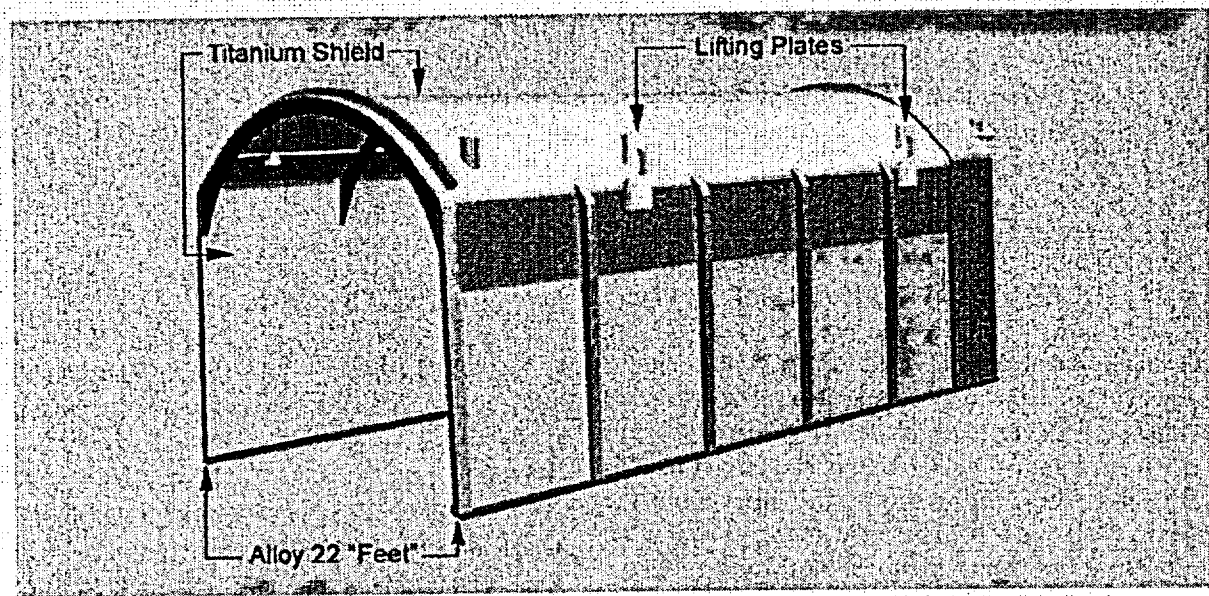


Figure 2-73. Drip Shield Isometric View
The drip shield is a self-standing structure built from structural grade titanium. Source: Modified from CRWMS M&O 2000ap, Figure 1.

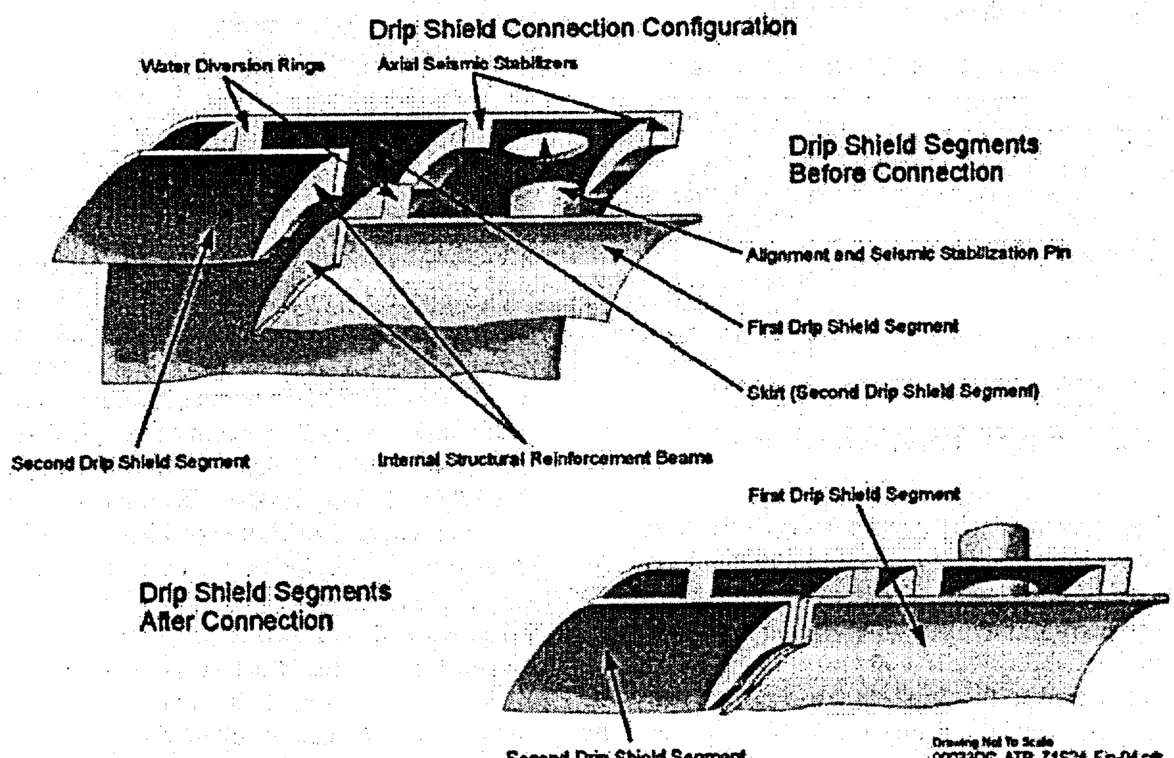


Figure 2-74. Drip Shield Interlocking Connection
The drip shield has a raised section at one end that overlaps and interlocks with the adjacent drip shield. Built-in water diversion rings keep moisture from penetrating the drip shield interface area. Source: CRWMS M&O 2000ap, Figure 2.

The geometry of the drip shield design will divert dripping water around the waste package and onto the emplacement drift floor. The interlocking mechanisms of the drip shield include water diversion rings to divert dripping water at the seams between drip shield segments and around the waste packages to the emplacement drift floor.

2.4.4.2 Drip Shield Interface with Invert

The drip shield feet would rest on the transverse support beams of the drift invert and alongside the longitudinal guide beams, as Figure 2-71 illustrates. The guide beams are also shown in Figure 2-72. The primary purpose of these longitudinal guide beams is to center the emplacement pallet; however, they also create a separation barrier between the pallet and the drip shield.

2.4.4.3 Drip Shield Emplacement

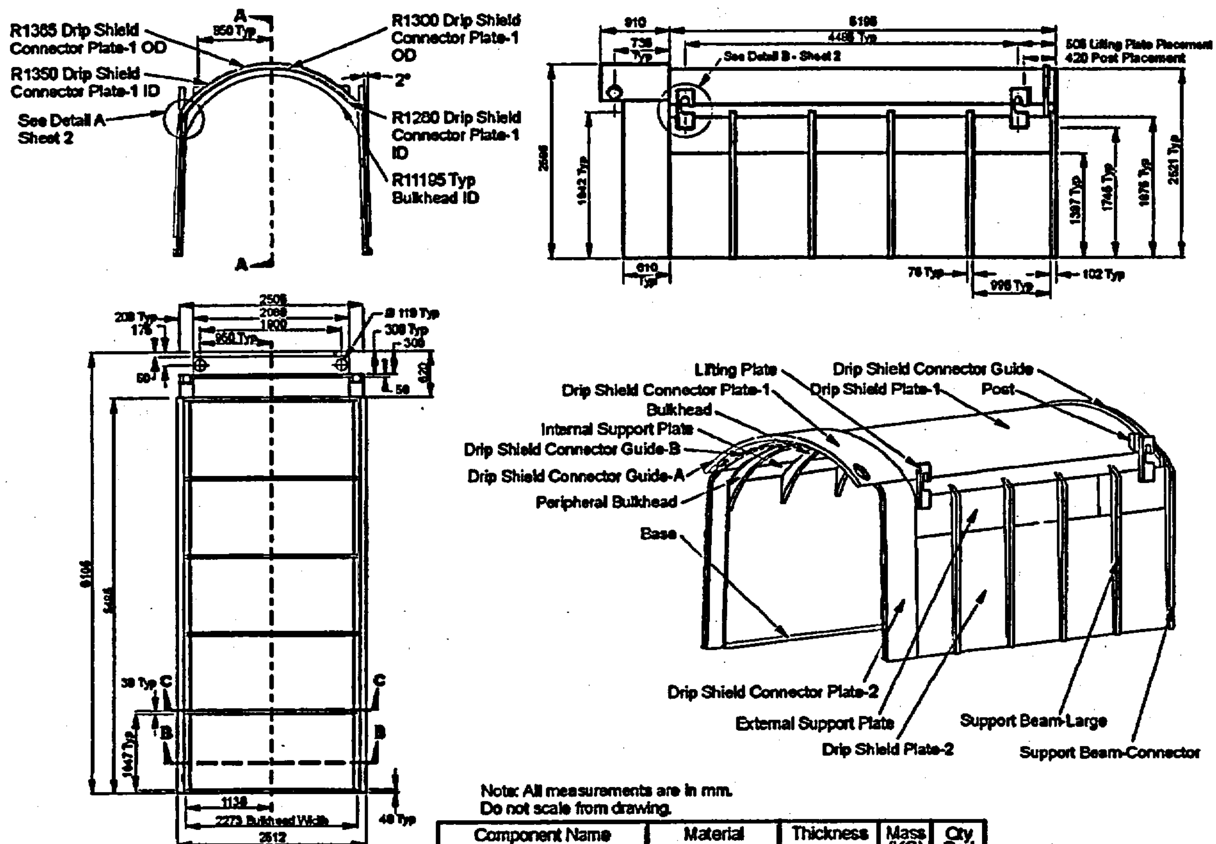
A drip shield emplacement gantry would emplace the drip shields (CRWMS M&O 2000ar, Section 6.1). The emplacement gantry is designed specifically to emplace drip shields over emplaced waste packages in the 5.5-m (18-ft) diameter emplacement drifts. The drip shield emplacement gantry design is based on the waste package emplacement gantry design, with similar structural and mechanical components. In addition to the drip shield emplacement gantry, other equipment used in drip shield emplacement operations includes:

- A flatbed gantry carrier railcar
- A flatbed railcar for transportation of the drip shields
- A locomotive to move the railcars.

The drip shield emplacement operation is similar to the waste package emplacement operation described in Section 2.3.4.5. The locomotive and the gantry carrier would transport the drip shield emplacement gantry to the emplacement drift turnout. After following the same sequence of events for switching to remote control operation of the locomotive and opening the drift isolation doors, the gantry carrier would dock at the emplacement drift. The remote control operators would then drive the drip shield emplacement gantry into the drift and put the gantry on standby inside the drift until a drip shield is delivered to the emplacement drift entrance. After a drip shield carrier car docks at the emplacement drift, the drip shield emplacement gantry would move over a drip shield by straddling the railcar. The gantry would engage the drip shield using a set of four lift pins that engage lifting plates on the shield. The gantry would lift the drip shield off the railcar, then carry the shield through the emplacement drift and over the waste packages to emplace the shield (CRWMS M&O 2000ar, Section 6.1). Figure 2-76 shows the drip shield emplacement gantry. An insert on this figure shows the lift plate and lift pin interface.

The weight of each drip shield unit has been estimated at about 3 metric tons (CRWMS M&O 2000ar, Section 5.4), but an upper-limit bounding value of 4 metric tons was used in the design. The lifting mechanism on the gantry uses ball-screw jacks, a common form of screw jack used for lifting heavy loads, to raise the drip shields. This mechanism has longer duty cycles and higher efficiencies than typical machine screw mechanisms. The gantry frame is a steel structure designed to support its own weight, the weight of the drip shield, and any seismic loads. The gantry would be self-propelled, using four two-wheeled trucks, with one truck placed at each corner of the gantry frame. A direct-current, variable-speed electric motor would drive one wheel in each truck through a gear reducer and chain drive. An integral brake within the direct-current electric motor would provide braking. Integral brake motors would work in a fail-safe configuration: if the electric motor loses power, a spring mechanism would automatically apply the brakes. All motors (including the ball-screw jack motors) and any other operating components of the gantry would be electrically operated and controlled (CRWMS M&O 2000ar, Sections 6.2.1 and 6.2.3).

The frame for the drip shield emplacement gantry would be made of ASTM A 36 steel and fabricated following American Welding Society standards. The selection of ASTM A 36 steel, or possibly other high-strength steels, will be reevaluated during final design of the gantry. The gantry frame and lifting mechanism dimensions would be sized to provide sufficient clearance for the drip shield over the rail transporter and emplacement drift



Note: All measurements are in mm.
Do not scale from drawing.

Component Name	Material	Thickness	Mass (KG)	Qty Rqd
Drip Shield Plate-1	SB-265 R52400	15	837	1
Drip Shield Plate-2	SB-265 R52400	15	750	2
Base	SB-575 N06022	10	55	2
Bulkhead	SB-265 R56400	38	45	4
Peripheral Bulkhead	SB-265 R56400	48	58	2
Post	SB-265 R56400	100	5.8	2
Internal Support Plate	SB-265 R52400	12.7	18	10
External Support Plate	SB-265 R52400	12.7	27	10
Lifting Plate	SB-265 R56400	50	17	4
Support Beam-Connector	SB-265 R56400	102	27	6
Support Beam-Large	SB-265 R56400	76	37	8
Drip Shield Connector Guide	SB-265 R52400	50	30	2
Drip Shield Connector Plate-1	SB-265 R52400	15	142	1
Drip Shield Connector Plate-2	SB-265 R52400	15	69	2
Drip Shield Connector Guide-A	SB-265 R52400	50	26	1
Drip Shield Connector Guide-B	SB-265 R52400	50	31	1
Drip Shield Assembly	-	-	4203	1

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Figure 2-75. Drip Shield Structural Components
Support beams, bulkheads, and support plates give the drip shield the strength and structural rigidity required to withstand backfill, if used, and rockfall. All measurements are in millimeters. Do not scale from drawing. OD = outside diameter; ID = inside diameter. Source: Modified from CRWMS M&O 2000at, Attachment II.

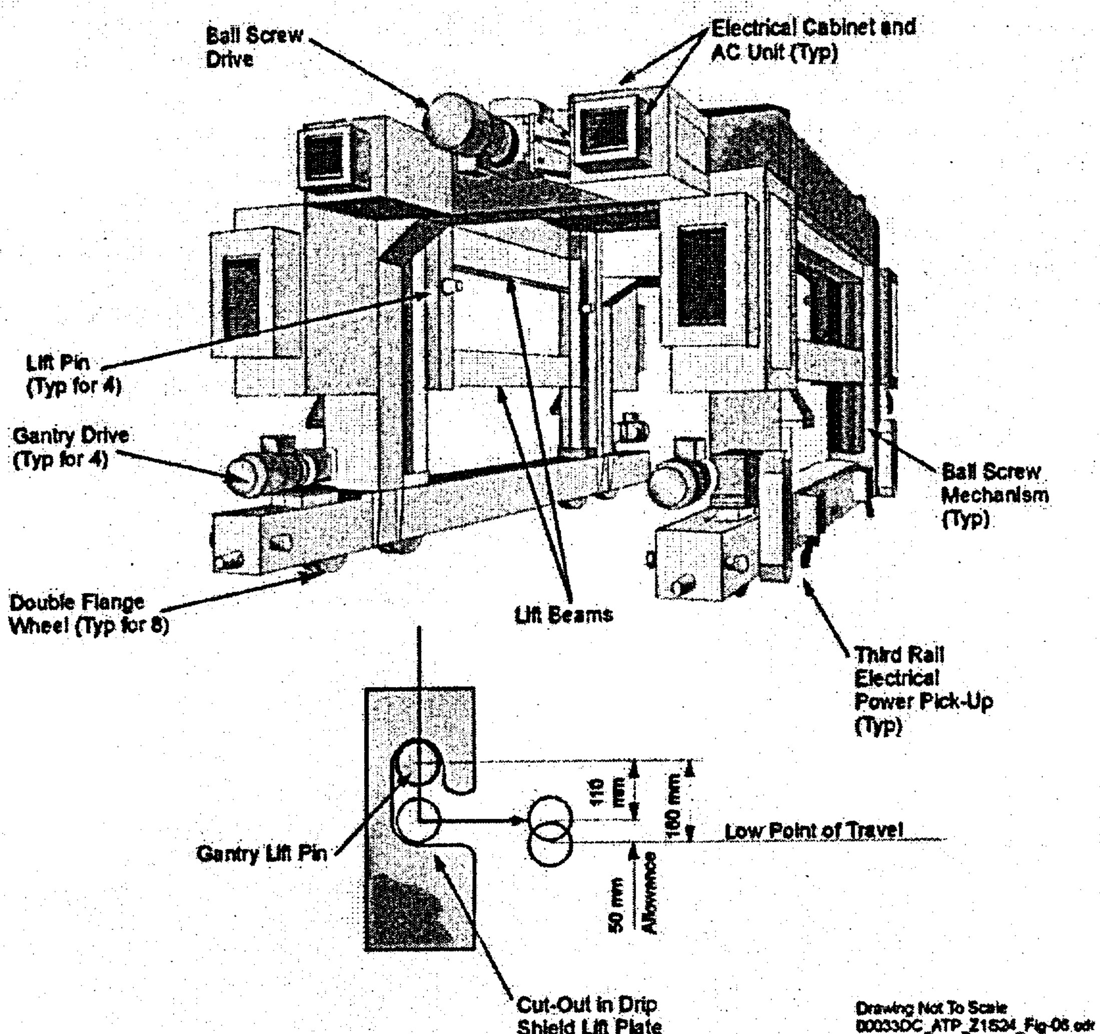

Figure 2-76. Drip Shield Emplacement Gantry and Lift Pin Mechanism
The drip shield emplacement gantry has a lift pin mechanism to engage the shield, lift it, and safely emplace it over the waste packages. AC = air conditioning. Source: CRWMS M&O 2000ar, Figures 2 and 4.

structural invert. These dimensions also take into account the vertical travel distance required for lifting pin disengagement (CRWMS M&O 2000ar, Section 6.2.3, Figure 3).

The drip shield emplacement gantry is designed to operate in the harsh environment inside the emplacement drifts, taking into account the moderately high temperature of 50°C (122°F), a relative humidity ranging from 10 to 100 percent, and high radiation levels. Operators at a remote control console located at the surface would remotely control the drip shield emplacement gantry (CRWMS M&O 2000ar, Section 6.2.5).

The primary source of electrical power for the drip shield emplacement gantry would be an electrified third rail (conductor bar) system. This is the same electrical supply system that would be used by the waste package emplacement gantry. Like the waste package emplacement gantry, the drip shield emplacement gantry would be supplied with redundant power pickup mechanisms to ensure a reliable and continuous source of power. The gantry would also carry an emergency backup power system that would provide a limited supply of electrical power. This backup system would provide enough power for the gantry to lower and release a drip shield and move to the emplacement drift entrance (CRWMS M&O 2000ar, Section 6.2.5.1).

Instrumentation and controls on the gantry would consist of high-resolution, articulated, closed-circuit television cameras and a series of high-intensity lights. The gantry would be equipped with thermal and radiological sensing instruments. Onboard cabinets and all temperature-sensitive components would be cooled with air-conditioning units mounted on each enclosure. The gantry would also be equipped with a fire protection system that would respond automatically if an onboard fire is detected. The fire detection portion of the system would immediately notify remote operators, through the gantry control computers, of the location and nature of the fire (CRWMS M&O 2000ar, Section 6.2.5.6).

Figure 2-77 illustrates a typical section of an emplacement drift with a series of emplaced waste packages of different sizes and with the drip shields in place. A portion of the drip shields has been cut out in the illustration for clarity in showing placement of the waste packages and interlocking of the drip shields.

No significant rockfall is anticipated during the preclosure period. It is envisioned that special equipment or attachments to the waste package or drip shield emplacement gantries will be used to "sweep" the mating surface of the emplaced drip shield on which the adjacent drip shield will be coupled. Debris on the invert surface that could interfere with emplacement of the drip shields would be removed or "plowed" aside. These operations would occur before or during drip shield emplacement. The design of such equipment will be developed as needed and as the detailed subsurface equipment designs evolve.

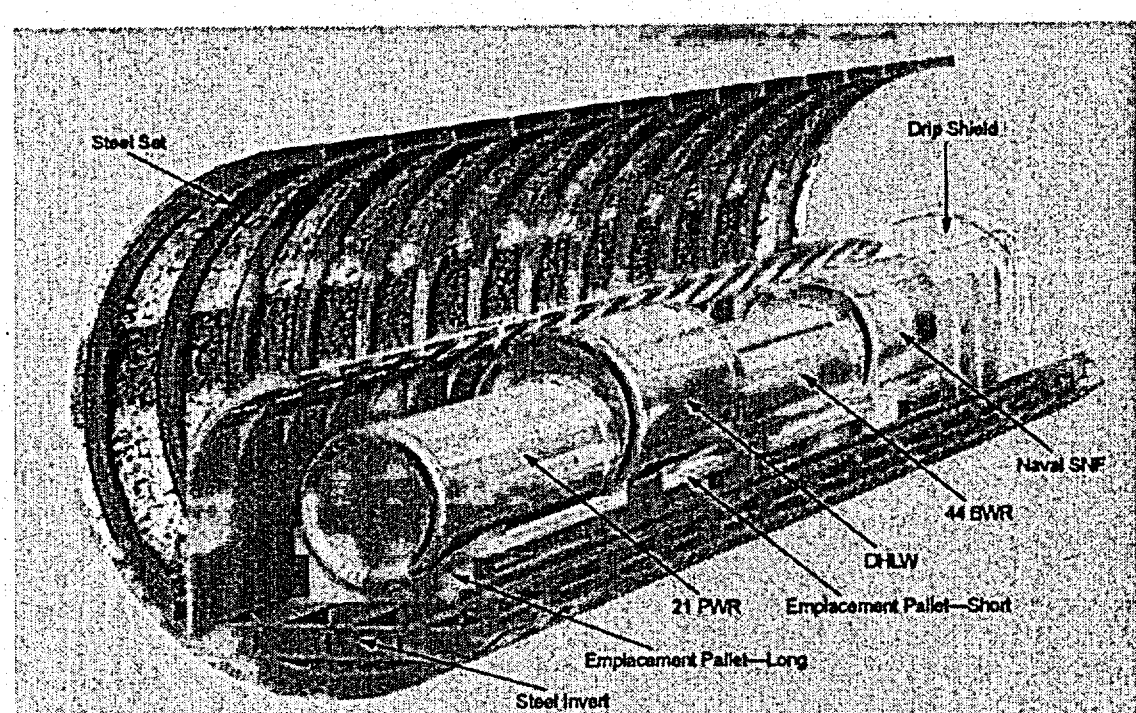

Figure 2-77. Typical Section of Emplacement Drift with Waste Packages and Drip Shields in Place
The final configurations of waste package emplacement and drip shield emplacement are illustrated in this figure. A cutaway view is provided to better illustrate the waste packages. The different types of waste packages are shown for illustration purposes only. One size drip shield accommodates all sizes of waste packages.

2.5 PERFORMANCE CONFIRMATION FACILITIES DESIGN

This section discusses the elements of repository subsurface design that support the *Performance Confirmation Plan* (CRWMS M&O 2000ag). Section 4.6 summarizes the objectives and scope of the plan. Section 5.4 outlines additional testing and evaluation performed during the preclosure years that will support the performance confirmation program. The description of performance confirmation facilities provided in this section is based primarily on *Performance Confirmation Plan* (CRWMS M&O 2000ag).

2.5.1 Facilities Functions and Types

The roles of the subsurface systems in support of performance confirmation are:

- Constructing the subsurface facilities required for performance confirmation program implementation

- Supporting the installation of monitoring equipment, instrumentation, data acquisition systems, and data communication networks

- Providing operations support and system maintenance

- Providing transportation, ventilation, and utilities

- Supporting planning and construction of future facilities, as needed

- Operating and supporting specialized emplacement drift access equipment, such as the remote inspection gantry.

The subsurface systems provide facilities to support monitoring of the natural and engineered barriers, emplacement drift and main drift environments, rock temperature and moisture regimes, water inflow into emplacement drifts, waste package integrity, and seismic activity. The subsurface facilities supporting the performance confirmation program can be grouped into the following types:

- Postclosure test drifts
- Observation drifts
- Alcoves
- Niches
- Boreholes.

A single type or a combination of these types of facilities can constitute a performance confirmation arrangement, depending on the requirements for access to the monitored geologic media and on the objectives of the monitoring. For example, monitoring the rock environment between emplacement drifts would require a combination of an observation drift, alcoves, and boreholes. The observation drift would provide the main access facility for personnel and equipment and would serve as a ventilation airway to maintain ambient temperatures suitable for personnel and instruments. Alcoves extend away from the observation drifts, like fingers, to provide advantageous positioning of monitoring equipment. Alcoves also provide space for installation equipment (e.g., drilling rigs) and monitoring equipment away from the observation drift transportation corridor. The alcoves are designed such that they provide strategic platforms from which instrumentation boreholes can be drilled. Boreholes drilled at different angles and to different lengths can provide monitoring access to the rock media at many locations around the emplacement drifts. The boreholes used in performance confirmation will be small-diameter, precision-drilled holes that allow remote access to monitoring locations within the rock. They offer a less intrusive way to reach areas around the emplacement drifts. Niches are small, excavated areas along the walls of the main drifts or observation drifts that would be used for setting up specialized instruments or monitoring equipment, such as seismometers or data communications equipment.

2.5.2 Proposed Performance Confirmation Facilities

Figure 2-78 illustrates the proposed locations of the performance confirmation facilities. This figure is a layout of the potential repository that shows how

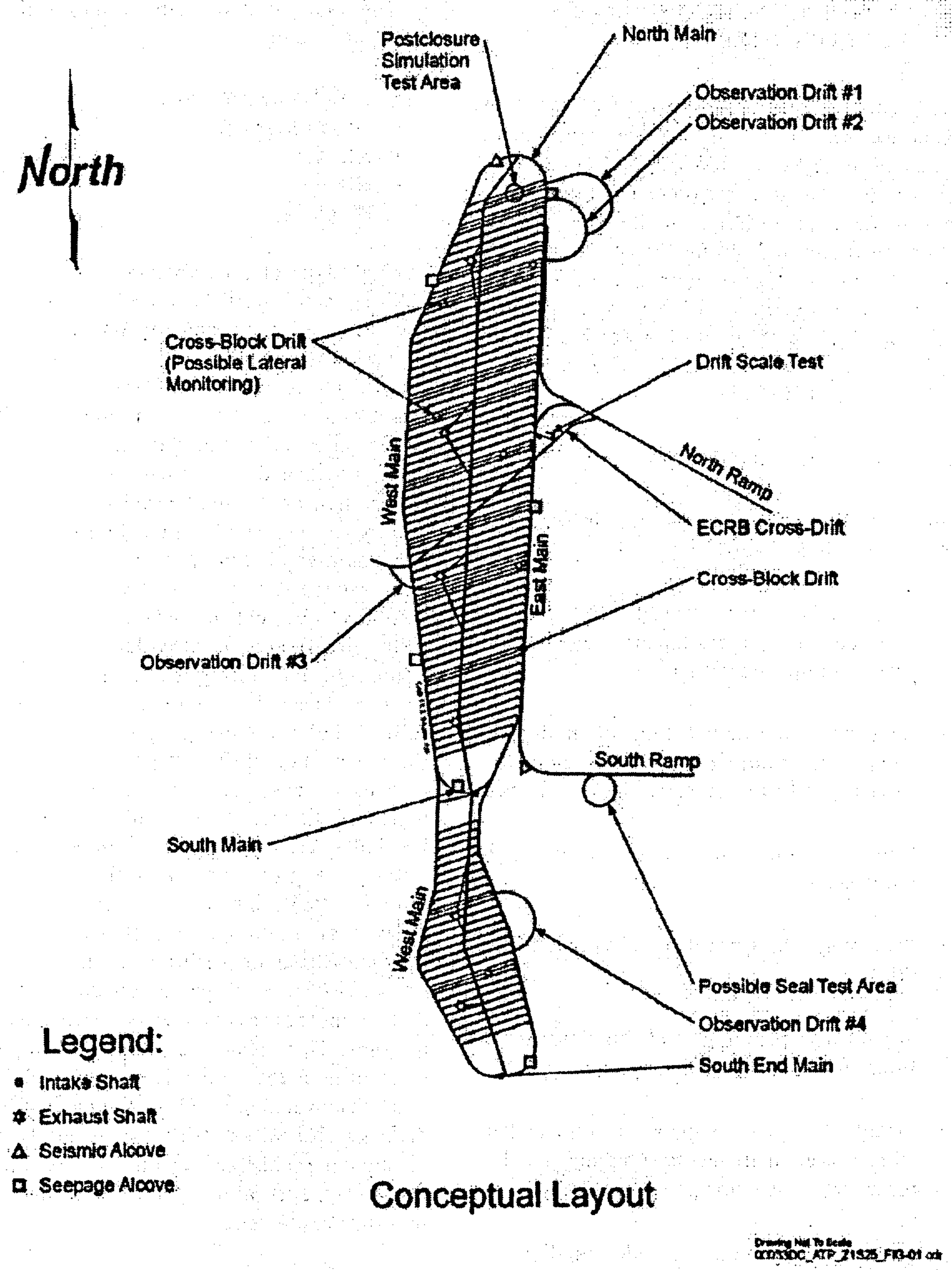

Figure 2-78. Subsurface Performance Confirmation Facilities Layout
This plan view shows the major subsurface facilities that support performance confirmation activities and how those facilities relate to the rest of the subsurface facilities layout. Source: Modified from CRWMS M&O 2000ag, Figure 5-4.

the proposed performance confirmation facilities have been completely integrated into the overall repository design for the best use of access, transportation, and ventilation networks.

2.5.2.1 Postclosure Simulation Test Area

The Postclosure Simulation Test Area consists of a single test drift, located at the edge of the emplacement horizon, with similar physical characteristics to a representative emplacement drift. This test drift would be separated into test sections, which would allow for the simulation of several different test cases. Within a particular test section, heaters or actual waste packages would be emplaced for a period long enough to allow for heating and drying effects on the adjacent rock mass. After a prescribed time, the postclosure configuration would be constructed in the section with heater-equipped surrogate or actual waste packages, and drip shields, using expected postclosure configurations. Instruments would be placed in the drift from adjacent alcoves associated with an observation drift (Figure 2-79) located below the emplacement level. The alcoves would be located below the postclosure test drift to minimize any potential impact on the moisture flow around the test drift opening (i.e., the near-field geohydrology).

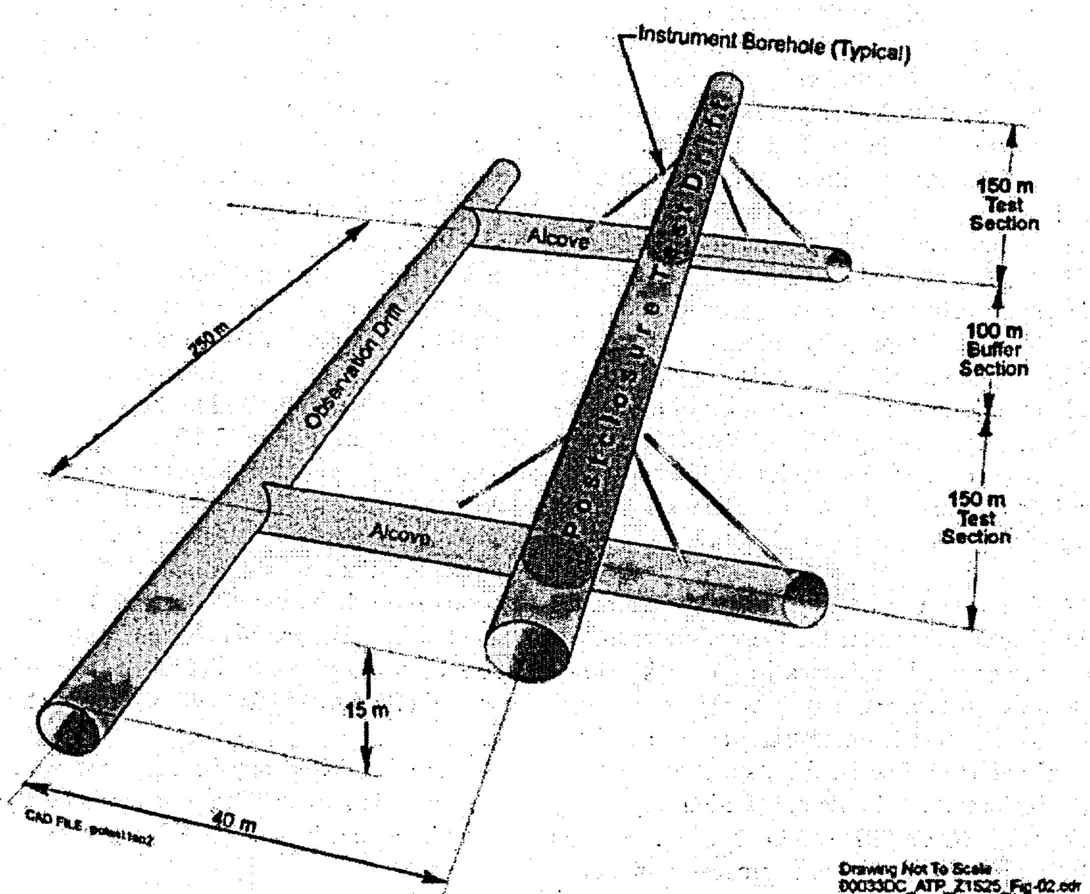

Figure 2-79. Conceptual Configuration for Postclosure Simulation Test Sections
The initial one or two drifts in the northern section of the repository block would be dedicated to performance confirmation activities, serving as postclosure test drifts. Both real and dummy waste packages would be used in these facilities to collect performance data and natural barrier response during the emplacement and monitoring phases of preclosure to support postclosure performance assessments. For clarity, only a portion of the Postclosure Simulation Test Area is illustrated, and only a few boreholes are shown. Source: Modified from CRWMS M&O 2000ag, Figure 5-3.

Alcoves at this location would also permit monitoring of the rock mass response below the drift, where greater temperature and response may be expected (CRWMS M&O 2000ag, Section 5.3.2).

Figure 2-79 provides a conceptual layout of the Postclosure Simulation Test arrangement, showing the relative locations of the test drift, observation drift, and alcoves. Also shown is a typical pattern of boreholes drilled into the rock from the alcoves and reaching areas in the rock around the test drift.

2.5.2.2 Observation Drifts

Figure 2-78 illustrates the proposed observation drifts. All are aligned parallel to the emplacement drifts and located along the centerline of the rock pillars but at different elevations than the emplacement drifts. Observation Drift #1, located below the emplacement drift horizon, is associated with the Postclosure Simulation Test area discussed in Section 2.5.2.1. Observation Drift #2 would be located above the emplacement drift horizon, within the first panel of emplacement drifts. Observation Drift #3, also located above the emplacement drift horizon, would be at about the midpoint in the repository block. Observation Drift #4, located above the emplacement horizon, would monitor the southern extension of the repository block. The existing ECRB Cross-Drift is also being proposed as an observation drift from which several emplacement drifts in the central area of the repository can be monitored.

These observation drifts and associated alcoves would be used to monitor the thermal-mechanical-hydrologic coupled processes in the rock. Monitoring these processes, along with the changes caused by the heat from emplaced waste and the eventual reduction in heat from spent nuclear fuel decay, is important in demonstrating repository performance (CRWMS M&O 2000ag, Section 5.4.1.2). Sensors to measure process parameters (see Section 4.6) would be installed and grouted into the boreholes. Data acquisition and communication equipment attached to the sensors would be located at the alcoves. The data communication lines would be linked to a central communications network, using fiber-optic cables leading from the alcoves to the performance confirmation control and data processing center at the repository surface facilities.

Observation Drift #1 would be located approximately 10 to 15 m (35 to 50 ft) below the emplacement drifts, while Observation Drifts #2, #3, and #4 would be located approximately 15 to 30 m (50 to 100 ft) above the emplacement drifts (CRWMS M&O 2000ag, Figure 5.5). These drifts will be thermally affected by the heat of the emplacement horizon; however, the separation will be enough to make the observation drifts accessible to personnel without additional ventilation. Rock temperatures around the observation drifts have been estimated to peak at about 46°C (115°F) during the preclosure period for the design and operating mode presented in this report. Therefore, the observation drifts will be exposed to rock temperatures within the design conditions for human access during the preclosure period without the aid of additional ventilation. Airflow can be maintained and regulated, as needed, to maintain the drift temperatures at a more comfortable level for human access. The pattern for ventilation of the observation drifts would be similar to the pattern for ventilation of the emplacement drifts: fresh air would be supplied at both ends of the observation drifts from the east and west mains and exhausted through a raise connected to the service side of the exhaust main. Figure 2-80 illustrates a typical layout for the ventilation system in an observation drift. The observation drifts would be at a different elevation plane than the east and west mains; therefore, ventilation raises at the ends of the observation drifts would connect the drifts to the main drift airways. Ventilation regulators would control the airflow to the observation drifts, as needed (CRWMS M&O 2000x, Figure 1).

Observation drifts may be equipped with rail for transportation of drilling and other heavy equipment into the drifts.

2.5.2.3 Other Performance Confirmation Facilities

The ventilation cross-drifts excavated parallel to and between emplacement drifts, at the same elevation plane, provide advantageous locations from which to drill boreholes into the rock pillars.

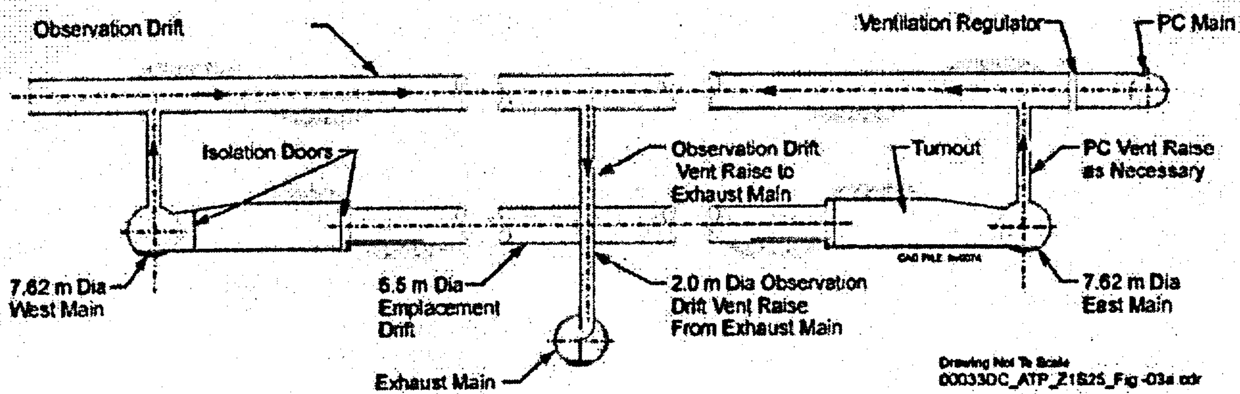

Figure 2-80. Observation Drift Airflow Concept
As illustrated in this figure, vertical raises are used to connect the observation drifts to the ventilation air supply from the east and west mains; another raise exhausts the air to the service side (unheated air) of the exhaust main. PC = performance confirmation. Source: CRWMS M&O 2000x, Figure 3.

These cross-drifts could therefore be used for additional monitoring installations, as needed.

The ECRB Cross-Drift (Figure 2-78), located approximately 15 to 30 m (49 to 98 ft) above the emplacement drifts, also offers a potentially suitable platform for performance confirmation facilities. With the use of alcoves and instrumentation boreholes, nine emplacement drifts could be monitored from this position.

Figure 2-78 also shows the location of the existing Drift Scale Test near the north ramp curve. After the test is completed, this facility will probably continue to serve performance confirmation functions during the preclosure period. The figure also shows the proposed location for the Seal Test Area, where constructibility and performance testing of the ramp and shaft seals would be performed in support of the performance confirmation program.

Several performance confirmation facilities would be located along the main drifts, in either alcoves or niches. Figure 2-78 shows proposed locations for two seismic alcoves to be equipped with seismometers at opposite ends of the repository block. Seismic facilities would monitor the subsurface response at the repository horizon level to seismic events. This subsurface seismic monitoring would require short alcoves, or niches, to house and protect a data acquisition system installed adjacent to a borehole containing a seismic probe (CRWMS M&O 2000ag, Section 5.4.1.3.4).

Figure 2-78 also shows the proposed locations of six seepage alcoves, where percolation from the rock strata above the emplacement horizon would be monitored around the perimeter of the emplacement block. In situ monitoring of seepage would require short alcoves hermetically sealed from the ventilation system with bulkheads. Inspection or maintenance of instrumentation inside the sealed alcoves would require access through the bulkhead seal (CRWMS M&O 2000ag, Section 5.4.1.3.2).

2.5.3 Subsurface Performance Confirmation Support Facilities

2.5.3.1 Data Acquisition Support Facilities

The subsurface data collection system would consist of groups of instruments reporting to a single local data acquisition system. The data acquisition system would collect data from the integrated instruments at set intervals, using input/output boards. The data acquisition system would store the data locally.

These local data acquisition systems would collect and transmit data from the various repository monitoring areas to a surface-based data control and storage facility. Based on current technology, data acquisition systems would transmit data through controller boards, over fiber-optic trunk lines installed in access ways and along observation drifts, and to the surface data control facility (CRWMS M&O 2000ag, Section 5.4.1.4).

2.5.3.2 Mobile Vehicle Control Systems

In addition to the stationary control system, mobile performance confirmation systems, such as the remote inspection gantry (Figure 2-81), would collect data. Such equipment would also be controlled from the surface facilities during monitoring incursions into the emplacement drifts. A mobile vehicle operations control system and an associated communications transmission system would be required to collect data from both mobile and stationary instruments and to provide real-time data to control these moving observation platforms accurately and safely. Microwave or radio wave transmissions from the gantry would be collected by in-drift wires and transmitted through a subsurface network to an operator control station at the surface. The operator control station would control and monitor a variety of systems throughout the repository, in addition to the mobile equipment (CRWMS M&O 2000ag, Section 5.4.1.5).

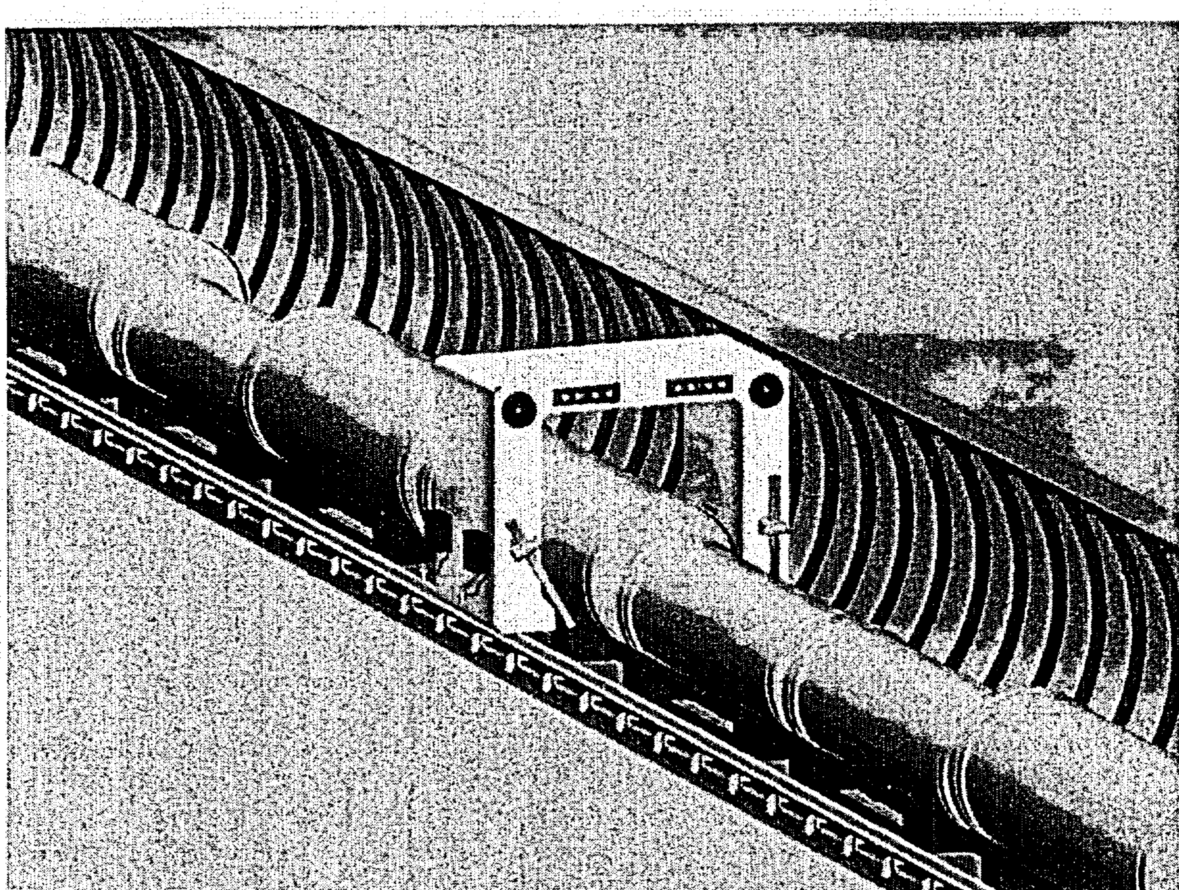

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Figure 2-81. Remote Inspection Gantry Used for In-Drift Performance Confirmation Activities
A remotely operated inspection gantry would be used to inspect and collect information in the emplacement drifts during preclosure. Lights, cameras, sensors, on-board computers, and robotic arms would be designed to make this gantry an efficient and safe inspection tool. Source: CRWMS M&O 2000ag, Figure 5-1.

3. DESCRIPTION OF THE WASTE FORM AND PACKAGING

Section 114(a)(1)(B) of the Nuclear Waste Policy Act of 1982 (NWPA), as amended (42 U.S.C. 10134(a)(1)(B)), requires "a description of the waste form or packaging proposed for use at such repository, and an explanation of the relationship between such waste form or packaging and the geologic medium of the site." This section describes the waste forms to be disposed, along with their packaging; Section 4 explains their relationship with the geologic medium of the site. An explanation of the important parameters considered in the design of the waste package is included in this section, as is a summary of the expected performance of the waste package design. This section:

- Presents an overview of the waste forms and the waste package designs
- Describes the waste package, its design bases, and its functions
- Discusses in detail the waste forms, the parameters considered in designing the waste package (and its variations), and the evaluations performed on the designs
- Describes the material selection and fabrication of the waste package
- Presents the results of design evaluations of the waste package.

Waste Form Overview—Waste forms to be received and packaged for disposal include spent nuclear fuel from commercial power reactors, spent nuclear fuel owned by the U.S. Department of Energy (DOE) (including naval fuel), and canisters of solidified high-level radioactive waste from prior commercial and defense fuel reprocessing operations, some of which would contain cans of immobilized plutonium.

Section 114(d) of the NWPA (42 U.S.C. 10134(d)) limits the first repository's capacity to no more than 70,000 MTHM "...until such time as a second repository is in operation." The types of waste that would be accepted at the potential repository have been allocated as follows (DOE 2002, Chapter 2):

- 63,000 MTHM of commercial spent nuclear fuel
- 7,000 MTHM of DOE high-level radioactive waste, commercial high-level radioactive waste, and DOE spent nuclear fuel.

The waste forms received at a potential repository will be in solid form. Materials that could ignite or react chemically at a level that would compromise containment or isolation will not be accepted by the potential repository. Neither the waste forms nor the waste packages will contain free liquids that could compromise waste containment. Materials that are regulated as hazardous waste under the Resource Conservation and Recovery Act of 1976 (42 U.S.C. 6901 et seq.) will not be disposed in the potential repository (DOE 1999c, Section 4.2.3).

Waste Package Overview—The design of a waste package is based on the characteristics of the waste forms that it would hold. Because commercial and DOE high-level radioactive waste forms have similar characteristics, both may be placed into a waste package of the same design. This has allowed the DOE to design waste packages capable of accommodating all the types of spent nuclear fuel and high-level radioactive waste currently generated or anticipated in the United States, whether commercial or governmental.

The waste package has been designed, in conjunction with the natural and other engineered barriers, to ensure compliance with applicable U.S. Nuclear Regulatory Commission (NRC) regulations, to contribute to safe operations during the preclosure phase, to make efficient use of the potential repository area, and to preserve the option of retrieving the waste. To perform its containment and isolation functions, the waste package described in this report has been designed to take advantage of a location in the unsaturated zone.

All the waste package designs consist of two concentric cylinders in which the waste forms would be placed. The inner cylinder would be composed of Stainless Steel Type 316NG. The outer cylinder would be made of a corrosion-resistant nickel-based alloy (Alloy 22). The waste package designs for DOE spent nuclear fuel and high-level radioactive waste are larger in diameter and thicker than those for commercial spent nuclear fuel. The outer layer of corrosion-resistant material protects the underlying layer of structural material from corrosion, and the structural material supports the thinner material of the outer layer.

Each waste package design has outer and inner lids. The outer (closure) lids would be made of Alloy 22. The inner lids would be made of Stainless Steel Type 316NG, and their thickness will vary, depending on the waste package design. In addition to the inner and outer lids, an Alloy 22 lid on the closure end of the waste package (flat closure lid) would provide additional protection against stress corrosion cracking in the closure weld area.

A titanium drip shield would be placed over the waste package before the repository is closed. This drip shield would protect the waste package against rockfall and dripping water. Figure 3-1 illustrates the waste package and drip shield materials.

Before the double-walled waste package is sealed, helium would be added as a fill gas. The helium will prevent oxidation of the waste form and help

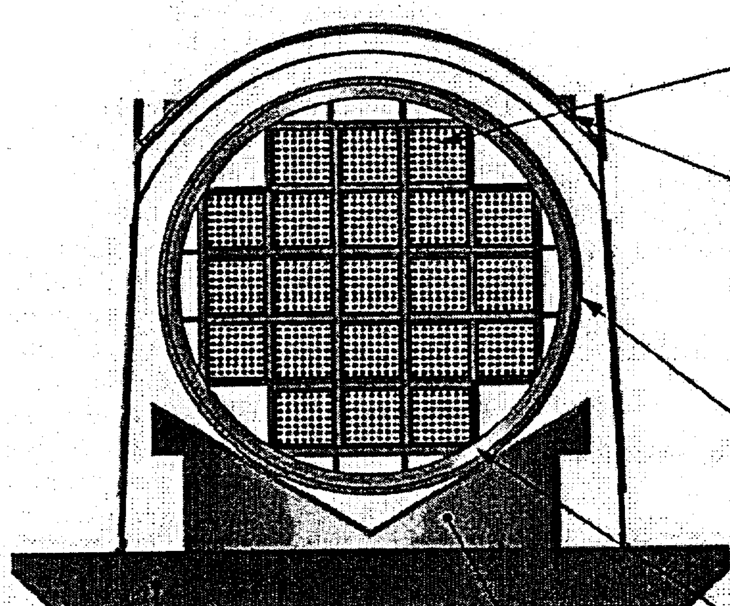

Figure 3-1. Cross-Sectional Illustration of an Alloy 22 and Stainless Steel Emplaced Dual-Metal Waste Package

The figure illustrates a waste package supported on an emplacement pallet and covered by a titanium drip shield, showing multiple engineered barriers that provide defense in depth. The use of engineered barriers of different materials protects against common mode failures.

transfer heat from the waste form to the wall of the inner shell of the waste package. Transferring heat away from the waste form is an important means of controlling waste form temperatures. This helps preserve the integrity of the metal cladding on the fuel rods, thus extending the life of an existing barrier to water infiltration.

All waste package designs will use a remote lifting-and-handling mechanism. The collar-sleeve-and-trunnion joint apparatus illustrated in Figure 3-2 will allow the necessary handling of the waste package before it is placed on an emplacement pallet and transferred to the designated drift. Each waste package would also have a unique permanent identifying label (CRWMS M&O 2000au, Section 1.2.1.14).

Although they share the features described previously, the waste package designs have different internal components to accommodate the different waste forms. For example, the waste package for uncanistered commercial spent nuclear fuel has an internal basket assembly to support fuel assemblies. In other waste packages (e.g., the high-level radioactive waste and DOE spent nuclear fuel waste packages), the internal basket has a different design, or, as is the case with naval spent nuclear fuel, the basket is contained inside the canister. Internal components are discussed in more detail in the respective waste package design sections.

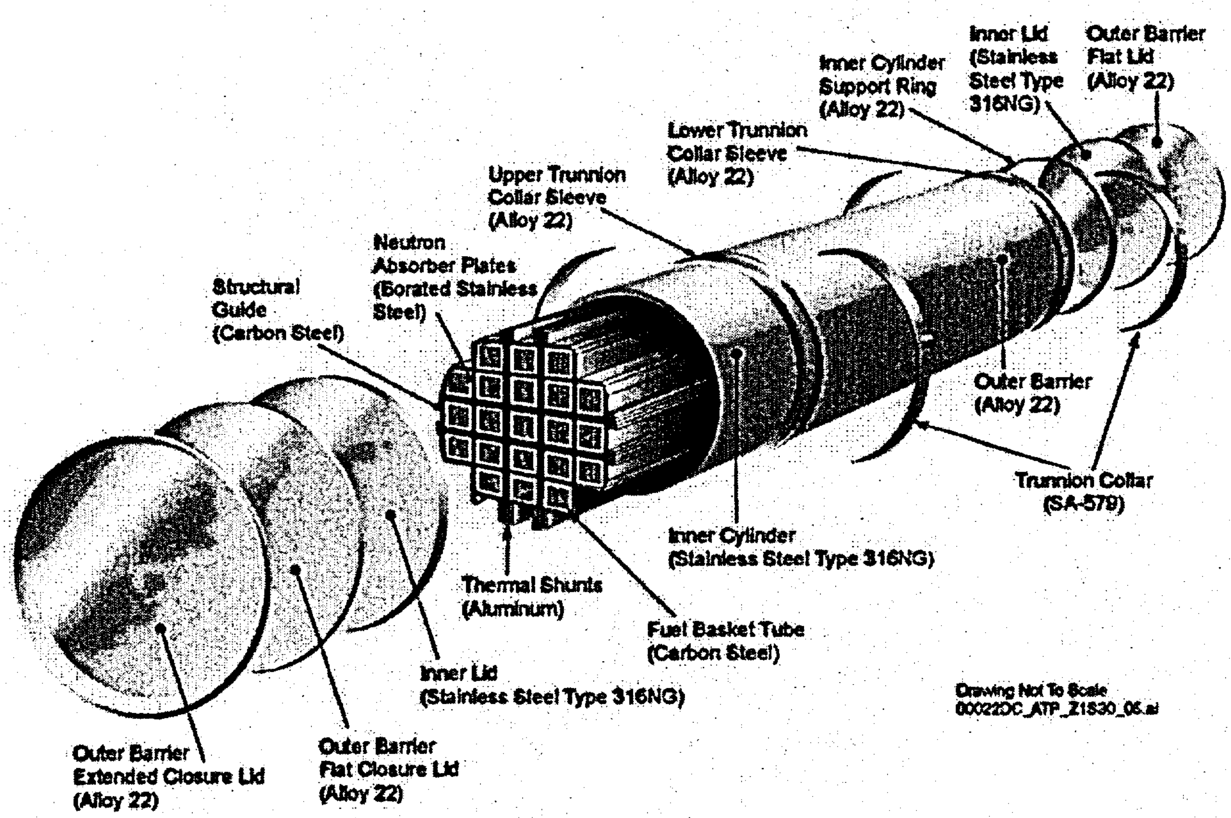

Figure 3-2. 21-PWR Absorber Plate Waste Package Design
Components for the 21-PWR commercial spent nuclear fuel assembly with an absorber plate (to prevent criticality) are illustrated. Internal strength is provided by carbon steel structural guides and fuel basket tubes. Aluminum thermal shunts assist in removing heat, and plates made of a stainless steel and boron alloy assist in preventing criticality by absorbing neutrons. Temporary trunnion collars would be installed on the waste package to facilitate handling operations. The inner cylinder support ring holds the inner cylinder to the outer barrier. All waste package designs have the same concept for lids, trunnions, and a multilayer configuration, although internal components vary among designs.

3.1 GENERAL DESIGN BASIS FOR THE WASTE PACKAGE

The engineered barrier system would be an important element of a potential repository. The primary component of the system would be the waste package. As defined in 10 CFR 63.2 (66 FR 55732), a waste package includes the waste form and any containers, shielding, packing, and other absorbent materials immediately surrounding it. The invert material does not immediately surround the waste package, so it is not considered part of the waste package. Figure 3-3 illustrates the waste package within the emplacement drift of the engineered barrier system.

The waste package has been designed to use materials that perform well under the anticipated conditions at Yucca Mountain. The design analyses performed on the waste package include evaluations of structural integrity, thermal performance, criticality safety, and shielding properties. In addition, data from the material and waste form testing programs have been used to model both the waste package and the cladding on the spent nuclear fuel as part of the total system performance assessment (TSPA) discussed in Section 4.

The waste packages emplaced in repository drifts will be affected by the atmosphere that surrounds them, the water that could come in contact with

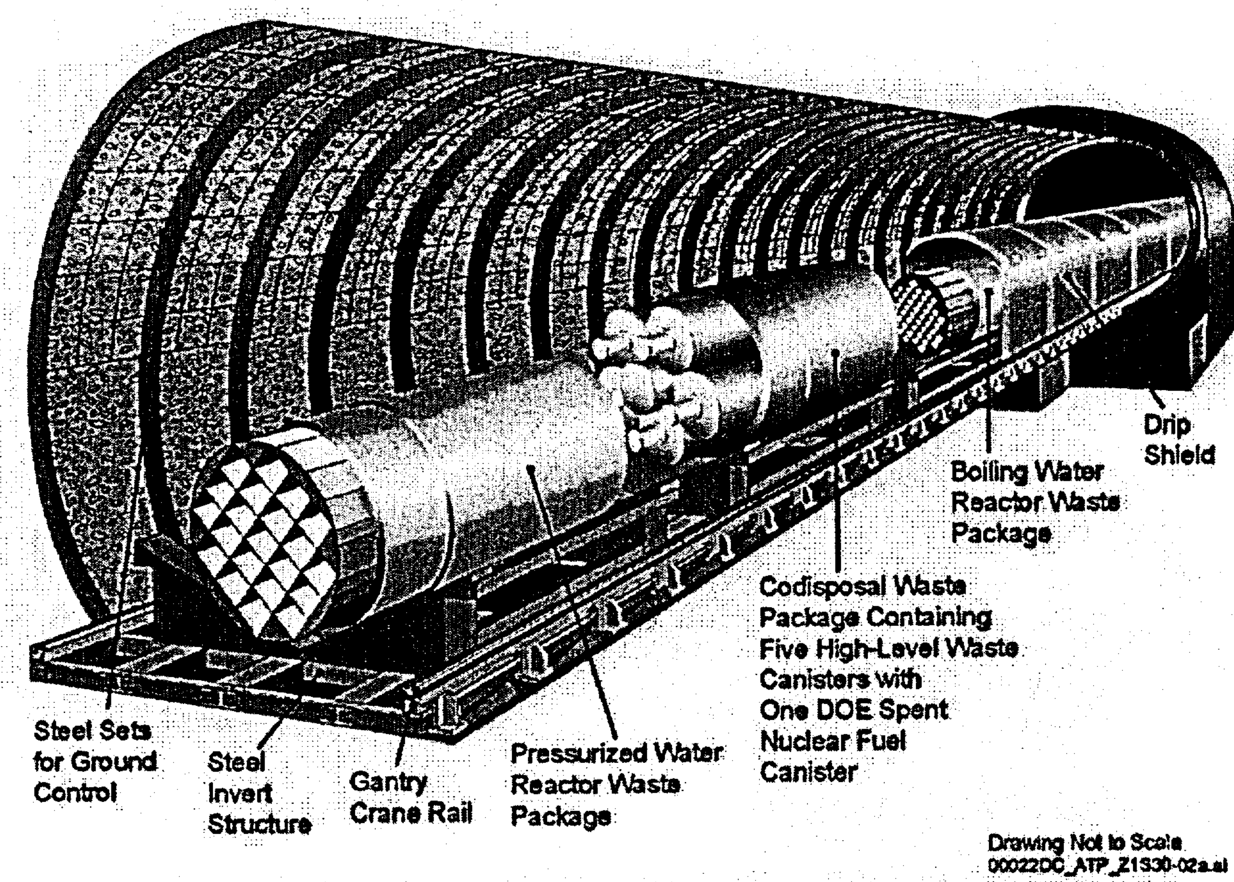

Figure 3-3. Schematic Illustration of the Emplacement Drift with Cutaway Views of Different Waste Packages
Ground support for emplacement drift walls is illustrated in the figure, which also shows three designs for dual-metal waste packages (representing various waste forms), a protective drip shield, and emplacement pallets supporting the waste package above the drift floor.

them, and the movement of the host rock in which they are emplaced. How the decay heat produced by waste forms is managed impacts the atmosphere surrounding the waste package. In the higher-temperature operating mode described in this report, fuel blending would be used to manage the amount of thermal output from the waste packages (i.e., how much heat they emit) to ensure that temperatures in most of the rock between the emplacement drifts stays below the boiling point of water.

3.1.1 Waste Package Functions

Waste containment begins when the waste form is sealed in the waste package. Once sealed, the waste package ensures a dry and stable physical and chemical environment for as long as it remains intact. Engineers have designed the waste package to work with the natural environment: the material for the outer barrier of the waste package was selected because of its resistance to corrosion in an environment such as the one expected at Yucca Mountain (CRWMS M&O 2000av).

The waste package performs a number of other functions. System Description Documents define each function as the basis for a waste package performance specification (CRWMS M&O 2000au, Section 1.1; CRWMS M&O 2000aw, Section 1.1; CRWMS M&O 2000ax, Section 1.1). The waste package, in conjunction with other systems, has been designed to:

- Restrict the transport of radionuclides
- Provide criticality protection during waste package loading and emplacement
- Manage the decay heat for the potential repository
- Provide identification (i.e., each waste package will be uniquely labeled and its contents identified)
- Enhance the safety of personnel, equipment, and the environment
- Prevent adverse reactions involving the waste form
- Maintain structural integrity during loading, onsite transportation, emplacement, and retrieval
- Resist corrosion in the emplacement drift environment
- Provide physical and chemical stability for the waste form
- Promote heat transfer between the waste form and the outside environment
- Facilitate decontamination of the waste package's outer surface.

The following sections discuss the general performance specifications that apply during repository operations (preclosure phase) and after closure of the repository (postclosure phase).

3.1.2 Preclosure Design Performance Specifications

The performance specifications for the functionality of the waste package during the repository's preclosure phase are consistent with 10 CFR 63.112(b) (66 FR 55732). This regulation provides for the DOE's analysis of the ability of the waste package's structures, systems, and components to perform their intended safety functions during an accident or event sequences. For the waste package, design basis events are determined by identifying the functions of the waste package and evaluating the effects on its performance of given events that could occur during normal handling of the waste package or during a credible accident scenario (i.e., events that have at least 1 chance in 10,000 of occurring before permanent closure of the geologic repository) (CRWMS M&O 2000ay, Section 4.2.1).

These event sequences and their effects on performance were defined by reviewing the *Preliminary MGDS Hazards Analysis* (CRWMS M&O 1996a), studying NRC standard review plans for similar facilities (e.g., the *Standard Review Plan for Dry*

Cask Storage Systems [NRC 1997]), and considering current surface and subsurface design information. This review process led to the classification of two types of event sequences that might affect waste packages during the preclosure period: internal (normal operations, mechanical or other failures, and operator error) and external (natural phenomena and man-made events not initiated by repository operations). The results constituted a bounding list of preclosure event sequences that could affect the waste packages. Using this list, engineers performed structural, thermal, and criticality analyses of the impacts such events could have on waste package performance (CRWMS M&O 2000ay). The event sequences were developed early in the design process and were based on conceptual designs for commercial spent nuclear fuel. The fuel design has evolved, but the event sequences evaluated still represent plausible accident scenarios that can be used to evaluate the adequacy of the waste package designs.

Table 3-1 summarizes the complete list of event sequences (CRWMS M&O 2000au, Section 1.2.2), which constitute the performance specifications for the waste package and support the waste package function specifications. In accordance with NRC guidance, designers considered many other types of events. However, because some events are either very low probability or their consequences are not significant, they were not included in the safety analysis. Section 3.5 presents the results of representative design evaluations for select waste packages.

3.1.3 Postclosure Performance Specification

10 CFR 63.113(b) (66 FR 55732) requires the entire repository system to meet specific dose limits for 10,000 years. The waste package is one of many barriers relied upon to meet this limit. The DOE's objective is to design a waste package that works in concert with the natural environment to meet performance standards while reducing the uncertainty associated with the current understanding of natural processes at the site.

3.1.4 Design Descriptions

An analysis was undertaken to determine the number of designs needed to handle the different waste forms that would constitute the anticipated waste stream in the most economical manner (CRWMS M&O 1997b). The objective of the evaluation was to determine:

- The number of different waste package designs needed
- The capacity of each waste package design (i.e., the amount of waste it would hold)
- The limits on spent nuclear fuel properties (e.g., age, thermal characteristics) that might apply to each waste package design.

The complete system of waste package designs is intended to allow reliable disposal of those waste forms that a repository would accept while still enhancing overall efficiencies.

To determine the most efficient set of waste package designs for commercial spent nuclear fuel, the DOE designed waste packages of various assembly-holding capacities and incorporated into the design methods for removing decay heat and preventing criticality. This resulted in the selection of a set of five waste package designs as the most efficient means of accommodating the anticipated waste stream of commercial spent nuclear fuel. A similar process led to three designs for DOE non-naval spent nuclear fuel and DOE and commercial high-level radioactive waste. Two other designs are specific to naval spent nuclear fuel, which will arrive presealed in canisters (CRWMS M&O 2000az, Sections 4.2 and 4.3). Some DOE non-naval spent nuclear fuel will be loaded into waste packages with high-level radioactive waste; this DOE spent nuclear fuel and high-level radioactive waste will also arrive in presealed canisters. Table 3-2 lists the waste package designs and a description of each. Table 3-3 provides a breakdown of the percentage of waste packages by waste package design and the percentage of MTHM by waste package design. Figures 3-4 and 3-5 illustrate the waste forms and associated waste package designs.

Table 3-1. Bounding Event Sequences for Waste Packages

Analysis Type	Event Group	Event	Performance Specification
Structural	Falling Objects—Side Impact on Waste Package	Rockfall from the drift onto the waste package	Withstand 13-metric ton (14-ton) rock falling 3 m (10 ft). Drop height based on a 5.5-m (18-ft) drift and a distance of 2.4 m (8 ft) between the top of the drift and the top of the waste package (CRWMS M&O 1999h, p. 40)[a]
	Falling Objects—End of Waste Package Impact	Handling equipment drop onto the waste package	Withstand 2.3-metric ton (2.5-ton) object falling 2 m (6.6 ft). Drop height based on the distance between the handling equipment and the top of the waste package
	Waste Package Vertical Drops and Waste Package End Collisions	Waste package vertical drop from the disposal container cell crane	Withstand 2-m (6.6-ft) drop. Drop height based on the maximum crane hook height; the bottom of the waste package cannot be lifted higher than 2 m (6.6 ft) above the floor
	Waste Package Horizontal Drops and Waste Package Side Collisions	Emplacement drift gantry drops waste package	Withstand 2.4-m (8-ft) drop
	Puncture Hazards	Waste package falls onto a sharp object while being transported in a horizontal position	Withstand 2-m (6.6-ft) horizontal drop onto a steel support or 2.4-m (8-ft) horizontal drop onto a concrete pier, whichever is worse
	Tipover	Tipover due to vertical drop or seismic event	Withstand tipover from a vertical position onto a flat surface
	Seismic Activity	Earthquake	Maintain structural integrity and prevent tipover during a design basis earthquake
	Missile	The missile identified was a valve stem being ejected at the surface facility	Withstand impact of a valve stem weighing 0.5 kg (1.1 lb), with a 1-cm (0.39-in.) diameter, inside a valve with 5 cm (2 in.) of packing and under a system pressure of 2.1 MPa (305 psi), which has become a missile with a velocity of 5.7 m/s (19 ft/s)
	Transporter Runaway	Failure to maintain the transporter at or below the maximum speed limit	Withstand maximum impact from a transporter runaway, derailment, and impact at a speed of 63 km/hr (39 mi/hr)
	Fuel Rod Rupture/Internal Pressurization	100% fuel rod rupture and fission gas release	Withstand internal pressure of 1 MPa (146 psi)
Thermal and Structural	Thermal Stresses and Peak Waste Package Temperature	Fire in disposal container cell	Survive a fire, defined as exposure of whole waste package for not less than 30 minutes to a heat flux not less than that of a thermal radiation environment of 800°C (about 1,500°F) with an emissivity coefficient of at least 0.9. Surface absorptivity must be at least 0.8. If significant, convective heat transfer must be considered on the basis of still air at 800°C (about 1,500°F).
Criticality	Criticality Safety	Criticality scenario inside a waste package	The effective multiplication factor (k_{eff}) is less than or equal to 0.95 under assumed accident conditions, considering allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation

NOTE: [a]This rock size requirement was lowered to 6 metric tons (BSC 2001m) since completion of the rock fall analysis in support of a potential site recommendation (CRWMS M&O 2000au, Section 2.5.2.1).

Table 3-2. Waste Package Design

Waste Package Design	Description
21-PWR Absorber Plate	Capacity: 21 commercial pressurized water reactor assemblies and an absorber plate for preventing criticality.
21-PWR Control Rod	Capacity: 21 commercial pressurized water reactor assemblies with higher reactivity, requiring additional criticality control that is provided by the placement of control rods in all assemblies.
12-PWR Long	Capacity: 12 commercial pressurized water reactor assemblies and an absorber plate for preventing criticality; longer than the fuel assemblies placed in the 21-PWR packages. Because of its smaller capacity, it may also be used for fuel with higher reactivity or thermal output.
44-BWR	Capacity: 44 commercial boiling water reactor assemblies and an absorber plate for preventing criticality.
24-BWR	Capacity: 24 commercial boiling water reactor assemblies with higher reactivity, requiring a thicker absorber plate to prevent criticality than that used in the 44-BWR design.
5-DHLW/DOE SNF Short	Capacity: 5 short high-level radioactive waste canisters and 1 short DOE SNF canister. When high-level radioactive waste includes immobilized plutonium cans, no DOE spent nuclear fuel is placed in the center.[a]
5-DHLW/DOE SNF Long	Capacity: 5 long high-level radioactive waste canisters and 1 long DOE SNF canister.[a]
2-MCO/2-DHLW Long	Capacity: 2 DOE multicanister overpacks and 2 long high-level radioactive waste canisters.
Naval SNF Short	Capacity: 1 short naval SNF canister.
Naval SNF Long	Capacity: 1 long naval SNF canister.

NOTE: [a]DOE non-naval spent nuclear fuel

Table 3-3. Breakdown of Waste Packages for 70,000 MTHM

Waste Package Design	Approximate Percentage of Waste Packages by Waste Package Design	Approximate Percentage of MTHM by Waste Package Design
21-PWR Absorber Plate	38%	55%
21-PWR Control Rod	1%	1%
12-PWR Long	2%	2%
44-BWR	25%	32%
24-BWR	1%	<1%
5-DHLW/DOE SNF Short[a]	14%	3%
5-DHLW/DOE SNF Long[a]	15%	4%
2-MCO/2-DHLW Long	1%	<1%
Naval SNF Short	2%	<1%
Naval SNF Long	1%	<1%

NOTE: [a]DOE non-naval spent nuclear fuel

3.2 COMMERCIAL SPENT NUCLEAR FUEL

Commercial nuclear fuel rods are arranged in assemblies that range in length from about 2 to 5 m (6.6 to 16 ft). These assemblies are arranged in a square, cross-sectional pattern and customized to meet the size and performance requirements of the reactor they will fuel. The fuel rods are sealed metal tubes, about 6.5 to 12.7 mm (0.26 to 0.50 in.) in diameter, that contain ceramic-like fuel pellets. The fissionable material in the fuel rods is uranium dioxide. Fissionable material has the ability to sustain a controlled nuclear chain reaction and, in so doing, release energy in a controlled manner. Spent nuclear fuel contains uranium-235 and uranium-238, short-lived fission products such as strontium-90 and cesium-137, and long-lived transuranic isotopes (i.e., isotopes with atomic numbers greater than 92) such as plutonium-239 and americium-243.

In most nuclear fuel assemblies, the tubes containing the fuel pellets are made of Zircaloy, a zirconium-based material. The generic name for the metal that the tubes are composed of is "cladding." Zirconium-based cladding is used for

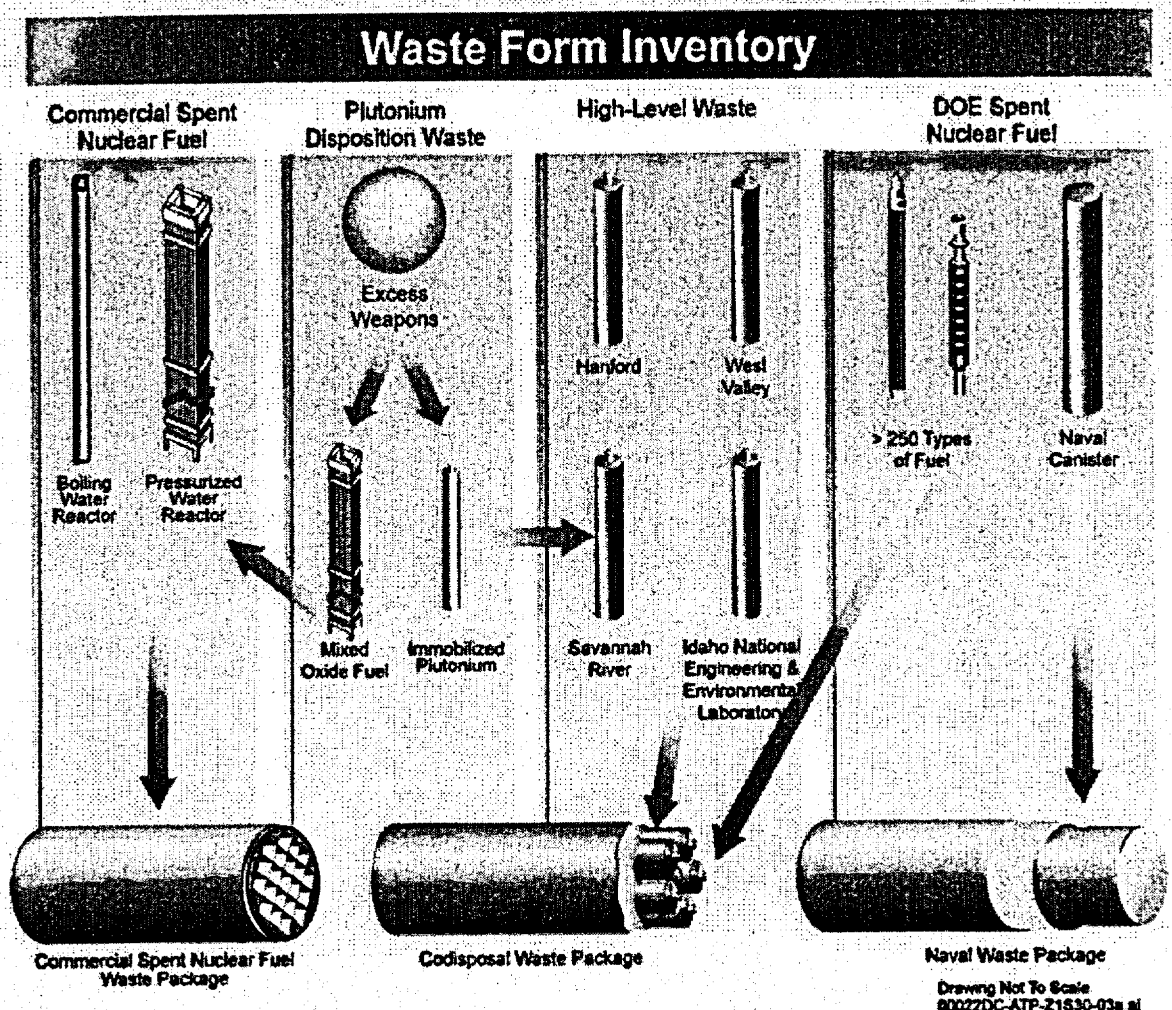

Figure 3-4. Waste Form Inventory
The figure depicts the types and quantities of waste forms to be disposed in a first repository and their representative waste package designs. Until a second repository is in operation, the Nuclear Waste Policy Act limits inventory in the first repository to 70,000 MTHM. Dispositioning excess plutonium from weapons programs into mixed-oxide fuel for burnup in reactors and disposal in a repository aids in the fight against nuclear proliferation.

98.5 percent of pressurized water reactor fuel assemblies and 99.8 percent of boiling water reactor fuel assemblies. The cladding on the remainder is made of stainless steel. Future fuel designs are not expected to change from mostly zirconium-based cladding (CRWMS M&O 1999a, Section 3.1.1). Figure 3-6 illustrates a typical commercial nuclear fuel assembly for a pressurized water reactor.

Approximately 292,000 commercial spent nuclear fuel assemblies will be generated by 2040: 167,000 from boiling water reactors and 125,000 from pressurized water reactors (CRWMS M&O 1999a, Section 3.1, Tables 3 and 4). About 220,000 of these assemblies would be emplaced in the potential repository. Up to 33 metric tons of U.S. surplus weapons-usable plutonium will be fabricated into uranium–plutonium fuel (called mixed-oxide fuel) and irradiated in commercial reactors. Use of mixed-oxide fuel will be limited to only a few specific commercial reactors and would involve, at most, no more than 1,800 assemblies (CRWMS M&O 1999a, Appendix B). Mixed-oxide spent nuclear fuel would become part of the commercial waste stream accepted for disposal at a repository.

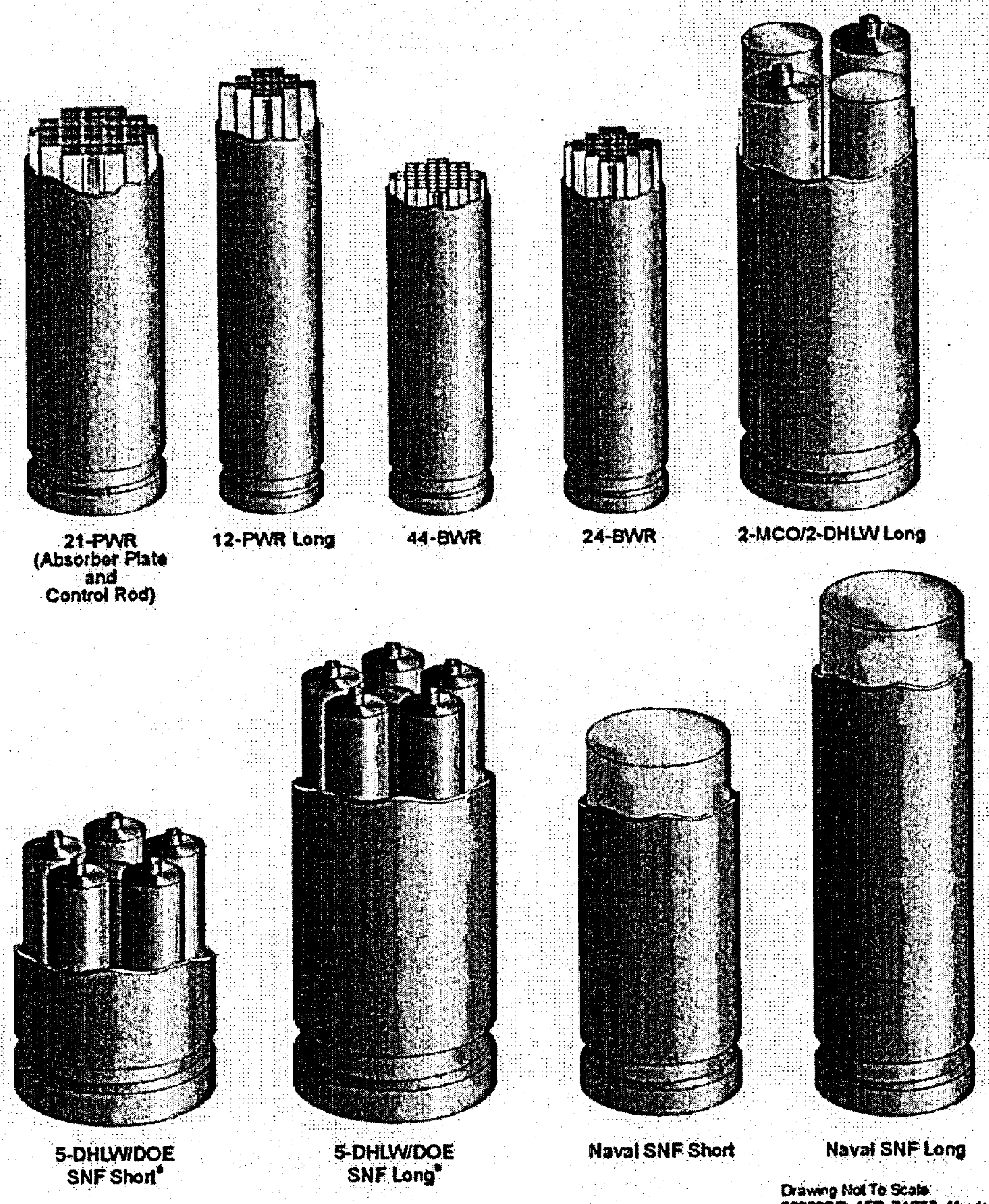

Figure 3-5. Waste Package Designs with Waste Forms
[a]Basket is excluded from illustration for clarity.
The waste package designs are depicted close to scale, along with the waste form configuration contained in each design. The 21-PWR waste package is representative of both the 21-PWR Absorber Plate and 21-PWR Control Rod designs.

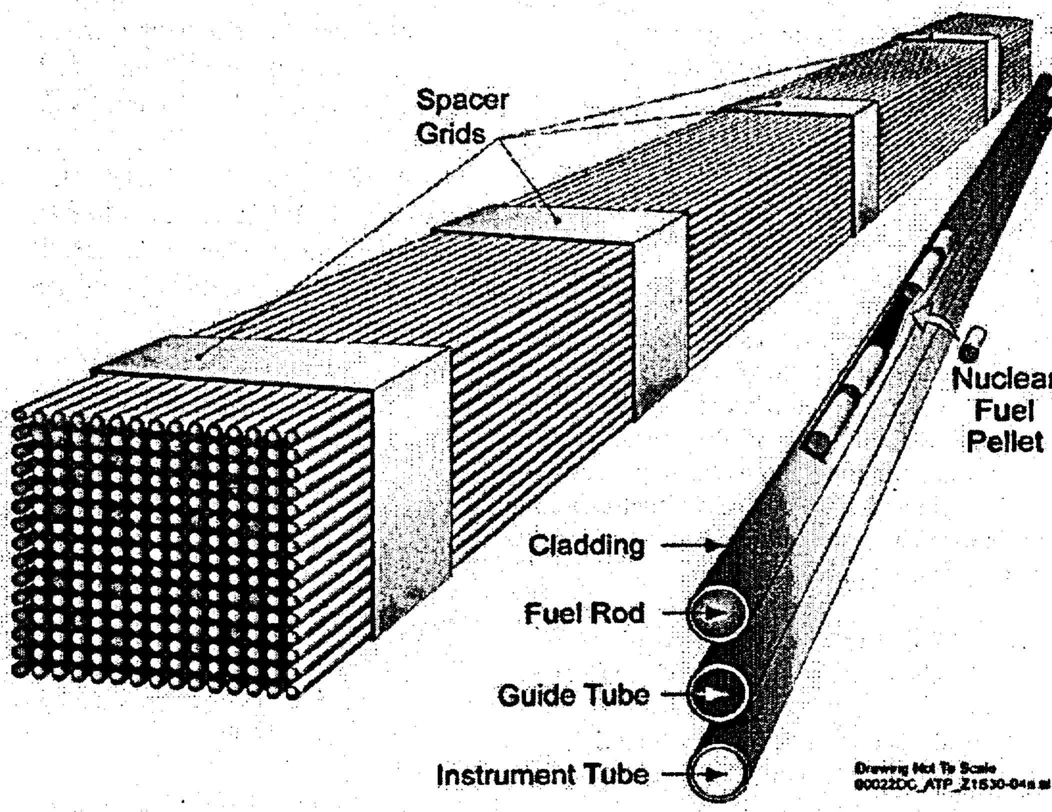

Figure 3-6. Cross-Sectional Illustration of a Typical Pressurized Water Reactor Fuel Assembly
The figure shows the spacer grids, which provide structural strength and maintain the cross-sectional configuration of the assembly; clad fuel rods, which contain fingertip-sized, ceramic-like fuel pellets that become highly radioactive in a reactor; and guide and instrument tubes. The array depicted contains 208 fuel rods, which would weigh about 550 kg (1,200 lbs), and would be 4.3 m (14 ft) long.

Each mixed-oxide fuel assembly irradiated and disposed would replace an energy-equivalent enriched uranium assembly. Preliminary evaluations indicate that mixed-oxide fuel can be accommodated within the suite of waste package designs (CRWMS M&O 2000ba, p. vii).

In addition to standard commercial fuel assemblies, a small portion (less than 2 percent) of the spent nuclear fuel will arrive in canisters containing individual fuel rods. Utilities repackage fuel rods that have damaged cladding in these canisters to confine radioactive materials during handling and shipment. To ensure that waste package designs have the flexibility to accommodate canistered fuel, the canisters would have sizes within the range of dimensions that qualify as standard fuel. Thus, it will be possible to handle and dispose canistered fuel in the same way as uncanistered spent nuclear fuel assemblies (CRWMS M&O 1999a, Section 3.2).

Most commercial spent nuclear fuel assemblies would arrive at the potential repository undamaged and suitable for immediate disposal. Some of the fuel rods in these assemblies are expected to have minor defects in their cladding (i.e., small cracks or pinholes due to manufacturing defects or corrosion). The initial condition of the cladding is considered in the TSPA analysis (see Section 4.2.6).

Based on prior experience, testing, and the design of the system and equipment, transport will not impact the integrity of spent nuclear fuel. Assemblies, canistered or not, would be placed intact into waste packages. The entire assembly, which may include nonfuel hardware components (such as control rods), would be packaged for disposal.

3.2.1 Commercial Spent Nuclear Fuel: Assigning the Right Waste Package

The characteristics of spent nuclear fuel assemblies (i.e., size, thermal output, and reactivity) will be used to select the appropriate waste package design. The size of assemblies is used to determine the size and configuration of the fuel within the waste package, and to perform structural analyses to evaluate the integrity of the waste package during normal handling and event sequences. The thermal output and reactivity of the fuel is used to determine which waste package can accommodate each given fuel assembly.

3.2.1.1 Physical Characteristics of Commercial Spent Nuclear Fuel

The physical characteristics of commercial spent nuclear fuel include length, cross section, weight, and cladding. Table 3-4 summarizes boiling water reactor assembly dimensions and weights by related groups; Table 3-5 provides similar information for pressurized water reactor assemblies (CRWMS M&O 1999a, Section B.1.1). The information provided in Tables 3-4 and 3-5 represents the full inventory of approximately 292,000 assemblies. Operating and shut down reactors are shown in Figure 1-2 of Section 1.

Based on physical dimensions, approximately 95 percent of the fuel assemblies can be emplaced using two waste package designs (the 21-PWR Absorber Plate and the 44-BWR). The 12-PWR Long is designed to accommodate assemblies from Combustion Engineering and the South Texas Project, which are longer than the others. Because the 12-PWR Long holds fewer assemblies, it can also be used to manage thermal load and criticality concerns about fuel with higher thermal output and reactivity.

Table 3-4. Design Basis Dimensions of Assemblies for Boiling Water Reactors

Assembly Group	Length mm (in.)	Width mm (in.)	Weight kg (lb)	Percent of Total BWR Assemblies	Total BWR Assemblies
Big Rock Point	2,071.6 to 2,154.0 (81.6 to 84.8)	165.1 to 183.1 (6.5 to 7.2)	207 to 268 (456 to 591)	<1	524
Humbolt Bay, Dresden 1, and LaCrosse[a]	2,413.0 to 3,591.5 (95.0 to 141.4)	101.6 to 160.0 (4.0 to 6.3)	125 to 218 (276 to 481)	<1	1,615
General Electric BWR (8x8 and 9x9)	4,343.4 to 4,521.2 (171.0 to 178.0)	132.1 to 154.9 (5.2 to 6.1)	252 to 329 (556 to 725)	99	164,800

NOTES: [a]See Figure 1-2 in Section 1 for a map of currently operating and shut down reactors.
BWR = boiling water reactor.

Table 3-5. Design Basis Dimensions of Assemblies for Pressurized Water Reactors

Assembly Group	Length mm (in.)	Width mm (in.)	Weight kg (lb)	Percent of Total PWR Assemblies	Total PWR Assemblies
Haddam Neck, Indian Point 1, San Onofre 1, and Yankee Rowe[a]	2,837.1 to 3,561.1 (111.7 to 140.2)	160.0 to 218.4 (6.2 to 8.6)	198 to 731 (437 to 1,612)	2	2,460
Westinghouse, Babcock & Wilcox, and Others	3,708.4 to 4,406.9 (146.0 to 173.5)	198.1 to 218.4 (7.8 to 8.6)	497 to 773 (1,100 to 1,700)	84	105,500
Combustion Engineering 16x16 and South Texas Project	4,490.7 to 5,110.5 (176.8 to 201.2)	203.2 to 215.9 (8.0 to 8.5)	649 to 882 (1,430 to 1,940)	14	16,900

NOTES: [a]See Figure 1-2 in Section 1 for a map of currently operating and shut down reactors.
PWR = pressurized water reactor.

3.2.1.2 Thermal Output

Commercial spent nuclear fuel arriving at the repository is expected to have a wide range of thermal outputs. The waste package has been designed to ensure that this anticipated range can be accommodated in a way that supports the range of thermal operating modes being considered for the potential repository (see Section 2).

The key factors used to determine the thermal output of spent nuclear fuel are its age (i.e., number of years out of the reactor), burnup (measured in gigawatt-days per metric ton of uranium), and initial enrichment of fissile material (i.e., uranium-235 or plutonium). To cover anticipated thermal outputs, waste package designers considered the average characteristics provided in Table 3-6. Maximum characteristics were also evaluated to ensure that these fuel assemblies could be placed in the waste package.

Determining which waste package design can accommodate a particular spent nuclear fuel assembly thermally will require calculating the assembly's thermal output at the time it is emplaced in the repository. The appropriate waste package is chosen to ensure that the maximum thermal output limit is not violated. In the higher-temperature operating mode, this limit has been set at 11.8 kW.

Of the five commercial waste package designs, the 21-PWR Absorber Plate is the most limiting in thermal output. The characteristics of spent nuclear fuel assemblies loaded into this waste package type will be carefully chosen to ensure that the thermal output limit is not violated.

The thermal output of the waste package can be reduced, if necessary, to accommodate either a range of thermal operating modes or potential changes in the characteristics of the waste stream. Reduction can be achieved by using one or more of the following waste package loading strategies: (1) fuel blending (i.e., combining low heat output fuel and high heat output fuel within a single waste package); (2) de-rating (i.e., loading fewer assemblies than the waste package is designed to hold); or (3) increasing the use of the 12-PWR Long waste package (i.e., placing high heat output fuel in smaller waste packages). For more information on variables that can be modified to accommodate a range of operating modes, see Section 2.1.

3.2.1.3 Criticality Control

Waste package designs will be evaluated to ensure that subcritical limits can be met, as well as the thermal limits described in the previous section. As Section 3.5.2 describes in detail, loading curves will be developed from commercial spent nuclear fuel parameters to determine the method of loading waste packages. This will ensure that the reactivity of the fuel being loaded is below the level at which criticality could occur.

3.2.2 Commercial Spent Nuclear Fuel Waste Package Designs

All the waste package designs for commercial spent nuclear fuel have similar components that perform multiple functions. Figure 3-2 illustrates a representative waste package design. The general features of waste package design were described in Section 3.1; the internal components of commercial spent nuclear fuel waste package design are described in this section, along with their functions.

Table 3-6. Fuel Assembly Characteristics at Arrival

Assembly Type	Average Assembly Age (years)	Average Assembly Burnup (GWd/MTHM)	Average Assembly Initial ^{235}U Enrichment (wt%)	Weight of Heavy Metal (MTHM)
Boiling water reactor	22.7	33.6	3.03	0.200
Pressurized water reactor	23.1	41.2	3.75	0.475

NOTES: GWd = gigawatt day. Source: CRWMS M&O 2000bb, Table 5.

3.2.2.1 Internal Basket Design

Baskets are composed of interlocking plates, fuel tubes, thermal shunts, and structural guides. These elements will displace any water that might be present inside a breached waste package, helping to prevent criticality. All four elements and their functions are described below.

Interlocking Plates—The interlocking plates set the pattern for how the fuel assemblies will be arranged inside the waste package. The basket for each design is customized to meet requirements for the size, type, and number of fuel assemblies it can hold, as well as the specific waste form being packaged. The material composition and thickness of the interlocking plates are tailored to provide enough structural strength to maintain fuel geometry during normal handling and design basis events, and to prevent criticality. However, this extra durability is not considered in the structural analyses (CRWMS M&O 2000au). The interlocking plates are made of either Neutronit A 978 (a stainless steel and boron alloy) or SA 516 Grade 70 carbon steel and range between 5 and 10 mm (0.2 and 0.4 in.) in thickness.

The neutron absorber materials that prevent criticality can be placed directly into the plates using Neutronit A 978 or can take the form of separate control rods. Plates that include neutron absorber material will vary in thickness because of the number of plates in the design. For example, the 44-BWR waste package has more plates than the 21-PWR waste package; therefore, the plates in the 44-BWR can be thinner, but the entire waste package will still have about the same mass of neutron absorber material as the 21-PWR.

The corrosion rate for Neutronit A 978 plates is slow, and the plates tend to corrode by pitting. Because of this, the plates remain in place between the fuel assemblies even as they corrode. To avoid processes that could accelerate stress corrosion cracking, there will be no bends or structural welds on the Neutronit A 978 plates.

Fuel Tubes—The fuel tubes are long, square containers that line the insides of the cavities created by the interlocking plates. They support the internal structure created by the interlocking plates while holding the fuel assemblies in place. The fuel tubes provide structural strength for the internal basket during event sequences they also help conduct heat away from the cladding. The fuel tubes for each waste package design are made of SA 516 Grade 70 carbon steel that is 5 mm (0.2 in.) thick (CRWMS M&O 2000au, Section 2.4.1.2).

Thermal Shunts—All the waste package designs for commercial spent nuclear fuel except the 24-BWR require thermal shunts. These shunts, which are made of 5-mm (0.2-in.) thick SB 209 6061 T4 (an aluminum alloy), are placed alongside the interlocking plates (CRWMS M&O 2000au, Section 2.4.1.3). The shunts are added to help transfer heat from the waste form to the walls of the waste package. Adding thermal shunts is a simple and effective method to improve heat conduction between the center of the waste package and the outer edge of the internal basket, providing a reliable means of keeping the temperature of the cladding within design limits. Limiting cladding temperatures helps protect the waste form by minimizing damage to the fuel cladding (CRWMS M&O 2000au, Section 2.4.1.3).

Structural Guides—The structural guides for each waste package are made of 10-mm (0.4-in.) thick SA 516 Grade 70 carbon steel and are placed inside the inner layer of the waste package to hold the basket structure in place. They help maintain fuel geometry, which can prevent criticality during event sequences. The structural guides also help conduct heat from the waste form to the walls of the waste package, where it is radiated to the surrounding drift walls (CRWMS M&O 2000au, Section 2.4.1.4).

3.2.2.2 Control Rods

Control rods similar to those used in reactors will be placed in waste packages that need additional long-term criticality control, such as those containing highly reactive fuel assemblies from pressurized water reactors. Control rods are made of boron carbide and have Zircaloy cladding. Because this is the same material used in most fuel rod cladding, it will have similar corrosion properties and longevity.

3.2.3 Preliminary Engineering Specifications for the Commercial Spent Nuclear Fuel Waste Package Designs

The preliminary engineering specifications for the waste package design include the waste form characteristics, the physical dimensions of the waste package, and material specifications. Tables 3-7 and 3-8 provide preliminary engineering specifications for the waste package designs for commercial spent nuclear fuel based on the physical dimensions, thermal output, and reactivity of the fuel. Table 3-9 shows the material specifications of the waste package components. These engineering specifications were developed to meet the performance specifications given in Table 3-1.

3.3 U.S. DEPARTMENT OF ENERGY SPENT NUCLEAR FUEL, HIGH-LEVEL RADIOACTIVE WASTE, AND IMMOBILIZED PLUTONIUM

Ten types of canisters of DOE spent nuclear fuel and high-level radioactive waste may be received at the potential repository (CRWMS M&O 2000az, Sections 4.2 and 4.3):

1. Naval spent nuclear fuel canisters, short
2. Naval spent nuclear fuel canisters, long
3. DOE spent nuclear fuel canisters, short
4. DOE spent nuclear fuel canisters, long

Table 3-7. Physical Dimensions of Commercial Waste Package Designs

No.	Waste Package Design	Outer Diameter mm (in.)	Outer Length mm (in.)	Mass of Empty WP kg (lb)	Mass of Loaded WP kg (lb)
1	21-PWR Absorber Plate	1,644 (64.7)	5,165 (203.3)	26,000 (57,300)	42,300 (93,300)
2	21-PWR Control Rod	1,644 (64.7)	5,165 (203.3)	26,000 (57,300)	42,300 (93,300)
3	12-PWR Long	1,330 (52.4)	5,651 (222.5)	19,500 (43,000)	30,100 (66,400)
4	44-BWR	1,674 (65.9)	5,165 (203.3)	28,000 (61,700)	42,500 (93,700)
5	24-BWR	1,318 (51.9)	5,105 (201)	19,400 (42,800)	27,300 (60,200)

NOTES: Control rods do not add any mass to the package because they displace the mass of nonfuel components (e.g., existing control rods, in-core detectors) included in the fuel assembly mass. WP = waste package. Source: CRWMS M&O 2000au; BSC 2001n.

Table 3-8. Commercial Spent Nuclear Fuel Characteristics by Waste Package Design

Waste Package Design	Average Waste Package Heat Generation Rate Based on Assembly Heat at Repository Arrival (kW)	Number of Assemblies	Assembly Average Burnup (GWd/MTHM)	Assembly Average Initial ^{235}U Enrichment (wt)	Average MTHM per Assembly	Average Assembly Age (years)
21-PWR Absorber Plate	11.53	90,262	41.5	3.74	0.430	23.00
21-PWR Control Rod	3.11	1,992	19.6	3.57	0.368	36.14
12-PWR Long	9.55	1,855	46.3	4.01	0.540	18.04
44-BWR	7.38	124,532	34.1	3.04	0.177	22.41
24-BWR	0.52	2,013	8.1	2.63	0.167	40.32

NOTES: Based on 63,000 MTHM. GWd = gigawatt day. Source: CRWMS M&O 2000bb, Table 10.

Table 3-9. Waste Package Design Component Materials

Component	Material
Dual-layer design: Inner structural shell Outer corrosion-resistant barrier	 Stainless Steel Type 316NG Alloy 22 (SB 575 N06022)
WP fill gas	Helium
Fuel tubes for commercial SNF WP basket design	Carbon steel (SA 516 Grade 70)
Neutron absorber interlocking plates for commercial SNF WP	Neutronit A 978 (borated 316 stainless steel)
Interlocking plates for 21-PWR Control Rod design	Carbon steel (SA 516 Grade 70)
Structural guides for commercial SNF WP basket design	Carbon steel (SA 516 Grade 70)
Canister guide for 5-DHLW/DOE SNF designs	Carbon steel (SA 516 Grade 70)
Thermal shunts for commercial SNF WP basket design	Aluminum plate (SB 209 6061 T4)

NOTES: SNF = spent nuclear fuel; WP = waste package.

5. Larger-diameter DOE spent nuclear fuel canisters, short
6. Larger-diameter DOE spent nuclear fuel canisters, long
7. Solidified high-level radioactive waste canisters, short
8. Solidified high-level radioactive waste canisters, long
9. Solidified high-level radioactive waste canisters containing immobilized plutonium cans, short
10. Multicanister overpacks containing spent nuclear fuel from the Hanford N Reactor.

The number of canisters of solidified high-level radioactive waste will greatly exceed the number of canisters of DOE spent nuclear fuel. Therefore, the DOE has developed an efficient arrangement for packing them together (CRWMS M&O 2000az, Section 4.2). This mixing of DOE spent nuclear fuel and high-level radioactive waste is called "codisposal." Codisposal also helps maintain criticality control for DOE spent nuclear fuel that contains highly enriched uranium. Naval spent nuclear fuel canisters, which are larger in diameter, will not be placed in codisposal waste packages; they will be placed one canister per waste package. Because the waste package designs being considered will contain both DOE spent nuclear fuel and high-level radioactive waste, the following section describes both waste forms, as well as the appropriate waste package designs.

3.3.1 U.S. Department of Energy Spent Nuclear Fuel

DOE spent nuclear fuel has a wide variety of physical, chemical, and nuclear characteristics and represents an inventory of approximately 2,500 MTHM; 2,333 MTHM of this is included in the waste allocation for disposal in the first repository (DOE 1999d, Section 8.1). The waste packages designed for DOE spent nuclear fuel will accept fuel irradiated at DOE facilities, naval spent nuclear fuel, and certain types of material irradiated at commercial nuclear reactors, including debris from the Three Mile Island-2 reactor and fuel from the Fort Saint Vrain reactor. All DOE waste canisters will be sealed before they are transported to the potential repository.

The largest single component of the DOE spent nuclear fuel inventory by weight is uranium metal fuel, at approximately 2,130 MTHM (DOE 1999d, Appendix C, Section 5.1, Table 1). Fuel from the N Reactor at Hanford, Washington, accounts for 2,100 MTHM of this inventory. During its 20-year life, the N Reactor produced nuclear isotopes for defense purposes. N Reactor fuel has an initial enrichment of less than 2 percent uranium-235. It will be placed in multicanister overpacks that will both store the waste onsite and transport it to the potential repository. The multicanister overpack is a stainless steel container that is slightly wider at the top than at the bottom (DOE 1999d, Appendix C, Section 5.1, Table 1). Although N Reactor fuel is the largest portion of the DOE spent nuclear fuel inventory by weight, it will be emplaced in the repository in only one percent of the waste packages (Table 3-3).

Approximately 184 MTHM of the DOE inventory is low-enriched uranium oxide, some of which is standard commercial spent nuclear fuel used for testing. Some is the fuel debris from the damaged reactor core at Three Mile Island-2, which is already stored in small canisters that can be placed inside a standard DOE canister. The DOE canister can then be inserted into a transportation cask and transported to the potential repository (DOE 1999d, Appendix C, Section 5.1, Table 1).

Approximately 125 MTHM of the DOE inventory includes uranium enriched initially to more than 20 percent uranium-235, uranium enriched initially to between 5 and 20 percent uranium-235, and thorium- and plutonium-based fuels (DOE 1999d, Appendix C, Section 5.1, Table 1).

Naval nuclear fuel is designed to operate in a high-temperature, high-pressure environment for many years. Naval fuel is highly enriched. To ensure it can withstand battle-shock loads, naval fuel is surrounded by large amounts of structural material made of Zircaloy. There are two canister designs for naval fuel; both use similar materials and mechanical arrangements. The DOE plans to emplace about 65 MTHM of naval spent nuclear fuel in the potential repository. This fuel will be contained within about 300 sealed canisters, which will be transferred directly from transport casks into waste packages (DOE 1999d, Appendix C, Section 5.1, Table 1).

3.3.1.1 Physical Characteristics

The canisters for DOE spent nuclear fuel will be standardized to efficiently utilize the waste package design (CRWMS M&O 2000aw). Table 3-10 gives the preliminary canister dimensions.

3.3.1.2 Thermal Output

DOE spent nuclear fuel has a low thermal output, with naval spent nuclear fuel being the hottest. The maximum thermal output of a naval spent nuclear fuel canister is 8.01 kW, which is well below the maximum limit of 11.8 kW (CRWMS M&O 2000ax, Section 2.5.4.2).

3.3.1.3 Criticality Control

The controlling factors in disposal criticality analyses of DOE spent nuclear fuel are fuel matrix,

Table 3-10. U.S. Department of Energy Waste Forms for Disposal, According to Waste Package Design

Waste Package Design	Canister Length mm (in.)		Canister Diameter mm (in.)		Canister Mass kg (lb)		Canisters/ Waste Package
	HLW	SNF	HLW	SNF	HLW	SNF	
5-DHLW/DOE SNF Short[a]	3,000 (118.1)	3,000 (118.1)	610 (24.0)	457 (18.0)	2,500 (5,512)	2,270 (5,004)	5 HLW / 1 SNF
5-DHLW/DOE SNF Long[a]	4,500 (177.2)	4,570 (179.9)	610 (24.0)	457 (18.0)	4,200 (9,259)	2,721 (6,000)	5 HLW / 1 SNF
Naval SNF Short	N/A	4,750 (187)	N/A	1,689.1 (66.5)	N/A	44,452 (98,000)	1
Naval SNF Long	N/A	5,385 (212.0)	N/A	1,689.1 (66.5)	N/A	44,452 (98,000)	1
2-MCO/2-DHLW Long	4,500 (177.2)	4,200 (165.3)	610 (24.0)	643 (25.3)	4,200 (9,259)	8,910 (19,642)	2 HLW / 2 SNF

NOTES: [a]DOE non-naval spent nuclear fuel.
HLW length and diameter are nominal; HLW mass is maximum. DHLW = defense high-level radioactive waste; MCO = multicanister overpack; SNF = spent nuclear fuel; N/A = not applicable; HLW = high-level radioactive waste.
Sources: DOE 1999c; Naples 1999; DOE 2000d.

primary fissile isotope, geometry, and enrichment (DOE 1999d, Section 5.2). The 250 types of DOE spent nuclear fuel have been divided into groups based on these four factors to perform criticality analyses.

DOE non-naval spent nuclear fuel from each of the four groups will be analyzed in the appropriate configuration, and data important to preventing criticality—such as fissile loading, enrichment, initial configuration of the basket and spent nuclear fuel, and neutron absorber loading in the canister—will be identified. From these results, waste acceptance criteria will be developed for each group. The canister design will also control criticality by limiting the amount of fissile material in each waste package. If required, neutron absorbers would be added into a canister for further criticality control.

A separate analysis will be performed for naval spent nuclear fuel to demonstrate that criticality would be precluded for all credible event sequence conditions during handling at the repository (Mowbray 1999).

3.3.2 High-Level Radioactive Waste and Immobilized Plutonium

About 22,000 canisters of high-level radioactive waste will be generated by 2035 (DOE 1997b, Section 1.5.4). Approximately 1.5 percent will come from reprocessed commercial nuclear fuel; the rest will come from treatment of materials from the defense nuclear program. The estimated number of high-level radioactive waste canisters to be emplaced in the first repository is approximately 8,300, based on the total inventory limit in the NWPA.

Liquid high-level radioactive waste will undergo a process at its current site that yields a solid leach-resistant material, typically a borosilicate glass. While still liquid, the glass is poured into stainless steel canisters. After the glass cools and solidifies, the canisters are sealed. The potential repository would accept solid high-level radioactive waste generated from activities at DOE's Savannah River, South Carolina, and Hanford, Washington, sites, as well as from the Idaho National Environmental and Engineering Laboratory. The waste will arrive in presealed canisters. The potential repository would also receive, subject to the execution of a disposal contract between the DOE and the state of New York, commercial high-level radioactive waste from the West Valley Demonstration Project in New York.

Up to 17 metric tons of surplus plutonium that is excess to national defense needs will be immobilized within ceramic discs that will have neutron absorber material evenly distributed throughout their matrix (65 FR 1608). The ceramic will resist the leaching of plutonium. Section 3.2 describes the additional 33 metric tons of surplus plutonium that will be converted into mixed-oxide fuel.

3.3.2.1 Physical Characteristics

The canisters containing high-level radioactive waste will be standardized to accommodate the waste package design and to reduce manufacturing costs. Table 3-10 gives the canister dimensions.

3.3.2.2 Thermal Output

DOE high-level radioactive waste has a low thermal output. The total heat generation rate will not exceed 1.5 kW per 3-m (9.8-ft) canister or 1.97 kW per 4.5-m (15-ft) canister (DOE 1999c, Section 4.2.3.1). The maximum thermal output of the hottest waste package, the 5-DHLW/DOE short, is 9.16 kW—well below the maximum limit of 11.8 kW (CRWMS M&O 2000aw, Section 2.5.4.2).

3.3.2.3 Criticality Control

With the exception of high-level radioactive waste canisters containing immobilized plutonium, evaluations have indicated that DOE high-level radioactive waste will not contain enough fissile material to pose a criticality risk.

A principal criticality control measure for the immobilized plutonium is the incorporation of neutron absorbing materials (i.e., gadolinium and hafnium) into the waste form. These materials are an effective criticality control measure for both the preclosure and postclosure phases. The planned loading strategy for immobilized plutonium is to transfer it into a codisposal waste package containing five high-level radioactive waste canisters but no DOE spent nuclear fuel canister in the center. Detailed criticality analyses (CRWMS M&O 2000ba) have shown that preclosure and postclosure criticality for a waste package that contains five plutonium-loaded canisters, but that does not have a center DOE spent nuclear fuel canister, is below the subcritical limit for criticality to occur. See Section 3.5.2.4 for a brief discussion of the criticality potential of the immobilized plutonium waste form.

3.3.3 U.S. Department of Energy Waste Package Designs

Three waste package design configurations have been developed for codisposal of DOE non-naval spent nuclear fuel and high-level radioactive waste. In two designs that differ only in length, the typical arrangement places a DOE spent nuclear fuel canister in the center of a ring of five high-level radioactive waste canisters. The exception occurs for high-level radioactive waste canisters containing immobilized plutonium, which will be packaged without a DOE spent nuclear fuel canister in the center. Structural guides will provide support to ensure that the DOE spent nuclear fuel canister is not damaged by an impact to the waste package. These guides also facilitate waste package loading. Such support is not needed for high-level radioactive waste glass, which maintains its own structural integrity (CRWMS M&O 2000aw).

The third waste package design will accept multicanister overpacks, which will have a diameter larger than those of DOE spent nuclear fuel or high-level radioactive waste canisters. To prevent criticality, no more than two multicanister overpacks will be put into a waste package. To improve packaging efficiency without compromising criticality prevention, two long high-level radioactive waste canisters will be codisposed with the two multicanister overpacks. A DOE study determined that codisposing two multicanister overpacks with two long DOE high-level radioactive waste canisters would be the most efficient arrangement (CRWMS M&O 2000aw).

Naval spent nuclear fuel will arrive at the potential repository in canisters suitable for long-term disposal. The canisters will fit one to a waste package. Because the naval fuel will arrive in canisters of two sizes (one short and one long), the DOE has devised two waste package designs for it. The larger of these two types will be the heaviest and longest of all the waste packages. No additional features would be necessary for structural support, heat transfer, and criticality control, since these are provided by the naval spent nuclear fuel or the canister (CRWMS M&O 2000ax).

3.3.4 Preliminary Engineering Specifications

The preliminary engineering specifications for the waste package include the waste form, the physical dimensions of the waste package, and material specifications. Tables 3-10 and 3-11 present preliminary engineering specifications for waste package designs for DOE spent nuclear fuel and high-level radioactive waste, which are based on the physical dimensions, thermal output, and criticality potential of the fuel. Table 3-9 shows the material specifications of the waste package components. These engineering specifications were developed to meet the performance specifications given in Table 3-1.

3.4 SELECTING MATERIALS AND FABRICATING WASTE PACKAGES

The selection of materials from which reliable waste packages could be fabricated followed a multistep analysis and design process. It began by analyzing the critical functions of a particular waste package and its various components. In selecting a material for a component, the designers considered both the material's availability and the critical functions the component would serve as part of the waste package. They identified eight major components and eight performance criteria for selecting materials to fabricate them (CRWMS M&O 1997c, Section 3). The eight major components are:

- Structural shell
- Corrosion-resistant barrier
- Fill gas
- Interlocking plates for commercial designs

Table 3-11. Physical Dimensions of Waste Packages Designed for U.S. Department of Energy Waste Forms

No.	Waste Package Design	Outer Diameter mm (in.)	Outer Length mm (in.)	Weight of an Empty Waste Package kg (lb)	Weight of a Loaded Waste Package kg (lb)
1	5-DHLW/DOE SNF Short	2,110 (83.0)	3,590 (141.3)	23,400 (51,600)	38,100 (84,000)
2	5-DHLW/DOE SNF Long	2,110 (83.0)	5,217 (205.4)	32,600 (71,900)	56,300 (124,000)
3	Naval SNF Short	1,949 (76.7)	5,430 (213.8)	25,800 (56,900)	70,300 (155,000)
4	Naval SNF Long	1,949 (76.7)	6,065 (238.8)	28,000 (61,700)	72,500 (159,000)
5	2-MCO/2-DHLW Long	1,815 (71.5)	5,217 (205.4)	21,800 (48,100)	48,100 (106,000)

NOTES: DHLW = defense high-level radioactive waste; MCO = multicanister overpack; SNF = spent nuclear fuel. Source: CRWMS M&O 2000bc, Section 4.3.

- Fuel tubes for commercial designs
- Structural guides for commercial designs
- Guide tube for codisposal designs
- Thermal shunts for commercial designs.

Not every waste package design requires all of these components; it varies according to the waste form each will hold. However, all eight of these components cover the major requirements of all ten waste package designs.

The eight criteria that contribute to performance are:

- Mechanical performance (strength)
- Chemical performance (resistance to corrosion and microbial attack)
- Predictability of performance (understanding the behavior of materials)
- Compatibility with materials of the waste package and waste form
- Ease of fabrication using the material
- Previous experience (proven performance record)
- Thermal performance (heat distribution characteristics)
- Neutronic performance (criticality and shielding).

Reasonableness of cost was considered as a discriminator.

3.4.1 Material Selection

The first step in selecting the waste package materials was identifying the functional requirements for each component. Next, the characteristics of materials that would help meet the requirements were selected. Candidate materials were chosen from commonly available materials (or, in the case of fill gas, from common gases). The materials were then analyzed in terms of how they would perform their intended functions. Once the candidate materials and alternates were selected, they were tested; the results of these tests are summarized in Section 4.2.4.

Table 3-9 lists the component materials selected after testing. The following sections explain the material selection process in more detail.

3.4.1.1 Waste Package Materials: Contributing to Containment

Corrosion-Resistant Materials—Corrosion performance has been determined to be the most important criterion for a long waste package lifetime. Essential performance qualities therefore include a material's resistance to general and localized corrosion, stress corrosion cracking, and hydrogen-assisted cracking and embrittlement. The effects of long-term thermal aging are also important. To address recommendations provided by the Nuclear Waste Technical Review Board, the DOE has initiated studies to gain a better understanding of the processes involved in predicting the rate of waste package material corrosion over the 10,000-year regulatory period.

Combinations and arrangements of materials as containment barriers were carefully considered from several perspectives. In the process, analysts considered such criteria as (1) material compatibility (e.g., galvanic/crevice corrosion effects); (2) the material's ability to contribute to defense in depth (e.g., because it has a different failure mode from other barriers); (3) the material's ease of fabrication; and (4) the potential impact of thin, corrosion-resistant materials used as containment barriers on a repository's essential operations, such as waste package loading, handling, and emplacement.

The major objectives centered on understanding the temperature and humidity conditions that would exist at different times for a range of thermal operating modes in a particular unsaturated zone, then designing the waste packages accordingly. Since the properties of any material selected for a corrosion barrier would inevitably be influenced by the temperature and humidity conditions in a repository of a particular design at a particular site, selecting the right corrosion-resistant material became one of the most important priorities.

After assessing potential materials available for waste package corrosion barriers, analysts selected nickel- and titanium-based alloys as the most promising candidate materials for corrosion resistance in an oxidizing environment such as Yucca Mountain. Using a corrosion-resistant material as the outer barrier of the waste package will significantly lower the risk of waste package failure from corrosion. Alloy 22 was selected as the preferred material for the outer barrier because it has excellent resistance to corrosion in the environment expected at Yucca Mountain; it is easier to weld than titanium; and it has a better thermal expansion coefficient match to Stainless Steel Type 316NG than titanium. A structurally strong material (stainless steel) was chosen for the inner layer of the waste package (CRWMS M&O 2000av, Section 7.6).

Alloy 22 also offers benefits in the areas of program and operating flexibility. It is extremely corrosion-resistant under conditions of high temperature and low humidity, such as those that would prevail for hundreds to thousands of years in a repository designed to allow a relatively high thermal output from the waste packages. At low temperatures, Alloy 22 is extremely corrosion-resistant in either low or high humidity. Thus, the selection of Alloy 22 supports the flexibility to operate the potential repository over a range of thermal modes (CRWMS M&O 2000av). Uncertainty about the waste package corrosion rate may be reduced by avoiding the conservatively defined window of corrosion susceptibility for Alloy 22, which can be accomplished by keeping waste package temperatures below 85°C (185°F) or maintaining the in-drift relative humidity below 50 percent (Dunn et al. 1999, p. xvi; CRWMS M&O 2000n, Section 3.1.3.1). Table 3-12 presents the chemical composition of Alloy 22.

Table 3-12. Chemical Composition of Alloy 22

Element	Composition (wt%)
Carbon (C)	0.015 (max)
Manganese (Mn)	0.50 (max)
Silicon (Si)	0.08 (max)
Chromium (Cr)	20.0 to 22.5
Molybdenum (Mo)	12.5 to 14.5
Cobalt (Co)	2.50 (max)
Tungsten (W)	2.5 to 3.5
Vanadium (V)	0.35 (max)
Iron (Fe)	2.0 to 6.0
Phosphorus (P)	0.02 (max)
Sulfur (S)	0.02 (max)
Nickel (Ni)	Balance

Source: ASTM B 575-97, *Standard Specification for Low-Carbon Nickel-Molybdenum-Chromium, Low-Carbon Nickel-Chromium-Molybdenum, Low-Carbon Nickel-Chromium-Molybdenum-Copper and Low-Carbon Nickel-Chromium-Molybdenum-Tungsten Alloy Plate, Sheet, and Strip.*

Structural Materials—The major functional requirement of the structural material for the inner layer of the waste package is to support the corrosion-resistant outer material. The DOE chose Stainless Steel Type 316NG for the structural layer (CRWMS M&O 2000av, Section 5.2). This material provides the required strength; has a better compatibility with Alloy 22 than carbon steel; and provides an economical solution to functional requirements. Table 3-13 presents the chemical composition of Stainless Steel Type 316NG.

Table 3-13. Chemical Composition of Stainless Steel Type 316NG

Element	Composition (wt%)
Carbon (C)	0.020 (max)
Phosphorus (P)	0.030 (max)
Silicon (Si)	0.75 (max)
Copper (Cu)	0.50 (max)
Titanium (Ti)	0.05 (max)
Tantalum (Ta) and Niobium (Nb)	0.05 (max)
Manganese (Mn)	2.00 (max)
Sulfur (S)	0.005 (max)
Nitrogen (N)	0.060 to 0.10
Cobalt (Co)	0.10 (max)
Boron (B)	0.002 (max)
Bismuth (Bi) + Tin (Sn) + Arsenic (As) + Lead (Pb) + Antimony (Sb) + Selenium (Se)	0.02 (max)
Chromium (Cr)	16.00 to 18.00
Molybdenum (Mo)	2.00 to 3.00
Nickel (Ni)	11.00 to 14.00
Vanadium (V)	0.1 (max)
Aluminum (Al)	0.04 (max)
Iron (Fe)	Balance

Sources: For all elements except carbon and nitrogen, values presented are within the ranges and maximum limits provided by ASTM A 276-91a, *Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes*. Values for carbon and nitrogen are given by Danko (1987, p. 931).

3.4.1.2 Waste Package Materials: Internal Components

The designs for commercial spent nuclear fuel and DOE codisposal waste packages include internal components (i.e., structural guides, interlocking plates, fuel tubes, and thermal shunts) that must be able to sustain the mechanical loads created by handling, emplacement, and, if necessary, retrieval. Thus, mechanical performance was a major selection criterion. Thermal performance was also an important selection criterion because these components provide an additional path for conducting heat from the waste form to the walls of the waste package. The fuel tubes contact both the waste form and the basket plates. If the material selected for the tubes causes the waste form to degrade, release rates could be increased; if it causes the plates to degrade, criticality control could be compromised. Therefore, compatibility with other materials was an important criterion. The waste package design does not rely on these components for postclosure performance, so corrosion-resistant materials are not needed. Two grades of carbon steel (SA 516 Grades 55 and 70) were found to be the best choices for these internal components, based on the criteria; the designers chose to use Grade 70 (CRWMS M&O 2000bd, Section 4).

Neutron Absorber Interlocking Plates—The most important function of the neutron absorber is to reduce the potential for criticality. The neutron absorber material is typically an additive to a carrier material (e.g., stainless steel alloyed with a boron compound). The neutron absorber is used in the interlocking plates in the internal basket. Corrosion behavior is important in keeping the neutron absorber material in place and effective long after emplacement, so chemical performance in a variety of environments was an important selection criterion. Mechanical performance was an evaluation factor because the interlocking plates must be able to sustain the mechanical loads created by handling, emplacement, and, if necessary, retrieval. Compatibility with other materials was considered, since the plates must not cause the waste form to degrade. The plates also provide an important path for conducting heat from the waste form to the walls of the waste package, so thermal performance was considered. The material of choice was Neutronit A 978. Its selection was based on its corrosion performance compared to the other candidate materials, as well as its available boron concentration. The composition of Neutronit is similar to SA 240 Stainless Steel Type 316 but with 1.6 percent boron added (CRWMS M&O 2000be, Section 3.1.3).

Thermal Shunts—The thermal shunts provide another important path for conducting heat from the waste form (in this case, commercial spent nuclear fuel) to the walls of the waste package. The thermal conductivity of the material is very important. The thermal shunts would be in contact with the waste form, so compatibility with spent nuclear fuel was an important evaluation criterion. The thermal shunts are only needed during the early period of repository performance, when the decay heat from spent nuclear fuel would be relatively high. The material selected does not need a high degree of corrosion resistance. The thermal shunts must have enough structural strength to withstand handling, emplacement, and possible retrieval operations. However, these service loads are not very large, so mechanical performance was not selected as an evaluation criterion. Aluminum alloys 6061 and 6063 were selected over copper because of concerns that, should a waste package be breached and water enter, copper may react with the chloride ions in the water. This could result in accelerated degradation of the Zircaloy cladding on the spent nuclear fuel, which would eventually release radionuclides from the waste (CRWMS M&O 2000be, Section 3.2.3).

3.4.1.3 Fill Gas

The fill gas can be a significant conductor of heat from the waste form to the internal basket, so thermal performance was deemed one of the most important criteria in choosing a gas. The fill gas should not degrade other components of the waste package, so compatibility with other materials was another important criterion. Helium is routinely used as the fill gas for fuel rods, which indicates that helium would have an excellent compatibility with spent nuclear fuel. Based on a review of data on thermal conductivity and the fact that helium is chemically inert, it was chosen over other candidate gases, such as nitrogen, argon, and krypton (CRWMS M&O 2000be, Sections 3.3.1 through 3.3.3).

3.4.2 Waste Package Fabrication Process

This section describes the fabrication process for the waste package, which is shown schematically in Figures 3-7 and 3-8. The fabrication process was based on both the design criteria and the physical characteristics of the selected materials.

The waste package will be fabricated, welded, and inspected in accordance with those portions of the *ASME Boiler and Pressure Vessel Code,* Section III, Division I, Subsection NB (Class 1 Components) (ASME 1995) that will ensure the waste package will perform in accordance with the design basis. Because the largest number of waste packages will be manufactured for commercial spent nuclear fuel, this type will serve to illustrate the basic fabrication process for all waste package types. The other types would be fabricated in a similar way, though some dimensions would vary.

The commercial fuel waste package uses Neutronit A 978 and carbon steel interlocking plates with carbon steel tubes (CRWMS M&O 2000bf, Section 8.1). All waste package fabrication would take place offsite, including the welding of the bottom lids. The top lids, although fabricated offsite, would be welded on in the Waste Handling Building after the waste package had been loaded.

3.4.2.1 Outer Cylinder Fabrication

Forming the outer cylinder of rolled and welded Alloy 22 plate requires two half-length cylinders (see Table 3-7 for completed waste package lengths) because of the limitations of most rolling fabricators. Initially, the plate would be approximately 5,080 mm (200 in.) long by 2,540 mm (100 in.) wide. The thickness would permit machining (for rounding) after welding. After being received by the fabricator, the plate would be inspected, laid out to establish the developed length, and thermally cut to size. The plate would then be rolled (CRWMS M&O 2000bf, Section 8.1.1).

The cylinder would then be adjusted to meet the required diameter and the inner circumference, taking into consideration the subsequent weld shrinkage of the longitudinal seam. The long seam weld preparations would be machined and prepared for welding. The cylinder would be braced to minimize the weld distortion and welded. The braces would be removed, and the weld seam would be prepared for nondestructive examination. One end of the cylinder would be prepared for circumferential seam welding. In parallel, a second cylinder would be prepared the same way (CRWMS M&O 2000bf, Section 8.1.1).

The two cylinders would then be joined and circumferentially welded, with subsequent nonde-

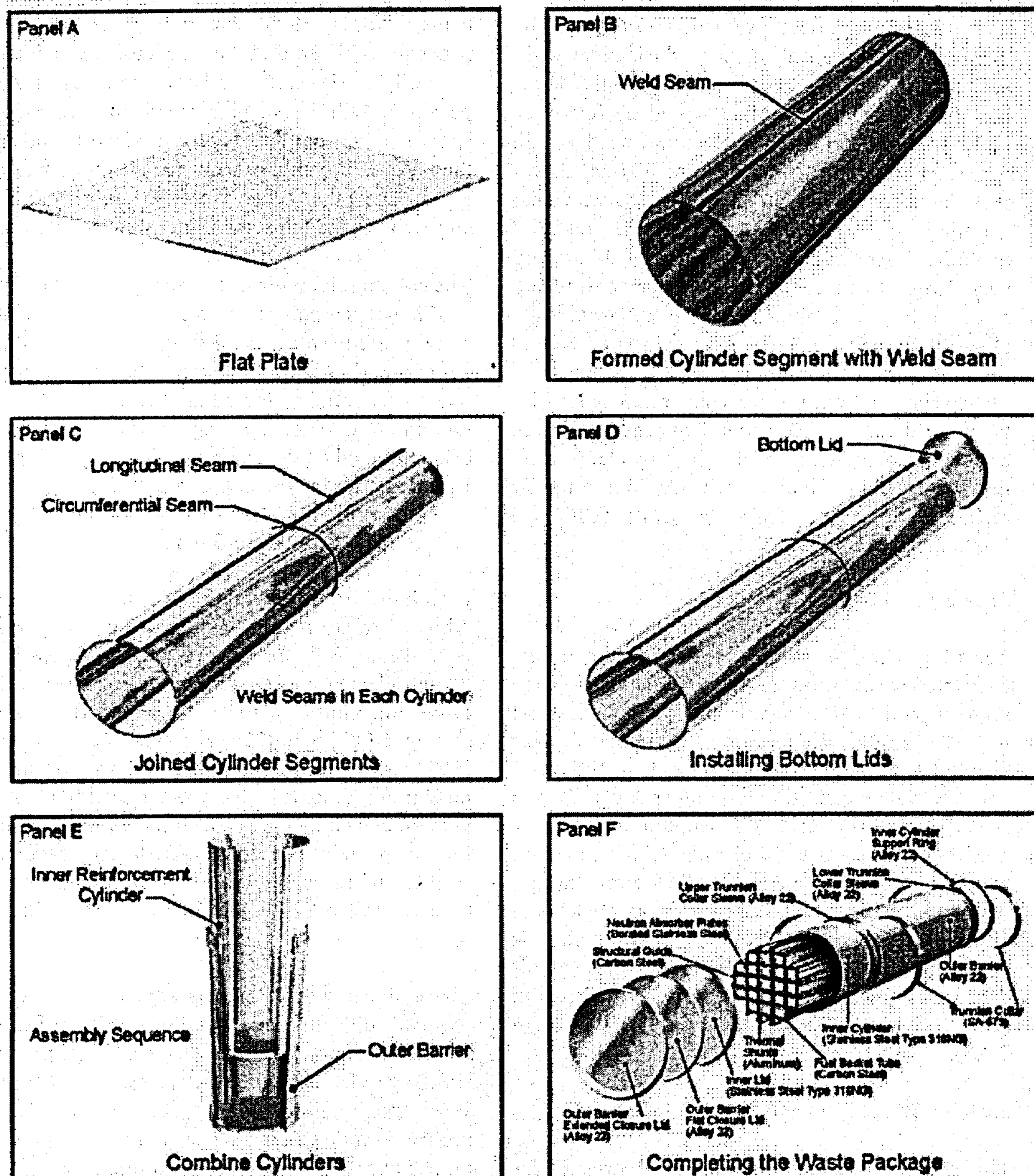

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Figure 3-7. Waste Package Fabrication Process
The waste package fabrication process starts with a flat plate of metal (Alloy 22 for the outer cylinder, Stainless Steel Type 316NG for the inner cylinder) (Panel A). The plate is rolled and welded to form a cylinder segment (Panel B). Two cylinder segments are combined and welded to form a waste package inner or outer cylinder (Panel C). Lids are attached to the inner and outer cylinders (Panel D). The inner cylinder is inserted into the outer cylinder to form the waste package (Panel E). The internal components, waste form, and lids are added to complete the waste package (Panel F).

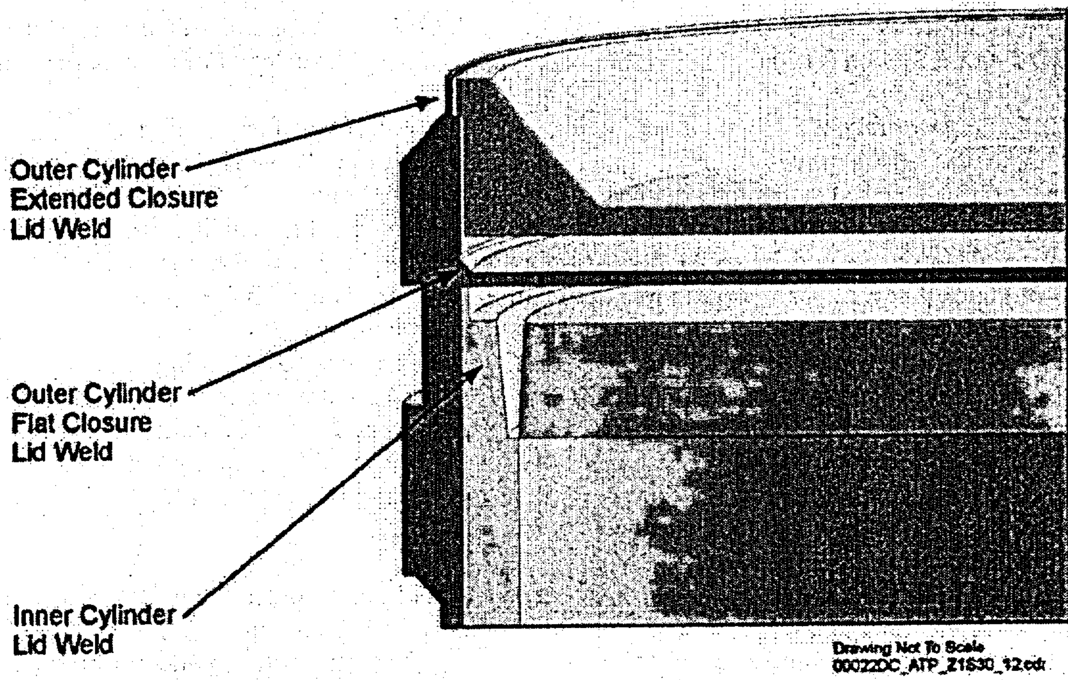

Figure 3-8. Waste Package Final Closure Welds
The process for the final closure welds on the top lids must be performed remotely in a high-radiation environment to produce high-integrity welds that can be inspected in the "as-welded" condition. Weld joint configurations, process techniques, and procedures have been developed for, and demonstrated on, inner and outer cylinder final closure welds for full-diameter waste package mockups. The narrow-groove, hot-wire, semiautomatic gas tungsten arc weld process can be done remotely, and has proven to produce single-bead layer welds with acceptable weld and corrosion properties.

structive examination testing performed on the circumferential seam. The outer cylinder would be inspected to verify that the inside diameter is within tolerance. The inside of the cylinder would then be machined (CRWMS M&O 2000bf, Section 8.1.1).

3.4.2.2 Inner Cylinder Fabrication

To form the stainless steel inner cylinder, workers would start with two plates of Stainless Steel Type 316NG large enough to make half the inner cylinder length (see Table 3-7 for completed waste package lengths). The full cylinder length would require two plates, each approximately 4,980 mm (196 in.) wide by 2,540 mm (100 in.) long. Each plate would be cut or machined for size and longitudinal weld preparations. The plates would be roll-formed to make two half-cylinders and welded. Inspectors would use nondestructive techniques to examine both the cylinders and the welds. The weld preparations for the circumferential seam would be machined, then the cylinder would be assembled and circumferentially welded. Nondestructive examination inspections would again be performed on the circumferential weld, and the cylinder would then be machined (CRWMS M&O 2000bf, Section 8.1.2).

3.4.2.3 Lid Fabrication

The outer and flat closure lids would be fabricated from Alloy 22 plates approximately 1,900 mm (75 in.) wide and 3,800 mm (150 in.) long. The plates would then be cut to the correct diameter and the edges machine cleaned to prepare for the weld (CRWMS M&O 2000bf, Section 8.1.3).

The inner lids would be fabricated from a Stainless Steel Type 316NG plate approximately 1,800 mm (71 in.) wide and 3,600 mm (142 in.) long. The plate would be laid out for the cutting of two circles. The plates would then be thermally cut to the correct diameter, and the edges would be machine cleaned to prepare for the weld (CRWMS M&O 2000bf, Section 8.1.3).

3.4.2.4 Assembly of Support Ring

A support ring attached to the inner diameter of the outer cylinder is required to hold the inner cylinder in place. This ring would be made from 20-mm (0.8-in.) thick Alloy 22 plate. A piece would be cut 100 mm (4 in.) wide and rolled into a ring, then weld preparations would be machined. The ring would be fit to the inside of the outer cylinder near the bottom end, welded, and inspected (CRWMS M&O 2000bf, Section 8.1.4).

3.4.2.5 Assembly of Lid to Cylinder

Once the inner and outer cylinders had been completed, the inner and outer bottom lids would be welded in place. The structures would be set in a vertical position and the lids assembled to each cylinder. The welding could then be done in the flat position. After the welding had been completed, radiographic examination, ultrasonic examination, and liquid penetrant examination would be performed on the inner and outer lid seams. Radiographic and ultrasonic inspection would ensure that all detectable flaws, regardless of their orientation, were identified. Liquid penetrant inspection would ensure that surface indications were identified (CRWMS M&O 2000bf, Section 8.1.5).

3.4.2.6 Annealing of Outer Cylinder

Annealing is a process in which a material is subjected to a controlled heating and cooling cycle to affect material properties, for example, to relieve residual stress. Residual stresses are a common by-product of fabrication processes, such as forming, machining, and welding. Stress mitigation techniques, such as annealing, will be applied to the outer cylinder to minimize the potential for stress corrosion cracking. Since the closure lids will be welded shut after the waste has been loaded, different mitigation techniques may be employed. The parameters for the annealing operation are still being developed, but the DOE expects that the cylinder assembly will be heated in a furnace and then quenched by water (CRWMS M&O 2000bf, Section 8.1.7).

3.4.2.7 Assembly of Commercial Spent Nuclear Fuel Waste Package

The objective of machining is to produce a gap ranging from 0 to 4 mm (0 to 0.16 in.) between the inner and outer cylinders. Over time, the stainless steel inner cylinder will expand in response to the heat emitted by the radioactive decay of its contents. Even with the cylinders touching, there is enough allowance for the inner cylinder to heat up and expand without putting excessive stress on the Alloy 22 outer cylinder. Static loads in the outer barrier shell will not produce tensile stresses above 10 percent of the yield strength of the outer barrier material (CRWMS M&O 2000au, Section 1.2.1.23).

After both the outer and inner reinforcement cylinders had been machined, they would be fitted together. The outer cylinder would be heated to about 370°C (700°F) to allow the inner cylinder to be lowered inside. The heat would then be removed and the cylinders allowed to cool (CRWMS M&O 2000bf, Section 8.1.8).

3.4.2.8 Basket and Internal Components

Once the inner and outer cylinders and the bottom lids had been assembled, the container would be ready for the addition of the internal components. The cylinder would be laid out to establish the location of the internal corner guide assemblies and the internal side guides. The corner guide assemblies and the side guides would be put in place and welded, using manual gas tungsten arc welding.

The bottom set of plates would be installed in an interlocking fashion, followed by three additional sets. The tubes would be inserted and the tube tops stitch-welded together, if required.

The waste package would then be cleaned, wrapped, and protected for shipment to the repository surface facility and storage until it was ready to be loaded and sealed (CRWMS M&O 2000bf, Section 8.1.10). The final fabrication step, performed at the repository surface facility, would be the annealing of the closure weld area to mitigate the stresses that may have been induced during the welding of the outer closure lid. For more details on the process of loading the waste package, refer to Section 2.2.4.

3.5 WASTE PACKAGE DESIGN EVALUATIONS

The waste package must satisfy defined performance specifications to protect the public and workers and to meet the performance objectives of a repository. An example of a performance specification is the ability of a waste package to withstand a tipover event without breaching. Performance specifications are discussed in the following sections, where they are categorized by relevant engineering discipline (i.e., thermal, criticality, structural, and shielding). Detailed discussions of performance specifications are available in System Description Documents (e.g., CRWMS M&O 2000au).

Some of the performance specifications and supporting evaluations depend on temperature. In these cases, the evaluation is based on the higher-temperature operating mode. Further evaluations of lower-temperature operating modes are part of ongoing engineering studies.

To show that waste packages can be successfully developed for the various waste forms expected to be received at the repository, a sensitivity analysis was performed to determine which waste package designs best represent the widest array of design configurations and waste forms (CRWMS M&O 2000az). This selection was based on the number of waste package types expected to be needed for the repository—for instance, the two commercial spent nuclear fuel waste package designs chosen for analysis represent over 95 percent of the total required—and the relationship of a waste package type to a limiting performance specification (e.g., a heavier waste package would be more susceptible to a drop event). Detailed design work was performed for four waste package types:

- 21-PWR Absorber Plate
- 44-BWR
- 5-DHLW/DOE SNF Short
- Naval SNF Long.

3.5.1 Thermal Evaluations Performed on the Waste Package Design

Thermal analyses have been performed to demonstrate that waste form temperatures will not exceed levels established to maintain waste form integrity. The thermal specification for commercial spent nuclear fuel ensures that the cladding temperature will not compromise the integrity of the cladding, a barrier to radionuclide release. A specification for DOE high-level radioactive waste ensures that the glass does not reach a transition temperature that would cause significant changes in its phase structure or composition. Such an alteration would increase the solubility of the glass and reduce the time required for movement of the radionuclides embedded inside.

With respect to these thermal functions, two performance specifications were selected for evaluation: (1) Zircaloy commercial spent nuclear fuel cladding must be maintained below 350°C (660°F) under normal conditions (CRWMS M&O 2000au, Section 1.2.1.6), and (2) the temperature of DOE high-level radioactive waste must be maintained below 400°C (750°F) under normal conditions (CRWMS M&O 2000aw, Section 1.2.1.6).

3.5.1.1 Spent Nuclear Fuel Cladding Temperature

To calculate the cladding temperature for commercial spent nuclear fuel, time-dependent heat generation rates of the waste packages were adjusted to ensure that the average heat generation rate of an emplacement drift segment was the same as that for the repository as a whole, as discussed in Section 2.3.1. The calculation used a representative section of a drift containing one 21-PWR Absorber Plate, one 44-BWR, and one 5-DHLW/DOE SNF waste package, arranged as shown in Figure 3-3. The 21-PWR Absorber Plate serves as the design basis waste package, with a maximum heat generation rate of 11.8 kW. The second waste package, the 44-BWR, is based on an average heat generation rate of 7.0 kW. The 5-DHLW/DOE SNF waste package serves as a balancing package in which the heat generation rate is varied to ensure the average in the drift segment is the same as that in the repository as a whole. The time-dependent waste package surface temperatures calculated were used to perform a two-dimensional analysis of the internal components of the waste package. This calculation gives the peak cladding temperature for the design basis waste package (CRWMS M&O 2000au). The peak cladding temperature for commercial spent nuclear fuel was calculated to be 282°C (542°F), with the peak occurring 35 years after emplacement. The waste package spacing for the calculation was modeled as 0.1 m (0.3 ft), with a 25-year ventilation period (CRWMS M&O 2000au, Section 2.5.1.6). Active ventilation for 25 years would provide heat removal, limiting the heat-up of the waste package during preclosure.

Peak cladding temperature is not an issue for DOE spent nuclear fuel because no credit is taken for the fuel cladding in performance assessment. A canister of naval spent nuclear fuel will have lower heat generation than a waste package containing commercial spent nuclear fuel. Thermal analysis indicates that thermal limits associated with naval spent nuclear fuel will not be exceeded in the repository (CRWMS M&O 2000ax, Section 2.5.4.2).

3.5.1.2 High-Level Radioactive Waste Canister Temperatures

The vitrified high-level radioactive waste form could undergo devitrification at temperatures above 400°C (750°F). A maximum peak value of 214.5°C (418°F) occurs in the glass under normal conditions, which is well below the 400°C (750°F) threshold (CRWMS M&O 2000aw).

3.5.2 Criticality Evaluations Performed on Waste Package Designs

3.5.2.1 Preclosure Evaluations—Commercial Spent Nuclear Fuel

This section describes the performance specifications for criticality of commercial spent nuclear fuel. Further details on preclosure criticality analysis are available in the *Preclosure Criticality Analysis Process Report* (CRWMS M&O 1999i).

Each specific fuel assembly or material received by the repository has an associated fuel reactivity, or capability of contributing to a self-sustained nuclear fissioning process (criticality). The major properties of fuel reactivity in available commercial spent nuclear fuel are the initial enrichment and the burnup of the fuel when it was discharged from the reactor. Enrichment is the weight percentage of a fissile isotope compared to the total amount of uranium in the fuel assembly. Burnup is a measure of the amount of energy produced by the assembly while it was in operation. The higher the initial enrichment of the fuel, the more it can contribute to a critical system. The higher the burnup, for domestic commercial nuclear fuel, the less it can contribute to a critical system. A detailed description of the physics of criticality is provided in Section 4.3.3.2.1.

During the preclosure period, criticality is prevented in waste packages by ensuring, before loading, that the reactivity for each fuel assembly is below the level required for criticality. This is accomplished through the use of loading curves. Loading curves show the minimum allowed burnup as a function of initial enrichment and provide a simple go/no-go check, using the available information, on whether a fuel assembly can be loaded unaltered into a standard waste package without any concern for criticality. The presence of a moderator will be controlled as necessary to ensure subcriticality.

There are two main aspects to loading curve development: the determination of the value of reactivity considered limiting and the determination of the burnup/enrichment pairs that correspond to this value.

The loading curve methodology summarized below is described in detail in the *Preclosure Criticality Analysis Process Report* (CRWMS M&O 1999i, Section A.1). The effective neutron multiplication factor (k_{eff}) of a system is the measure of criticality; when k_{eff} is greater than or equal to 1.0, criticality occurs. If the effective neutron multiplication is less than 1.0, the system is considered subcritical. When designing a system (e.g., a waste package) to be subcritical, there must be a means for ensuring that a k_{eff} value conservatively represents the true k_{eff} of the system. This is accomplished by choosing a value below 1.0 at which nuclear criticality is assumed to occur. This value is known as the critical limit. The critical limit is determined by accounting for and bounding the bias in the calculational method used, and the uncertainty in the experiments used to validate the method of calculation. For the calculations performed thus far, the critical limit for k_{eff} is 0.98. Because ensuring subcriticality is important to the health and safety of the repository workers, the calculations add an administrative safety margin of 5 percent for preclosure to provide additional conservatism. The loading curve is therefore based on a k_{eff} value of 0.93, referred to as the administrative limit. Any criticality calculation that uses the same tools on which the critical limit was based and shows a k_{eff} less than or equal to 0.93 provides high confidence that the system will not become critical (CRWMS M&O 2000bg, Section 6.2.1).

Once the criticality limit is established, many calculations are performed to determine, for a specific value of initial enrichment, what burnup results in a k_{eff} that equals the administrative limit (Figure 3-9). These calculations are performed for a point where the waste package is fully loaded with the same fuel assembly. Criticality calculations are performed in increments over the range of initial enrichments and burnups for each commercial fuel type to be received at the repository. The points where the different burnup/enrichment pairs equal the administrative limit can then be represented as a single curve of initial enrichment versus burnup, referred to as a loading curve.

Figure 3-9 shows an example of a loading curve. Loading curves will be developed for each waste package type containing commercial spent nuclear fuel. Any fuel that has a burnup/enrichment pair equal to or above the loading curve line can be loaded unaltered into that waste package type, and criticality will not occur in that waste package during preclosure. Some assemblies that fall below the curve will have to be loaded into the waste package with additional neutron absorber material so they will be above the line. Another option is to load the fuel assembly into a type of waste package that contains fewer fuel assemblies (e.g., the

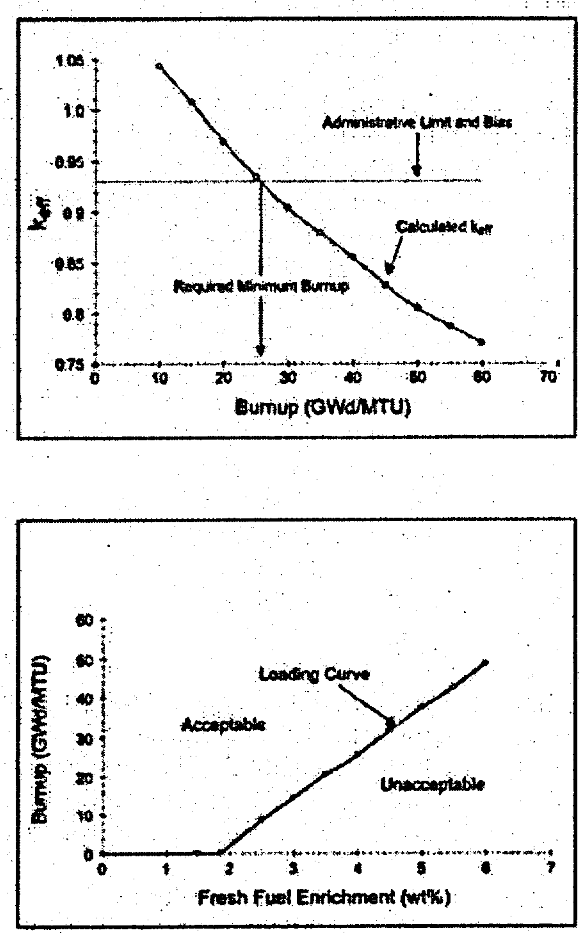

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Figure 3-9. Administrative Limit for Calculated k_{eff} and Typical Loading Curve
Criticality calculations are performed in increments over the range of initial enrichments and burnups for each commercial spent nuclear fuel type to be received at the repository. The points where the different burnup/enrichment pairs equal the administrative limit can then be represented as a single curve of initial enrichment versus burnup. This representation is a typical criticality loading curve. Fuel that meets the initial enrichment (in this case, the 4 percent initial enrichment is shown) and has the minimum required burnup (shown in the area noted as "acceptable") may be placed in the waste package for which the evaluation is being performed. GWd = gigawatt day.

12-PWR and 24-BWR), making the waste packages subcritical.

In the loading curve evaluation, margin to criticality is shown assuming the highly unlikely (beyond design basis), but essential to criticality, event of a waste package that is completely filled with water. A loading curve evaluation has been performed for the 21-PWR Absorber Plate model, as documented in the *21 PWR Waste Package Loading Curve Evaluation* (CRWMS M&O 2000bg, Section 6). The 21-PWR waste package design was chosen because more information is available about it than about the other designs. The 21-PWR waste package is also considered to be the design with the highest inherent reactivity (CRWMS M&O 2000az, Table 8). The 21-PWR loading curve evaluation is representative of calculations that will be performed for the other waste package types.

The waste package will be designed so that during preclosure, nuclear criticality cannot be possible unless at least two unlikely, independent, and concurrent or sequential changes occur in the conditions essential to nuclear criticality safety.

3.5.2.2 Postclosure Criticality Evaluation: Commercial Spent Nuclear Fuel

The methodology that has been developed to evaluate potential for criticality and to ensure that significant impacts are prevented during the postclosure period is described in detail in *Disposal Criticality Analysis Methodology Topical Report* (YMP 2000c, Section 3). Section 4.3.3.2 summarizes this methodology and the analyses conducted using it.

Evaluations show that there is a very small probability that a potential critical configuration might result under the repository and fuel conditions described in this report, but that if a postclosure criticality were to occur, the effects would not compromise the ability of the repository to protect public health or meet design objectives or regulatory limits. The evaluations were performed assuming the most reactive combinations of such factors as the amount of fissile uranium and plutonium, the physical arrangement of the fuel, and the presence and amount of moderator in the waste package. The results indicate that all applicable limits can be met and public health protected using the current waste package designs.

3.5.2.3 Evaluations of Criticality Potential of U.S. Department of Energy Spent Nuclear Fuel

Analyses to demonstrate the viability of disposal have been performed or are in process for seven groups of DOE non-naval spent nuclear fuel. A separate analysis is being performed for naval spent nuclear fuel to demonstrate that criticality will be prevented for all credible event sequence conditions in the repository (Mowbray 1999).

In general, the amount of DOE spent nuclear fuel allowed per canister is a function of the physical size and weight limitations of the canister. The limitation on the amount of fissile material per canister provides criticality control. However, insoluble neutron absorbers (gadolinium compounds and alloys) are required for criticality control within the canister for some DOE spent nuclear fuel groups. To date, several items have been identified as important to criticality (DOE 1999d, Section 5.2). The performance and distribution of the neutron absorber material is important in preventing criticality.

The canister shell is also important in preventing criticality because it initially confines the fissile elements and neutron absorber material so they cannot be separated. The canister baskets developed for the representative fuel types are particularly important in cases where they provide the distribution mechanism for neutron absorber material.

The configurations evaluated for each fuel type include varying degrees of degradation, resulting in many different geometric configurations and fissile distributions. These degraded configurations also bound the other types of fuels in a group as long as the limits on fissile mass, linear fissile loading, and enrichment are not exceeded (DOE 1999d, Section 5.2). Further details on DOE spent nuclear fuel criticality analysis are available in completed viability evaluations (CRWMS M&O

2000bh; CRWMS M&O 2000bi; CRWMS M&O 1999j; CRWMS M&O 2000bj; CRWMS M&O 2000bk).

3.5.2.4 Evaluation of Criticality Potential of the Immobilized Plutonium Waste Package

The criticality potential of the immobilized plutonium ceramic discs is effectively controlled by neutron absorbers (i.e., gadolinium and hafnium), which are fabricated into the waste form itself. These materials are an effective criticality control measure. During postclosure, about 50 percent of the hafnium and about 1 percent of the gadolinium is needed to maintain subcriticality under all dilution situations. The DOE has exhaustively examined the physical and chemical processes that might be able to cause a separation, or removal of the neutron absorbers from the waste package entirely, and concluded they are insufficient to warrant further consideration. Detailed criticality analyses (CRWMS M&O 2000ba, p. xii) have shown that five plutonium-loaded canisters, without a center DOE spent nuclear fuel canister, can be placed in the same waste package.

3.5.3 Structural Evaluations Performed on Waste Package Designs

A performance specification for the waste package requires that it not breach during normal operations and event sequences. To address the term "breach" in a quantified manner, threshold limits for failure from the American Society of Mechanical Engineers code will be used. The waste package is designed to meet American Society of Mechanical Engineers code requirements. For event sequences, breach is assumed to have occurred when 90 percent of the ultimate tensile strength has been exceeded. To demonstrate the design adequacy of the waste package with respect to these structural functions, several performance specifications were selected, as documented in the *Waste Package Design Sensitivity Report* (CRWMS M&O 2000az, Section 7). These include:

- Internal pressurization
- Retrieval
- Rockfall
- Vertical drop
- Tipover
- Missile impact (i.e., from an accidental airborne projectile).

3.5.3.1 Internal Pressurization

Pressurization of a commercial spent nuclear fuel waste package could be caused by the rupture of all the fuel rods. Evaluations have been performed over uniform waste package temperatures, ranging from 20° to 600°C (68° to 1,100°F). The peak stresses at the junction of the waste package shell and lid were then compared to the ultimate tensile stress of the waste package materials.

The 21-PWR Absorber Plate waste package was evaluated because it would have the maximum internal pressure due to fuel rod failure. Detailed calculations show that the resulting stresses for all components of the waste package are less than 90 percent of the ultimate tensile strength of the materials; therefore, the waste package will not breach as a result of pressurization, and the criterion is met (CRWMS M&O 2000au, Section 2.5.2.10).

3.5.3.2 Retrieval

The waste package is being designed to allow retrieval up to 300 years after emplacement. The Naval SNF Long waste package is the heaviest waste package for retrieval. The ability of the waste package and pallet to be lifted together as a single unit was calculated. The results show that the maximum stress intensities from lifting among the emplacement pallet Alloy 22 and Stainless Steel Type 316L components are 96 MPa (14,000 psi) and 39 MPa (5,700 psi), respectively. These stress intensity magnitudes are less than one-third of the yield strength and one-fifth of the tensile strength for each of the corresponding materials.

Atmospheric corrosion penetration rates for Alloy 22 and Stainless Steel Type 316 are 0.0093 μm/yr and 0.025 μm/yr, respectively. The calculated cumulative decrease of thickness of the structural members of the emplacement pallet over 300 years of preclosure emplacement is 2.8 μm for Alloy 22 and 7.5 μm for Stainless Steel Type 316.

This negligible level of corrosion renders the waste package retrieval calculation unnecessary, since the consequential change of the results presented in this calculation would be insignificant (CRWMS M&O 2000ax, Section 2).

3.5.3.3 Rockfall

The 21-PWR Absorber Plate waste package design was selected for this evaluation because it would be the most vulnerable to rockfall. Its thinner walls in the outer barrier and its greater sensitivity to internal basket deformation cause greater vulnerability. It is also the most common waste package, and hence the most likely to actually suffer a rockfall impact. This calculation modeled a representative rockfall from the roof of the drift onto the unprotected waste package during preclosure. A height of 3.1 m (10 ft) and a rock size of 13 metric tons (14 tons), which were determined in the event sequence hazards analysis, were modeled (CRWMS M&O 2000au, Section 2.5.2.1). The calculated results, presented in Table 3-14, indicate the survivability of a 21-PWR Absorber Plate waste package in a rockfall event (CRWMS M&O 2000au, Section 2.5.2.1). For event sequences such as rockfall, breach has occurred analytically when 90 percent of the ultimate tensile strength has been exceeded.

Table 3-14. Summary of Results for Rockfall Calculation

Shell Composition	Calculated Maximum Stress Intensity MPa (psi)	Ultimate Tensile Stress MPa (psi)	Percent of Ultimate Stress
Alloy 22	563 (81,600)	690 (100,000)	82
Stainless Steel	291 (42,200)	517 (75,000)	60

3.5.3.4 Vertical Drop

The vertical drop evaluation was performed using the Naval SNF Long waste package because it is the heaviest design and has the highest internal load; therefore, it will have the highest stresses in the lids during a vertical drop. This calculation modeled a waste package being dropped from a distance of 2 m (6.6 ft).

Detailed calculations demonstrate the survivability of a Naval SNF Long waste package in a vertical drop. The results show that the maximum stresses among the waste package (made of Alloy 22, except for the lower trunnion collar sleeve) and Stainless Steel Type 316NG components are 433 MPa (63,000 psi) and 275 MPa (40,000 psi), respectively. Since these stress intensities are less than 90 percent of the ultimate tensile strength for each of the corresponding materials, the performance specifications are met. The maximum stress in the lower trunnion collar sleeve contacting the unyielding surface is 799 MPa (116,000 psi), which exceeds the tensile strength of Alloy 22. However, this does not constitute failure because the collar sleeve is designed to protect the waste package by acting as a crush zone at the point of impact (CRWMS M&O 2000ax, Section 2.5.2.3).

3.5.3.5 Tipover

A waste package might tip over because of a vertical drop or a seismic event. The 21-PWR Absorber Plate waste package was selected for the preclosure tipover evaluation because of its thinner walls in the outer barrier (made of Alloy 22) and its greater sensitivity to internal basket deformation. It is also the most common waste package. The tipover analysis was simulated in a detailed calculation. Table 3-15 shows the results. The stresses for all components that make up the waste package are less than 90 percent of the ultimate tensile strength of those materials (CRWMS M&O 2000au, Section 2.5.2.6).

Table 3-15. Summary of Results of Tipover Calculation for 21-PWR Absorber Plate Waste Package

Waste Package Component	Calculated Maximum Stress MPa (psi)	Ultimate Tensile Stress MPa (psi)
Outer Shell and Lids	553 (80,000)	690 (100,000)
Inner Shell and Lids	327 (47,000)	517 (75,000)

3.5.3.6 Missile Impact

A potential internal missile event sequence could take the form of a valve stem being ejected from equipment operating at high pressures. The valve was estimated to have a mass of 0.5 kg (1.1 lb), a diameter of 1.0 cm (0.39 in.), and a velocity of 5.7 m/s (19 ft/s). The missile impact evaluation was performed for the 21-PWR, 44-BWR, 5-DHLW/DOE SNF, and Naval SNF waste package designs.

The calculated minimum velocity is significantly less that would be required to compromise the integrity of the waste package (CRWMS M&O 2000au, Section 2.5.2.8).

3.5.4 Shielding Evaluations Performed on the Waste Package Design

Shielding analyses evaluate the effects of ionizing radiation on personnel, equipment, and materials. The primary sources for waste package radiation are gamma rays and neutrons emitted from spent nuclear fuel and high-level radioactive waste. Loading, handling, and transporting of waste packages would be carried out remotely to keep personnel exposure as low as is reasonably achievable (e.g., having the human operators behind radiation shield walls, using remote manipulators, viewing operations with video cameras). Shielding analyses were performed for waste package designs to assess the effects of radiation on material and equipment. These analyses provide information used by both the subsurface and surface design programs to determine shielding requirements in the surface facility, on the waste package transporter, and in the subsurface facility.

Because they were designed to contain the waste forms for thousands of years, the waste packages must reduce radiation levels at their surfaces so that radiolytically enhanced corrosion under aqueous conditions is negligible. The shielding analyses determined radiation exposure rates on the surface of the waste package and evaluated whether radiolytically induced corrosion would be a contributing factor to the overall degradation of the waste package.

Shielding analyses were also performed on equipment to determine radiation exposure during the welding of the waste package closure lids. Various pieces of monitoring and control equipment, such as the welding heads and camera, would be close to radiation sources. The results of the shielding analyses will be used to quantify the shielding necessary for a piece of equipment to function properly at a given location for a required period of time.

In emergency situations, which could occur during the transport of waste packages from the surface facilities to the emplacement drifts or during the emplacement of waste packages in the drifts, personnel may have to enter areas near waste packages. Shielding analyses provide an evaluation of the radiation environment surrounding the waste packages so that worker safety can be ensured.

3.5.4.1 Source Term

Engineering calculations were performed to generate source terms, which are used to evaluate an upper limit for the surface dose rate of waste package. The source terms calculated for both pressurized water reactor and boiling water reactor spent nuclear fuel have the following characteristics: 5.5 percent (by weight) initial uranium-235, 75.0 GWd/MTU, and a 5-year decay time for the active fuel region; and 0.711 percent (by weight) initial uranium-235, 75.0 GWd/MTU burnup, and a 5-year decay time for the hardware regions of the assembly. The rationale for these assumptions is discussed in *PWR Source Term Generation and Evaluation* (CRWMS M&O 1999k) and *BWR Source Term Generation and Evaluation* (CRWMS M&O 1999l).

3.5.4.2 Results

The calculated maximum dose rate at the external surface of a 21-PWR Absorber Plate waste package is 1,130 rem/hr (+/-60 rem/hr) (CRWMS M&O 2000bl). The calculated maximum dose rate at the external surface of a 44-BWR waste package is 1,409 rem/hr (+/-32 rem/hr) (CRWMS M&O 2000bl, Section 6.2.3). Radiolytically enhanced corrosion is expected to be insignificant because the gamma dose on the surface of the waste package will not affect the corrosion properties of the waste package (see Sections 4.2.3.1.4 and 4.2.4.3.3).

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4. DISCUSSION OF DATA RELATING TO THE POSTCLOSURE SAFETY OF THE SITE

Section 114(a)(1)(C) of the Nuclear Waste Policy Act of 1982 (NWPA), as amended (42 U.S.C. 10134(a)(1)(C)), requires "a discussion of data, obtained in site characterization activities, relating to the safety of such site." This report presents a summary of the results of site investigations, design studies, and analyses of the performance of a potential repository at Yucca Mountain that began in 1978. Since 1986, the U.S. Department of Energy (DOE) has performed these studies as part of a formal program of site characterization that addresses regulations of the U.S. Nuclear Regulatory Commission (NRC). The program is also reviewed by the Nuclear Waste Technical Review Board, the State of Nevada, affected units of local government, and others. This section presents a summary of the data collected during site characterization as they relate to analyses of the postclosure safety of the site. The discussion is divided into six major parts:

- Section 4.1 presents an overview of the postclosure safety assessment approach used by the DOE to evaluate whether Yucca Mountain can safely isolate nuclear waste. It describes the methods the DOE has used to qualitatively and quantitatively assess the safety of the system. The information presented in this section provides a context for understanding how the data collected are related to safety. The discussion includes explicit recognition of the inherent uncertainty in analyses of future performance, and the potential consequences of alternative conceptual models or unexpected events. Methods to accommodate or mitigate uncertainty are summarized. The regulatory requirements for the performance assessment are also explained.

- Section 4.2 describes the data collected during site characterization and explains the DOE's conceptual understanding of the possible future behavior of the potential repository system, based on the data collected. The description is focused on processes important to safety (i.e., those that could affect a radionuclide release). The discussion is organized around the hydrologic and geologic processes that would operate in the repository system over time. For each component of the system, the text describes the data and information that form the basis of the DOE's understanding, and how the physical processes have been captured, or "abstracted," in the performance assessment. The processes affecting system components described in Section 4.2 include, for example, flow in the unsaturated zone, coupled thermal-hydrologic-geochemical processes near the repository, degradation of drip shields and waste packages, flow into and out of the waste packages, dissolution of the waste form, and transport of radionuclides away from the repository.

- Section 4.3 describes features, events, and processes (FEPs) that, if they occurred at Yucca Mountain, could affect repository performance. The discussion explains how scenarios (i.e., combinations of FEPs used to represent possible future conditions) have been defined that represent possible future conditions and behavior in the potential repository. These scenarios are the basis for the numerical analyses captured in the total system performance assessment (TSPA) and include all of the conditions and processes expected to operate at the repository, as well as disruptive events that could affect performance. Specific potentially disruptive events relevant to Yucca Mountain are described in detail.

- Section 4.4 explains the methods used to quantitatively assess the performance of the potential Yucca Mountain repository and presents the quantitative results of the TSPA for a nominal scenario (i.e., for the 70,000-MTHM base-case layout and higher-temperature mode of operations) and disruptive scenario (i.e., igneous activity), as well as a human intrusion scenario. The purpose of the TSPA analysis is to determine whether a Yucca Mountain repository could adequately protect public health and safety where the safety standard is defined by regulations that specify limits on allowable radiation doses to the public over the next 10,000 years. The effectiveness of the potential repository system is assessed by probabilistically analyzing performance (i.e., calculating dose rates, expressed as mrem per year) for a wide variety of possible future behaviors. Sensitivity analyses for the various scenarios of environmental conditions are presented to provide insight to the processes and model parameters that most influence TSPA results.

- Section 4.5 describes and explains qualitatively and quantitatively the performance capability of the natural and engineered barriers at Yucca Mountain.

- Section 4.6 describes the proposed program of performance monitoring, testing, and site stewardship that would be conducted at the potential repository. This program has been designed to provide additional confidence that a repository could be safely constructed and operated. It would include a performance confirmation program designed to monitor repository performance during and after its operation to verify that the technical basis for the program is sound. It would also include measures to maintain the ability to retrieve any or all of the waste at any time prior to closure.

This section emphasizes the scientific and engineering data and analyses related to the safety of the Yucca Mountain site. Most of the detailed scientific and engineering data is presented in Sections 4.2 and 4.3, which describe in detail the subsystem processes and the possible disruptive events that would control the performance of the potential repository. Although many of the concepts and descriptions presented are technically complex, the discussion is, to the extent possible, presented in nontechnical terms, so the information is accessible to non-technical readers.

4.1 THE POSTCLOSURE SAFETY ASSESSMENT METHOD

Assessing how a repository will perform over the next 10,000 years and beyond is a challenge for both the DOE and regulators. The limitations to the analyses and the uncertainties inherent in future system behavior cannot be completely eliminated by further testing or modeling. For this reason, the DOE has adopted an approach that relies on multiple lines of evidence to evaluate whether or not a repository at Yucca Mountain could adequately isolate and contain waste during the compliance period. This approach is documented in the *Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations* (CRWMS M&O 2001a, Volume 2). The postclosure safety case depends on combining sound science and engineering practice with informed judgment and planning. The postclosure safety case is described here because it provides a context for understanding how data and analyses presented throughout the rest of this section are related to safety.

The first element of the postclosure safety case is a thorough and quantitative evaluation of the possible future performance of the repository. This element is based on a comprehensive testing program that has evolved to address identified uncertainties, and an engineered barrier design developed specifically to work in combination with the natural barriers of the site. U.S. Environmental Protection Agency (EPA) and NRC regulations specify the method by which the DOE will analyze whether a repository can safely isolate spent nuclear fuel and high-level radioactive waste (i.e., a TSPA). The TSPA is described briefly in Section 4.4 but in more detail in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a) and several subsequent documents including *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). These reports include analyses of all the processes expected to operate at the repository that could affect its ability to isolate waste. They also explicitly consider both disruptive events and alternative process models that could result in unanticipated behavior (i.e., identifies what could go wrong). These evaluations directly address uncertainty in both the DOE's knowledge of the site and in future conditions and include numerical sensitivity analyses to test how the repository might perform if current or future conditions differ from those expected.

To capture the technical inputs used in developing the overall TSPA system level model, a set of analysis model reports have been prepared. These reports contain the detailed technical information regarding data, analyses, models, software, and supporting documentation that is used in the development of the process models. The analysis model reports provide the direct input into the TSPA analyses, as well as document the abstraction of the process level models for use in the overall TSPA system level model.

Using the analysis model reports as a basis, the descriptions of these process level models are documented in a suite of process model reports that cover the following areas:

- Integrated site model (CRWMS M&O 2000i)
- Unsaturated zone flow and transport (CRWMS M&O 2000c)
- Near-field environment (CRWMS M&O 2000al)
- Engineered barrier system degradation, flow, and transport (CRWMS M&O 2000as)
- Waste package degradation (CRWMS M&O 2000n)
- Waste form degradation (CRWMS M&O 2000bm)
- Saturated zone flow and transport (CRWMS M&O 2000bn)
- Biosphere (CRWMS M&O 2000bo)
- Disruptive events (i.e., seismicity and volcanism) (CRWMS M&O 2000f).

The process model reports synthesize the information contained in the individual analysis model reports and provide an integrated perspective for understanding each of the process level models. This hierarchical process of documentation was used to ensure the traceability of supporting information from its source through the analysis model reports and process model reports to its eventual use in the TSPA. Supplemental analyses at both the process model level, and the total system level, are presented in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b).

Because the DOE recognizes that uncertainty about the future performance of the repository cannot be completely eliminated, the postclosure safety case includes several additional measures designed to provide confidence and assurance that the repository will meet postclosure performance standards. These measures include:

- Qualitative (and sometimes quantitative) insights gained from the study of natural and man-made analogues to the repository or to processes that may affect repository performance. Analogue observations are especially useful in the analysis of 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 postclosure health and safety requirements. The DOE has implemented this approach through the selection of a specific location in the unsaturated zone at Yucca Mountain and the development of a design with multiple natural and engineered barriers to the migration of radionuclides. The engineered components of the site are designed specifically to complement the natural attributes of the potential repository host rock. 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.

- A commitment to a performance confirmation program and long-term management and monitoring 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. This commitment includes maintaining, for a period of up to 300 years, the ability to retrieve the spent nuclear fuel and high-level radioactive waste before closure for any reason, if future generations decide that doing so would be desirable.

This approach is similar to that recommended by many national and international professional organizations that have studied 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).

The various elements of the postclosure safety case contribute in different ways to building confidence in analyses of the long-term performance of the repository. Quantitative numerical models permit scientists to test their understanding of the site and assess the consequences of uncertainty or assumptions in their models. Observations of natural or man-made analogues can help scientists determine whether the results of repository models are consistent with the behavior of actual systems. They can also be used to qualitatively evaluate the reliability and uncertainties associated with modeling. Safety margin and defense in depth provide one way to compensate for uncertainty in analyses. Long-term monitoring can help scientists verify that the uncertainties in site and design performance have been appropriately characterized. In total, the elements documented in the *Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations* (CRWMS M&O 2001a, Volume 2) support the postclosure safety case for the Yucca Mountain site. The sections that follow contain additional descriptions of each element of the safety case.

4.1.1 Total System Performance Assessment

Analysis of the future performance of the potential repository is fundamental to the DOE's understanding of the Yucca Mountain site. Therefore, the first element of the safety case is a thorough analysis of how a repository at Yucca Mountain would behave in the future. As noted previously, the methods used and the results of the TSPA for Yucca Mountain are described in Sections 4.3 and 4.4.

Performance assessment is a method or tool defined and provided by the EPA and the NRC (in 40 CFR Part 197 and 10 CFR Part 63 [66 FR 55732], respectively) for the evaluation of a Yucca Mountain repository. The objective of the total system performance assessment for site recommendation (TSPA-SR) for Yucca Mountain is to provide a basis for evaluating whether the safety of the general public will be protected. However, the DOE has also used the performance assessment for broader purposes during site characterization of Yucca Mountain. For instance, it has been used as a tool to evaluate the effects of uncertainty on total system performance and to identify areas where further work is needed. This has been accomplished in an iterative manner. For this updated *Yucca Mountain Science and Engineering Report*, TSPA results are presented and discussed in one comprehensive report, summarizing several additional supplemental documents that describe analyses performed to address specific technical and/or regulatory issues. The key TSPA references include:

- *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a). This report describes a comprehensive analysis of the performance of a repository at Yucca Mountain. It documents the TSPA methodology and explains how process models have been incorporated in the TSPA-SR analysis.

- *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b). This two-volume report describes additional analyses performed to examine the uncertainty in TSPA analyses of Yucca Mountain. Volume 1 (BSC 2001a) describes new technical information and models of processes that may be important to performance; it provides additional quantitative analysis of uncertainties and investigates the effect of the thermal operating mode of the repository on the uncertainty associated with process models. Volume 2 (BSC 2001b) documents supplemental total system analyses based on an updated TSPA model (referred to in this report as the supplemental TSPA model), based on information in 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 document describes revised TSPA analyses (referred to in this report as the revised supplemental TSPA model) consistent with EPA's final 40 CFR Part 197 rule, including distance to the accessible environment, and calculation of radionuclide concentrations in groundwater at approximately 18 km (11 mi) from the repository.

- *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b). This report describes additional TSPA sensitivity analyses to address final 10 CFR Part 63 (66 FR 55732) provisions, including the possible treatment of unlikely events (igneous intrusion) in assessing groundwater protection and human intrusion, and the effects of a change to 3,000 acre-ft/yr water demand for evaluation against the individual protection standard.

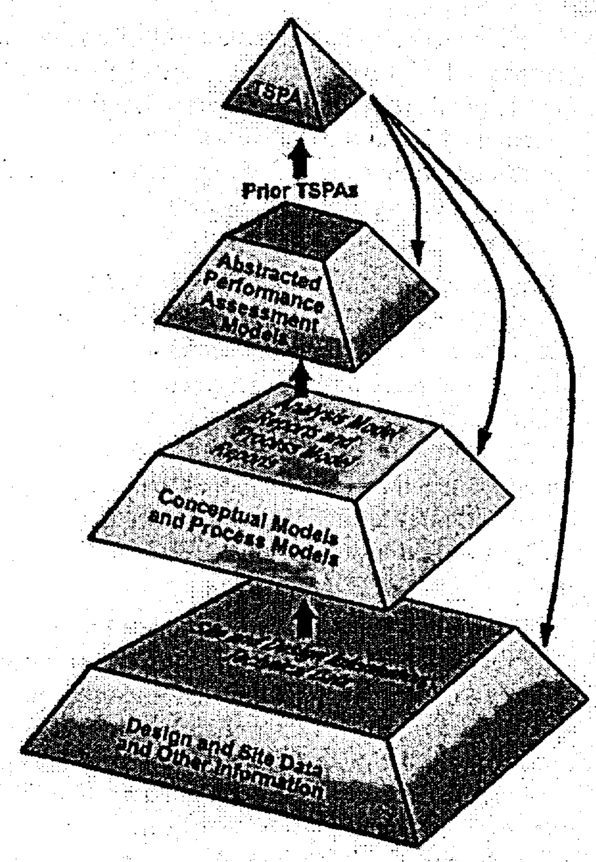

Figure 4-1. Total System Performance Assessment Pyramid Illustrating the Progressive and Iterative Process of Synthesizing Design Information, Site Data, Process Models, and Total System Performance Assessment Expertise

Figure 4-1 schematically presents the information flow within an iterative TSPA approach. Information gained during studies and testing flows upward through the development of conceptual and numerical models that provide the basis for the TSPA. Information gained through analysis flows downward in the form of specific information needs for the next iteration of the TSPA.

The foundations of the TSPA are site characterization and engineering design data. Project scientists and engineers use this information to formulate conceptual models of the FEPs that could affect the performance of the natural and engineered barriers at the site. An important step in the formulation of these conceptual models is the identification of uncertainties in the current state of knowledge.

These conceptual models, the level of uncertainty associated with each, and the essential assumptions used in their formulation are documented in Sections 4.2 and 4.3, as well as supporting documents describing the DOE's understanding of the Yucca Mountain repository system.

The conceptual models are next cast into process-level models. These are typically numerical computer models that range from simple to quite complex representations designed to capture and simulate the fundamental physical phenomena that influence the process being modeled. The process models, together with important uncertainties and assumptions, are also described in Sections 4.2 and 4.3, as well as in supporting references.

Many process level models are needed to analyze the various subsystems that could affect the performance of the potential repository. Some of the individual models are so complex that it is not possible or desirable to include them in a single linked total system model, due to computing limits or because simplified models of certain processes may be equally defensible. A total system model that depended on a single representation of a process might overlook credible alternative models. Therefore, the total system model is generally based on simplified (abstracted) models that enable assessment of the effects of alternative representations of potentially important processes.

The process level models described in Sections 4.2 and 4.3 provide the foundation for the abstracted models contained in the TSPA described in Section 4.4. These abstracted models include the important details of the process level models, so they can be used to simulate or bound the results of the process level models. Scientists and engineers evaluate the output of the detailed process level models to identify key results, uncertainties, and assumptions that must be captured by the abstracted models. For example, it may be determined that, of the many processes and parameters contained within a process level model, only a few have a significant effect on overall behavior. Analysts use test results, comparisons with alternative process level models, and judgment to determine how best to incorporate the uncertainty associated with each specific process in the abstracted models. The complexity of an abstracted model is governed by how sensitive total system performance is to the specific process and by how well the model incorporates uncertainty. The abstracted TSPA models, associated uncertainties, and important assumptions are described in Section 4.4 and in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a). More recent 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).

The abstracted models are combined into a total system model, which is used both to assess the potential future performance of the repository and to evaluate how uncertainty in the understanding of FEPs might affect performance. Throughout site characterization, the DOE has used this information to identify and prioritize future work activities. In this manner, the testing program has been updated and the repository design modified to continually improve the DOE's confidence in assessments of future performance. In the past decade, the DOE completed comprehensive analyses of total system performance in 1991, 1993, 1995 and 1998 (Barnard et al. 1992; Eslinger et al. 1993; Wilson, M.L. et al. 1994; CRWMS M&O 1995; DOE 1998, Volume 3). Each of these represented a significant advance in the DOE's understanding of how the potential Yucca Mountain repository might perform, and each resulted in modifications to the site testing program or the repository design to address key uncertainties. This report and the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a) reflect the knowledge and insights gained through this process. Additional analyses and insight are presented 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).