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Engineering for a Disposal Facility Using the In-Room Emplacement Method

Études techniques d'une installation de stockage permanent faisant appel à la méthode de mise en place en chambre

P. Baumgartner, D.M. Bilinsky, Y. Ates, R.S. Read,
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THE IN-ROOM EMPLACEMENT METHOD

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Atomic Energy of Canada Limited
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ABSTRACT

This report describes three nuclear fuel waste disposal vaults using the in-room emplacement method. First, a generic disposal vault design is provided which is suitable for a depth range of 500 m to 1000 m in highly stressed, sparsely fractured rock. The design process is described for all components of the system. The generic design is then applied to two different disposal vaults, one at a depth of 750 m in a low hydraulically conductive, sparsely fractured rock mass and another at a depth of 500 m in a higher conductivity, moderately fractured rock mass. In the in-room emplacement method, the disposal containers with used-fuel bundles are emplaced within the confines of the excavated rooms of a disposal vault.

The discussion of the disposal-facility design process begins with a detailed description of a copper-shell, packed-particulate disposal container and the factors that influenced its design. The disposal-room generic design is presented including the detailed specifications, the scoping and numerical thermal and thermal-mechanical analyses, the backfilling and sealing materials, and the operational processes. One room design is provided that meets all the requirements for a vault depth range of 500 to 1000 m. A disposal-vault layout and the factors that influenced its design are also presented, including materials handling, general logistics, and separation of radiological and nonradiological operations. Modifications to the used-fuel packaging plant for the filling and sealing of the copper-shell, packed-particulate disposal containers and a brief description of the common surface facilities needed by the disposal vault and the packaging plant are provided. The implementation of the disposal facility is outlined, describing the project stages and activities and itemizing a specific plan for each of the project stages: siting, construction, operation; decommissioning; and closure.

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ÉTUDES TECHNIQUES D'UNE INSTALLATION DE STOCKAGE PERMANENT
FAISANT APPEL À LA MÉTHODE DE MISE EN PLACE EN CHAMBRE

par

P. Baumgartner, D.M. Bilinsky, Y. Ates, R.S. Read,
J.L. Crosthwaite et D.A. Dixon

RÉSUMÉ

Le présent rapport donne une description de trois installations de stockage permanent de déchets de combustible nucléaire aménagées selon la méthode de stockage en chambre. Ce rapport donne en premier lieu un modèle générique d'installation de stockage qui convient à des profondeurs de 500 à 1 000 m dans la roche légèrement fracturée sous fortes contraintes. Il précise le processus de conception de tous les éléments du système. Le modèle générique est ensuite appliqué à deux installations de stockage différentes dont une est située à une profondeur de 750 m dans un massif rocheux de faible conductivité hydraulique et peu fracturé, et l'autre à une profondeur de 500 m dans un massif rocheux modérément fracturé, mais de conductivité hydraulique plus élevée. Dans la méthode de mise en place en chambre, les conteneurs de stockage remplis de grappes de combustible irradié sont placés à l'intérieur des chambres de l'installation de stockage.

Le processus de conception de l'installation de stockage est présenté en commençant par une description détaillée d'un conteneur de stockage à particules tassées à enveloppe de cuivre et des facteurs qui ont influé sur sa conception. Le modèle générique de chambre de stockage y est présenté comprenant toutes les spécifications, l'analyse de la portée et les analyses thermiques numériques et thermomécaniques, l'étude des matériaux de scellement et de remblai, et les processus d'exploitation. Un type particulier de chambre satisfait à toutes les exigences d'une installation creusée entre 500 et 1 000 m de profondeur. On présente également dans le rapport l'aménagement d'une installation de stockage et les facteurs qui ont influé sur sa conception, y compris la manutention des matériaux, la logistique générale et la démarcation entre les opérations à caractère radiologique et non radiologique. On y trouve aussi les modifications apportées à l'usine de conditionnement du combustible irradié qui servira au remplissage et au scellement des conteneurs de stockage à particules tassées à enveloppe de cuivre et une brève description des installations de surface communes nécessaires à l'installation de stockage et à l'usine de conditionnement. On y trace les grandes lignes de la mise en oeuvre de l'installation de stockage, en s'arrêtant aux différentes étapes et activités du projet et en dressant une liste des jalons particuliers à toutes les étapes du projet : choix du site, construction, exploitation; déclassement; et fermeture.

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1. INTRODUCTION

In the Nuclear Fuel Waste Management Program (NFWMP), Atomic Energy of Canada Limited (AECL) with the assistance of Ontario Hydro have been developing and assessing the technology for the safe geological disposal of nuclear fuel waste from CANDU® reactors. The Canadian concept for nuclear fuel waste disposal requires the creation of underground excavations at depths ranging from 500 to 1000 m in plutonic rock of the Canadian Shield. Some of these excavations (e.g., the disposal rooms) will hold the nuclear fuel waste that will be contained in corrosion-resistant containers. After the waste is suitably emplaced, all the disposal rooms and access excavations will be backfilled and sealed with engineered barrier materials (Simmons and Baumgartner 1994). The Environmental Impact Statement (EIS) (AECL 1994) on the concept for disposal of Canada's nuclear fuel waste contains pre- and postclosure assessment case studies based on a conceptual design of a single-level, used-fuel disposal vault using a borehole emplacement method for the titanium-shell disposal containers (Simmons and Baumgartner 1994). In this design, emplacement boreholes are drilled into the rock of the floor of the disposal rooms (Figure 1).

AECL, since submitting the EIS in 1994 October, performed several additional studies including:

- A study to show how to identify a favourable vault location that would ensure long groundwater travel times from the vault to the accessible environment (Ophori et al. 1995, 1996; Stevenson et al. 1995, 1996); and

- A study that illustrates (i) the flexibility for designing engineered barriers to accommodate a permeable host-rock condition in which advection is the dominant contaminant transport process, and (ii) the flexibility of the modelling methodology to simulate the long-term performance of different design options and site characteristics (Wikjord et al. 1996).

This report provides a new waste emplacement method to support the objectives of these additional studies and illustrates an approach to disposal vault design that, along with carefully selected engineered barriers, is applicable to a broad range of conditions in the Canadian Shield.

1.1 STUDY OBJECTIVES

The objectives of this engineering study are to show:

- a rational approach to developing a robust disposal facility design with the flexibility for broad application within the high stress, sparsely fractured rock conditions in the Canadian Shield (i.e., the robust vault objective);

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- a disposal facility design that is suitable for a favourable vault location that would ensure long groundwater travel times from the vault to the accessible environment (i.e., the favourable vault site objective); and

- a disposal facility design that is suitable for a permeable host-rock condition in which advection is the dominant contaminant transport process (i.e., the permeable geosphere design objective).

Three disposal vault designs using the in-room emplacement method are described to satisfy the three objectives. The first disposal vault is a generic design to meet the robust vault objective, suited for the high horizontal in situ stress, sparsely fractured granitic rock conditions of the Whiteshell Research Area throughout a depth range of 500 to 1000 m. It allows the reader to follow a robust and flexible design process through to a preliminary design. The disposal vault with the in-room emplacement method is expected to have a broad application within the highly stressed, sparsely fractured conditions expected at potential sites in the Canadian Shield.

The second disposal facility is a specific design to meet the favourable vault site objective. It is a derivative of the generic design because the favourable geosphere conditions, specified by Ophori et al. (1995, 1996) and Stevenson et al. (1995, 1996), are defined to be the high horizontal in situ stress, sparsely fractured rock conditions of the Whiteshell Research Area. They specified a vault depth of 750 m at their favourable location.

The third disposal facility is also a specific design to meet the permeable geosphere design objective. It is a derivative of the generic design with a section of the waste emplacement area[1] of the vault removed around a transecting low-angle fault. The geosphere conditions are defined as a permeable, moderately fractured granite rock mass with a vault depth set at 500 m. Although the specific rock strength and ambient in situ stresses are not as well defined for these rock conditions as for sparsely fracture rock, the disposal-room geometries for sparsely fractured rock are retained. Arguments are made for changes to the major and minor dimensions of the rooms in moderately fractured rock to accommodate differences in strength and in situ stresses between sparsely fractured and moderately fractured rock.

The disposal facility designs are produced in the context of other important engineering objectives including:

- Providing a feasible design that is constructible, operable and has a reasonable implementation schedule while meeting the safety criteria;

- Demonstrating a design integration process whereby the disposal facility design evolves to meet changing objectives and specifications; and

---

1 The waste emplacement area is defined as the plan area of the disposal vault that contains the waste disposal rooms and their associated access tunnels.

- Providing a specific design description of a disposal facility with sufficient detail to allow the conduct of a credible performance assessment.

The approach taken in this report is to produce a generic disposal-room, vault, packaging plant and surface facility design to meet these objectives, followed by the application of the generic design to two specific vaults at the different depths and site conditions.

In this study, the method of waste emplacement is within the confines of the disposal room (i.e., in-room emplacement) and the disposal-container shell is copper (Figure 2). In addition to the choice of the disposal container design, several of the design assumptions in this study are revised from those used in the in-floor, borehole emplacement method that is the reference case for the Environmental Impact Statement (AECL 1994, Simmons and Baumgartner 1994). The major changes are as follows:

- the assumed ambient in situ stress conditions, the Young's modulus, and the coefficient of thermal expansion of rock are increased to be representative of the conditions likely to exist in sparsely fractured granite;

- the average burnup of the used fuel is increased to reflect current experience in reactor operations;

- the design limit for the temperature of the outer surface of the disposal container is decreased to provide an appropriate design margin to the 100°C design requirement used in the in-floor borehole emplacement case (Simmons and Baumgartner 1994);

- the in-room emplacement method is selected to provide greater flexibility in excavation design for the high stress environment and to provide greater thicknesses in buffer and backfill materials around the disposal containers; and

- the rock mass strength design limits are updated based on the improved understanding of the performance of sparsely fractured rock.

The major emphasis of this report is the disposal vault design and the implementation requirements for a disposal facility. The balance of the used-fuel disposal facility, such as the packaging facilities and common support facilities to the packaging and disposal operations, are very similar to that of the disposal facility study using in-floor borehole emplacement (Simmons and Baumgartner 1994, AECL CANDU et al. 1992) and are not fully described in this report. The components of these facilities that are changed in this study are highlighted and described in sufficient detail for report completeness.

The conceptual-level designs discussed in this report are generic. They are not fully related to conditions at any particular site, although many of the site conditions assumed are drawn from the results of AECL's studies at the Whiteshell Research Area and surrounding region, including the Underground Research Laboratory. Therefore, the data that would normally evolve from site-specific field activities have been taken from available sources so the design analyses necessary for the conceptual design could be completed. This report discusses the assumptions made and the analyses undertaken.

No formal system or component performance or cost optimization has been done in these studies because many of the criteria for optimization should be established with the implementing organization. However, the design process has examined optional disposal arrangements and vault layouts before arriving at the presented practicable conceptual designs.

1.2 ORGANIZATION OF THE BALANCE OF THIS REPORT

This report is arranged to address the objectives stated above. Chapter 1, this chapter, provides an introduction to and the objectives for the study and provides the relationship to other program activities. It also provides a generic description of the used-fuel disposal facility and its operation that will aid the reader in forming a mental picture of the facility before the design, the design process and the details are presented.

Chapter 2 describes the design process that yielded the general in-room emplacement concept and generic vault design, the two specific vault designs and the implementation description. The design process includes the setting of specifications in their initial and final form, the scoping analyses that were performed to arrive at the final disposal-room arrangement and the integration of the design components into a generic and two specific vault designs.

Chapter 3 describes the details of the generic disposal facility design. It begins with a description of a copper-shell, packed-particulate disposal container and the factors that influenced its design. A second container, the steel-shell-supported copper container, is introduced as an alternate design. The detailed disposal-room generic design specifications, the scoping and numerical thermal and thermal-mechanical analyses, the backfilling and sealing materials, and the operational processes are presented. One room design is provided that meets all the requirements for a vault depth range of 500 to 1000 m. Finally, the disposal vault layout and the factors that influenced its design including material handling and movement, general logistics, and the separation of radiological and nonradiological operations; the modifications to the used-fuel packaging plant for the filling and sealing of the copper-shell, packed-particulate disposal containers; and the common surface facilities needed by the disposal vault and the packaging plant are described.

Chapter 4 presents the specific vault design at a depth of 750 m suitable for the favourable vault site objective. A description is provided for the implementation of the disposal facility. It outlines the project stages and activities and itemizes a specific plan for each of the project stages; siting, construction, operation; decommissioning and closure.

Chapter 5 provides the second disposal vault design that supports the permeable geosphere design objective. The modification of the specifications for this vault is presented, together with the details of the derived vault layout. Since the purpose of this vault design is to provide information for the postclosure assessment of a vault in moderately fractured rock (i.e., the quantity of used fuel and its distribution within the vault), only these details are provided because the details of the disposal rooms are identical to that described in Chapter 3.

Finally, Chapter 6 reviews the objectives of this report, and discusses the technical feasibility of the in-room emplacement method and its implications on structural and operational performance.

1.3 GENERAL DESCRIPTION OF A DISPOSAL FACILITY

The used-fuel disposal facility includes surface facilities for the receipt and packaging of used fuel within corrosion-resistant disposal containers and a disposal vault excavated in plutonic rock. The disposal facility is self-contained, except for the supply of materials and disposal-container components and is located on a suitable rock body of the Canadian Shield. The facility is designed to receive, package and dispose of used-fuel bundles at a rate of 250 000 bundles/a. The used-fuel bundles are assumed to have been discharged from the reactors for 10 years.

During operation, the used fuel is received at the packaging plant in either a road or rail transportation cask that contains the used-fuel bundles in storage/shipping modules. The modules are unloaded from the casks in a module-handling cell. In the packaging cell, the used-fuel bundles are transferred from the shipping modules to stainless steel disposal-container fuel baskets, 72 bundles to a basket, and each fuel basket is installed within a copper-shell disposal container. Each bundle and container is monitored for nuclear material safeguards purposes during all the transfer and disposal operations. The heat generated by 72 used-fuel bundles in a container is about 330 W.

The disposal-container shell and end closures assumed in this conceptual design are fabricated from deoxidized, low-phosphorus grade copper with a minimum thickness of 25.4 mm. The baskets, shells and top heads are assumed to be fabricated by an off-site supplier and shipped to the disposal facility. The loaded container is filled with a dry particulate, such as glass beads or fractionated silica sand, which is vibrationally compacted to fill all the residual void, allowing the container to resist the external loads. A top head is pressed into the container and the seam is electron-beam welded. The derived packaging rate is about 3470 containers/a.

After nondestructive testing of the electron-beam weldment by ultrasonic inspection to establish the integrity of the top head-to-shell joint, the outer surface of each container is decontaminated and the container is loaded into a shielding container cask. Each full cask is mounted horizontally on a rail-mounted cask car and transferred to the disposal vault using the conveyance in a shaft dedicated for waste transfers to the vault level. The loaded cask car is removed from the conveyance at the vault level and an empty cask, also mounted on a cask car, is returned to surface. The loaded cask car is rolled by locomotive to either an underground surge-storage area or directly to a disposal room.

The disposal vault is a single-level, room-and-pillar type of excavation designed for in-room emplacement of individual disposal containers. The waste emplacement area of the vault is essentially square with a plan area of about 4 km^2. It consists of 512 disposal rooms arranged within 8 panel areas serviced by over 20 km of access tunnels. Each disposal room has an elliptical-shaped cross-section, nominally 3-m high and 7.3-m wide, and a length of 238 m. Each room contains 158 containers, placed horizontally, two abreast at 2.21-m centre-to-centre spacing and at a longitudinal centre-to-centre spacing of 2.7 m, within a mass of precompacted buffer blocks and associated sealing materials and structures (Figure 2). The centre-to-centre spacing between disposal rooms is 30 m.

This disposal vault arrangement limits the maximum increase in temperature at the surface of the disposal container to 64°C, which is reached at 15 years after emplacement. For a disposal vault depth of 500 m, the maximum container surface temperature is 75°C and at a depth of 1000 m, the surface temperature is 81°C, when the temperature rise is added to the ambient temperatures at these depths. These maximum temperatures are less than the design limit requirement of 90°C because the longitudinal spacing of 2.7 m between container centres is controlled by the radiological shielding required of the sealing materials placed between the disposal containers and the area that is occupied by the workers. The radiation dose to workers placing the sealing materials and disposal containers should not exceed 1 μSv/h.

The disposal rooms are excavated by the drill-and-blast method. A low-heat, high-performance concrete floor structure is placed, rails are installed for the rail-mounted equipment and ventilation and utilities are installed. The concrete floor provides a uniform base for the accurate placement of precompacted blocks of the dense backfill and buffer sealing materials and for the rail. The rail provides a horizontal datum for the alignment of the container cask for container transfer to the prepared chambers formed in the preplaced mass of sealing materials.

Before a container cask is received in the disposal room, 21 specially shaped precompacted blocks of dense backfill and 24 precompacted blocks of buffer are placed along a 2.7-m long section of the room. The shape and arrangement of blocks provides two horizontal, cylindrical chambers, each receiving a disposal container and three shielding/sealing plugs. The gap between the top of the placed blocks and the roof of the room is filled with a pneumatically placed light backfill. When ready, the container cask car is positioned before one chamber and the container is slid into the back of the chamber. Three precompacted buffer plugs are slid, one at a time, into the remaining portion of the chamber to fill it and to provide shielding to the workers when the container cask is withdrawn. Any void between the container and the buffer mass is filled by a pneumatically placed dry granular bentonite and rounded sand mixture that is installed through a hollow lance inserted into the top gap formed between the buffer plugs and preplaced blocks. The container emplacement procedure is repeated for the second chamber.

The block placement procedure followed by container emplacement is repeated as work progresses in a retreat fashion out of the disposal room. After 158 containers have been placed, a low-heat, high-performance concrete bulkhead is constructed at, and grouted into, the room entrance to seal the room and to withstand the buffer and backfill swelling and the groundwater pressures. A safeguards seal system may be incorporated into the bulkhead to detect unauthorized entry.

The operational sequence in the disposal vault consists of disposal-room excavation, room preparation, repetitious precompacted block, pneumatic light-backfill and waste container emplacement, and concrete bulkhead placement and sealing. Eight disposal rooms are being filled at any one time on a three-shift per day, seven-day per week basis. While assemblages of precompacted blocks are being placed in four rooms, waste containers are being installed in the horizontal chambers of the prepared assemblages in the other four rooms. As waste emplacement occurs in one panel of the vault, excavation occurs in the next panel on the opposite side of the vault central access tunnels. In addition, sealing material blending and mixing and block compaction is simultaneously performed in the buffer and backfill preparation and block compaction plants located underground adjacent to the service shaft complex. All underground transportation is provided by rail-mounted equipment. The duration of the operational period of a disposal vault is a function of the annual throughput rate and the total quantity of waste to be disposed. For the 750-m deep vault, with an inventory of 5.8 million used-fuel bundles, the operational period is about 23 years, and for the 500-m deep vault with 4.3 million bundles, it is about 17 years.

When all the waste is emplaced, sufficient monitoring data have been collected and assessed to show design and regulatory compliance, and the regulators have approved the decommissioning and closure plan for the disposal facility, the facility will be decommissioned, backfilled and sealed. Backfilling and sealing of tunnels and shafts are achieved using similar materials and methods to those used in the disposal rooms. The surface facilities will be decommissioned, decontaminated and fully disassembled, and the site will be returned to a state suitable to allow public use of the surface.

2. DESIGN PROCESS LEADING TO DISPOSAL FACILITY DESCRIPTIONS

The disposal facility design process that leads to the two disposal vault descriptions provided in this report involved the setting of specifications and the production of preliminary designs and analyses to scope out the specific design options (Baumgartner et al. 1995).

2.1 GENERAL STUDY SPECIFICATIONS

The original specifications for the in-room emplacement study were similar to those used in the reference borehole emplacement case (Baumgartner et al. 1993). The major key exceptions are as follows (see also Table 1):

- The nuclear fuel waste is to be emplaced within the confines of the disposal room unlike the in-floor borehole emplacement method where the waste is emplaced within vertical boreholes drilled 5 m into the floor of the disposal room (Simmons and Baumgartner 1994).

- The depth of the disposal vault is nominally between 500 and 1000 m; no specific depths were specified until analyses from the other studies (Ophori et al. 1995, 1996, Stevenson et al. 1995, 1996, Wikjord et al. 1996) yielded decisions on preferred disposal facility locations and depths.

- The primary design of the copper-shell, packed-particulate disposal container has overall dimensions of 860-mm in diameter and 1189-mm in length and contains 72 used-fuel bundles. A secondary design, the steel-shell-supported copper disposal container, is provided and has overall dimensions of 994-mm in diameter and 1349-mm in length and also contains 72 used-fuel bundles.

- The reference used-fuel bundle is a Bruce Nuclear Generating Station bundle that has an average burnup of 720 GJ/kg U (200 MW·h/kg U) and that has been cooled for 10 years after discharge from a reactor before disposal. This burnup value is used for the thermal source term and the disposal vault temperature calculations. A second, above average burnup of 1008 GJ/kg U is used for the radiation shielding calculations needed to ensure the protection of the workers should occasional batches of high burnup fuel enter the disposal system.

- The maximum design limit for the surface temperature of the disposal container is 90°C.

- The maximum allowable radiation dose to workers is 20 mSv/a to be consistent with an Atomic Energy Control Board proposal (AECB 1991). Under normal operating conditions, this is reduced in the study to 2 mSv/a, that is, 1 µSv/h when a nominal duration of 2000 h/a is used for individual workers within the disposal room.

- The minimum thickness of buffer used to surround each disposal container is 0.5 m.

- The maximum hydraulic conductivity of the buffer is less than 10^{-11} m/s and less than 10^{-10} m/s for the other clay-based sealing materials.

- Low-heat, high-performance concrete is used within the disposal room.

- The ambient in situ stresses used in this study for the sparsely fractured plutonic rock of the Canadian Shield are in the upper range of measurements, including those at the Underground Research Laboratory.

- The rock mass properties are based on measurements in granites of the Canadian Shield including the measurements made in the Underground Research Laboratory.

- The Hoek and Brown (1980) empirical failure criterion model is used for the rock mass stability analyses. The peak strength design limit of the rock under excavation mechanical load conditions is σ_c = 100 MPa, m = 16.6 and s = 1. The peak strength design limit of the rock under full thermal-mechanical load conditions is σ_c = 150 MPa, m = 25 and s = 1. The m and s parameters define the triaxial strength characteristics of the rock mass. The factor of safety is calculated as the ratio of the strength to the stress acting at any point in the rock mass. Stable excavations are defined when all points around the excavations have factors of safety of 1.0 or greater.

- The nominal extraction ratio is set between 0.25 and 0.3. The extraction ratio is calculated by comparing the width of the disposal rooms with the combined width of the rooms and their separating pillars (i.e., the centreline distance between rooms).

- The nominal waste emplacement area of the disposal vaults is set at 4 km^2, similar to the value derived in the in-floor borehole emplacement method (Simmons and Baumgartner 1994). The total quantity of used fuel to be disposed in a disposal vault is derived based on the design specifications listed above.

Further details of the specifications are presented in the following subsections and some discussions of their derivations are disseminated throughout the balance of the report.

2.1.1 Used-Fuel Characteristics

The reference CANDU fuel bundle is the one designed for the Bruce Nuclear Generating Station, and it is the basis for the disposal-container design, the disposal-vault thermal analyses and the radiological shielding analyses. This bundle consists of 37 fuel elements and is about 495-mm long and 102 mm in overall diameter (Figure 3). The mass of the bundle is 23.7 kg and it contains 18.9 kg of elemental uranium (kg U) when initially inserted into a reactor. This forms the basis of all further analyses.

The average fuel burnup chosen for the thermal and thermal-mechanical calculations is increased to 720 GJ/kg U (200 MW·h/kg U) (Novak and Hastings 1991, Floyd et al. 1992) from the previously used value of 685 GJ/kg U (190 MW·h/kg U) (Baumgartner et al. 1993). The heat output per kilogram of contained uranium, 10 years after discharge from a reactor, is about 0.242 W or about 4.6 W for a used-fuel bundle. A conservative fuel burnup value of 1008 GJ/kg U (280 MW·h/kg U) was retained for the radiation shielding calculations, which takes into account the relatively wide range of burnup. About 90% to 95% of the used-fuel bundles would have a burnup less than this value. Fuel bundles for other CANDU nuclear generating stations are similar in composition and geometry and are amenable to the same packaging and disposal methods.

In this conceptual design, the cooling time for all used fuel received at the disposal facility is assumed to be 10 years after discharge from reactors. In practice, much of the used fuel in Canada would be considerably older than this and would have a correspondingly reduced heat and radiation output, by the time disposal would be implemented. This 10-year cooling period introduces a degree of conservatism to the design in terms of both radiological safety for workers and heat output from each disposal container. Further information about used-fuel characteristics is given by Johnson et al. (1994a, 1996).

2.1.2 Used-Fuel Quantities and Disposal Rate

The quantity of used fuel assumed for disposal in the in-room emplacement conceptual design is based on the waste emplacement area of the disposal vault. In the reference borehole emplacement case (Simmons and Baumgartner 1994), the waste emplacement area was about

4 km^2. Note that the waste emplacement area for the postclosure safety assessment in the EIS case study was truncated from 4 km^2 to 3.2 km^2 and the quantity of used fuel was reduced from 10.1 million to 8.5 million bundles. A similar 4 km^2 waste emplacement area is retained in developing the in-room emplacement vault designs to simplify comparisons between the designs, subject to any adjustments necessary to accommodate the assumed geological structure. Scoping thermal and radiological shielding analyses for this vault size showed that the quantity of CANDU used-fuel bundles that could be accommodated in this 4 km^2 area varied between a maximum of 5 to 5.8 million bundles (i.e., about 95 000 to 110 000 Mg of elemental uranium) depending on the specific disposal container spacing within the disposal rooms (Baumgartner et al. 1995). A container spacing of 2.7 m along the disposal-room length is selected from the detailed thermal and radiation-shielding analyses, described in Section 3.2. For a disposal vault at the 750-m depth with the full 4 km^2 waste emplacement area, the maximum quantity of waste is 5.8 million used-fuel bundles (Chapter 4).

The waste emplacement area, for the disposal vault at a depth of 500 m, is divided into two sections, one section on either side of a low-angle thrust fault that is assumed to pass though the disposal vault (Section 5.1). The horizontal distance between the two vault sections is 375 m, based on the geometry of the geosphere model (Stanchell et al. 1996). When this area is removed from the waste emplacement area, and allowances are made in the disposal vault layout for access tunnels (Section 5.1), the total quantity of waste that may be disposed in the 500-m deep vault is reduced to about 4.3 million fuel bundles. Although the overall disposal vault area excluding the shaft service areas is about 4 km^2, the effective waste emplacement area (i.e., the sections where disposal containers are emplaced) is about 3.4 km^2.

These quantities of waste are in the range of the estimated quantity of used fuel that is forecast to be generated by the currently operating nuclear power reactors in Canada (AECL 1994) (i.e., Curve 3 in Figure 4).

The derived conceptual design should be flexible and modular, allowing the capacity to be changed with no fundamental change in the proposed operations. The assumed capacity of the used-fuel transportation system to send used fuel from the nuclear generating stations to the disposal facility has been retained at 250 000 used-fuel bundles per year (Baumgartner et al. 1993).

2.1.3 Used-Fuel Disposal Container

The used-fuel disposal-container design chosen for the in-room emplacement study is a copper-shell, packed-particulate container with overall dimensions of 860 mm in diameter and 1189 mm in height (Figure 5, Section 3.1). It consists of a copper shell holding a basket of 36 bundle-retaining tubes. The basket consists of thin-walled, stainless steel tubes, 110 mm in diameter and 995-mm long, placed around a central thick-walled pipe. Each of the 36 tubes holds two used-fuel bundles, for a total of 72 bundles in the basket. The basket is contained in a 25.4-mm-thick shell, composed of deoxidized, low-phosphorous copper. This replaces the previously specified oxygen-free electronic copper (Section 3.1.1). The top and bottom heads are 31.7-mm thick and are made of the same material. The extra thickness allows for the machining of a rabbet for fitting the head to the shell and provides backing for the electron-beam weldment. All seams within the container shell and head assemblies are closed with electron-beam welds. All the residual voids around the basket and fuel bundles within the container are filled with a vibrationally compacted particulate such as glass bead or fractionated silica sand. The particulate is less than 1 mm in diameter.

The nominal maximum temperature that the outer surface of the disposal container is allowed to reach at any time within the vault environment is set at 100°C (Johnson et al. 1994b). For this report, however, a temperature limit of 90°C is used to allow for variability in fuel burnup. The averaged properties of the container assumed for heat transfer analysis are shown in Table 2 and the container heat output is shown in Figure 6 and Table 3. Details of the copper-shell, packed-particulate disposal container design and an alternate steel-shell-supported copper container are given in Section 3.1.

2.1.4 Radiation Protection Requirements

The requirements for radiation protection are based on AECB C-122 (AECB 1991), a proposal to reduce current radiation exposure limits. Systems which, during normal operation and minor maintenance, exceed a whole body dose equivalent to 20 mSv/a for atomic radiation workers, or 1 mSv/a for the public, are unacceptable. Radiation protection systems are to be designed for much lower exposure levels in order to deal with process upset or accident conditions. For example, larger doses tend to occur from non-routine operations (i.e., major maintenance, upgrades, accidents and decommissioning). In designing a facility, the objective is to keep the doses to as low as reasonably achievable, social and economic factors taken into account. The assumed limit in this study for a routine dose to an individual worker during normal operations is 2 mSv/a. This corresponds to an average dose rate of 1 μSv/h, based on a nominal time estimate for individual worker exposure of 2000 h/a. This radiation protection requirement is less than the 2.5 μSv/h value specified in the reference borehole emplacement case study (Baumgartner et al. 1993).

2.1.5 Vault Sealing Materials And Components

Two groups of materials have been identified as having the necessary characteristics to meet the requirements for vault sealing for the in-room emplacement design; clay-based materials, and high-performance cements and concretes (Baumgartner et al. 1995).

The use of high-performance cements and concretes has been extended in this study to the interior of the waste disposal rooms, as compared with previous studies (e.g., Simmons and Baumgartner 1994) where their use was limited to the sealing bulkheads at the disposal-room entrances. Low-heat, high-performance concrete is used for the construction of a uniform platform on the floor of the room for supporting rails and equipment and for placing and aligning precompacted backfill and buffer blocks, as well as the construction of bulkheads at the disposal-room entrances, in access tunnels and in shafts. Cement-based grouts may be used to control groundwater movement into the excavation and around seals.

In addition to these groups of sealing materials, a thin layer (i.e., 50 mm) of a dry granular bentonite and rounded sand mixture is placed in the gap between the buffer and the container, to provide for conductive heat transfer and to maintain the density of the clay-based sealing system. Rounded silica sand and granular bentonite, screened to specific sizes (i.e., fractionated) and dried, have good flow properties to fill the gap. This mixture replaces the previously specified glass bead or silica sand without any bentonite. Bentonite has been introduced to reduce the diffusion-controlled mass transport rate within the annulus around a defected disposal container.

Three clay-based sealing materials are specified for the in-room emplacement design: the buffer material, the dense backfill material and the light backfill material. The specifications for the clay-based materials are presented in Section 3.2.3.4. Note that the composition of the light backfill material is changed from sodium-bentonite clay and silica sand, assumed in the preliminary study (Baumgartner et al. 1995), to sodium-bentonite clay and crushed granite, because crushed granite strongly sorbs technetium. Sealing materials specified for the conceptual design are based on the results of the vault-sealing research program presented by Johnson et al. (1994a) and from studies in progress (Johnson et al. 1996).

Concrete bulkheads are required in the entrance to each disposal room immediately after waste emplacement, and later at strategic locations in the access tunnels and shafts as they are backfilled during the decommissioning stage. One purpose of the bulkheads is to provide a means of closing the rooms to protect the integrity of the sealing materials. Without the bulkhead as an extrusion restraint, any volumetric expansion of the bentonite clay in the buffer would reduce the dry density of the clay and could reduce its effectiveness as a sealing material. The bulkhead also provides an opportunity for applying a nuclear-materials safeguards seal that would allow detection of human intrusion into the filled room.

2.1.6 Ambient In Situ Stress State

A volume of sparsely fractured granite was selected as the host medium for the waste emplacement area of the disposal vault. The ambient principal in situ stresses assumed for the in-room emplacement vault within the Canadian Shield are based on measurements from the Underground Research Laboratory (Martin 1990, Read 1994), the Medika pluton (J.B. Martino, unpublished memorandum, 1993)[2] and from CANMET investigations (Herget and Arjang 1991). The in situ stresses used for the design analyses are shown in Figure 7 and are based on the following equations:

$$\sigma_3 = \sigma_v = 0.0260 \text{ MPa/m (depth)}; \tag{1}$$

$$\sigma_2 = 0.1112 \text{ MPa/m} + 9.9 \text{ MPa, from 0 to 300 m}; \tag{2a}$$

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[2] J.B. Martino. 1993. Medika USBM overcore testing. Unpublished AECL Internal Memorandum, GSEB-93-402.

$$\sigma_2 = 0.00866 \text{ MPa/m} + 40.7 \text{ MPa, from 300 to 1660 m;} \quad (2b)$$

$$\sigma_2 = 0.0293 \text{ MPa/m} + 6.4 \text{ MPa, greater than 1660 m;} \quad (2c)$$

$$\sigma_1 = 0.1345 \text{ MPa/m} + 18.5 \text{ MPa, from 0 to 300 m;} \quad (3a)$$

$$\sigma_1 = 0.00866 \text{ MPa/m} + 56.3 \text{ MPa, from 300 to 1400 m; and} \quad (3b)$$

$$\sigma_1 = 0.0403 \text{ MPa/m} + 12.1 \text{ MPa, greater than 1400 m.} \quad (3c)$$

where σ_v = vertical stress; and
$\sigma_1, \sigma_2, \sigma_3$ = major, intermediate and minor principal stresses respectively.

The stress levels at specific depths are given in Table 4. The detailed rationale for selecting these functions to represent the in situ stress state is presented in Appendix B of Baumgartner et al. (1995).

2.1.7 Rock Mass Material Properties and Design Limits

The rock mass material properties and the derived strength limits used in the design analyses are largely, but not exclusively, based on up-to-date measurements taken in the Underground Research Laboratory.

2.1.7.1 Rock Mass Material Properties

The sparsely fractured rock mass is assumed to be linearly elastic, isotropic and homogeneous. The elastic constants for the rock mass (Table 5) are based on measurements for Lac du Bonnet granite from the Underground Research Laboratory. The thermal properties of the rock mass, based largely on generic properties of granite (Simmons and Baumgartner 1994), are also shown in Table 5. Note that the Young's (elastic) modulus is increased from 35 GPa, assumed for the reference borehole emplacement case (Baumgartner et al. 1993), to 60 GPa, which is in the range of values estimated in the sparsely fractured rock of the Underground Research Laboratory (Read and Martin 1992).

2.1.7.2 Rock Mass Thermal Properties

The assumed thermal properties of the rock mass are given in Table 5 and are largely based on generic properties of granite (Simmons and Baumgartner 1994). Only the coefficient of thermal expansion is increased from 8×10^{-6}/°C (assumed in the reference in-floor borehole emplacement case study) to 10×10^{-6}/°C to reflect the value observed for the Lac du Bonnet granite at the Underground Research Laboratory. The geothermal gradient is assumed to be 0.012°C/m of depth, with the Earth's average surface temperature being 5°C (after Drury and Lewis 1983, Jessop and Lewis 1978). For example, at a depth of 750 m, the ambient temperature is 14°C.

2.1.7.3 Rock Mass Strength Design Limits

The rock mass strength data are taken from the sparsely fractured granitic Lac du Bonnet batholith. For this granite under uniaxial conditions, the stress for the onset of stable crack initiation (σ_{ci}) is about 70 to 75 MPa. In comparison, the stress for the onset of unconfined unstable crack growth (σ_{usc}) is about 150 MPa, and the peak unconfined compressive strength (σ_f) is about 210 MPa (i.e., the conventional value from laboratory testing). The assessment of mechanical and thermal-mechanical stability is made by calculating the factor of safety around an excavation perimeter for a given set of strength parameters. The factor of safety is defined as the ratio of the rock strength to the rock stress under triaxial conditions. At the tunnel perimeter, the design limit for strength is defined as

$$\sigma_t < \sigma_\theta < \sigma_c \tag{4}$$

where σ_t = tensile strength of the rock mass, indicated by the strength criterion;
σ_θ = tangential stress at the disposal-room perimeter; and
σ_c = compressive strength of the rock mass under unconfined conditions.

Note that the failure criteria reflect an intact rock tensile strength of 6 MPa, which is below the average 10.4 MPa value observed for the wet Lac du Bonnet granite at the Underground Research Laboratory (Martin 1993).

The Hoek and Brown (1980) empirical failure criterion model is used, defined as follows:

$$\sigma_{1f} = \sigma_{3f} + (m\sigma_c\sigma_{3f} + s\sigma_c^2)^{1/2} \tag{5}$$

where σ_{1f} = major principal stress at failure,
σ_{3f} = minor principal stress at failure,
σ_c = uniaxial compressive strength, and
m, s = empirical strength parameters.

Two peak strengths, with associated empirical strength parameters, are used with this failure model to calculate the factors of safety in sparsely fractured rock, as follows:

1. The peak strength design limit of the rock under excavation mechanical (EX) load conditions is σ_{EX} = 100 MPa, m = 16.6, s = 1;

2. The peak strength design limit of the rock under full thermal-mechanical (TM) load conditions is σ_{TM} = 150 MPa, m = 25, s = 1, if and only if, the peak strength under excavation load (see item 1. above) is not exceeded;

The rationale for selecting two design limits is discussed further in Section 3.2.3.3.

A criterion is also set for the structural performance of the geosphere near the ground surface. The uplift of the geosphere immediately surrounding the vault, caused by thermal expansion from heat from the disposed waste, may open or extend near-surface, subvertical fractures and, thus, enhance groundwater flow. This near-surface extension zone (also called the perturbed fracture or perturbed fissure zone) is defined as the volume of rock overlying the disposal vault that could experience uplift, loss of horizontal confining stresses (i.e., horizontal stress equals zero for a "no-tension analysis (Zienkiewicz et al. 1968)) and potential opening or extension of subvertical fractures. The maximum depth of the near-surface extension zone, measured from ground surface, is set at 100 m, as in previous studies (Baumgartner et al. 1993).

2.1.8 Disposal Room And Container Spacing Considerations

Previous AECL designs specified vault-level extraction ratios (ER) ranging between 0.25 to 0.30 (Baumgartner et al. 1993). This ER range is maintained in this study and is defined as follows:

$$ER = W/(W+P) \tag{6}$$

where ER = extraction ratio,
W = width of the disposal rooms (m), and
P = width of the pillars between disposal rooms (m).

2.2 SCOPING ANALYSES

A series of scoping calculations was performed (Baumgartner et al. 1995) to derive the main design features of a disposal facility. The key aspects of the analyses were to develop stable disposal-room shapes and waste emplacement arrangements that would meet the thermal and mechanical specifications for a disposal-vault depth range of 500 to 1000 m in a high in situ stress regime. An elliptical cross-sectional shape was selected for the disposal room because this shape can provide the lowest stress concentrations at the excavation perimeter (see Section 3.2 for details). The minimum aspect ratio of the ellipse (major axis/minor axis) that met the excavation and thermal stability specifications for the entire depth range was determined to be about 2.3.

Vertical emplacement of the disposal containers within a disposal room was initially considered in the scoping analyses of the in-room emplacement method study (Baumgartner et al. 1995). However, to accommodate the equipment, to provide for radiation protection and to maintain stability of the rock mass, a minimum room height of about 8 m was required; this resulted in a room width exceeding 18 m throughout the design depth range if the rooms are oriented perpendicular to the direction of the major principal stress. This room size would require significant volumes of rock to be excavated and backfill to be placed. In addition, thermal analyses showed poor heat transfer because of the presence of large volumes of backfill and sealing materials that have lower thermal conductivities than rock. Because of these findings, the study was refocused to reducing the size of the rooms, and this led to the selection of a horizontal emplacement method for the disposal containers (Figure 2).

A scheme for placing the sealing and backfill materials and for emplacing the waste was derived. It is practicable and provides the radiological protection requirements for workers. This scheme included the use of precompacted buffer and dense backfill blocks assembled on a low-heat, high-performance concrete floor that provided two central chambers for horizontal placement of the disposal containers. The gap between the assemblage of blocks and the roof of the room is filled with pneumatically placed light backfill.

A rail haulage system was selected to provide for aligning the container cask to each central chamber in the assemblage of blocks for the disposal-container transfer operation. Buffer plugs to seal the emplacement chamber and to provide radiation sealing are also transferred through the container cask.

A minimum disposal-room height of 3 m was selected to provide the minimum thickness of buffer (i.e., 0.5 m) and sufficient operational space to place the other sealing materials with the materials handling equipment. The minimum room width was set at 7.3 m, which provides for an aspect ratio of about 2.4. Currently, excavation by the drill-and-blast method is the only method that can achieve the elliptical shape for a disposal room. An allowance of 150 mm is provided at the perimeter for the blast-hole drilling "look-out angle", which results in an ellipse at the end of the blast round with dimensions 3.3-m high and 7.6-m wide. The corresponding aspect ratio is 2.3. The minimum dimension perimeter is termed in construction practice as the "A-line" (i.e., 3.0 m x 7.3 m) and the maximum excavation perimeter is termed the "B-line" (i.e., 3.3 m x 7.6 m).

Radiation shielding calculations were performed (G.R. Penner, unpublished memorandum, 1995)[3] to establish the sealing material thickness required between the emplacement disposal container and the workers to provide protection. A thickness of about 1.5-m of buffer, in the form of three 0.5-m-thick plugs, was needed to provide the 1 μSv/h protection(see Section 3.2.3.5). When the 1.5-m thickness of buffer is added to the 1.2-m length of the chamber to house the 1.189-m long disposal container, the emplacement unit length of 2.7 m for the assemblage of blocks is finally established. Thus, the centre-to-centre spacing of containers along the length of the room is 2.7 m.

Thermal and thermal-mechanical finite-element analyses were subsequently performed to confirm the scoping calculations for a stable disposal-room shape and for the distribution of the heat-generating waste (Wai and Tsai 1995, Tsai 1995). Details of these numerical near- and far-field analyses are provided in Section 3.2.4.2.

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[3] G.R. Penner. 1995 January 20. In-room emplacement radiation shielding calculations. Unpublished AECL Internal Memorandum, GRP-95-011.

2.3 DESIGN INTEGRATION

On the basis of study specifications and scoping analyses that bracketed vault layout options for a range of disposal vault depths (Baumgartner et al. 1995), together with the review of the requirements for the additional studies (Ophori et al. 1995, 1996, Stevenson et al. 1995, 1996, Wikjord et al. 1996), refinements are made to the specifications of the in-room emplacement method for the specific disposal vault arrangements, as follows:

- The depth of one disposal vault is at 750 m in a low hydraulically conductive, sparsely fractured rock mass within the Lac du Bonnet batholith of the Whiteshell Research Area to meet the objective of the favourable vault site study.

- The depth of the second disposal vault is at 500 m in a higher conductivity, moderately fractured rock mass of the Lac du Bonnet batholith to meet the permeable geosphere design objective.

- The disposal vault at a depth of 500 m has a 20-m thick, low-angle (18^o) fault transecting the waste emplacement area of the vault. The total horizontal distance between the two vault sections separated by the transecting fault is 375 m (Stanchell et al. 1996).

- The overall disposal vault area, excluding the shaft service areas, of 4 km^2 is retained and a spacing between the disposal containers along the length of the disposal room of 2.7 m is calculated to meet both thermal and radiation protection design limits, resulting in the quantity of used fuel being about 5.8 million bundles for the vault at 750-m depth; and about 4.3 million bundles for the vault at 500-m depth with a fault transecting the waste emplacement area.

- The copper-shell, particulate-packed disposal container increased in design height from 1184 mm to 1189 mm, and the copper alloy is changed from oxygen-free, electronic copper to deoxidized, low-phosphorous copper. A steel-shell-supported copper disposal container, with dimensions of 994-mm outside diameter and 1349-mm length, is also provided as an alternative design.

- the composition of the light backfill pneumatically placed over the assemblage of precompacted blocks is changed where the silica sand component is replaced by crushed granite.

- the particulate in the eccentric annulus between the disposal container and the buffer is changed from either glass beads or silica sand to a mixture of granulated bentonite and rounded silica sand; and

- the disposal rooms in the two specific vault designs, one at a depth of 750 m in the favourable site and the other at a depth of 500 m in the moderately fractured rock, are oriented parallel to the major principal stress direction;

This report focuses on a generic disposal vault design using the in-room emplacement method for waste disposal that is then applied to two specific vault designs, as defined above.

The generic vault design considers a number of factors including: the layout of the disposal rooms; shaft and tunnel accesses; traffic, material and ventilation flows; and separation of radiological and nonradiological operations. In the 750-m deep vault design, adaptations of the generic design are made. All aspects of the used-fuel disposal facility implementation are defined and established as a project and scheduled, beginning with the siting stage and stepping through the sequence of the construction, operation, decommissioning and closure stages. Optional extended monitoring stages are also briefly defined. In the 500-m deep vault design, adaptations of the generic vault design are made in the details of the vault layout and disposal-room design relative to the conditions of the specific site. All other aspects of the used-fuel disposal facility design including project implementation are addressed throughout the body of the report and are not repeated

3. DESIGN DESCRIPTION OF A DISPOSAL FACILITY WITH IN-ROOM EMPLACEMENT

This chapter steps through one disposal facility design. It begins with a detailed description of a copper-shell, packed-particulate disposal container, an engineered barrier containing the used fuel to be disposed. This is followed by the generic design and description of the disposal room within a vault at a depth range of 500 to 1000 m. Then the design of the disposal vault, its layout and the factors that influenced its design are described. Finally, the modifications to the used-fuel packaging plant and a brief description of the common surface facilities needed by the disposal vault and the packaging plant are provided.

3.1 COPPER-SHELL, PACKED-PARTICULATE CONTAINER

The disposal container for the in-room emplacement design is based on an outer corrosion barrier of deoxidized, low-phosphorus copper with a corrosion performance expectancy of no less than a million years (Johnson et al. 1996).

3.1.1 Container Design

The container design is based upon the structural-support concept developed for the titanium-shell design used as the reference for the engineering assessment for a used-fuel disposal centre in the Environmental Impact Statement (AECL 1994a, Johnson et al. 1994a). In this concept, support for the outer corrosion-resistant shell against external pressure is provided by a particulate material (i.e., glass beads or fractionated silica sand) that is vibrationally compacted into the container after the used fuel is loaded.

The material selected for construction of the shell is a deoxidized, low phosphorous (DLP) copper proposed for the construction of disposal containers in the Swedish nuclear fuel waste management program, based on that program's long-term mechanical-behaviour investigations of various copper grades (Henderson et al. 1992, R. Dutton, unpublished memorandum, 1995).[4] This better data base is the reason for the change from the original material specification, presented in Baumgartner (1995), which was oxygen-free, electronic (OFE)[5] copper. An assessment (F. King, unpublished memorandum 1995)[6] of the probable corrosion performance of DLP copper, relative to that of OFE copper, concluded that the copper-container failure model was equally applicable to OFE and DLP copper.

The container design was developed by Ontario Hydro Technologies (B. Teper, unpublished memorandum, 1992).[7] For in-room emplacement, the container is 1189 mm in height and 860 mm in diameter, with a capacity of 72 CANDU used-fuel bundles (Figure 5).[8] To facilitate horizontal installation into the precompacted buffer, the container-lifting ring attached to the top head is designed to maintain a level profile along the container length.

3.1.2 Fabrication and Inspection

For this study, it is assumed that the container shell and bottom head are fabricated and preassembled off-site, as are the top head and lifting ring.

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4 R. Dutton. 1995 May 18. Creep fracture in various grades of copper. Unpublished AECL Internal Memorandum, OCE:AFC-95-006.

5 ASTM material designation number according to the Society's Unified Numbering System (UNS): C10100. The chemical requirements for OFE copper (according to ASTM Specification B 170) are that the minimum copper content shall be 99.99% and that the maximum oxygen content shall be 0.001%.

6 F. King. 1995 June 12. Corrosion consequences of selection of Finnish DLP copper as the reference container material for alternative post-closure assessment. Unpublished AECL Internal Memorandum, FWTB-95-201.

7 B. Teper. 1992 December 10. Proposed design of copper containers for IES study. Unpublished Ontario Hydro Technologies Memorandum, 825.123(T).

8 These dimensions reflect the effects of certain design changes (B. Teper, unpublished memorandum, 1994, J.L. Crosthwaite, unpublished memorandum 1994)[9,10] from the original container concept presented in Teper (1992), only some of which are included in Baumgartner et al. (1995).

9 B. Teper. 1994 May 3. Proposed redesign of top lid weld of the PASCOS container. Unpublished Ontario Hydro Technologies Memorandum, 825.123.

10 J.L. Crosthwaite. 1994 October 7. In-room emplacement study - new bottom head design for Cu-shell reference container and new particulate-porosity value. Unpublished AECL Internal Memorandum, FWTB-94-422.

For ease of manufacture, casting of the shell is attractive, and copper-based materials are among the easiest to cast, with the notable exception of unalloyed copper. Attempts to cast pure copper usually result in a product that contains coarse, columnar grain structures, rough surfaces and a tendency to form shrinkage cavities (Peters and Kundig 1994). Overcoming these problems requires strict adherence to proper foundry practices, but there is still risk of producing an inferior product. As has been shown in the Swedish program, large grain size can result in severely reduced creep ductility in copper, and early failure of the metal under sustained stress (Henderson et al. 1992). Thus casting of pure coppers is usually reserved for products that require either the highest thermal or electrical conductivities, neither of which is a requirement for the construction or structural durability of used-fuel disposal containers.

U.S. researchers have been investigating centrifugal casting of composite shells consisting of an outer cylinder of pure copper and an inner shell of nickel-aluminum-bronze alloy (Peters et al. 1993). Casting of the inner Ni-Al-bronze component is conducted prior to full solidification of the outer cast-copper component so that fusion of the two at their interface occurs, giving a single, but two-layered shell. Most of the container's mechanical strength is provided by the Ni-Al-bronze layer so that the columnar grain structure of the copper is of little consequence to structural durability. This work is an advanced technique that may warrant further consideration, particularly for container designs employing an internal shell for structural durability (Section 3.1.3). However, production cost, long-term structural performance, corrosion durability of large-grained copper, etc. would require investigation prior to identifying such a manufacturing technique for containers at this time. For these reasons, casting of the container shell has been eliminated for this study.

The cylindrical shell, for a single-shell copper container, can be constructed in two ways:

- by rolling two flat plates into semicircular configurations and joining them along their length by electron-beam welding, and

- by extruding a seamless shell from a heated copper ingot.

Both fabrication methods are under investigation in the Swedish program, in which it is proposed that the container be constructed with a 880-mm diameter, 50-mm-thick copper shell with overall length 4.85 m (Figure 8) (SKB 1992a,b, Johnson et al. 1994a). On the basis of the current experience in the Swedish program, we have selected rolling/welding as the method of manufacture of the copper-shell disposal container for the in-room emplacement method.

To weld the two half-shell components together, a backing strip, made of the same grade of copper as the shell material, is attached first, by tack welding, along the length of the mating junctions of the half-shell components, on the inside container surface. Electron-beam welding is then performed along each shell junction, with the electron beam penetrating from the outer surface of the container, through the shell junction and into the backing strip. After both longitudinal welds are completed, the backing strips are machined off, likely on a large lathe. If necessary, the outer surface of the welds is machined or ground smooth (e.g., to remove any weld spatter).

After the cylindrical shell is assembled, the bottom head is attached by electron-beam welding. In the current design, both the bottom and top heads have been designed to provide self-backing to the electron beam to eliminate the need for a separate backing strip. Again, the weld surface is machined or ground smooth, if required.

For this stage of container construction, all welds are inspected by an appropriate volumetric technique such as radiography and, if necessary, a surface technique such as dye-penetrant examination. If further inspection is necessary, leak detection examination employing procedures proposed by Zane and Boag (1995) could be employed to ensure the integrity of the container-shell/bottom head configuration.

The lifting ring is electron-beam welded to the top head. A surface-inspection technique would likely be adequate for this weld, since it does not form part of the containment boundary. To ensure a good match and intimate contact between the top head and shell, which is required to achieve a high quality final-closure weld, each ring/top-head assembly is individually matched to a specific prefabricated shell/bottom-head assembly and the mating edges at the final-closure-weld position trimmed if necessary. Matched heads and shells are then stamped or otherwise identified with matching serial numbers and shipped as a combination to the disposal facility.

At the disposal facility, 72 fuel bundles are loaded into a basket (also prefabricated off-site), and the loaded basket installed into the shell/bottom-head assembly (Figure 5). In the initial design developed by Ontario Hydro Technologies, the basket-construction material was carbon steel, but stainless steel could be used. For this study, stainless steel that has the same dimensions as the original carbon-steel design was selected (Johnson et al. 1996). After the residual void in the container is filled with vibrationally compacted particulate (e.g., glass beads or fractionated silica sand), the matching top-head/lifting ring assembly is attached by electron-beam welding. The final-closure weld is inspected ultrasonically (Moles 1986, Maak 1988), and the container then is transferred to either surge storage or to the vault for disposal. The masses and volumes of the disposal container and its components are presented in Table 6.

3.1.3 Structural Analysis

Depending on variables such as impurity content, plate-fabrication processes and heat treatment, the mechanical properties of copper can vary widely. For purposes of the current container structural analysis (Teper 1995), the yield strength of the material was specified as 86 MPa, which is near the lower end of the range common for copper (48 to 350 MPa).[11] Analyses were conducted for both linear-elastic and elastic-plastic container-shell material properties using glass

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[11] A more precise specification of required material properties would form part of a future container-optimization stage.

beads as the particulate. The properties of the particulate were taken from earlier work by Teper (1987).

The design load for the structural analysis of the copper-shell, packed-particulate disposal container is based on an external pressure of 13 MPa.[12] However, there is a potential that loads associated with a future glaciation event could increase the external hydraulic pressure to as high as 40 MPa if the water table is near the top of a 3000-m-thick ice sheet and the disposal vault is at a depth of 1000 m in a pluton. For this case, an alternate container, the steel-shell-supported copper container, is described later in this section. For the assumptions that the container is perfectly packed (e.g., no residual void or further settlement of the particulate), local strains of up to 2.1% can occur, at an external pressure of 13 MPa. Copper is a ductile material, typically able to deform plastically to strains exceeding 30% at its ultimate tensile strength (G.R. Kasprick, unpublished data),[13] and such strains are, therefore, well within the limits of acceptability for this container design.

Teper (1995) also conducted an analysis of an in-room emplacement container in which an internal void develops through further settlement of the particulate. The potential for such settlement is based upon the observation of this phenomenon in the testing of a full-scale, titanium-shell packed-particulate container prototype of borehole-emplacement design (Teper 1988, Johnson et al. 1994a).

For a volumetric settlement of the same magnitude that was observed in the prototype titanium-shell packed-particulate container for borehole emplacement, the maximum gap between container shell and the particulate would be 25.1 mm. The container shell, under 13 MPa hydrostatic pressure, would initially deform to produce a through-wall strain of 8.7% and this could, under the effects of material creep, increase to as high as 24%.

Creep tests on DLP copper conducted for the Swedish program (Lindblom et al. 1995) have shown that strains to failure of as high as 80% can be achieved under some test conditions. However, some tests incurred strains to failure of as low as 9% and therefore, although DLP copper generally exhibits good creep-rupture ductility, it is currently uncertain whether 24% would fall within an acceptance criteria for allowable creep strain. Such acceptance criteria have not yet been established.

Alternative approaches are underway to further develop a container design that can meet the essential short- and long-term structural requirements during disposal:

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[12] Consisting of 10 MPa hydrostatic pressure caused by groundwater at a maximum vault depth of 1000 m, plus a maximum of 3 MPa buffer-swelling pressure.

[13] G.R. Kasprick. 1995. Unpublished data for OFE, OFHC and DLP copper tested at AECL Whiteshell Laboratories.

- use of alternative particulates with higher strength and less tendency for additional settlement, and

- use of a steel inner shell for structural support.

Investigations into alternative particulate materials are focused on the use of fractionated silica sands. Previous studies (Teper 1987) have shown that such sands have the potential for very high compressive strength (>50 MPa) which, for a container designed with a lifetime of a million years or more, would be able to withstand the combined effects of glacial loading (maximum hydrostatic pressure of 40 MPa) and of the buffer-swelling pressure (3 MPa). The current investigations include studies of ease of compaction, short-term mechanical properties, tendencies for further settlement and creep-deformation behaviour. The creep-deformation studies will generate data that will be used to predict the creep-rupture life of the copper shell according to a methodology proposed and described by Dutton (1995b).

Investigations into container designs employing a steel inner shell for structural support are also discussed by Garroni et al.(1996). From these studies, a steel-shell-supported copper disposal container has been developed (Figure 9) to withstand an external pressure of 50 MPa without collapse, including the pore-water pressures associated with glaciation if they are assessed to be transmitted to the emplaced container. Note that in this design, the use of a 65-mm-thick inner steel shell would require some increase in the overall container dimensions from those of the packed-particulate design. Similar to studies being conducted on designs employing a compacted particulate for structural support, studies are underway to determine the creep-rupture life of the copper shell in steel-shell-supported designs.

3.1.4 Summary

A copper-shell container design for in-room emplacement has been developed. The design uses compacted particulate (e.g., glass beads or fractionated silica sand) to provide internal support to the shell, under pressures that could occur at a vault depth of up to 1000 m. If well compacted, and if no further settlement occurs after final closure-welding of the container, this design would provide adequate support to the shell. Because further settlement of some types of particulate (in particular, single-size glass beads) has been observed under some conditions, the use of alternative particulate materials, such as graded silica sand, is under study. In addition, alternative container designs employing a steel inner shell to support the outer corrosion-resistant shell have been developed.

3.2 DISPOSAL ROOM

This section presents an overview of the disposal-room design process, describes the main room components, discusses the source and rationale for selected design specifications, summarizes the results of design analyses and presents the operations associated with waste disposal.

3.2.1 Overview of Disposal-Room Design

The process for the derivation of a design for the disposal room is briefly described, including the analytical and numerical methods used in assessing its stability.

3.2.1.1 Evolution of the Disposal-Room Design

The disposal-room design used in the in-floor borehole emplacement method was based on an opening with a rectangular lower cross-section, an arched roof and closely spaced waste emplacement boreholes drilled into the floor (Simmons and Baumgartner 1994). Thermal-mechanical analyses of this complex room shape showed that it meets design requirements at a depth of 500 m in the average stress conditions of the Canadian Shield. However, this arrangement is not well suited in average in situ stress conditions anticipated below a depth of 500 m or in above average stress conditions. To support the favourable vault site and the permeable geosphere design objectives, other, more optimal, room shapes are investigated. The preliminary design studies for in-room emplacement (Baumgartner et al. 1995) introduced in Section 2.2, considers both oval- and elliptical-shaped disposal rooms. These two excavation shapes result in lower stress concentrations around the room perimeters than do other excavation shapes, when the major axes of the room elliptical cross section is oriented horizontally and parallel to the above-average horizontal in situ stresses assumed in this design (Section 2.1.6). The stress concentrations at the perimeter of the excavation are lowest for an ellipse when the room width-to-height (i.e., W/H) ratio, or room aspect ratio, is equal to the ratio of the major and minor principal stresses (i.e., σ_1/σ_3) acting in the plane of the elliptical section. For the condition $\sigma_1/\sigma_3 = 60/26$, an aspect ratio of 2.3 represents the "ideal" room shape to minimize the stress concentrations acting on the room perimeter. However, an opening with an oval shape is usually preferred for operational reasons as it provides more useable space at the outer edges of the room. Using a two-dimensional closed-form solution for single openings by Greenspan (1944), Baumgartner et al. (1995) conducted scoping analyses of oval-shaped excavations. Their results show that, depending on the radius of curvature of the outer quadrants ("corners") of the ovals, significant stress concentrations are developed in these regions when compared with ellipses having equivalent aspect ratios (Figure 10). Consequently, remaining analyses were limited to elliptical-shaped excavations. For ellipses with their axes aligned with the principal in-plane stresses, the critical stress points are in the centres of the floor (i.e., the invert), the roof (i.e., the crown) and the walls (i.e., the springlines), which represent the perimeter intersections with the major and minor axes of the ellipse.

The goal of the design analyses (Baumgartner et al. 1995) was to establish a disposal-room shape suitable for the upper range of in situ and thermal stress conditions anticipated over a range of depths (i.e., 500 to 1000 m), thus enhancing the robustness of the design and reducing the need for design iterations upon selection of the disposal-vault depth. Scoping analyses were performed for disposal-room aspect ratios ranging between 1.5 and 3.25, using a range of stress conditions representative of those between 500 and 1000-m depth. In each analysis, the maximum and minimum tangential stresses at the intersections of the elliptical axes and the room perimeter were calculated and compared with mechanical and thermal-mechanical design limits. The initial results of these analyses (Baumgartner et al. 1995) were used to produce a design envelope relating room aspect ratio to depth and stability (Figure 11). Based on this envelope, a disposal room with an aspect ratio ranging between 2.2 and 2.6 was found suitable for the desired depth range. These analyses were based on the premise that the room axis is perpendicular to the major principal stress direction; hence the $\sigma_{max}/\sigma_{min}$ ratio at each depth represents an upper bound.

After detecting an input data error for in situ stresses in the initial scoping analyses in Baumgartner et al. (1995), the correct in situ stress data was used in the updated calculations, and the recalculated results for the disposal-room with its longitudinal axis perpendicular to the major principal stress direction are shown in Figure 12. Also the results for the disposal room, parallel to the major principal stress direction, are shown in Figure 13. In Figure 12, the estimated roof compressive stresses under thermal conditions show that the stability of the room appears to be somewhat marginal for the depth range of 500 to 750 m for the geometric design range. If this is confirmed by detailed thermal-mechanical analyses using the finite-element method, then the situation can be avoided by rotating the longitudinal axis of the disposal room by 90° so that the intermediate principal stress (i.e., σ_2) is acting on the elliptical section (Figure 13).

The preliminary design envelope for this restricted orientation of the disposal room appears reasonable for the entire depth range (Figure 13). The simplified scoping calculations for a single opening are confirmed, by the detailed thermal-mechanical modelling of multiple openings (Wai and Tsai 1995, Tsai 1995) described in Section 3.2.4.2, to provide reasonable trends for disposal-room stability, although the scoping calculations are somewhat inaccurate. As discussed in Section 3.2.4.2, only a very minor part of the design envelope infringes upon the wall stability envelope, based on the detailed finite-element modelling, for the case of the disposal room at a depth of 1000 m with the major principal stress oriented parallel to the longitudinal axis of the room (i.e., factor of safety of 0.99). All other cases of disposal-room depth and orientation are stable around the room perimeter.

The minimum centreline room height required for this emplacement method is 3 m. An acceptable room aspect ratio ranges between 2.3 and 2.43, taking into account the drill-and-blast borehole lookout angle (Figure 13). Consequently, a minimum centreline room width of 7.3 m is derived (Baumgartner et al. 1995).

3.2.1.2 Analytical Approach Used in Disposal-Room Design

The definition of the suitable disposal-room size is retained for the design analyses presented in this report. The analyses consider disposal rooms oriented perpendicular and parallel to the major principal stress direction at vault depths ranging from 500 to 1000 m. Findings from the Underground Research Laboratory regarding in situ stresses and rock strength are considered in the design analyses (Section 2.1.6).

The preliminary design process (Baumgartner et al. 1995) consisted of a number of steps for deriving a mechanically and thermal-mechanically stable disposal-room and pillar geometry. This process involved the specification of material properties and design criteria, the definition and analysis of room shapes under ambient temperature and stress conditions, the distribution of heat-generating nuclear fuel waste (i.e., used fuel) within each room of a disposal vault, and analysis of the same rooms under backfilled, elevated temperature and stress conditions. Excess pore pressures caused by thermal expansion effects were assumed to be minimal since the rate of pore pressure dissipation is assumed to match the rate of temperature and expansion effects. This assumption is the subject of ongoing investigation.

The excavation design process uses physical material properties and stresses derived or specified from the findings of the Nuclear Fuel Waste Management Program. This preliminary phase of the design process used analytical design methods. The confirmatory design phase described in this report is based primarily on the finite-element design method. The preliminary design phase is discussed in detail in Baumgartner et al. (1995), and the results are summarized in Section 3.2.4.1.

3.2.2 Description of Room Components

The disposal room is designed on the basis of the multibarrier approach (Simmons and Baumgartner 1994), with the waste form, the container, the buffer material, the backfill and the rock mass each representing one of the engineered or natural barriers. The disposal container and used-fuel packaging plant are described in Sections 3.1 and 3.4 respectively. The disposal room comprises three main components: the opening excavated within the rock mass, the permanent furnishings required to conduct emplacement operations, and the sealing materials. The arrangement of the various room components is shown schematically in Figure 2, and their relative positions and dimensions are illustrated in Figure 14 (Wai and Tsai 1995). The main components are described in the sections that follow.

3.2.2.1 Disposal-Room Excavation

As discussed in Section 3.2.1.1, at depths between 500 and 1000 m the preferred room shape is elliptical with an aspect ratio between 2.2 and 2.6. A nominal aspect ratio of 2.3 was chosen as a robust design value. The preferred disposal-room orientation is such that the room axis is parallel to the maximum principal stress direction, and the major axis of the elliptical cross section is parallel to the intermediate principal stress direction. In comparison with the other room shapes and orientations, this design minimizes the tangential stress concentrations around the opening.

The disposal-room size is based on the minimum space required to place a floor of low-heat high-performance concrete, the blocks of precompacted sealing materials, the disposal containers and the light backfill. On this basis, the minimum centreline room height is 3 m. The drill-and-blast method used to excavate the room will also enlarge the excavation by 150 mm in all directions at the end of a blasted section because of the necessity to drill the blastholes slightly outward (i.e., at a "look-out" angle) to maintain minimum excavation size. Therefore, the maximum room height is 3.3 m, and for an aspect ratio of 2.3, the maximum width is 7.6 m. At its smallest cross section, the room is 7.3 m wide at the centreline, yielding a maximum local aspect ratio of 2.43. Both aspect ratios are within the stable regime of the room design envelope for the 500 to 1000 m depth range (Figure 13). For the finite-element method design analyses, discussed in Section 3.2.4, a room size of 3.3 m by 7.6 m was used. The rooms are nominally 238 m in length, based on the panel layout described in Section 3.3.1.

3.2.2.2 Permanent Furnishings

Low-heat, high-performance concrete is used for the construction of a uniform platform on the floor of the room for supporting rails and equipment and for placing and aligning precompacted backfill and buffer blocks. Although the rail and other temporary furnishings are removed as the room is sealed, the concrete floor remains as a permanent part of the sealing system. The properties and specifications of high-performance concrete are discussed in Section 3.2.3.4. Referring to Figure 2, the pad is shaped to provide a flat 3.7-m-wide platform from which to conduct emplacement operations. At the centre of the disposal room, the concrete pad is about 0.45-m thick.

3.2.2.3 Sealing Components

The two main types of sealing materials in the in-room emplacement design are clay-based materials and high-performance cements and concretes. The clay-based materials include dense backfill in the lower part of the room, buffer material (buffer blocks and plugs) in the central part surrounding the waste container, and light backfill in the upper part of the room. In addition, a dry granulated bentonite and rounded sand mixture, is placed in the gap between the buffer and the container to provide for conductive heat transfer and to maintain the density of the clay-based sealing system. Low-heat, high-performance concrete is used in the construction of bulkheads at the disposal-room entrances. These bulkheads are nominally 12-m long and may be keyed into the rock mass depending on the condition of the disposal-room perimeter. The properties and specifications for these materials are presented in Section 3.2.3.4.

As shown in Figure 2, the waste containers are 860 mm in diameter and 1189 mm in length. These are emplaced in horizontal chambers, 960 mm in diameter and 2.7 m deep, centrally spaced across the room at 2.21 m, centre-to-centre. The minimum thickness of buffer surrounding these holes is 0.5 m. In the crown of the room, light backfill is used to seal the region between the buffer and the rock. In the sidewall, the region between the rock and the buffer is sealed with a combination of light and dense backfill. The region between the concrete floor and the buffer comprises dense backfill. The pitch distance (i.e., the centre-to-centre distance from one container to another along the longitudinal axis of the disposal room) used for the in-room emplacement design is 2.7 m, based on radiological and thermal considerations. The materials separating the containers along the pitch include 1.5 m of buffer and 0.05 m of dry particulate.

3.2.3 Design Criteria and Parameters Used for Numerical Analyses

The specifications and, in some cases, the reasons for the design specifications are discussed here.

3.2.3.1 Ambient In Situ Stress and Temperature Conditions

The ambient in situ stress and temperature conditions in the geosphere are provided in Sections 2.1.6 and 2.1.7 respectively. The structural analyses of the disposal rooms are performed with the long axes of the rooms oriented perpendicular and parallel to the major principal stress (σ_1) direction for the depth range of 500 to 1000 m.

3.2.3.2 Rock Mass Properties

The sparsely fractured rock mass is assumed to be linearly elastic, isotropic and homogeneous, as defined in Section 2.1 and as specified in Table 5.

3.2.3.3 Rock Strength Design Limits

Mechanical and thermal-mechanical design limits on rock strength have been defined in terms of the empirical Hoek and Brown (1980) failure criterion as given in Section 2.1.7.

The factor of safety (FS) associated with this criterion is defined as the ratio of the maximum principal stress, σ_1, at a point to the failure stress, σ_{1f}, calculated from Equation 5 using the minimum principal stress, σ_3, value at the point. The choice of appropriate parameters for this criterion was based on results from extensive laboratory testing, and from in situ testing at the Underground Research Laboratory. Both types of testing were necessary to identify differences in laboratory and field behaviour, in particular, to assess the effects of damage and loading path on the rock strength around underground openings.

A typical stress-strain curve for Lac du Bonnet granite is shown in Figure 15. In uniaxial and triaxial laboratory tests, Martin (1993) showed that microcracks initiate at 0.3 to 0.4 σ_f, where σ_f is the peak strength of the sample. This point represents the start of stable crack growth, and for the unconfined condition it occurs at about 70 to 75 MPa in Lac du Bonnet granite. The true strength of a brittle rock in compression is defined by the crack-damage stress, σ_{cd}, which in monotonically loaded samples occurs at about 0.7 to 0.8 of the peak strength, σ_f. This point represents the start of unstable crack growth (Bieniawski 1967), and it is commonly referred to as σ_{usc}, or the "long-term" strength, for monotonically loaded samples. In the unconfined case, σ_{usc} for Lac du Bonnet granite is between 150 and 160 MPa. In laboratory tests, loads above the crack-damage stress can be maintained for a short duration, but generate a through going, sliding crack that provides a mechanism for ultimate collapse of the sample. By inducing damage in a sample (e.g., through cyclic loading), Martin (1993) demonstrated that the crack-damage stress, and consequently the rock strength, was reduced largely through cohesion loss.

Excavation at the 420 Level of the Underground Research Laboratory has shown that the in situ rock strength can be variable with location around an excavation perimeter. Results from the Mine-By Experiment at the Underground Research Laboratory (Read and Martin 1992) showed that macroscale failure in regions of compressive stress concentration at the periphery of a circular test tunnel initiated at a stress level about 0.5 to 0.6 σ_f (i.e., between 110 and 125 MPa), well below the strength obtained from undamaged laboratory samples. This difference between the laboratory and in situ strengths was attributed to damage induced in situ by stress changes that occurred ahead of the advancing tunnel face (Martin 1993). Points in localized regions, ahead of the face of the test tunnel, experienced changes in the principal stress magnitudes and orientations as the tunnel advanced, causing microscale damage. This damage locally weakened, or "pre-conditioned", the rock mass. The degree of damage was greatest about one radius from the projected tunnel centreline (i.e., at the tunnel periphery), and decreased rapidly with increasing radial distance from the tunnel centreline.

Upon tunnel advance, these localized regions of damage were exposed at the tunnel perimeter. Hence the strength of the rock mass around the underground opening varied from point to point, with the strength at any given point depending on the stress history and damage accumulated there as a result of the excavation process. Points located at the tunnel wall in the maximum compressive stress concentration accumulated the most damage as the tunnel advanced, and therefore experienced the largest strength reduction. Points located further from the opening, or away from the zones of maximum compressive stress concentration, experienced less damage and thus had strengths closer to the undamaged laboratory strength.

Martin (1995) showed that excavation-induced microseismicity could be used to identify the regions where cracking initiates in situ. The stress level associated with these events in conditions where σ_3 approaches zero is referred to as the in situ crack-initiation stress, σ_{ci}. Three-dimensional numerical stress analyses, conducted to determine the crack-initiation stress associated with individual microseismic events around the Mine-By test tunnel, showed that the threshold for the initiation of in situ cracking could be approximated by

$$\sigma_1 - \sigma_3 \approx 70 \text{ to } 75 \text{ MPa} \tag{7}$$

This implies that σ_{ci} = 70 to 75 MPa in situ. However, it was only in those areas ahead of the face where the principal stresses rotated and the confining stress was low that this crack initiation resulted in damage significant enough to cause strength reduction.

The variability in strength at the perimeter of a 1.24-m-diameter vertical borehole at the 420 Level of the Underground Research Laboratory was much less than that experienced in the Mine-By test tunnel. The excavation perimeter strength in the region beyond the influence of the room (i.e., ~3-m below the floor of the room) was between 150 and 158 MPa (Read 1994), very similar to the stress level at the onset of unstable crack growth in unconfined laboratory samples (i.e., σ_{usc}). Stress analyses around the opening confirmed that the pattern of in situ crack initiation predicted using Equation 7 was significantly different from that for the Mine-By test tunnel and would not be expected to "pre-condition" the rock exposed at the borehole wall beyond about 3 m below the floor of the room. The difference in stress patterns ahead of the faces of the two excavations is largely a result of the difference in orientation of the openings relative to the maximum and minimum principal stress directions (Read et al. 1995).

Thus the in situ strength immediately around an underground opening in a highly stressed rock mass can vary within a large range. The rock strength is a function of the amount of damage accumulated during the excavation process, which is in part related to the three-dimensional stress effects associated with the advancing face. To simplify the assessment of mechanical stability of an opening, derated strength parameters can be used with the Hoek and Brown (1980) failure criterion to reduce the complex three-dimensional, stress-path dependent analysis of an advancing excavation face to a simple two-dimensional plane strain analysis. This simplified approach is considered appropriate for a conceptual-level design. In terms of thermal-mechanical behaviour, the assessment of rock mass stability under thermal loading is less complex because the thermally induced stresses are superimposed on the plane strain conditions existing around the completed excavation (i.e., there are no three-dimensional effects to consider). Stresses at the end of the disposal room are lower than in the plane strain condition because of the support provided by the rock mass of the unexcavated face. At the disposal-room intersection with the panel access tunnel, stress conditions are more severe and ground support may be required. The first disposal container is about 25 m away from the entrance of the disposal room, separated from the panel access tunnel by a 12-m long low-heat high-performance concrete sealing bulkhead, which begins about 11 m from the disposal-room entrance. After disposal of the waste, the installed bulkhead provides ground support.

A proper three-dimensional thermal-mechanical numerical analysis (e.g., using the finite-element method) would require the simulation of the excavation sequence. For each excavation step, the calculated stress at any point (i.e., in any element) would be compared with the initiation of the in situ cracking stress (Equation 7). The strength would be reduced if the calculated stress exceeded the cracking stress. The computer code needs to keep track of the stress path history as well as the value of the reduced strength for all of the elements and use this information in the next excavation step. In each excavation step and, following excavation, each heating time step, the strength and stress values need to be updated based on a strain-softening material behavioural model representative of the rock mass. Though possible in principle this is monumental in practice and unnecessary for a conceptual-level study.

Figure 16 illustrates the various envelopes associated with rock mass strength. The upper envelope (σ_c = 213 MPa, m = 31 and s = 1) represents the peak "short-term" strength, as determined from typical laboratory test results. The envelope forming the upper bound of the shaded area (σ_c =150 MPa, m = 25 and s = 1) represents the in situ rock mass strength around a tunnel under plane strain conditions where there has been no "pre-conditioning" resulting from three-dimensional face effects. This envelope also represents, σ_{usc}, the "long-term" strength from laboratory tests. The envelope forming the lower bound of the shaded area (σ_c = 100 MPa, m = 16.6 and s = 1) represents the derated strength parameters in plane strain conditions used to simulate the three-dimensional conditions associated with the onset of "pre-conditioning" ahead of the face. It should be noted, however, that this envelope underestimates the rock mass strength further back from the opening surface. Below the Hoek and Brown (1980) envelopes is the linear envelope representing the in situ crack initiation threshold (Equation 7).

The assessment of mechanical and thermal-mechanical stability is based on comparison of the calculated factor of safety around an excavation for a given set of strength parameters. At the tunnel perimeter, the design limit is defined as $\sigma_t < \sigma_\theta < \sigma_c$, where σ_t is the tensile strength indicated by the failure criterion, σ_θ is the tangential stress at the excavation perimeter, and σ_c is the compressive strength of the rock mass under unconfined conditions. Note that the failure criterion, described above, reflects an intact rock tensile strength of 6 MPa, which is below the 10.4 MPa average value observed for wet Lac du Bonnet granite at the Underground Research Laboratory (Martin 1993). The mechanical and thermal-mechanical stability analyses of an opening are conducted in sequence as follows:

1. The factor of safety around an excavation is determined under excavation (i.e., ambient temperature) conditions based on the envelope for the derated strength parameters (σ_c = 100 MPa, m = 16.6 and s=1, the excavation conditions design limit). This approach provides a conservative estimate of the factor of safety at the tunnel perimeter but is overly conservative further back from the opening surface.

    - If the excavation conditions criterion is satisfied everywhere on the excavation perimeter (i.e., FS > 1), it is assumed that no significant strength reduction has occurred. Therefore, the failure criterion representing the "long-term" strength of Lac du Bonnet granite (σ_c = 150 MPa, m = 25 and s = 1) is used for subsequent analyses.

    - If the excavation criterion is violated somewhere on the excavation perimeter (i.e., FS < 1), it is assumed that significant strength reduction has occurred, and the tunnel perimeter is likely to fail.

2. If there is no failure indicated by the excavation conditions analysis, the factor of safety around an excavation under thermal conditions is then determined based on the envelope for the "long-term" strength of the rock mass granite (σ_c = 150 MPa, m = 25 and s = 1, the thermal conditions design limit). Note that no credit is taken in the excavation thermal stability analyses for any support that may be provided by the backfill materials to the rock mass. Therefore, this thermal conditions analysis is conservative.

    - If the thermal criterion is satisfied everywhere around the excavation, it is assumed that no failure will occur under the combined excavation and thermally induced stresses.

    - If the thermal criterion is violated, it is assumed that failure will occur.

A criterion is also set for the structural performance of the geosphere near the ground surface. The uplift of the geosphere immediately surrounding the vault, caused by thermal expansion attributable to heat from the disposed waste, may open or extend near-surface, subvertical fractures and, thus, enhance groundwater flow. This near-surface extension zone (also called the perturbed fracture or perturbed fissure zone) is defined as the volume of rock overlying the disposal vault that could experience uplift, loss of horizontal confining stresses and potential opening or extension of subvertical fractures. The maximum depth of the near-surface extension zone, measured from ground surface, is set at 100 m, as in previous studies (Baumgartner et al. 1993).

3.2.3.4 Sealing Material Properties and Specifications

The in-room emplacement method focuses on the emplacement of disposal containers within the confines of an excavated room (Figure 2). The basic requirements of sealing components for the vault can be summarized as follows:

- Restrict water flow around the container to constrain and limit the rate of container corrosion and, subsequent to its failure, limit the rate of fuel dissolution and radionuclide transport;

- Swell sufficiently when water is absorbed from the surrounding rock to seal any opening between the container and the host rock. The shear strength of the swelled material will be sufficient to support the container without significant deformation; and

- Sorb and retain released radionuclides to significantly retard the rate and extent of radionuclide migration.

Two groups of materials were identified as having the necessary characteristics to meet the requirements for vault sealing for the in-room emplacement design: clay-based materials, and high-performance cements and concretes. Considerable research has been conducted in the Nuclear Fuel Waste Management Program to establish specifications for clay-based materials to be used as sealing components in the vault. Numerous studies of mineralogical, physical, chemical and mechanical properties were performed in Canada and abroad and the results were reported in the literature. The characteristics and the suitability of clay-based materials as sealing components are well documented (Yong and Warkentin 1975; Mitchell 1976; Pusch et al. 1987; Graham et al. 1989; Gray 1993).

Bentonite clays that predominantly contain montmorillonite, a member of a group of clay minerals termed smectites, have a set of special properties that make them particularly attractive as sealing materials. Montmorillonite is a highly surface-active clay mineral that confers the special properties of swelling, plasticity and very low hydraulic conductivity to bentonite clay, and it also provides the ability to sorb and retain cations.

Cement-based materials were specified for limited use as a functional sealing component in the in-floor borehole emplacement case study because of uncertainty regarding the effects of the cement on the local groundwater chemistry, the waste form and the other engineered barrier materials. Recent test information suggests that newly developed high-performance cements and concretes would have very low porosity, reduced pH and extremely low hydraulic conductivities. Microcracks generated in the high-performance materials would tend to self-seal (Onofrei et al. 1992).

The use of high-performance cements and concretes has been extended in the in-room emplacement study to the interior of the waste disposal rooms, as compared with the in-floor borehole emplacement case study where their use was limited to the sealing function of structural bulkheads at the disposal-room entrances. In the in-room emplacement method, high-performance concrete is used as a functional structure for the construction of a smooth platform on the floor of the room for supporting rails and equipment and for placing and aligning precompacted backfill and buffer blocks, as well as the construction of bulkheads as a sealing function at the disposal-room entrances, in access tunnels and in shafts. Cement-based grouts may be used to control groundwater movement into the excavation and around seals.

In addition to these groups of sealing materials, a thin (50 mm) layer of dry granular bentonite and rounded sand is placed in the gap between the buffer and the container to provide for conductive heat transfer and to maintain the density of the clay-based sealing system. Sealing materials specified for the conceptual design are based on the results of the vault sealing research program presented by Johnson et al. (1994a) and on results from studies in progress.

The hydraulic conductivity is the key property of clay-based sealing materials. This characteristic depends on the predominant clay mineral, the quantity of clay-sized material in the mixtures, and on the dry density, that in turn depends on the placement method. Results of the tests performed on materials delivered to AECL from 1982 to 1990 indicate that hydraulic conductivities of compacted bentonite-based materials are less than 10^{-11} m/s for bentonite contents of 50 wt% or greater and densities greater than 1200 kg/m^3. Three clay-based sealing materials are specified for the in-room emplacement design: the buffer material, the dense backfill material and the light backfill material. The specification for these clay-based materials composition is presented in Table 7.

The buffer (Dixon and Gray 1985) is a mixture of sodium-bentonite clay (a montmorillonite-rich clay found in commercial quantities in the central plains of North America) and well-graded silica sand mixed in a 1:1 dry mass ratio. The buffer material is the same reference buffer proposed for the reference borehole emplacement configuration (Simmons and Baumgartner 1994), but the form in which it is placed is different.

For the in-room emplacement method, the buffer is placed around the container in the form of close-fitting, precompacted blocks since the disposal-container emplacement arrangement and the limited disposal-room height precludes in situ placement and compaction to the required density specifications. A composite structure of compacted blocks can be constructed yielding a near-homogeneous buffer mass with a minimum dry bulk density of about 1.67 Mg/m^3 and an optimum gravimetric moisture content of 17-19 wt%. The minimum dry density is 95% of the dry density attainable in ASTM test D-1557-78 (ASTM 1982). The maximum dimension of any side of a precompacted block is about 1 m. A minimum thickness of 0.5 m of buffer around the disposal container is specified for this study. To improve the thermal contact between precompacted blocks, a mortar-like material with a composition similar to the buffer material and with a high moisture content to form a paste can be used. The buffer material specifications are presented in Table 8.

The dense backfill material is a variant of the reference backfill material proposed for the reference borehole emplacement method (Simmons and Baumgartner 1994). It is a mixture of glacial lake clay (an illite-rich lake clay deposited in glaciated regions of North America), sodium-bentonite clay and crushed granite mixed in a proportion of 25/5/70% by dry weight, with a dry bulk density of about 2.1 Mg/m^3 and an optimum gravimetric moisture content of 8%.

The dense backfill is placed as close-fitting blocks of highly precompacted backfill material. The maximum dimension of any side of a precompacted block is about 1 m. The dense backfill was specified for the lower portion of the disposal rooms, for the lower portion of all other horizontal excavations (tunnels and ancillary service rooms) and for the shafts. The thermal contact between precompacted blocks may be improved by the use of a buffer-based mortar.

A light backfill material is specified to fill the upper part of the room to ensure good contact with the upper perimeter of a disposal room,. The light backfill is placed by a blowing technique. Because this technique will yield a different bulk density for the light backfill than the bulk density of the dense backfill, the light backfill was specified to have a composition of 50% sodium-bentonite clay and 50% crushed granite, by dry weight. Based on the minimum dry density of this mix and the minimum clay dry density, it is judged that this material will yield about the same hydraulic conductivity as the dense backfill. The specifications for the basic physical properties of clay-based sealing materials are presented in Table 9.

Important factors for the disposal vault design are the container heat output and its change with time, and the temperature limits for the outer-shell of the container and sealing materials. For the buffer, a thermal conductivity of $K^* = 1.5$ W/(m°·C) is selected (Baumgartner et al. 1995). For all other sealing materials, the values of thermal conductivity corresponding to the initial condition of moisture content have been specified. The specifications for the thermal and thermal-mechanical properties of sealing materials are presented in Table 10. The values of thermal conductivity presented are preliminary estimates. They were inferred from laboratory experiments and in situ test results, and from the numerical approximations of coupled hydro-thermal-mechanical processes in unsaturated clay. Investigations of the material properties as the conditions evolve within a disposal vault are ongoing.

3.2.3.5 Radiation Protection Considerations

A radiation shielding analysis was performed to determine the effect of the dual-container geometry of the in-room emplacement method on the total dose rate to an exposed worker at the face of the assembled precompacted blocks (G.R. Penner, unpublished memorandum, 1995) and to confirm the previous dose attenuation analyses for the materials (G.R. Penner, unpublished memorandum, 1993).[14] Figure 17 shows the geometric and material model analysed with the computer code, TWOSYS (Brinkley 1990). It is modelled using two-dimensional, cylindrical geometry. Reflective surfaces are used in the model to reduce its size for calculation purposes. The burnup of the used fuel, for this analysis, is assumed to be 1008 GJ/kg U, and its age is 10 years after discharge from a reactor (Section 2.1.1).

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[14] G.R. Penner. 1993 April 06. In-room emplacement shielding analysis. Unpublished AECL Internal Memorandum, SAB-93-159.

Dose rates are determined at points through the buffer material axially opposite the disposal container and along the face of the assembled blocks in the direction of the neighbouring disposal container. The effective chamber length for the disposal container and annular bentonite/sand layer is modelled as 1206 mm. The overall length of the disposal container including the 50.8-mm high lifting ring is 1189 mm. A minimum of 50 mm of bentonite/sand mixture and 1310 mm of buffer is required to achieve the 1 μSv/h maximum dose specification. This arrangement would provide a net minimum of 2516 m between disposal-container centres along the length of the room.

On examination of placing buffer plugs between the container and the workers and inserting them in the breech mechanism of the container cask, a nominal buffer plug thickness of 500 mm is judged to be practicable for handling and placement. Three such plugs provide a total thickness of 1.5 m. Together with the chamber length of about 1.2 m, a spacing of 2.7 m between disposal-container centres is achieved. The net result is a worker dose of about 0.1 μSv/h for the high burnup fuel, of which about 1×10^{-4} μSv/h is the neutron dose rate. The second of three buffer plugs may be substituted by a dense backfill plug if material economies are required. Lower dose rates are expected for the majority of the fuel with lower burnups and for a greater cooling time of the used fuel when disposal is implemented.

3.2.4 Results of Disposal-Room Design Analyses

A staged design process was used in the development of a disposal-room size and shape. In the preliminary analyses, analytical approximations were used in the scoping calculations to arrive at a reasonable distribution of the heat-generating waste within the vault to ensure that disposal-container temperature design limits were not exceeded. Two-dimensional structural analyses were then performed to define the general stability bounds for the openings throughout the depth range of the vault. After a specific geometry for the shape and size of the room was determined, more precise numerical methods were used to confirm and refine the estimates from the preliminary analyses.

3.2.4.1 Preliminary Thermal and Mechanical Analyses

Preliminary Thermal Analyses

Preliminary thermal calculations (Baumgartner et al. 1995) were performed using the HOTROK thermal analysis code (Mathers 1985). This code provides an analytical solution for a large array of heat sources in a semi-infinite half-space assuming constant thermal properties (i.e., conductivity and diffusivity). The code has been modified to provide temperature estimates when additional materials with different thermal properties are placed around a heat source (e.g., sand and buffer around a disposal container).

Results (Baumgartner et al. 1995) showed that the temperature rise on the outer surface of a disposal container at the centre of the disposal vault is 69°C at about 10 years after waste emplacement, after which the container temperature begins to decrease. The maximum temperature at the outer surface of the container is 86°C at a vault depth of 1000 m, given an ambient temperature at this depth of 17°C. A vault situated at a depth of 500 m with the same waste emplacement geometry would undergo a similar temperature rise (i.e., 69°C) in 10 years, but the maximum temperature at the outer surface of the container would reach 80°C because the ambient temperature at this depth is 11°C. At 750-m depth, the maximum temperature is estimated to be 83°C.

The far-field or geosphere-scale thermal analysis (Baumgartner et al. 1995) performed with HOTROK showed that the thermal pulse from the decay of radionuclides in the used fuel would heat a significant volume of rock for a period of over 50 000 years, and it would take over 100 000 years to approach the initial ambient temperature. The thermal analysis indicated that the averaged-peak temperature rise in the rock at the vault horizon is about 40°C.

Preliminary Thermal-Mechanical Analyses

The thermally induced stress at the vault elevation was calculated (Baumgartner et al. 1995) using a simple, order of magnitude analytical solution for a uniformly heated, thick horizontal plate under plane stress conditions (Timoshenko and Goodier 1970). The representation of the rock mass overlying the disposal vault as a thick plate was considered valid because the thickness (500 to 1000 m) is less than the major dimensions of the disposal vault (e.g., 2 km by 2 km) and the associated heat-affected area, and because the rock mass is confined laterally. However, the temperature is not constant in the vertical dimension but follows a gradient. Therefore, this simple analytical solution should overestimate the horizontally generated stresses. This was confirmed by subsequent detailed analyses (Section 3.2.4.2).

The principle of superposition for linear elasticity was used to add the thermally induced stresses to the ambient in situ stresses at the vault elevation to define the range of stable disposal-room aspect ratios under combined excavation and thermally induced stresses (Baumgartner et al. 1995). The results are shown as the "thermal" values in Figures 10 to 13. As discussed in Section 3.2.1.1, an aspect ratio of 2.3 was chosen as the basis for further analyses.

3.2.4.2 Thermal and Thermal-Mechanical Analyses Using the Finite-Element Method

A series of three-dimensional near- and far-field thermal and thermal-mechanical analyses were conducted by Wai and Tsai (1995) and Tsai (1995). These analyses were undertaken to refine the estimates produced in the preliminary (or scoping) analyses (Baumgartner et al. 1995). The detailed analyses are based on the finite-element method of numerical modelling. The studies were conducted using the commercial finite-element code, ABAQUS, on an ALPHA 2100 5/250 computer.

Studies by Wai and Tsai (1995) and Tsai (1995) considered far-field analyses of disposal vaults at two different depths (i.e., 500 and 1000 m), and near-field analyses of 7.6-m-wide by 3.3-m-high elliptical disposal rooms oriented perpendicular and parallel to the maximum principal stress direction at three different depths (i.e., 500, 750 and 1000 m). The far-field analyses considered a sealed condition under elevated temperature, and a glaciated condition under elevated temperature. The conditions considered in the near-field analyses included an excavated condition under ambient temperature and a sealed condition under elevated temperature. The excavated condition is the state of the excavated disposal room prior to the emplacement of disposal containers and sealing materials. The sealed condition is the state of the disposal room and vault after the emplacement of disposal containers and sealing materials. The glaciated condition is the state of the vault at 10 000 years after the vault is sealed when an estimated 3000 m of ice uniformly overlies the surface of the region.

Far-field Analyses

The thermal and thermal-mechanical model representation is shown in Figure 18 as a quarter section, and the thermal and mechanical boundary conditions for the model are shown in Figure 19. The initial gross thermal loading for this model is 6.4 W/m^2. This thermal loading is the weighted average of both the heating (i.e., the disposal rooms and intervening pillars - about 8.0 W/m^2) and non-heating (i.e., all access tunnels, and pillars between tunnels, rooms and tunnels, and ends of rooms - 0 W/m^2) areas in the disposal vault. The peak temperatures in the first 100 years generated by this far-field model are less than that generated by the near-field model (presented below) because the heat generated by the individual disposal containers in the plane of the disposal vault is averaged over the entire waste emplacement area, as defined by the initial gross thermal load. Generally, this model is accurate for periods beyond 2000 years in the immediate plane of the vault and earlier in time as the distance from the plane of the vault increases (i.e., the localized heating effects are "smeared" out).

The major difference between the two far-field cases (vaults at 500- and 1000-m depths) is that the thermal gradients associated with the vault at 500 m are greater than those for a vault at 1000 m (Wai and Tsai 1995). Therefore, temperatures surrounding the vault at 500-m depth start to decline earlier (i.e., <2000 years at 500-m depth and <4000 years at 1000-m depth - Figures 20 and 21). For both of these cases, however, the peak averaged-temperature rise[15] in the rock mass above the centre of the vault is about 31°C, and it occurs about 76 years after waste emplacement. Corresponding, the peak rise in horizontal stress (i.e., the thermally induced stress), for both cases at the vault horizons, is about 26 MPa, about 100 years after waste emplacement. The total maximum horizontal stresses acting along a vertical line passing through the centre of the vault are shown in Figure 22 for the 500-m deep case and Figure 23 for the 1000-m deep case.

The analyses show that the thermally induced stresses reduce the overall horizontal stress magnitude by a maximum of 6 MPa at the ground surface. However, because of the high horizontal ambient stresses in the model (i.e., 9.9 and 18.5 MPa for σ_2 and σ_1 respectively), the stresses remain compressive throughout the 100 000-year period of the analysis. Thus, the near-surface extension zone does not develop (i.e., the depth of the zone is zero metres). The factor of safety (i.e., strength-to-stress ratio) in the region considered in the far-field analyses is greater

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15 The term "peak averaged-temperature rise" represents the highest (i.e., the peak) rise in temperature calculated by the model, at any time. Temperature rise is used as the common basis for comparison because the initial ambient temperature at each vault level differs.

than 2.0 for all post-emplacement times. The maximum uplift is about 100 mm on the ground surface above the centre of the vault at about 10 000 years after waste emplacement.

The weight of 3000 m of ice uniformly distributed over the region of the disposal vault increases the far-field vertical stress by approximately 27 MPa and the horizontal stress by about 9 MPa. These incremental stresses increase the compressive state of the rock mass and, generally, increase the far-field factor of safety (Wai and Tsai 1995). The effect of an advancing glacial front is not analysed and needs to be considered in the context of changes in pore-water pressure.

Near-field Analyses

Wai and Tsai (1995) and Tsai (1995) also analyze the near-field region of a disposal room for six cases of a vault located at depths of 500, 750 and 1000 m; three of which have the room longitudinal axis oriented perpendicular to the major principal stress direction, and the other three cases with the room parallel to the major principal stress direction. The thermal and thermal-mechanical model representation is shown in Figure 24 as a "unit" cell.

The near-field calculations were performed for a three-dimensional "unit" cell of the disposal vault for the first 100 years of heating. The unit cell consists of a parallelepiped portion of the disposal vault and geosphere that is bounded on top by the earth's surface and at the bottom by a plane 2000 m below the vault horizon; on one set of opposing sides by the vertical mid-plane along the longitudinal axis of the disposal room and by the vertical mid-plane along the longitudinal axis of the interroom pillar; and on the second set of opposing sides by the vertical mid-plane between two sets of containers across the room and by the vertical mid-plane passing through the first of the two sets of containers.

For the thermal portion of the analyses, the top (representing ground surface) is a constant temperature (i.e., isothermal) boundary set at 5°C to represent average Canadian Shield surface temperature, and the bottom is also an isothermal boundary set to the ambient temperature 2000 m below the vault (i.e., assuming a geothermal vertical gradient of 0.012°C/m). All of the four vertical boundaries are adiabatic planes of mirror symmetry to reflect the heat generated within the cell (Figure 25). This mirror symmetry mimics the thermal contribution from all the surrounding "unit" cells, in effect replicating an infinite tabular array of infinitely long parallel disposal rooms.

For the structural analyses under ambient and thermal conditions, the boundary conditions are as follows. The top boundary is free to displace vertically, and the perimeter is rigidly constrained laterally. The bottom boundary is rigidly fixed against displacement, both vertically and laterally. The four vertical boundaries are fixed against out-of-plane lateral displacement and are attached to the top and bottom boundaries to maintain the appropriate continuity (Figure 25). This also constrains the "unit" cell to effectively displace consistently with the surrounding "unit" cells and to allow the build-up of horizontal stress caused by thermal expansion. The boundary conditions for the analyses are set such that the unit cell effectively represented the central portion of an infinite tabular array of disposal rooms and heat sources.

The initial gross thermal loading for this model is 8.0 W/m^2, achieved by spacing the disposal containers 2.7-m apart along the length of the room. The calculated peak temperature rise is 64°C at the container skin about 15 years after waste emplacement (Figure 26). Given an ambient temperature of 17°C at 1000-m depth, the peak temperature is 81°C, less than the 90°C design limit. Because the ambient temperatures at depths of 500 and 750 m are less than that at 1000 m (i.e., 11°C and 14°C respectively, yielding peak temperatures of 75°C and 78°C, respectively), the temperature design limit is not reached over this depth range. Figure 27 shows the entire temperature variation with time of the container outer surface at depths of 500, 750 and 1000 m. Figure 27 is created by superimposing the near-field temperature rise (i.e., the first 100 years of heating at an initial gross thermal load of 8 W/m^2 (Wai and Tsai 1995)) onto the ambient temperature (i.e., 11, 14 and 17°C respectively) and adding the far-field temperature rise (i.e., beyond 100 years) produced by the HOTROK computer code at an initial gross thermal load of 6.4 W/m^2 at the corresponding depths, which is representative of the disposal vault with combined heating and nonheating areas.

The analyses show that a state of compression prevails everywhere in the near-field for all cases immediately after excavation and at all post-emplacement times (Table 11). For both room orientations relative to the major principal stress direction, the maximum compressive stress around the room is within the excavation and thermal design limits described in Section 3.2.3.3. There is only one exception where the factor of safety is less than 1: the disposal room immediately after excavation at a depth of 1000 m oriented parallel to the major principal stress direction (i.e., factor of safety = 0.99). Based on Figure 13, the simple design solution to eliminate wall yielding at this depth is to reduce the opening aspect ratio slightly. Because vault designs in Chapters 4 and 5 are produced for depths of 500 and 750 m, this modification is not required for these design applications.

The detailed finite-element analyses (Table 11) also show that the simple analytical stability envelope (Figure 12) developed in the scoping analyses is conservative for roof stability. This envelope in Figure 12 shows the potential for roof instability for rooms oriented perpendicular to the major principal stress direction for depths shallower than about 700 m. However, the factors of safety for roof stability in Table 11 (after Tsai 1995) shows that this clearly is not the case, the values are greater than 1.0. Therefore, the scoping methodology described in Section 3.2.4.1 provides a useful tool for rapidly assessing the stability of excavations under ambient and thermal conditions in sparsely fractured rock. Preliminary designs derived from this scoping tool should be confirmed by more definitive methods (e.g., the finite-element method), as was done in this study.

Excavation-induced displacements at the room perimeter are directed inward (i.e., convergence), and are in the order of 2 mm, in general. Thermal loading causes a further convergence of about 1 mm in the walls of the room, and an expansion of about 2.5 mm in the roof and floor at 100 years (Wai and Tsai 1995, Tsai 1995).

In terms of the factor of safety, values are generally well above 2.0, at all times, beyond a perimeter annulus of about 0.5-m thick or less, depending on the location around the room perimeter, the room orientation and the vault depth. Values generally approach 1.0 at selected locations depending on the case. Figures 28 and 29 shows typical values of the factor of safety surrounding a disposal room at depths of 500 and 1000 m with the long axis of the room oriented perpendicular to the major principal stress direction immediately following excavation and after 100 years of heating respectively. As shown in Table 11, orienting the room parallel to the maximum principal stress direction improves the factor of safety in the roof of the room. However, there is an associated decrease in factor of safety in the sidewall. The net effect is an improvement in the overall stability of the opening under thermal loading conditions.

Although the effect of glacial loading on the near-field stability of the disposal room was not explicitly considered in the analyses, it can be estimated using an analytical solution for boundary stresses around an elliptical opening (Jaeger and Cook 1979). For the case of an ellipse with a horizontal semimajor axis "a" and a vertical semiminor axis "b", the tangential stresses in the roof and at the walls induced by a horizontal stress σ_h and a vertical stress σ_v are given by

$$\sigma_\theta = \sigma_h (1 + 2b/a) - \sigma_v \text{ in the crown of the roof, and} \quad (8a)$$

$$\sigma_\theta = \sigma_v (1 + 2a/b) - \sigma_h \text{ at the springline of the walls.} \quad (8b)$$

Substituting the incremental far-field stresses caused by glacial loading (i.e., $\Delta\sigma_h = 8$ MPa and $\Delta\sigma_v = 26$ MPa) for the horizontal and vertical stresses respectively, in the above equations, the incremental tangential stress (i.e., $\Delta\sigma_t$) in the crown is -11 MPa, and, at the springline (see definition in Section 3.2.1.1), it is 138 MPa. The net effect of superimposing this load on the sealed vault condition is to increase the factor of safety in the crown, but to reduce the factor of safety at the springline. Depending on the confinement provided by the sealing materials within the room, and the effective stiffness (or resistance to deformation) of the sealing materials, the factor of safety may be less than 1.0 over a limited region of the sidewall, potentially resulting in a small zone of damage with increased local hydraulic permeability. However, because of the high stress gradient in this region, this phenomenon would be limited to a region very close to the wall of the room. Moreover, because the 12-m-long room sealing bulkheads would limit displacement of the rock perimeter into the room and may be keyed into the rock mass, this potential damage zone could be discontinuous in nature, being either prevented, or cut off, by the stiff concrete bulkheads. Although glacial loading is speculative at this point, its effects may be accommodated in the disposal-room design. Further analysis is required during detailed design to estimate the response of the sealing bulkhead to the long-term glacial loading.

3.2.5 Disposal-Room Operations

The basic disposal-room operations are described. They include room preparation to provide the logistical and buffer and backfill structures for the disposal containers, the emplacement of the waste within the prepared chambers in the assembled emplacement unit and the plugging of the emplaced container within a chamber and, after the disposal room is filled, the placement of a low-heat, high-performance concrete bulkhead to seal the entrance to the room.

3.2.5.1 Disposal-Room Preparation

Prior to the commencement of waste emplacement operations, the disposal room is prepared for the planned operations. Key installations are the low-heat, high-performance concrete floor, the rail system, the auxiliary ventilation system and the mechanical and electrical utilities.

The concrete floor is placed during the room excavation process, immediately after excavation. After the concrete has sufficiently cured, the double-track rail system is installed and secured to the floor. In the first eight rooms in a panel, four on either side of the panel, the auxiliary ventilation system is installed.

This ventilation system consists of two rigid ducts, each fitted with a 50-kW exhaust fan, which flows about 14 m^3/s. During operation, a portable high-efficiency, particulate air (HEPA) filter is provided on the exhaust from the disposal room where waste emplacement is occurring. Each duct is equipped with a radiation monitor and bypass damper. Under normal conditions, the HEPA filter is bypassed. However, when the monitor detects releases of radioactive contaminants, the damper is activated, an alarm is sounded for disposal-room evacuation and the exhaust air is routed through the HEPA filter. This concept of auxiliary ventilation is similar to the system described by Simmons and Baumgartner (1994).

The following activities take place during block placement:

- retraction of ventilation ducting, electrical and mechanical services and rail;
- emplacement of dense backfill and buffer blocks; and
- emplacement of light backfill.

The operations within a single panel are based upon the maximum emplacement rate of 12 containers/day. The average disposal rate to maintain the required schedule is 9.65 containers/day (i.e., 3.2 containers/shift). The placement of sealing materials and containers alternate from one side of the panel to the other. That is, while crews are placing containers in rooms from one of the panel access tunnels, other crews are placing dense backfill and buffer blocks from the other panel access tunnel.

The first operation in preparing a room for block placement is the retraction of the ventilation ducting, electrical and mechanical services and rail to permit installation of the next emplacement unit (Figure 30). This ventilation ducting and electrical services are retracted in 2.7-m lengths, which is the length of the emplacement units. After each vertical section of precompacted blocks is installed, the rail is removed in 0.9-m sections. The rail is surveyed in place during installation to ensure proper alignment of the precompacted blocks. Removal of the rail in 0.9-m segments is required for equipment operations during installation of the precompacted blocks.

The emplacement of dense backfill and buffer blocks begins after the retraction of the ventilation ducting, services and rail. The precompacted blocks are transported from the block compaction plant to the disposal rooms on rail cars. Equipment for placing the blocks (Figure 31) is positioned at the working face of the room.

A total of 45 dense backfill and buffer blocks are placed for each emplacement unit. They are placed in 3 rows of 15 blocks each, consisting of 7 dense backfill and 8 buffer blocks. The blocks range in size from 0.3 m^3 to 1.0 m^3 and have closely controlled tolerances so there is a tight fit between blocks. Seven dense backfill blocks are first placed on top of the high-performance concrete floor. The concrete floor is appropriately formed, and the rail is accurately positioned to ensure the placement of the blocks in the correct alignment. The buffer blocks are shaped so that the emplacement chambers, that will receive the containers, are formed as the blocks are assembled. If required, a bentonite-clay based "mortar" or dry powder can be placed along block surfaces to fill any excessive gaps. This would be investigated further during detailed design, together with the staggering of seams between placed blocks to prevent the potential for radiation beams passing along gaps.

After all 45 blocks are placed, light backfill is placed into the void between the roof and upper walls of the disposal room and the blocks. This light backfill is placed by pneumatic blowing (Figure 32) with modified shotcrete placement equipment similar to that used in the Stripa Buffer Mass Test (Pusch et al. 1985). A minimum space of 300 mm was provided above the buffer material to allow access for the pneumatic nozzle. This method ensures that the light backfill completely fills all the remaining indentations and irregularities on the tunnel surface. Each container chamber of the emplacement unit is ready to receive one disposal container, followed by three buffer plugs, and a dry particulate, such as glass bead or fractionated silica sand, to fill the eccentric annulus around the container (Figure 2).

3.2.5.2 Disposal-Container Emplacement

The following activities take place during container emplacement:

- emplacement of the container,
- emplacement of the three buffer plugs, and
- infilling of the annulus around the container with dry particulate.

The first step in the container emplacement sequence is to visually measure and inspect each container chamber in the buffer blocks of the emplacement unit into which a container is to be placed. If a chamber is rejected due to quality control deviations or the immediately surrounding rock mass is deemed unsuitable for waste disposal in the particular locale, the chamber(s) is completely filled with the standard buffer plugs and annular particulate to seal the chamber. If the state of the chamber walls in the emplacement unit is acceptable, it is approved for waste emplacement.

A conceptual design has been developed for the disposal-container cask (Figure 33) that is used to transfer disposal containers from the packaging plant to the disposal room in the vault. The cask is exempt from many of the design, testing and qualification requirements of the

Transportation Packaging of Radioactive Materials Regulations (AECB 1990) because it will not be airtight and will not leave the site.

The container cask is designed to attenuate the radiation field from a full disposal container of used fuel to an on-contact dose rate of less than 1 μSv/h. The gamma and neutron attenuation is achieved through the use of steel, lead and polyethylene in cask construction. The overall mass of the cask, including container, is about 36 Mg.

A full container cask is moved from the waste-shaft headframe cask laydown area on a cask car (Figure 34) to the waste shaft. The cask car is a 40-Mg capacity rail car chassis fitted with a specially designed frame to accommodate the disposal-container cask in a horizontal position. The cask and car are loaded into the waste-shaft cage and delivered to the disposal vault. The cask and car are then taken to a disposal room for container emplacement or to the underground cask storage area were it is queued for transport to the disposal room. The underground cask storage normally contains 12 casks (i.e., one-day operation) to meet surge requirements. After emplacement of the container, empty casks mounted on cask cars are returned to surface.

The container emplacement process begins with the positioning of the disposal-container cask in the room (Figure 35). The design accommodates the necessary radiation shielding to allow for the presence of personnel within the disposal room during the emplacement operations. Shielding calculations have determined (Section 3.2.3.5) that 1.5 m of sealing materials is sufficient to meet the 1 μSv/h exposure design limit for workers involved in emplacement operations. Thus the 2.7-m pitch distance, on centres, between disposal containers is effective from a radiological protection perspective. This approach should provide a high level of operational safety for all personnel by having either a container cask or buffer plugs placed between the disposal container and the workers. Administrative procedures may be required to ensure workers withdraw from the immediate vicinity of the interface between the cask and the emplacement unit during the transfer of the disposal container and installation of the buffer plugs. This is an example of the additional design details to be addressed in the future, which are beyond the scope of this conceptual-level design.

The container is installed by opening the cask-bottom split gate and pushing the container into the buffer chamber with the ram through a top head access port by the ram tractor. The bottom leading edge of the disposal container is curved to reduce gouging within the buffer as the container is pushed into the chamber. The cask is also designed to install the buffer plugs (Figure 35). After the container is installed in the buffer chamber, the ram is retracted and the cask split gate is closed. The cask breach is then opened to receive a 0.5-m-thick buffer plug from the plug car. The breach is then closed, the bottom split gate is opened and the buffer plug is installed using the ram. The process is repeated until the three buffer plugs are installed.

Various interlocks are provided to prevent inadvertent operations, such as opening the breach with the split gate opened or with a container in the cask. The massive construction required for radiation shielding results in a very robust cask that resists damage from handling impacts. If a loaded cask is subjected to a severe impact in handling, its interior could be monitored for any airborne contamination. If present, the cask is wrapped in plastic to prevent the spread of contamination (i.e., containment) and returned to the used-fuel packaging cell for further examination.

The final emplacement operation is to fill the eccentric annular and top- and bottom-head residual void around the container with a dry granular bentonite and rounded sand mixture to improve thermal conductivity between the container shell and the buffer and to reduce the residual void for buffer swelling. Placement of this material is accomplished pneumatically through a tube inserted through a small upper gap in the buffer plugs (Figure 36). The block placement and container emplacement operations repeat until the disposal room is filled.

3.2.5.3 Disposal-Room Sealing

The in-room emplacement design requires that disposal rooms be sealed as soon as container emplacement within a room is complete. Concrete bulkheads provide a means of closing the rooms to protect the integrity of the sealing materials. Without a bulkhead as an extrusion restraint, any volumetric expansion of the bentonite clay in the buffer would reduce the dry density of the clay and reduce its effectiveness as a sealing material. The bulkhead also provides an opportunity for applying a nuclear-materials safeguards seal that would allow detection of intrusion into the filled disposal room.

The disposal-room entrance is sealed with a 12-m-long sealing bulkhead, as shown in Figure 37. The bulkhead is assumed to be composed of poured high-performance concrete behind formwork. The final operation is cement grouting of the rock/concrete interface and the surrounding rock mass, as appropriate, for the excavation disturbed zone and natural fracturing conditions. The complete backfill and sealing of a disposal room requires an average of 17 working days for the placement of 3570 precompacted blocks and 158 disposal containers in the 238-m-long room.

During the later decommissioning stage, all tunnels within the vault are backfilled and sealed in a similar fashion as the disposal rooms. The tunnels have a similar high-performance low-heat concrete roadbed as the disposal rooms. Precompacted buffer and backfill blocks are placed in a similar fashion as the disposal rooms although no container chamber is formed. The light backfill would then be placed in the space between the blocks and the tunnel upper perimeter. Periodically, an assemblage of buffer blocks would be placed in conjunction with a concrete bulkhead to form a sealing bulkhead at strategic locations along the tunnels, particularly on both sides of an intersected fault (Johnson et al. 1994a).

3.2.6 ADAPTATIONS FOR STEEL-SHELL-SUPPORTED COPPER CONTAINER

No disposal-room design is specifically created for the steel-shell-supported copper container. By inspection, only minor dimensional changes need to made for this slightly larger container (i.e., 994-mm-diameter x 1349-mm-length vs. 860-mm-diameter x 1189-mm-length for the copper-shell, packed-particulate container).

3.3 DISPOSAL VAULT

The disposal vault consists of the waste emplacement area and the underground access ways and infrastructure required to safely conduct the disposal operations. Essential components are the assemblage of disposal rooms in the waste emplacement area, the shafts for vertical access to the vault level, the tunnels that connect all of the rooms, and the ancillary services required to excavated rock, prepare and distribute the sealing materials, transport personnel, materials and equipment, and provide maintenance.

3.3.1 General Requirements

The disposal-vault arrangement for in-room emplacement of nuclear fuel waste is a system of access tunnels and disposal rooms arranged into eight distinct panels (Figure 38). The overall dimensions of the container emplacement area are about 2 km by 2 km. These dimensions are based on an ideal site and do not account for any adaptations that may be required at an actual site because of local conditions (e.g., specific rock structures, faults and stress anomalies).

The in-room emplacement vault design uses central access and perimeter tunnels that join at opposite ends of the vault where two sets of shafts (i.e., shaft complexes) are located. With the central access tunnel twinned, two independent halves of the vault are created. Each half is divided into 4 panels of disposal rooms by panel tunnels for a total of 8 panels. Panel tunnels are perpendicular to the central access tunnels. A panel consists of the 64 disposal rooms contained between two adjacent panel tunnels (Figure 38). Each panel is divided into distinct halves, each half containing 32 disposal rooms. With a 2.7-m pitch spacing of containers along the length of the disposal room, and with 2 containers placed across the with of the room, each 238-m-long disposal room can hold 158 containers. The arrangement, shown in Figure 38, has a maximum capacity of 80 896 containers or 5 824 512 intact fuel bundles. The nominal capacity of the vault is stated as 5.8 million fuel bundles.

The following requirements and factors were considered in determining the vault layout for this study:

- providing a vault-level extraction ratio of about 0.25;
- spacing the used-fuel containers to limit the maximum temperature of the container outer surface, or the peak buffer temperature, to 90°C;
- providing five shafts for operations;
- providing for flexibility of operations;
- separating radioactive and nonradioactive working environments;
- providing appropriate ventilation;
- ensuring reasonable traffic flow patterns;
- ensuring that excavation and emplacement operations retreat from the upcast-shaft complex to the service-shaft complex as disposal rooms are filled (Simmons and Baumgartner 1994);
- providing underground ancillary support facilities outside the waste emplacement area;

- establishing the shaft complexes at least 200 m away from the waste emplacement area to reduce the temperature increase around the shafts;
- directing the underground drainage-water flow and exhaust ventilation air towards the upcast-shaft complex, where it is discharged to surface under controlled and monitored conditions (Simmons and Baumgartner 1994); and
- applying a nuclear-material safeguards method for used-fuel disposal although no requirements have yet been established by the International Atomic Energy Agency (IAEA).

Excavation equipment within the disposal vault is a combination of rubber tired and rail-mounted equipment. Underground movement of materials and personnel is provided by a rail system which is installed throughout the vault. The rail system consists of 80 lb/ft (119 kg/m) ASCE (American Society of Civil Engineers) rail on steel ties with a gauge of 1.26 m. The rail system provides

- stable and rapid equipment movement and alignment,
- simplified repeated positioning of the equipment for disposal-room block placement,
- simplified repeated positioning of the equipment for container emplacement,
- reduced friction and low effort for movement of equipment,
- bulk handling of materials (i.e., a multi-unit train),
- reduced materials handling and transfer operations, and
- reduced requirement for heavy lift equipment underground.

The perimeter access tunnel surrounds the disposal vault. Entry to the emplacement panel is made via the perimeter or central access tunnels. The central access tunnels separate the vault into two halves. Excavation and installation of disposal-room services takes place in one half of the vault while buffer and/or backfill block placement and container emplacement is taking place in the other half of the vault (Figure 39). This separation in activities is essential to smooth vault operation and worker safety. Such a system of tunnels allows independent flow of containers for emplacement from one direction (i.e., along a central access tunnel) while the sealing materials can move into the disposal room from another direction (i.e., along a perimeter tunnel). One panel tunnel is used to gain access to one half of the disposal rooms for placement of containers, while the other panel tunnel is used for block placement activities.

Having excavation panel operations physically separated from emplacement panel operations by the central access tunnels minimizes worker exposure to potentially radioactively contaminated air, vault drainage water, or disposal-container transporters and to complex traffic flows.

Ventilation airflows can be readily distributed, controlled and segregated using the tunnel network. Two independent ventilation circuits are provided, one for the emplacement side and one for the excavation side of the vault. Ventilation control doors are used to direct and control the quantity of fresh air required for emplacement and excavation activities. The doors are equipped with interlock alarms and position monitors to ensure that the proper flows are maintained.

Because operations retreat from the upcast shaft-complex to the service-shaft complex, fresh air flows through the operation areas before exhausting through the completed excavations or emplacement areas. This reduces the potential for blasting gases, dust or radioactive contamination from entering the occupied operating areas. Within a given panel, fresh air is supplied from the perimeter access tunnel and is exhausted through the central access tunnel.

Drainage water from the excavation and emplacement panel operations collects in a sump located at the bottom of the upcast-shaft complex. The drainage water is then pumped to the surface settling pond and water-treatment plant where potentially contaminated water can be treated. The drainage-water is then sent for reuse underground or released into the environment after meeting regulatory requirements.

At any given time in the operation stage, the central access tunnel and perimeter access tunnel associated with one half of the vault are used for excavation. The nature of this operation allows the use of one-way traffic flows, thereby simplifying the procedures. Empty rail cars are shunted down the central access tunnel to the panel tunnels where disposal rooms are being excavated. The cars are loaded with blasted rock by loaders working in the disposal rooms, and then the cars travel along the perimeter access tunnel to the service shaft complex. Here the rail cars are emptied of waste material and are dispatched to the central access tunnel for the next loading cycle.

Disposal containers and directly associated personnel use the other central access tunnel associated with waste emplacement operations. Remaining personnel, equipment, supplies and materials associated with waste emplacement use the corresponding perimeter access tunnel.

3.3.2 Excavation Method

The excavation method selected for in-room emplacement method is the same as in the reference borehole emplacement case (Simmons and Baumgartner 1994), the drill-and-blast method. The drill-and-blast method is favoured when excavating elliptical shaped rooms in highly anisotropic in-situ stress fields such as those assumed for in-room emplacement of waste. Experience in AECL's Underground Research Laboratory indicates that the damage induced by drill-and-blast excavation can be minimized by careful blast-round design and application (Kuzyk et al. 1987, Favreau et al. 1987).

Where ground control is required, standard rock-bolting methods using scissor-lift trucks are used. The actual bolting requirements are dictated by the observed ground conditions, and could vary from no bolts in regions of sparsely fractured rock to heavy bolting with screening and/or shotcrete in fractured zones.

Either "pilot-and-slash" or "full-face" excavation techniques can be used depending on specific site conditions. The full-face technique is more productive than the pilot-and-slash method and is the preferred excavation method where the rock quality is appropriate and acceptable excavation-perimeter quality can be achieved.

Estimates of the time required for room excavation operations are based on underground construction methods using careful drill-and-blast methods. Three years are needed to develop 64 disposal rooms from the panel access tunnels. Each disposal room is nominally 7.3 m wide by 3.0 m high and 238 m long. To achieve this, excavation occurs simultaneously in three rooms, with one crew working in each. Flexibility in assigning work areas is achieved by having several excavation faces available in one area. The three crews are assumed an average advance rate of 16 m/d or 5.3 m/shift. The excavated rock removal rate is about 425 Mg/shift during the construction stage and drops to about 260 Mg/shift during the operating stage of the disposal vault. The rock excavation and handling equipment used in the construction and operating stages are shown in Table 12. The panel-tunnel and disposal-room services, including electrical power, water, compressed air and ventilation, are installed as part of the excavation operations.

3.3.3 Shafts

Five shafts are included in the conceptual design and are equipped similar to the shafts presented in the previous disposal facility design employing in-floor borehole emplacement (Simmons and Baumgartner 1994). The shafts are divided into two groups, the service-shaft complex and the upcast-shaft complex (Figure 40). Both complexes include other underground installations. The service-shaft complex includes the surface and underground systems of the service shaft, the waste shaft and the downcast ventilation shaft, the underground service and maintenance areas, the buffer/backfill preparation and compaction plants, the component test area (see Section 4.1.2) and associated ancillary systems. The upcast-shaft complex includes the emplacement and the excavation panel upcast ventilation shafts, the associated ancillary systems, and the retrieved-container transfer facility.

The service shaft is lined with 0.3 m of concrete and has an internal diameter of 7.3 m. Once operational, the service shaft provides the main services and underground access for workers, materials and equipment for the end of the construction stage, the operation, decommissioning and the optional predecommissioning extended monitoring stages. A permanent headframe is erected on completion of the sinking operations and the service shaft is equipped with

- two 9-Mg capacity skips for in-balanced hoisting of excavated rock and/or lowering of buffer and backfill component materials;

- one 10-Mg capacity service cage in balance with a counterweight for hoisting workers, equipment and supplies; and

- one 1-Mg capacity auxiliary cage without a counterweight for hoisting workers.

The service shaft is also equipped with power and communication cables and service pipes, including a concrete supply line, lake clay and bentonite supply lines, and compressed air, water and diesel-fuel supply lines. The service shaft is ventilated with an upward air flow of 69 m^3/s, supplied by the downcast ventilation shaft.

The waste shaft, which is equipped with a 40-Mg-capacity cage in balance with a counterweight, is dedicated to the transport of container casks. The hoisting-system capacity is increased from 38-Mg described by Simmons and Baumgartner (1994) to accommodate the weight of the container-cask car. No personnel, operating supplies or equipment are hoisted in this shaft other than for inspection and maintenance. The waste shaft is lined with 0.3 m of concrete and has an internal diameter of 4.0 m. The concrete lining is installed to facilitate decontamination if the shaft becomes radioactively contaminated and to eliminate the possibility of loose rock falling off the shaft walls and interfering with the container cask transfers.

The minimum cycle time to load a full cask and car onto the cage at the surface, lower it to the emplacement level, unload it, load an empty cask onto the cage, return to the surface and unload the empty cask is estimated to be 44 minutes for a disposal vault at a depth of 1000 m. This capacity exceeds the peak requirement of 12 containers/day to maintain the disposal schedule.

The downcast ventilation shaft provides the fresh ventilation air supply for all operations and activities in the disposal vault. This shaft is lined with a 0.15 m of concrete to reduce resistance to air flow and has an internal diameter of 4.6 m. The air is heated to above freezing temperatures for winter operation.

There are two upcast ventilation shafts located in the upcast-shaft complex at the end of the vault opposite to the service-shaft complex (Figure 40). Both shafts are lined with 0.15 m of concrete and have internal diameters of 3.65 m. One shaft, the emplacement panel upcast ventilation shaft, provides exhaust ventilation for the panels where container emplacement and associated activities are in progress. This flow is classified as potentially radioactive, and the concrete lining accommodates decontamination and prevents the movement of radionuclides into the rock surrounding the shaft. A HEPA filtration system is installed as part of the shaft system to remove potential contaminants from the air flow. The HEPA filters are normally operated in "bypass" mode and come on-line when radiation monitors detect unacceptable contamination levels. Two drainage water discharge pipelines are located in this shaft. Potentially contaminated vault drainage water is pumped to a surface water-treatment plant from an underground sump located in the upcast shaft complex. The excavation panel upcast ventilation shaft provides exhaust ventilation for the panel where excavation is occurring.

3.3.4 Tunnels

The panel tunnels are sized to accommodate the underground container-cask car, transport of material, such as buffer and backfill blocks, and the room-to-room transfer of the equipment. The panel tunnels are wide enough to allow these transport vehicles to pass each other, if required. The panel tunnels are nominally 10 m wide and 4.4 m high. The central access and perimeter tunnels have the same requirements as the panel tunnels and these tunnels are also 10 m wide and 4.4 m high.

During the construction stage, all 64 disposal rooms of Panel A are excavated and serviced, and 16 rooms of Panel B and 16 rooms of Panel C are also excavated. The panel excavation and emplacement sequence during the early part of the operation stage is shown in Figure 39a.

Commencing in the operation stage, the emplacement of used-fuel disposal containers begins in Panel A, excavated and serviced as part of the construction stage, while crews are excavating and installing services in Panel B. Excavation and emplacement operations are performed on opposite sides of the central access tunnels to allow separation of ventilation airflows and material flows. Excavation operations progress at a rate consistent with emplacement operations, which are estimated to take about three years per panel using a 3 shift/d, 7 d/week operation (i.e., 360 d/a) basis.

An operational changeover occurs as the emplacement operations in a panel are completed. For example, assume that waste emplacement is occurring in Panel A and room excavation and service installation are occurring in Panel B (Figure 39a). When the emplacement operations in Panel A are completed, emplacement activities begin in Panel B. Room excavation and service installation then begin in Panel C, the next panel retreating back from Panel A (Figure 39b). At this time, the roles of the central, perimeter and panel access tunnels and the ventilation system upcast shafts are exchanged.

At any time, emplacement operations take place within a group of eight disposal rooms associated with a panel; four rooms off each panel tunnel. Four rooms and their associated panel tunnel are used for the installation of the buffer and dense backfill blocks. The other four rooms and panel tunnel are used for container emplacement operations.

3.3.5 Disposal Rooms

The 7.3-m-wide by 3.0-m-high disposal rooms in the vault are excavated during two stages. Room development during the construction stage includes the 64 rooms of Panel A, and 16 rooms in Panel B and 16 rooms in Panel C (Figure 39). During the operations stage, the remaining disposal rooms are excavated and prepared for emplacement.

Excavation panel operations consist of disposal-room excavation, followed by the installation of room services required for emplacement operations. High-performance low-heat concrete is placed on the floor of the disposal rooms. Concrete forms would be surveyed in place to ensure the proper positioning of the dense backfill and buffer precompacted blocks. The installation of rail, ventilation, electrical and mechanical services are also conducted in the excavation panel. The emplacement operations within each disposal room follows the sequence shown in Figure 41, which takes advantage of the parallel panel access tunnels to separate waste emplacement operations from other nonradiological operations. Emplacement operations are based upon placing a maximum of four containers/shift, with the requirement to achieve an average of 3.2 containers/shift. Such a schedule allows up to 25% down time for any unscheduled delays.

By scheduling concurrent container emplacement and disposal-room excavation operations, the duration of the construction stage and, therefore, the costs incurred prior to beginning disposal, are reduced. As well, the time between excavation and sealing of a disposal room is minimized, which minimizes the amount of effort required to maintain a continuing safe working environment in disposal rooms.

The estimated time for moving a loaded container cask to a disposal room and return with an empty cask to the waste-shaft station is 72 minutes. A maximum of 12 locomotive trips/day are required to transfer the required maximum 12 containers/day if only one cask car is transferred per trip. However, a train unit of 2 to 4 cask cars per trip is conceivable. In this case, the cask cars may be parked in an unused disposal room and a locomotive may shunt the loaded and unloaded cask cars into and out of the disposal room throughout the shift as required. Figure 42 shows an empty cask car being retracted from the working face in a disposal room after the last container has been installed within an emplacement unit.

Sealing of the disposal rooms encompasses installation of room seals and final bulkheads upon completion of emplacement operations within the room.

3.3.6 Sealing Materials Handling

Component materials, used to formulate the buffer, backfill and concrete, must meet material specifications and inspections before they are approved for use in the disposal vault (see Section 3.2.3.4). Bulk carriers supply most of the component materials, other than the crushed granite, which are then stored on surface in the sealing materials storage bins or at the concrete batching plant.

The crushed rock and concrete used within the disposal vault are produced at surface facilities located within the disposal facility. Full-time qualified operators and inspectors monitor the production process to ensure the end products meet the specified requirements.

The rock crushing plant produces crushed granite with a size distribution suitable for the dense and light backfill and the high-performance, low-heat concrete. The crushing plant uses excavated rock brought to surface from the underground excavations. Of the total 8.9 Tg of granite excavated from the disposal vault, about 33% or 2.9 Tg is estimated to be crushed and returned to the disposal vault as aggregate for these products.

The high-performance low-heat concrete is moved from the batching plant to the service shaft using rotating-drum trucks. The concrete pumping system in the service shaft delivers the concrete to a remixing and truck filling station underground. The other materials, used to formulate the buffer and backfills, are moved by pneumatic conveyors from the sealing materials storage bins to the service shaft. The quantities and methods of transfer for materials within the service shaft from surface to underground are shown in Table 13 for the operating stage.

The skip and transfer pipe capacities in the service shaft are sized to satisfy the average demands. The effect of peak demands and details of dust suppression and collection systems would be fully considered in a future design optimization process.

Sealing materials, used to formulate the buffer, dense backfill and light backfill, are stored in underground bins located above the main level of the service shaft complex and adjacent to the service shaft. The batch mixing and block compaction areas are also located underground

(Figure 40). The material components are taken from the bins in metered loads and mixed to prescribed process requirements. The batch mixing process is similar to that presented in the previous in-floor borehole emplacement vault (Simmons and Baumgartner 1994). It comprises two batching circuits, any one of which can produce the desired dense backfill, buffer or light backfill products as required.

To produce the dense backfill, the crushed granite, glacial-lake clay and bentonite clay are simultaneously withdrawn from their bins and transferred by belt and screw conveyors respectively in the backfill batching circuit, to individual weigh hoppers (Simmons and Baumgartner 1994). When the weigh hoppers are loaded with the required quantity of material, they automatically discharge through feeders into a rotating-pan mixer. The moisture content of the mix is adjusted by the metered addition of water into the mixer from the domestic-water storage tank. The dense backfill is mixed until it reaches the specified degree of homogeneity by applying a method specification that is periodically confirmed by sampling. The mixed dense backfill is then delivered by conveyor to the block compaction plant.

The mixing process for the buffer and light backfill products are similar to the dense backfill process. For buffer, bentonite clay and silica sand are withdrawn and transferred to weigh hoppers by a screw conveyor, discharged into a rotary-pan mixer and water is added and mixed until the desired material specifications are met. The buffer is then transferred to the block compaction plant by conveyor. For the light backfill, the blended material of bentonite clay and crushed granite is discharged into rail cars equipped with rotating mixers for direct delivery to the disposal rooms. The product quality is achieved by following a method specification that is confirmed regularly by material sampling and testing of grain size, moisture content, compaction and swelling characteristics.

From the batch mixers, the buffer and backfill material is belt-conveyed to hoppers in the block compaction area. Fourteen block compaction machines (Figure 43) are located in the compaction area. Nine compaction machines are required to meet the daily supply of compacted blocks, leaving five spares. The spare machines would be used when other machines are being serviced for mold and compaction head changeouts and repair/maintenance or if block production falls behind schedule. All 14 block compaction machines are located in a gallery near the service shaft.

Each block compaction machine receives material from the metering hopper located above the block mold. Enough material to form a 50-mm-thick compacted layer is poured from the metering hoppers into the mold of the compaction machine. The mold is vibrated to level the material. The compaction head is lowered onto the layer of material and the mold is strongly vibrated again, creating the desired compaction density. This process of material placement, levelling and compaction is repeated until the block is completed.

Completed blocks are conveyed from the compaction machines to a block-loading gallery where the blocks are inspected, sorted, and placed on rail flat cars. The blocks are grouped logically on the flat cars to provide the desired combination for assembly when they reach the disposal room. Trains of compacted blocks are taken from the block-loading gallery to the disposal rooms that are being prepared. The compacted blocks are assembled into the emplacement units that receive the disposal containers.

The block-loading gallery is located parallel to the block compaction plant. It is connected to the compaction plant by short cross-cutting tunnels. Seven tunnels serve to transfer the compacted blocks from the compaction machines, one tunnel for two machines. An eighth tunnel serves as an access way from the block-loading gallery to the compaction machine gallery. In emergency situations, all of the cross-cutting tunnels could be used by personnel for egress.

A considerable amount of material will be handled during disposal-room preparations. The rate of container emplacement determines the rate of all the other operations including material handling. Table 13 summarizes the material handling requirements per shift for the waste emplacement operations in the disposal rooms and the total quantities during the operation stage of the disposal vault.

3.4 USED-FUEL PACKAGING FACILITIES

The used-fuel packaging facilities consist of the used-fuel packaging plant, the auxiliary building, and the radioactive waste management areas that are located within the radiologically active zone of the disposal facility. The used-fuel packaging facilities for the in-room emplacement case study are essentially the same as the used-fuel packaging facilities described in the reference borehole emplacement case study (Simmons and Baumgartner 1994). This section of the report primarily focuses on the used-fuel packaging plant because it is the only part of the facilities that is modified from the reference study to accommodate the copper-shell, packed-particulate disposal container.

3.4.1 General Requirements

Used-fuel handling operations are performed in the used-fuel packaging plant. The associated hazards require remote operation, reliable process systems and special support facilities. All used-fuel handling processes are assumed to be capable of meeting nuclear material safeguard control measures that the International Atomic Energy Agency may formulate for disposal facilities (see detailed discussion in Simmons and Baumgartner (1994).

The used-fuel packaging plant is a two-storey reinforced concrete structure assumed to be designed and constructed according to Canadian practice for concrete containment structures of the CANDU nuclear generating stations (CSA 1982), which also accommodates seismic loading (CSA 1980). The used-fuel packaging plant would be constructed and commissioned prior to the start of disposal operations.

Packaging plant operations include the transfer of used-fuel bundles from the shipping modules to the disposal container fuel baskets, 72 bundles to a basket, and the installation of each basket is within a corrosion-resistant, copper container for subsequent placement in the disposal vault. Other operations include the receipt and storage of shipping modules, container welding, inspection of container welds, repair and/or rework of containers that fail quality inspections, decontamination of the container, and the transfer of the container to the container cask, or to the container surge-storage pool prior to disposal.

The used-fuel packaging plant processes about 3470 containers per year. This is achieved by operating two parallel packaging lines, 230 d/a, 5 days a week, on a two, 8-hour shifts/day basis. The following is a brief review of the used-fuel packaging plant and the differences attributable to use of the copper-shell disposal container instead of a titanium-shell container.

3.4.2 Used-Fuel Receipt and Storage

At full operating capacity, about 50 storage/shipping modules, each containing 96 fuel bundles, are received at the disposal facility each week. This requires receiving either 25 road casks (Figure 44) or 9 rail casks (Figure 45) or some combination of the two cask types per week.

The road or rail transporter is received at the transportation cask receiving and shipping area (Figure 46). In the normal situation where the transportation cask is not damaged in transit and at least one of the safeguards seals is intact, the cask is handled following normal procedures. If the cask is damaged, or both safeguards seals are broken, the cask is moved to the damaged transportation-cask hot cell. Normally, a filled transportation cask is removed from the transporter, the impact limiter is removed, and the cask is placed in the full-cask laydown area. An empty, decontaminated cask, loaded with empty storage/shipping modules, and an impact limiter is placed on the transporter for return to the nuclear generating stations. Metal tags and safeguards seals are used to identify loaded casks, and metal tags are used to identify empty, decontaminated casks. To further identify the status of the casks, the metal tags would have appropriate wording (e.g., full or empty) and could be colour-coded to improve recognition.

Depending on the state of the packaging plant operations, the module is either transferred directly to the used-fuel packaging cell (Section 3.4.4) or to the receiving surge-storage pool. In the latter case, the modules are lowered into the pool on an inclined elevator, and they are transferred using the pool manbridge and module-handling tool into secure stacking frames within the pool (Figure 47). The modules are retrieved and put into the packaging cell by reversing this storage sequence.

The probability of a transportation cask being damaged on receipt or of safeguards seals being broken is low. In this event, the cask would be transferred to the damaged transportation-cask hot cell, which is accessible from the transporter receiving and shipping area (Figure 46). This operation is essential for any damaged casks. The operations in this hot cell would likely be manually controlled because of the wide variability in the possible physical condition of the transportation casks.

3.4.3 Container and Basket Receipt and Storage

Disposal containers and used-fuel baskets may be fabricated at the disposal facility or by an off-site fabricator. Off-site fabrication is assumed to take place in this study. Segregated areas are provided at the basket and container storage building and at the used-fuel packaging plant for receipt, inspection and storage of the copper container shells and top heads, and the stainless steel baskets. Inspection facilities are provided at the fabrication plants and within the disposal facility to ensure complete dimensional and fabrication quality control of the baskets and containers.

3.4.4 Used-Fuel Packaging

An empty storage/shipping module is placed on the empty module trolley in the used-fuel packaging cell (Figure 47) and a full module is picked up from the full module trolley. The bridge and carriage positions the module relative to the used-fuel transfer assembly.

The used-fuel transfer assembly is loaded with a used-fuel container basket in a horizontal position. The basket is positioned using rotary and lateral motion so that fuel bundles can be transferred to all positions of the basket (Figure 48).

The individual used-fuel bundles are transferred sequentially from the storage/shipping module via the transfer carousel into one of the 36 stainless steel tubes of the disposal-container basket. The end plates on the bundle can be cleaned during this operation so that the manufacturer and serial numbers can be read and recorded for additional accounting purposes, if necessary. If desired, the optional gamma-radiation monitor can measure the magnitude and energy spectrum of the radiation being emitted from the bundle to confirm the presence of used fuel for nuclear-materials safeguards purposes. In the event that a bundle is damaged from shipping or handling and is unable to be transferred to the basket, or cannot be adequately identified and needs further examination, it is transferred into a special handling area using the bundle-retrieval service ram (Figure 48).

When a storage/shipping module is empty, the operation stops, the empty module is returned to the appropriate transfer trolley, and a full module is picked up by the bridge/carriage assembly. When a container basket is filled with 72 fuel bundles, the used-fuel transfer operation stops and the basket is rotated to the vertical position. The loaded basket is moved in its vertical orientation to the container loading station.

An empty container shell is placed on the container shaker table at the container-loading station. The loaded basket is lowered into the container. The residual void in the container is then filled with a fixed volume of dry particulate such as glass bead or fractionated silica sand from the particulate-metering hopper, and the particulate is compacted by the vibration of the table. Dry fractionated silica sand (see Section 2.1.5) and glass beads have good flow properties to fill the residual void within the disposal container. After the particulate is compacted, the top edge of the container shell is cleaned of any particulate that may be resting on it by using a vacuum cleaning system.

If necessary, from a nuclear-materials safeguards perspective, the serial numbers of all fuel bundles can be recorded against the serial number of the disposal container into which they are loaded and sealed.

The container of used fuel and compacted particulate is transferred to the rotary table in the vacuum chamber for electron-beam welding. The mating surface for the top head on the container is brushed clean of any residual debris. The top head is then lowered onto the container shell. If necessary, the top head may be pressed down with a ram to ensure that the flange of the head is resting firmly and evenly on the top edge of the container. The turntable is then raised or lowered to precisely align the head-to-shell junction with the electron beam. After closure and sealing of the vacuum chamber, the chamber is evacuated of air to a maximum pressure of 0.7 Pa (Blakely 1983), and then electron-beam welding begins. The weld is made by rotating the container around its axis on the rotary table so that the electron beam impinges on the head-to-shell junction. Only a single weld pass is required. The total time required to install the container and head into the vacuum chamber, align the junction with the electron gun, seal and evacuate the chamber and make the weld is estimated to be one hour.

After removal from the vacuum chamber and a visual examination, the sealed container (Figure 49) is moved to the rotary table of the ultrasonic inspection station. Equipment is provided at the station to inspect the integrity of the final closure weld between the lid and the container using ultrasonic methods. Development studies conducted in Canada and Sweden on ultrasonic inspection (Moles 1986, Maak 1988, Sanderson et al. 1983, SKBF/SKB 1983) have indicated the viability of this procedure for determining the integrity of the electron-beam weld.

Containers that fail to pass inspection are moved to the container repair station for repair or disassembly. If the ultrasonic inspection identifies an unacceptable defect in the top head joint, an attempt may be made to repair it by performing a second electron-beam weld at the location of the defect. If a container fails inspection after being repaired, the container is disassembled and the loaded basket retrieved for reloading into another container at the container-loading station.

A container that passes inspection is decontaminated at a decontamination station using water jets to wash the container exterior. A wet-vacuum system is used to remove water sitting on the top-head closure, and air driers complete the process. Swipe tests are performed to check for surface contamination. The dried container is then ready to be loaded into a container cask or placed in the headframe surge-storage pool (Figure 46). The headframe surge-storage pool provides temporary disposal-container storage/supply in the event that either packaging or disposal operations are stopped, preventing interference with the other facility's operation. The container cask provides a safe method of handling the disposal container from the packaging plant to the waste shaft and within the disposal vault.

3.4.5 Decommissioning of Used-fuel Packaging Facilities

The decommissioning of the used-fuel packaging facilities consists of removing all radioactive and nonradioactive systems, installations and structures, so that the site can be released to the public for other use. The decommissioning of the used-fuel packaging facilities is dependant upon the total radioactive contaminant inventory. The radioactive contaminant inventory depends on the internal release of contaminants and the degree to which these releases are cleaned up during plant operation. Waste generated during the used-fuel packaging plant operation are assumed to have been disposed of and all equipment has been kept to normal operating contamination levels. The radioactivity released during used-fuel packaging plant operations consists mainly of corrosion products detached from the fuel elements.

Most of this loose contamination would have been collected by the routine cleaning of floors and walls and by the storage pool and ventilation filtration systems. Any remaining loose contamination is usually removed and contained using a fluid wash or vacuuming with special machines. The key emphasis throughout the operation and decommissioning stages is waste minimization and its segregation from uncontaminated materials.

During decommissioning, waste can be generated from structural materials (mostly contaminated concrete and hot cell liners) and from equipment that cannot be, or that is not economical to be, decontaminated. Such waste is generally disassembled or demolished and placed in containers for disposal in approved facilities. The structural and equipment solid wastes are classified by the IAEA (1987) as follows :

- Low-level waste that does not require shielding during normal handling and transportation because of its low radionuclide content and that does not contain alpha-emitting radionuclides in quantities over the regulatory limits for uncontrolled release.

- Intermediate-level waste that has a lower radioactivity and heat output than high-level waste but generally requires shielding during handling and transportation. An exception to the shielding requirement may be intermediate-level waste that contains one or more alpha-emitting radionuclides, usually actinides, in quantities above the regulatory limits for uncontrolled release.

- High-level waste that has a radioactivity level comparable with nuclear fuel waste.

The volume of radioactively contaminated material collected during the decommissioning stage is estimated to be about 2000 m^3, assuming extensive use of compaction methods for volume reduction proposed by AECL CANDU et al. (1992).

After contaminated materials are removed from the used-fuel packaging facility, a final detailed radiological survey is conducted to confirm acceptability. The remaining task is then conventional dismantlement, demolition and removal of the building and substructures to an approved disposal area on the site.

3.5 COMMON SURFACE FACILITIES

The common surface facilities for the in-room emplacement study are essentially the same as the used-fuel packaging facilities described in the reference borehole emplacement case (Simmons and Baumgartner 1994). The common surface facilities represent the buildings and services that are common to the operation of the surface and underground facilities.

3.5.1 General Requirements

The processes and services required are water (supply, collection and treatment, sewage and settling ponds), normal and emergency electrical power, ventilation, compressed air, security, administration offices, storage and maintenance areas. The common surface facilities are designed to provide adequate site infrastructure (roads, drainage, lighting, security), appropriate buildings and structures, service systems and components that are required to meet the functional requirements of both the surface and underground facilities.

The service requirements of the following underground infrastructure facilities are also accommodated within the disposal facility common surface facilities design:

- underground disposal vault,
- waste shaft headframe,
- service shaft complex at surface,
- downcast ventilation fanhouse, and
- concrete batching and rock crushing plant.

Because the disposal facility conceptual design presented is not site specific, assumptions have been made about the nature of the site for design purposes. The site is assumed to be

- relatively flat and undeveloped;
- within 300 km of a populated centre (~15 000 inhabitants);
- within 25 km of a suitable highway, railway and electrical lines;
- adjacent to a suitable source of fresh water (at least 250 L/s);
- in a plutonic rock body of the Canadian Shield;
- in a zone of low seismic hazard; and
- unpopulated within the required surface property area boundary.

The disposal facility (Figure 50) is a self-contained complex. The site exclusion boundary is extended over the underground disposal vault portion of the facility as well. The disposal facility (Figure 51) site has overall dimensions of 5.2 km x 3 km. The site is divided into a nonradioactive, unfenced supervised area to which public access is discouraged by signs posted on the perimeter, and two potentially radioactive, protected areas that are fenced to inhibit and aid in the detection of unauthorized entry.

Installation of the common surface facilities early in the construction stage makes these facilities available to support ongoing underground and surface facility construction work. This reduces the costs associated with temporary installations, and provides ample opportunity to test the permanent installations under service conditions.

3.5.2 Water Supply, Collection and Treatment

It is assumed the used-fuel disposal facility is located near a lake or river of sufficient size in order to have a suitable source of fresh water. This is where the pumphouse, housing the process water supply, the rock-crushing-plant-water supply and the fire-water supply is located. Two 50%-capacity electrically driven fire-water pumps supply the fire-water system at a maximum capacity of 17 280 m^3/d. For this in-room emplacement study, there is a 30% reduction in the process water demand (mainly because of the reduction in backfill quantities) as compared with the reference borehole emplacement case (Simmons and Baumgartner 1994). Process water is drawn through screens at a peak rate of 5 600 m^3/d by three 50%-capacity electrically driven pumps that supply the following peak daily load requirements:

- General surface operations require 2000 m^3/d;
- The used-fuel packaging plant requiring an additional 900 m^3/d;
- Underground operations are estimated to require 1200 m^3/d;
- Two 100%-capacity electrically driven pumps are used to supply the rock-crushing plant demand of 1000m^3/d; and
- The water-treatment plant requires about 500 m^3/d to meet domestic and demineralized water requirements.

The water-treatment plant has the capacity to clarify about 500 m^3 of process water per day and supply this treated domestic water via the domestic-water storage tank at peak loads of up to 50 L/s to meet the following daily load requirements:

- A total estimated workforce of about 900 people, at an assumed requirement of 200 L/d per person, requires about 180 m^3/d distributed between surface (162m^3/d) and underground (18 m^3/d);
- Decontamination operations at the used-fuel packaging plant requires about 8m^3/d;
- The concrete-batch plant requires about 10 m^3/d for disposal-room concrete operations and about 34 m^3/d for 1 or 2 days per month during placement of room seals;
- The buffer/backfill mixing and compaction plant requires about 150 m^3/d for operations at 16-h/d;
- Domestic water is further treated to make demineralized water. Demineralized water is made available to the used-fuel packaging plant surge storage pools to make up for evaporation losses estimated at 6 m^3/d.

Collected wastewater can be classified as either nonradioactive or potentially radioactive, depending upon the source. Nonradioactive water can further be classified as run-off, sewage water and water that requires settling. Radioactively contaminated water is collected and treated in the packaging plant prior to reuse or release. A system of drains, pipes and pumps brings the collected water to ponds sized for each type of water. The rock-crushing-plant water and the underground service water are collected in separate ponds because of the different geographical locations and the potential for contamination of the underground service water. Collected sewage water from the surface and underground is pumped to the sewage-treatment plant, which then forwards it into the sewage-holding pond before discharge. It is possible to recycle some of the water, particularly the rock-crushing plant settling water, with a minimum capital investment and reduce the overall water supply needs of the disposal facility.

3.5.3 Electrical-Power Distribution System

The electrical-power distribution system provides reliable electrical power throughout the disposal facility. The distribution system comprises the switchyard connected to a regional grid and a network of transformers, power lines, power cables and circuit breakers. These form the primary- and secondary-distribution systems that supply the buildings and the main power loads. The power distribution system comprises lower voltage transformers down to 3-phase, 600 V and 120 V, bus ducts, power cables, motor starters, distribution panels and cable trays, as required by the various plant loads and detailed configuration. The total estimated load for the disposal facility design is 22 MW: 7 MW is required for underground operations and 15 MW is required by the surface facilities. Approximately 4 MW of standby power is provided as back-up power should the primary-power supply be disrupted.

3.5.4 HVAC and Compressed Air Supply

Heating, ventilation and air conditioning (HVAC) units are provided, as required in each building, to maintain comfortable environmental conditions for workers.

Compressed air serves primarily the used-fuel packaging plant and the underground infrastructure. The compressed air flow required for underground operations is quite small because most underground equipment is either diesel or electrically powered. The compressed-air system also supplies breathing air to the used-fuel packaging plant. For breathing air, all compressed air contaminants such as oil mist, dust and moisture are removed by refrigerated dryers and filters. The breathing air is humidified on location at the time of use.

3.5.5 Administration Offices, Security and Fire Protection

Offices are provided for managerial and administrative workers in the administration building. The site is provided with facilities and security guards to protect the workers, the property and, in particular, to protect spent fuel and other nuclear materials from sabotage or diversion. Security is achieved by an unfenced, supervised, outer exclusion area and second by the high-security fence around the two potentially radioactive inner zones. Security guards control the entrance and exit of persons and equipment at all times.

The disposal facility is provided with a firehall, a fire truck, a network of hydrants and all other facilities normally associated with a comprehensive fire protection system. Each building is equipped with a fire alarm system, automatic, manual and portable fire extinguishing equipment. Carbon dioxide and dry chemical extinguishing systems are installed where electrical fires pose a hazard. Security guards, also trained as firefighters, monitor the disposal facility around the clock.

3.5.6 Storage and Maintenance Areas

Heated-indoor and outdoor storage areas are made available for storage of materials, equipment, and spare parts throughout the disposal facility. Maintenance facilities are provided in various buildings, primarily in the auxiliary building, to provide service capabilities for nonradioactively contaminated equipment used in surface or underground facilities. Maintenance of radioactively contaminated equipment is carried at the packaging plant.

3.6 SUMMARY

In this chapter, a generic disposal facility is described with a disposal vault that can be located throughout a depth range of 500 m to 1000 m in highly stressed, sparsely fractured rock conditions. On the basis of the stability calculations for an elliptically shaped disposal room, the rooms can be oriented either perpendicular or parallel to the major principal stress direction. A practicable scheme for emplacing the wastes within the bounds of the room perimeter is described; the in-room emplacement method. This in-room design requires that the buffer and dense backfill materials be compacted into blocks for their placement in the rooms. After placed in a disposal room, they form two horizontal chambers to allow the emplacement of the disposal containers in a horizontal fashion. The use of a low-heat, high-performance concrete is introduced to provide a uniform floor for the accurate placement of the blocks and for the alignment of rail track and equipment.

Descriptions of the two disposal containers are provided; a copper-shell, packed-particulate container and a steel-shell-supported copper container. Both containers can be fabricated and filled with used-fuel bundles, sealed and emplaced within a disposal room. The details of the disposal-vault design focuses on the copper-shell, packed-particulate container and its emplacement. However, only minor dimensional modifications to the disposal room and its components would be needed to accommodate the slightly larger steel-shell-supported copper container.

Details of the generic disposal facility, its design and the factors that influenced its design are presented. The design is feasible and practicable, in that the facility is constructible and operable. A rational approach to developing a robust disposal facility design with the flexibility for broad application within the highly stress, sparsely fractured conditions in the Canadian Shield is demonstrated (i.e., the robust vault objective).

4. ENGINEERING A DISPOSAL VAULT FOR A DEPTH OF 750 M IN SPARSELY FRACTURED ROCK

A complete design description for a used-fuel disposal facility has been presented in Chapter 3. The disposal vault is designed to be structurally stable within a depth range of 500 to 1000 m in a sparsely fractured granitic pluton. The disposal vault, shown in Figure 38, and the surface facilities including the used-fuel packaging plant, shown in Figure 50, meet the objective of producing a disposal facility design with a disposal vault at a depth of 750 m in a relatively impermeable, sparsely fractured granite rock mass. This vault design is appropriate for a preferred site in the Whiteshell Research Area (Ophori et al. 1995, 1996, Stevenson 1995,1996).

The siting, construction, operation, decommissioning and closure of a disposal facility would be a complex and large-scale engineering project extending over many decades. The project would progress by discrete stages, each stage having a specific objective. Many sequential, concurrent and overlapping activities would be associated with these stages to support and assist the validation and confirmation of the specific geotechnical conditions of the site, the designs and the high-performance models. One possible set of stages and activities is illustrated in the following section.

4.1 PROJECT STAGES AND ACTIVITIES

The sequence for the implementation of the used-fuel disposal facility follows the stages and activities defined in the previous studies(AECL 1994, Simmons and Baumgartner 1994). The relationship between the stages and activities is shown in Figure 52.

4.1.1 Project Stages

The stages consist of siting, construction, operation, decommissioning and closure. Optional extended monitoring stages may also be included. The stages are shown on the project schedule for the disposal vault at a depth of 750 m (Figure 52).

The Siting Stage would involve developing the siting process, and site screening and site evaluation substages to identify suitable sites(s) for waste disposal. Data would be gathered during site evaluation to develop an understanding of the surface and underground physical and chemical conditions in and around the site(s) to confirm their potential for safe disposal. During the siting stage, preliminary disposal facility designs would be prepared for each site being evaluated. A specific design for the preferred site would be completed and approved prior to deciding to proceed with underground site evaluation(i.e., exploratory excavation). The end point of the siting stage would be a design based on the results obtained from the surface and underground site evaluation studies and approval for construction at the site selected for a disposal facility.

The Construction Stage would involve constructing the infrastructure and surface facilities needed to transport and dispose of nuclear fuel waste, the underground access ways and service areas, and a portion of the underground disposal rooms. Monitoring would continue during this stage.

The Operation Stage would involve receiving nuclear fuel waste transported to the disposal facility, sealing it in corrosion-resistant containers, placing and sealing the containers in disposal rooms, and constructing and preparing additional disposal rooms at a rate consistent with the waste disposal rate. Monitoring would continue during this stage.

The Extended Monitoring Stages, if required, would involve monitoring and assessing conditions in the vault, geosphere, and biosphere between the operation and decommissioning stages (i.e., predecommissioning monitoring) and/or between the decommissioning and closure stages (i.e., postdecommissioning monitoring). Predecommissioning monitoring makes use of the underground access while it is still available prior to disposal-vault sealing in the decommissioning stage. Postdecommissioning monitoring makes use of the surface-based boreholes while they are still available prior to borehole sealing in the closure stage. The durations of these stages are not specified because they depend on agreements with the public and the regulators during the preceding stages of project implementation.

The Decommissioning Stage would involve the decontamination and removal of the surface and subsurface facilities; the sealing of the tunnels, underground service areas, shafts, and underground exploration boreholes; and the return of the surface of the site to a state suitable for public use. Monitoring would continue during this stage.

The Closure Stage would involve the removal of monitoring instruments from any boreholes drilled from surface that could compromise the safety of the disposal vault, the sealing of those boreholes, and the return of the site to a state where safety would not depend on institutional controls (i.e., to a passively safe state). Monitoring could continue beyond closure if desired provided that such monitoring did not compromise the long-term passive safety of the sealed disposal vault.

The major activities associated with the implementation of nuclear fuel waste disposal, public involvement, characterization, design, monitoring, component testing, performance assessment and construction, can span two or more stages and are discussed further in Section 4.1.2.

4.1.2 Project Activities

Project activities can occur concurrently and, generally, can continue through more than one project stage. The general objectives of the various activities are discussed below.

The Public Involvement activity would involve providing information to the public, particularly those communities hosting the disposal facility and transportation system, and then obtaining input from them throughout all stages of nuclear fuel waste disposal. This input would be a major factor in initiating the siting process, establishing the criteria for siting, assessing socio-economic effects, deciding how these effects should be dealt with, and identifying and resolving public issues and concerns. As well, public input would be considered in planning and conducting other activities, such as characterization, design, monitoring and performance assessment.

The Characterization activity would involve the surface and subsurface investigation of regions, areas and sites to determine the conditions in the geosphere, biosphere and human communities. The data obtained would be used for site selection, facility design and performance assessment. Many of the measurement instruments installed for characterization would also be used for ongoing monitoring. Characterization would be a major activity during the siting stage, and would continue at the selected site during the construction, operation, decommissioning and any extended-monitoring stages.

The Design activity would involve the development of designs for the surface facilities and transportation system, and the development of vault designs of increasing detail for each of the sites under investigation throughout the preclosure phase, on the basis of data collected from characterization, monitoring, performance assessment, construction experience, regulatory requirements and public input. Information on specific site geology, hydrogeology and hydrogeochemistry would be used to recommend container geometry, container material and sealing system requirements and alternatives, as well as vault location and layout. Environmental assessments conducted at each of the sites would help develop any constraints on the design needed to provide acceptable performance for the disposal system.

The Monitoring activity would consist of the continuous or intermittent measurement of conditions in the region influenced, or potentially influenced, by the presence of the disposal facility and associated transportation system. Monitoring would be done to determine the baseline conditions and to identify any changes from the baseline conditions. Parameters indicating conditions in the vault, geosphere, biosphere, and human communities would be measured. Monitoring would be initiated early in the siting stage and would be continued until closure. It could also be continued after closure if required by regulators and/or the public.

The Component Testing activity would consist of conducting and analyzing tests to measure the performance of elements of the disposal facility and the associated transportation system. These tests would be initiated during the underground site evaluation substage and could continue through the construction and operation stages. For example, the performance of the container, the sealing materials, and the rock surrounding the excavations could be studied in underground test areas. Prior to the operation stage, heaters could be used to simulate the heat that would be produced by nuclear fuel waste.

The Performance Assessment activity would consist of the evaluation of the functioning of a disposal system or system component in terms of one or more standards and criteria. This would involve evaluating the current and future behaviour of the disposal system or a subsystem on the basis of data obtained by site characterization, monitoring and component testing, improved knowledge and understanding, and the standards and criteria in use at that time.

Performance assessment is equivalent to a safety assessment when the future effects on humans and nonhuman biota are evaluated in terms of safety standards. For a safety assessment, the system of interest depends on the disposal phase being assessed: for the preclosure phase, it is the disposal facility and associated transportation system; for the postclosure phase, it is the closed disposal vault.

Construction, as an activity, would consist of the development, fabrication and assembly of surface and underground installations for the disposal facility and associated transportation system. Construction activities would begin during the siting stage with the development of a support infrastructure for surface and underground evaluation. It would continue in the construction stage with construction of the surface disposal facilities, site installations and underground excavations, and in the operation stage with development and sealing of the disposal rooms. It would continue with disassembly of the facilities and sealing of tunnels, shafts and boreholes during the decommissioning and closure stages.

4.2 SITING STAGE PLAN

The objective of the first stage, the Siting stage, is to obtain permission to commence the construction of a specifically designed disposal facility at a specific site on the Canadian Shield. The siting stage would initially involve site screening and site evaluation.

No decisions have yet been made about the type of siting process that will be applied to select a site to dispose of Canada's nuclear fuel waste. The siting process would likely be developed at the beginning of the implementation of the project in consultation with the public, governments, and regulators. The objective would be to develop an agreed set of principles and procedures for effective and equitable siting of a disposal facility, and to use these to guide the site screening and site evaluation activities (Greber et al. 1994).

4.2.1 Site Screening

The objective of the site screening is to identify a small number of areas that have the characteristics desired for a disposal site, and warrant detailed investigation, within siting regions on the Canadian Shield (Davison et al. 1994). The activities would include analyzing existing regional-scale data, performing some reconnaissance surveys to gather additional data, developing and applying criteria for accepting or rejecting locations and ranking them for further investigation. The selection of the siting regions to be screened could involve a great deal of government and public input.

Preliminary conceptual design work on surface and underground facilities will begin during site screening, primarily to establish the access, utility and infrastructure requirements. These requirements would be considered during site screening to ensure that they could be met at potentially suitable site locations in the areas selected for detailed evaluation. Details of the environmental and vault monitoring program would also be developed, and the plan to incorporate this program into subsequent site evaluation activities would be prepared during site screening.

4.2.2 Surface-Based Site Evaluation

Site evaluation follows from site screening. The objective of site evaluation would be to identify a preferred location for a disposal site and to obtain approval to construct a disposal facility at that site (Davison et al. 1994). The activities would include thorough site characterization, disposal facility design, and performance assessment. Work would first begin at a relatively larger regional scale to identify preferred disposal locations in the broader context of the geological setting, and then in more detail in the area surrounding the location of the preferred site(s). Site characterization would involve airborne and surface investigations and borehole studies first of the regional areas, followed up by studies at smaller areas where potentially suitable sites might exist.

Note that the schedule and activities of the site screening and surface-based site evaluation substages, and those of the closure stage remain identical to that of the reference in-floor borehole emplacement case (Simmons and Baumgartner 1994) and are not discussed in any further detail here. The schedule and activities of the other stages are different due to the difference in the assumptions and parameters described in Section 1 and 2.

4.2.3 Underground Evaluation

Underground evaluation follows the surface-based site evaluation. By that time the surface-based site evaluation program is completed, a preferred disposal site has been selected, and much of the geotechnical characteristics of the preferred site are known and understood to allow for the underground site evaluation. The purpose of the Underground Evaluation substage is

- to gain direct access to the disposal-vault level environment;
- to verify and refine the surface-based evaluation interpretation of site conditions and behaviours;
- to delineate in detail the acceptable areas for waste emplacement;
- to perform geotechnical mapping, characterization and component testing for deriving engineering design values and constraints; and
- to develop final construction and operation designs of the disposal vault and its components.

More details can be found in Simmons and Baumgartner (1994) and Davison et al. (1994). Figure 53 shows the disposal vault layout at this stage. The underground evaluation plan consists of the following major activities (Figure 54):

- upgrade the site infrastructure to perform exploration shaft sinking, tunnelling and underground characterization;
- sink and equip two exploration shafts, with 4.6-m and 3.65-m-finished internal diameters, at the opposite ends of the potential vault area to about 800-m depth.
- construct a service bay area equivalent to 615 m of 4.4-m x 10-m size tunnels.
- conduct about 6000 m of 76-mm and 96-mm-diameter horizontal and subhorizontal exploratory diamond drilling in and around the projected disposal vault horizon;
- excavate 18 340 m of 3.5-m x 4-m exploration tunnels through and around the waste emplacement area;
- conduct an additional 37 000 m of 76-mm and 96-mm-diameter exploratory diamond drilling in and around the disposal vault horizon;

- characterize the geotechnical environment by core and borehole logging and sampling, excavation mapping, borehole sampling and testing, excavation deformation measurements, and geophysical imaging;

- excavate the equivalent of about 2000 m of exploration-sized tunnels and begin rock mass behavioural testing in the component test area;

- conduct appropriate research and development as needed; and

- produce the detailed engineering specifications, plans and safety case for the construction of the disposal facility and apply for a construction licence.

Some of the activities outlined here are sequential in nature (e.g., infrastructure, shaft sinking, initial drilling, tunnelling and component testing), whereas others are parallel activities associated with the sequential activities (e.g., characterization and additional drilling during the excavation process), as shown in Figure 52. Note that about two years of component testing for deriving engineering design values and constraints will be required; this is necessary to develop final construction and operation designs of the disposal vault and its components after excavation of the component test area and prior to the completion of this project stage (Figure 54).

The exploration shafts are located such that they should fit in the plans for the subsequent stages of the implementation. The exploration tunnels and other underground facilities are also located and constructed such that they should be easily adapted to be used as the actual disposal vault elements.

All excavation, drilling and construction activities in this stage are based on 3 shifts/d, 360 d/a. Component testing is assumed to occur over 1 shift/d, 261 d/a.

4.3 CONSTRUCTION STAGE PLAN

After regulatory bodies have granted licences the construction of the full-scale disposal vault can begin. The purpose of the construction is to build all the facilities necessary for the operation of the disposal vault and its components. Provision is made in the design for concurrent excavation during the operation stage. The construction stage plan consists of the following activities (Figure 55):

- upgrade the site infrastructure to perform large-scale shaft sinking, tunnelling and underground characterization;

- construct used-fuel packaging plant and associated facilities;

- refurbish the two exploration shafts, one to serve as the downcast ventilation shaft and the other to serve as one of the upcast ventilation shafts for the balance of the disposal-vault life;

- sink and equip the service shaft, the second upcast ventilation shaft, and the waste shaft to depths of about 800 m, 770 m, and 780 m respectively;

- enlarge 2042 m of the 3.5-m x 4-m central exploration tunnel to serve as the 4.4-m x 10-m central access tunnel, and drive the second 2042-m-long central access tunnel to the same dimensions;

- enlarge 1523 m of the 3.5-m x 4-m tunnels of the service shaft complex to the 4.4-m x 10-m size, and excavate an additional 293 m of the 4.4-m x 10-m size tunnels;

- enlarge 375 m of the 3.5-m x 4-m tunnels of the upcast shaft complex to the 4.4-m x 10-m size, and excavate an additional 315 m of the 4.4-m x 10-m size tunnels;

- enlarge all the remaining 3.5-m x 4-m exploration tunnels to full-size, 4.4-m x 10-m tunnels;

- excavate and equip a buffer/backfill and block compaction plant; and, at surface, the rock-crushing plant;

- construct the permanent headframes and install the hoisting plants and fans;

- excavate 96 disposal rooms (i.e., 1.5 panels), 3.3-m x 7.6-m in size, for a total of 22 848 m in preparation for the operation stage;

- characterize the geotechnical environment by core and borehole logging and sampling, excavation mapping, borehole sampling and testing, excavation deformation measurements, and geophysical imaging;

- complete rock mass behavioural tests in the component test area;

- prepare the access tunnels with services, such as rail and ventilation tubing; pour concrete for floors, and install rails;

- prepare 64 disposal rooms (i.e., one panel) with services, such as rail and ventilation tubing; pour concrete for floors, and install rails;

- commission all the underground equipment and produce detailed operating procedures;

- conduct appropriate research, as needed, and development,; and

- prepare the detailed safety case for the operation of the disposal facility and apply for an operating licence.

Some of the activities outlined here are sequential in nature (e.g., infrastructure, shaft sinking, initial drilling, and tunnelling), whereas others are parallel activities associated with the sequential activities. Figure 56 show the vault layout at the end of the construction stage.

As in the Underground Exploration stage, all excavation, drilling and construction activities are based on 3 shifts/d, 360 d/a. Component testing is assumed to occur over 1 shift/d, 261 d/a.

4.4 OPERATION STAGE PLAN

The purpose of operation is to emplace and seal the disposal containers in the disposal vault. There are three major concurrent groups of operational activities occurring during the Operation stage:

- Room Excavation: Drilling and blasting, muck removal and ground support installation;

- Room Preparation: Installation of concrete floors, installation of the rail, and other support services; and

- Container Emplacement: Installation of backfill blocks, placement of buffer blocks, placement of upper backfill, emplacement of disposal containers, and placement of sand.

After all the disposal containers are emplaced in a room, the room bulkhead is constructed.

The three major activities are scheduled to take place concurrently, such that when containers are being emplaced in one panel on one side of the central access tunnel, the room preparation and room excavation takes place in another panel on the other side of the central access tunnel. As in the borehole emplacement case, two separate ventilation system are maintained; one for the radiological operations (i.e., container emplacement) and the other for nonradiological operations (i.e., room excavation and room preparation). The cycle times for the three major operational activities can be modified by adjusting the crew sizes such that all three activities take about the same amount of time to complete.

Sufficient rooms are excavated and prepared during the construction stage such that at the start of the operation stage, the crews for these activities are at staggered locations and operate in a non-interfering mode. Specifically, the rooms in Panel A have been excavated and prepared, while the rooms in one half of Panel B have been excavated and are ready for room preparation. At the beginning of the operation stage room, block placement and waste emplacement starts in Panel A, while room preparation takes place in the excavated rooms of Panel B, and room excavation takes place in the remaining half of Panel B (see Figure 56).

The principle of segregating the radiological operations (i.e., buffer and waste emplacement) from the nonradiological operations (i.e., room preparation, borehole drilling, and room backfilling) is maintained. The central access tunnels are twinned to reduce the potential for traffic accidents, particularly with radioactive loads (i.e., cask cars with disposal containers) and to provide a secondary route for worker and material transport. The emplacement operations retreat from the upcast shaft complex towards the service shaft complex. Thus the work progresses from a potentially contaminated area towards a clean area with a fresh air source, enhancing the environment for workers.

At the end of each cycle when the waste emplacement operations are completed in a room section, each functional activity is moved to the next sequence of rooms in the opposite panel, across from the central access tunnels. The general operation activity plan is given in Figure 57.

4.5 DECOMMISSIONING STAGE PLAN

The purpose of the decommissioning stage is to

- decontaminate and remove all the related underground support works;

- backfill and seal the balance of the disposal vault, which consists of all of the exploratory and instrumented boreholes drilled from underground, the tunnels, the service and upcast shaft complexes, the buffer and backfill preparation plant, the block compaction plant, the component test area and the shafts;

- decontaminate and dismantle the used-fuel packaging plant and associated facilities;

- dismantle all surface buildings and associated facilities;

- dismantle and remove the rock-crushing plant, the concrete-batch plant, and the shaft headframes, fans and collarhouses; and

- dismantle and remove all surface infrastructure including rock crushing plant, the concrete batch plant, the buffer/backfill preparation plant, and the shaft headframes, fans and collarhouses.

The start date of this stage is dependent on whether or not an extended monitoring stage is required. The decommissioning stage plan consists of the following activities (Figure 58):

- remove instruments from all underground boreholes and seal each borehole;

- backfill the upcast shaft complex, installing sealing bulkheads at strategic locations;

- ream, with raise boring machines, the waste and upcast shafts to remove the concrete linings and any wall rock degradation, re-equip each shaft with services and stagings, and backfill the shafts including the installation of shaft sealing bulkheads at strategic locations;

- backfill the central access tunnels, installing tunnel sealing bulkheads at strategic locations;

- dismantle and backfill the component test area, the service shaft complex, and the material/truck storage area, installing sealing bulkheads at strategic locations;

- ream the service shaft to remove the concrete lining and any wall rock degradation, re-equip the shaft and backfill, installing shaft sealing bulkheads at strategic locations; and

- prepare the safety case and apply for approval to release the site.

As in the Operation stage, all sealing and decommissioning activities are scheduled for 3 shifts/d, 360 d/a. Tables 14 and 15 provide estimates for the total quantities of materials and their components that are removed (i.e., excavated granite) and emplaced within the disposal vault throughout its lifecycle.

4.6 CLOSURE STAGE PLAN

The purpose of the closure stage is to

- remove instruments from all surface boreholes and backfill and seal each borehole, except those that are needed for monitoring in the postclosure phase. Note that any monitoring in the postclosure stage must not compromise the passive safety of the facility;

- recondition the site surface to a state suitable for public use with the provision that subsurface use be restricted; and

- prepare the safety case and apply for approval to release the site.

The activities and related data for this stage is the same as described for the borehole case. Closure work is assumed to occur over 1 shift/d, 261 d/a.

4.7 SUMMARY

In this chapter, a disposal facility with a disposal vault at a depth of 750 m is described that is suitable for the sparsely fractured rock conditions. On the basis of the disposal-room stability calculations in Chapter 3, the disposal rooms can be oriented either perpendicular or parallel to the major principal stress direction at this site, whichever is convenient for its implementation. All the major activities required for its implementation throughout its lifecycle are described. The described disposal facility satisfies the favourable vault site objective.

5. ENGINEERING A DISPOSAL VAULT FOR A DEPTH OF 500 M IN A PERMEABLE, MODERATELY FRACTURED ROCK

A complete design description for a used-fuel disposal facility was presented in Chapter 3. The disposal vault is designed to be structurally stable within a depth range of 500 to 1000 m in a sparsely fractured granitic pluton. However, the disposal vault design needed at a depth of 500 m should be suitable for a permeable, moderately fractured granite rock mass.

5.1 GENERAL REQUIREMENTS

The key requirements from a disposal vault design perspective are to provide (i) a disposal-room design consistent with the rock properties and ambient in situ stresses anticipated for a depth of 500 m in moderately fractured rock, and (ii) a vault layout consistent with the geosphere condition where a low-angle fault transects the waste emplacement area of the 500-m-deep vault.

5.1.1 Assumed Rock Mass Properties and Ambient In Situ Stress Conditions

AECL has limited experience in excavating in moderately fractured rock, other than sinking shaft and drilling boreholes through such rock. Without a data base for moderately fractured rock properties, the rock properties and the in situ stresses for sparsely fractured granite are assumed as the basis for this specific design at a depth of 500 m in moderately fractured rock. However, the engineering implications for this assumption are understood and a discussion for adapting to moderately fractured rock conditions is presented in Section 5.3.

5.1.2 Prescribed Geosphere Conditions

The geosphere model, which includes the disposal vault at a depth of 500 m, has a 20-m-thick, low-angle (18^o) fault that transects the waste emplacement area of the vault. Therefore, two waste emplacement areas are defined, one section on the hangingwall side and one on the footwall side of the fault. The total plan area of the vault is 1900 m wide by 2153 m long (i.e., approximately 4 km^2) (Stanchell et al. 1996). The horizontal distance between the two vault sections where no waste is emplaced is 375 m long to allow for the transecting fault zone and a rock pillar on each side of the fault at the vault level. The effective or net waste emplacement area (i.e., the sections where disposal containers are emplaced) is about 3.4 km^2.

5.2 DISPOSAL-VAULT DESIGN SUITABLE FOR THE GEOSPHERE CONDITIONS

The basic disposal-vault design assumed for this application is one suited to rock that is highly stressed and sparsely fractured as discussed above. The layout of the vault is modified to adapt to the geosphere condition of a transecting fault.

5.2.1 Disposal-Room Design for a Vault at a Depth of 500 m

The disposal-room designs presented in Chapter 3 are used for this disposal vault because of the lack of detailed rock mass stress and property measurements in moderately fractured rock (Section 5.1.1.).

5.2.2 Layout Design for a Vault at a Depth of 500 m

The layout for the disposal vault at a depth of 500 m is shown in Figure 59. The design principles outlined in Section 3.3 are retained, providing the separation of radiological and nonradiological operations. A total of 8 panels is also retained although two panels are divided by the transecting fault. In 6 of the panels, the room lengths are shortened to about 230 m to provide uniformity in block and waste emplacement operations. The two panels straddling the fault have the disposal rooms shortened to about 110 m, again to provide uniformity in block and waste emplacement operations.

Two additional 1900-m-long perimeter access tunnels run parallel to the fault to provide easy access to the fault zone for characterization purposes during the underground evaluation substage (Section 4.2.3). The total quantity of waste to be disposed in the 500-m-deep vault is reduced to about 4.3 million fuel bundles, as shown in Figure 59.

5.3 ENGINEERING IMPLICATIONS OF MODERATELY FRACTURED ROCK CONDITIONS ON DESIGN

Qualitatively, it can be argued that the moderately fractured rock mass of this application would have reduced strength compared with sparsely fractured rock, which would likely depend on the extent and frequency of fracturing. The presence of a higher degree of fracturing would suggest that the ambient horizontal in situ stresses should be lower because of a likely reduction in the Young's modulus of the large-scale rock-mass. The vertical stress component should remain the same because it is derived from lithostatic loading (i.e., the density and depth of overlying rock).

5.3.1 Sensitivity Analysis of Design Parameters

An examination of the role that in situ stress, rock strength and Young's modulus play in a disposal-room stability analysis is illustrated by using the scoping methodology described in Section 3.2.4.1. As noted in Section 3.2.4.2, this tool proved to be useful for rapidly assessing the stability of excavations under ambient and thermal conditions. This simple analytical approach assumes that the moderately fractured rock mass effectively behaves in a homogenous, linear elastic manner. The assumption may not be fully accurate depending on the degree of fracturing and the roughness and alteration characteristics of the fractures, as well as the orientation of the fractures with respect to the excavation. However, the linear elastic, continuum approach may have some validity if stress transmission across fractures is not severely influenced by the fracture shear strength.

As a basis for an analysis of the significance in a reduction in the in situ stress conditions, the values for the average ambient in situ stress condition in the Canadian Shield are selected from Herget and Arjang (1991). The average stress condition is illustrated in Figure 60 and can be compared with the high stress condition shown in Figure 7. By using these average stresses, the scoping analysis shows that the disposal-room shape developed for the sparsely fractured rock conditions at a depth of 500 m is suitable for the reduced stress conditions, with the longitudinal axis of the room oriented either perpendicular (Figure 61a) or parallel (Figure 61b) to the major principal stress direction. These figures can be compared with Figures 12 and 13 for the high stress conditions analyzed in Chapter 3. Note that the rock mass strength design criteria (σ_{EX} = 100 MPa and σ_{TM} = 150 MPa, Section 2.1.7.3) and the Young's modulus (i.e., 60 GPa, Table 5) are unchanged, in this particular analysis, from their values used in Chapter 3.

If the rock strength design criteria are reduced (e.g., arbitrarily reduced by 20 MPa from the values in the above paragraph, σ_{EX} = 80 MPa and σ_{TM} = 130 MPa) for the average in situ stress condition (Figure 60), as may be expected for good to very good quality (Barton et al. 1974) moderately fractured rock, the disposal-room shape is still suitable, at a depth of 500 m, for both room orientations with respect to the major principal stress direction (Figures 62a,b). The Young's modulus remains at 60 GPa in this case.

Finally, when the Young's modulus is reduced (e.g., arbitrarily reduced by 15 GPa from the value in the above paragraph, E = 45 GPa), together with the reduced strength criteria and the in situ stresses, the result is an improvement in room stability for the thermal condition (Figures 63a,b). The excavation condition is unchanged. The improved thermal stability is a direct result of the reduction in the thermally induced expansionary stresses. In the scoping analysis, the thermally induced horizontal stress is calculated by using a simple analytical solution for a uniformly heated, thick horizontal plate under plane stress conditions (i.e., no lateral strain) (Timoshenko and Goodier 1970) as follows:

$$\Delta\sigma = \alpha E \Delta t/(1-\nu) \tag{9}$$

where $\Delta\sigma$ = change in horizontal stress (MPa);
α = coefficient of thermal expansion ($^{o}C^{-1}$);
E = Young's modulus (MPa);
Δt = change in temperature (^{o}C); and
ν = Poisson's ratio.

This brief and simple sensitivity analysis illustrates that the specific shape of the disposal room, developed for sparsely fractured rock conditions, may be suitable for the moderately fractured rock conditions assumed in the geosphere model illustrated in Stanchell et al. (1996). A discontinuum analytical approach may be better suited for this type of analysis when the actual conditions of a moderately fractured rock site are known. The wide range of fracturing possibilities for moderately fractured rock precludes a meaningful stability analysis at this conceptual stage. The intent of the this analysis is to explore, with the reader, a potential implication for siting a disposal vault in moderately fractured rock and that this may be a reasonable possibility from a structural stability perspective.

5.3.2 Additional Design and Construction Considerations

If the qualitative assumptions on rock mass strength and in situ stresses are valid, the reaction for disposal-room design is to reduce the aspect ratio of the elliptical room cross section to make the design more compatible to the conditions for providing greater margins of stability. This can be accomplished by changing room width or height. The disposal-room stability envelopes may be applied to achieve the desired result following a site investigation program.

For example, the reduction of the disposal-room width depends upon a number of factors, both beneficial and detrimental. Beneficial effects would be a reduced requirement for ground support to control the potential for wedge failures in the excavation perimeter. Wedges of rock, formed by the intersection of bounding fracture planes, have the potential to loosen or dislodge from the roof and walls of the excavation. Compared with sparsely fractured rock, moderately fractured rock could require more extensive ground support measures, such as regular patterns of rock bolting, and the use of shotcrete is likely to be required. However, reduction of room width may require the closer spacing of disposal containers across the width of the room. Although some margin is available for the resulting increase in the temperatures at the container outer surfaces, the effect on the radiation shielding provided by the interstitial web of buffer material between the two containers would have to be considered.

An increase in room height has two potential effects: it can increase the quantities of excavated and sealing materials (i.e., increase cost) and it can increase the container surface temperature because of the increased thickness of the low thermal-conductivity buffer and backfill materials. However, it also increases the amount of sorption sites in the buffer and backfills for radionuclides released from the disposal containers.

Ultimately, the aspect ratio may be reduced to 1.0, providing the horizontal stress component reduces sufficiently (i.e., Figure 61b and Figure 63b). In this case, the disposal room could be amenable to the machine boring of stable circular tunnels, and disposal containers may be emplaced along the tunnel axes similar to that proposed by NAGRA (1985) and SKB (1992b).

Another factor that may influence the design is the potential need for grouting as a construction expedient. The increased permeability of the moderately fractured rock mass relative to sparsely fractured rock may hamper the construction process (i.e., excavation of rock and placement of concrete floors, precompacted blocks and wastes). The need for construction grouting is dependent on the rate of groundwater ingress, the quantity of stored groundwater, the hydraulic conductivity of the local rock mass and the nature of the specifics of the activities which may differ in their ability to tolerate groundwater influx. If the depletion of the stored groundwater is rapid and the continual resupply is low, the influx may reduce to a low continuous value that is amenable to diversion or evaporation into the ventilating air and grouting may be unnecessary. However, if the groundwater influx rate at any locale is intolerable for the specific construction activity, crews may need to be augmented to deal with the specific grouting situation. Grouting can have a positive effect on the local stability of moderately and highly (i.e., faults) fractured rock by providing shear-resisting fracture infilling material and dilation control.

5.4 SUMMARY

A disposal facility with a disposal vault at a depth of 500 m is described that is suitable for the sparsely fractured rock conditions and, following a brief sensitivity analysis, appears likely to be suitable for moderately fractured rock conditions. Furthermore, the geosphere condition of a fault transecting the disposal vault level can be accommodated in the vault layout by reducing vault capacity while maintaining the separation of radiological and nonradiological operations. On the basis of the disposal-room stability calculations in Chapter 3 for sparsely fractured rock conditions, and on the sensitivity analysis for moderately fractured rock conditions, the disposal rooms can be oriented either perpendicular or parallel to the major principal stress direction at this site and depth, whichever is convenient for its implementation. Coincidentally, the in situ stress measurements at depth in the Underground Research Laboratory together with the structural geological inferences of thrust faulting suggest that the long axes of the disposal rooms are oriented parallel with the major principal stress direction for the geosphere model being used for moderately fractured rock (Stanchell et al. 1996).

6. CONCLUSION

Three disposal vault designs using the in-room emplacement method have been described to meet three objectives. One objective was to provide a rational approach to developing a robust disposal facility design with the flexibility for broad application within the sparsely fractured rock in the Canadian Shield. A generic design a of disposal vault was presented that could be applied over a depth range of 500 to 1000 m. A scoping and detailed design process for distributing the thermal loading of the waste within the disposal vault and a disposal room and for the thermal-mechanical stability analyses was demonstrated.

The second objective was to produce a suitable disposal facility design for a depth of 750 m at a favourable vault location in the highly stressed, sparsely fractured granite of the Lac du Bonnet batholith in the Whiteshell Research Area. The requirements of this case were within the specific design parameters of the generic disposal vault design. No new design or analyses were needed. The details for the implementation of the specific design were illustrated for all stages of the project lifecycle.

The third objective was to produce a specific disposal facility design with appropriate engineered barriers that are suitable for a permeable host-rock condition in which advection is the dominant contaminant transport process. A specific vault design is provided in Chapter 5 for the simulation of the long-term performance of the design for the specific site characteristics, and the generic vault design (Chapter 3) provides the specific discussion of the design of the engineered barriers which accommodate the permeable host-rock condition. Arguments for adapting the specific disposal-room design to differing geosphere conditions (i.e., reduced rock mass strength, Young's modulus and ambient in situ stresses) by changing the disposal-room dimensions and aspect ratio (i.e., the ratio of the major ellipse axis to the minor ellipse axis) to suit the specific conditions were provided in Chapter 5 for the proposed design of a disposal vault at a depth of 500 m in moderately fractured rock. The design process would remain the same, only the specific design parameters would change. The vault layout in Chapter 5 also introduced the flexibility available in the design to include the effect of a fault transecting the waste emplacement area.

In all cases, the specific engineered barriers are the copper-shell, packed-particulate disposal container and its alternate, the steel-shell-supported, copper disposal container, and the clay-based buffer and backfill materials in the forms of precompacted blocks of buffer and dense backfill and pneumatically placed light backfill. The flexibility of the engineered barriers is illustrated by their choice (i.e., the disposal container or its alternate), by their composition (i.e., fraction and type of clay and aggregate in the buffer and backfill mixes) and by their thickness to suit the radionuclide retention needs. The shapes of the precompacted blocks can be adjusted to suit the size of the disposal container and the size and shape of the disposal room to accommodate the specific rock mass strength and in situ stress conditions, as discussed in Chapter 5. The precompacted blocks provide a second service, the radiation protective shielding for the workers from the emplaced disposal container.

The designs are feasible in that they are constructible and operable and are applicable for a much wider range of geosphere conditions than that of a disposal vault with the in-floor borehole emplacement method. They are constructible in that the excavations can be constructed in the elliptical configuration, as is currently being demonstrated in the Underground Research Laboratory, and in that the concepts of precompacted block formation and their placement and the pneumatic placement of clay-based materials have been performed at experimental facilities (Gray 1993). They are operable in that concepts for rail haulage, ventilation, concrete placement and materials handling are well established in civil and mining practise and that the radiation protection measures for waste handling and emplacement are designed for the presence of workers in the room and to meet recent proposed regulations (AECB 1991).

The in-room emplacement method using copper-shell, packed-particulate disposal containers provides an alternate means of nuclear fuel waste disposal in geological media under more adverse conditions than specifically addressed in the Environmental Impact Statement (AECL 1994) for the in-floor borehole emplacement method. Specifically, the in-room emplacement method can accommodate a wider range of in situ stress and geological conditions than in-floor boreholes because the in-room method eliminates the stress concentrations generated by the complex borehole intersection geometry at the excavation perimeter (Simmons and Baumgartner 1994). The elliptical shape of the disposal rooms also minimizes and redistributes the stress concentrations associated with flat floors and right-angle corners at the floor-wall intersections. The thermal and thermal-mechanical analyses demonstrate that the elliptical disposal-room shape can be applied to the extreme in situ stress range measured in the Canadian Shield for the 500- to 1000-m-depth range assumed in the Nuclear Fuel Waste Management Program. The aspect ratio and the dimensions of the disposal room and the waste arrangement within the room can be rearranged to accommodate different ambient in situ stress and rock mass conditions.

On the basis of the experimental program at the Underground Research Laboratory, specific rock mass strength design criteria have been proposed for both the excavation conditions and the thermal conditions assumed to be generated under disposal vault conditions. These strength design criteria are based on the dependence of the stress path on the rock mass physical properties at the excavation perimeter as the conditions in the vault evolve. The general elliptical and oval disposal-room shapes proposed in this report, at the time this report is being written, are being excavated and tested in the Underground Research Laboratory. The proposed strength criterion for excavation conditions appears to be reasonable. Continuance of this excavation testing to and beyond the strength criterion should establish the accuracy of the premise. Future physical testing under thermal conditions is also needed to finalize the entire premise.

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*** Unrestricted report available from Ontario Hydro, 700 University Avenue, Toronto, Ontario M5G 1X6.

**** Unpublished report available from the principal author at Ontario Hydro Technologies, 800 Kipling Avenue, Toronto, Ontario M8Z 5S4.

TABLE 1

COMPARISON BETWEEN DISPOSAL FACILITY STUDIES

Item	Emplacement Method	
	In-Floor Borehole	In-Room
Depth	500 to 1000 m	500 to 1000 m
Quantity of Waste	10.1 million bundles	4.3 to 5.8 million bundles
Ambient *In situ* Stresses	Average for Canadian Shield	Upper range for Shield
Disposal Container	Particulate-packed, Ti-shell 633-mm-diameter, 2246-mm-long	Particulate-packed, Cu-shell 860-mm-diameter, 1189-mm-long
Rock Strength		
- Excavation	σ_c = 190 MPa, m=17.5, s=0.19	σ_c = 100 MPa, m=16.6, s=1
- Thermal	σ_c = 190 MPa, m=17.5, s=0.19	σ_c = 150 MPa, m=25, s=1
- Rock Web	σ_c = 110 MPa, m=30, s=1	Not Applicable
- Factor of Safety	2 (avg. for rock web and pillars)	1 (at excavation perimeter)
Young's Modulus	35 GPa	60 GPa
Container Heat Output	297 W	330 W
Maximum Container Surface Temperature	100°C	90°C
Minimum Buffer Thickness	250 mm	500 mm
High-Performance Concrete	Only at disposal-room bulkhead	Entire floor of disposal room
Maximum Worker Radiation Dose Rate	5 mSv/a	2 mSv/a

TABLE 2

OVERALL DISPOSAL-CONTAINER MATERIAL PROPERTIES

Thermal Conductivity (W/(m·°C))	1.3
Specific Heat (kJ/(kg· °C))	0.86
Density (Mg/m^3)	5.22
Young's Modulus (GPa)	0.2
Poisson's Ratio	0.1

TABLE 3

USED-FUEL DISPOSAL CONTAINER HEAT OUTPUT AS A FUNCTION OF TIME

Time Out-of-Reactor (a)	Container Heat Output (W)
10	329.6
11	319.2
12	310.4
13	302.7
14	295.8
16	283.6
18	272.8
20	262.9
30	220.8
40	187.4
50	160.4
60	138.7
70	121.2
80	107.0
90	95.7
100	86.5
135	66.0
160	58.0
200	49.7
300	42.6
500	35.1
1 000	24.6
2 000	17.1
5 000	12.6
10 000	9.1
20 000	5.4
35 000	3.0
50 000	1.9
100 000	0.6
150 000	0.3
250 000	0.2
600 000	0.2
1 000 000	0.2

TABLE 4

PRINCIPAL IN SITU STRESSES AT SELECTED DEPTHS

Vault Depth (m)	Maximum Principal Stress σ_1 (MPa)	Intermediate Principal Stress σ_2 (MPa)	Minimum Principal Stress σ_3 (MPa)	Stress Ratio σ_1/σ_3	Stress Ratio σ_2/σ_3
500	60.6	45.0	13.0	4.7	3.5
750	62.8	47.2	19.5	3.2	2.4
1000	65.0	49.4	26.0	2.5	1.9

TABLE 5

GRANITE ROCK MASS MATERIAL PROPERTIES

Thermal Conductivity (W/(m·°C))	3.0
Specific Heat (kJ/(kg·°C))	0.845
Mass Density (Mg/m^3)	2.65
Young's Modulus (GPa)	60
Poisson's Ratio	0.25
Coefficient of Thermal Expansion (10^{-6}/°C)	10

TABLE 6

COMPONENTS OF COPPER-SHELL, PACKED-PARTICULATE DISPOSAL CONTAINER

Component	Mass (kg)	Volume (m^3)
Copper Shell	1028	0.115
Stainless Steel Basket	314	0.040
Used-Fuel Bundles	1709	0.176
Particulate	526	0.337
Total	3577	0.667
Elemental Uranium in Initial Fuel	1363	

TABLE 7

SPECIFICATION FOR CLAY-BASED MATERIALS COMPOSITION

Composition	Material		
	Buffer	Dense Backfill	Light Backfill
Clay:			
Type	Bentonite	Lake Clay/Bentonite	Bentonite
Content (dry wt%)	50	25/5	50
Aggregate:			
Type	Silica Sand	Crushed Granite	Crushed granite
Content (dry wt%)	50	70	50

TABLE 8

SPECIFICATION FOR BUFFER MATERIAL

Parameter	Silica Sand	Bentonite	Buffer
Moisture Content (wt%)			
Liquid limit	N.A.*	210	135
Plastic limit	N.A.	45	18
Air dried	0.5	6 - 7	3 - 5
Minimum Specific Surface (m^2/g)	N.A.	590	290
Predominant Clay Mineral	N.A.	Sodium montmorillonite	Sodium montmorillonite
Cation Exchange Capacity (Meq/100 g)	N.A.	»80	»40
Particle Size Distribution (mm)	(% Passing)	(% Passing)	(% Passing)
2	92-100	100	96-100
1.5	86-98	100	92-100
1	76-88	100	88-94
0.5	52-64	100	76-82
0.25	24-35	100	62-68
0.1	1-11	100	51-56
0.06	0-6	100	50-53
0.002	0-2	70	35-36
Minimum Dry Density (Mg/m^3)	N.A.	N.A.	1.67
Compaction Water Content (wt%)	N.A.	N.A.	17-19

* Not Applicable

TABLE 9

SPECIFICATION FOR CLAY-BASED SEALING MATERIALS

Selected Physical Properties	Material: Buffer	Dense Backfill	Light Backfill
Dry Density when Placed (Mg/m^3)	>1.67	>2.1	>1.2
Potential Swelling Pressure (kPa)	800 - 2000	<50	100 - 200
Free Swell (%)	80 - 175	<10	20 - 50
Hydraulic Conductivity when Placed (m/s)	$<10^{-11}$	$<10^{-10}$	10^{-10}
Drying Shrinkage (%)	<2	2 - 5	<5
Effective Angle of Internal Friction (°)	14	32	14
Degree of Saturation when Placed (%)	80	80	33
Moisture Content when Placed (wt%)	18	8	15

TABLE 10

SEALING MATERIAL PROPERTIES

Property	Low-Heat High-Performance Concrete	Fractionated Silica Sand	Buffer	Dense Backfill	Light Backfill
Thermal Conductivity (W/(m·°C))	1.80	0.45	1.70	2.00	0.70
Specific Heat (kJ/(kg·°C))	0.90	0.82	1.38	1.19	1.30
Bulk Density (Mg/m^3)	2.43	1.45	1.97	2.27	1.38
Young's Modulus (GPa)	50	0.10	0.10	0.20	0.01
Poisson's Ratio	0.10	0.10	0.10	0.10	0.10
Coefficient of Thermal Expansion (10^{-6}/°C)	10	N.A.*	N.A.	N.A.	N.A.
Swelling Pressure (kPa)	0	0	800 - 2000	<50	100 - 200

* Not Applicable

TABLE 11

TANGENTIAL STRESSES AND FACTORS OF SAFETY FOR NEAR-FIELD CASES (after Tsai 1995)

Location	Time (a)		Vault Depth and Room Orientation to Major Principal Stress Direction (Room aspect ratio = 2.3) 500 m ($\perp : \sigma_1$)*	500 m ($= : \sigma_1$)+	750 m ($\perp : \sigma_1$)	750 m ($= : \sigma_1$)	1000 m ($\perp : \sigma_1$)	1000 m ($= : \sigma_1$)
Roof	0	σ_1	95.5	67.8	93.4	65.7	91.3	63.6
		FS#	1.05	1.47	1.07	1.52	1.09	1.57
Roof	15	σ_1	134.7	107.1	132.6	105.0	130.5	102.9
		FS	1.11	1.40	1.13	1.43	1.15	1.46
Roof	100	σ_1	145.8	118.5	143.7	116.1	141.6	114.0
		FS	1.03	1.27	1.04	1.29	1.06	1.32
Wall	0	σ_1	32.3	54.6	53.1	66.6	87.0	101.3
		FS	3.09	1.83	1.88	1.50	1.15	0.99
Wall	15	σ_1	50.6	72.8	58.4	80.6	83.6	97.7
		FS	2.97	2.06	2.57	1.86	1.79	1.53
Wall	100	σ_1	50.4	72.7	58.2	80.5	67.6	88.3
		FS	2.97	2.06	2.58	1.86	2.22	1.70

* Perpendicular to major principal stress direction
\+ Parallel to major principal stress direction
\# Factor of safety

TABLE 12

DISPOSAL-VAULT EXCAVATION EQUIPMENT (without allowance for spares)

Equipment	Construction Stage	Operating Stage
Three-Boom Hydraulic Drill Jumbo	4	4
Load-Haul-Dump (LHD) 6-Mg Capacity	6	5
Truck 24-Mg Capacity	8	7
Explosive Truck	2	2
Scissors-Lift Truck (for roof bolting)	2	2

TABLE 13

VAULT MATERIALS HANDLING IN SERVICE SHAFT DURING OPERATION STAGE

Material	Handling Method in Service Shaft	Average Handling Capacity per Shift (Mg)	Total Quantity Handled (Gg)
Crushed Granite[(1)]	Skip	41	1 023
Bentonite Clay	Pneumatic Transfer Pipe	38	958
Glacial-Lake Clay	Pneumatic Transfer Pipe	11	269
Silica Sand[(2)]	Skip	26	656
Concrete[(3)]	Transfer Pipe	33	825
Excavated Granite	Skip	194	4 865

Notes: 1. Crushed granite used in dense and light backfill.
2. Silica sand used in buffer and annular sand.
3. Concrete is used at a higher rate during construction of bulkhead seals.

TABLE 14

TOTAL QUANTITY OF MAJOR MATERIALS IN ENTIRE DISPOSAL VAULT

PRODUCT	DRY QUANTITY (Gg)	MOIST QUANTITY (Gg)
Concrete	1 260	1 320
Buffer	1 400	1 660
Dense Backfill	2 570	2 770
Light Backfill	1 100	1 270
Annular Sand	30	N.A.*
Container Materials	150	N.A.
Total	6 650	N.A.
Containers (numbers)	80 900	N.A.
Used-Fuel (bundles)	5 825 000	N.A.
Used-Fuel as Uranium (Gg U)	110	N.A.

* Not Applicable

TABLE 15

TOTAL QUANTITY OF MATERIAL COMPONENTS IN ENTIRE DISPOSAL VAULT

COMPONENTS	QUANTITY (Gg)
Granite	2 910
Cement	50
Pozzalans	160
Superplasticizer	10
Silica Sand	1 210
Sodium-Bentonite Clay	1 380
Lake Clay	640
DLP Copper	80
Stainless Steel	30
Glass Bead	40
Used-Fuel Bundles	140
Water in Materials	680
Total	6 710
Excavated Granite	8 930

EIS 4-ES.4

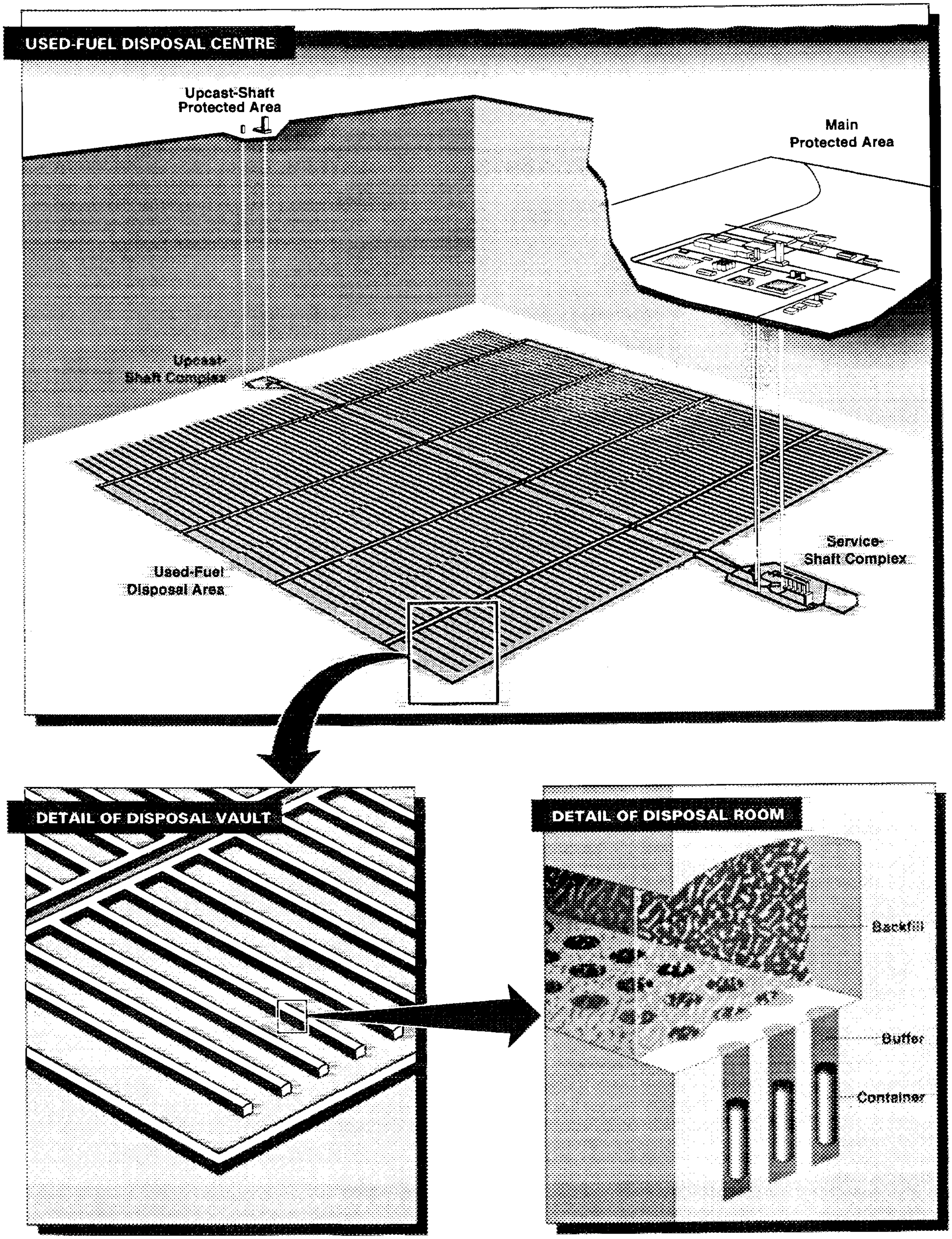


FIGURE 1: Disposal Room Showing the Borehole Emplacement of Titanium-Shell Disposal Containers (Simmons and Baumgartner 1994)

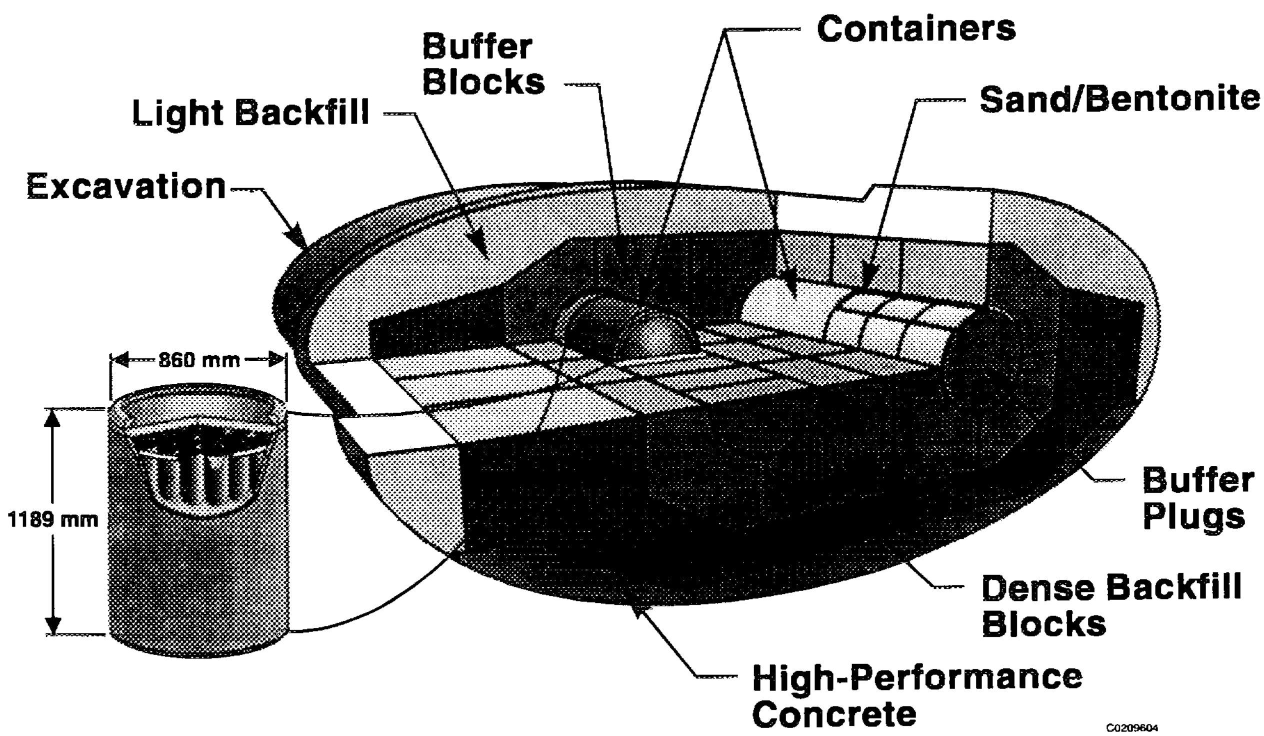


FIGURE 2: Disposal Room Showing the In-Room Emplacement of Copper-Shell Disposal Containers

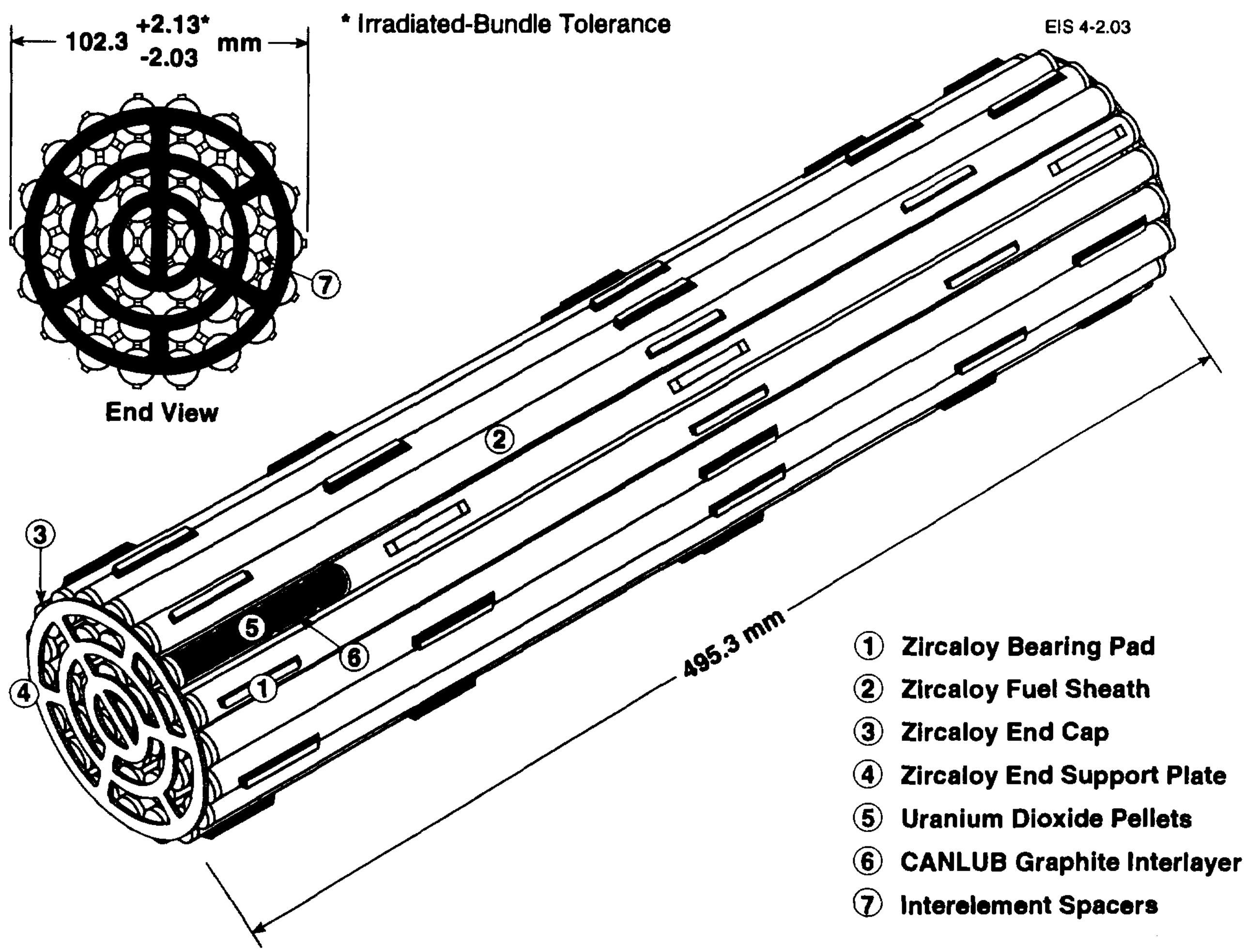
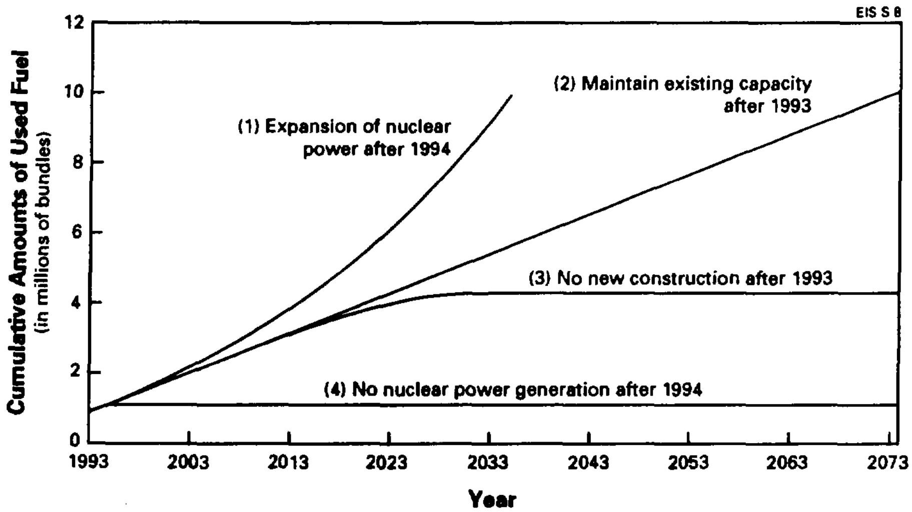


FIGURE 3: Typical CANDU Fuel Bundle for Bruce Nuclear Generating Station (after AECL CANDU et al. 1992)



ASSUMPTION FOR PROJECTIONS

1. Expansion of Nuclear Power after 1994: Canadian nuclear generating capacity would increase by 3% per year after 1994. By the end of 2035, a total of about 10 million bundles of used fuel would have been produced in Canada.

2. Maintain Existing Capacity after 1993: Canadian nuclear generating capacity existing as of 1993 March 31 would be maintained. By the end of 2073, a total of about 10 million bundles of used fuel would have been produced.

3. No New Construction after 1993: Canadian nuclear power reactors existing as of 1993 March 31 would be operated for 40 years, but no new reactors would be constructed. By the end of 2033, when the last reactor would be shut down, a total of about 4.3 million bundles of used fuel would have been produced.

4. No Nuclear Power Generation after 1994: Canadian nuclear power reactors existing as of 1993 March 31 would be shut down as of 1995 January 1, by which time a total of about 1.1 million bundles of used fuel would have been produced.

FIGURE 4: Used-Fuel Projections in Canada (AECL 1994)

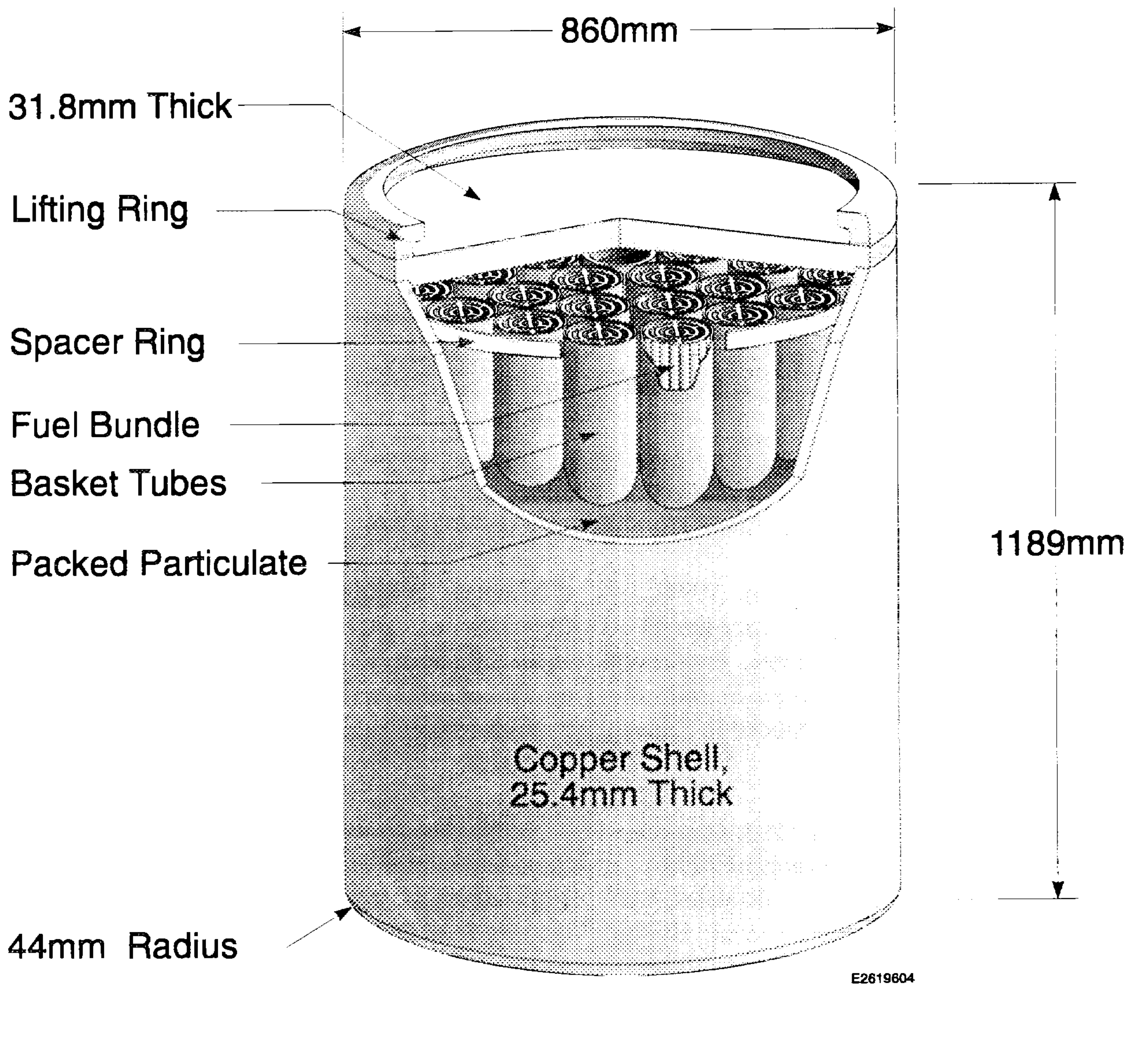


FIGURE 5: Copper-Shell, Packed-Particulate Used-Fuel Disposal Container

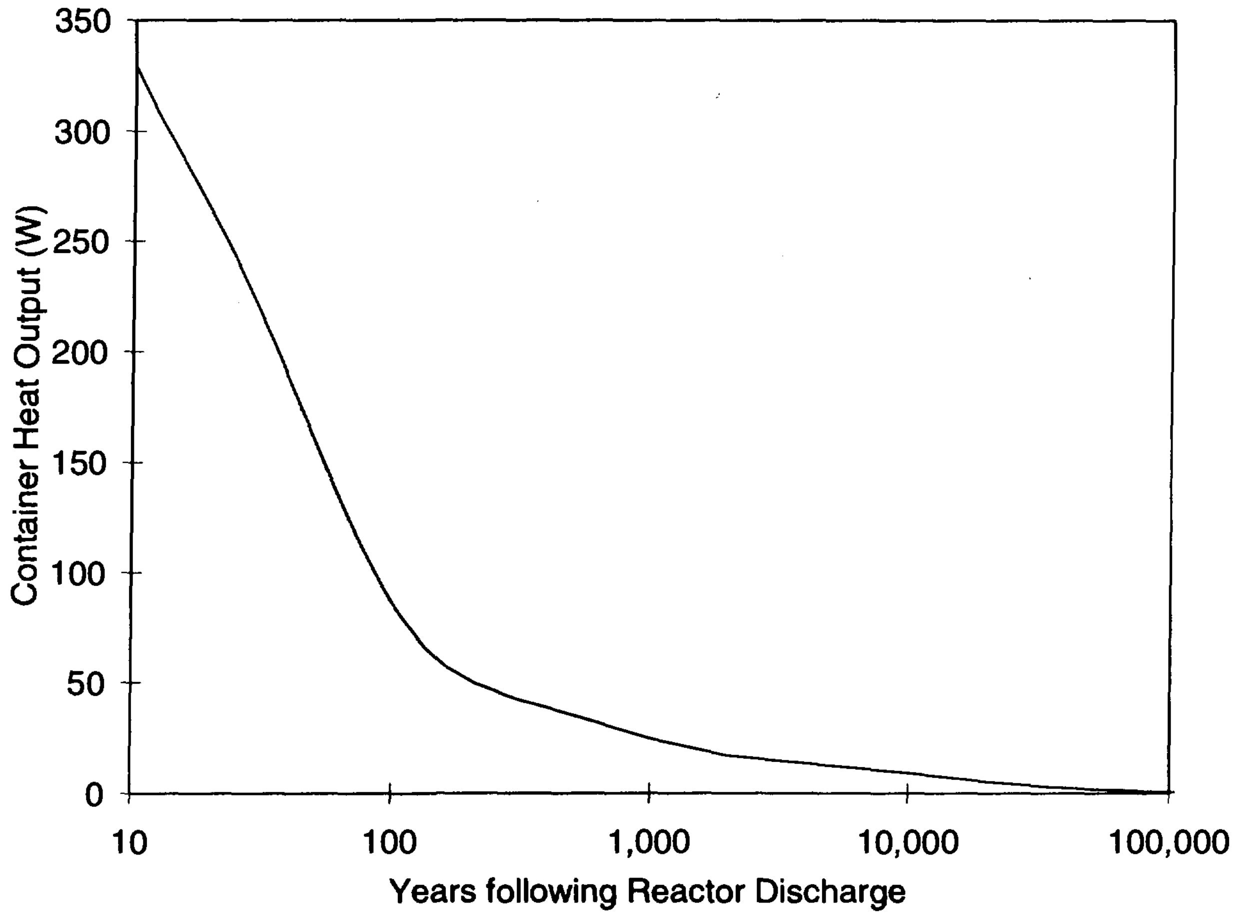


FIGURE 6: Used-Fuel Disposal-Container Heat Output as a Function of Time

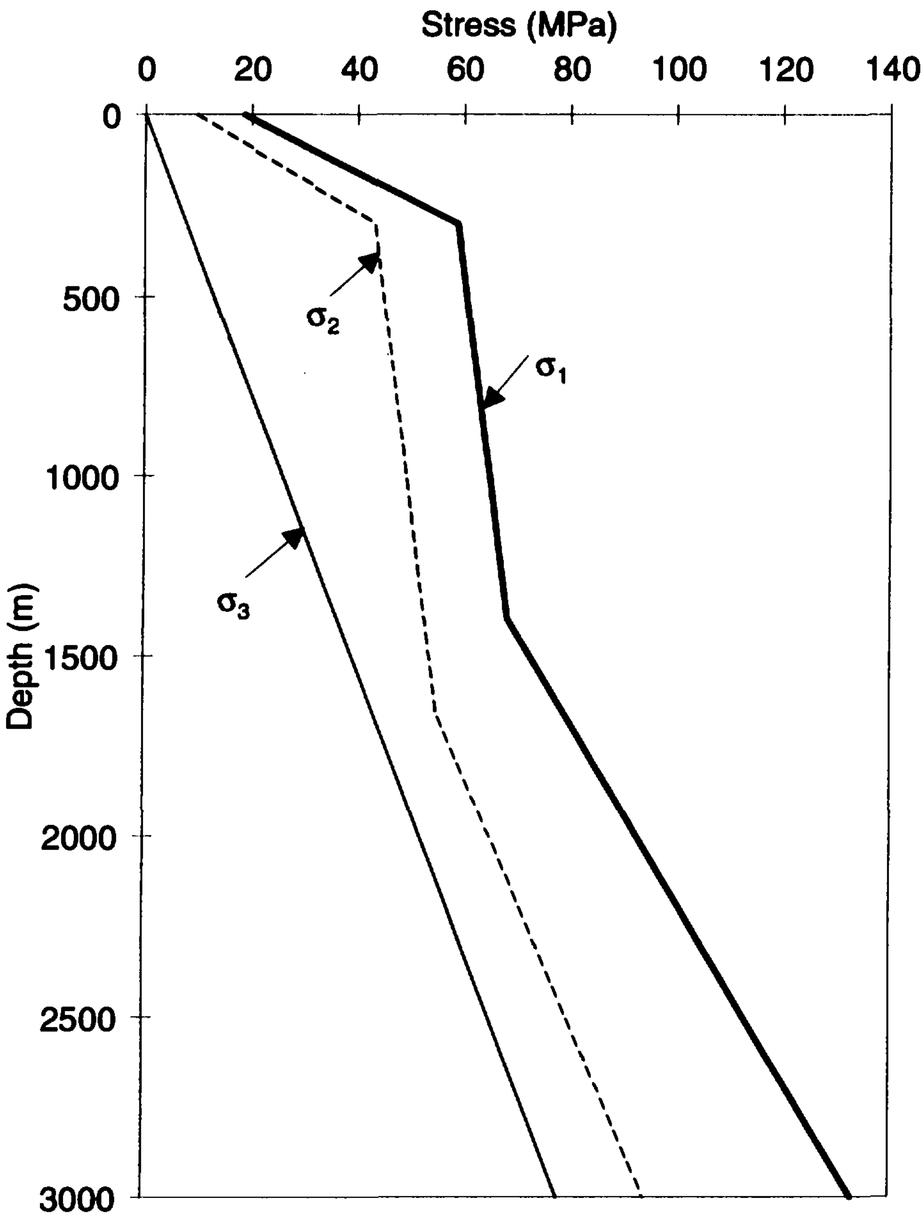


FIGURE 7: Ambient In Situ Stress State

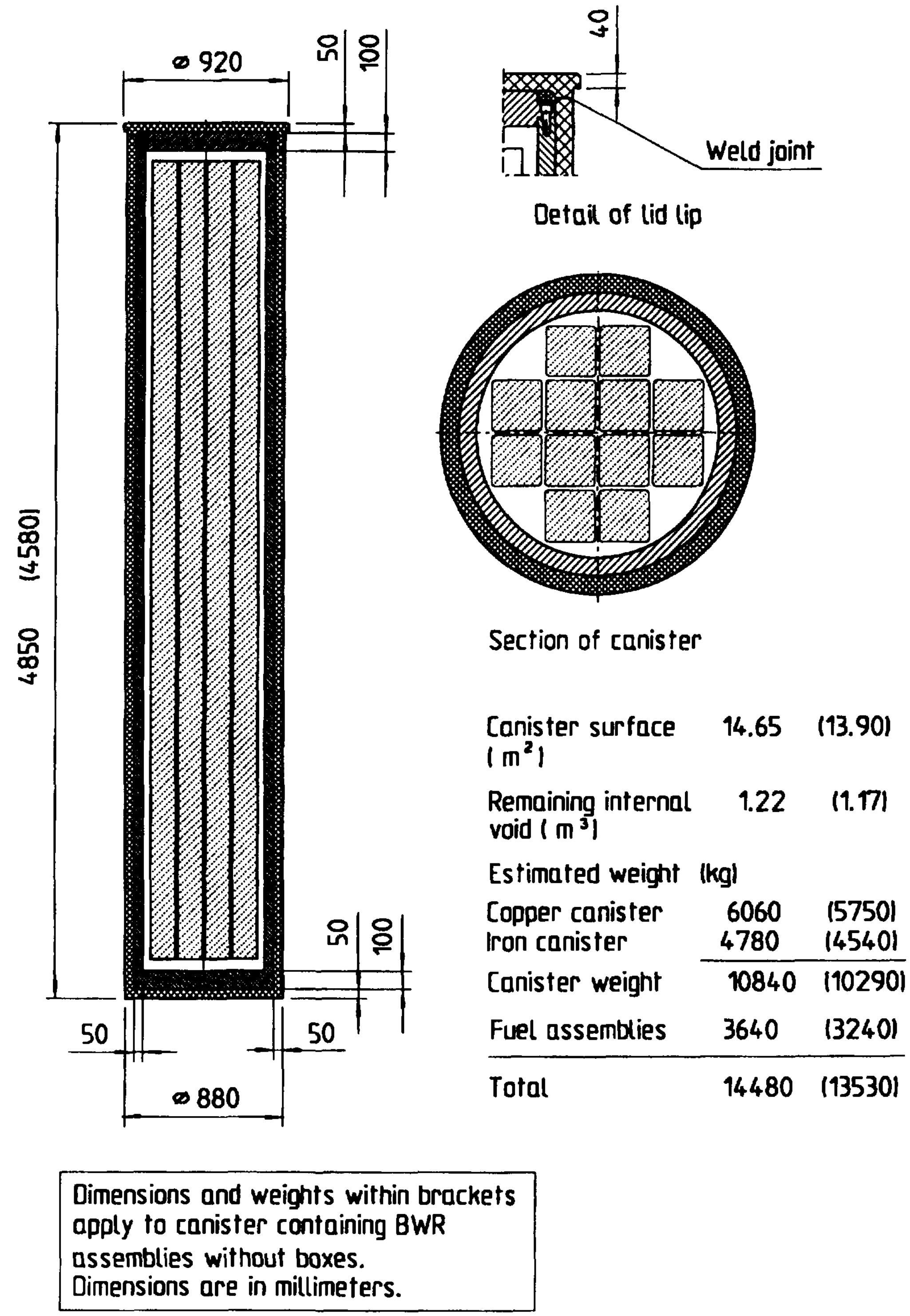


FIGURE 8: Swedish Copper/Steel Canister Alternative with BWR Assemblies (SKB 1992b - Fig B1-2a)

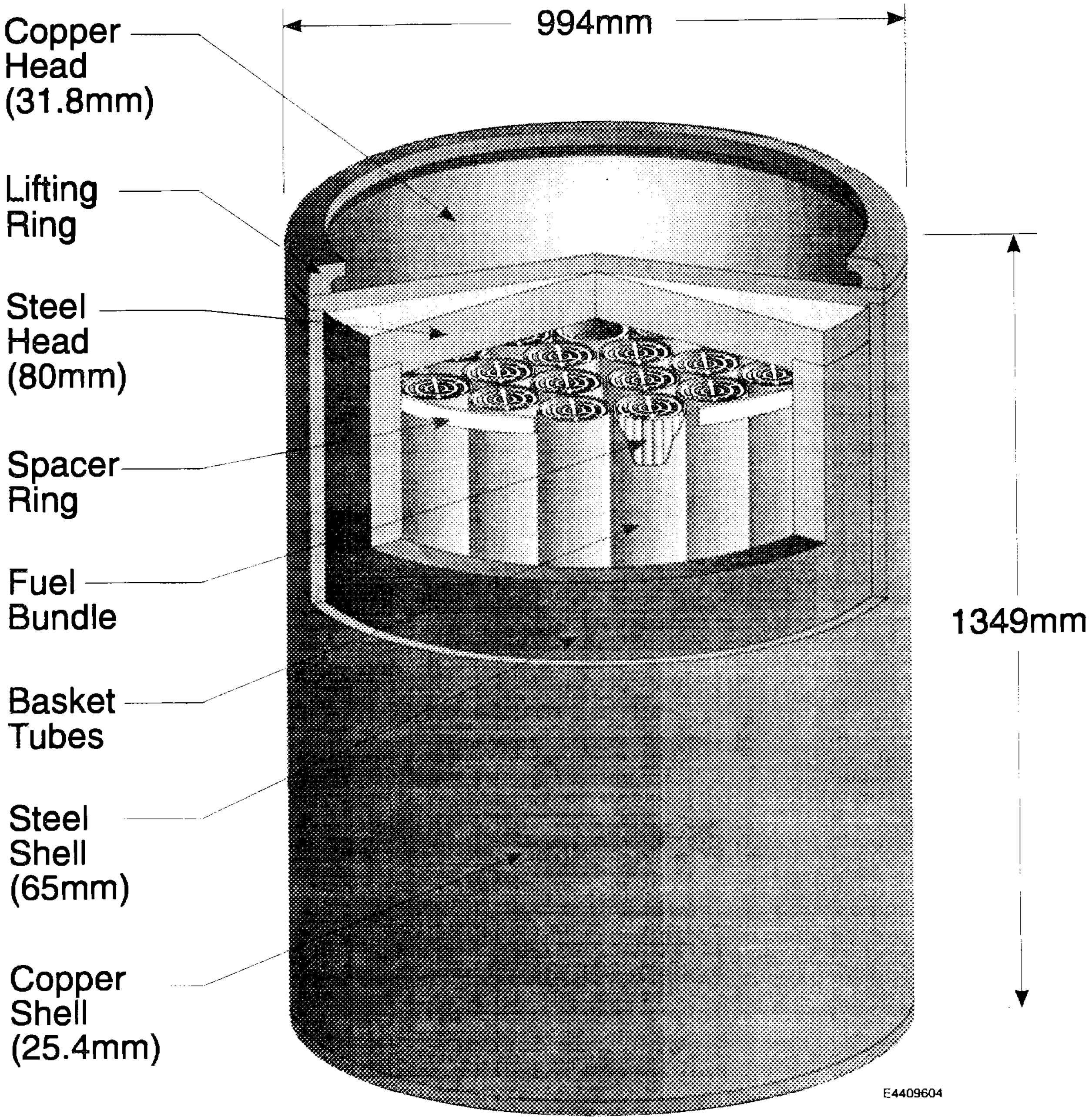


FIGURE 9: Steel-Shell-Supported Copper Disposal Container

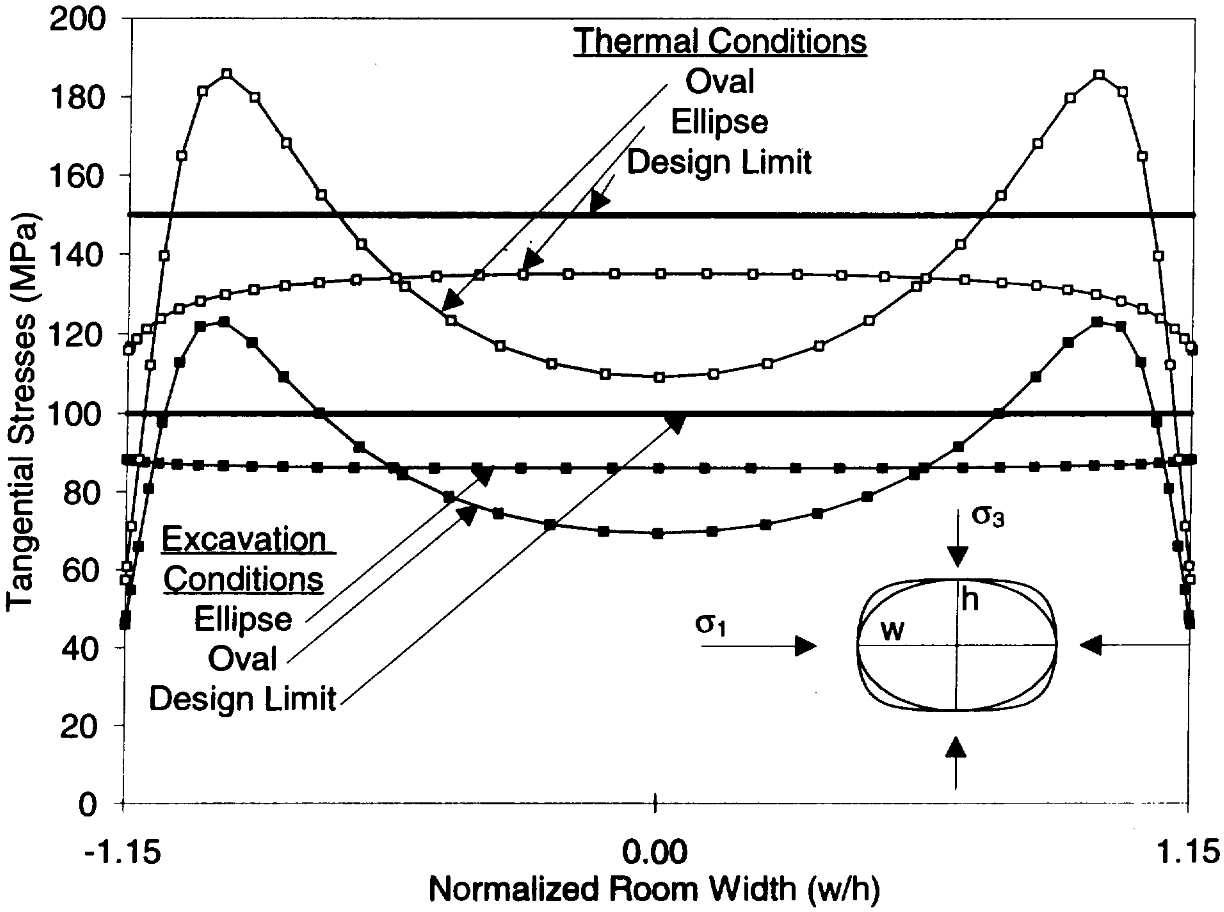


FIGURE 10: Tangential Stress along the Upper Excavation Perimeter at a Depth of 1000 m for Oval- and Elliptical-Shaped Excavations under Excavation and Thermal Conditions (The point 0 represents the centre of an excavation with an aspect ratio of 2.3)

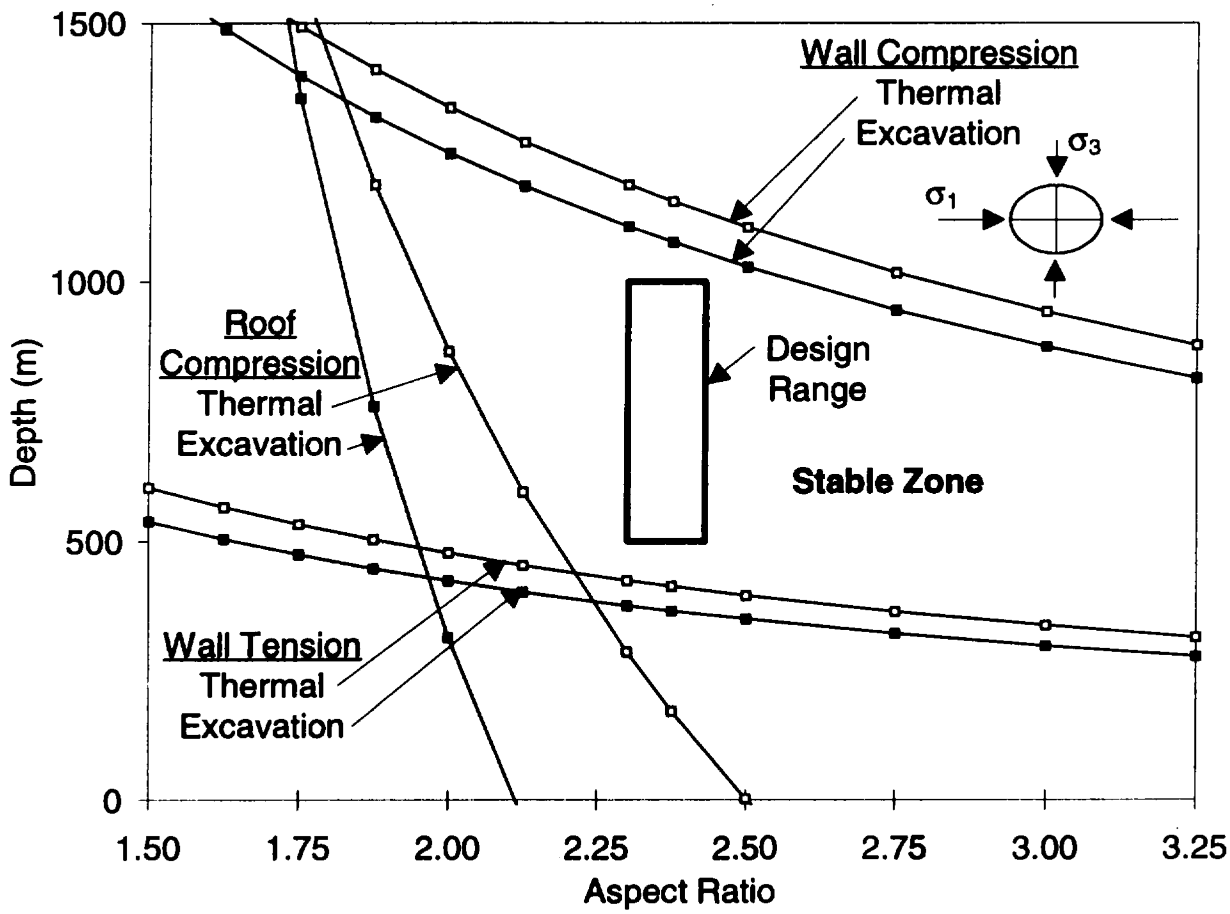


FIGURE 11: Disposal-Room Stability Design Envelope for Scoping Analyses of Both Excavation and Thermal Conditions (Baumgartner et al. 1995)

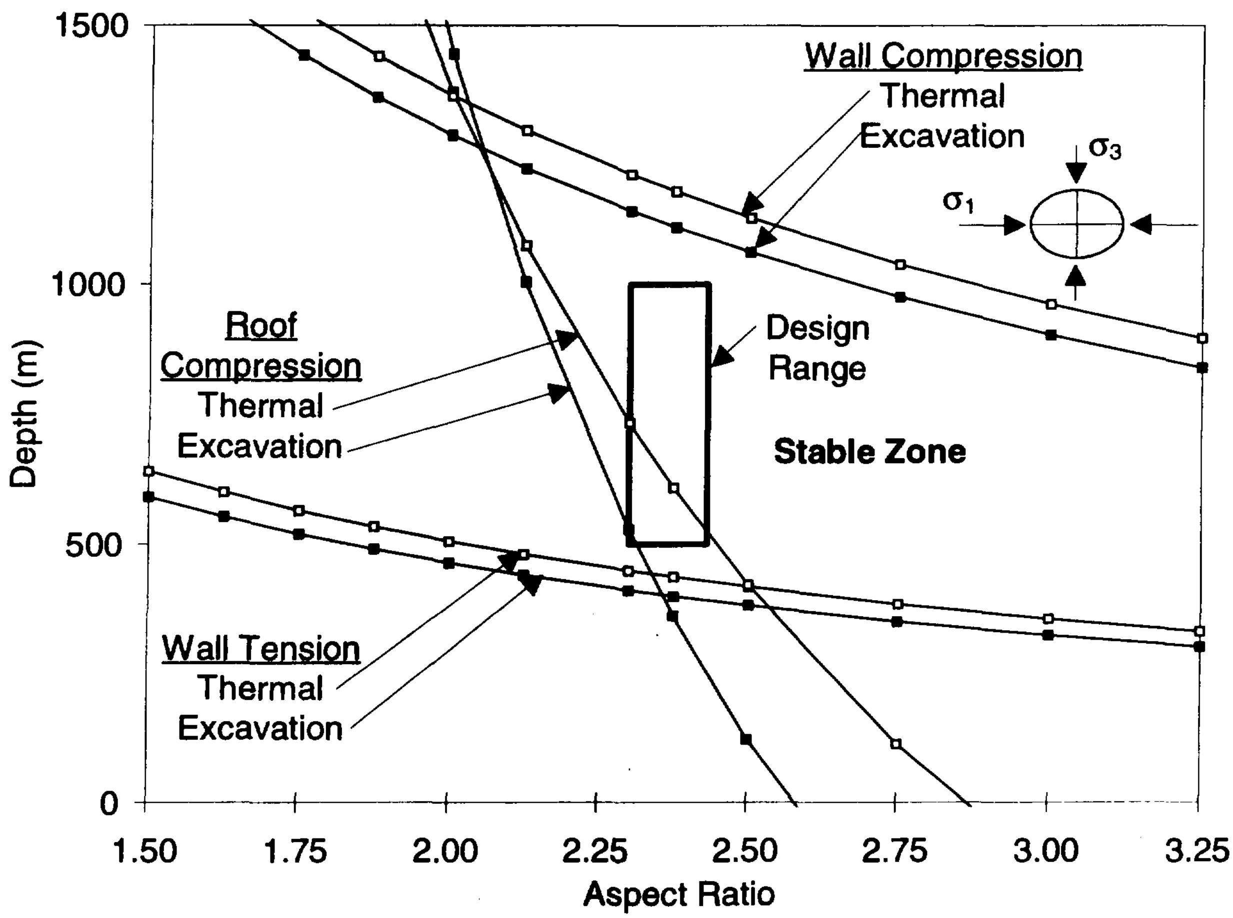


FIGURE 12: Revised Disposal-Room Stability Design Envelope for Scoping Analyses of Both Excavation and Thermal Conditions (Room longitudinal axis oriented perpendicular to the major principal stress direction)

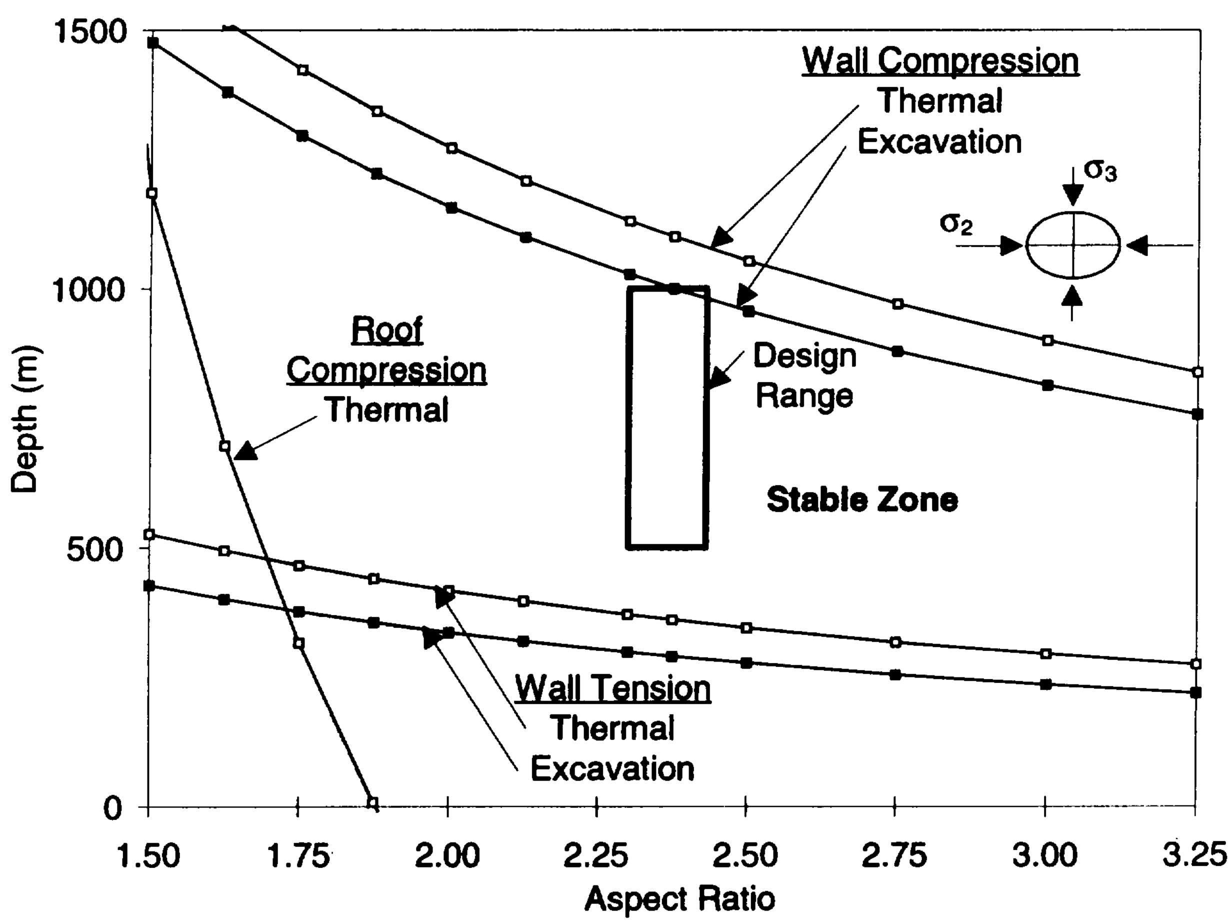


FIGURE 13: Disposal-Room Stability Design Envelope for Scoping Analyses of Both Excavation and Thermal Conditions (Room longitudinal axis oriented parallel to the major principal stress direction)

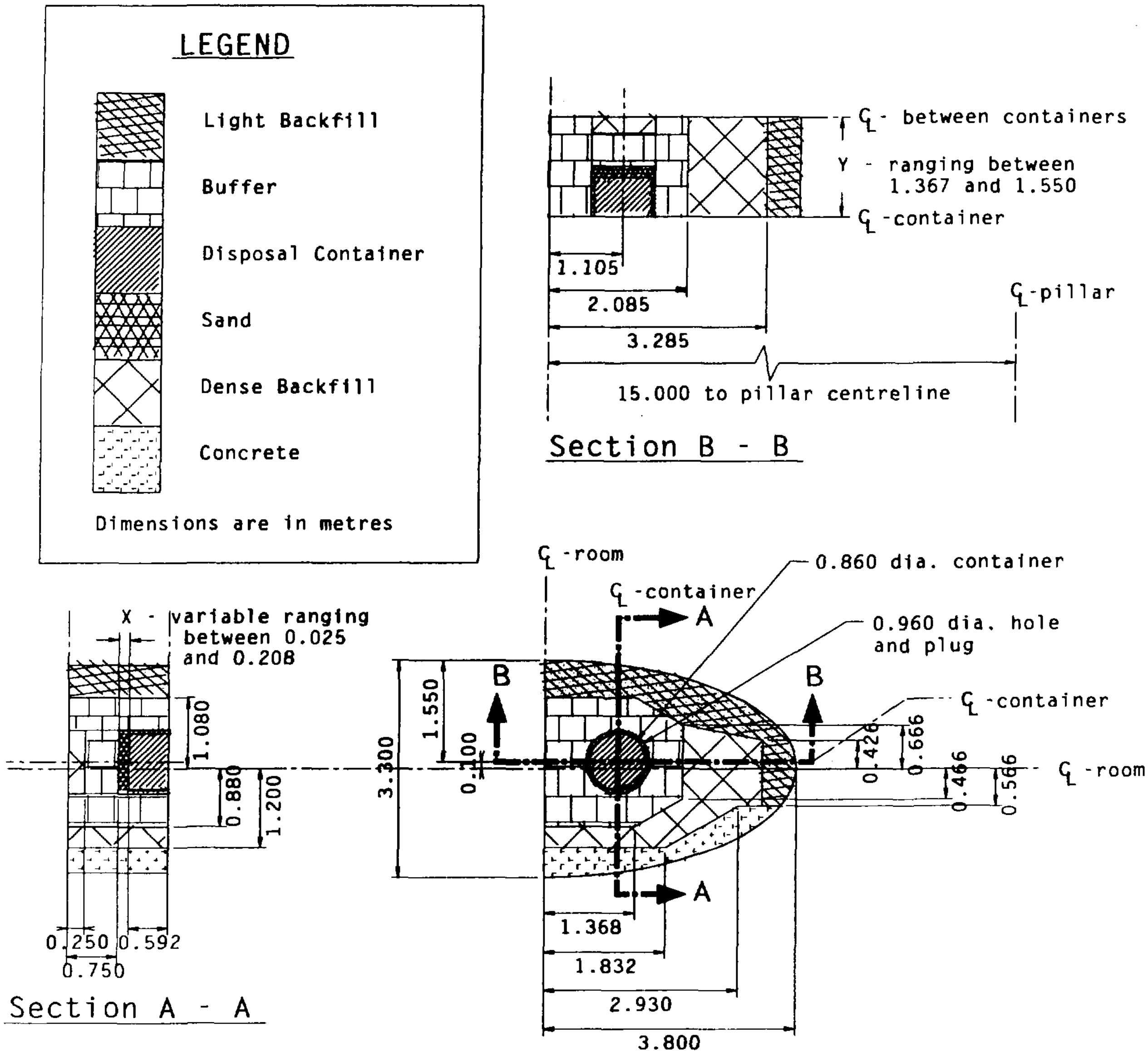


FIGURE 14: Sectional Views of the Disposal Room in the Plane of the Unit Cell (Wai and Tsai 1995)

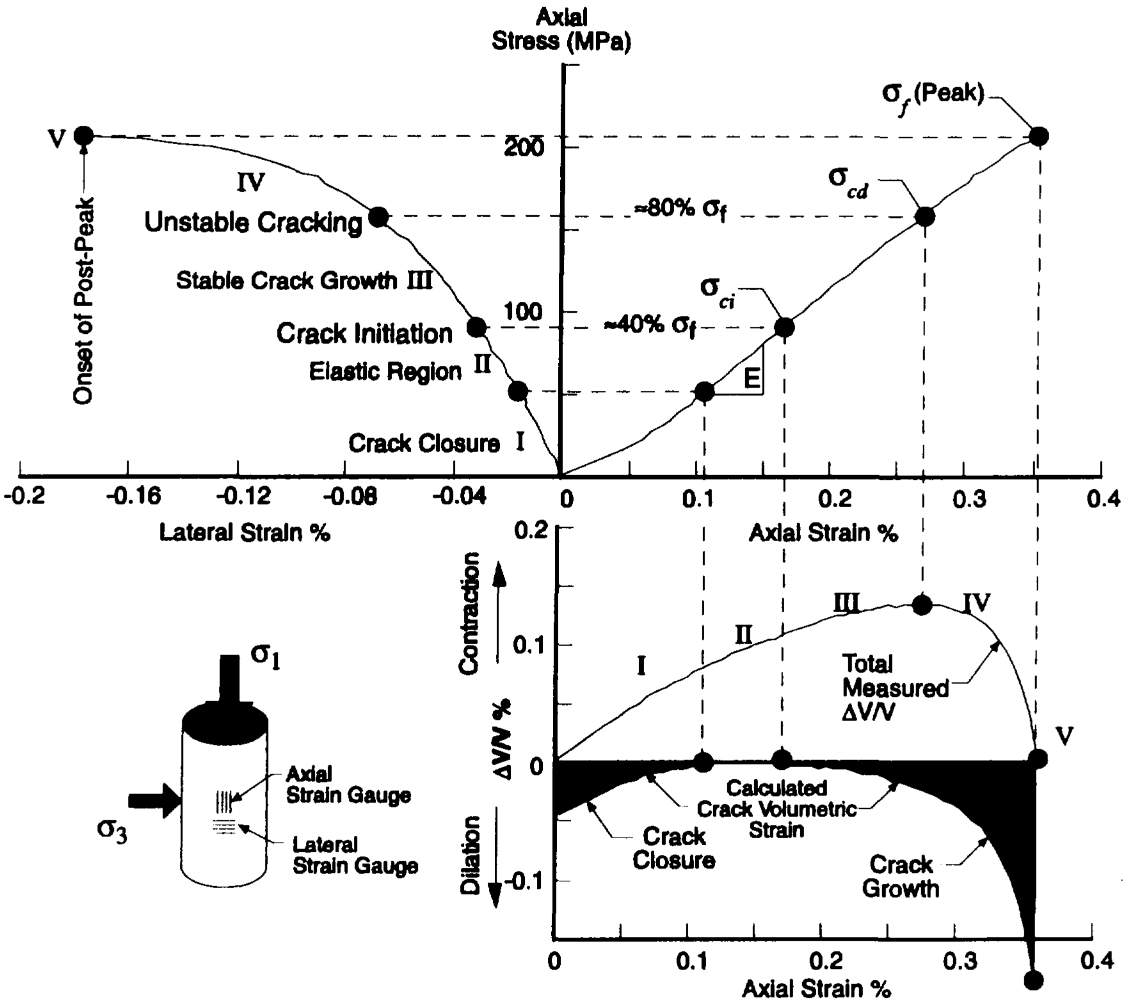


FIGURE 15: Stress-Strain Diagram Obtained from a Single Uniaxial Compression Test for Lac du Bonnet Granite (after Martin 1993) (Definitions for crack initiation (σ_{ci}), crack damage (σ_{cd}) and peak strength (σ_f). Axial and lateral strains are measured; volumetric strain and crack-volumetric strain are calculated).

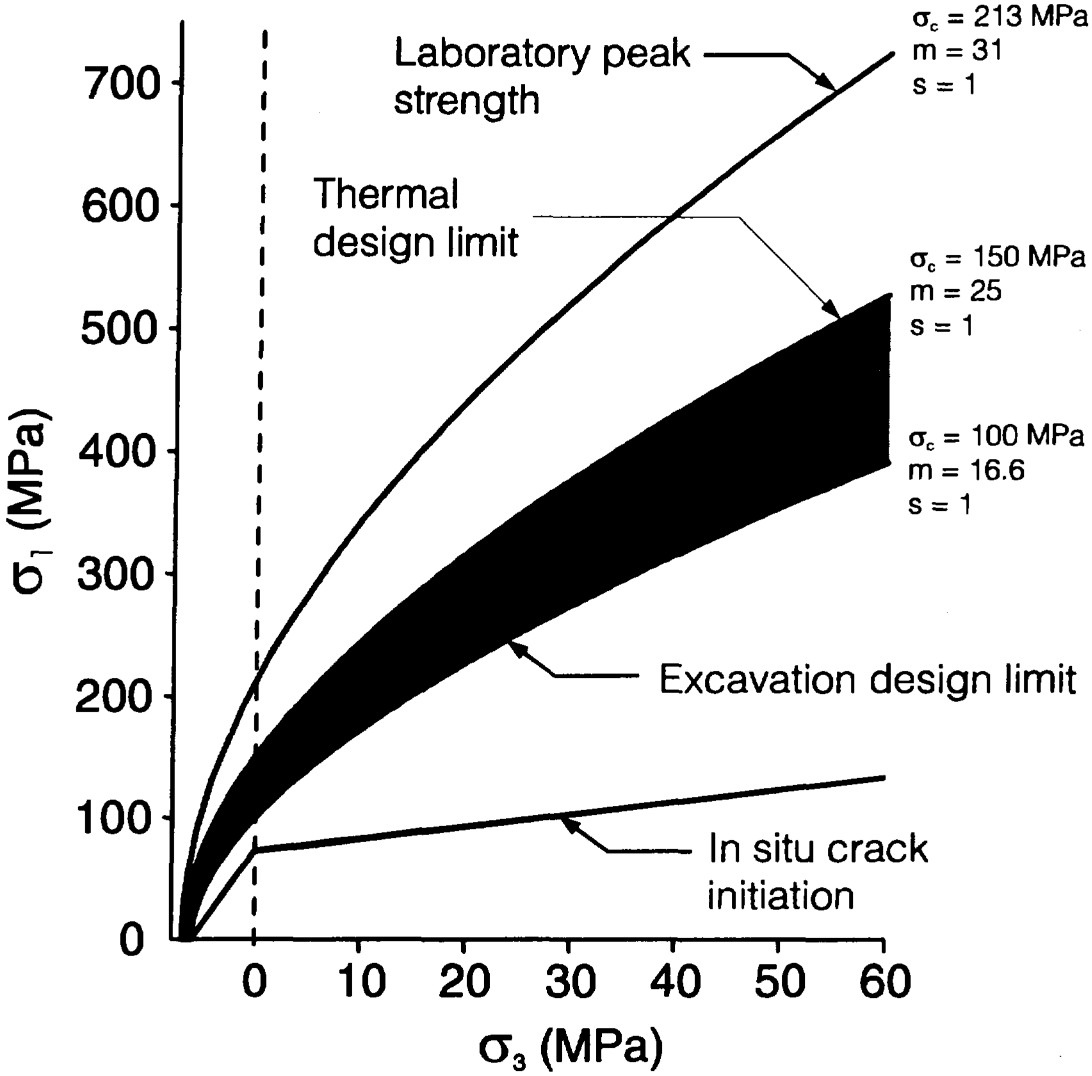


FIGURE 16: Hoek and Brown (1980) Failure Envelopes Defining the Design Limits Used for Stability Analyses and the Peak Laboratory Strength (Also shown is the in situ crack initiation envelope)

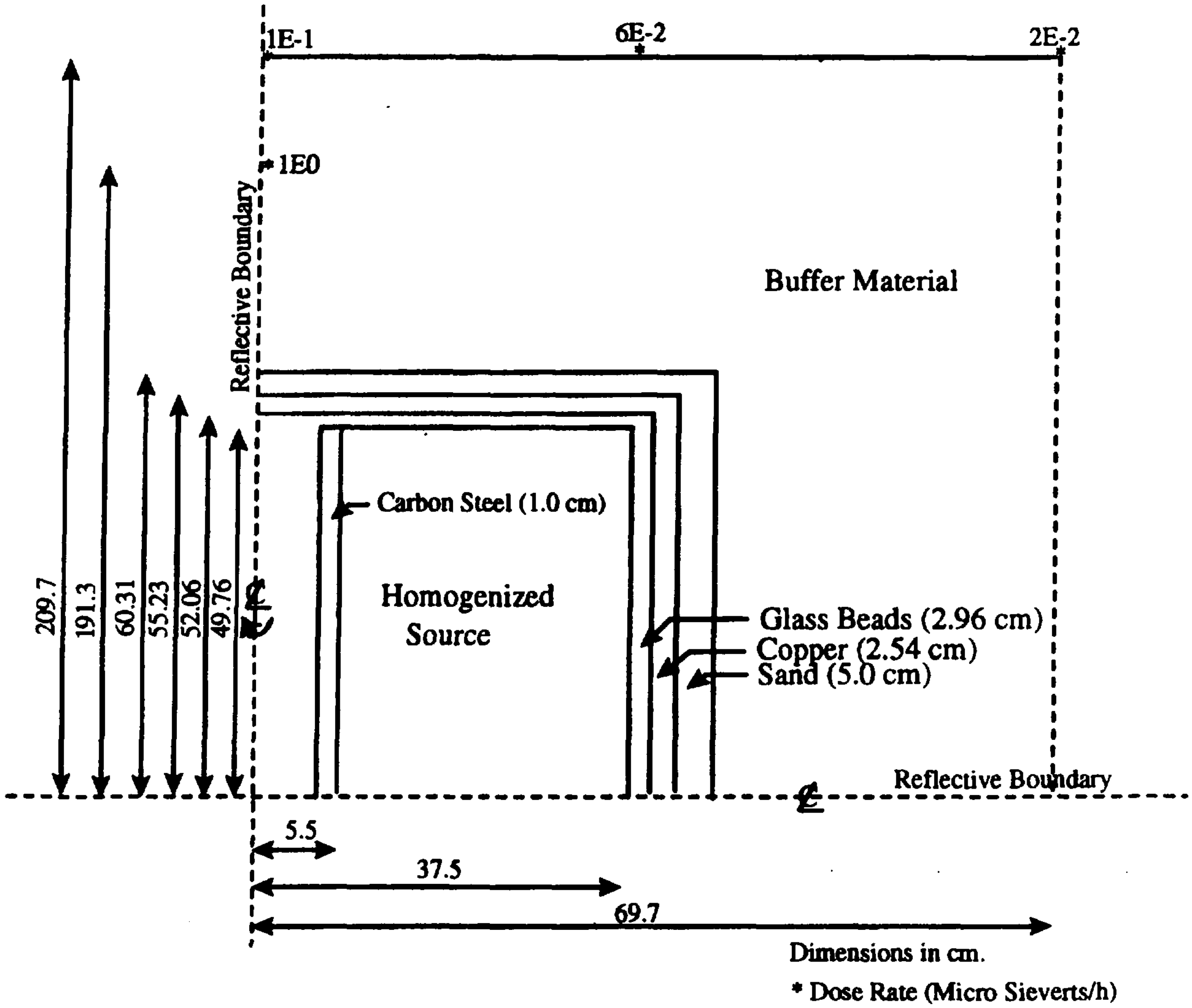


FIGURE 17: Axisymmetric Two-Dimensional Shielding Model at the Face of the Disposal Room (Source: G.R. Penner. 1995. January 20. In-room emplacement radiation shielding calculations. Unpublished AECL Internal Memorandum, GRP-95-011.)

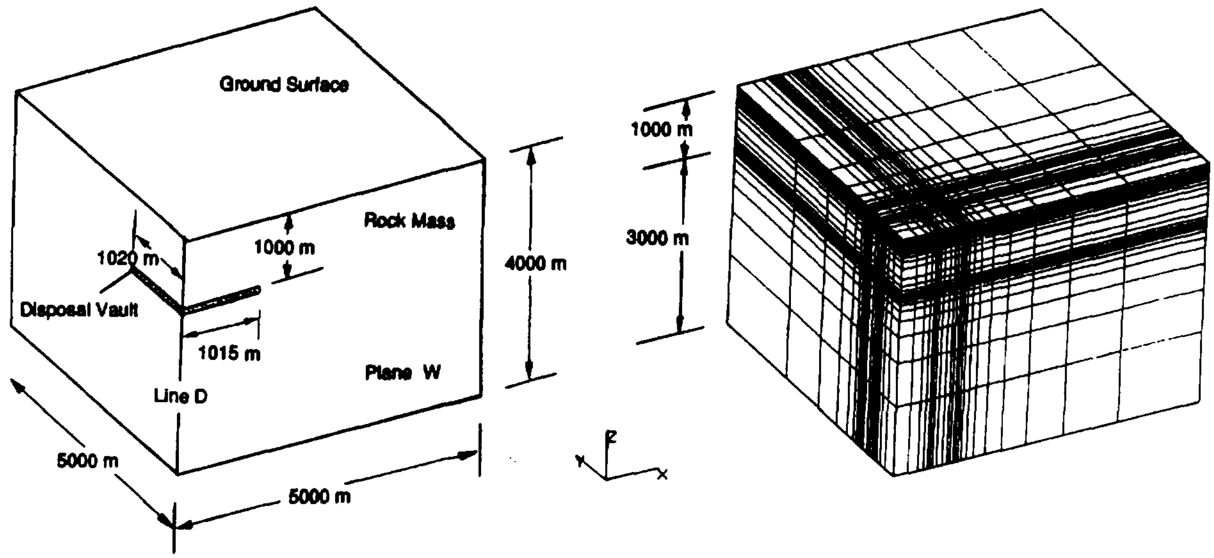


FIGURE 18: Perspective View and Finite-Element Discretization of the Far-Field Model (Vault at 1000-m depth) (Wai and Tsai 1995)

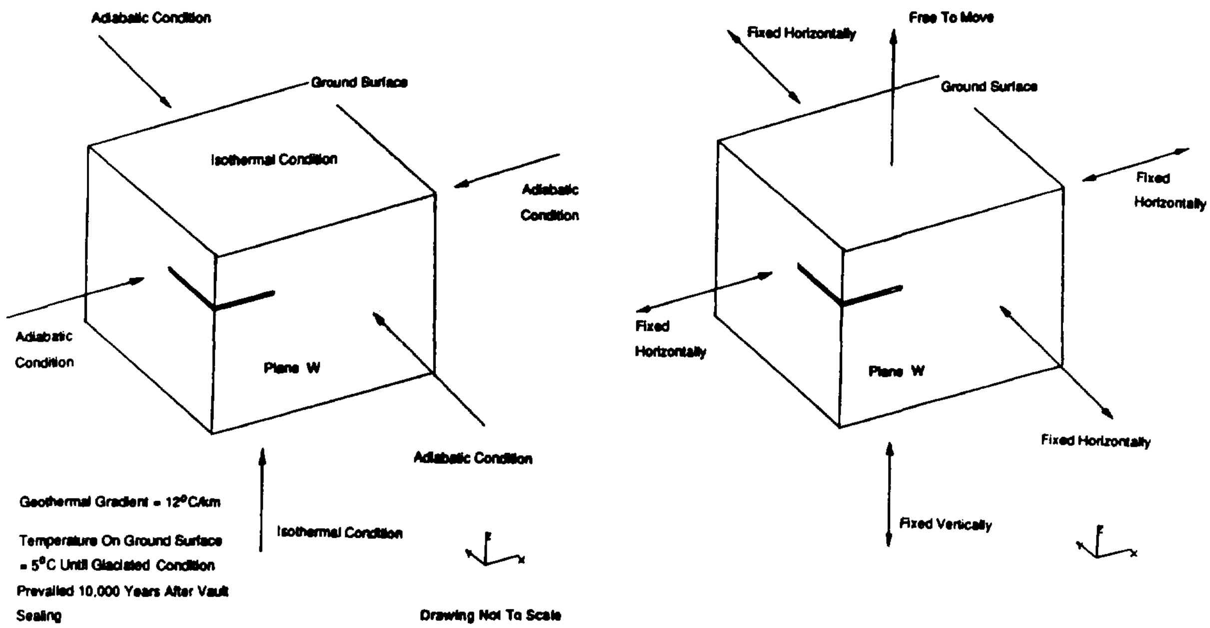


FIGURE 19: Thermal and Mechanical Boundary Conditions of the Far-Field Model (Vault at 1000-m depth) (Wai and Tsai 1995)

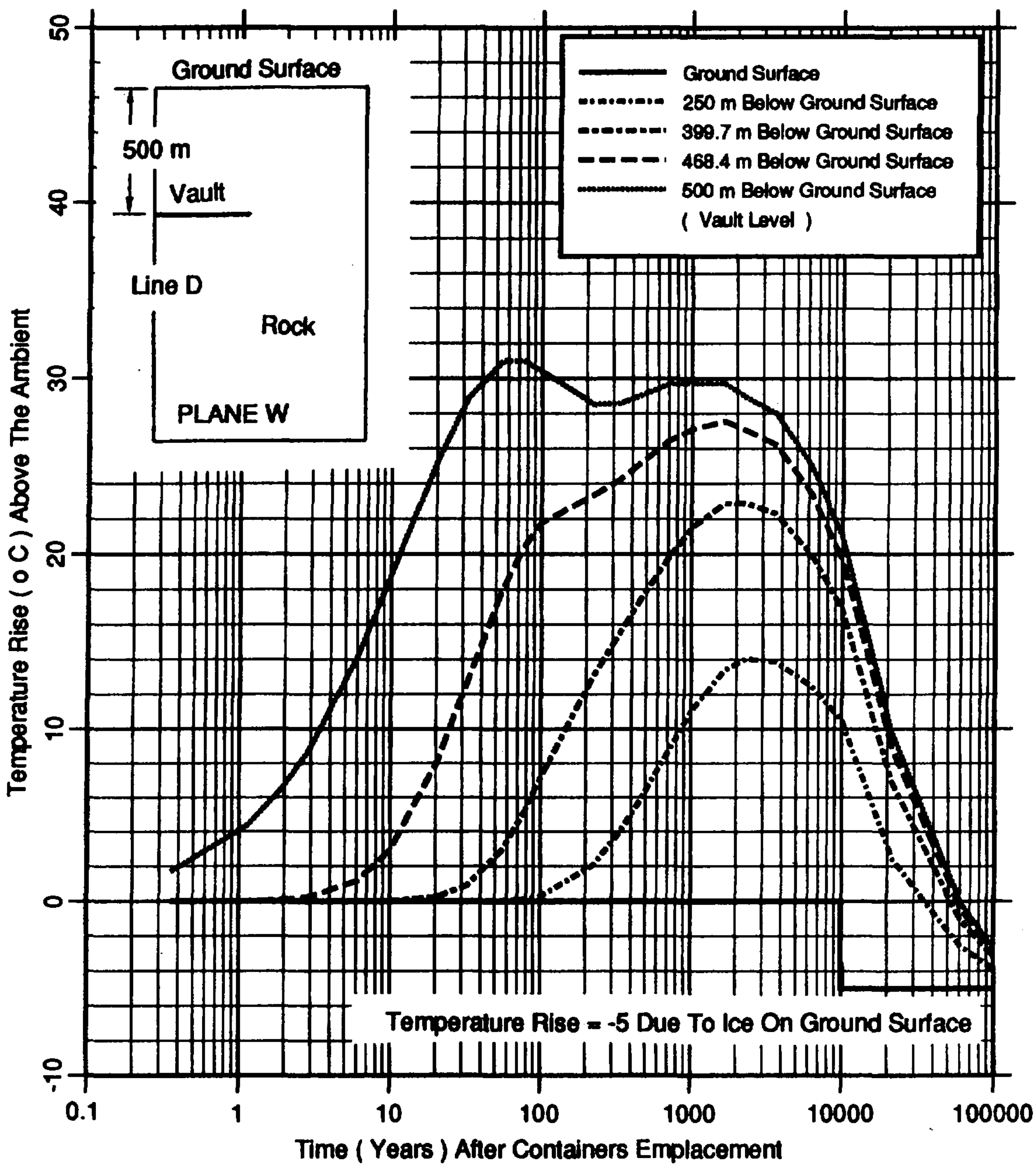


FIGURE 20: Variation of Averaged Temperature with Time (Vault at 500-m depth, after Wai and Tsai (1995))

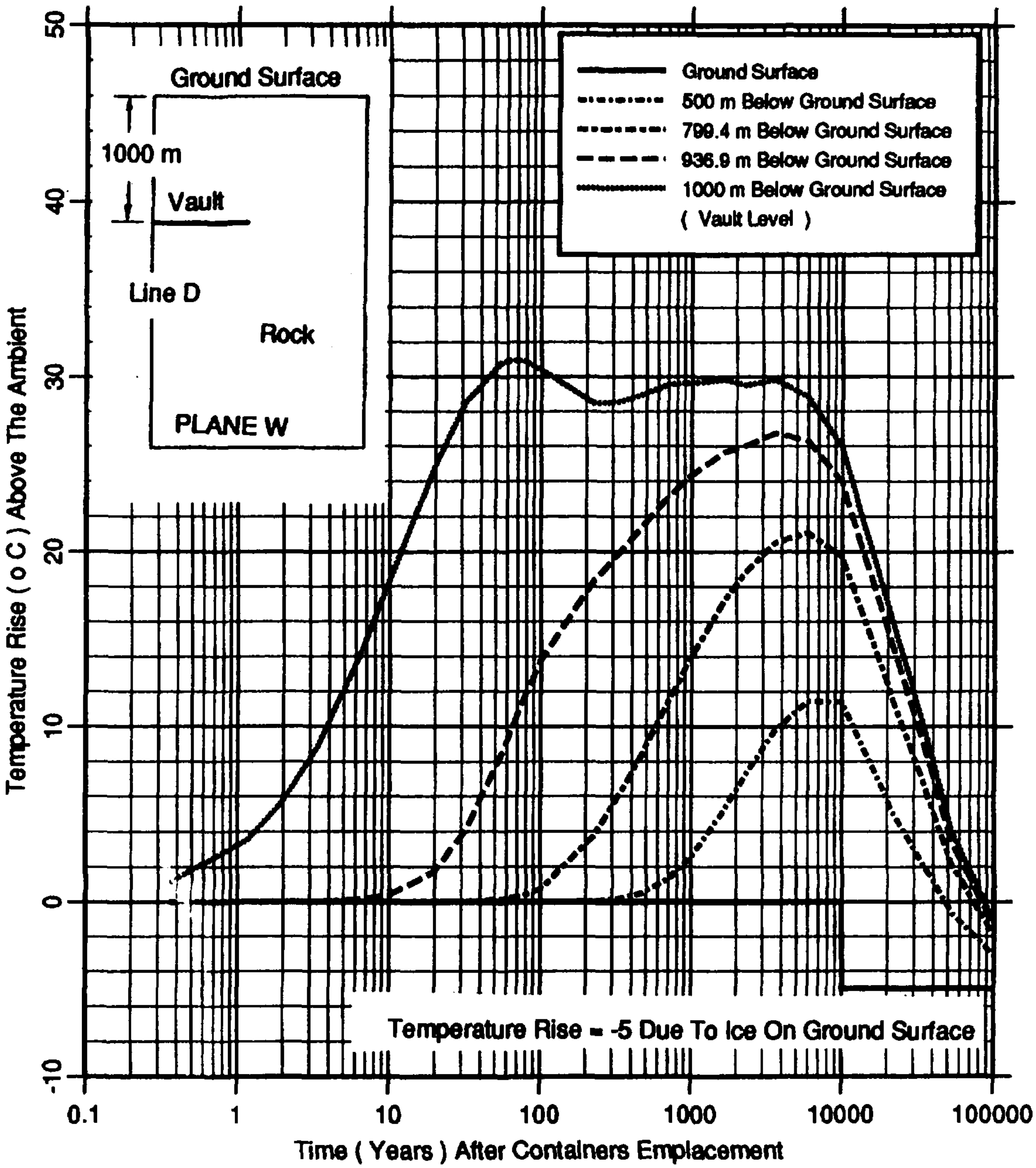


FIGURE 21: Variation of Averaged Temperature with Time (Vault at 1000-m depth, after Wai and Tsai (1995))

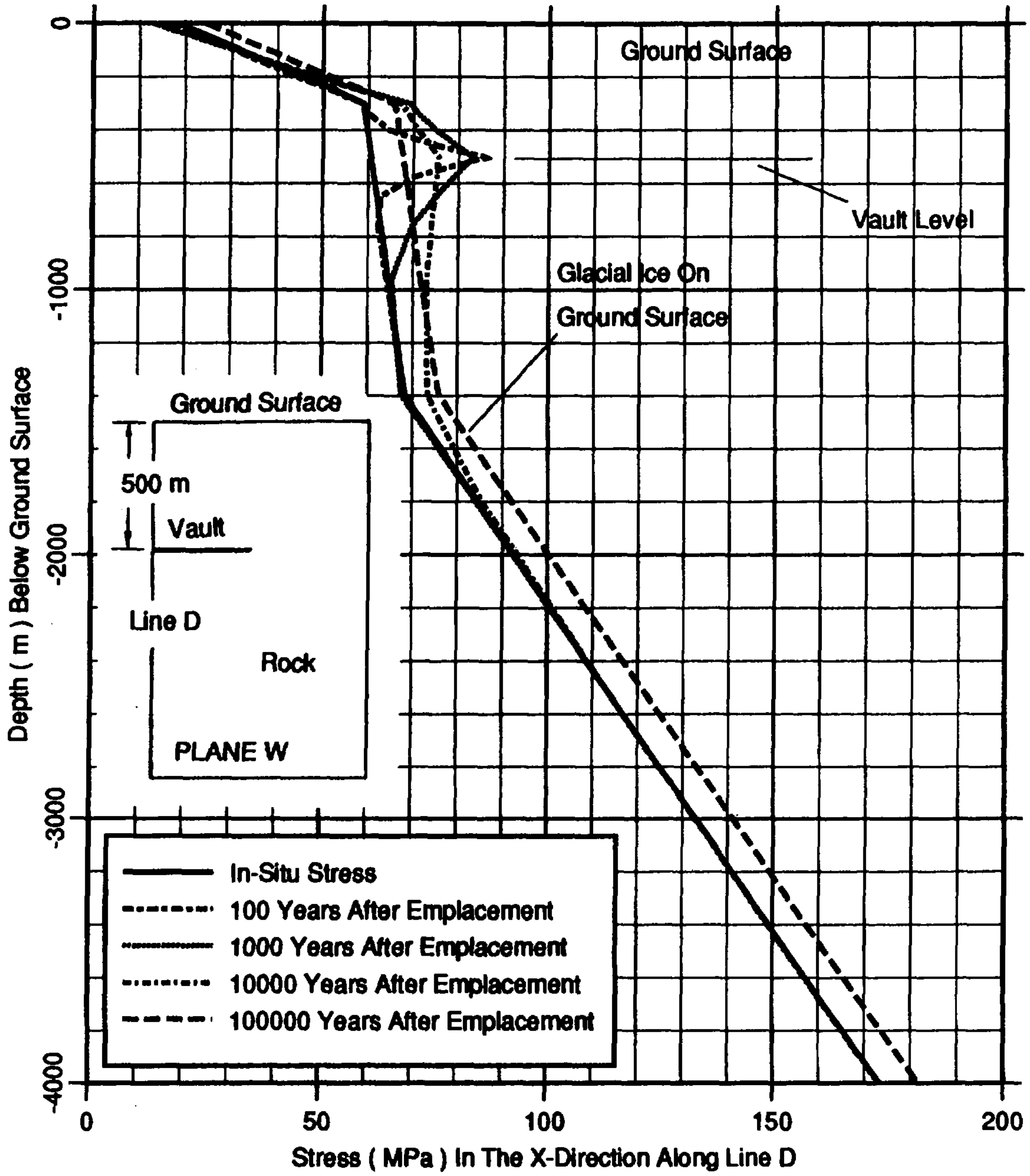


FIGURE 22: Variation of Major Horizontal Stress with Depth and Time (Vault at 500-m depth, after Wai and Tsai (1995))

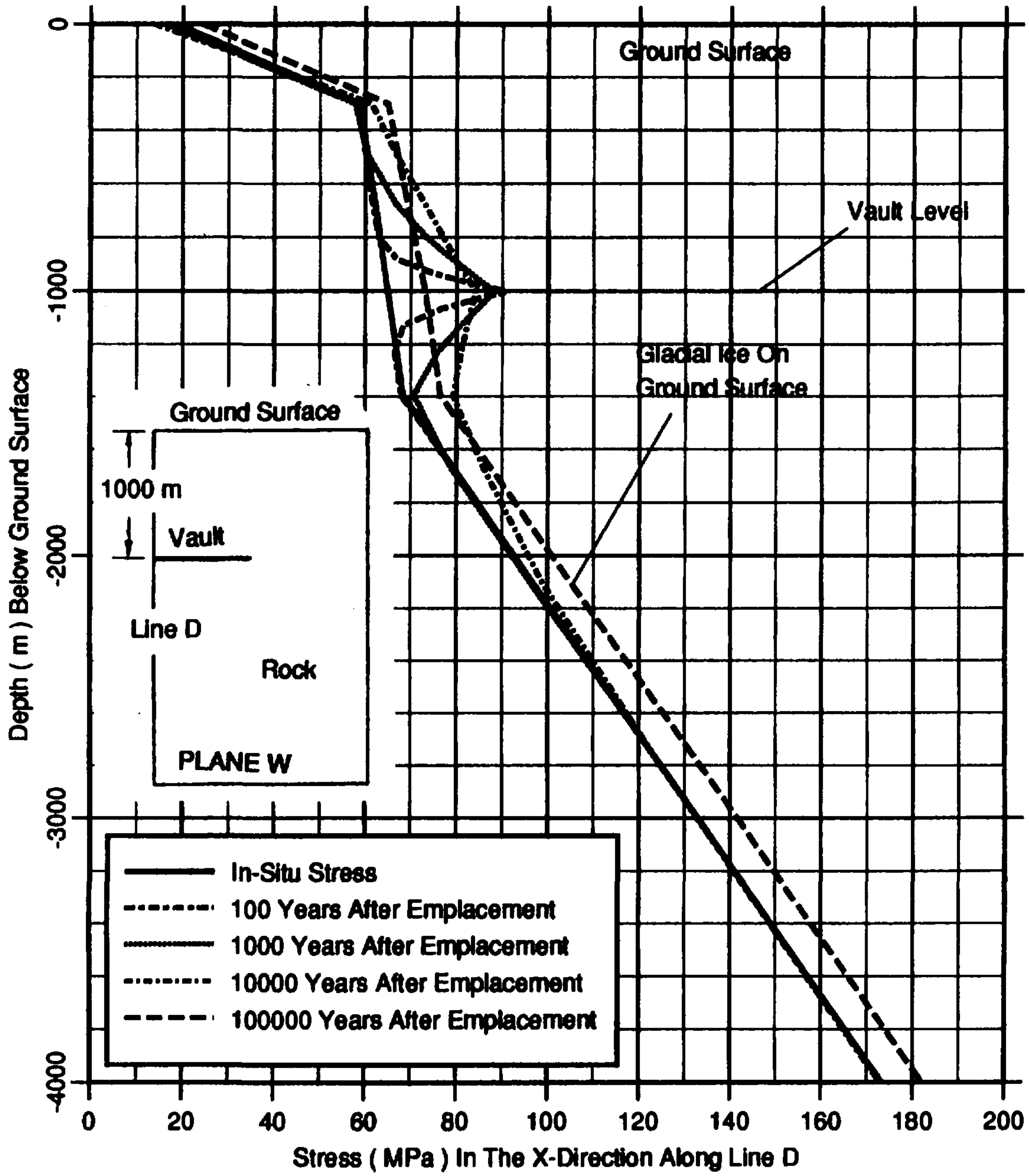


FIGURE 23: Variation of Major Horizontal Stress with Depth and Time (Vault at 1000-m depth, after Wai and Tsai (1995))

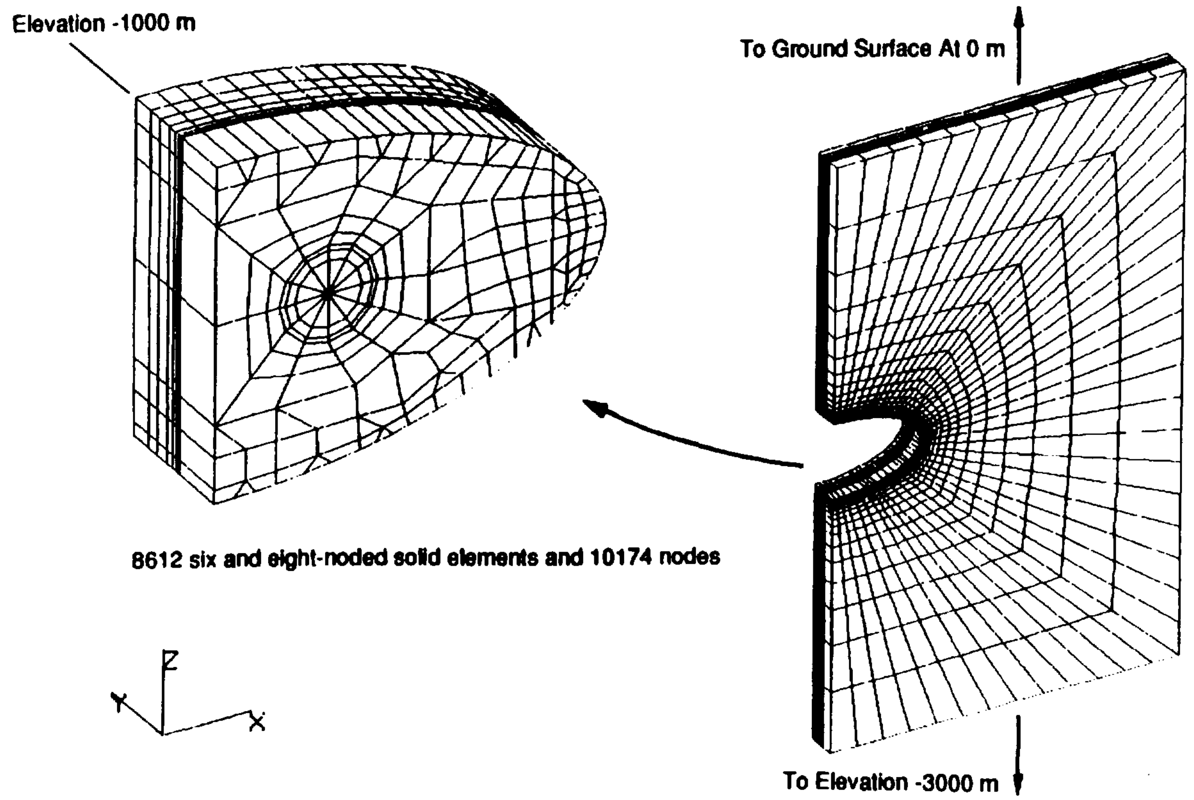


FIGURE 24: Finite-Element Discretization of the Central Part of the Unit Cell for Near-Field Analyses (Wai and Tsai 1995)

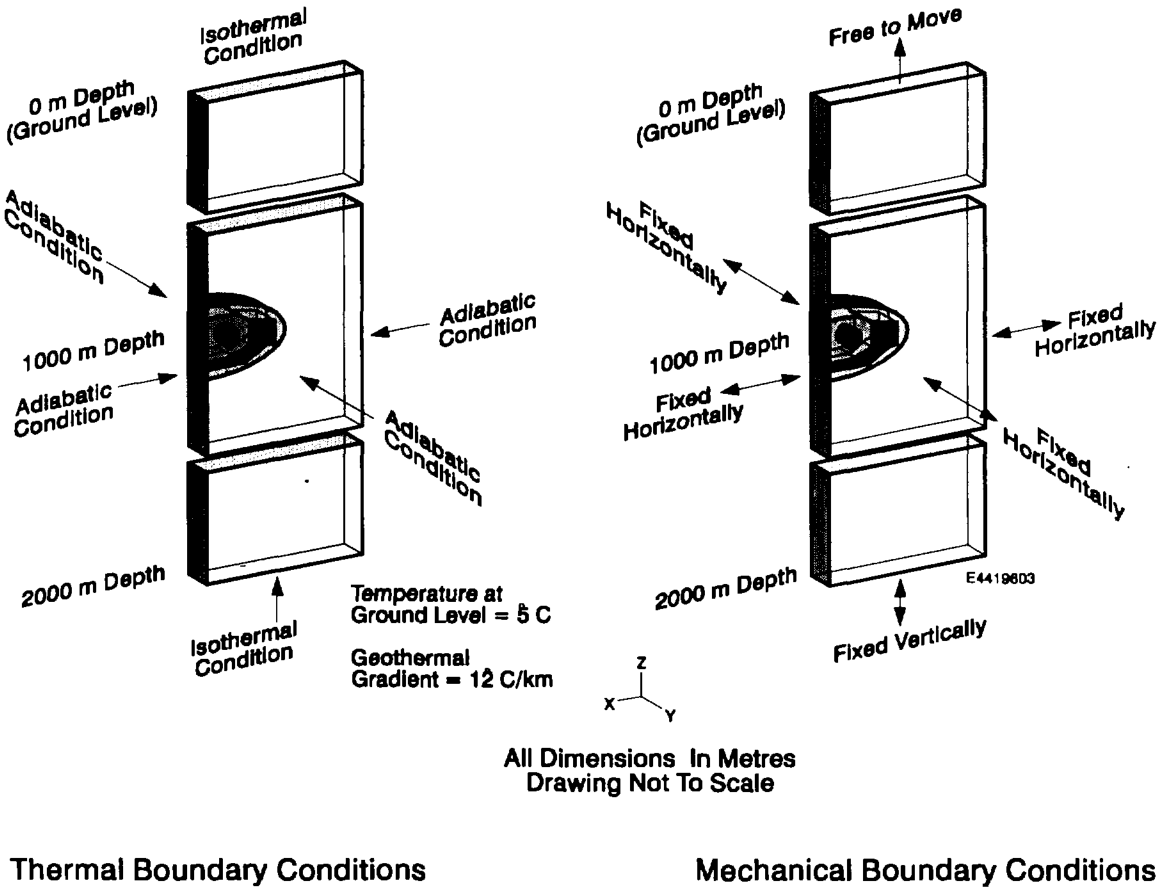


FIGURE 25: Thermal and Mechanical Boundary Conditions of the Near-Field Model (after Wai and Tsai 1995)

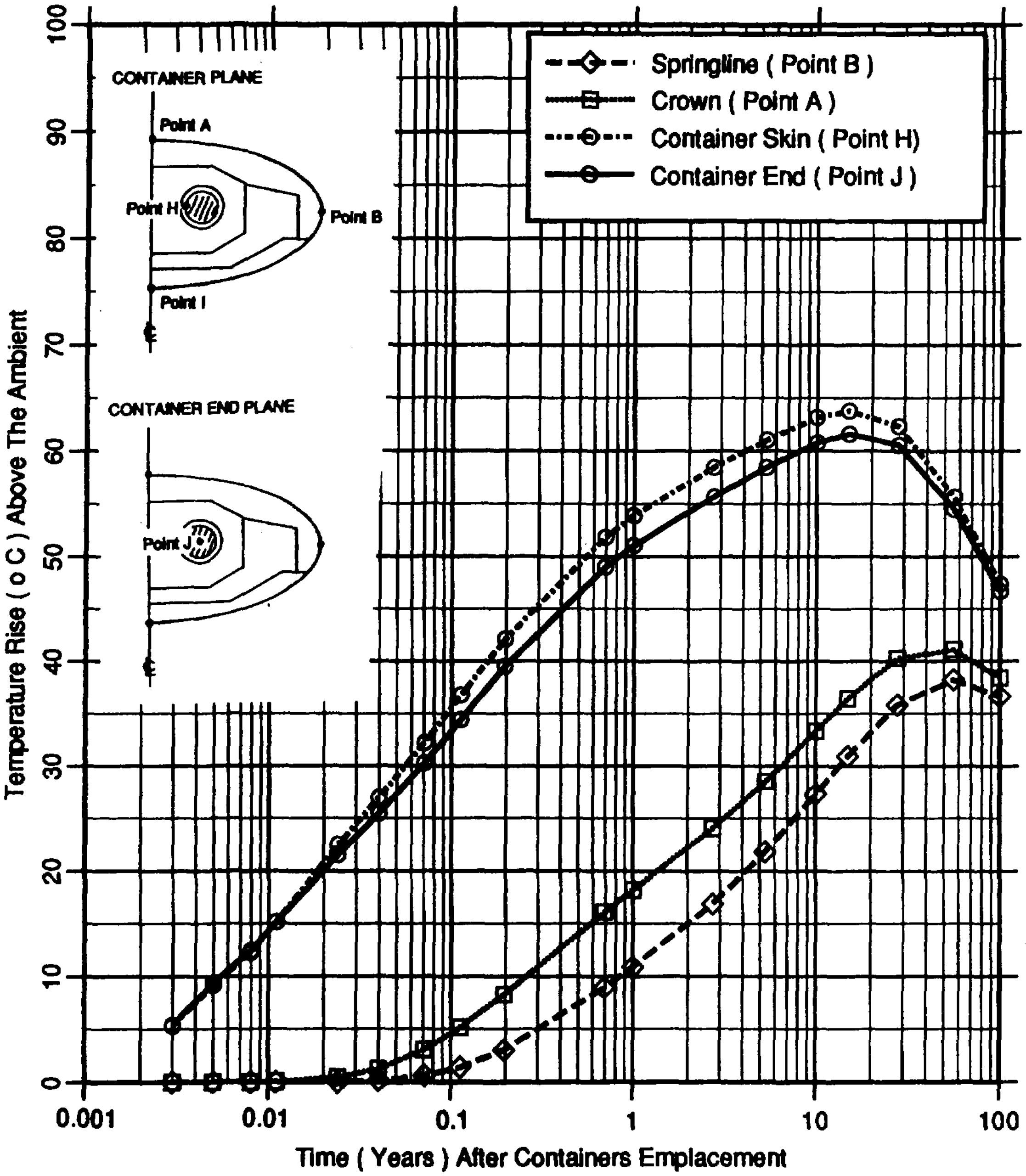


FIGURE 26: Variation of Temperature Rise with Time at Points within the Unit Cell (Wai and Tsai 1995)

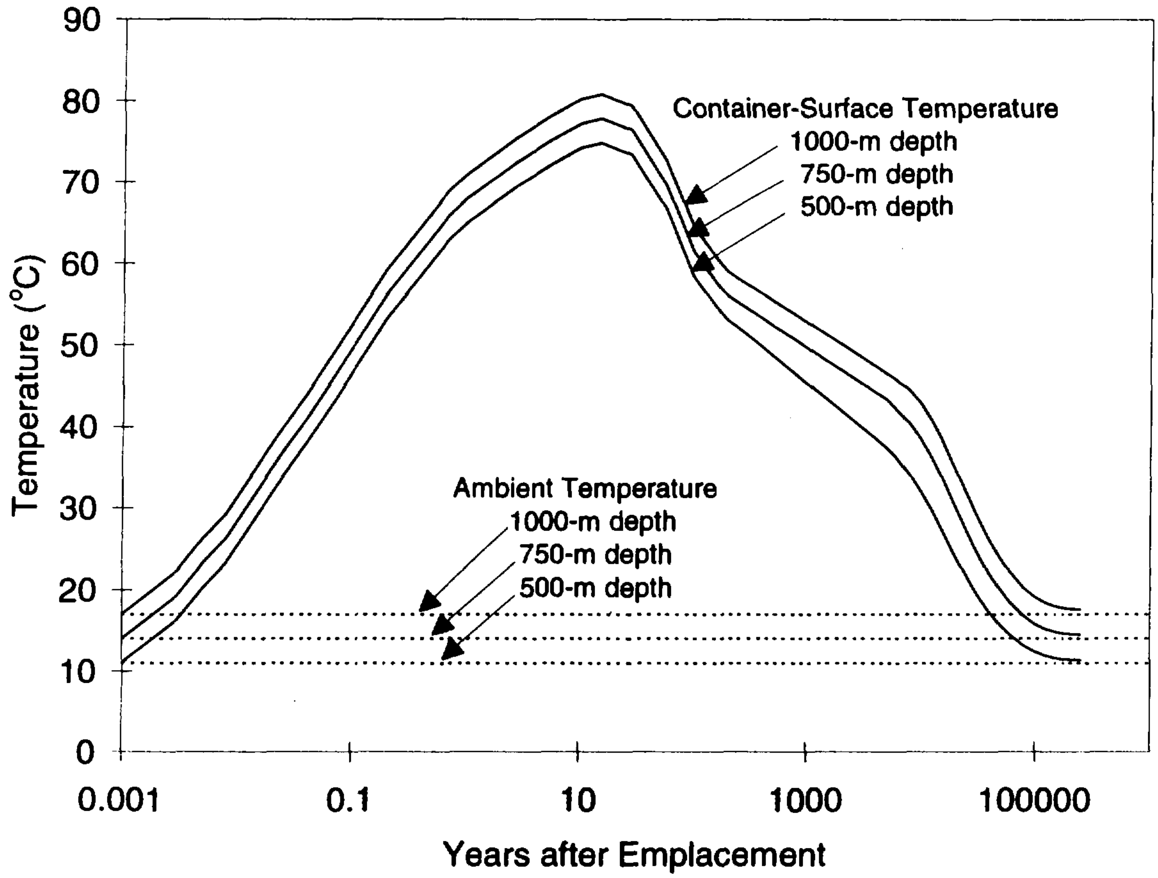


FIGURE 27: Variation of Temperature with Time at the Surface of the Disposal Container Located at Three Vault Depths (i.e., 500, 750 and 1000 m)

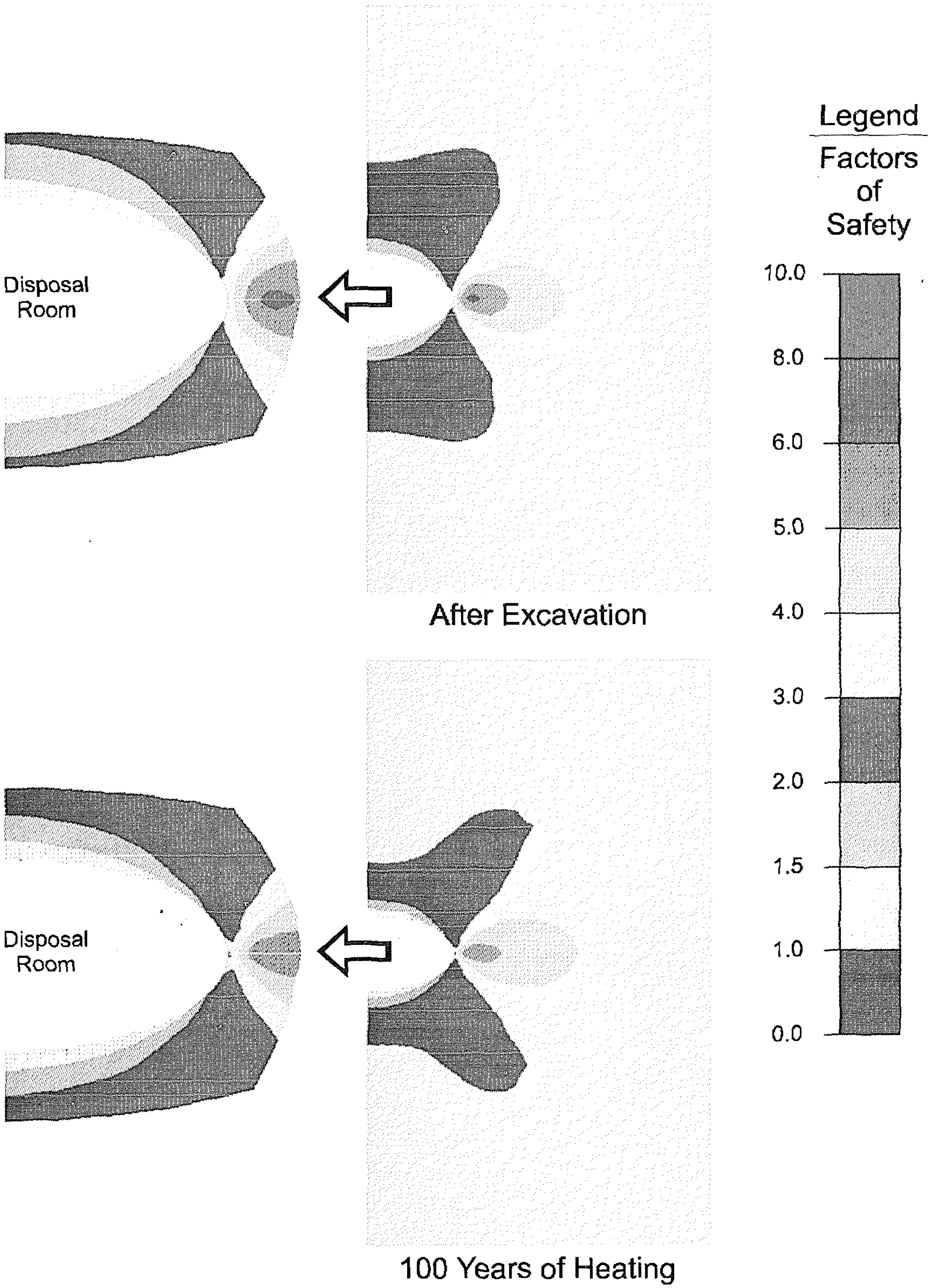


FIGURE 28: Factors of Safety Immediately after Excavation and after 100 Years of Waste Emplacement (500-m depth, perpendicular to major principal stress direction) (after Wai and Tsai 1995)

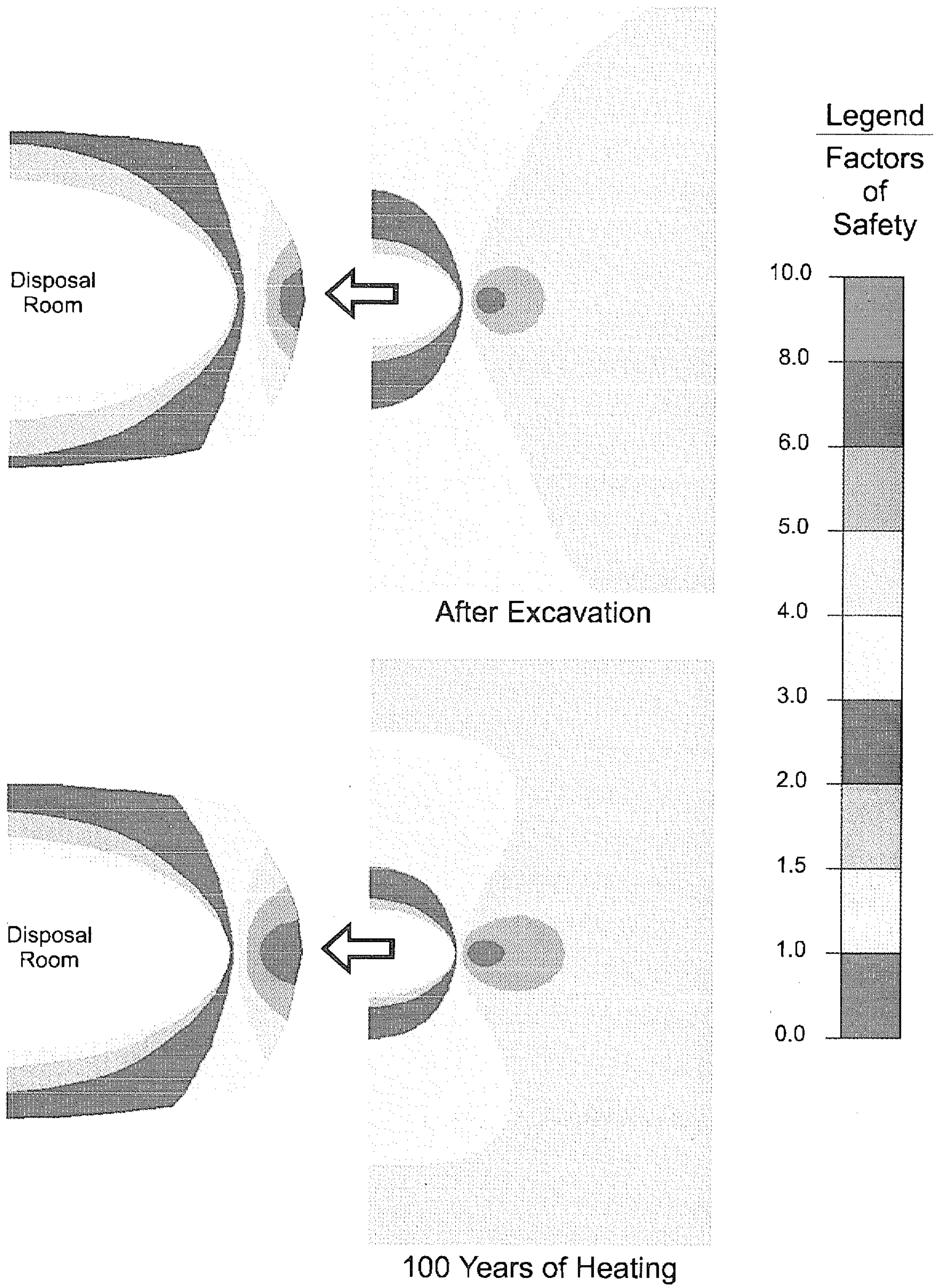


FIGURE 29: Factors of Safety Immediately after Excavation and after 100 Years of Waste Emplacement (1000-m depth, perpendicular to major principal stress direction) (after Wai and Tsai 1995)

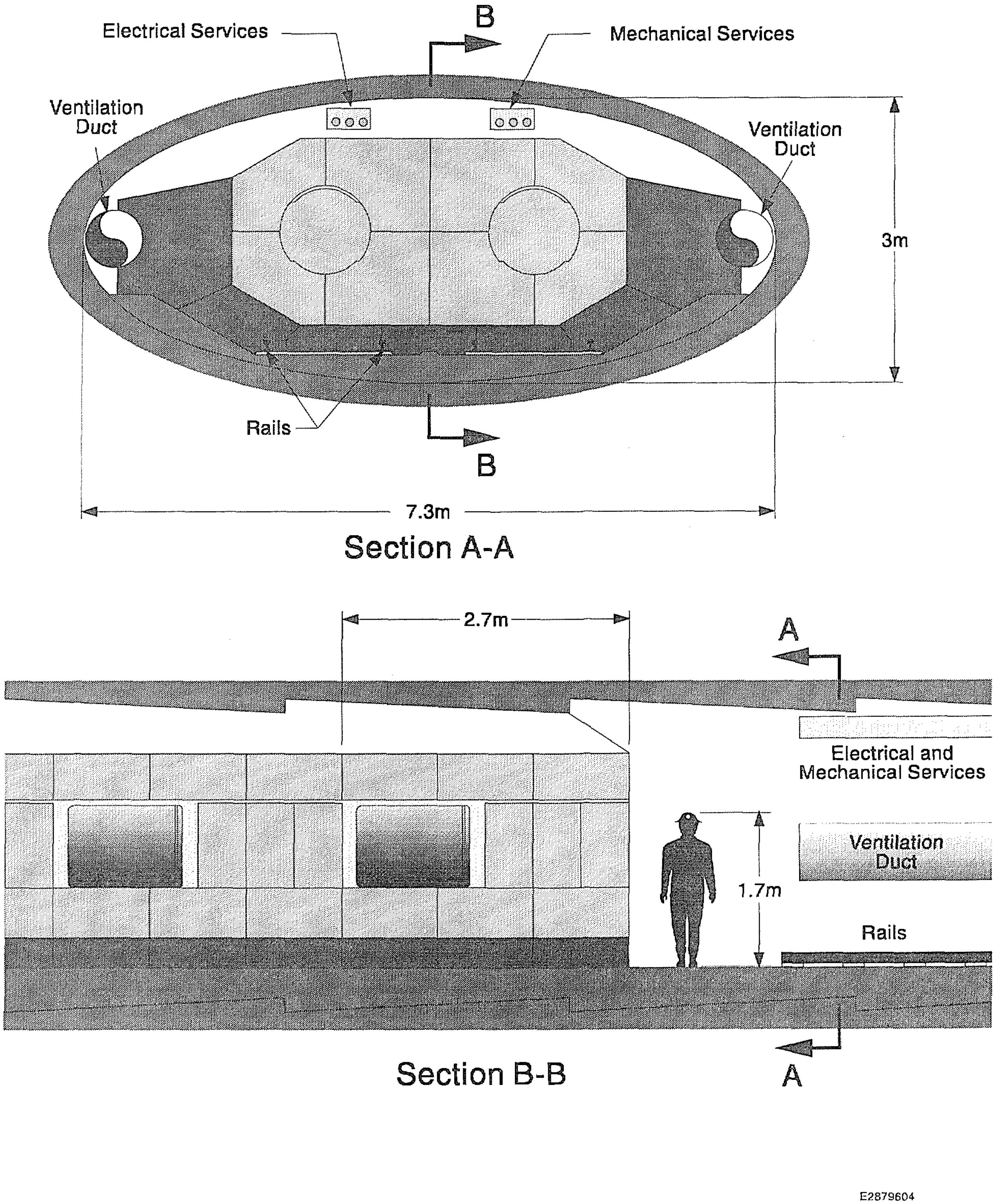


FIGURE 30: Disposal Room Showing General Arrangement of Emplaced Blocks and Waste, Including Ventilation, Electrical and Mechanical Services

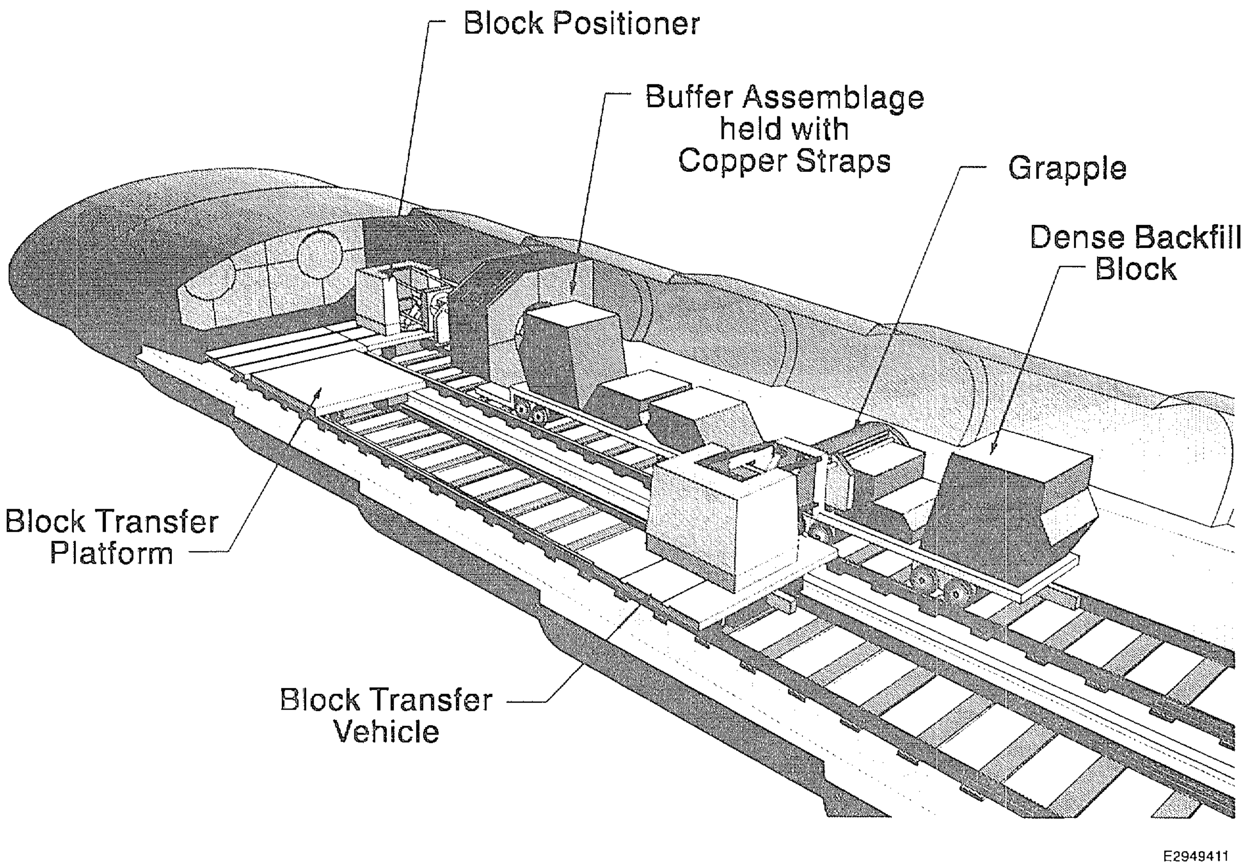


FIGURE 31: Precompacted-Block Handling Equipment within a Disposal Room

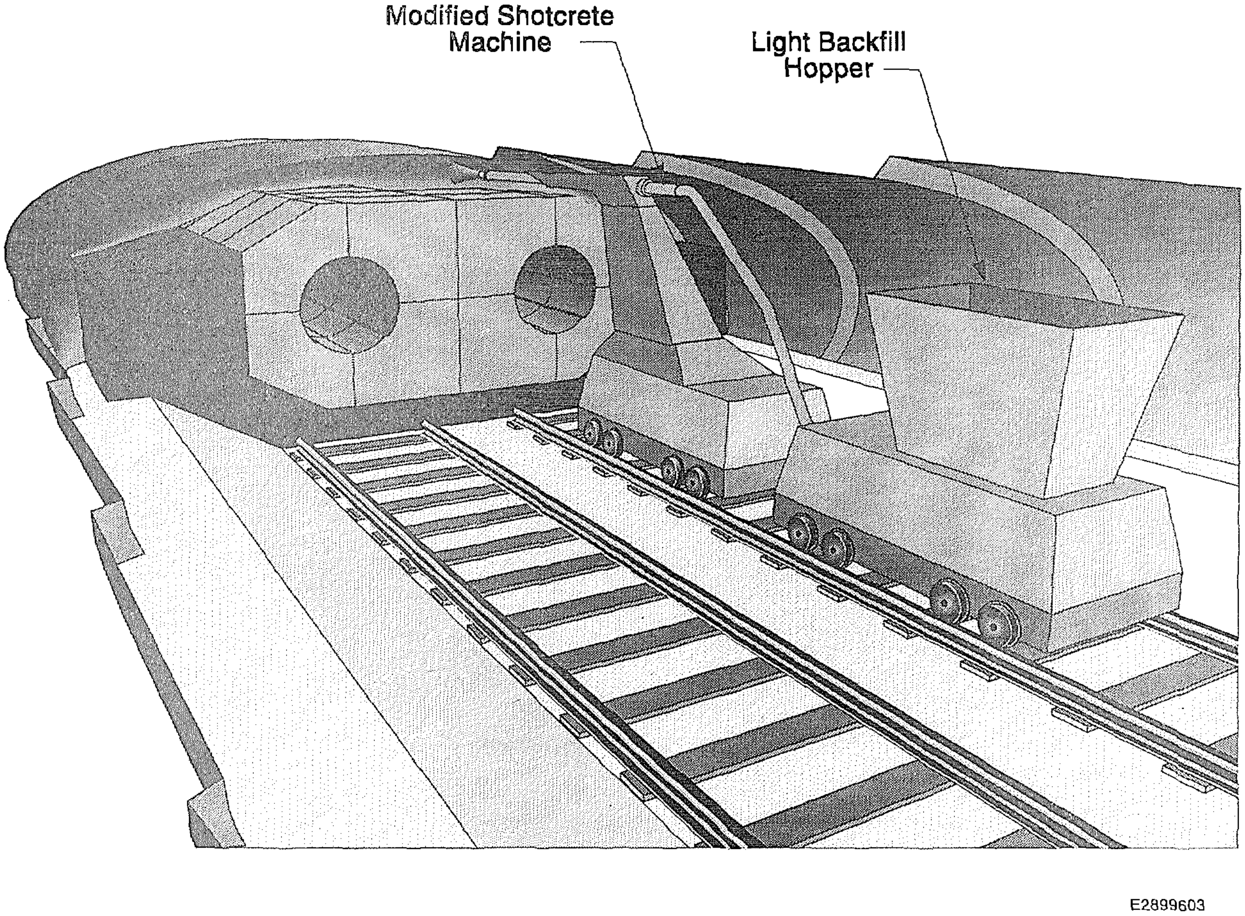


FIGURE 32: Light Backfill Placement

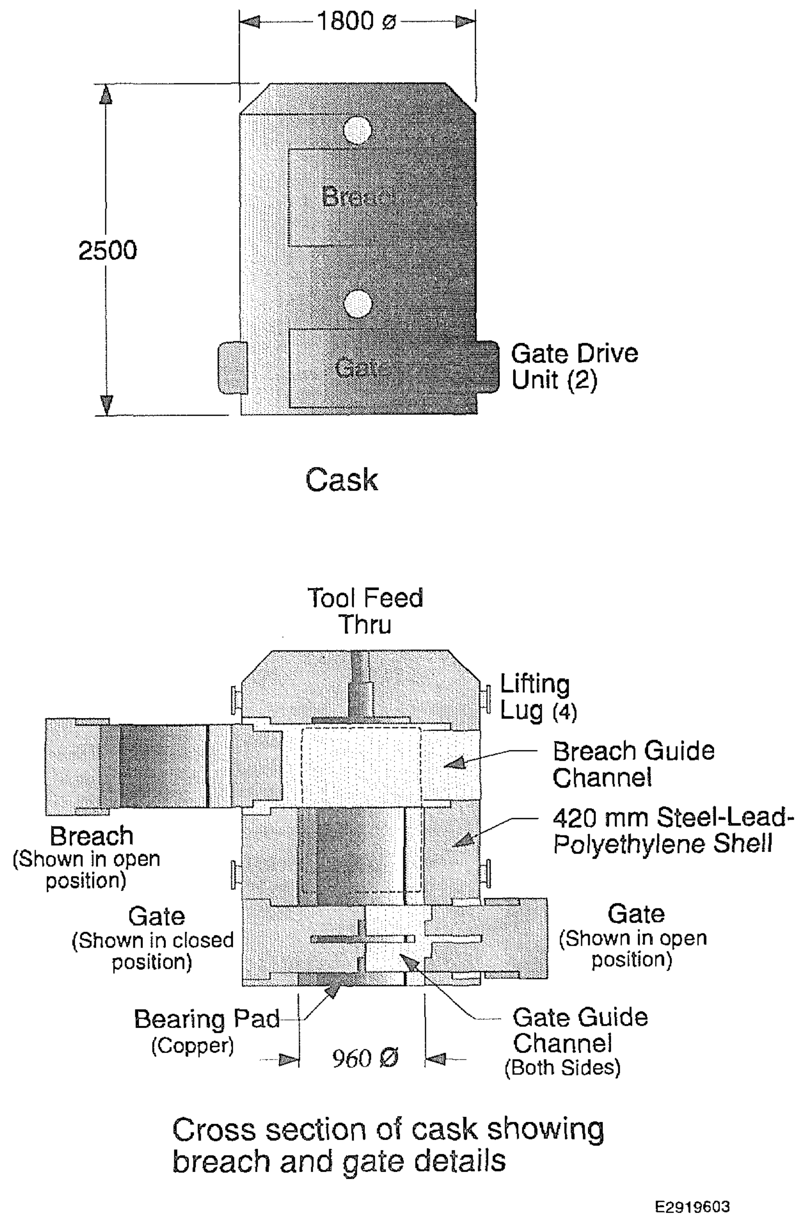


FIGURE 33: Disposal-Container Cask for the Copper-Shell, Packed-Particulate Container

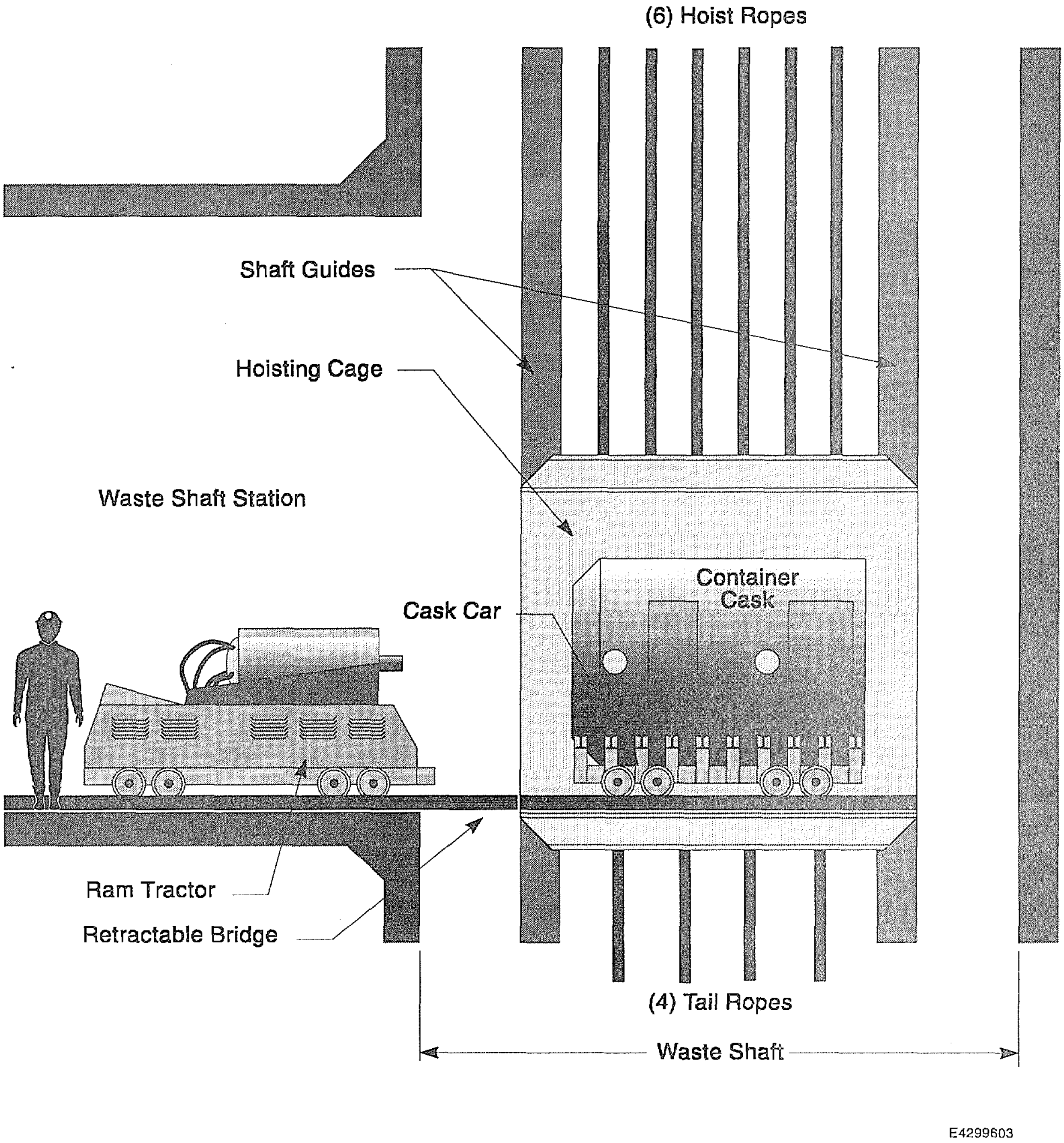


FIGURE 34: Transfer of Disposal-Container Cask Car from Waste-Shaft Cage

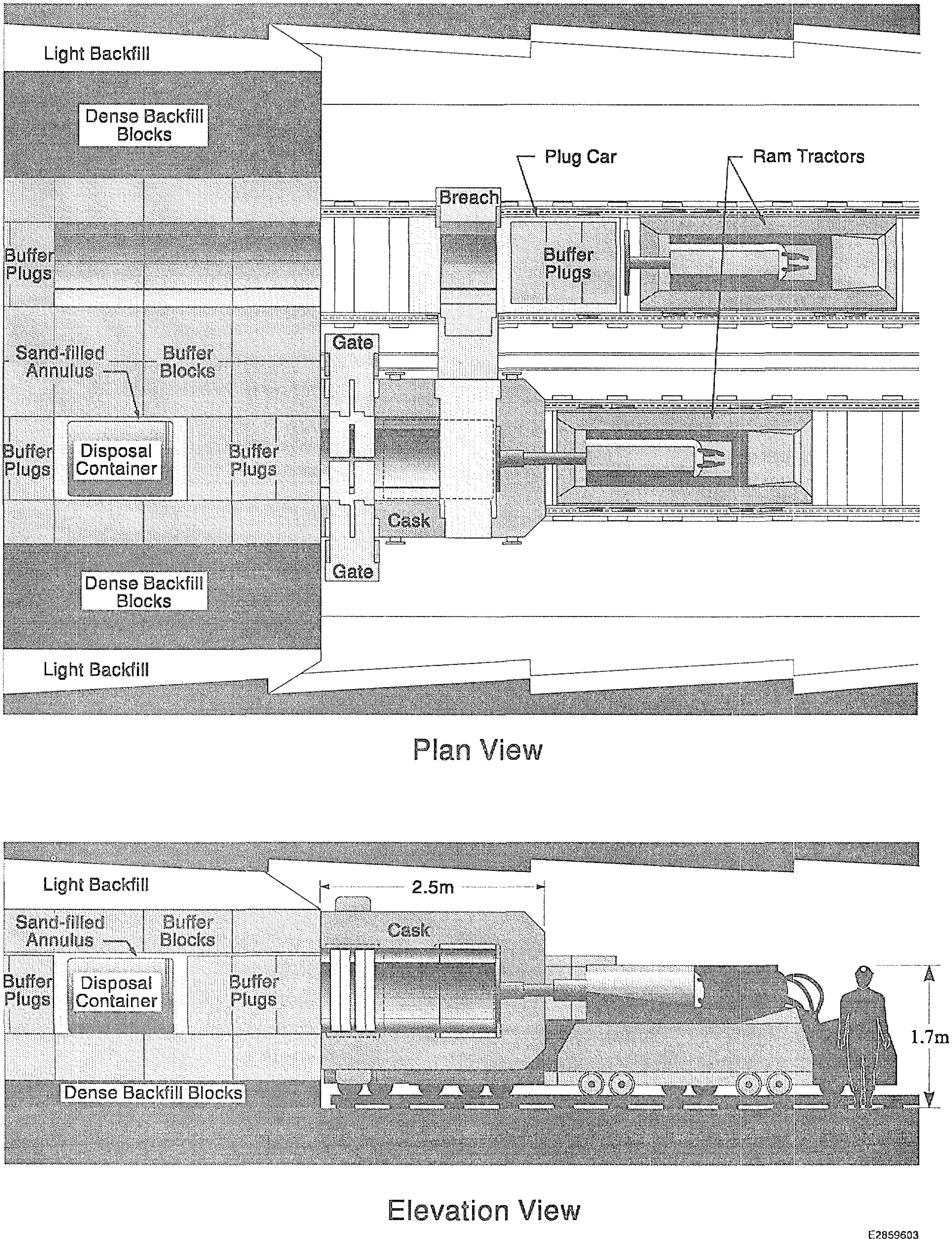


FIGURE 35: Disposal-Room Emplacement Operations

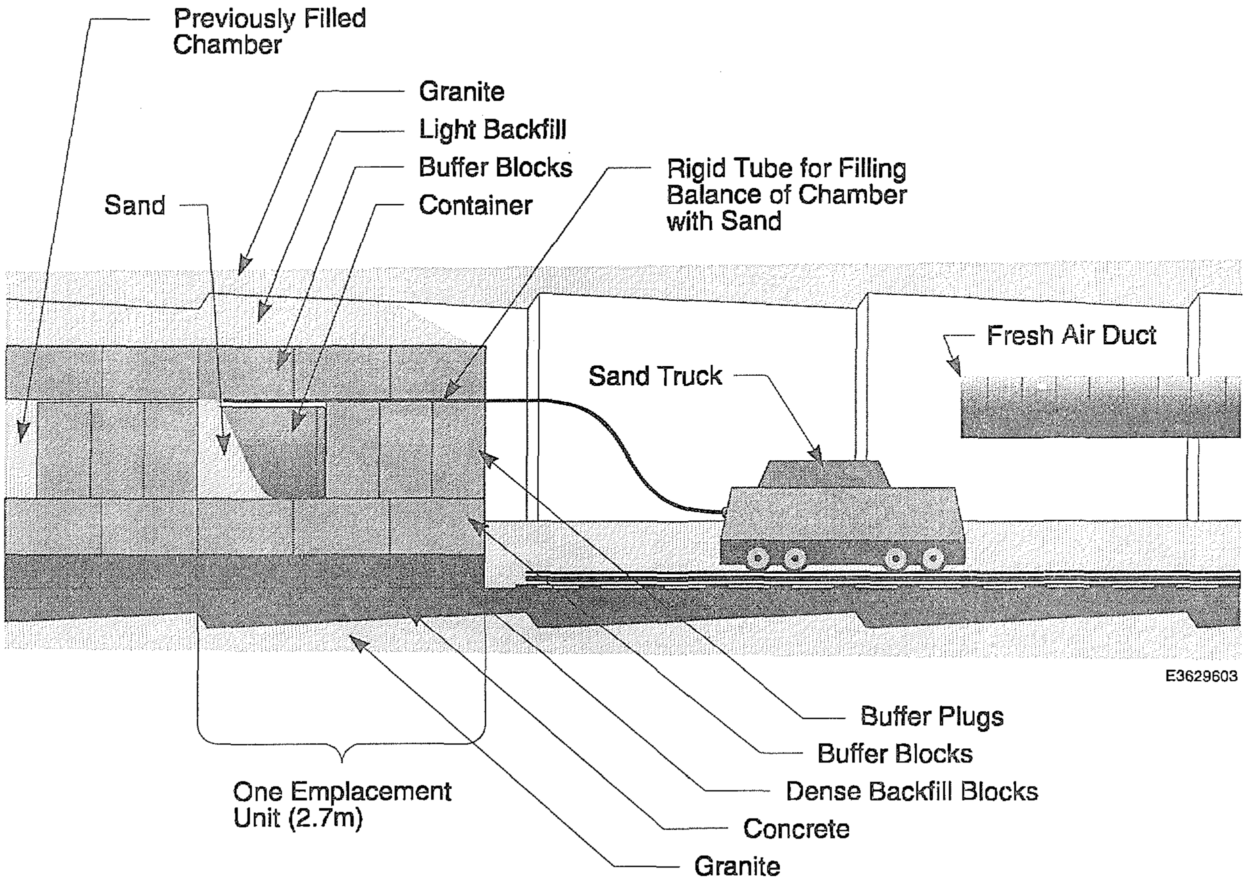


FIGURE 36: Annular-Fill Particulate Placed by Pneumatic Lance

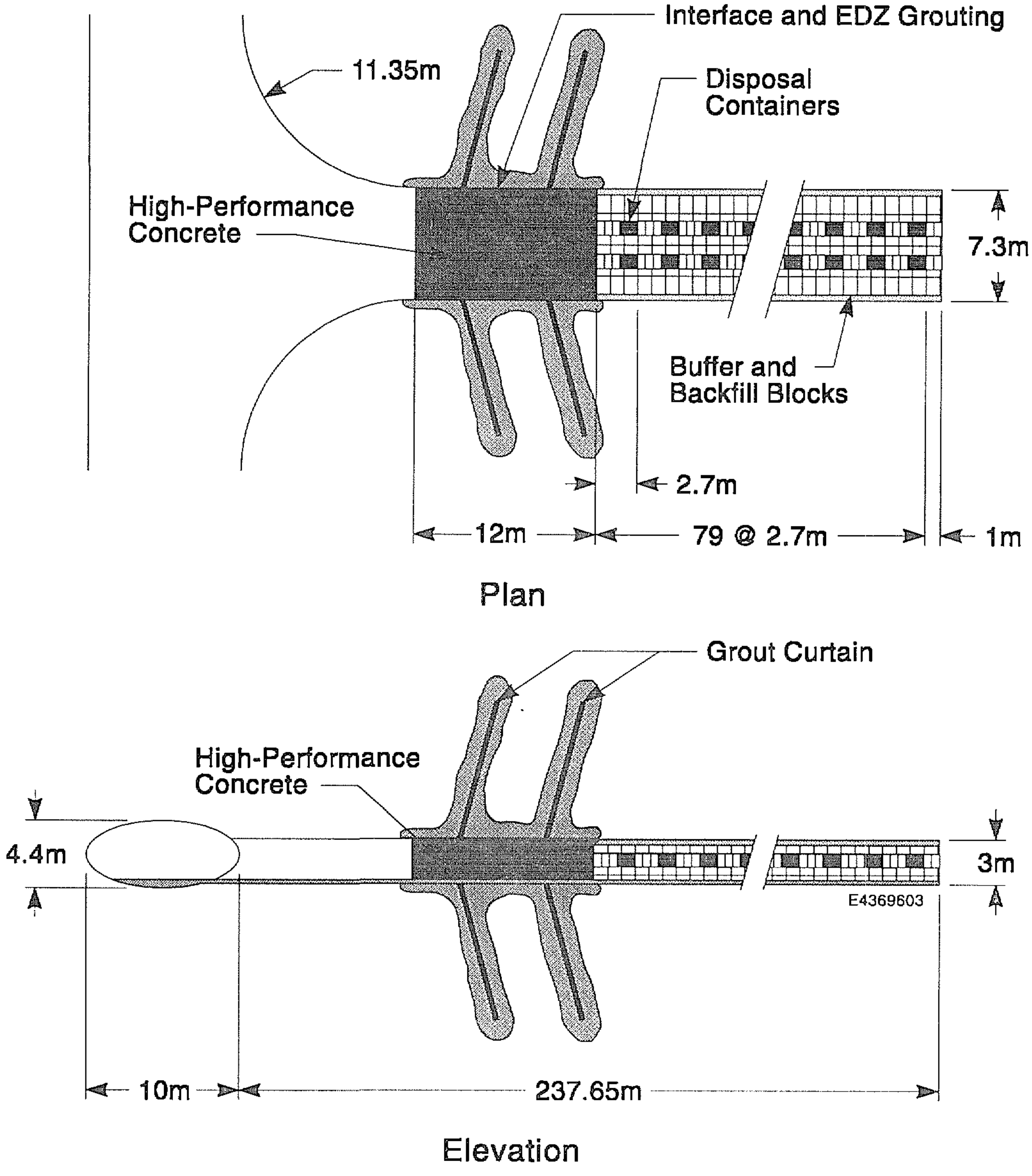


FIGURE 37: Sealing Bulkhead at Disposal-Room Entrance

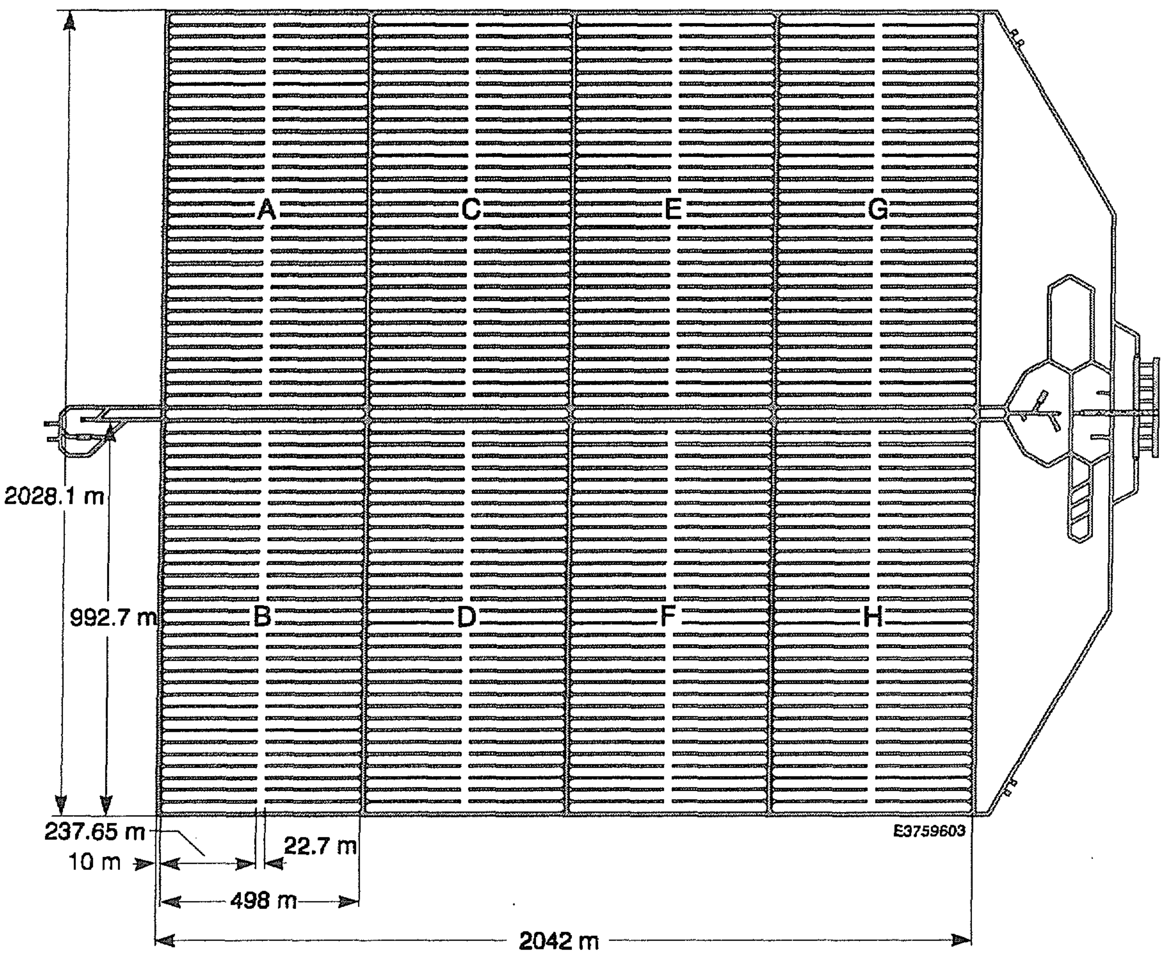


FIGURE 38: Plan of Used-Fuel Disposal Vault

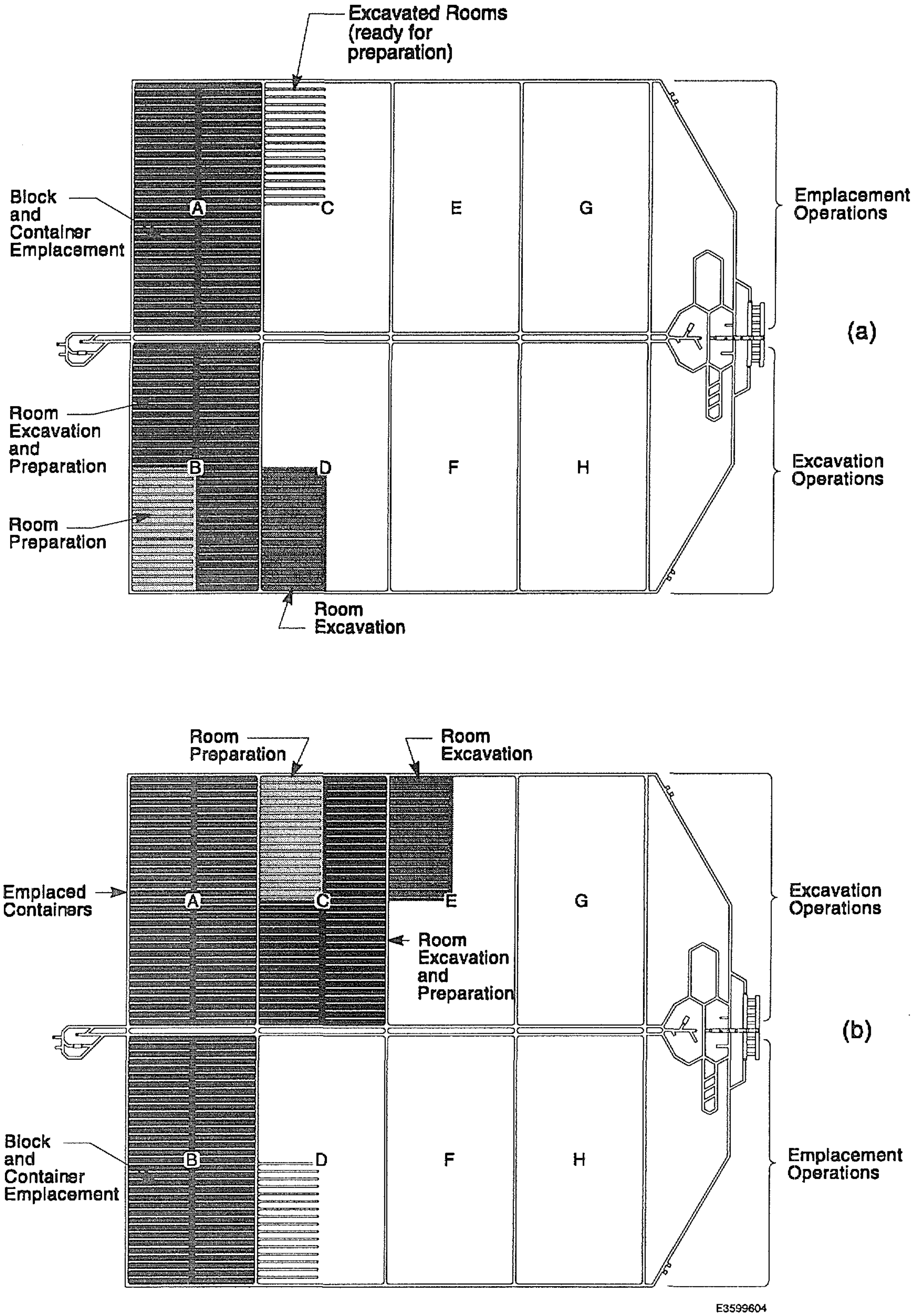


FIGURE 39: Panel Excavation and Waste Emplacement Sequence

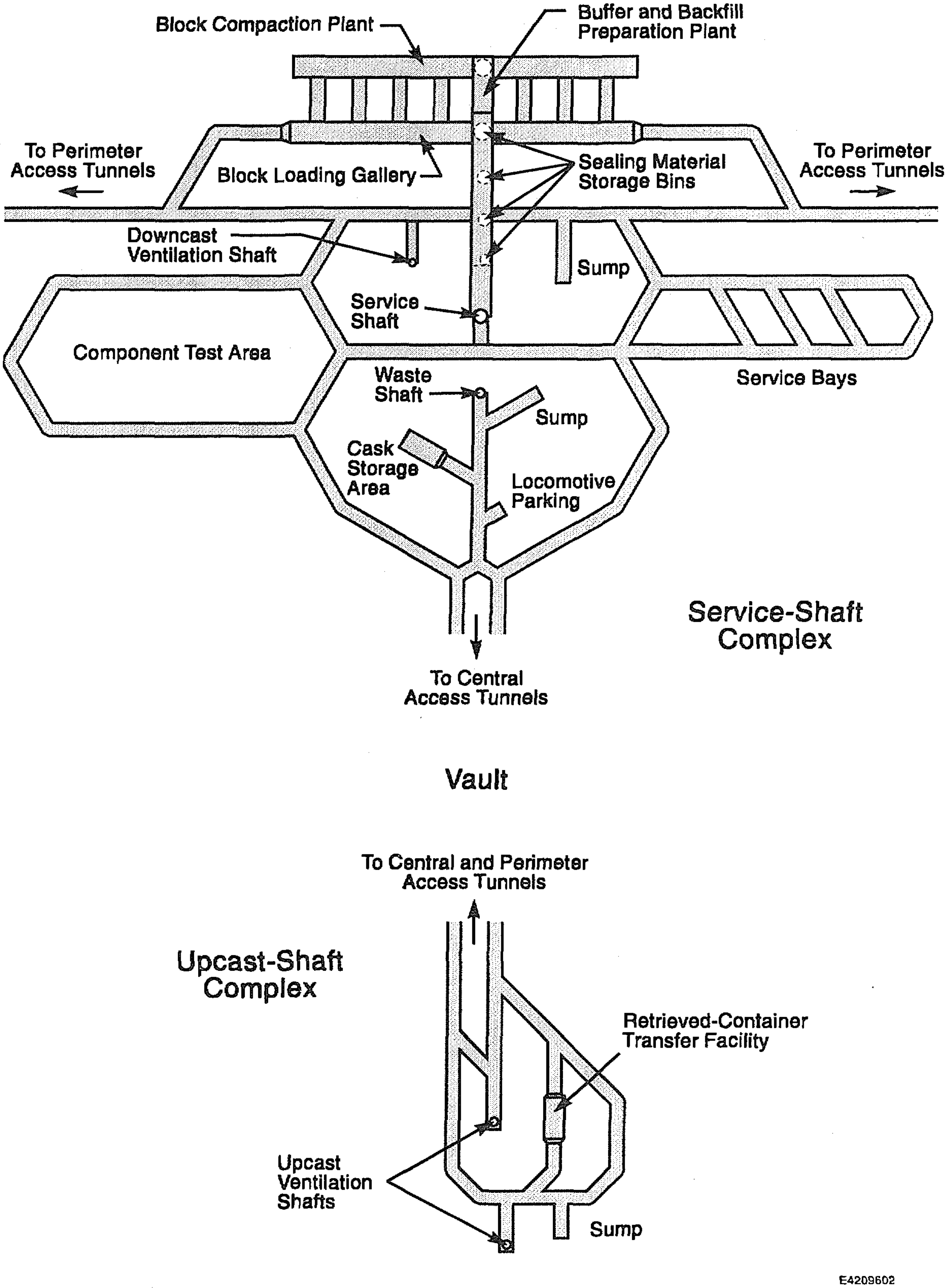


FIGURE 40: Service-Shaft and Upcast-Shaft Complexes

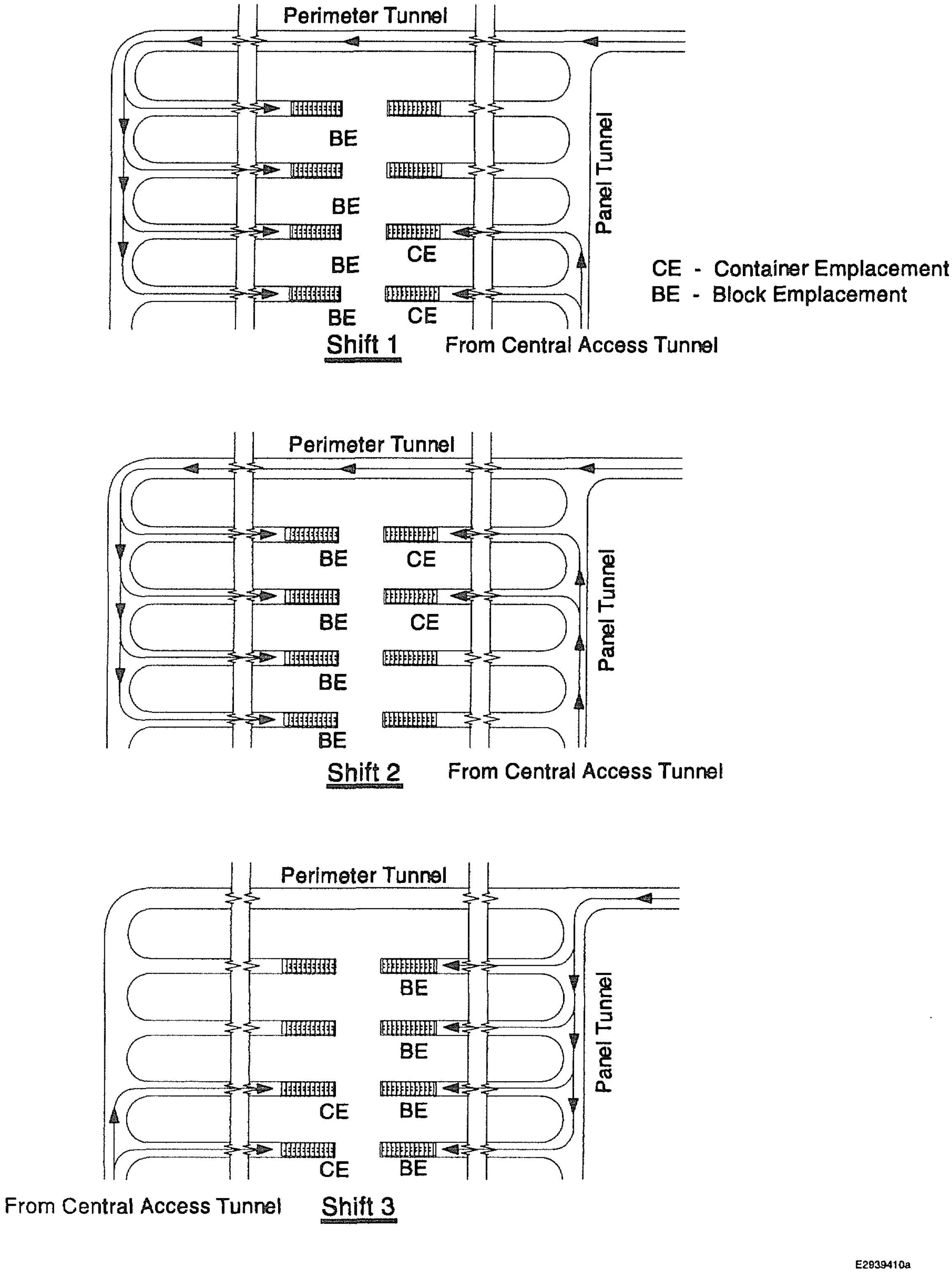


FIGURE 41: Disposal-Room Operating Sequence and Material-Flow Diagram (One day of activities detailing three shifts of emplacement operations in a panel)

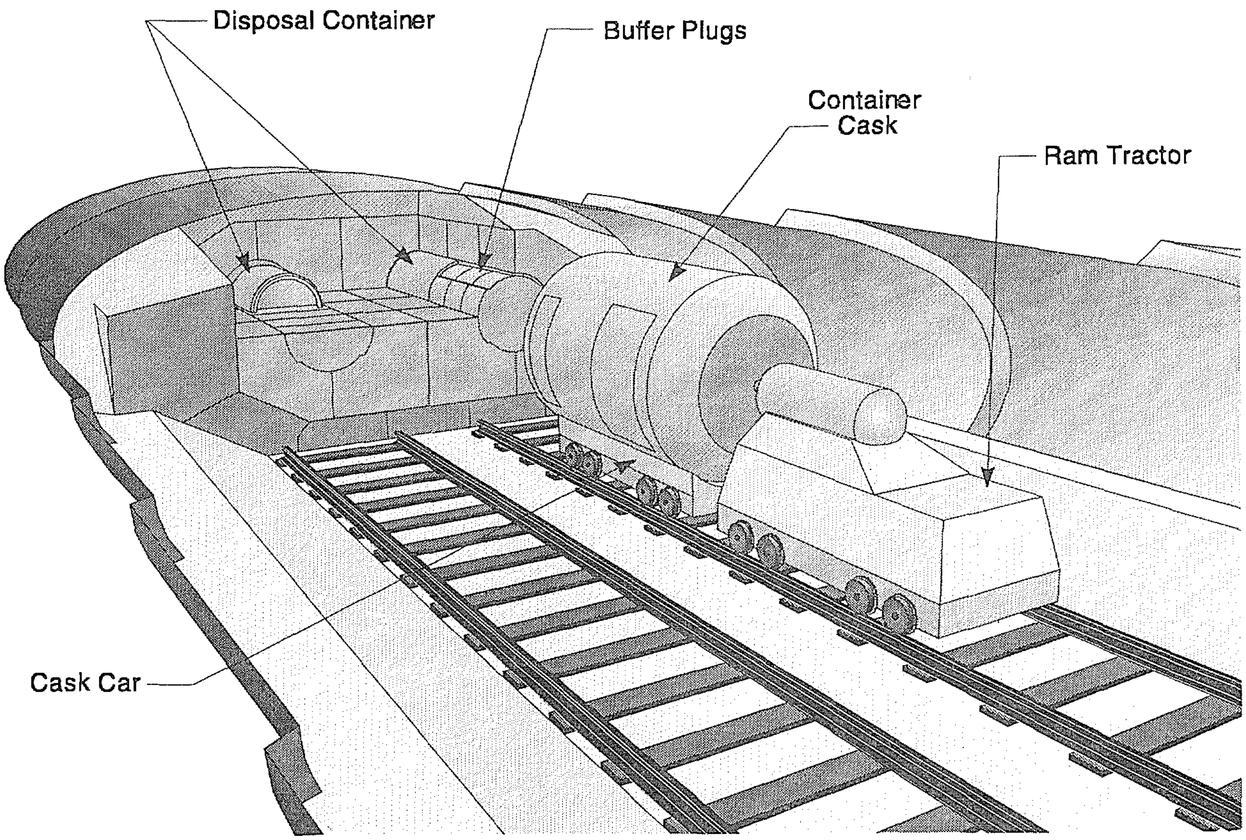


FIGURE 42: Retraction of an Empty Cask Car from the Working Face in a Disposal Room (The emplacement unit of blocks is cutaway to show general arrangement)

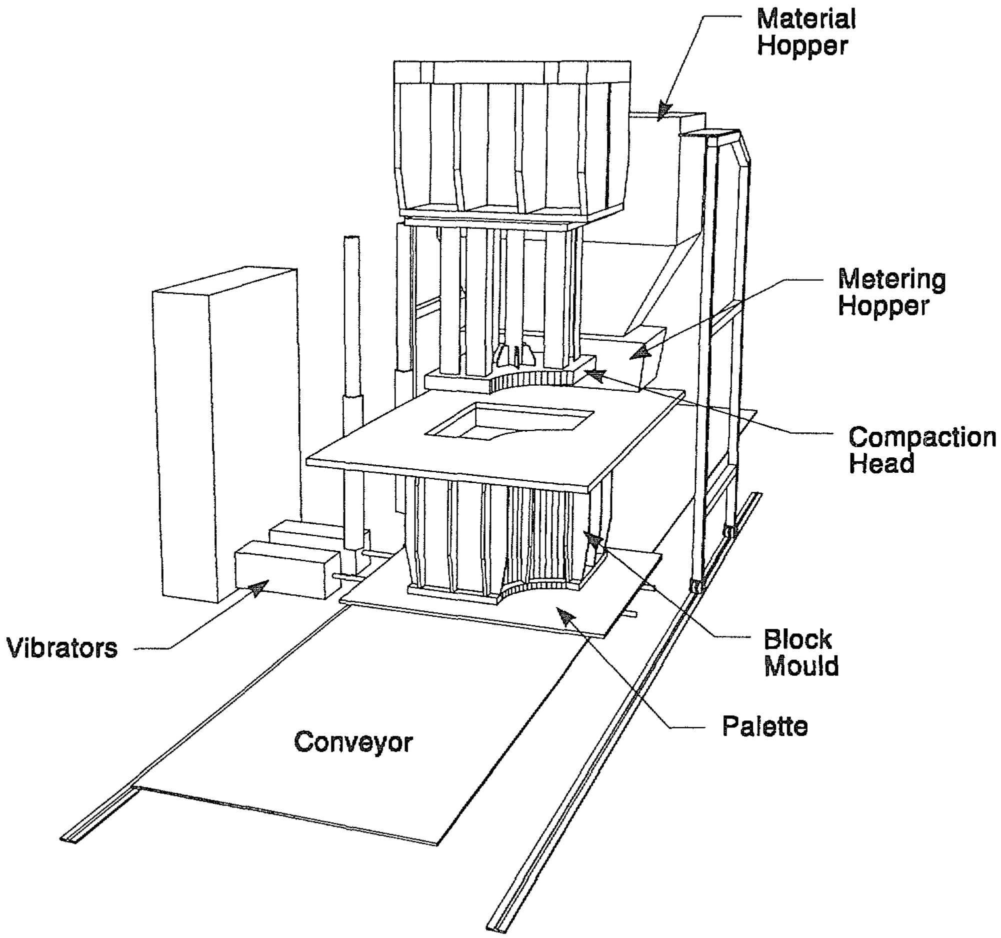


FIGURE 43: Block Compaction Machine

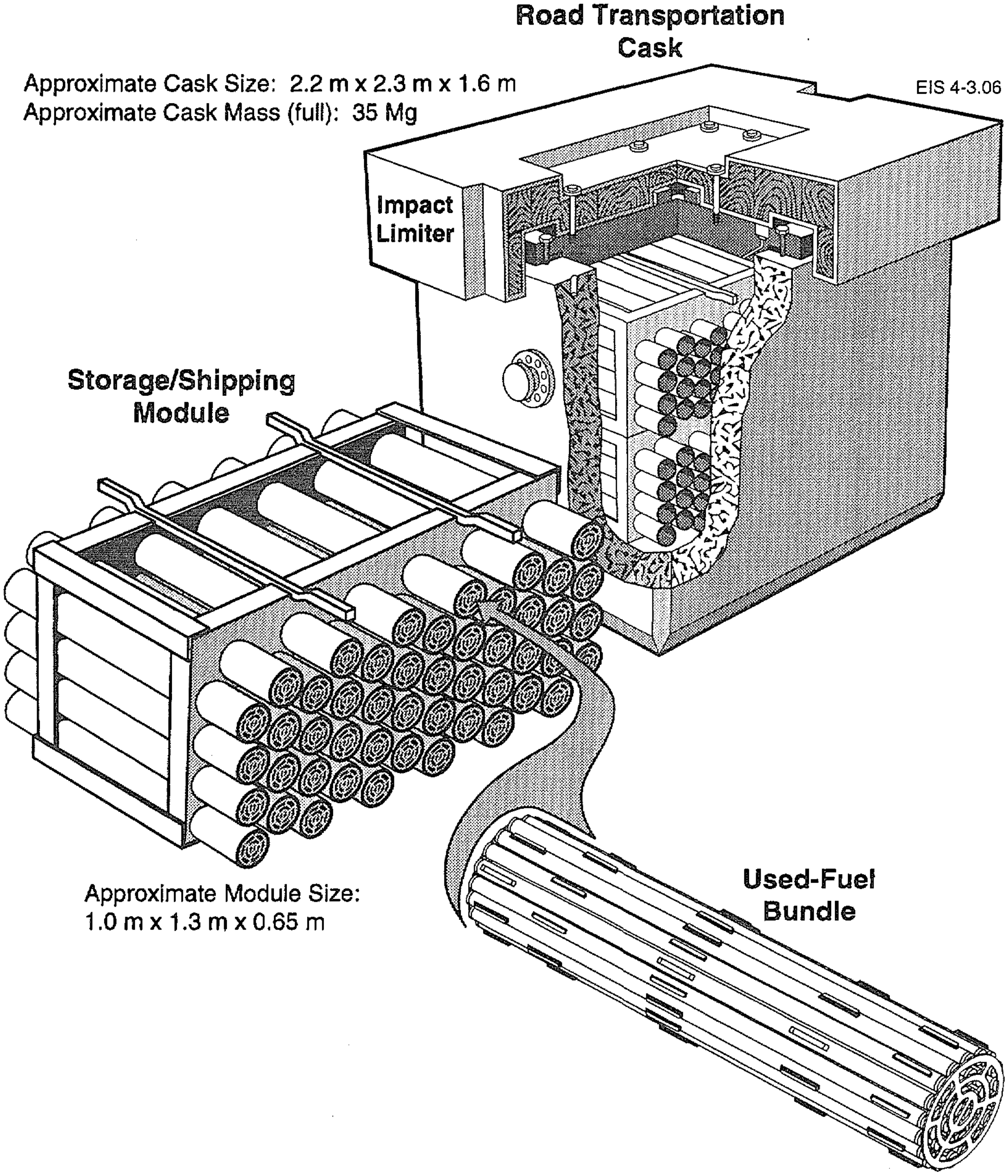


FIGURE 44: Ontario Hydro Road Transportation Cask with Two Shipping/Storage Modules and Used-Fuel Bundles (96 bundles/module)

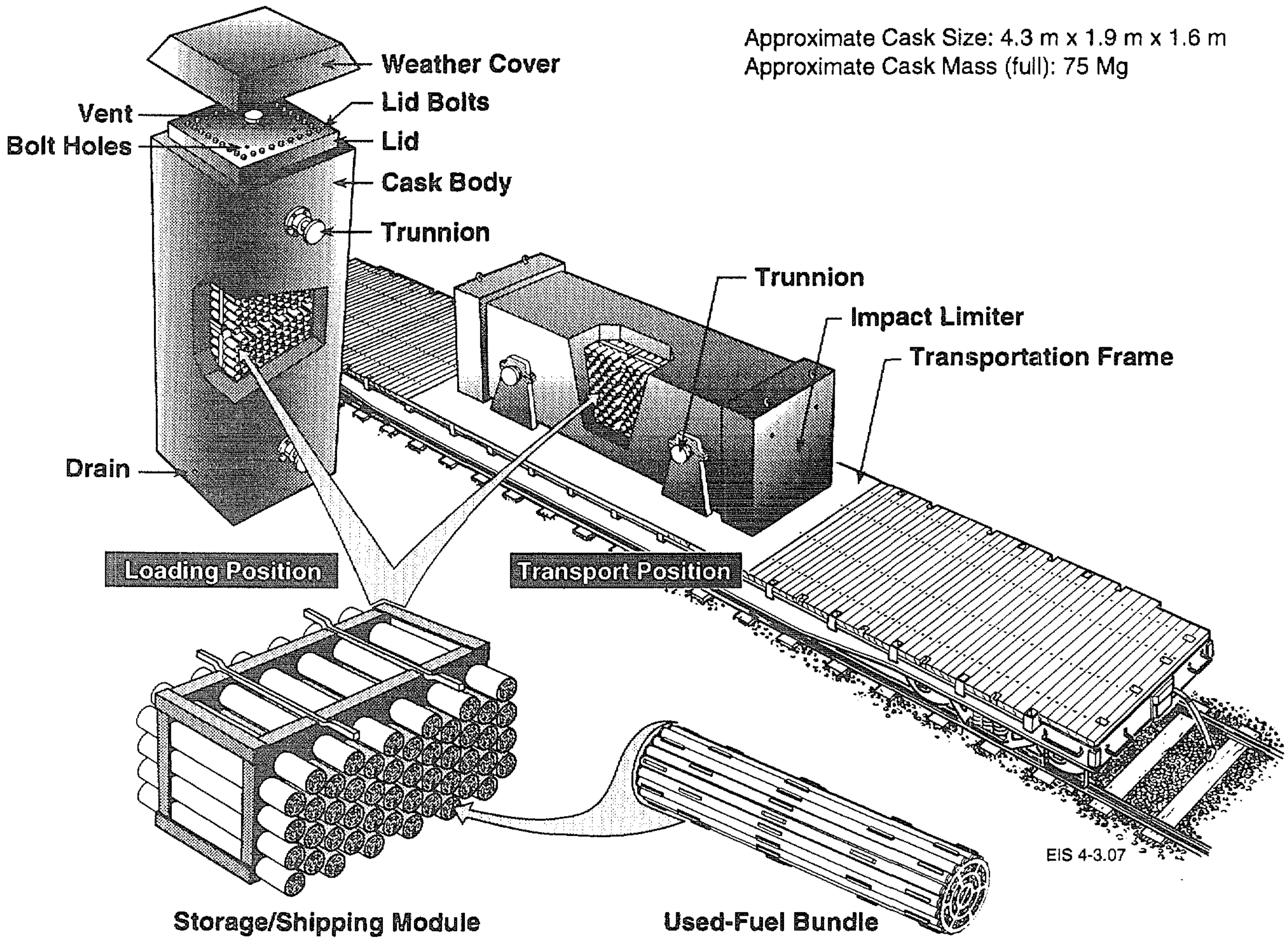


FIGURE 45: Ontario Hydro Road Transportation Cask with Six Shipping/Storage Modules and Used-Fuel Bundles (96 bundles/module)

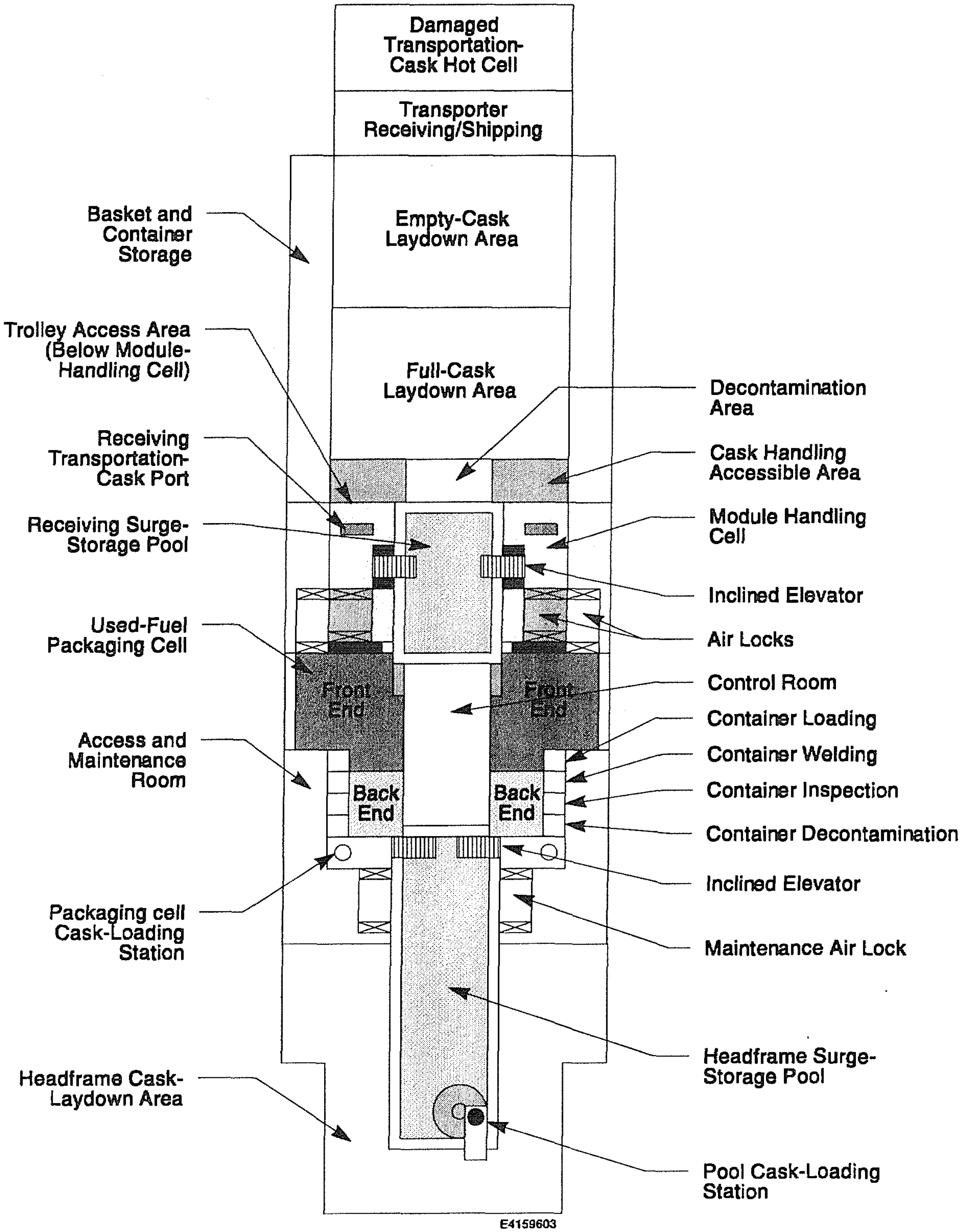


FIGURE 46: Simplified Plan of the Used-Fuel Packaging Plant (Simmons and Baumgartner 1994)

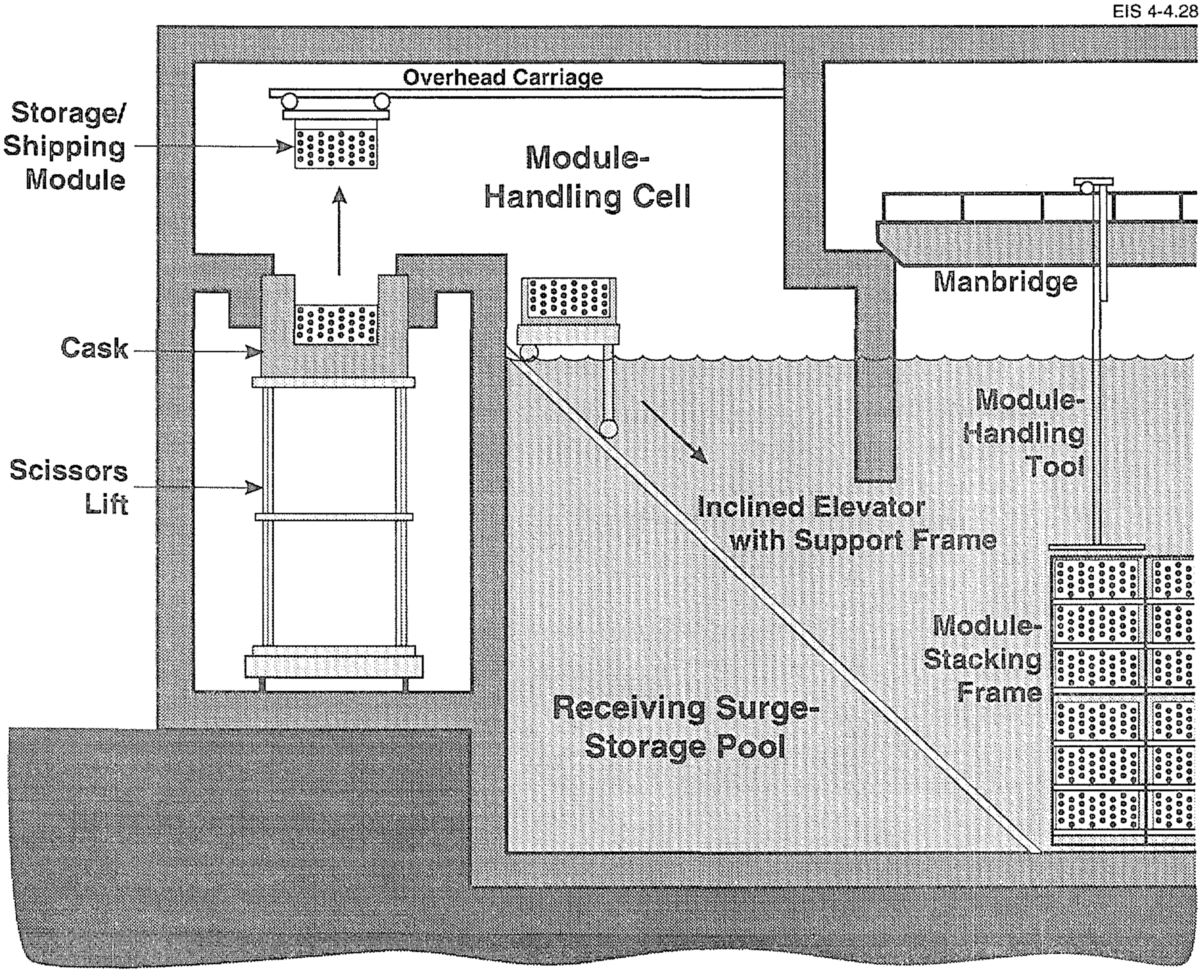


FIGURE 47: Receiving Surge-Storage Pool Module Handling (after AECL CANDU et al. 1992)

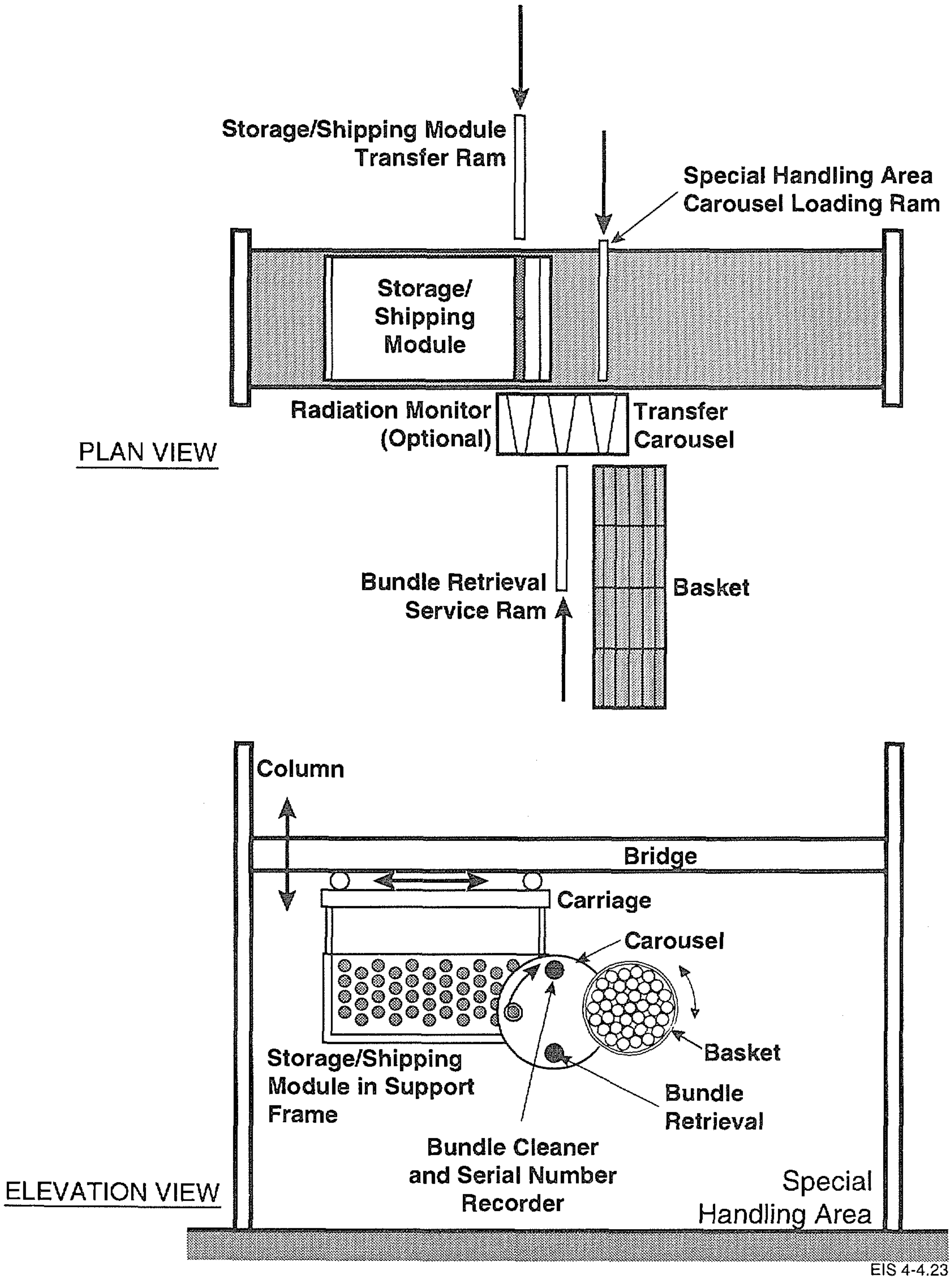


FIGURE 48: Bridge/Carriage and Used-Fuel Transfer Assemblies
(after AECL CANDU et al. 1992)

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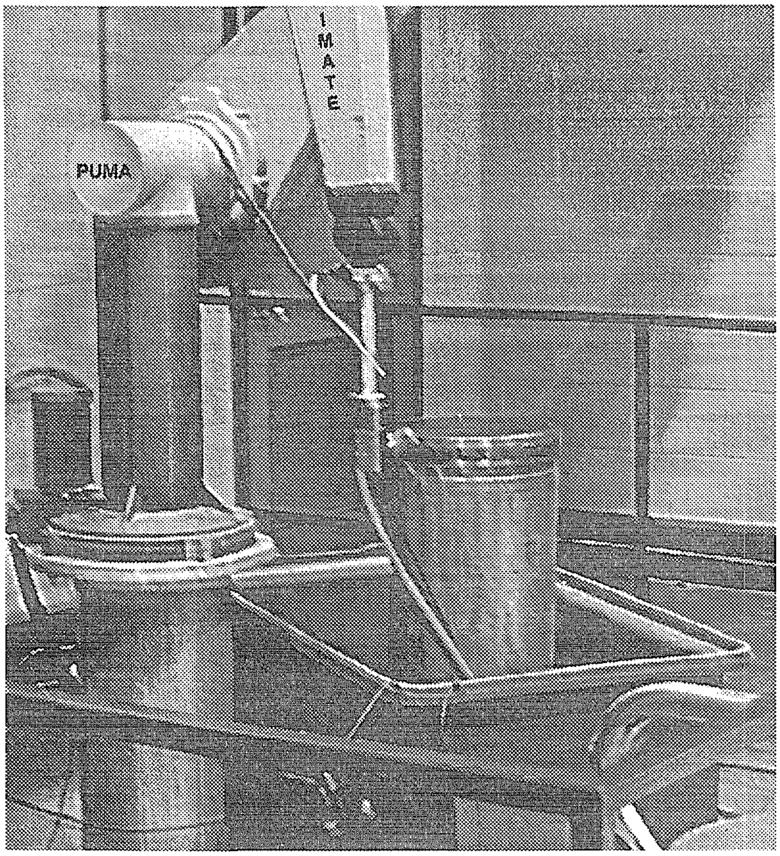
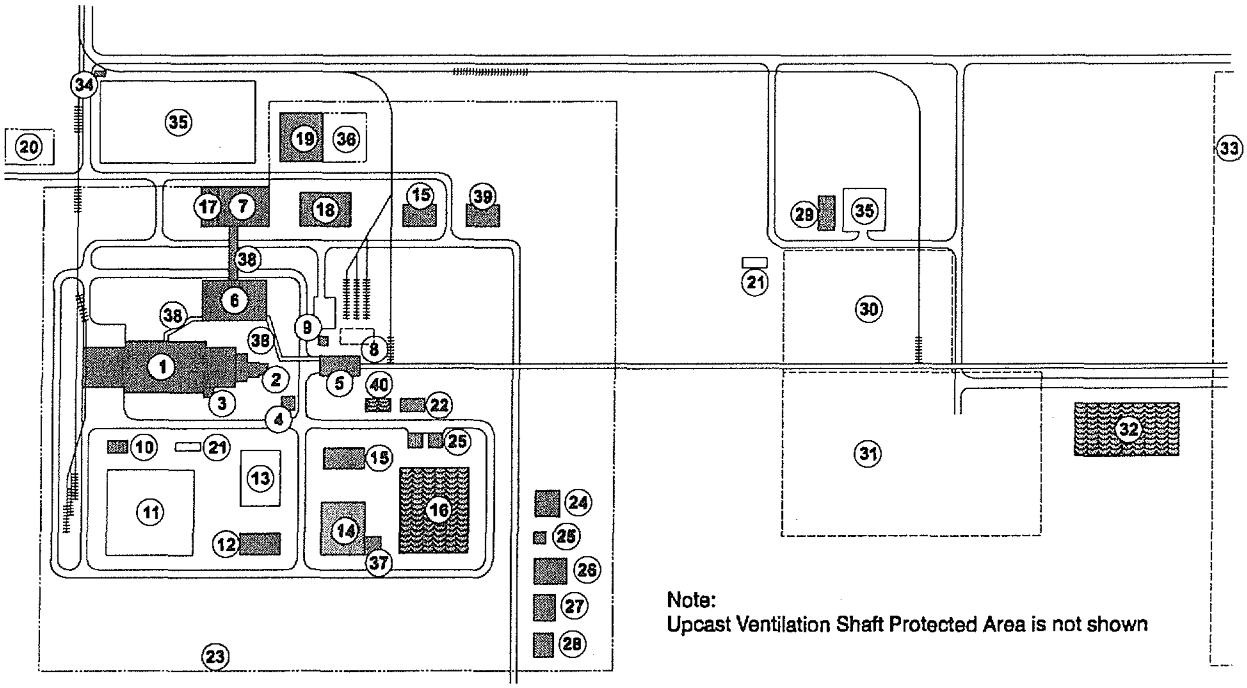


FIGURE 49: Inspection of Weldment by a Water-Jet-Coupled Ultrasonic Transducer Mounted on an Industrial Robot



1. Used-Fuel Packaging Plant
2. Waste-Shaft Headframe
3. Stack
4. Downcast Ventilation Shaft
5. Service-Shaft Complex
6. Auxiliary Building
7. Admin. Bldg. Including Firehall
8. Sealing Material Storage Bins
9. Dust Collection Bag House
10. Active-Solid-Waste Handling Bldg.
11. Waste Management Area
12. Active Liquid Waste Treatment Bldg.
13. Low-Level Liquid Waste Storage Area
14. Sewage Holding Pond
15. Garage
16. Storm Runoff Holding Pond
17. Cafeteria
18. Basket and Container Storage Building
19. Warehouse
20. Switchyard
21. Transformer Area
22. Air Compressors
23. Security Fence (Main Protected Area)
24. Powerhouse
25. Fuel Tanks
26. Water Storage Tanks
27. Water Treatment Plant
28. Pumphouse and Intake
29. Quality Control Offices and Laboratory
30. Concrete Batching Plant Area
31. Rock Crushing Plant Area
32. Process-Water Settling Pond
33. Rock Disposal Area
34. Guard House
35. Parking Area
36. Storage Yard
37. Sewage Treatment Plant
38. Overhead Corridor
39. Hazardous Materials Storage Building
40. Service-Shaft Complex Water Settling Pond

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FIGURE 50: Used-Fuel Disposal Facility Surface Layout (after AECL CANDU et al. 1992)

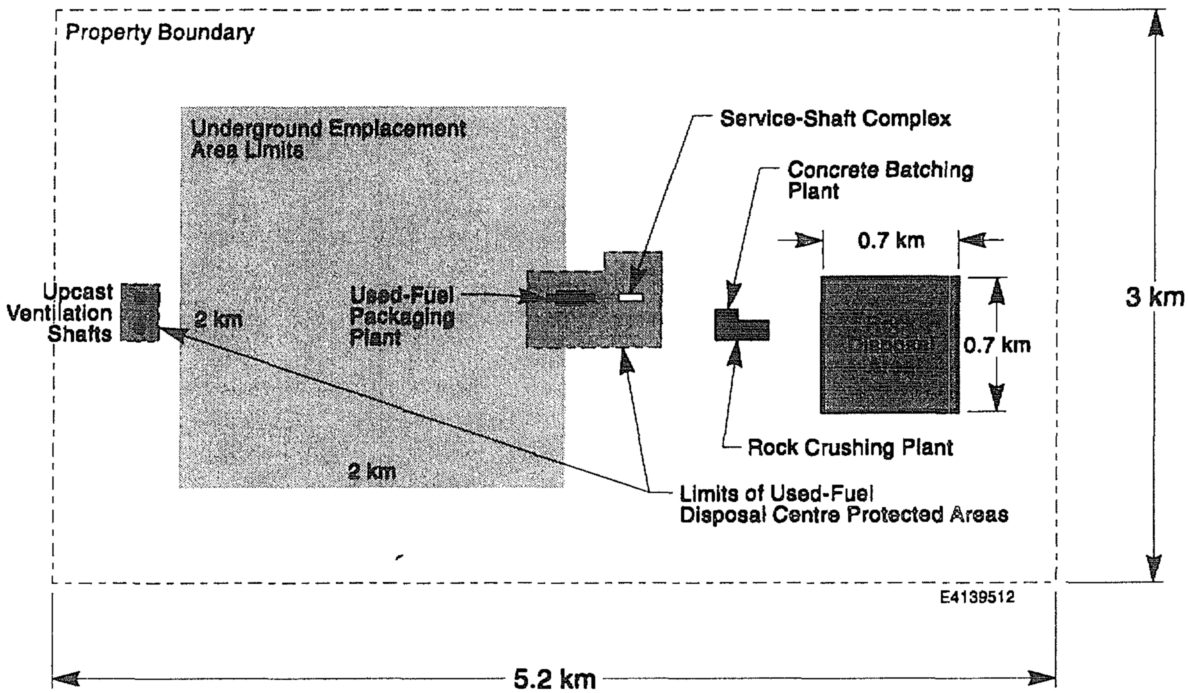


FIGURE 51: Used-Fuel Disposal Facility Land Requirements (after AECL CANDU et al. 1992)

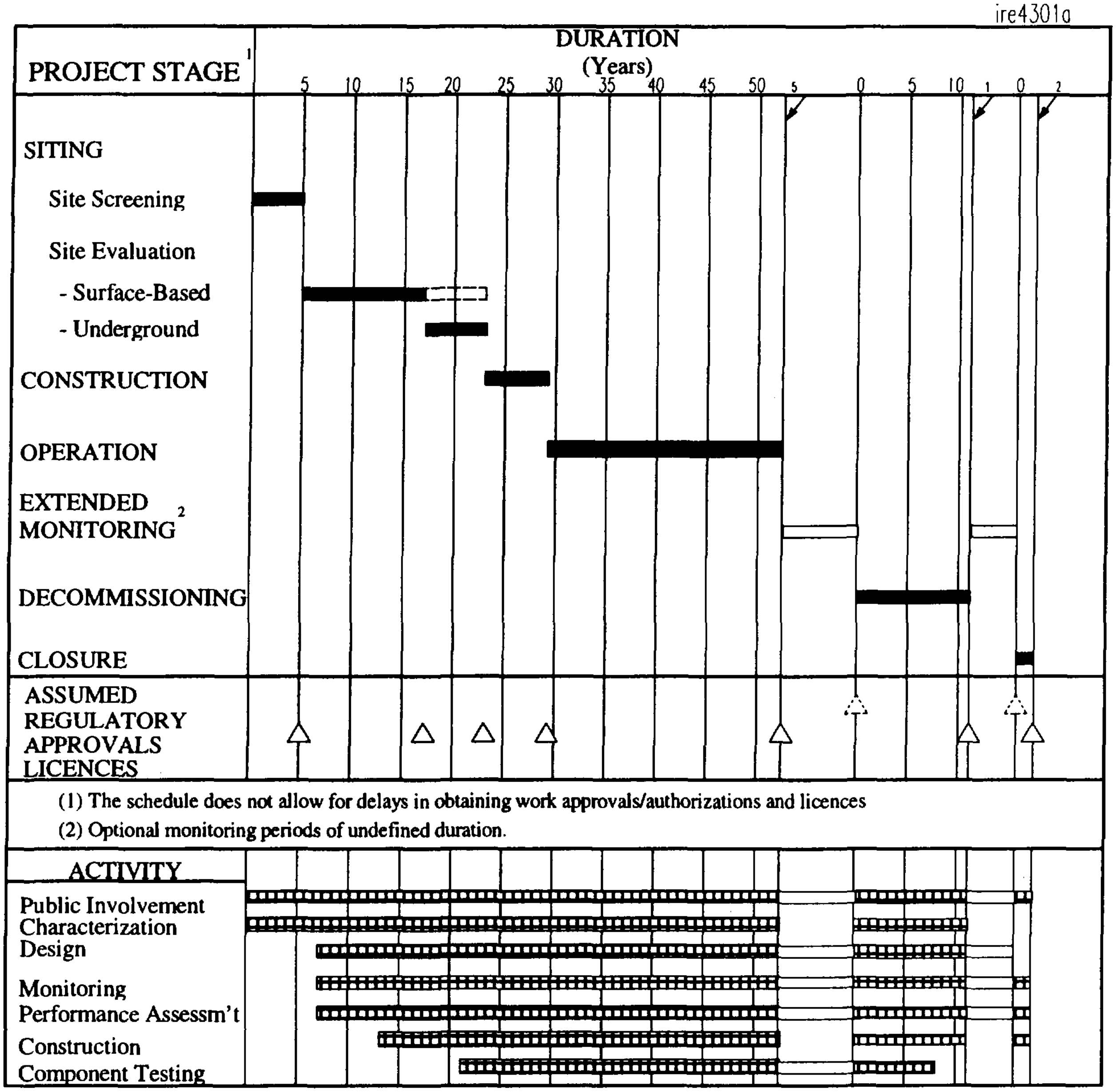


FIGURE 52: Project Schedule for a Used-Fuel Disposal Facility with a Vault at a Depth of 750 m

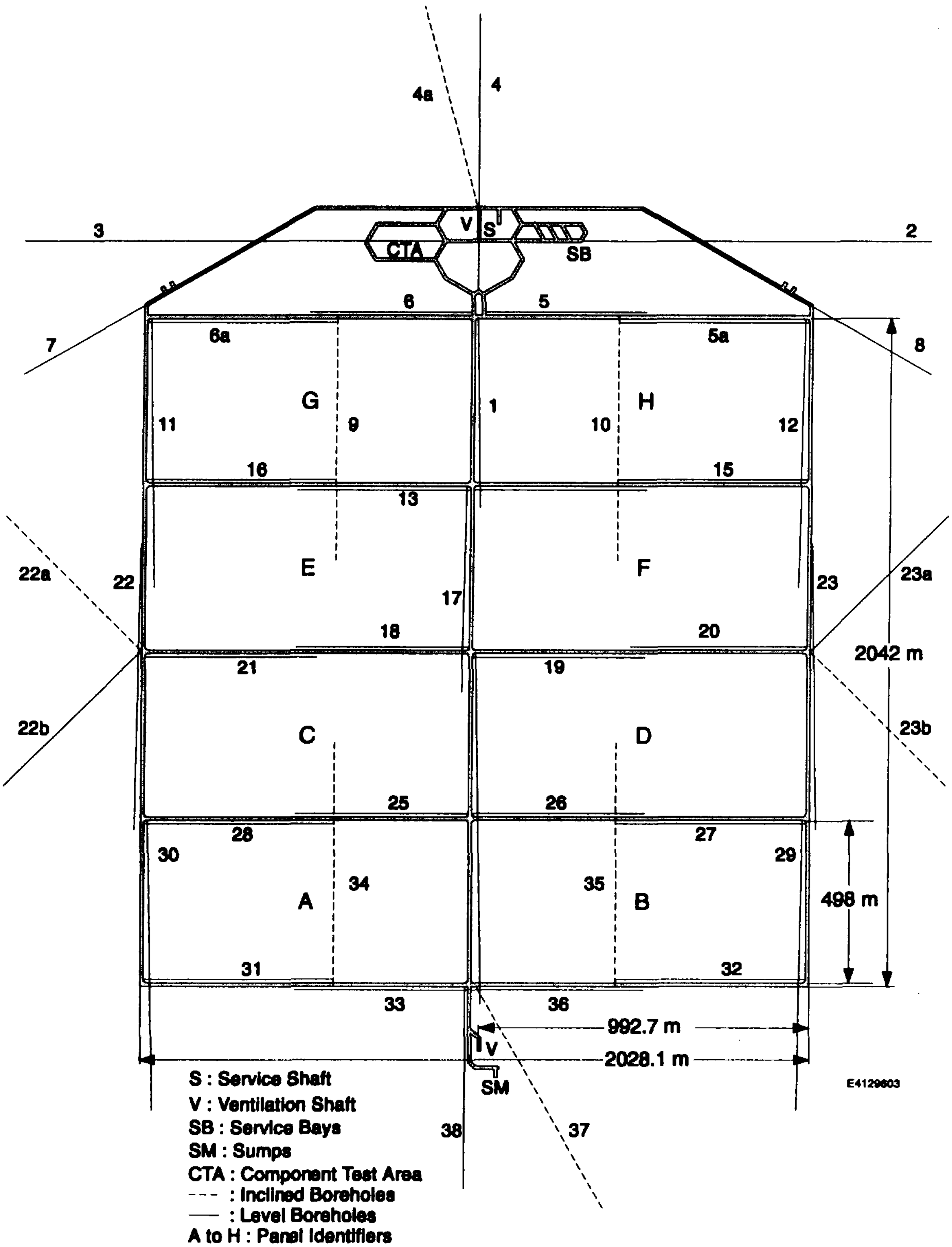


FIGURE 53: Vault Layout at the End of the Underground Evaluation Substage
(Numbering of exploration boreholes indicates sequential order of drilling)

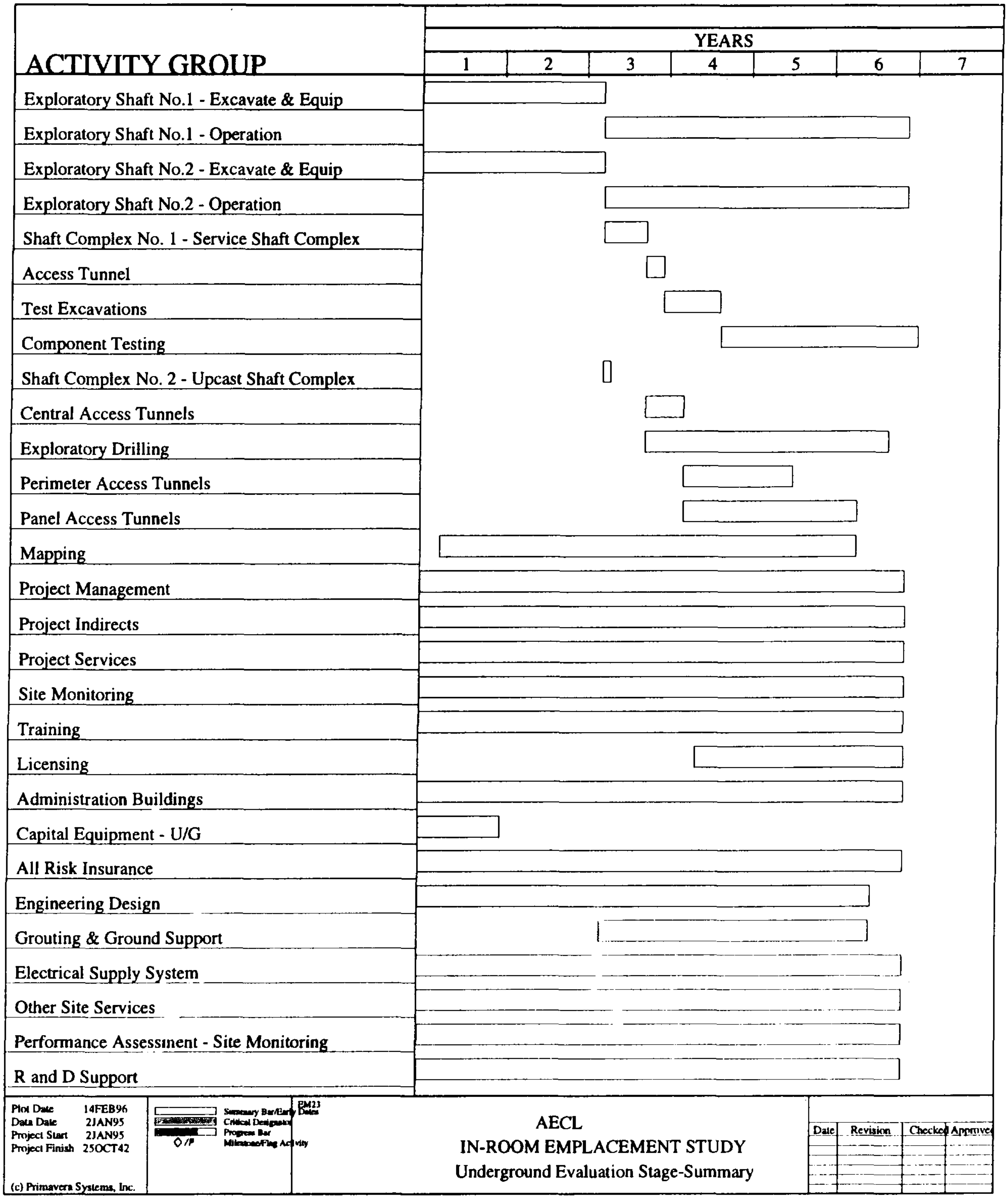


FIGURE 54: Summary Activity Schedule of the Underground Evaluation Stage

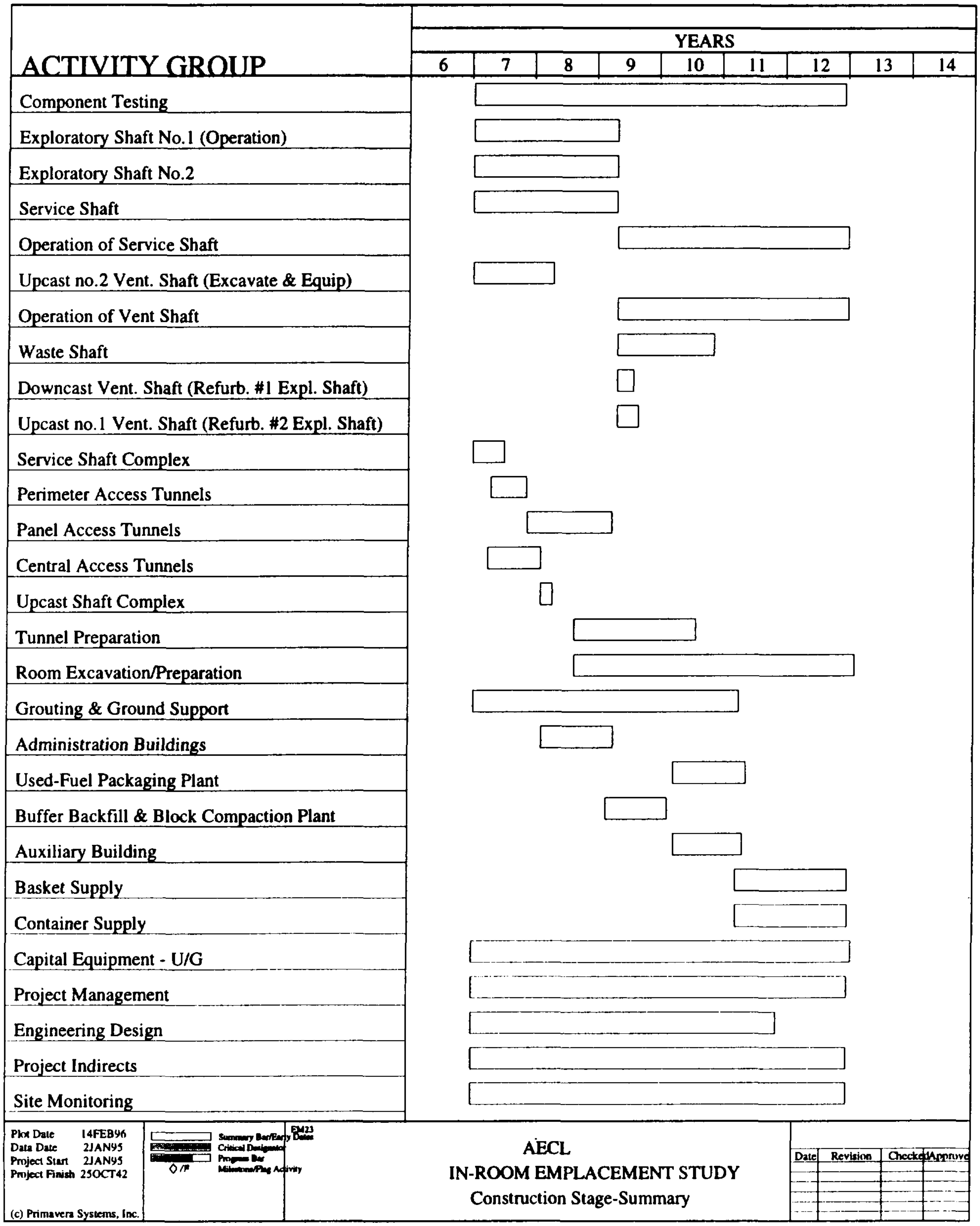


FIGURE 55: Summary Activity Schedule of the Construction Stage

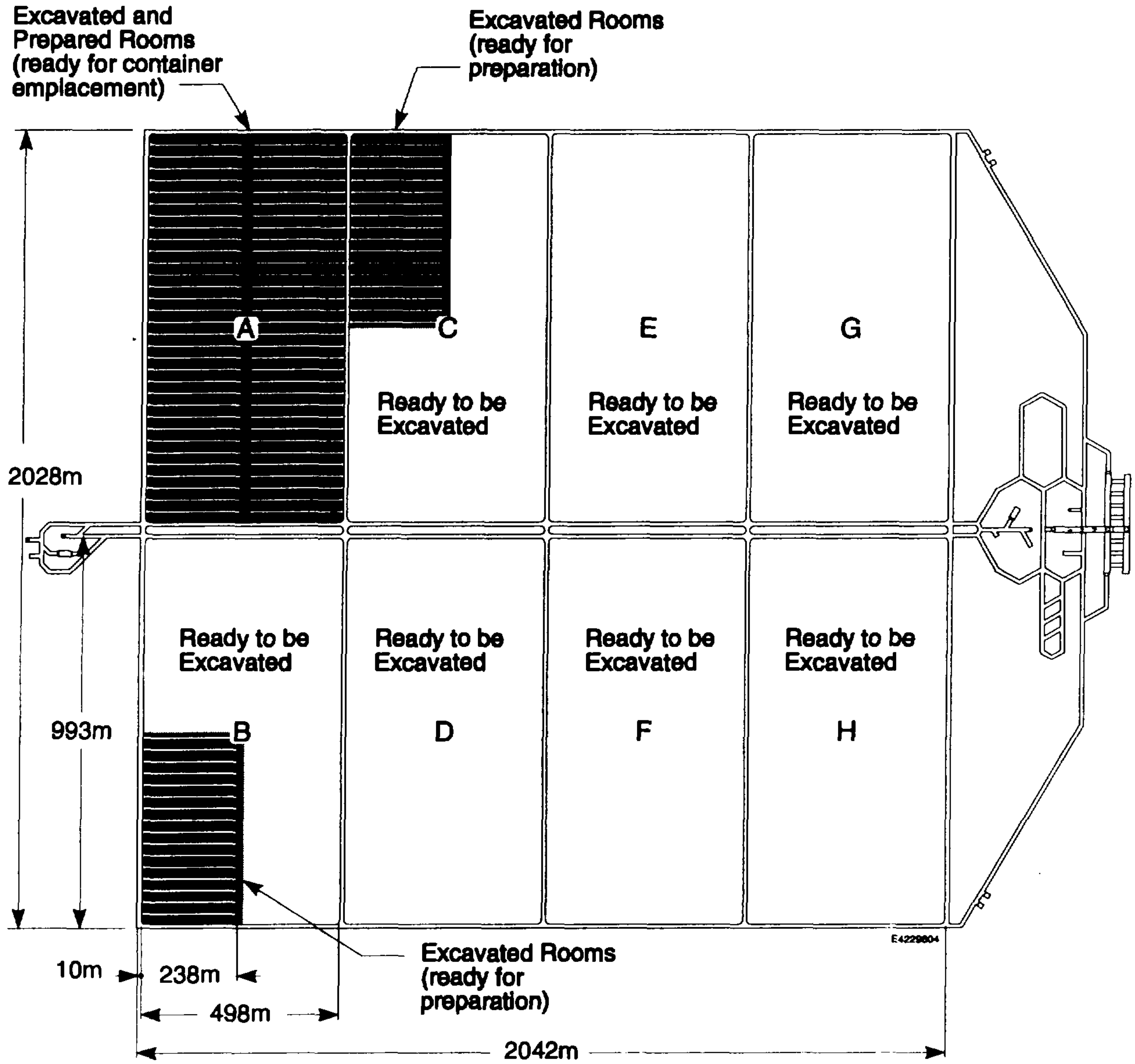


FIGURE 56: The Vault at the End of the Construction Stage

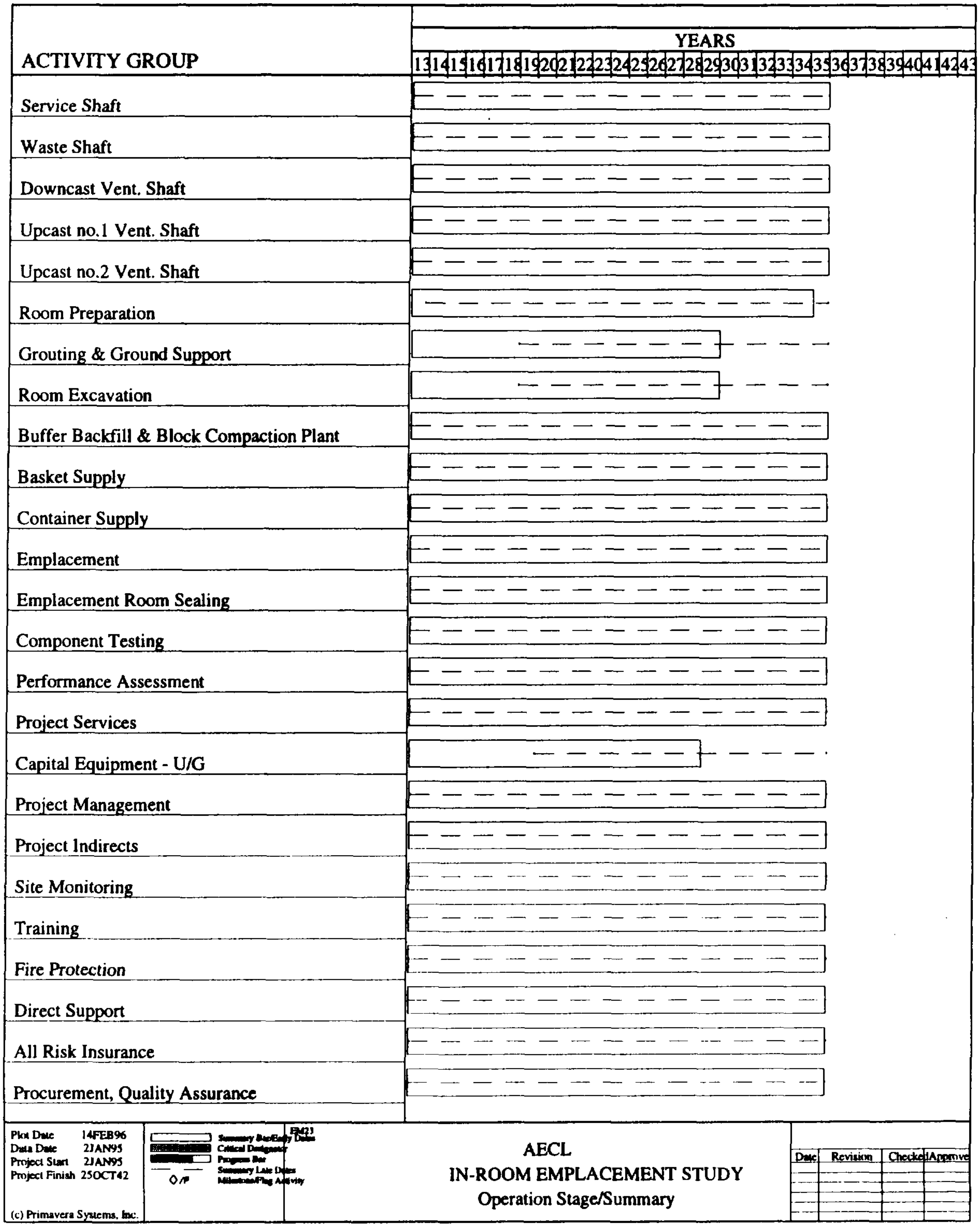


FIGURE 57: Summary Activity Schedule of the Operation Stage

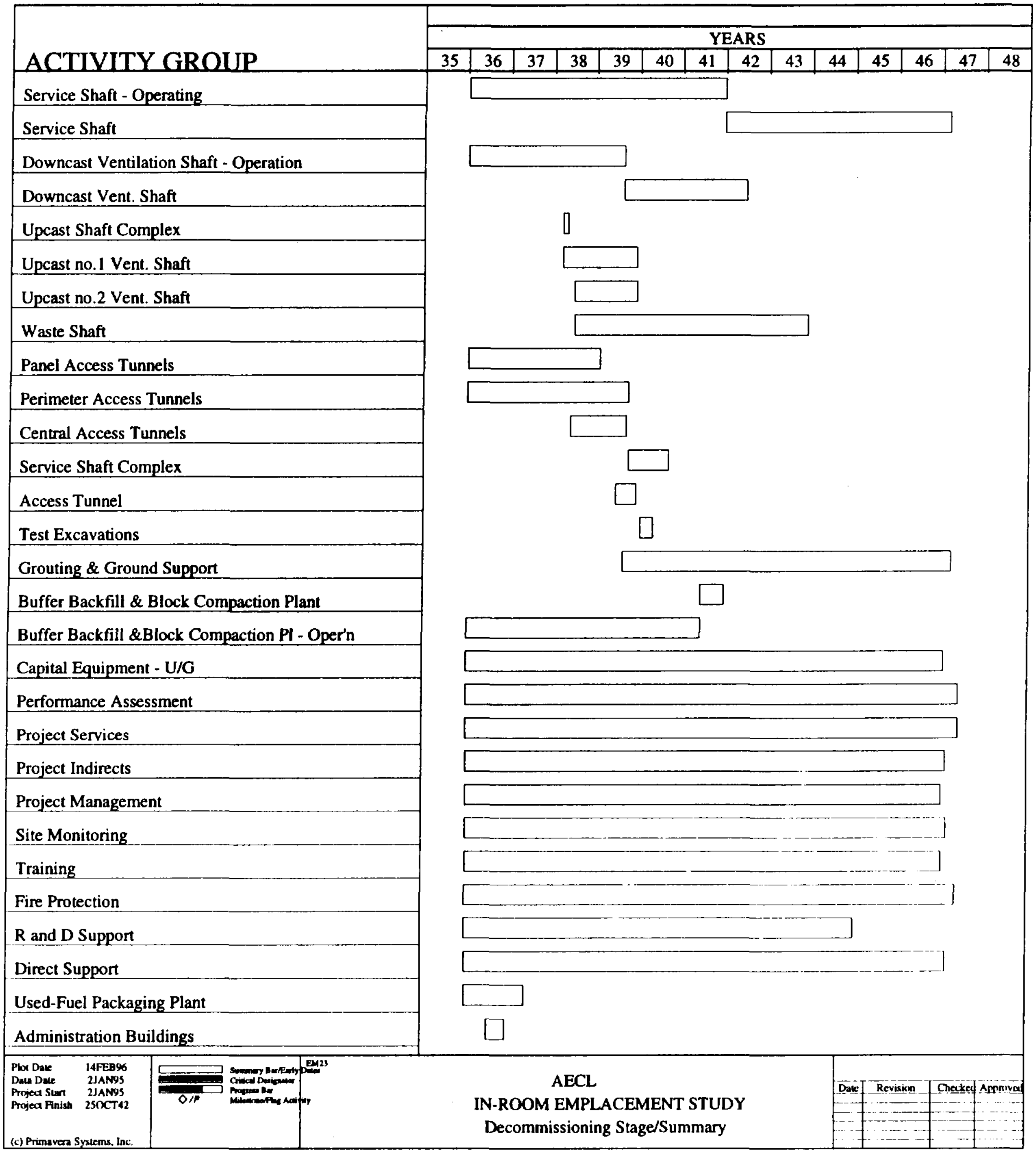
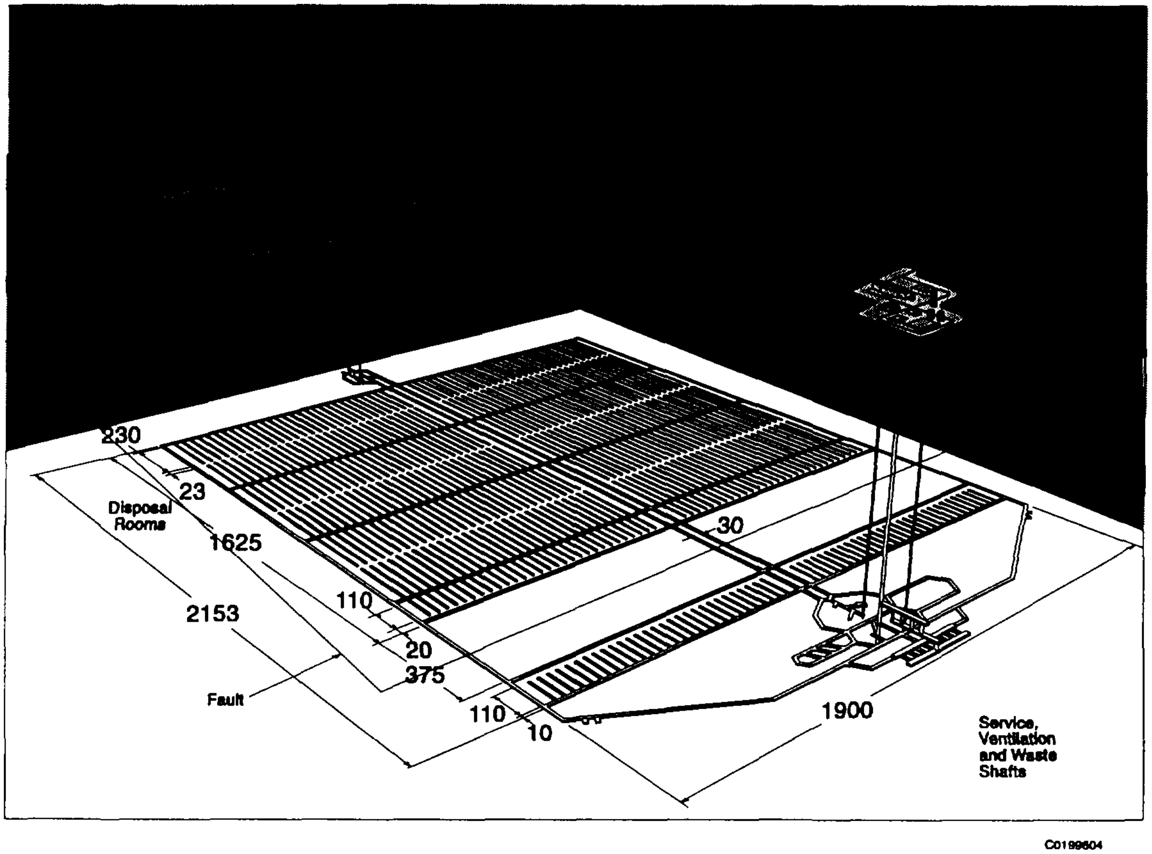


FIGURE 58: Summary Activity Schedule of the Decommissioning Stage

Large Vault Section

(1900 m x 1625 m)

3.09	km^2 area
4.07	Million Bundle Capacity (56 490 containers max.)
152	Containers/normal-length room
348	Normal-length rooms
62	Containers/short-length room
58	Short-length rooms
329.6	W/container at emplacement
6.03	W/m^2 at emplacement



Small Vault Section

(1900 m x 153 m)

0.29	km^2 area
0.26	Million Bundle Capacity (3596 containers max.)
62	Containers/short-length room
58	Short-length rooms
329.6	W/container at emplacement
4.08	W/m^2 at emplacement

Unless otherwise specified all pillars are 22.7 m

Ⓐ = 30.2 m pillars

All tunnels are 10 m

All rooms are 7.3 m

All room spacings are 30 m on centres

FIGURE 59: Disposal Vault at a Depth of 500 m in Moderately Fractured Granite

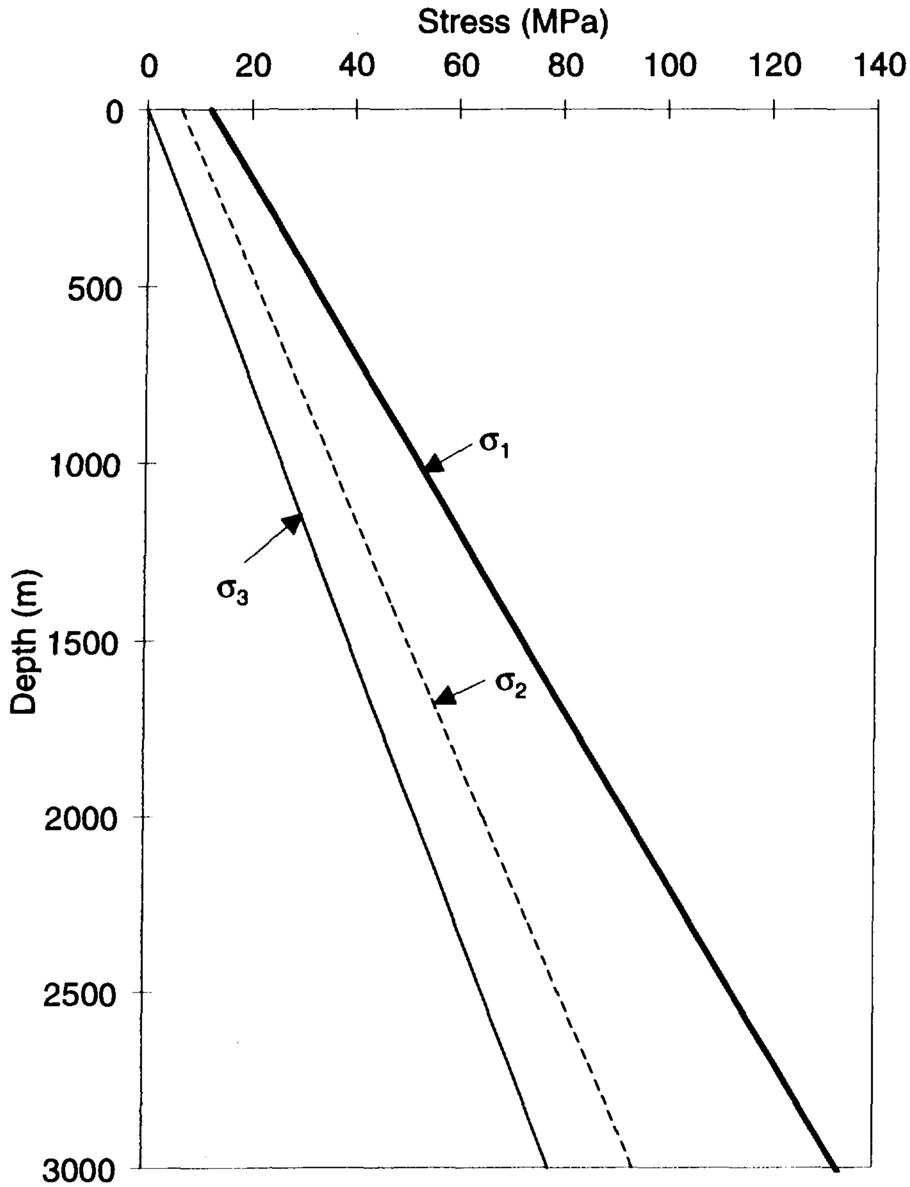


FIGURE 60: Average Ambient In Situ Stress State (after Herget and Arjang 1991)

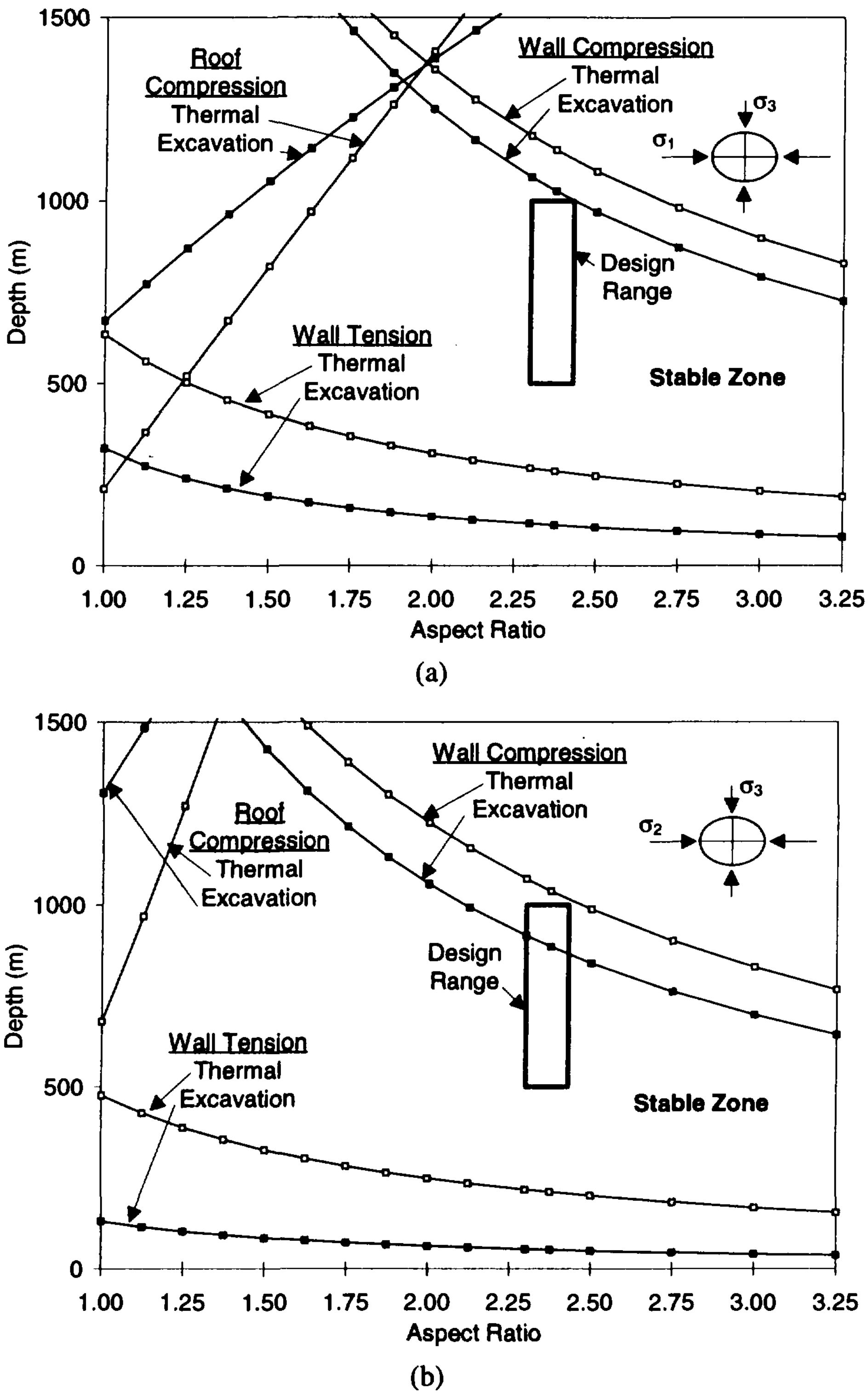
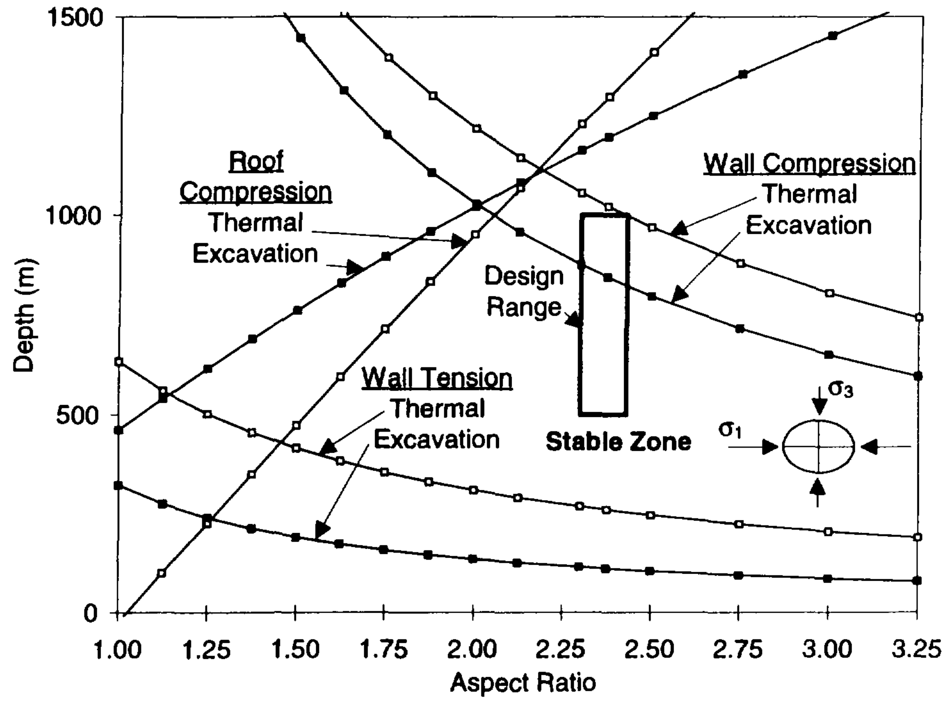
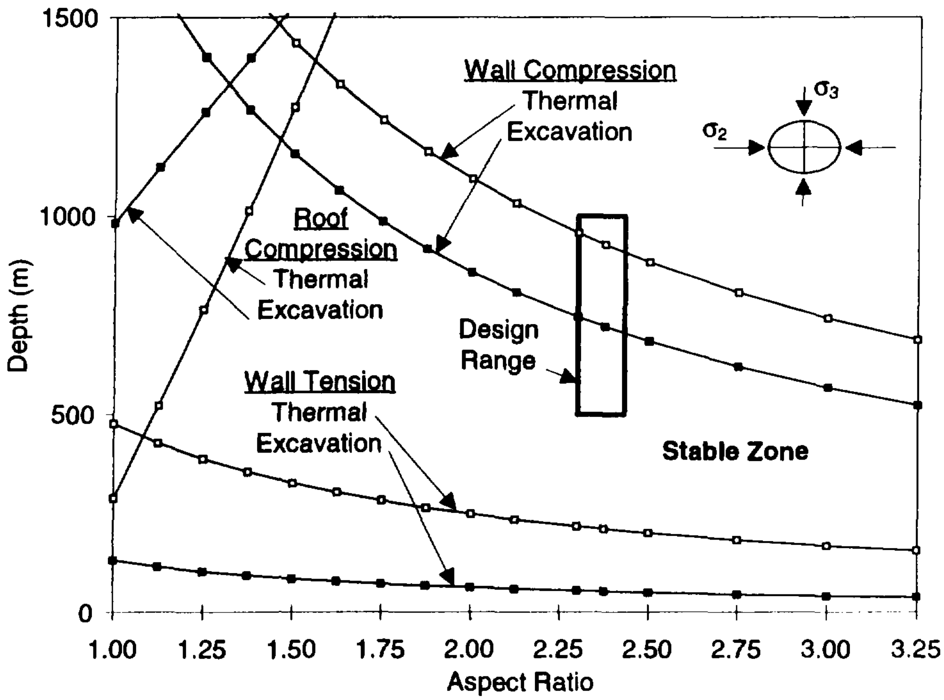


FIGURE 61: Disposal-Room Stability Design Envelopes for Average Ambient In Situ Stresses and Strength Design Limits for Sparsely Fractured Rock (σ_{EX} = 100 MPa and σ_{TM} = 150 MPa, Young's modulus = 60 GPa)



(a)



(b)

FIGURE 62: Disposal-Room Stability Design Envelopes for Average Ambient In Situ Stresses and Reduced Strength Design Limits (σ_{EX} = 80 MPa and σ_{TM} = 130 MPa, Young's modulus = 60 GPa)

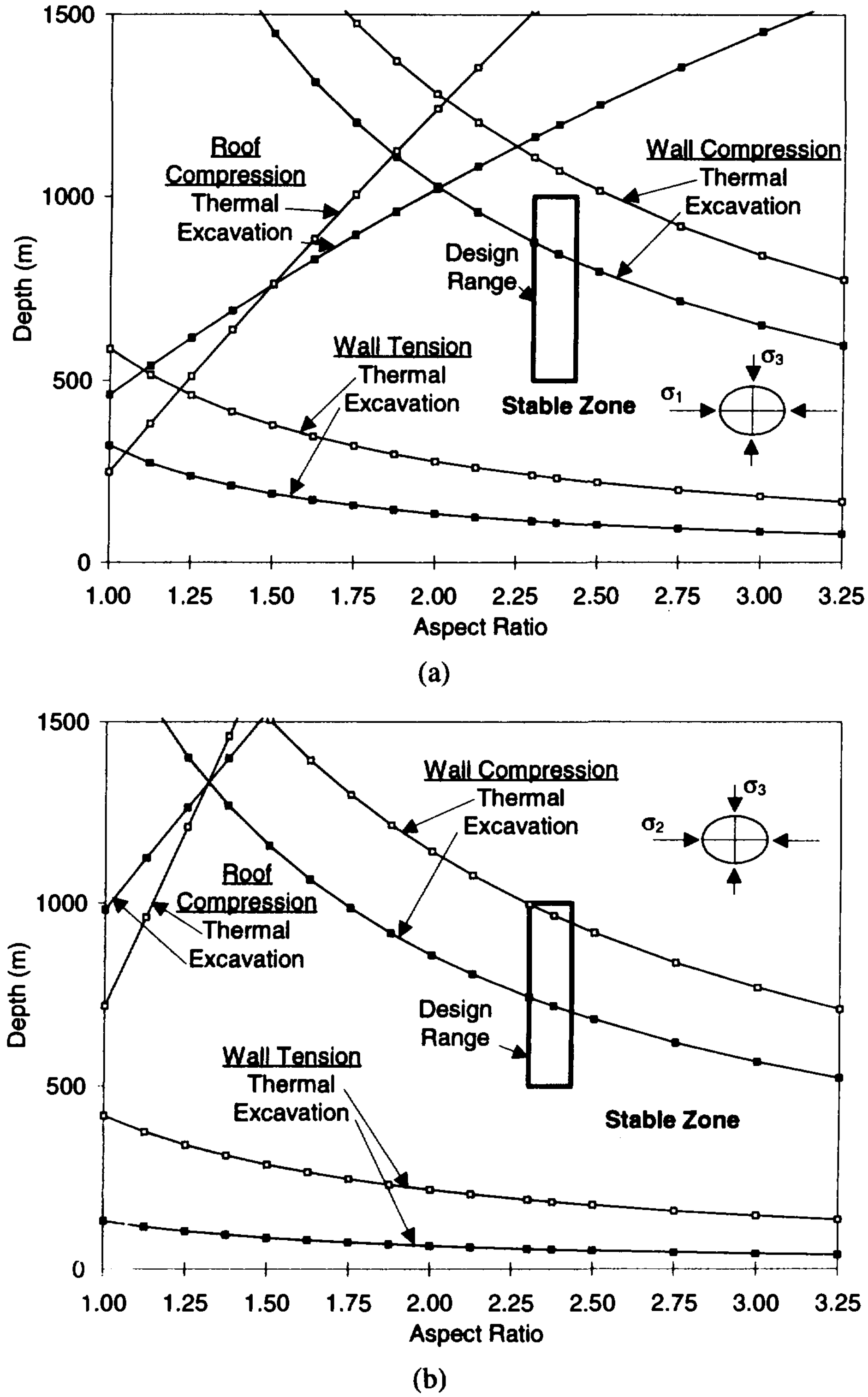


FIGURE 63: Disposal-Room Stability Design Envelopes for Average Ambient In Situ Stresses and Reduced Strength Design Limits (σ_{EX} = 80 MPa and σ_{TM} = 130 MPa, Young's modulus = 45 GPa)

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