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SUMMARY REPORT OF FIRST AND FOREIGN HIGH-LEVEL WASTE REPOSITORY CONCEPTS

Technical Report

(Working Draft 001)

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4 November, 1987

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ABSTRACT

Reference repository concepts designs adopted by domestic and foreign waste disposal programs are reviewed. Designs fall into three basic categories: deep borehole from the surface; disposal in boreholes drilled from underground excavations; and disposal in horizontal tunnels or drifts. The repository concepts developed in Sweden, Switzerland, Finland, Canada, France, Japan, United Kingdom, Belgium, Italy, Holland, Denmark, West Germany and the United States are described.

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TECHNICAL STATUS

This technical report is being transmitted in advance of DOE review and no further dissemination or publication shall be made of the report without prior approval of the DOE Project/Program Manager.

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1.0 INTRODUCTION

1.1 BACKGROUND

Battelle's Office of Waste Technology Development (OWTD) in support of the Department of Energy's (DOE) Repository Technology Program, has been assigned the job of developing a strategy for the siting of a second radioactive waste repository within the contiguous United States. The strategy is based on identifying geologic concepts with suitable isolation characteristics. A geologic concept is described by a specific set of features such as host-rock type, associated geochemistry, nature of the host-rock formation and overburden, and local and regional hydrological conditions. Identification of preferred geologic concepts involves system evaluations that include not only geologic and hydrologic factors but also engineering considerations related to the ability to safely construct, operate, and seal a repository in a given geologic setting and the ability to provide a waste package whose performance will satisfy regulatory containment and controlled release requirements.

1.2 PURPOSE

It is the object of this report to deal with the engineering considerations related to the ability to safely construct, operate and seal a repository. The report reviews and summarizes the configurations of the first repository projects in the United States and repository programs abroad and how they selected these configurations and presents some of the advantages and potential drawbacks of the various concepts.

1.3 SCOPE

The review is concerned mainly with the high-level waste and/or spent fuel repository configurations, although the low- and intermediate-level concepts for some of the foreign nuclear waste disposal programs have been briefly outlined for completeness. It also deals primarily with reference repository designs and emplacement concepts both for domestic and foreign repository programs although some of the alternate concepts that have been considered are also summarized. The report describes the repository configurations as well as construction and operations related aspects, outlines briefly the nuclear waste disposal program for the different countries, and discusses some of the potential advantages and problems of the concept.

1.4 REPORT STRUCTURE

The review was conducted basically by country or national nuclear waste disposal program and the report structure evolved in basically the same fashion as this made description and summary the easiest. The following describes the various sections in the report:

- INTRODUCTION
  - Background, purpose, scope and report structure
- SOURCES AND TYPES OF NUCLEAR WASTE
  - Summary of types of high-level wastes and how they are produced
- THE DEEP GEOLOGIC DISPOSAL CONCEPTS
  - Description of the some of the features of the deep geologic disposal concept
- MAJOR FACTORS INFLUENCING NUCLEAR WASTE DISPOSAL CONCEPTS
  - Summary of major factors affecting disposal concept designs
- RESEARCH PROGRAMS
  - Descriptions of the major investigation programs and facilities related to nuclear waste disposal
- CRYSTALLINE ROCK FORMATIONS
  - Descriptions of the repository concepts in countries considering crystalline host rocks. The descriptions include geographical and geologic summaries, nuclear waste disposal strategy, repository configuration and construction, waste treatment and emplacement, backfilling and sealing, summary, and low- and intermediate-level concepts
- SEDIMENTARY ROCK FORMATIONS
  - Descriptions of the repository concepts in countries considering sedimentary rock formations. Subsections are as described for the crystalline concepts
- EVAPORITE FORMATIONS
  - Descriptions of the repository concepts in countries considering evaporite formations. Subsections are as described for the crystalline concepts

- BASALT FORMATIONS

  - Descriptions of the repository concepts in countries considering basalt formations. Subsections are as described for the crystalline concepts

- TUFF FORMATIONS

  - Descriptions of the repository concepts in countries considering tuff formations. Subsections are as described for the crystalline concepts

- OTHER CONCEPTS

  - Descriptions of concepts (geologic and non-geologic) not considered mined geologic concepts or that have been found not viable at this time

- SUMMARY

  - General observations about the concepts studied.

- REFERENCES

2.0 SOURCES AND TYPES OF NUCLEAR WASTES

This chapter describes the types of long-lived nuclear wastes arising from various nuclear applications. These applications include nuclear power generation, nuclear defense programs, research programs, and industrial and medical applications. The nuclear wastes produced by these uses varies widely in terms of form and activity content, all the way from virtually inactive trash to spent fuel, which has a very high radioactivity and heat output. The different waste forms thus impose different demands on handling and final disposal requirements.

Nuclear wastes are generally categorized as low-level, intermediate-level and high-level wastes. Low-level wastes can be handled and stored in simple packages with minor protection requirements. Intermediate-level wastes must be radiation-shielded for safe handling and require isolation for some time during storage (e.g., ASSE II mine). High-level wastes require not only radiation shielding but also cooling for a certain time in order to allow safe isolation. It is these isolation concepts for high-level wastes that are subject of this report.

The following subsections give a short summary of the types of high-level nuclear wastes that are generated in the nuclear fuel cycle and require special treatment and isolation techniques due to their intense radioactivity and heat production.

2.1 SPENT FUEL

Most of the radioactive material that is formed in nuclear power plants is present in the spent fuel. A fuel assembly (Figures 2.1.1 to 2.1.3) can contain up to 500 kg of uranium, depending on the type of reactor and the manufacturer. The CANDU (CANada Deuterium Uranium) reactor fuel rods (Figure 2.1.3) use natural uranium while the other fuel elements use enriched uranium fuel. Tables 2.1.1 and 2.1.2 summarize the makeup of fuel assemblies for a boiling water reactor and a pressurized water reactor. The examples shown in the tables are for fuel used in the Swedish reactors. Spent fuel consists of irradiated fuel removed from a commercial reactor (after 3 or 4 years of use) or special fuels from test or research reactors. The spent fuel consists mainly of unfissioned uranium, while most of the radioactivity is due to the fuel's content of fission products and transuranics. The major difference between the radionuclide content of spent fuel and high-level waste is the presence in spent fuel of unextracted uranium and plutonium. The high level of radioactivity of spent fuel means that it continues to emit heat for a long time after it has been discharged from the reactor. Unreprocessed fuel can be stored in water-cooled facilities or dry storage for a long time before final disposal to allow decay of the radioactivity and heat. Storage for up to 50 years is planned in some countries (e.g., Canada, Sweden, Finland). Additional packaging and conditioning must be envisaged for the disposal of spent fuel.

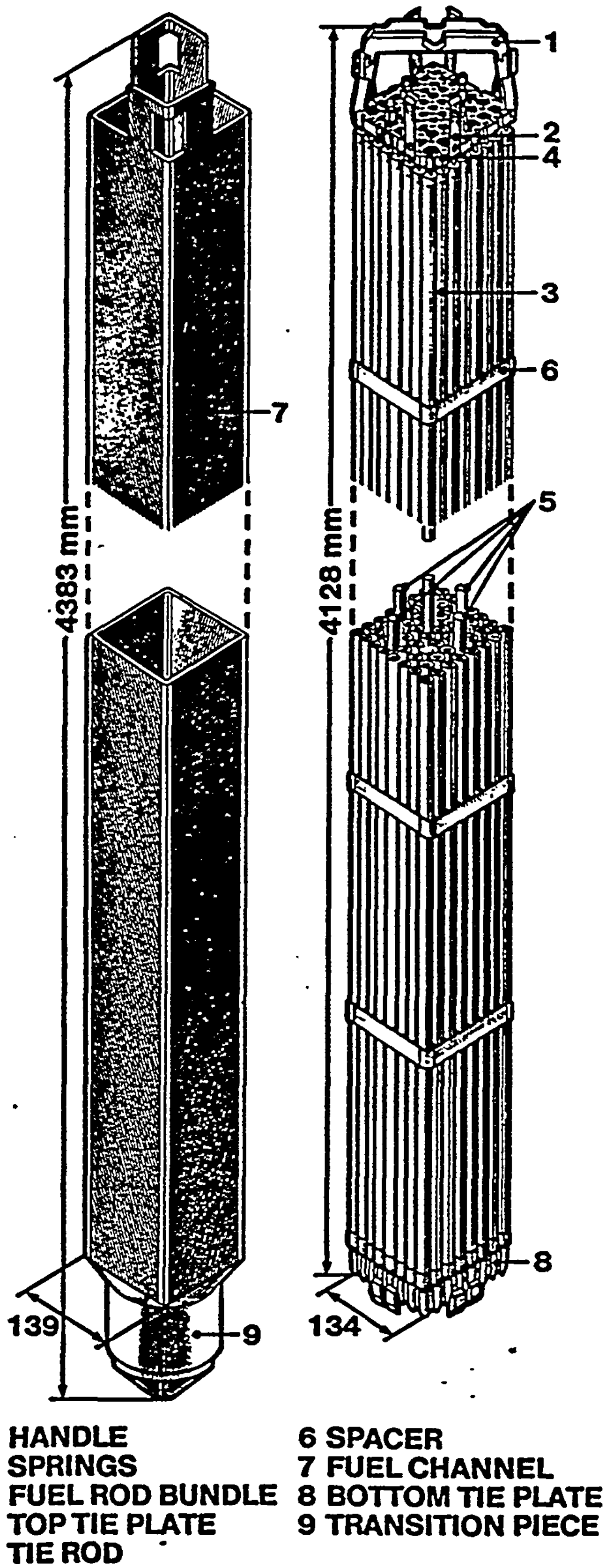
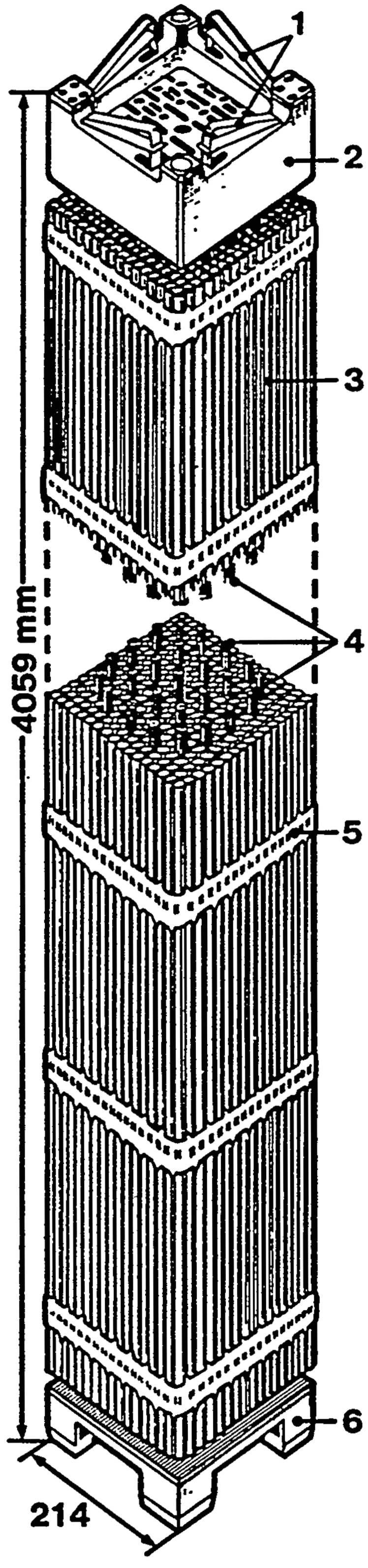
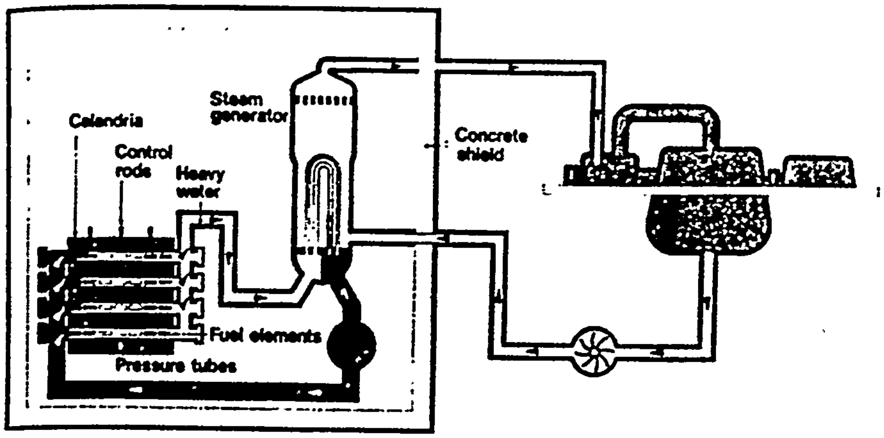


Figure 2.1.1. Boiling Water Reactor Fuel (Sweden)



1 SPRINGS
2 TOP NOZZLE
3 FUEL ROD
4 CONTROLROD GUIDE TUBE
5 SPACER
6 BOTTOM NOZZLE

Figure 2.1.2. Pressurized Water Reactor Fuel (Sweden)



Pressurised heavy water reactor (CANDU)

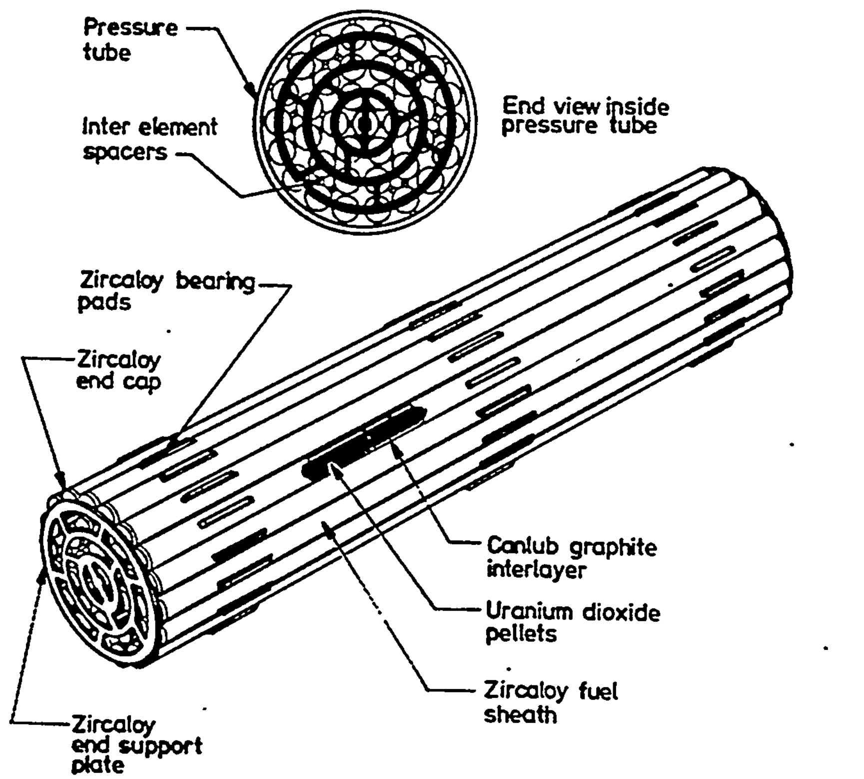


Figure 2.1.3. CANDU Reactor Fuel

Comparison between the main data for AA, Exxon and KWU-fuel elements. (The weights are given per element where nothing else is indicated).

	ASEA-ATOM (F3)	EXXON (B1)	KWU (F1)
Fuel bundle			
Length (mm)	4128	4127	4126
Width (mm)	151	147	151
Weight (kg)	256	262	258
Weight of uranium	178	174	176
Top plate incl. handle and springs (kg)	1,5 SS 0,14 Inconel	1,5 SS 0,18 Inconel	1,6 SS
Bottom tie plate (kg)	1,1 SS	1,0 SS	1,2 SS
Spacers (kg)	0,87 Inconel	1,5 Zr-2 0,3 Inconel	1,7 Zr-4 0,3 Inconel
Fuel rods number	63	63	62
Length (mm)	3998	4011	4016
Outer diameter (mm)	12,25/11,75	12,29	12,3
Cladding (kg/rod)	0,76 Zr-2	0,86 Zr-2	0,79 Zr-2
Uranium dioxide (kg/rod)	3.265/2.959	3.136	3.214
Plenum spring (kg/pc)	0,035 SS	0,043 Inconel	0,04 SS
Spacer capture rod number	1	1	2
Weight (kg/rod)	0,77 Zr-2	2,87 Zr-2	0,94 Zr-2
Springs	0,04 Inconel	0,15 Inconel	
Nuts	0,06 SS	0,07 SS	0,1 SS
Total amount of material in the fuel bundle			
Uranium oxide (kg)	202	197	199
Zircaloy (kg)	48,4	58,4	52,5
Inconel (kg)	1,1	3,3	0,5
Stainless steel (kg)	4,9	2,5	5,4
Al_2O_3 (kg)	-	0,2	0,2

Table 2.1.1. Boiling Water Reactor Fuel Assembly Makeup

Data for different PWR-fuel elements (the weights are given per fuel element where nothing else is indicated).

	Ringhals 3	Ringhals 2	
	Westinghouse	Westinghouse	KWU
Type	17x17	15x15	15x15
Length (mm)	4059	4059	4057
Width (mm)	214	214	214
Weight (kg)	665	655	645
Weight of uranium (kg)	464	460	435
Top nozzle (kg) incl. springs	7,0 Stainless Steel 0,5 Inconel	8 SS	6,7 SS
Bottom nozzle (kg)	5,7 Stainless Steel	6 SS	5,7 SS
Spacers (number)	8	7	7
Weight	6,3 Inconel 718	6 Inconel	5,6 Inconel
Fuel rods (number)	264	204	204
Length	3852	3856	3864
Outer diameter (mm)	9,50	10,72	10,77
Cladding (kg/rod)	0,4 Zircaloy-4	0,5 Zr-4	0,59 Zr-4
Uranium dioxide (kg/rod)	1,99	2,55	2,42
Plenum spring (kg/pc)	0,02 Stainless Steel	0,02 SS	0,02 SS
Guide thimbles (number)	25	21	21
Length (mm)	3900	3900	3918
Outer diameter (mm)	12,3	13,9	13,9
Weight (kg)	9,6 Zircaloy-4	9 Zr-4	13,2 Zr-4
Screws	0,2 Inconel 718	0,2 Inconel	0,07 SS

Total amount of material in the fuel bundle

Uranium dioxide	526	521	494
Zircaloy-4	115	112	133
Stainless steel	18	16	13
Inconel	7	7	5

Table 2.1.2. Pressurized Water Reactor Fuel Assembly Makeup

2.2 HIGH-LEVEL WASTES

High-level waste is produced by the reprocessing of either commercial spent fuel or defense production reactor fuel. Reprocessing is the chemical separation of the uranium and plutonium from the fission products and transuranic elements in the spent fuel. The principal radioactive constituents in high-level wastes from reprocessing are the low volatility fission products, small quantities of unextracted uranium and plutonium, and other transuranium elements such as neptunium, americium and curium. High-level waste is liquid and for any long-term surface storage and for geological disposal it is generally accepted that it needs to be solidified. A number of processes exist for the incorporation of high-level wastes in borosilicate glass or other solid matrices. The vitrification processes have been demonstrated in France on an industrial pilot plant scale, and vitrification has been accepted for industrial scale operation in some countries as part of the high-level waste disposal concept (e.g., France, the United Kingdom, Japan, Belgium and the Federal Republic of Germany). The vitrification process produces a solid, low-volume waste form, which is chemically durable and has suitable thermal and physical properties for long-term storage. High-level waste generates a lot of heat and requires substantial shielding to control penetrating radiation.

2.3 TRANSURANIC WASTES

Transuranic wastes come mainly from the reprocessing of spent fuel and from the use of plutonium in the manufacture of nuclear weapons. The United States Department of Energy defines it as "waste contaminated with alpha-emitting radionuclides of atomic number greater than 92 (i.e., uranium, hence the term transuranic) and half-lives greater than 20 years in concentrations greater than 100 nanocuries per gram." Transuranic wastes are less intensely radioactive and generate less heat than fission products, but take a long time to decay and therefore require long-term isolation similar to high-level wastes. Techniques for preparing the wastes for chemical and physical stability are relatively well established. Many countries are already practicing immobilization in concrete, bitumen or plastic resins on a large scale.

2.4 OTHER NUCLEAR WASTES

Other types of nuclear wastes with lower radioactivity levels are also produced during the operation and decommissioning of reactors and interim high-level waste storage facilities, operation of research facilities, hospitals, and industrial laboratories. Disposal of these types of wastes generally does not present the types of long-term problems associated with high-level wastes since the radioactivity decays to safe levels in a relatively short time. Although they are not subject of this report, a summary of these other types of wastes is given in this section for completeness.

2.4.1 Operating Wastes

Operating wastes are low- and intermediate-level wastes produced during the operation of nuclear power plants, interim storage facilities for spent fuel and high-level wastes and research facilities. The operating wastes consist of filters and ion exchange resins obtained during the reactor water clean up operations, replaced reactor components, protective clothing, paper, insulating material, tools, medial treatment and research materials, etc. These types of wastes are treated to reduce their volume, e.g., burning, and solidified in concrete, bitumen, resin or some other material in mold, barrels, drums, etc. The solidified wastes can be disposed of by burial in surface trenches, storage in near-surface underground excavations (e.g., Forsmark, Sweden) or in converted mines (e.g., ASSE or Konrad mines in Germany). These waste drums and mold can often be contact-handled although some require additional shielding or remote handling.

2.4.2 Reactor Components

Core components and rector internals located in or near the reactor core are exposed to radiation and are therefore treated as low- or intermediate-level wastes when they are removed during replacement or repair. Spent fuel cladding or hulls and other components removed form the fuel assemblies before storage, reprocessing or encasing in canisters are also included in this category. The radioactivity of these components is lower than that of the transuranic wastes and final disposal requirements are less involved than those for high-level wastes.

2.4.3 Decommissioning Wastes

The radioactive wastes arising during the dismantling of nuclear reactors are all low- or intermediate-level wastes. They consist mainly of the reactor parts such as the reactor vessel, reactor core, concrete around the reactor vessel and components such as steel beams, pipes, tanks, valves, etc., in addition to the wastes produced during the dismantling process (e.g., water and air cleaning systems). The activity levels vary greatly between parts and some can be reused, some deposited in simple landfills while others have higher activity and require longer isolation. The latter are often included for disposal in the tunnels of the underground repositories during the final repository backfilling and sealing operations.

3.0 THE DEEP GEOLOGIC DISPOSAL CONCEPT

Both alpha and heat-generating radioactive wastes can be stored in surface facilities for several decades but this cannot be considered a permanent solution. Continued maintenance, surveillance and monitoring are associated with surface storage and it is necessary to provide a disposal option where these institutional activities will not be required.

The basic idea of geologic disposal is to isolate the radioactive wastes in a geologic environment that has retained its isolation capability for many millions of years. The disposal of waste packages at great depths on land in sufficiently stable and impermeable rock formations can ensure that they will be isolated and undisturbed until the radioactivity has decayed to safe levels. Emplacement of wastes at depth is done in boreholes, specially constructed mined excavations, or in existing mines.

The rationale for underground disposal of radioactive wastes arises from five main considerations:

1. The disposal system is entirely **passive** and requires no human involvement for its safety. It can be abandoned after closure and provides shielding against radiation and dissipates the heat produced by the wastes without the need for active cooling systems.

2. A major feature is the **safety** of the disposal system. Radioactive wastes present no hazard while they remain in the repository and deep disposal decreases the likelihood of inadvertent intrusion by man. A multibarrier system can be used to ensure isolation of the radionuclides for a long time.

3. A variety of geological settings provide a great degree of **flexibility and convenience** in siting and operation of a repository.

4. Many years of mining and civil engineering and construction experience supports the **practicality** of the geologic system.

5. Even though disposal implies no plans for retrieval, some concepts do allow for **retrieval** of the waste packages.

Although there is a high degree of confidence in the feasibility of underground disposal, there may be a need to demonstrate that a facility can be built, operated and closed safely at acceptable costs, using available mining and engineering experience. Studies such as these are already underway in many countries in old mines (Sweden, West Germany) or specially built underground research facilities (Belgium, Canada). These facilities cover a range of potential host rocks from crystalline rocks to salt formations.

The choice of containment of radioactive waste in deep underground repositories provides natural isolation barriers. In addition, engineered barriers can be used to provide further protection, depending on characteristics of geological location and the desire for redundancy. These multiple barriers thus strengthen the containment of a repository, thereby reducing the potential for migration of radionuclide to the biosphere.

Typical barriers which may be found or could be used are:

- Waste form of high stability
- Encapsulation of wastes in materials of high corrosion resistance and long service life
- Use of buffer materials with low permeability and high sorptive capacities around the waste and possibly for filling of rooms, tunnels and shafts
- Impervious, self-healing rock with no circulating ground water or rock with low permeability and low ground-water flow rates
- Long transport times for the ground water to reach the biosphere
- Retardation of migrating nuclides in rock fractures by ion exchange and other processes
- Sorption of nuclides in overlying strata and soils.

Some of the manmade barriers may well be optional. In principle, however, the waste treatment, encapsulation and emplacement steps can be designed to add considerably to the overall safety of the repository concept. This report describes the steps that are being taken by various countries to meet these objectives.

4.0 MAJOR FACTORS INFLUENCING NUCLEAR WASTE DISPOSAL CONCEPTS

Management of radioactive wastes involves a sequence of processes starting with the generation of the wastes, through their collection, treatment and storage, to eventual final disposal. Considerable flexibility exists in the selection of waste management strategies relating to the choice of technological processes, storage times, and selection of the appropriate disposal options. The decisions on waste management strategies and issues will involve various national regulatory agencies and the decisions will have to take into account overall national waste management policies and specific situations such as existing facilities for treatment or processing of wastes, radiation levels of the effluents, operation of the various facilities and personnel radiation exposure levels. This chapter summarizes some of the major factors which influence the choice of the waste management concept.

4.1 GEOLOGICAL

The geological formations chosen for the disposal of radioactive wastes must provide a high isolation capability for radionuclides and adequate stability. Various specific properties have been defined to provide guidelines for the selection of potentially suitable formations. The most important geological factor is to minimize contact with flowing ground waters. A wide range of rock types and potential host formations have been identified throughout the world and are being studied to determine their suitability. These rock types include salt (domal and bedded), crystalline rocks, clay and shales, basalt flows and tuff. All countries with nuclear wastes disposal programs are investigating at least one of these rock types. The choice of rock type depends mainly on the availability of suitably large occurrences of the formations within which an isolation system could be located. After selection of potentially favorable formations, sites will be selected and investigated in a detailed, site-specific reconnaissance program.

4.2 ENGINEERING ASPECTS

The feasibility of the proposed engineering design for deep underground repositories can be demonstrated through the extensive mining and civil construction experience available today and through specific in situ experiments involving actual radioactive materials or heat sources. Many countries have conducted engineering studies and have proven that the capability exists to construct and operate repositories in various geologic media and at the depths proposed. Further technological development is still needed in some areas such as excavation in deep plastic clay formations and in long-term in situ instrumentation and measurement techniques. These developments are being pursued through extensive in situ investigations in underground research laboratories before actual construction and operation of repositories begins.

4.3 SAFETY ASSESSMENT

Since direct demonstration of the long-term safety of particular geologic disposal options over the many thousands of years for which radioactivity exists is impossible, the demonstration of long-term safety must be indirect. The safety assessments of the repository system at a particular geologic disposal site must rely heavily on predictive modelling techniques of repository behavior with time, the duration of waste isolation and the movement of radionuclides that may released. The credibility of this predictive modelling is based on the scientific understanding of the local geology, hydrogeology, the rock mechanics, the chemical, physical and geological processes that may lead to a breach of the isolation system, and the subsequent migration of radionuclides through the geosphere and biosphere. Computer codes will allow evaluation of the overall risk associated with an isolation system by incorporation of basic site data, barrier models and probabilities. Some of these codes are already operational and others are under development. Efforts need to continue, however, in order to refine the codes, verify their applicability, define their limitations, and incorporate results of other ongoing research.

4.4 COSTS OF DISPOSAL

The substantial cost of underground radioactive waste disposal will provide strong incentive to optimize the disposal system and make fullest use of each disposal site developed. Costs for the disposal options including site investigation and technology development as well as repository construction and operation have been estimated to fall generally between 1-3% of the cost of electricity generation.

4.5 AMOUNT AND TYPES OF WASTES

The character (i.e., heat production and radioactivity level) of the radioactive waste will determine the type of isolation system that will be required and for how long this system has to be effective. Less heat production means a closer spacing in the repository and therefore a smaller repository. Reprocessing and vitrification of spent fuel will reduce the amount of wastes but will require additional facilities to achieve this. The reprocessing option is still under investigation in most countries and, though some prefer this option, the direct disposal of spent fuel is also being studied as a possible alternative.

4.6 INTERIM STORAGE

The length of time the wastes are stored before disposal will affect the repository concept. Longer interim storage means less heat output in the repository but carries the additional risk and cost associated with extended interim storage. Storage times being considered in various disposal options range from a relatively short 10 year period (United States) to over 100 years (France).

4.7 SCHEDULE

The repository concept options will also depend on the rate of production of the nuclear wastes and the required rate of disposal. Storage requirements, waste handling methods, repository construction plans and operational schedules will be affected by this.

4.8 REGULATORY

Existing regulations on radiation protection generally apply in situations where exposure of the public can be influenced by control of the radiation source and operational procedures, as well as by control of the transfer routes such as control of consumption of contaminated food after accidental radiation release. Disposal options for radioactive wastes are, however, essentially passive and, after a short period of institutional control, should not require continued human monitoring to assure their safety in the long term. All countries pursuing a radioactive waste disposal system have regulations and laws controlling the disposal process. Regulations cover such aspects as:

- Policies regarding the reprocessing of spent fuel
- Organizations responsible for the various aspects of waste disposal and the working relationships between them
- Funding of the disposal program
- Scheduling for siting, construction, and operation of disposal facilities
- Licencing aspects and support data for licence applications
- Site characterization activities
- Radioactive release rates to the accessible environment
- General nuclear power generation policies
- Environmental impacts of the repository program
- Repository site selection and evaluation criteria
- Engineered barrier considerations and their function in the overall isolation system
- Retrievability of the wastes
- Interim storage.

4.9 RETRIEVABILITY

The United States' high-level waste disposal program, for example, has a requirement in regulations that the waste canisters must be retrievable for a period of 50 years after emplacement begins. This will influence design and operational aspects of the repository since safe reentry into the disposal area must be considered and disposal cannot be conducted in a way that compromises this requirement.

5.0 RESEARCH PROGRAMS

A vital part of the high-level nuclear waste repository programs in all countries is considered to be the investigation and testing of the host rock formations in situ. These investigation and testing programs generally take place in a small scale facility (e.g., one shaft and limited level development) constructed especially for the purpose of conducting research related to nuclear waste storage or in a former working mine which has been taken over by the research organizations and converted to a research facility. This section briefly describes the major research facilities where nuclear waste disposal investigations were or are being conducted. All countries which are considering a mined geologic repository for spent fuel or high-level waste disposal, plan to construct an underground research laboratory at the proposed repository location in order to conduct in situ site characterization before finally constructing the repository.

5.1 AVERY ISLAND - UNITED STATES

A series of experiments were conducted between 1977 and 1983 at the Avery Island mine (International Salt Company) as part of the United States Department of Energy nuclear waste repository program. The mine is located in a Gulf Coast salt dome in southwestern Louisiana. Three basic tests were conducted as part of the Avery Island experimental program and centered mainly around waste canisters which were electrically heated to simulate waste heat and placed in boreholes in the floor of the test room:

- Full-scale heater tests (implanting electrical heaters in the salt and monitoring the mechanical response of the salt formation)

- Brine-migrations tests (monitor the movement of the brine in the salt as a result of the thermal disturbance caused by the electrical heaters)

- Core-jack tests (place circular hydraulic jacks around a roughly 1 m (3.6 ft) diameter core and monitor the response of the salt core to the application of various stresses via the jacks).

The salt mine is an operating room-and-pillar mine and the experiments were conducted in disused caverns on the 168 m (550 ft) level.

5.2 CLIMAX - UNITED STATES

The Spent Fuel Test - Climax (SFT-C) was a series of nuclear waste storage tests using heaters as well as actual spent fuel rods. The tests were conducted between 1980 and 1985 under the technical direction of the Lawrence Livermore National Laboratory for the United States Department of Energy and are the most extensive that have been conducted in the United States to date.

The SFT-C facility (Figure 5.2.1) was located at a depth of 420 m in the Climax stock granite at the Nevada Test Site near Las Vegas. The facility was constructed between 1978 and 1980. Spent fuel was emplaced in April and May 1980 and retrieved in March and April 1983. Post-test characterization was completed in 1985. The objectives of the test were to demonstrate the safe and reliable packaging, transport, short-term storage, and retrieval of spent nuclear fuel. The technical measurement program was implemented to gather data to aid in qualifying granite as a potential repository host rock, aid in the design of repositories, and predict the behavior of the granite under repository conditions.

To create the SFT-C, an existing nuclear testing facility consisting of a furnished shaft and headframe, which had been constructed during the 1960s, was refurbished and used for construction of the three test rooms (Figure 5.2.2). Figure 5.2.3 shows the dimensions of the shaft and the excavated test rooms. A 0.76 m (30 in) diameter borehole was drilled from the surface to the test rooms to serve as spent fuel canister access shaft. During the tests, 13 spent fuel assemblies were used.

Detailed descriptions of the facility and tests are contained in the final report by Patrick published in 1986.

5.3 PROJECT SALT VAULT - UNITED STATES

Salt formations were thought to be potentially suitable host rocks for a high-level nuclear waste repository as long ago as the 1950s. The then United States Atomic Energy Commission's Committee on Waste Disposal recommended that salt formations be studied further to determine their acceptability as repository locations. As a result, tests were commenced in 1965 in an abandoned salt mine near Lyons, Kansas, to demonstrate the feasibility of disposing radioactive wastes in salt. The project, known as Project Sault Vault, continued for about 2 years during which time-spent nuclear fuel was placed in the mine to determine effects of heat and radiation on the bedded salt formation. Electrical heaters were also placed in the salt and used for control data for the same effects. Spent fuel canisters and electrical heaters were installed in similar arrays and data from these experiments included measurement of salt temperature, radiation doses, radiolytic production of gas, thermal expansion of the salt and stress changes and displacements in the surrounding pillars. The most significant results of this first research and demonstration project were:

- Observations of brine migration in the salt towards the heat source
- Development of constitutive relationship for time-dependent behavior of salt
- Refinement of models to predict the temperature distribution around a heat source
- Radiolytic production of chlorine gas caused by gamma radiation was insignificant.

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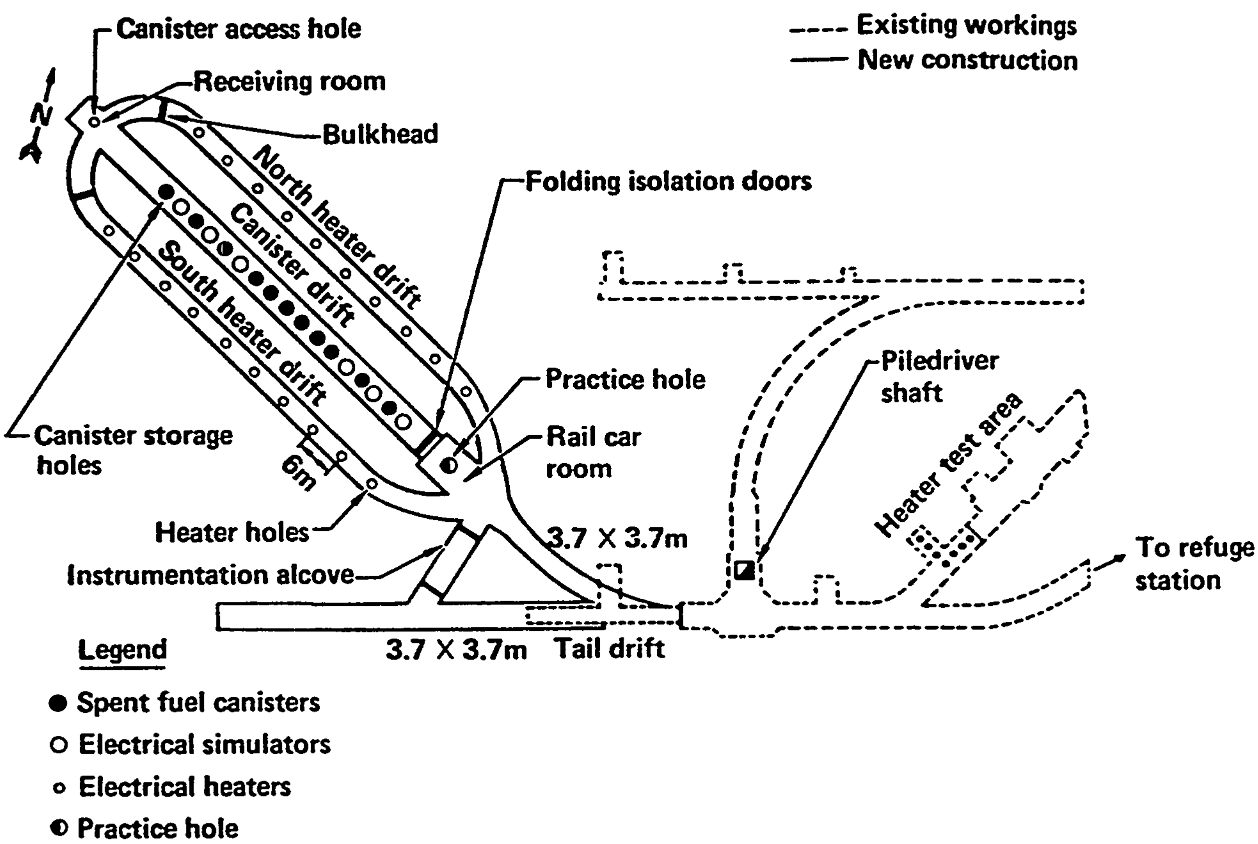

Figure 5.2.1. The SFT-C Facility in granite at Climax, Nevada

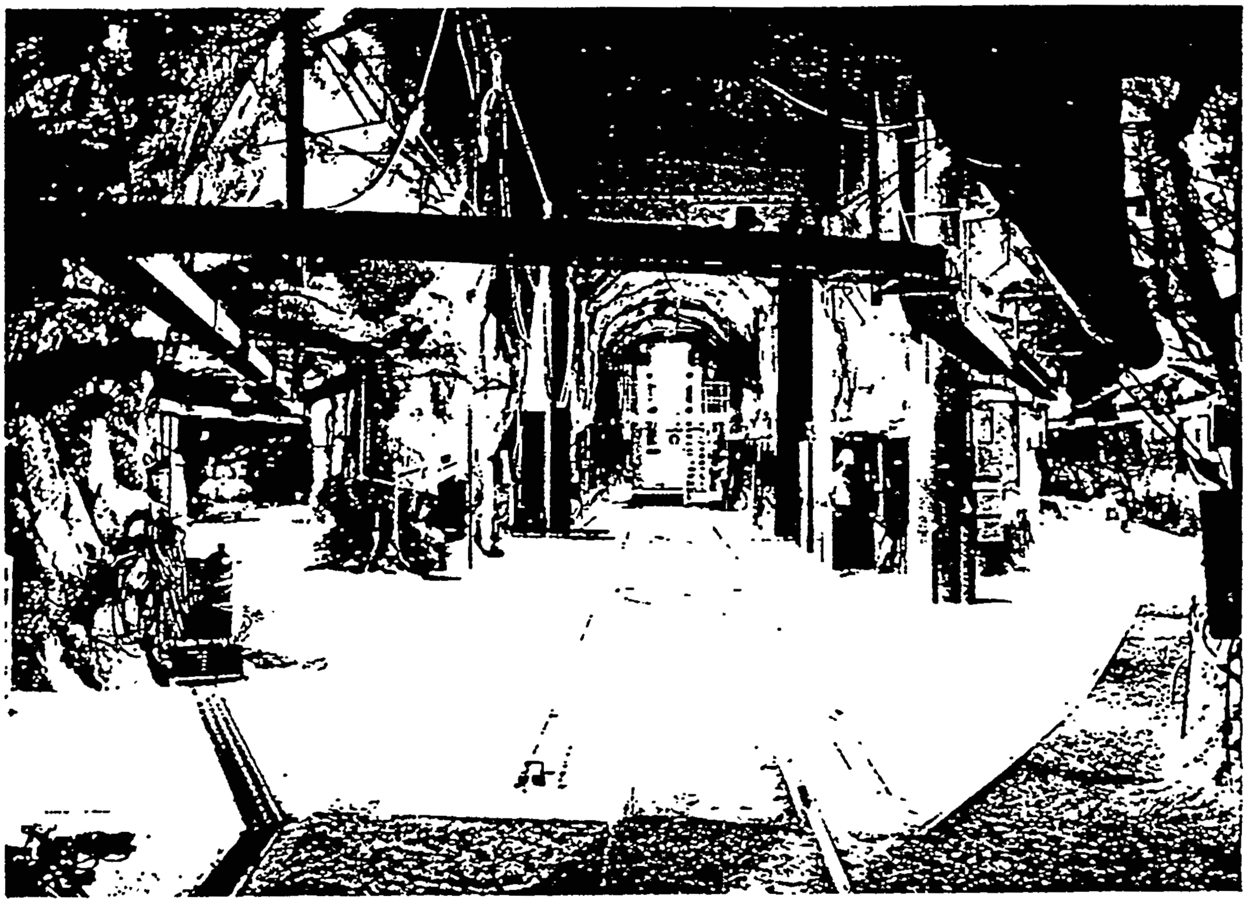

Figure 5.2.2. Test drifts for Spent Fuel Test at Climax, Nevada

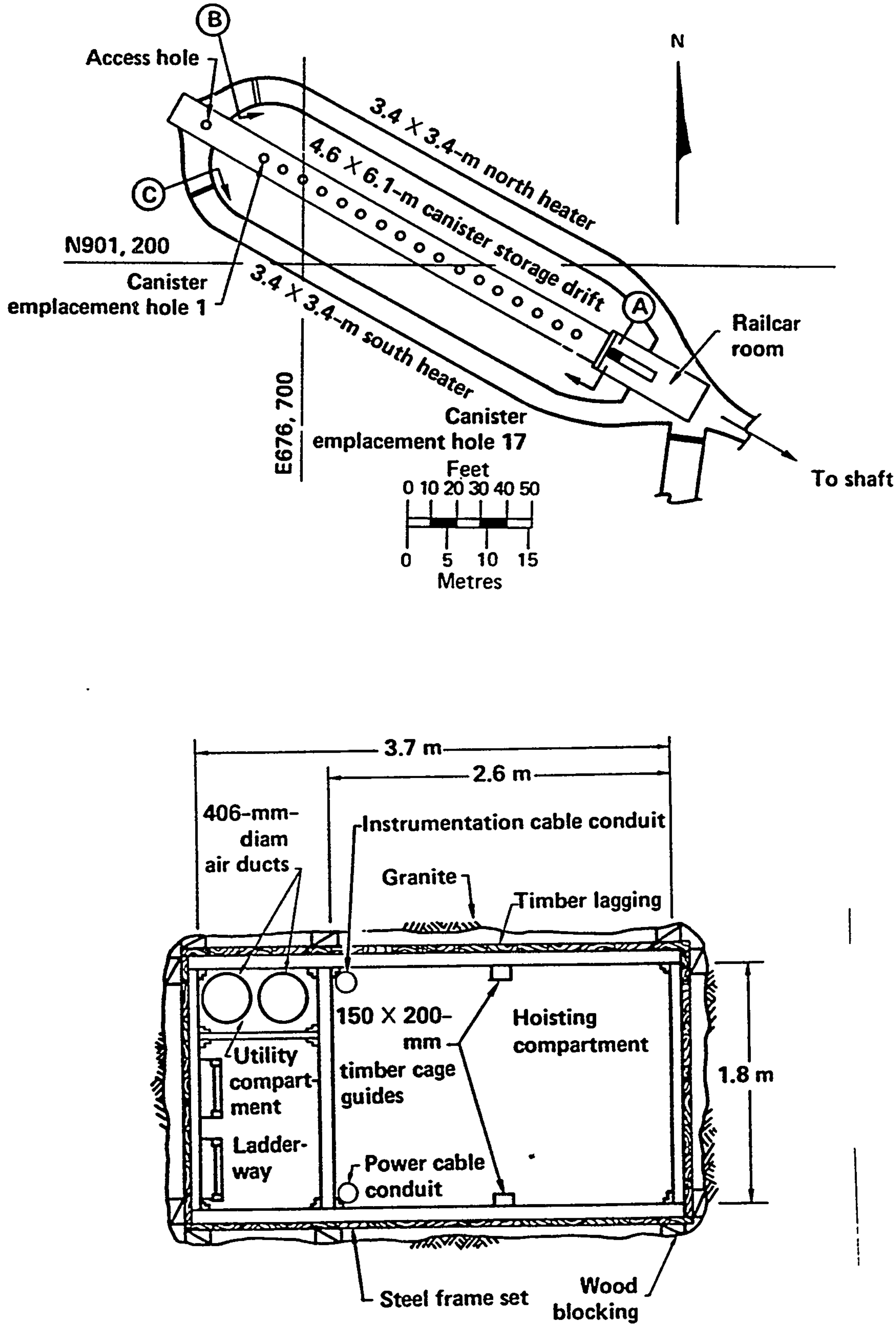

Figure 5.2.3. Configuration of shaft and test drifts at Climax, Nevada

5.4 WASTE ISOLATION PILOT PLANT - UNITED STATES

The Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico, is nearing completion. The facility (Figure 5.4.1) is being constructed in bedded salt at a depth of about 655 m (2,150 ft) and will become the repository for most of the military's retrievable transuranic wastes. Although the WIPP facility will not be associated with the commercial radioactive waste program, the type of in situ instrumentation used at WIPP will be helpful to the research underway for the commercial high-level waste facility. The WIPP facility also represents a significant development in the design, construction, and operations of underground test facilities in salt. A wide range of tests are being conducted to verify the design and predictive computer modelling used in that design in preparation for the start of waste disposal in 1988. 5 years after that, with Department of Energy approval, disposal operations will proceed from pilot to full scale. During the initial 5-year pilot scale operation, the waste can be retrieved if problems occur. The plan is currently to operate the repository until 2015, after which the facility will be backfilled and all access sealed.

5.5 COLORADO SCHOOL OF MINES - UNITED STATES

The Colorado School of Mines in Golden, Colorado, operates an experimental mine located in Idaho Springs, Colorado, where the school has developed many of the test procedures being used in underground facilities. The experimental mine is located in granitic gneiss and is being used for several experiments and testing procedure development programs for the high-level waste repository program:

- In situ evaluation and calibration of rock instrumentation and measurement techniques
- Rock mechanics tests to determine the thermomechanical response and fluid-flow properties of jointed crystalline rocks (crystalline rocks are one of the formations under investigation for the second high-level waste repository)
- Blasting studies to determine the nature and extent of the rock damage induced during excavation
- Large heated-block experiment to collect basic data on the behavior of jointed rock under uniaxial stress and increased thermal load.

5.6 NEAR SURFACE TEST FACILITY - UNITED STATES

The Near Surface Test Facility was constructed at about 100 m (330 ft) below ground in the basalt flows at the Hanford Site in southeastern Washington, one of the sites under investigation as a first repository

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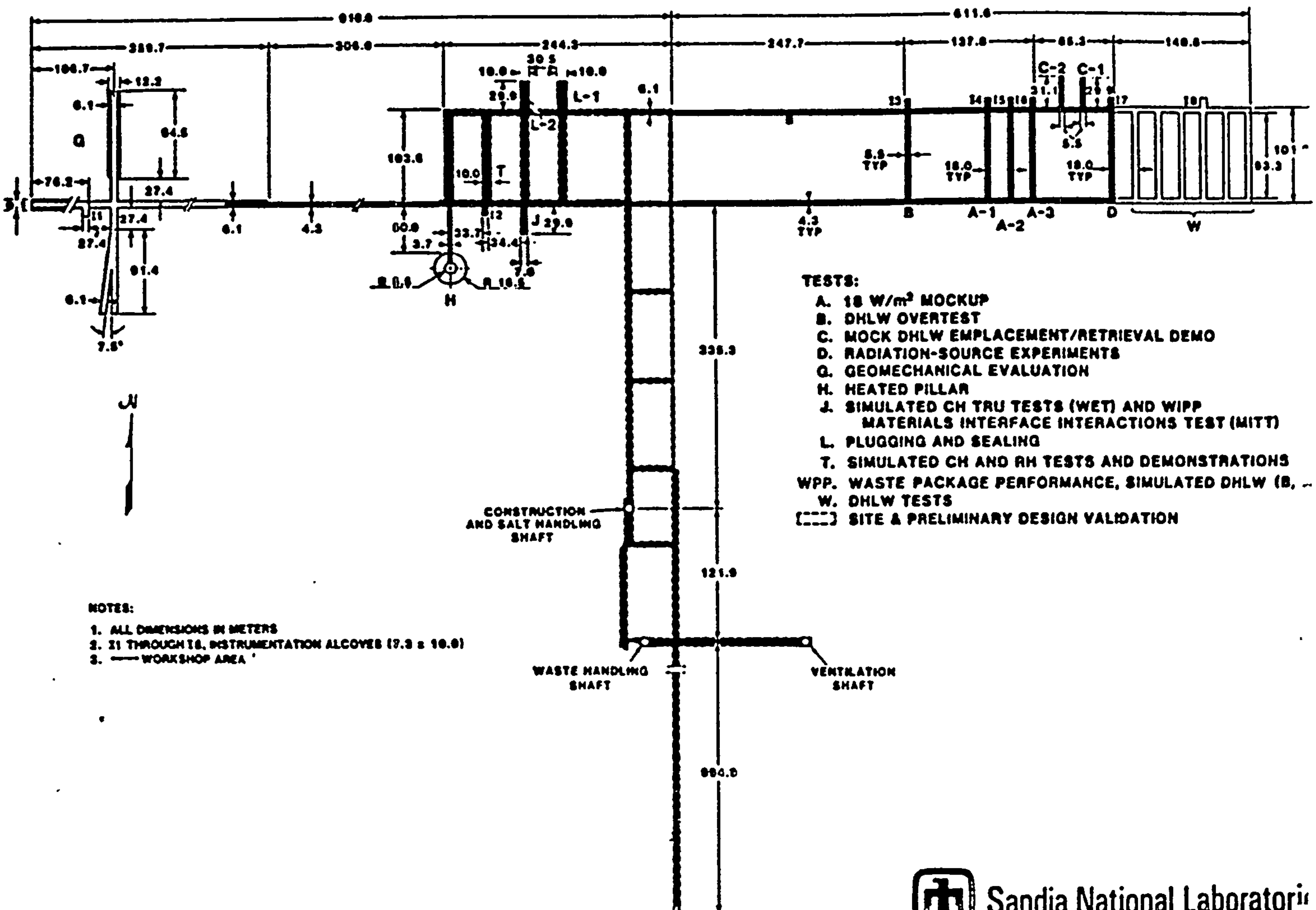

Sandia National Laboratori

Figure 5.4.1. The Waste Isolation Pilot Plant in Carlsbad, New Mexico

location. The purpose of the facility was to establish experimental techniques and monitor the response of the basalt to high temperatures and pressures. Three 200 m (660 ft) long adits were constructed into the side of Gable mountain and connected by 2 100 m (330 ft) long test rooms. Three testing programs were conducted in the facility:

- Full-scale electrical heater tests
- Block tests to determine the pressure- and temperature-dependent deformation of the basalt by cutting slots into the rock to form a separate block which was pressurized at various temperatures with flat jacks
- Overcoring tests to evaluate in situ stresses and overcoring techniques.

5.7 G-TUNNEL - UNITED STATES

An experimental program in tuff was conducted in the previously constructed G-Tunnel complex on the Nevada Test Site near Las Vegas, Nevada. In situ physical and mechanical properties of tuffaceous rocks similar to those at Yucca Mountain are currently being measured under simulated repository conditions in the tunnel, which is an underground test facility. G-Tunnel is being used for preliminary investigations because it is in a layer of welded tuff whose thermal and mechanical properties are similar to some of the welded tuffs at Yucca Mountain. The completed and ongoing tests include small-diameter heater tests and a heated-block experiment. The purpose of these experiments is to measure the thermal and mechanical behavior of welded tuff in situ. Predictions can then be made of the rock's response to heat that radioactive waste would introduce into a repository. The heated-block experiment used an in situ block of welded tuff 2 m (6 ft^2) square bounded by vertical slots. Both stress and thermal loads were imposed on the block to achieve combinations of stress and temperature for evaluating deformation, thermal conductivity, thermal expansion, and fracture permeability. Moisture changes within the block were examined with piezometers, ultrasonic instruments, and a neutron probe. These tests provide valuable experience for developing instrumentation and field techniques that can be used for in situ testing during site characterization.

5.8 UNDERGROUND RESEARCH LABORATORY - CANADA

The Underground Research Laboratory is under construction by the Atomic Energy of Canada Limited, in the undisturbed Lac Du Bonnet Batholith in the southeastern part of Manitoba. The Lac Du Bonnet Batholith is a large granite pluton which outcrops in the Whiteshell Nuclear Research Establishment property.

Experiments being conducted at the Underground Research Laboratory include:

- Surface to subsurface prediction methods and modelling
- Assessment of the effect of the excavations in the rock mass properties
- Study and model the hydrogeological and geochemical environment
- Validate models in a scale and in an environment similar to that of the full-scale repository
- Shaft and borehole sealing methods
- Excavation response experiments
- Heat impact experiments
- Excavation method (blasting) investigations
- Site characterization and mapping methods
- Measuring and monitoring instrumentation development

The first phase of the Underground Research Laboratory was constructed between 1982 and 1986 and includes the initial shaft excavation to 255 m (836 ft) and the completion of the experimental and support excavations on the 240 m (785 ft) level (Figure 5.6.1). The United States Department of Energy has been involved in the experimental program with the Atomic Energy of Canada Limited, in the form of bilateral agreements since 1977 and is currently funding the extension of the shaft to 465 m (1,525 ft) depth. This will allow testing at depths more representative of those expected for the repositories in the United States. The extension is planned to be complete in 1988 and testing is expected to last until about 2000 when the Atomic Energy of Canada Limited lease expires and the facility will have to be backfilled and sealed and returned to its original condition.

5.9 NATURAL ANALOG STUDIES - BRAZIL AND UNITED STATES

Natural analog studies are being conducted in an international agreement effort by Sweden, Great Britain, Brazil, Switzerland and the United States under management by SKB, the Swedish Nuclear Fuel Supply Company. The purpose of these studies is to determine the mass transport of radionuclides in a hydrothermal system within crystalline rock formations, to evaluate the parameters affecting the transport of nuclides under natural conditions, and to evaluate mineral migration characteristics using laboratory techniques. The field portion of these studies involves the analysis of natural uranium and thorium deposits in the Pocos de Caldas district in Minas Gerais, Brazil, and at Marysvale, Utah.

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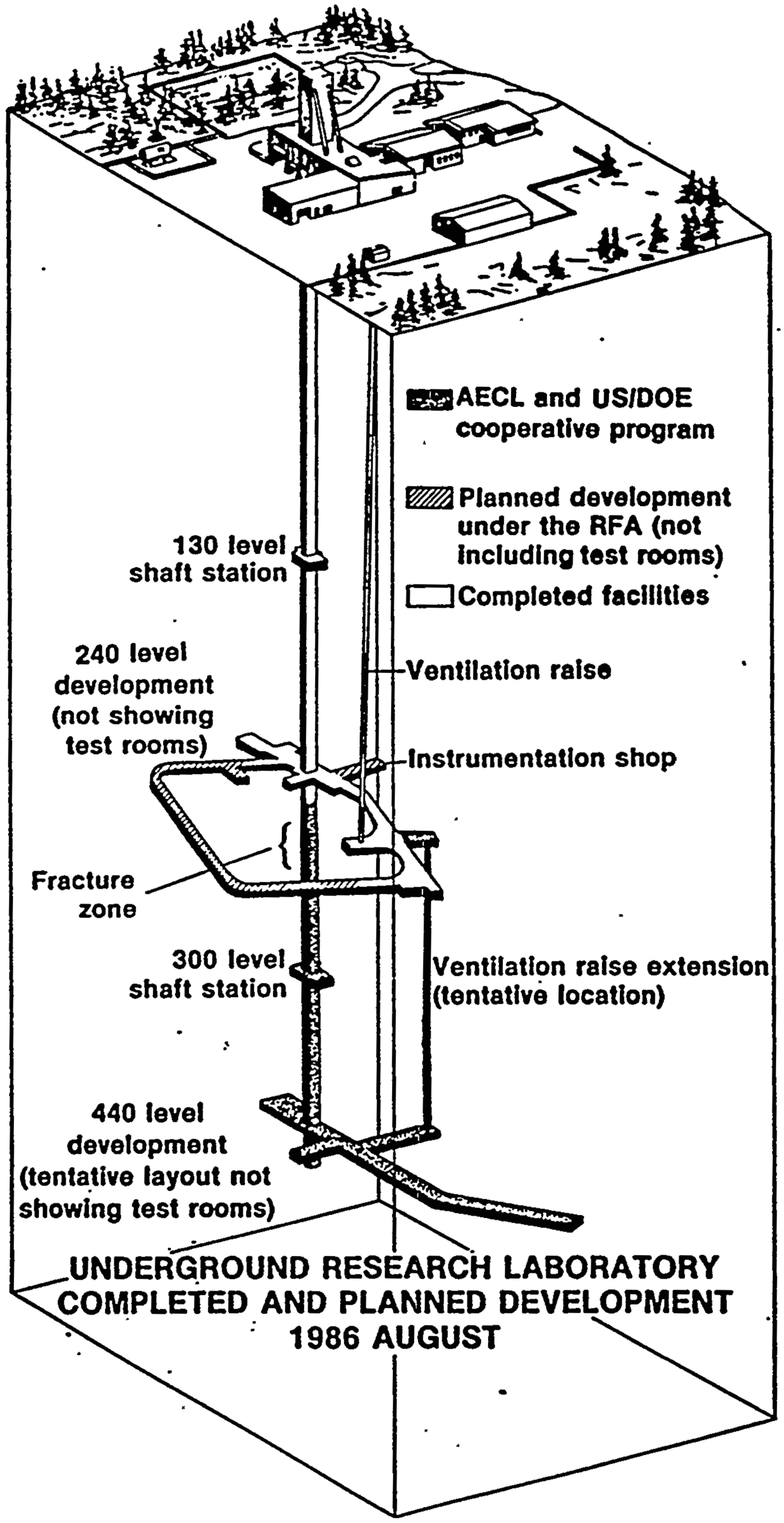


Figure 5.6.1. The Underground Research Laboratory in Manitoba, Canada

5.10 GRIMSEL - SWITZERLAND

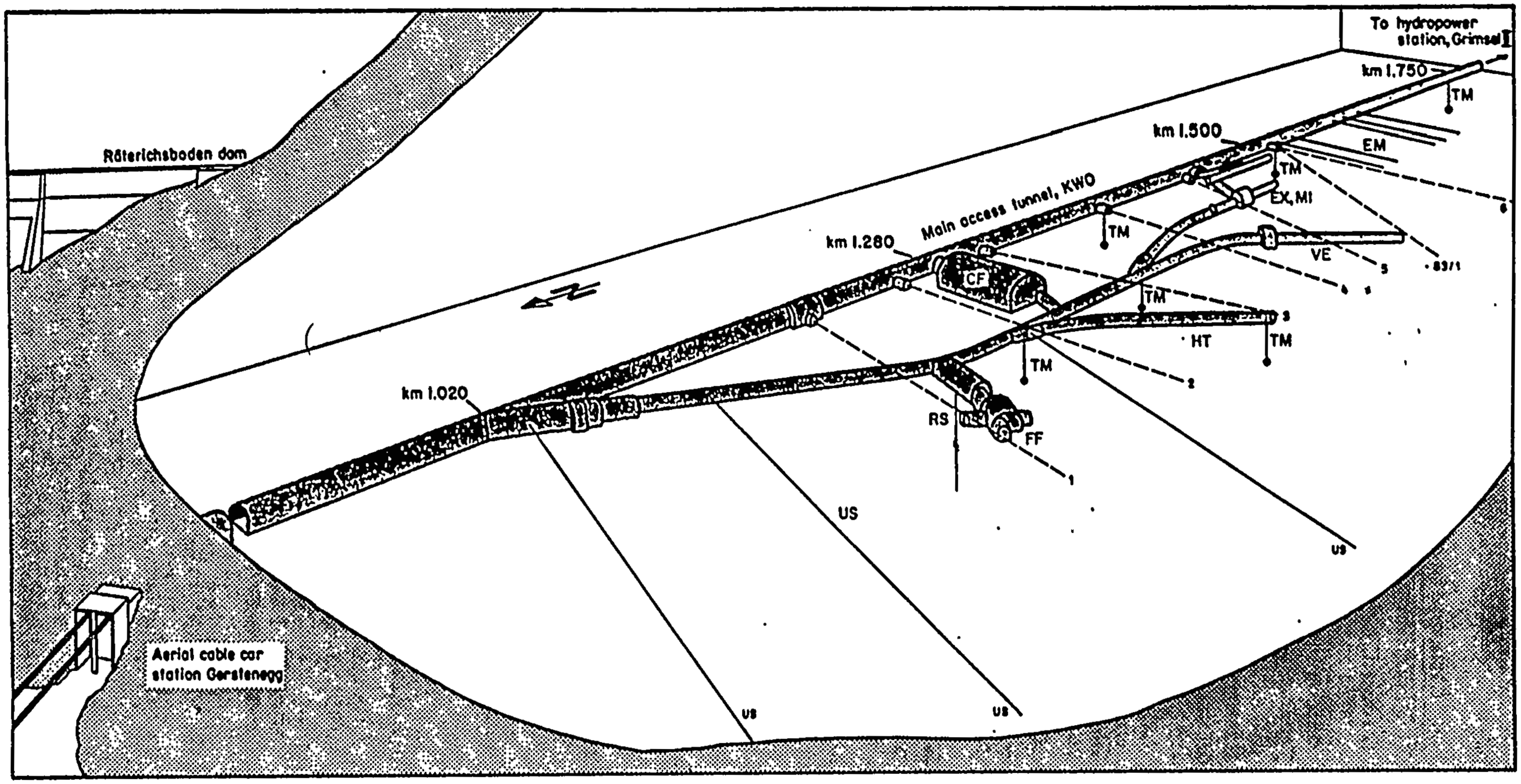
The Grimsel Test Site was established between April 1983 and May 1984 and lies at a depth of 450 m (1,480 ft). The test facility was excavated from the access tunnel to the Grimsel II hydropower station using tunnel boring machine and drill-and-blast methods. The laboratory was built as a branch tunnel from the main access tunnel into the granite Juchlistock. The test tunnels have a total length of about 900 m (2,950 ft) and a diameter of 3.5 m (11.5 ft). Figure 5.10.1 shows the configuration of the Grimsel Test Site and location of the various experiments that have been carried out and are planned. The main aims of the Grimsel test programs can be summarized as follows:

- Evaluation of foreign test results and their application to the specific geologic conditions at potential repository sites in Switzerland
- Conduct specific experiments related to the special conditions of the NAGRA repository concept.
- Gain experience in the planning, operation, and interpretation of various underground testing programs
- Gain practical experience in the development, investigation, and application of the appropriate test procedures and instrumentation.

The Grimsel Test Site is operated by NAGRA, the Swiss National Cooperative for the Storage of Radioactive Wastes, and the experiments are carried out by NAGRA and two German research establishments.

5.11 ASSE II MINE - WEST GERMANY

The Asse II salt mine near Braunschweig was used between 1965 and 1978 as a salt test facility and experimental repository for low- and intermediate-level wastes by the Gesellschaft für Strahlen- und Umweltforschung - Institute für Tieflagerung. Commercial mining in the salt dome between 1916 and 1965 had produced about 130 rooms on 13 different levels at depths between 490 to 750 m (1,600 and 2,500 ft). During the test period about 130,000 low-level waste drums were emplaced in the rooms using various methods including vertical stacking, horizontal stacking, and random placement. About 1,300 intermediate-level waste canisters were stored on the 511 m (1,680 ft) level in a specially prepared cavern during the same period. Since 1978 when the operating permits expired, the Asse facility has been used for non-radioactive testing only to support the nuclear waste programs in Germany and other countries.



LEGEND

CF Central facilities
EM El. magn. HF measurements
EX Tests for excavation effects
FF Fracture system flow test
HT Heat test
MI Migration tests
RS Rock stress tests
TM Tiltmeter
US Underground seismic testing
VE Ventilation test
--- Exploratory boreholes

Figure 5.10.1. The Grimsel Test Site in Switzerland

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5.12 KONRAD IRON ORE MINE - WEST GERMANY

The Konrad iron ore mine near Salzgitter is being evaluated as a repository for non-heat producing low-level nuclear wastes. The mine shut down after 13 years in 1976 due to economic reasons and the existing operational levels are 800 to 1,200 m (2,600 to 4,000 ft) below surface. Two 7 m (23 ft) diameter shafts connect the workings to the surface. The mine is extremely dry and is overlain by thick claystone formations which are impermeable to water. The current geotechnical and rock mechanical investigations aim to prove that the stability of the mine openings and overburden was not affected by the mining operations and that construction of the repository and emplacement and sealing operations will not influence the long-term stability of the repository environment.

5.13 STRIPA IRON ORE MINE - SWEDEN

The Stripa Project is located in an abandoned iron ore mine near Stripa in central Sweden. The underground experiments are being conducted at a depth of about 350-400 m (1,150-1,320 ft) in granitic crystalline bedrock. The project was initiated in 1980 as an autonomous project out under the sponsorship of the Nuclear Energy Agency of the Organization for Economic Cooperation and Development. The Swedish Nuclear Fuel and Waste Management Company acts as manager for the project which includes Canada, Finland, France, Japan, Spain, Sweden, Switzerland, United Kingdom, and the United States as participants.

The experiments are carried out mainly on the 360 m (1,180 ft) level in a massive, grey to light red, medium-grained granite. Research has been conducted in the following four main areas:

- Geohydrological investigations of the granite and migration tests with nuclides in simple and complex fractures
- Chemical testing of the ground water in the granite
- Develop techniques for the detection and characterization of fracture zones in the granite
- Investigation of bentonite clay as a backfill and sealing material.

Phase 3 experiments are scheduled to be conducted from 1986 to 1991 and will include:

- Investigation of a large undisturbed body of granite approximately 125 m x 125 m x 50 m (410 x 410 x 160 ft) in size (Figure 5.13.1) by developing ground-water flow models before comparing them with actual field results
- Continuation of tracer tests started in Phase 2

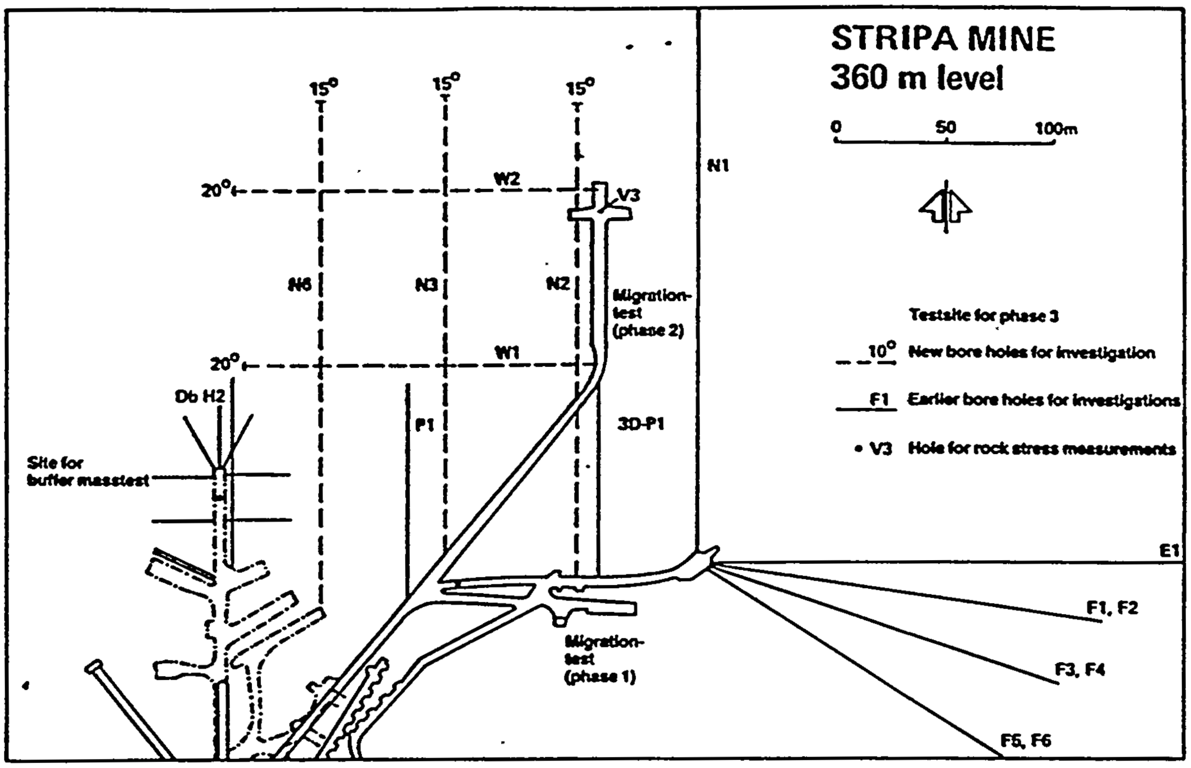

Figure 5.13.1. The Stripa Test Facility in Sweden

- Flow characteristics of ground water in fractures
- Technology development for measuring hydraulic length and width of fractures
- Continue development of rock investigation methods such as borehole radar and high-resolution seismics in boreholes
- Continue investigation of repository backfill and sealing materials
- Grouting tests.

5.14 UNDERGROUND RESEARCH LABORATORY - BELGIUM

The underground research laboratory was constructed between 1978 and 1984 at the Belgian Nuclear Research Establishment in Mol, and was the start of an extensive investigation program into the Boom clay formations at the site. The laboratory is part of the HADES (High Activity Disposal Experimental Site) project and results to date have been encouraging. The decision has been taken to expand the facility to a pilot scale repository in which all repository operations will be performed to demonstrate the feasibility of disposing high-level wastes in clay formations.

The underground research laboratory (Figure 5.14.1) consists of the following:

- A shaft of 225 m (740 ft) depth
- A 35 m (115 ft) long horizontal gallery, lined with galvanized cast iron segments to a finished diameter of 3.5 m (11.5 ft)
- A 25 m (82 ft) deep shaft, diameter of 1.5 m (5 ft), lined with concrete blocks
- A 7 m (23 ft) long experimental drift at 250 m (820 ft) depth.

Site investigations carried out to date have included studies of corrosion behavior of various candidate material, geotechnics, hydrology, physio-chemistry, heat transfer and radionuclide migration. The most important geotechnical result from the construction phase was the possibility of constructing openings in non-frozen clay at depth of 250 m (820 ft). The facility will therefore be expanded between 1986 and 1994 to accommodate additional excavations in the clay, a second shaft and a full suite of experiments and tests including actual waste package handling and emplacement technology, backfilling methods, retrieval of waste packages, monitoring of the response of the clay formation to the repository operations, etc.

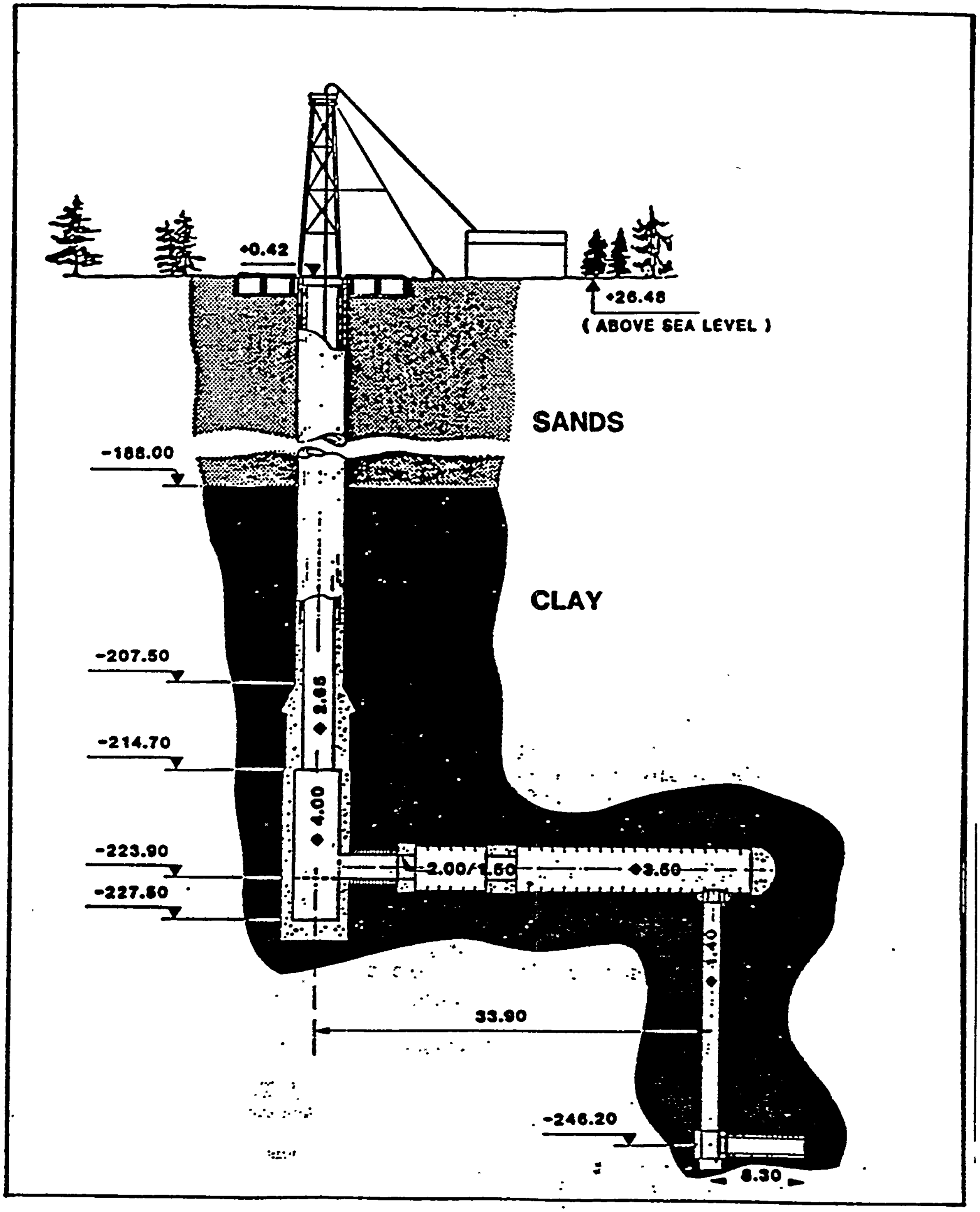

Figure 5.14.1. The Mol Underground Research Laboratory in Belgium

5.15 FANAY-AUGERES URANIUM MINE - FRANCE

A research program, investigating the properties of fractured rock formations, has been initiated within the framework of its program on the safety evaluation of radioactive waste disposal in crystalline formations by the "Departement de Protection Technique de l'Institut de Protection et de Surete Nucleaire" (CEA, Atomic Energy Authority). One of the studies currently in progress is the evaluation of the scale effects on the determination of permeability and dispersion coefficients. The objective is to devise optimal hydraulic tests and improve the interpretation of the results obtained which could be dependent on the rock volume in heterogeneous media.

The experiments are being conducted in an underground laboratory at a depth of about 170 m (560 ft) in a drift of the Fanay-Augeres uranium mine. This mine is located near the city of Limoges in a granitic batholith in the Massif Central. Figures 5.15.1 and 5.15.2 show the location and layout of the test laboratory. After the site investigation phase, which included collection of structural data, core sampling and measurement of ground-water pressure and flow rates, about 10 core holes, drilled from the drift, were equipped with a series of packers to create individual measurement chambers. Via a system of tubes in two of the boreholes, it is possible to purge the system, measure the pressures and flow rates, and inject tracers. Using the measurement chambers created by the packers in all the core holes, it is possible to measure these stabilized pressures and ground-water flow rates.

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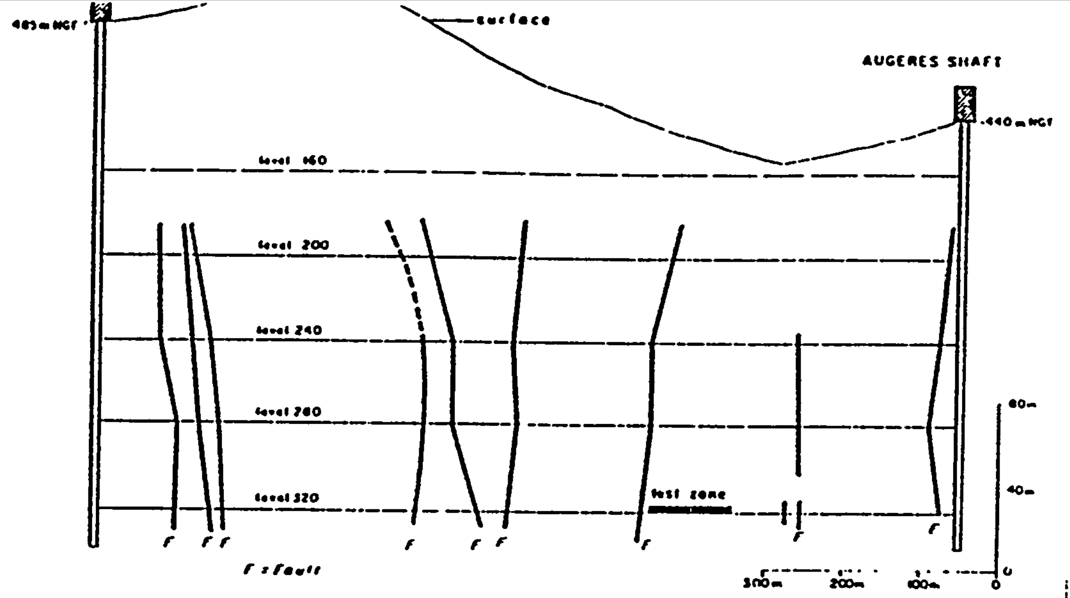
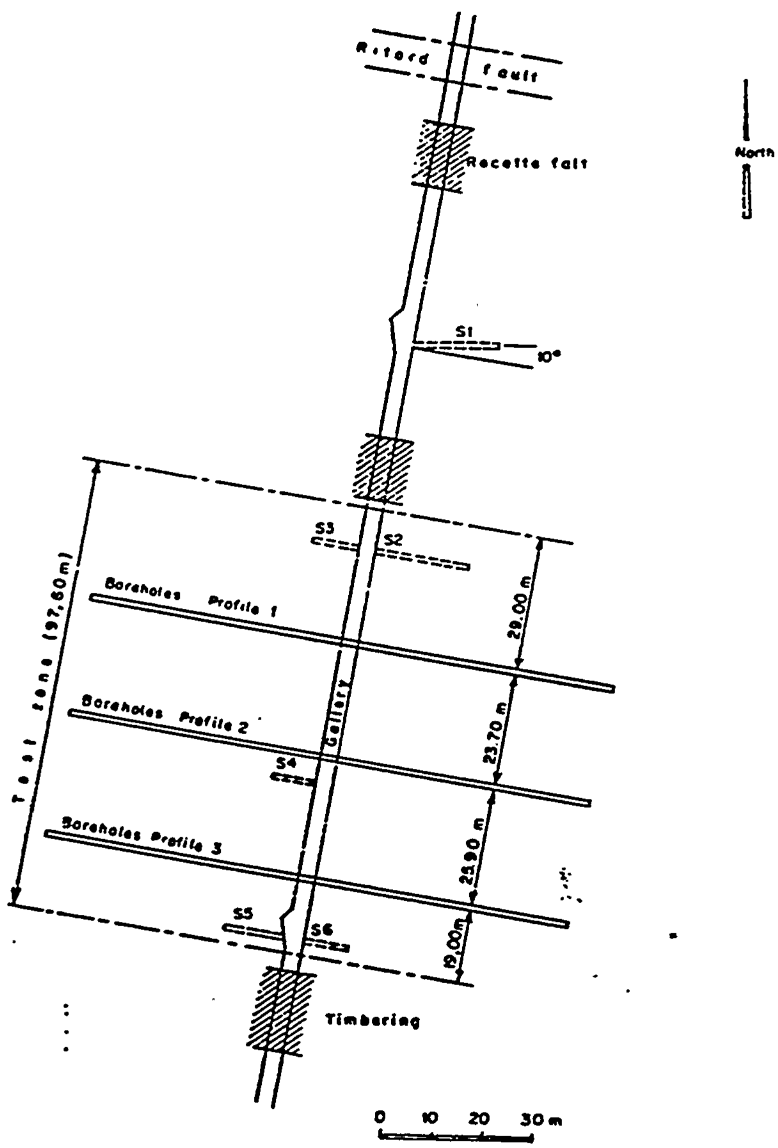


North-South Cross-Section of the Mine



Plan View of the Test Zone Fanay-Augeres Mine, Level 320
(175 m below ground)

Figure 5.15.1. The Fanay-Augeres Mine Test Facility

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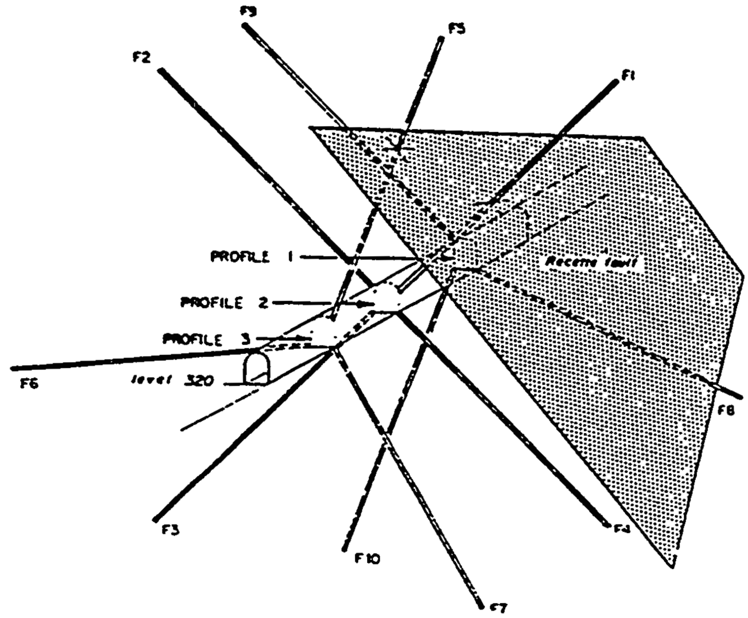

Perspective View of the Experimental Gallery at Fanay-Augeres and of the Three Sections of the Radial Exploration Borehole

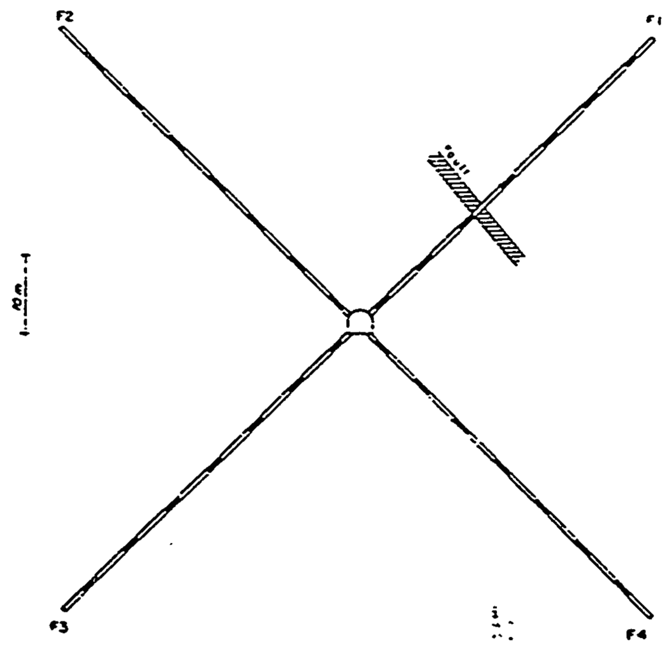

Diagram of the Location of the Packers and the Measurements Chambers

Figure 5.15.2. The Fanay-Augeres Experimental Drift

6.0 CRYSTALLINE ROCK FORMATIONS

6.1 CRYSTALLINE ROCKS: THE SWEDISH SKB DISPOSAL CONCEPT

This chapter describes the current Swedish nuclear waste disposal strategy. The main topic of this chapter is the spent fuel disposal concept because of its relevance to the U.S. program. Brief descriptions of the disposal program for low- and intermediate-level wastes, however, are also included since the handling and preparation of the spent fuel produces wastes of this type. The current Swedish nuclear program is based on a total phase out of nuclear power production by the year 2010. At that time, 12 reactors will have to be decommissioned and the wastes disposed of. Repositories will therefore have to be constructed and operational by that date. The overall strategy adopted to achieve safe nuclear waste disposal in Sweden is basically as follows:

- Interim storage of spent fuel (CLAB)
- Encapsulation of spent fuel in copper canisters
- Disposal of canisters in deep underground repository (SFL-2) in crystalline bedrock
- Disposal of power plant operating and decommissioning low- and intermediate-level wastes in shallow underground repositories (SFR-1 and SFR-3)
- Disposal of operating and decommissioning wastes from CLAB and BS facilities in deep underground repositories (SFL-3, SFL-4 and SFL-5).

The Swedish program does not include spent fuel reprocessing and therefore all "high-level" waste will be in the form of spent fuel. The total spent fuel amount is expected to be about 7,800 tonnes (8,580 tons) of uranium. The strategy is to store the spent fuel from all reactors for an interim period of about 40 years at a monitored interim storage facility. The spent fuel is then to be encapsulated in copper canisters at an encapsulation facility located at the repository and placed in an underground repository. The repository is to be located about 500 m (1,640 ft) underground in crystalline bedrock and will consist of a series of deposition tunnels accessed from surface via several shafts. The canisters will be deposited into bentonite-filled boreholes drilled into the tunnel floors. When the repository is full (around the year 2051), the tunnels and shafts will be sealed to isolate the repository.

The interim spent fuel storage facility (CLAB) was taken into operation in 1985 after completion of Phase I construction and will provide storage initially for about 3,000 tonnes (3,300 tons) of spent fuel. Expansion to full capacity of 8,000 tonnes (8,800 tons) is planned for the mid-1990s.

The site for the spent fuel repository, designated SFL-2, is scheduled to be chosen by about the year 2000. After site investigation, including test construction of an underground test facility, actual repository construction is to begin about 2010 to allow spent fuel disposal to start by the year 2020.

Repositories for the low- and intermediate-level wastes (SFL 3-5) generated during spent fuel storage and encapsulation as well as decommissioning of the storage and encapsulation facility, are scheduled to be constructed concurrently with SFL-2. They are planned to be located near the SFL-2 facility to allow common use of some surface facilities. Similarly to the SFL-2, these repositories are also to be located in crystalline bedrock at a depth of about 500 m (1,640 ft).

Construction of a repository (SFR-1) for the low- and intermediate-level operating wastes from the nuclear power plants and research facilities such Studsvik was started in 1985 and waste emplacement is scheduled to begin in 1988. The repository will be a series of caverns and silos excavated off-shore about 60 m (200 ft) below the sea bed. Access to the repository will be via two declined tunnels originating at the surface facilities located near the coast. Around the year 2000, the facility will be expanded to provide additional disposal caverns (SFR-3) for the decommissioning wastes from the nuclear power plants.

Since all the nuclear reactors, the CLAB storage facility and the SFR-1 repository are located near the sea, special sea terminal facilities were constructed at each site. This allows loading and unloading of the waste transport containers onto a specially designed ship which will transport the containers from the reactors to the CLAB facility or SFR repositories and, later, also from the CLAB to the SFL repositories. The majority of the transport has thus been removed from land. If the SFL repositories will be located inland, a rail system will be required to transport the casks from a coastal terminal to the site.

Figure 6.1.1 shows the locations of the nuclear reactors and the facilities currently operating or under construction and Figure 6.1.2 shows gives an overall timetable for the construction of the nuclear waste disposal facilities. Table 6.1.1 summarizes the radioactive wastes to be disposed of under the program.

6.1.1 Geographic Location

In Sweden, as in many other countries with similar natural conditions, efforts have been concentrated on the study of disposal in deep geologic crystalline rock formations. The Swedish bedrock is geologically very stable and no major changes can be expected in the hydrological or geochemical conditions of the bedrock at a depth of several hundred meters over the next million years or so. A final decision on the site for the final repository is expected towards the end of the 1990s. In order to ensure that a reliable and extensive body of data is available for this decision, a relatively large number of sites will be investigated during the

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Product	Principal origin	Unit	No. of units	Volume in final repository m^3
Spent fuel		canisters	5 600	12 600
Alpha-contaminated waste	Low- and intermediate-level waste from Studsvik	drums	18 000	6 000
Core components	Reactor internals	moulds	2 300	19 000
Low- and intermediate level waste	Operating waste from nuclear power plants and treatment plants	drums and moulds	102 700	95 000
Decommissioning waste	From decommissioning of nuclear power plants and treatment plants	10-20 m^3 containers	5 600	113 000
Total quantity			**134 000**	**246 000**

Table 6.1.1. Radioactive Waste Types

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Swedish reactors

Reactor		Power MW$_e$	Commercial operation	Energy Availability in 1984 %
Oskarshamn 1	BWR	440	1972	82
Oskarshamn 2	BWR	580	1974	93
Oskarshamn 3	BWR	1050	1985	–
Barsebäck 1	BWR	580	1975	88
Barsebäck 2	BWR	580	1977	86
Ringhals 1	BWR	760	1976	80
Ringhals 2	PWR	820	1975	67
Ringhals 3	PWR	915	1980	72
Ringhals 4	PWR	915	1982	78
Forsmark 1	BWR	900	1980	92
Forsmark 2	BWR	900	1981	81
Forsmark 3	BWR	1050	1985	–

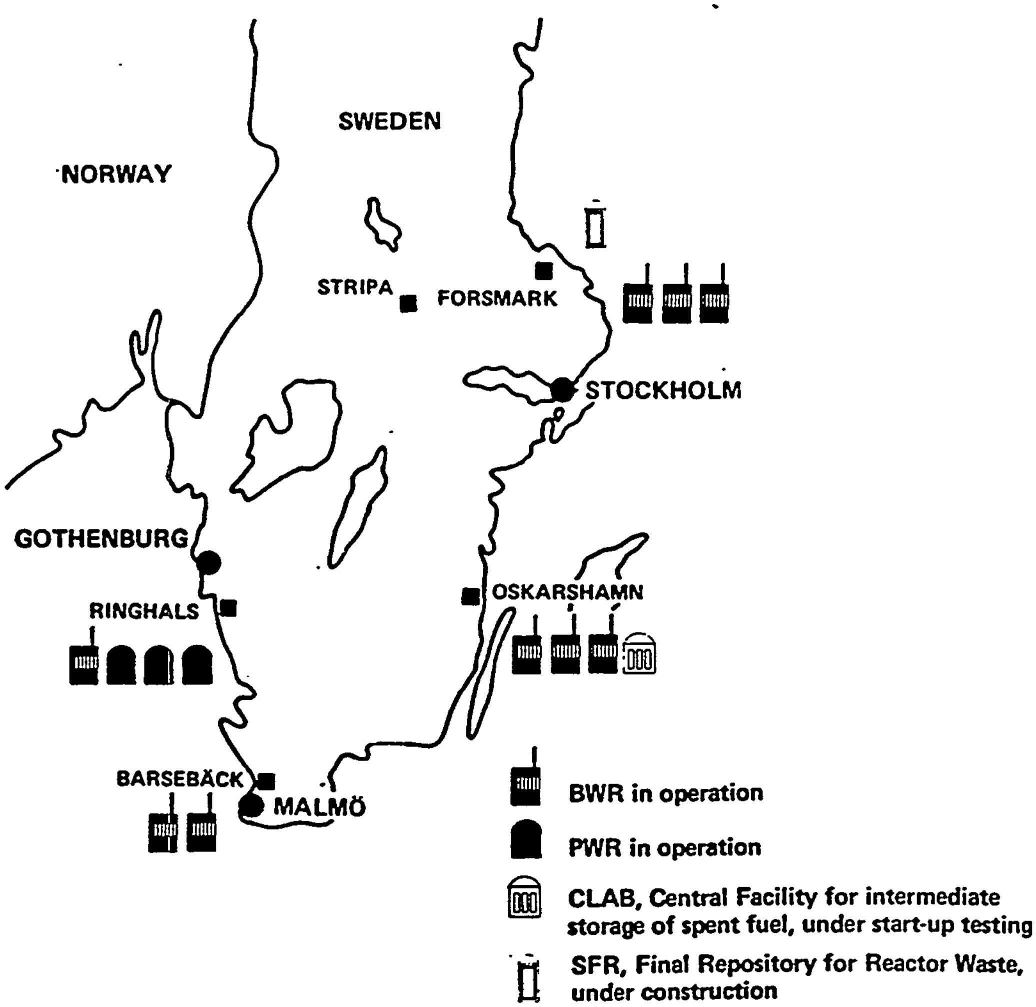

Figure 6.1.1. The Swedish Nuclear Power Program

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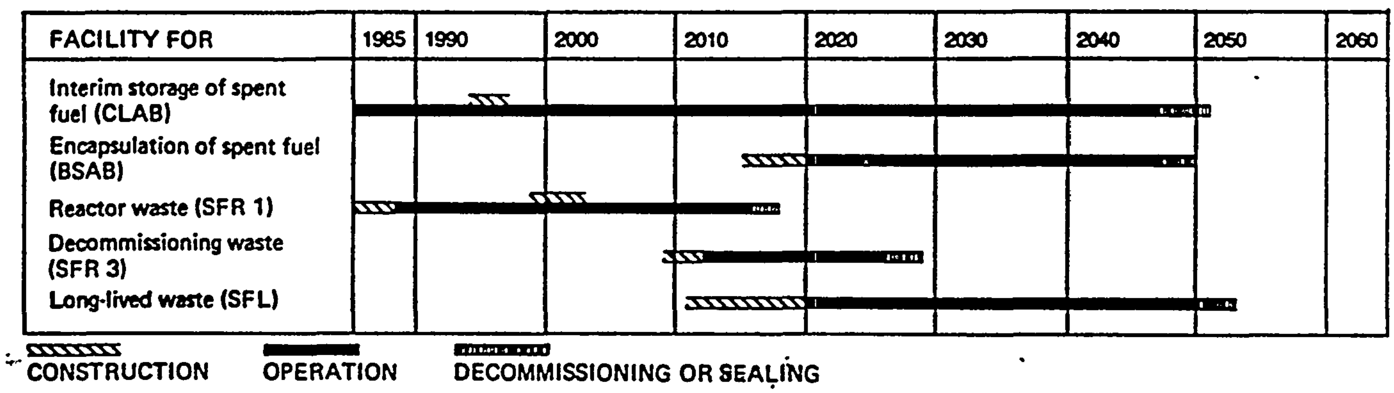

Figure 6.1.2. Timetable for Nuclear Waste Program

1980s and 1990s. Figure 6.1.3 shows the sites that have been investigated to date and Figure 6.1.4 shows the schedule for the site investigation and selection process. Additional areas will be investigated so that a sufficient body of data will be available for an optimum site choice. Factors that also influence the choice of study sites are the previous experience from geologically similar areas, land ownership, accessibility, population structure, potential for mineral exploitation, etc.

6.1.2 Summary of Geologic Setting

The Swedish studies of the final high-level nuclear waste disposal systems focus on disposal below the ground water table in crystalline rock and at such a depth that the repository can be regarded as being protected from even extreme surface forces such as glaciation or/and near-surface activities such as explosions, underground construction, and drilling. Crystalline formations are the predominant formations in Sweden and studies have therefore focused on these because of the large number of possible sites as well as the suitable repository host rock characteristics exhibited by these formations. The crystalline basement rocks are characterized by the fact that they contain "blocks" of sound rock, surrounded by more or less pronounced fracture zones. These fracture zones have arisen during geologically more active periods, the majority more than 650 million years ago. The probability that the general fracture pattern will be dramatically altered over the next million years is therefore very low. However, the possibility of occasional local displacements, found to have occurred in recent geological time, cannot be disregarded. Such movement tends to naturally follow previously fractured and therefore weakened zones, which are avoided when selecting the final repository site.

The Swedish site investigation studies (Figure 6.1.3) have primarily focused on bedrock of granite and gneiss. The studies have concentrated on characterization of:

- Large-scale fractures in the bedrock
- Hydraulic properties of the bedrock
- Chemical composition of the ground water
- Chemical properties of the rock types and fracture minerals.

An example of a potential location for a repository is the site at Gideå, located in the northern part of Sweden. The site was found potentially suitable for further investigation for the following main reasons:

1. The site is characterized by a flat topography.
2. The rock mass is dominated by veined gneiss of low hydraulic conductivity. This rock has been found suitable for the

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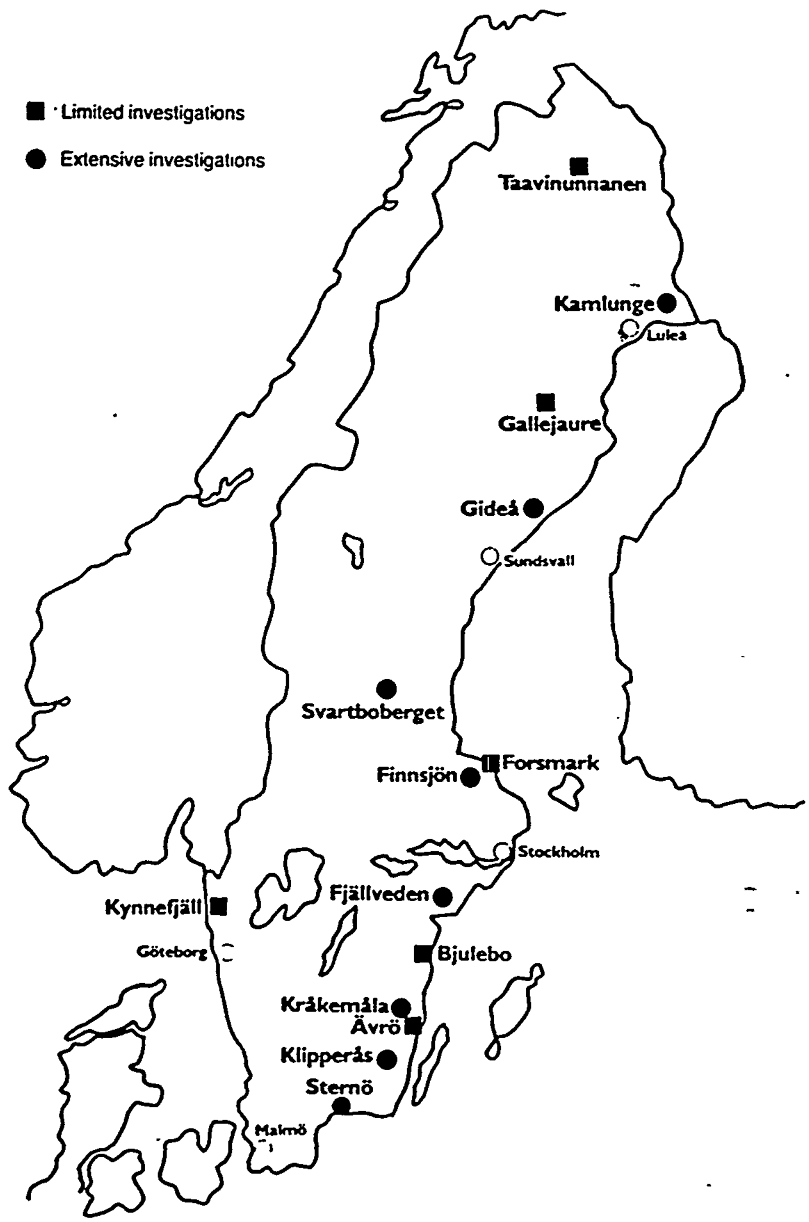

Figure 6.1.3. Investigation Sites

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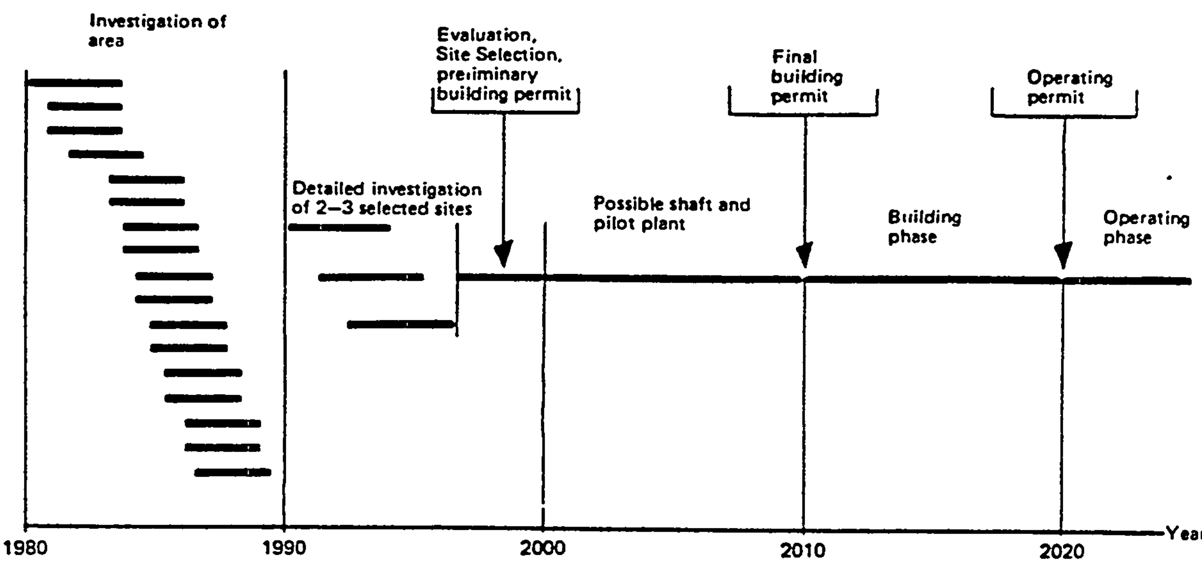

Figure 6.1.4. Schedule for Spent Fuel Disposal

construction of underground openings due to low water inflow and the structural integrity of the openings.

3. The study site displays a low frequency of fractures.

4. The hydraulic conductivity in the local fracture zones is insignificantly higher than that in the rock mass.

5. The estimated area suitable for repository locations is about 2 km^2 (500 acres).

Figure 6.1.5 shows the results of the initial investigations and how a repository could fit into the in situ hydrogeologic setting.

6.1.3 Repository Concept Description

This section describes the current strategy of the Swedish high-level nuclear waste disposal program and the disposal concepts being considered. The descriptions of the repository are still conceptual at this point since site investigation activities, research into waste package and rock properties, fracture behavior and flow modeling studies, and other aspects related to the storage of nuclear waste are still being conducted. The results of these studies and other future influences such as legislative changes, technological advances, etc., will undoubtedly impact the final disposal concept. What is presented in this section is the concept that is considered viable at this time given the present state of knowledge and technology in the disciplines related to the disposal of high-level nuclear waste.

For completeness, brief descriptions are also included on the repositories envisioned for some of the other types of wastes generated during the handling of the spent fuel and decommissioning of the various facilities described in this section. These will be located in the vicinity of the spent fuel repository for more efficient execution and administration of the overall waste disposal program. Section 6.1.8 summarizes these repository concepts.

6.1.3.1 High-Level Waste Disposal Strategy

At this time, the Swedish high-level waste disposal strategy includes only the direct (i.e., no reprocessing) storage of spent fuel in a deep mined geologic repository after a period of interim monitored storage. The Swedish power industry has had reprocessing contracts with BNFL in Great Britain and COGEMA in France. Sweden does not currently plan to use these reprocessing contracts and is working to transfer them to other customers. Reprocessing waste is therefore no longer included in the Swedish plans for a high-level waste repository. Since the high-level waste will not be finally disposed of until well into the 21st century, new methods may lead to changes in the design of the disposal concept as discussed above.

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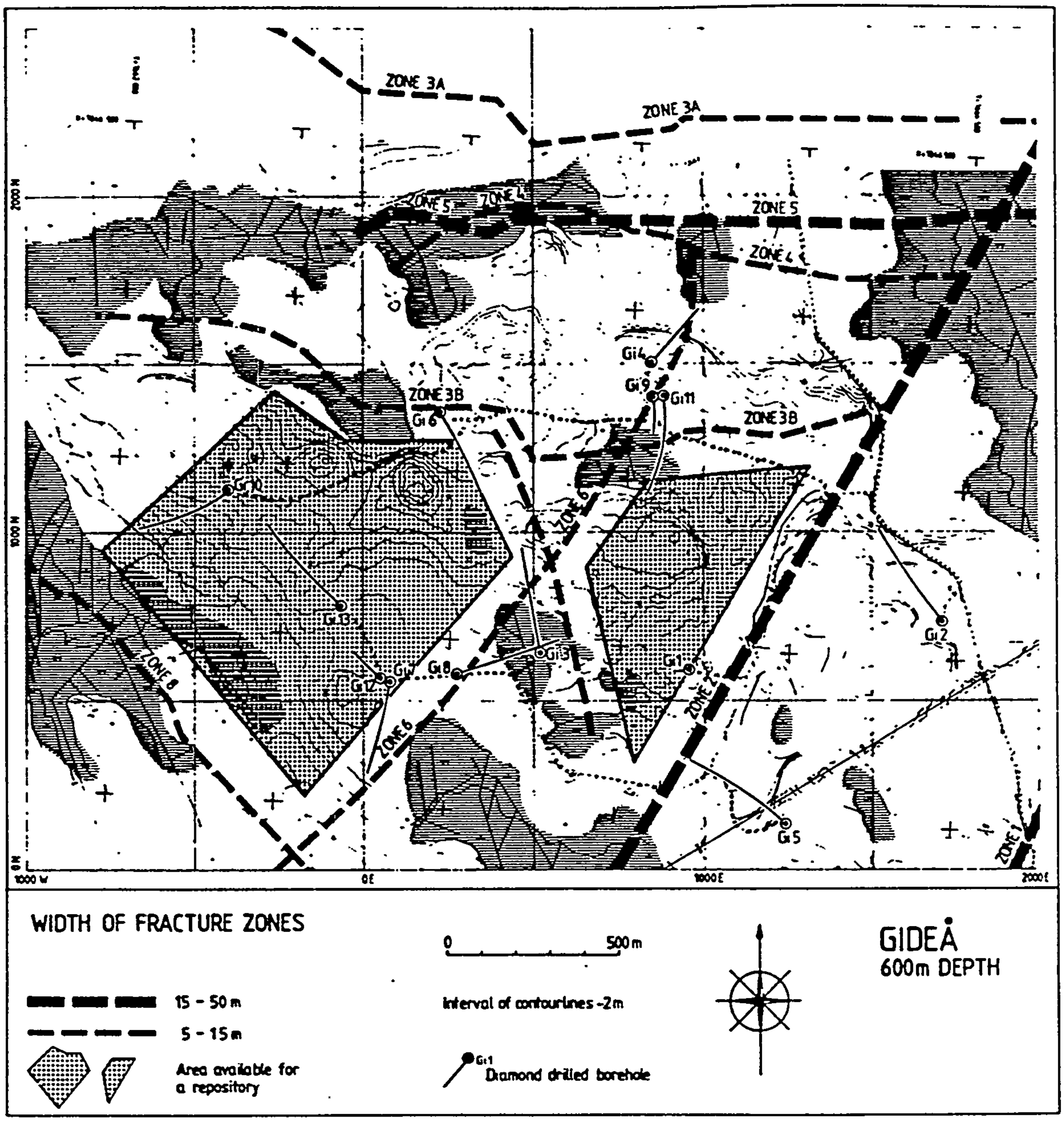

Figure 6.1.5. Study Site at Gideå

A general overview of the Swedish waste disposal strategy for spent nuclear fuel is shown in Figure 6.1.6:

1. Once discharged from the reactor, the spent fuel is stored at the reactor in storage pools for a minimum of six months.

2. The spent fuel is then transported by sea in a specially designed ship to the central interim storage facility for spent fuel (CLAB) shown in Figure 6.1.7. During transport, the spent fuel is enclosed in a special cask, meeting specific international requirements. All nuclear power stations and CLAB have specially designed harbors with suitable terminal equipment and vehicles for handling the spent fuel shipments. Figure 6.1.8 shows the cask and the handling equipment.

3. The spent fuel will be stored at CLAB for about 40 years. During this period, the fuel's radioactivity content and residual heat will reduce by about 90%.

4. After 40 years of storage, the fuel will be transported to an encapsulation station. This transport may take place in the same manner as that from the power station to the CLAB. The shipping cask (Figure 6.1.8) can be used for both sea and land transport.

5. The encapsulation station will be located on the surface at the repository. In the encapsulation station (Figure 6.1.9), the spent fuel is enclosed in copper canisters. The amount of fuel per canister will be limited to about 1.4 tonnes (1.5 tons) to keep the temperature at the canister surface well below 100°C (212°F). The voids between the fuel rods will be filled with lead or copper. Approximately 5,600 copper canisters will be required for the Swedish repository program.

6. From the encapsulation station, the canisters are transferred to the final underground repository. The repository will be designed as a series of parallel tunnels approximately 500 m (1,640 ft) underground. Emplacement holes of about 1.5 m (5 ft) diameter and 7.5 m (25 ft) depth will be drilled into the floors of the tunnels and one canister stored in each hole. The repository may have a single or multiple level arrangement. Tunnel spacing in a single-level repository will be about 25 m (82 ft) and in a two-level arrangement about 33 m (108 ft).

7. The copper canisters in the emplacement holes will be surrounded by blocks of highly compacted bentonite which swells when it absorbs water.

8. After all canisters have been emplaced in the repository, the facility will be sealed by filling all tunnels and shafts with a mixture of sand and bentonite.

9. Some wastes produced during the handling of the spent fuel, such as the fuel boxes and boron glass rods removed during encapsulation

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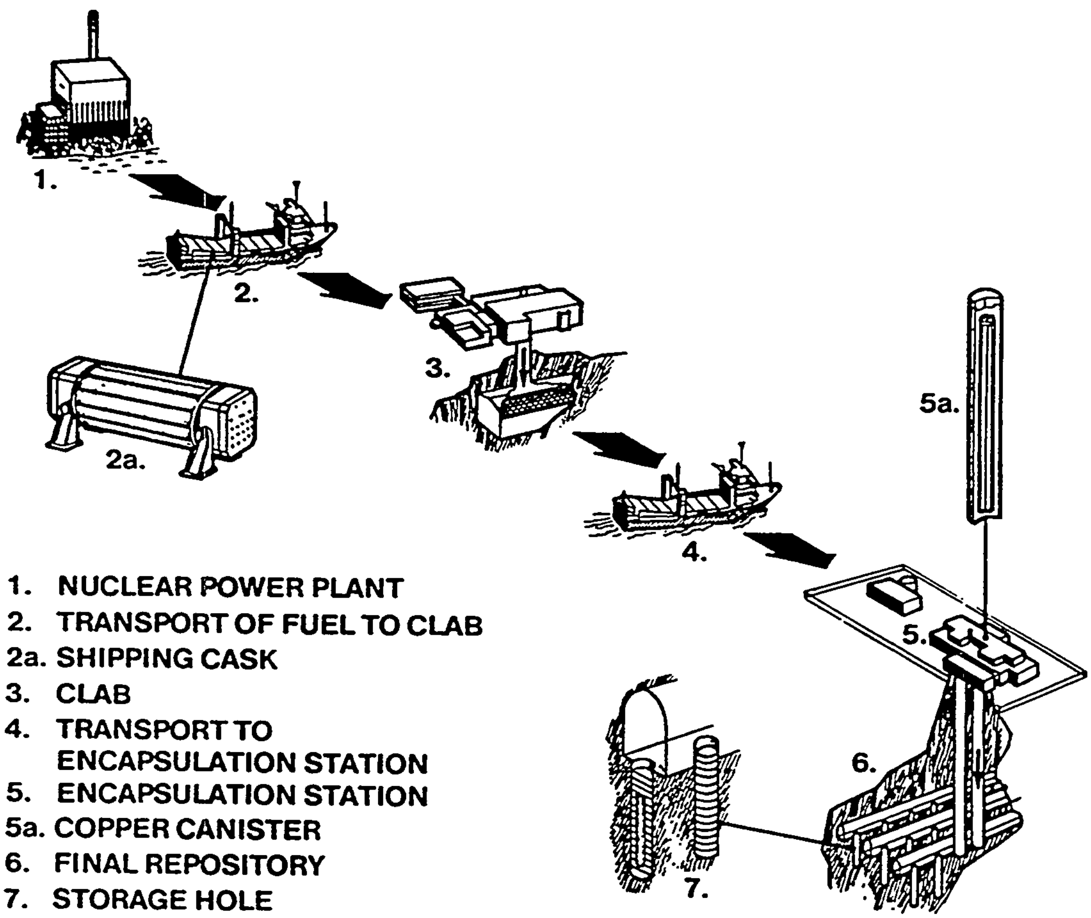

Figure 6.1.6. Spent Fuel Handling Sequence

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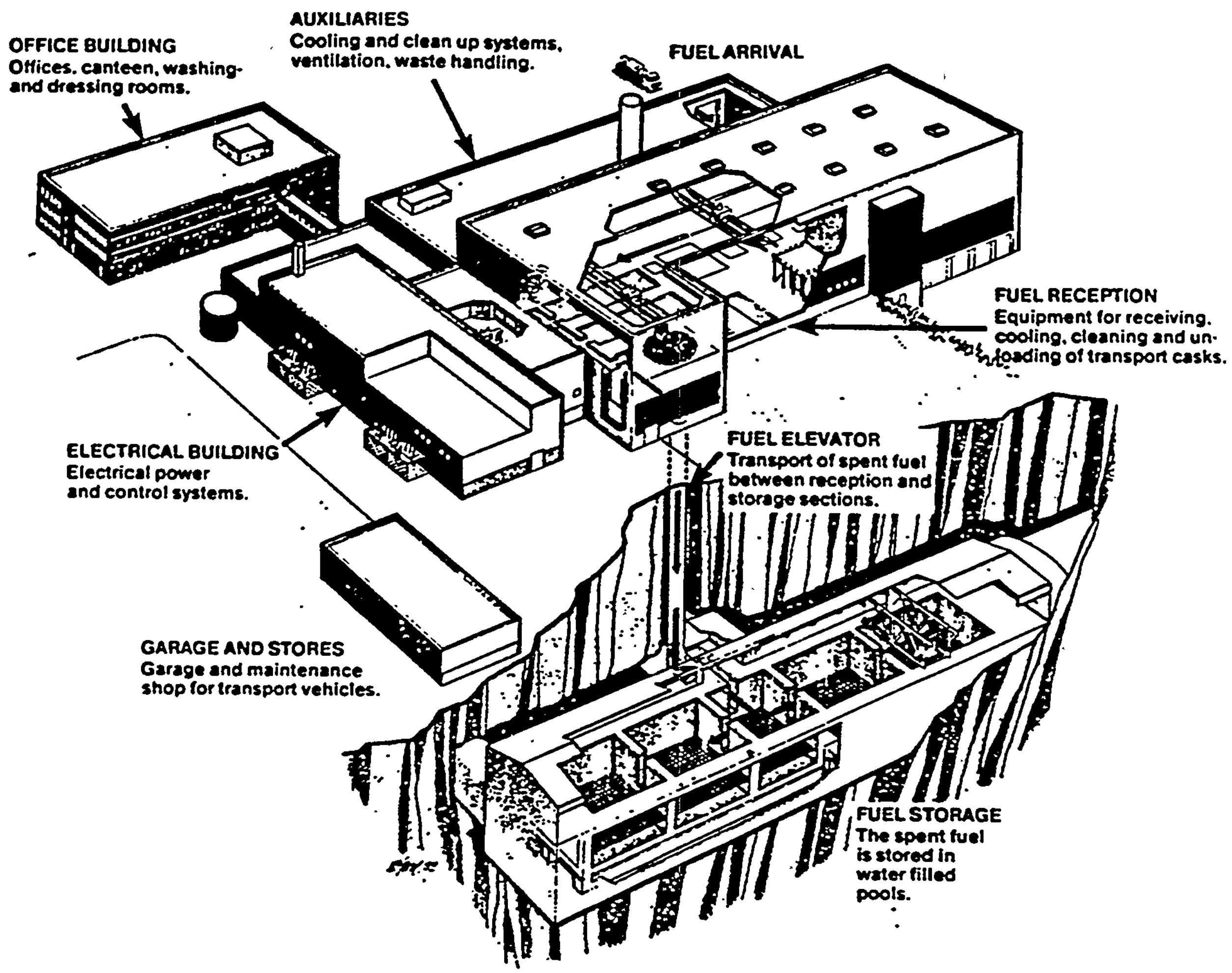

Figure 6.1.7. CLAB General Arrangement

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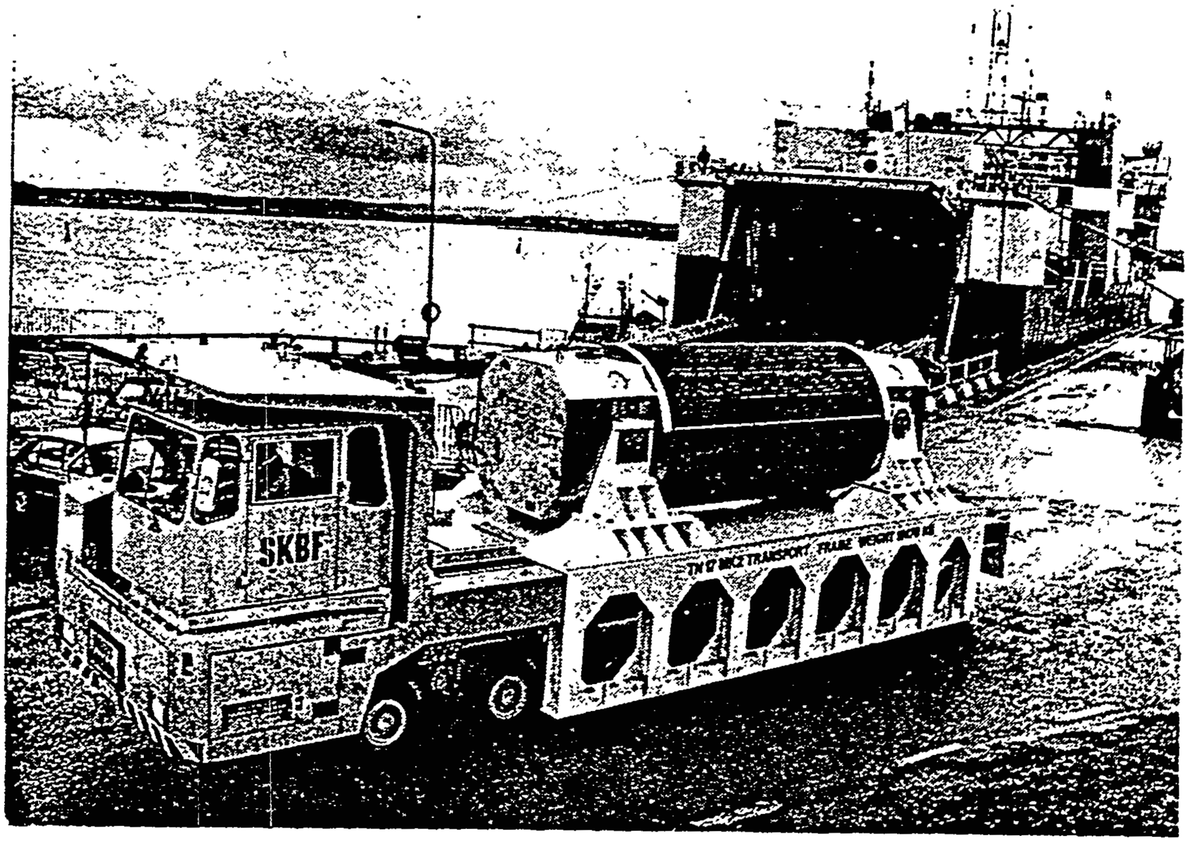

Figure 6.1.8. Spent Fuel Transport Equipment

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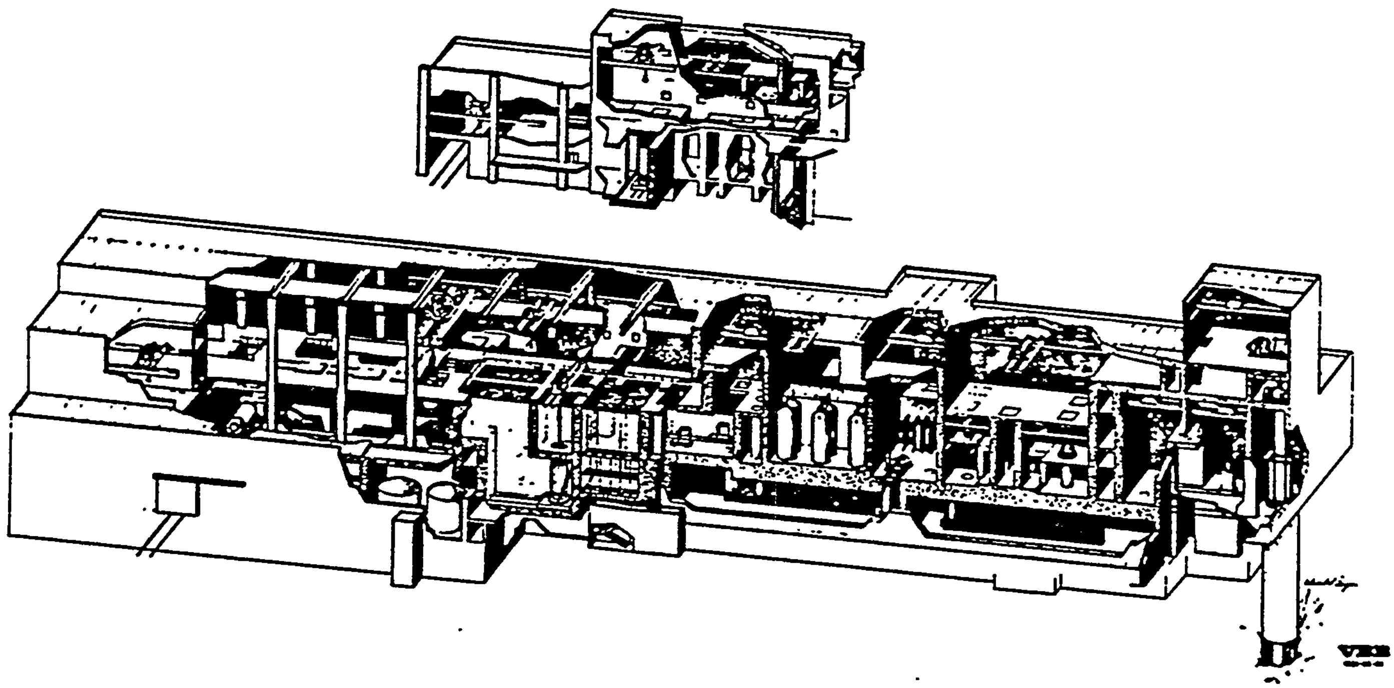

Figure 6.1.9. Encapsulation Station

and decommissioning wastes, will be embedded in concrete molds at the encapsulation facility and taken to a separate, repository for final disposal.

Compared to some other radioactive waste disposal programs such as the Unites States', the Swedish waste disposal program places a high degree of importance on the engineered barriers (waste form, waste canisters, buffer material around the canister, and the repository sealing materials) for the containment of the radionuclides in addition to the host geologic setting. This "multi-barrier" system presents several radionuclide containment systems for added safety and thus some of the uncertainty associated with the radionuclide containment in a geologic setting may be diminished by adding more predictable, engineered containment barriers.

The spent fuel and other long-lived radioactive wastes will be finally disposed of in geologic repositories located in crystalline bedrock approximately 500 m (1,640 ft) below the surface. Four basic types of repositories are planned, each intended for a particular type of waste:

1. SFL-2 - Intended for the encapsulated spent fuel. The repository will consist of tunnels with waste emplacement holes drilled into the floor.

2. SFL-3 - Intended for the transuranic and intermediate-level operating wastes, mainly from CLAB and the encapsulation facility. The repository will consist of concrete troughs placed in a rock cavern.

3. SFL-4 - Intended for the decommissioning wastes, mainly from CLAB and the encapsulation station. The repository will consist of the tunnels and other chambers that are left open after filling of SFL-3 and SFL-5 is completed.

4. SFL-5 - Intended for core components and reactor internal parts embedded in concrete molds. The repository will consist of tunnels into which the molds are stacked and grouted with concrete.

The SFL-2 repository will be a single, separate underground facility. The spent fuel will be encapsulated in copper canisters before emplacement in the repository. The encapsulation will take place at the encapsulation station and it is currently assumed that it will be located at the SFL-2 repository. This means that the fuel can be taken directly from encapsulation via an access shaft to the underground repository SFL-2. Figure 6.1.10 shows a general view of the SFL-2 repository.

The SFL 3-5 repositories will be combined into one underground facility served by several shafts. Figures 6.1.11 and 6.1.12 show the envisioned layout for these repositories. The facility will be located in the vicinity of the SFL-2 repository so that the encapsulation station can be used for preparation of all wastes.

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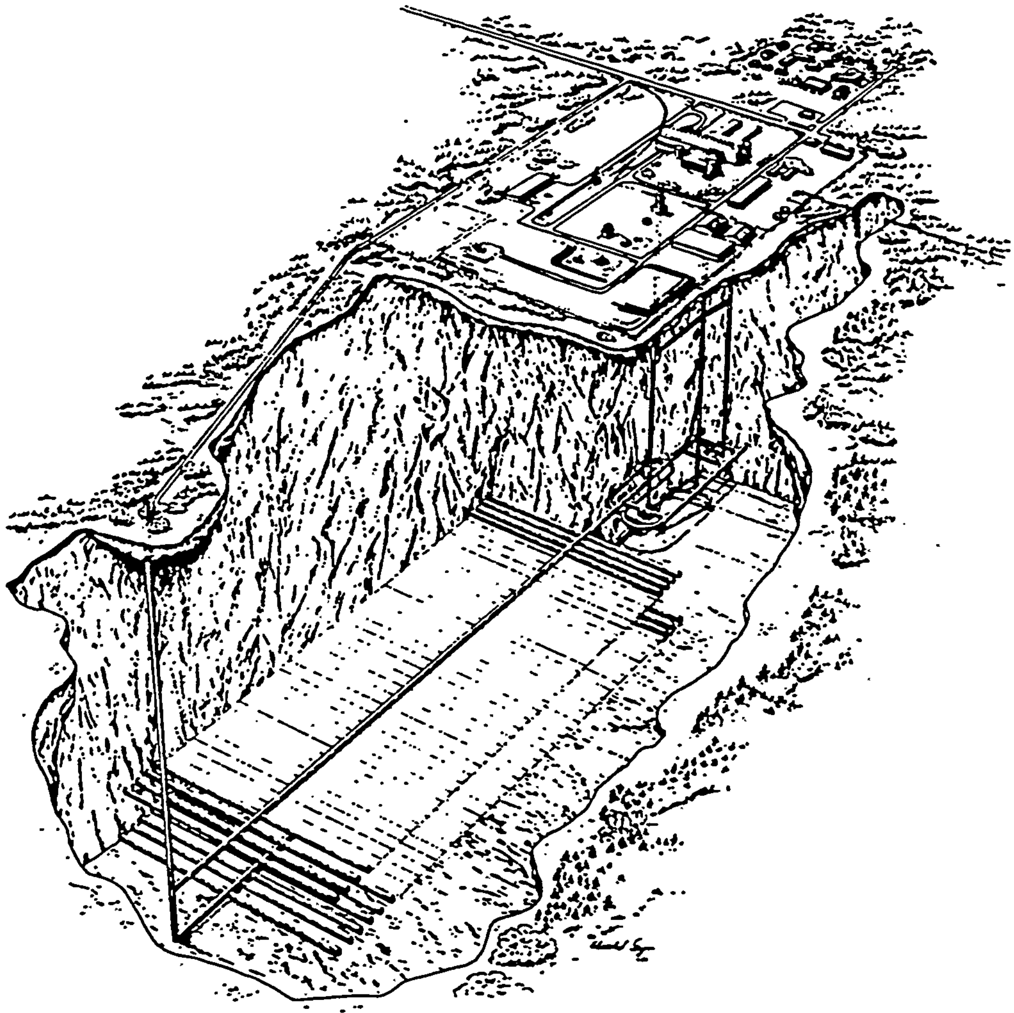

Figure 6.1.1Ø. SFL-2 Repository Overview

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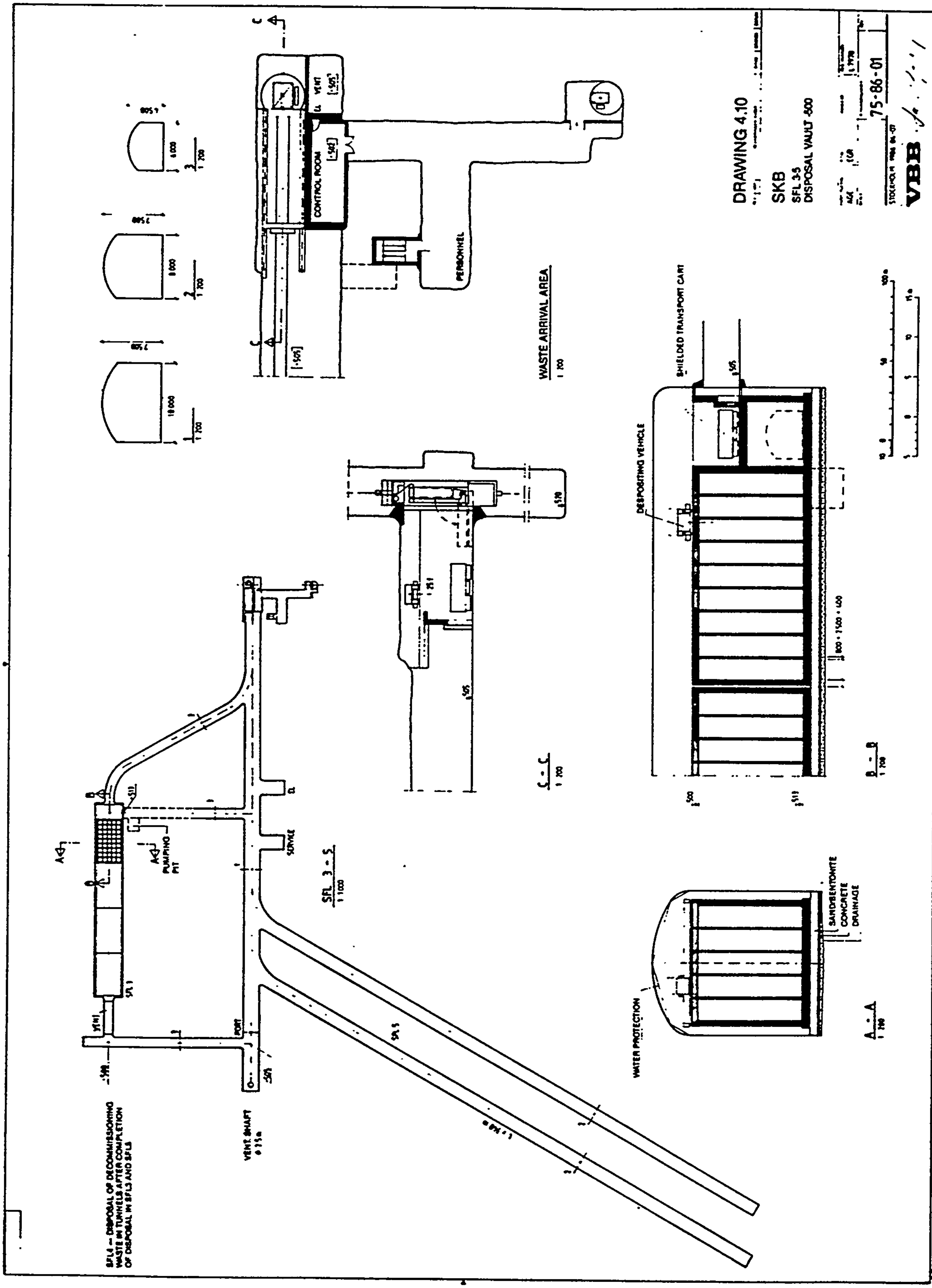

Figure 6.1.11. SFL 3-5 Repository Layout

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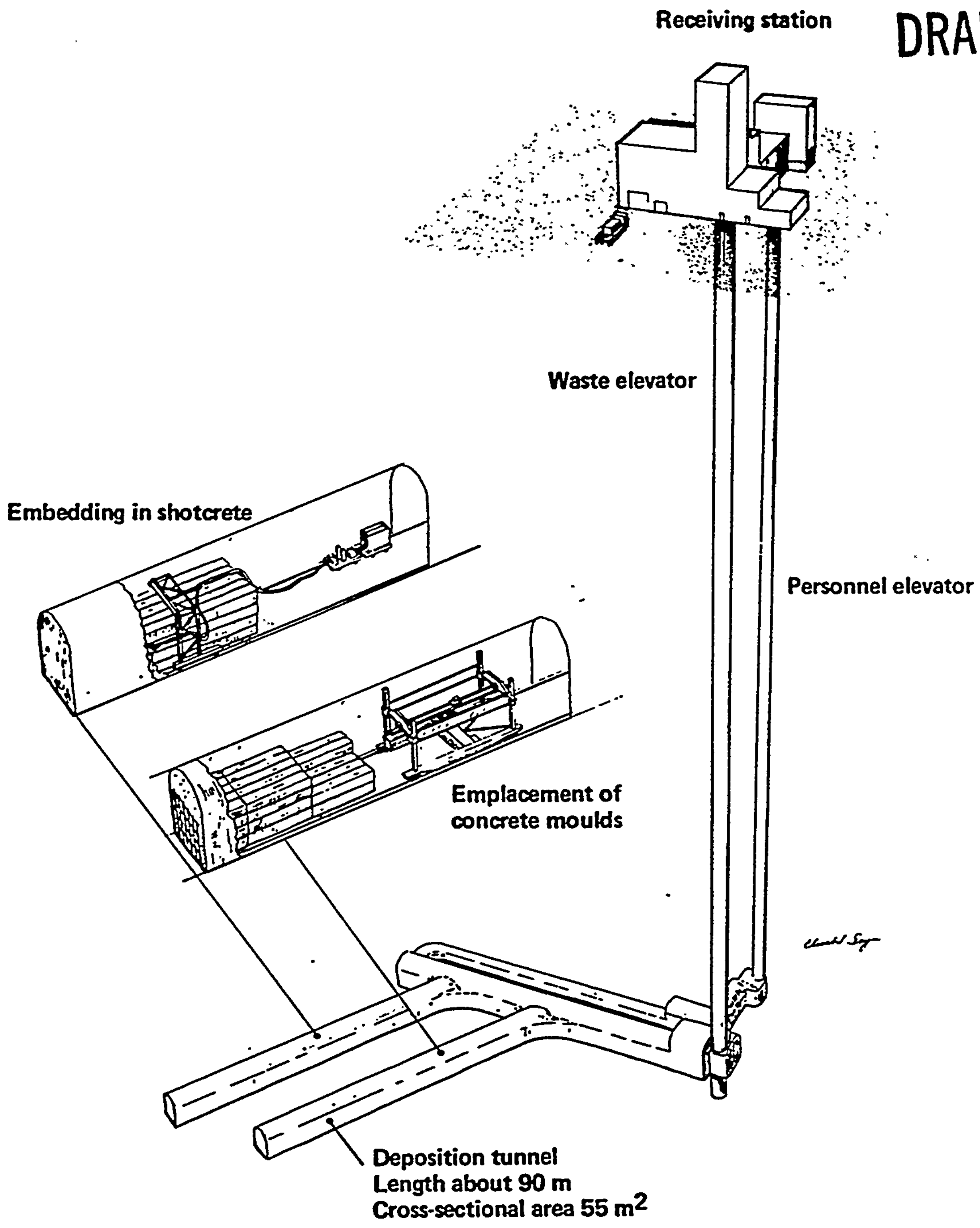

Figure 6.1.12. SFL-5 Repository Layout

By constructing the encapsulation station and the various SFL repositories as one large facility, a large number of services and supply systems can be common to all facilities. This applies to items such as the transportation system, site area security services, facility administration, etc.

It is assumed that the waste coming from CLAB and the Studsvik research facility will arrive by ship to the nearest suitable harbor and from there is transported by rail to the SFL site. Improvement will most likely be required in the harbor area to provide adequate navigational clearances and suitable terminal facilities as well as the construction of railway facilities (where existing facilities are not available) from the harbor to the repository site.

After all wastes have been emplaced in the various repositories, all facilities will be dismantled and the site restored as close to its original condition as possible. The decommissioning wastes, mainly from CLAB and the encapsulation station, will be placed in SFL-4 before sealing the repository shafts, and all operations are estimated to be completed by the year 2055.

6.1.3.2 Surface Facilities

A number of facilities must be planned, built and operated in order to handle and dispose of the spent fuel and high-level radioactive waste products in Sweden. This section describes the surface waste handling facilities in more detail and the next section outlines the underground facilities. Section 6.1.8 briefly describes the facilities for the SFL 3-5 and SFR-1/3 repositories.

6.1.3.2.1 Central Interim Storage Facility. The Central Interim Storage Facility (CLAB) is a spent fuel interim storage facility. Its purpose is to provide an efficient means of monitoring and storing all spent fuel from the Swedish power plants before encapsulation and final disposal. When completed, CLAB will therefore be able to accommodate a spent fuel quantity equivalent to approximately 7,800 tonnes (8,580 tons) of uranium. In addition to spent fuel, other items such as reactor parts and decommissioning products that have been irradiated will be stored at CLAB pending future disposal.

CLAB consists of an above-ground receiving building and underground storage complex in rock. The facility will be constructed in two phases (Figure 6.1.13). Phase I, completed in 1985, comprises the surface handling facilities and an underground rock cavern with storage pools for about 3,000 tonnes (3,300 tons) of uranium. In Phase II, the facility will be expanded to accommodate all 7,800 tonnes (8,580 tons) and is scheduled be completed in the mid-1990s. The facility is planned so that the expansion operation and the fuel storage process can be carried out simultaneously. Storage operations at CLAB started in 1985 and the capacity is now about 3,000 tonnes (3,300 tons) of spent fuel in four pools. Expansion is planned

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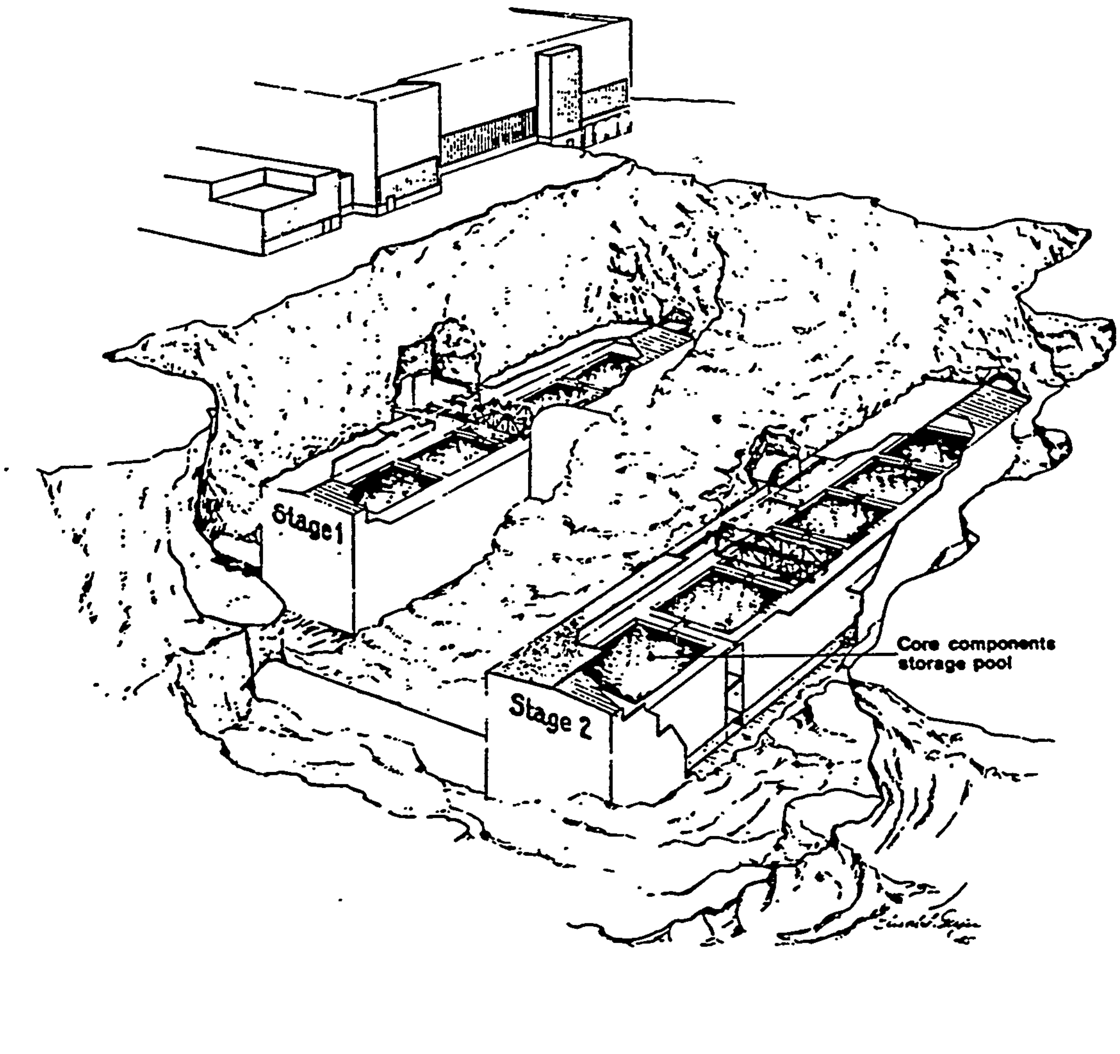

Figure 6.1.13. CLAB Storage Areas

for the mid-1990s so that all approximately 8,000 tonnes (8,800 tons of fuel from the Swedish nuclear program can be accommodated at CLAB).

The surface facilities consist of several buildings which house the various functions for handling the spent fuel. The reception building houses the equipment to unload and load the transport casks for receiving and dispatching the spent fuel and other wastes at CLAB. All handling of spent fuel in the reception building, as in the rest of the facility, takes place in water-filled pools to provide cooling and effective radiation protection. There are seven pools in the reception building. Four of these are used for the two unloading lines and the others for temporary storage and other service requirements, such as receipt of other transport containers.

The water cooling and purification, waste handling, ventilation, etc., are housed in the auxiliary building, connected directly to the reception building. The operations center and all equipment for power supply, control and monitoring of the facility are housed in the electrical building. A separate office and personnel building is connected to all the above by separate passages.

The spent fuel and other waste is stored underground in pools constructed in rock caverns. The rock caverns are about 30 m (100 ft) underground and are supported with rock bolts and concrete. The Phase I cavern (Figure 6.1.14) is about 120 m long, 21 m wide and 27 m high (394 ft x 69 ft x 89 ft). It contains four storage pools, each with 300 storage compartments for the transportable storage modules (canisters) as well as a smaller central pool connected to the elevator shaft via a transport channel. The pools are constructed of reinforced concrete and are lined with stainless steel. Each pool holds 3,000 m^3 (790,000 gal) of water and can accommodate about 800 tonnes (880 tons) of uranium. The Phase II cavern will be similar in design and will contain six pools for spent fuel and one for reactor core components.

A fuel transport cask arriving at CLAB (Figure 6.1 8) is inspected, raised into an upright position and transferred to one of the cooling cells in the reception building. The cask seals are removed and the cask filled with water. A cooling circuit circulates the water to cool the cask and flush out some of the radioactive particles in the cask. The cask then travels on a transport wagon via a connecting channel to a position under the unloading pool. A watertight connection fitting seals the cask to the base of the pool to separate the uncontaminated water in the cask pool from that in the unloading pool. After removing the cover from the cask, a crane transfers the fuel assemblies from the cask to fuel canisters. Several types of canisters are used. A canister for boiling water reactor fuel holds 16 fuel assemblies, a pressurized water reactor canister holds 5. Figure 6.1.15 shows examples of the fuel assemblies. A full fuel canister is transferred by crane to the elevator and taken down to the storage cavern. An overhead crane transfers the canister to one of the storage pools. The empty cask is drained, reassembled, and, after a final inspection, removed from the CLAB facility. Loading of the transport casks

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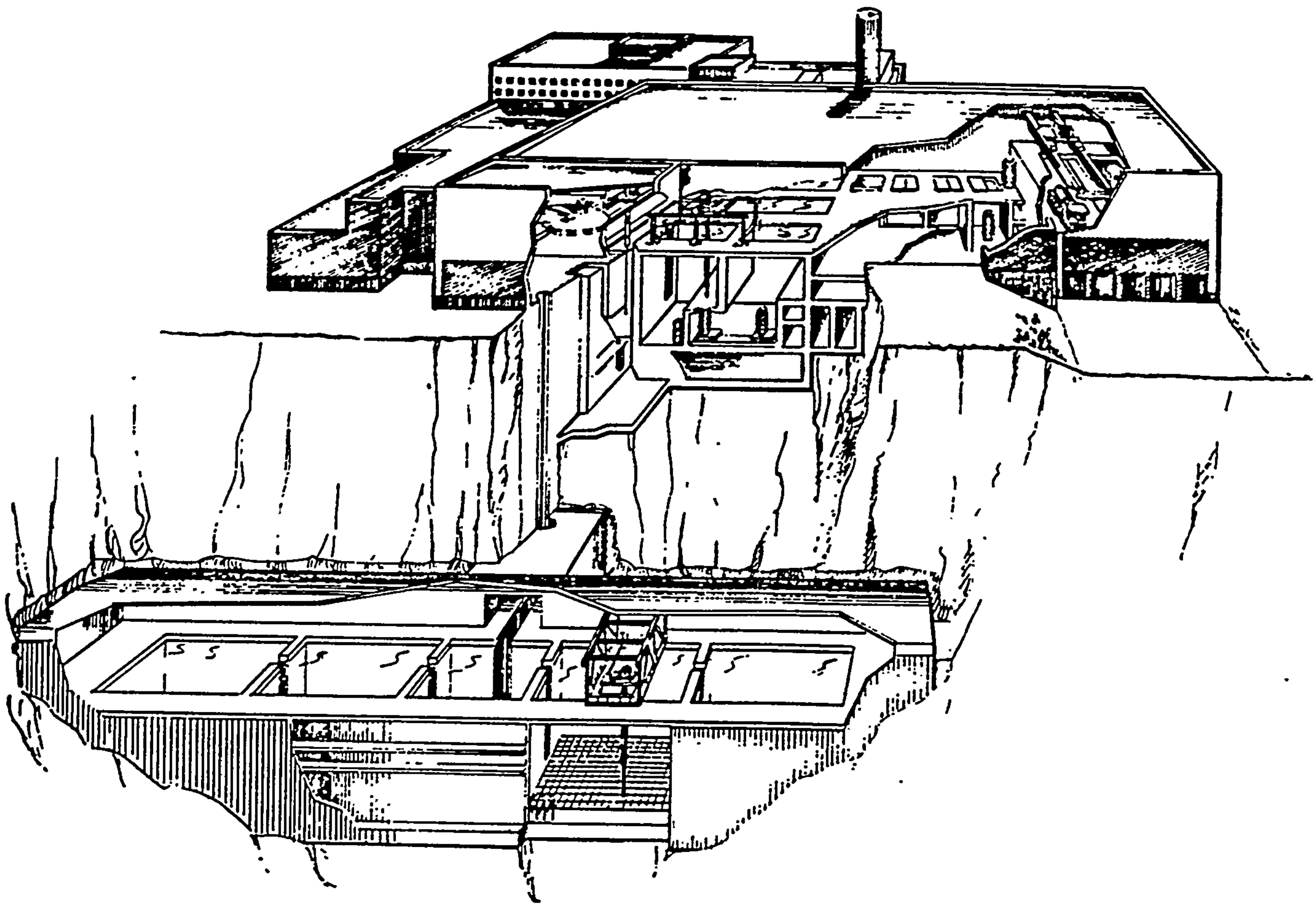

Figure 6.1.14. CLAB Phase 1

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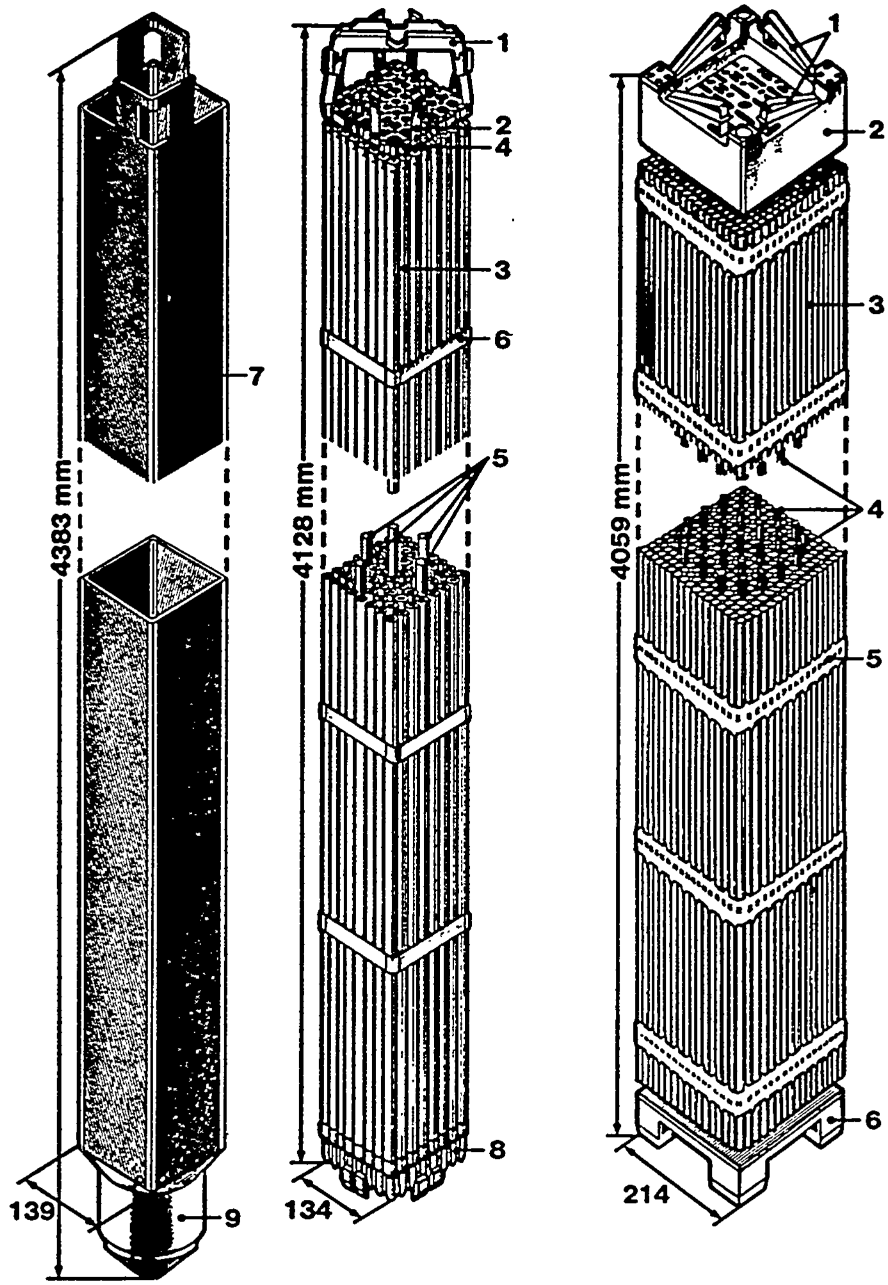

1 HANDLE
2 SPRING CLIPS
3 FUEL ROD BUNDLE
4 TOP TIE PLATE
5 TIE ROD
6 SPACER GRID
7 FUEL CHANNEL
8 BOTTOM TIE PLATE
9 TRANSITION PIECE

1 SPRING CLIPS
2 TOP TIE PLATE
3 FUEL ROD
4 CONTROL ROD GUIDE TUBE
5 SPACER GRID
6 BOTTOM TIE PLATE

Fuel assembly for a BWR reactor (ASEA-ATOM).

Fuel assembly for a PWR reactor (Westinghouse Ringhals 3 and 4).

Figure 6.1.15. BWR and PWR Fuel Assemblies

will follow the reverse procedure of unloading. During operation, the manpower requirement is about 75 for fuel handling with about 70 additional service personnel. After removal of all fuel and wastes from CLAB to the final repository, the surface facilities will be dismantled as well as any contaminated underground component. The radioactive wastes will be transferred to the SFL-4 repository for final disposal.

6.1.3.2.2 Encapsulation Station. Spent fuel assemblies arriving at the SFL-2 repository from CLAB will be encapsulated in copper canisters before they are transferred underground for final deposition. Figure 6.1.16 shows the encapsulation station where the spent fuel will be encased in the canisters and other wastes such as core components and reactor internals embedded in concrete molds. Figures 6.1.17 and 6.1.18 show examples of the copper canisters and concrete molds, respectively. Capacity of the encapsulation station is to be one canister per day or roughly 210 canisters per year, each canister containing about 1.4 MT (1.5 tons). This production capacity will provide an adequate safety margin to the annual spent fuel production from the 12 reactors.

Figures 6.1.19 to 6.1.22 detail the layout of the encapsulation station. The length of the building will be about 170 m (558 ft) and will be constructed mainly of concrete to meet radiation shielding and ventilation tightness requirements. The main parts of the facility are as follows:

1. Receiving station, and unloading and fuel handling bay. Included will be a transport cask repair shop, buffer storage for several transport casks, storage pool for fuel for about three months of encapsulation work, washing pits for washing the outside and flushing the inside of the casks, and pools for unloading the casks and transferring the fuel to the encapsulation section.

2. Fuel encapsulation section. This will be designed as a hot cell with no access for personnel. This is where the fuel will be actually encased in the copper canisters as described later in this section.

3. Dispatch section. This is where the finished canister will be cleaned, inspected and stored before being taken underground for final emplacement. Direct access to the shaft will be provided from the dispatch area.

4. Encapsulation area for core components (concrete casting).

5. Adjacent to encapsulation area there will be a service section containing warehouse facilities, lead melting equipment, etc.

6. Auxiliary systems such as cooling and purification systems.

7. Electrical and control section.

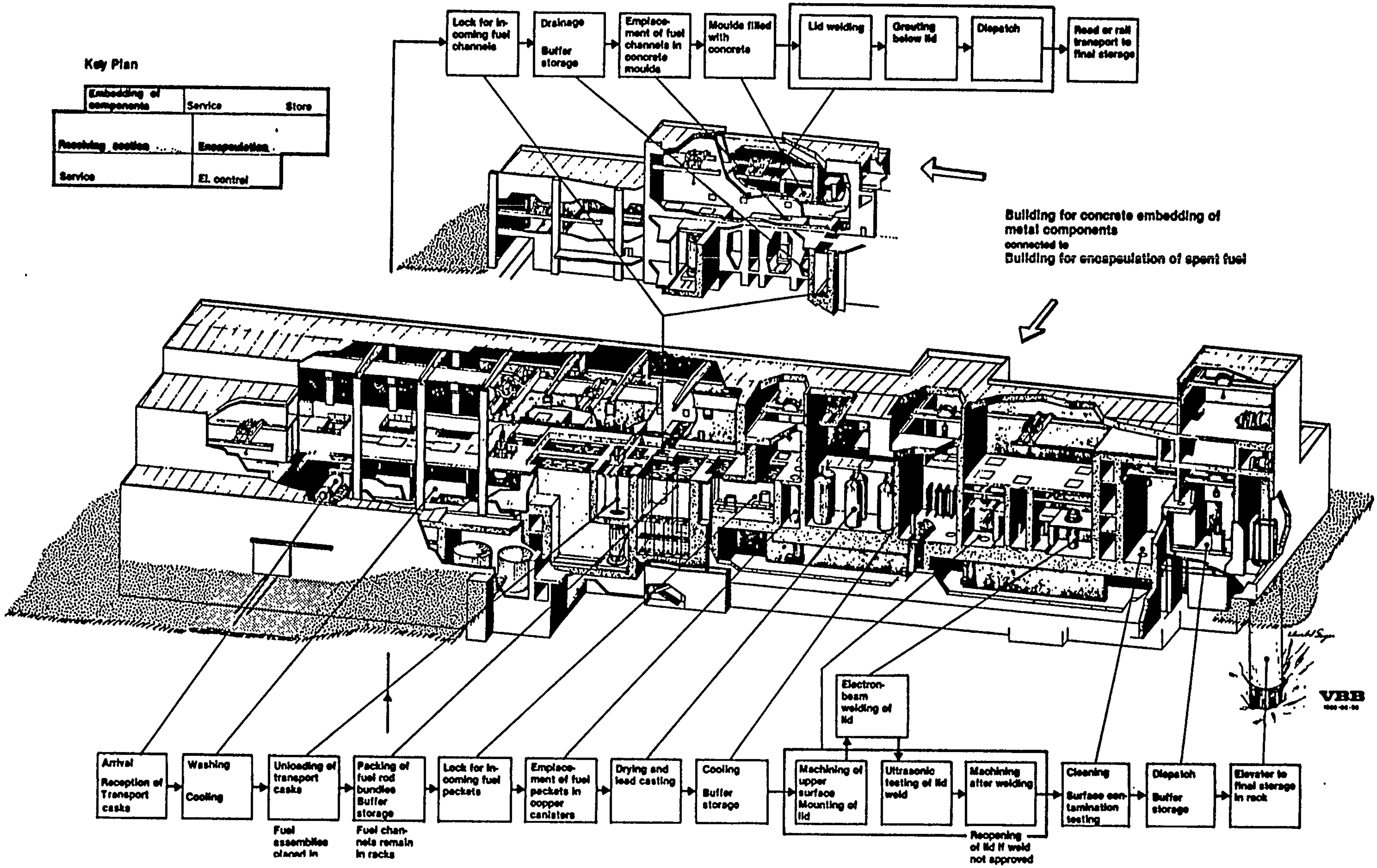

Figure 6.1.16. Encapsulation Station

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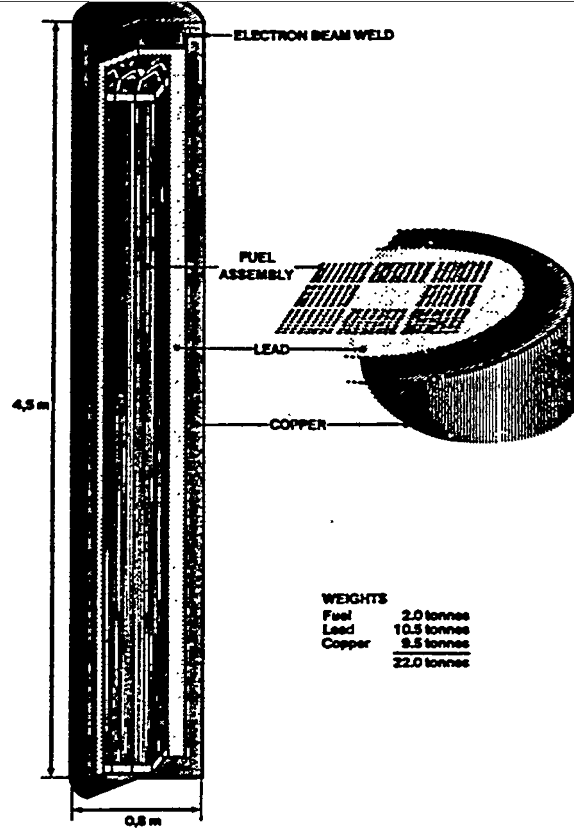
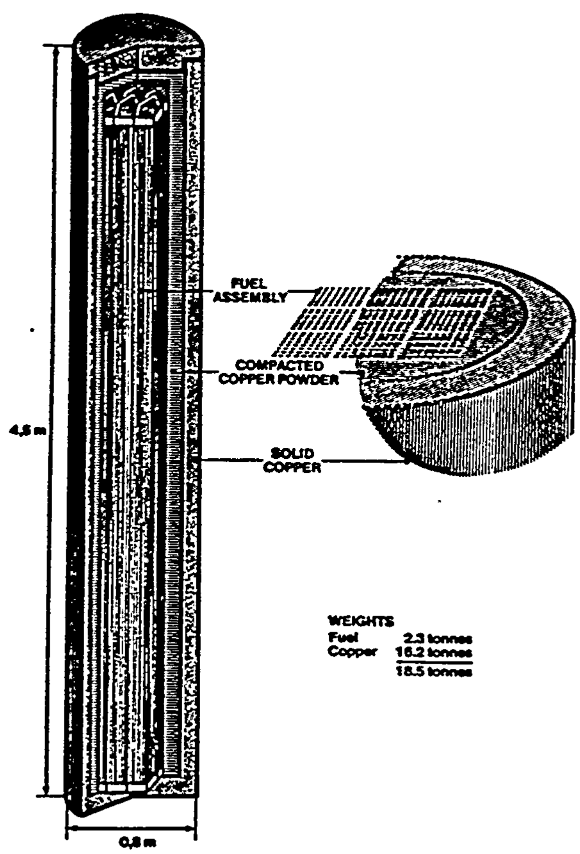


Welded copper canister.



Copper canister produced by hot isostatic pressing.

Figure 6.1.17. Spent Fuel Copper Canisters

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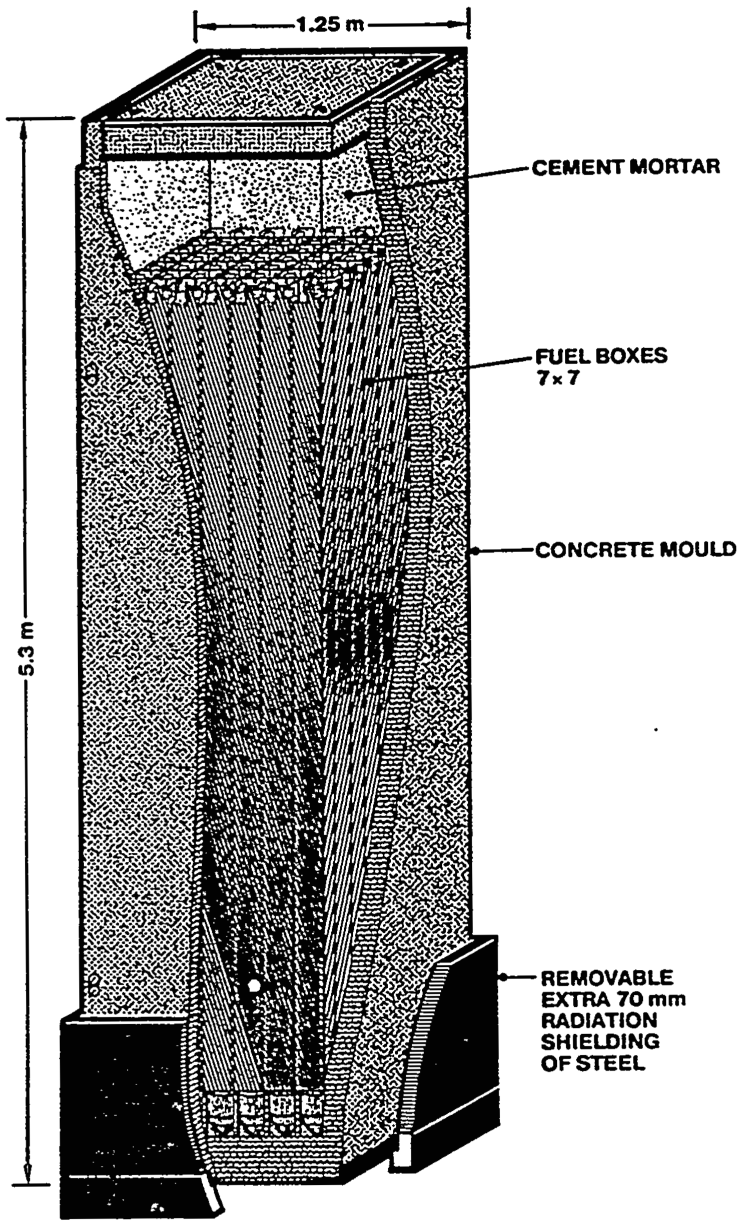

Concrete mould with fuel boxes.

Figure 6.1.18. Concrete Molds for Fuel Boxes

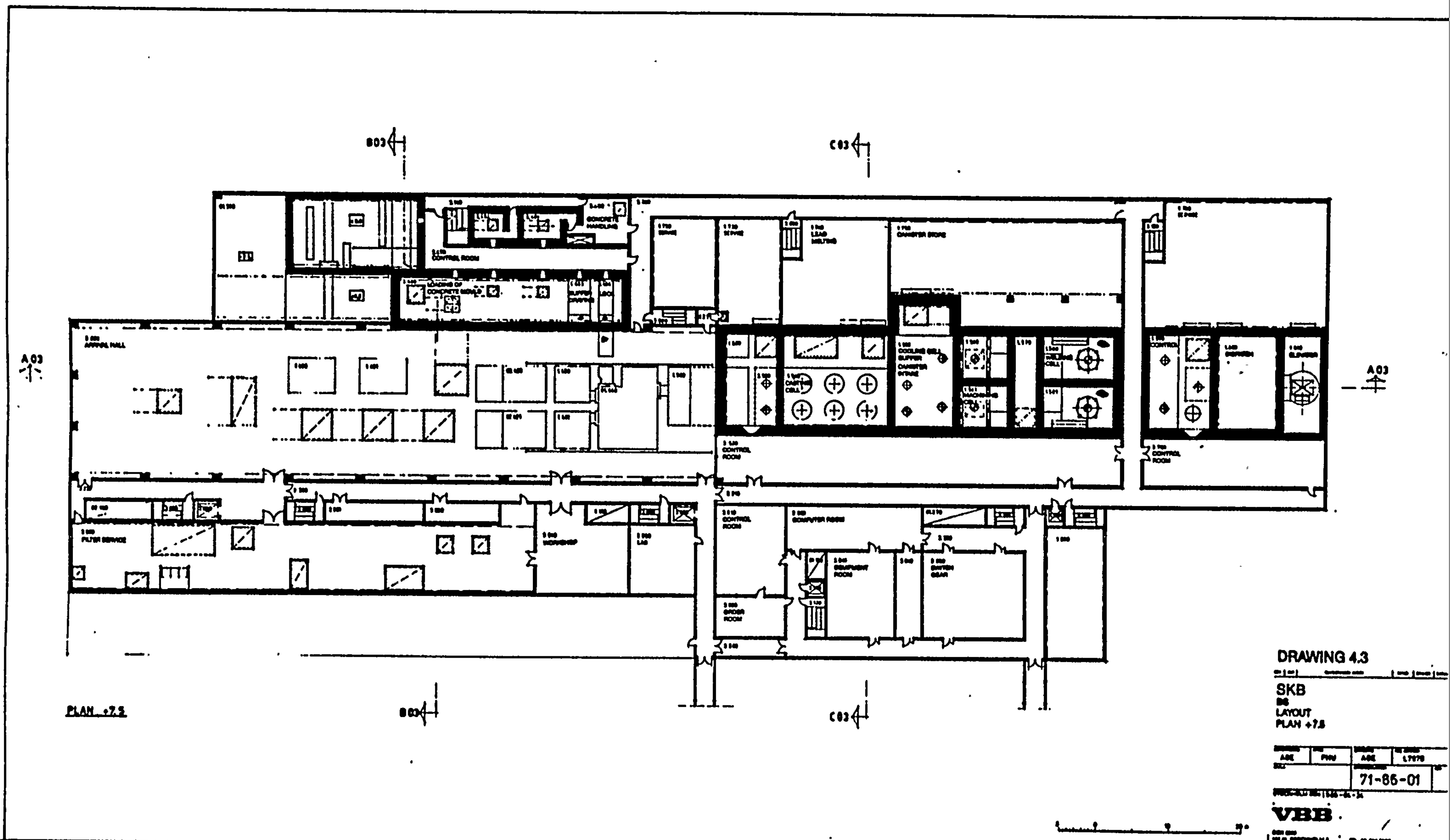

Figure 6.1.19. Encapsulation Station Detail 1

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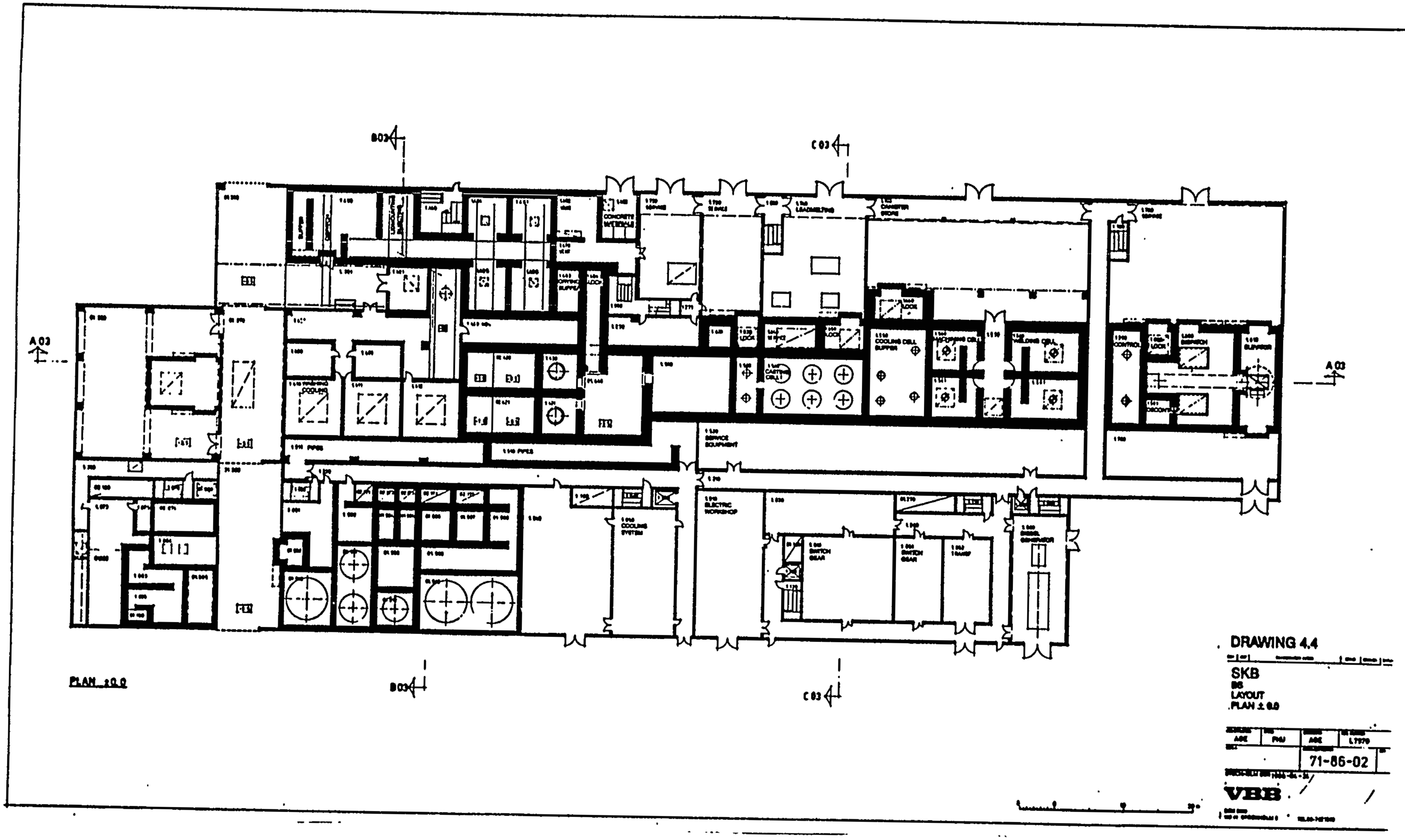

Figure 6.1.20. Encapsulation Station Detail 2

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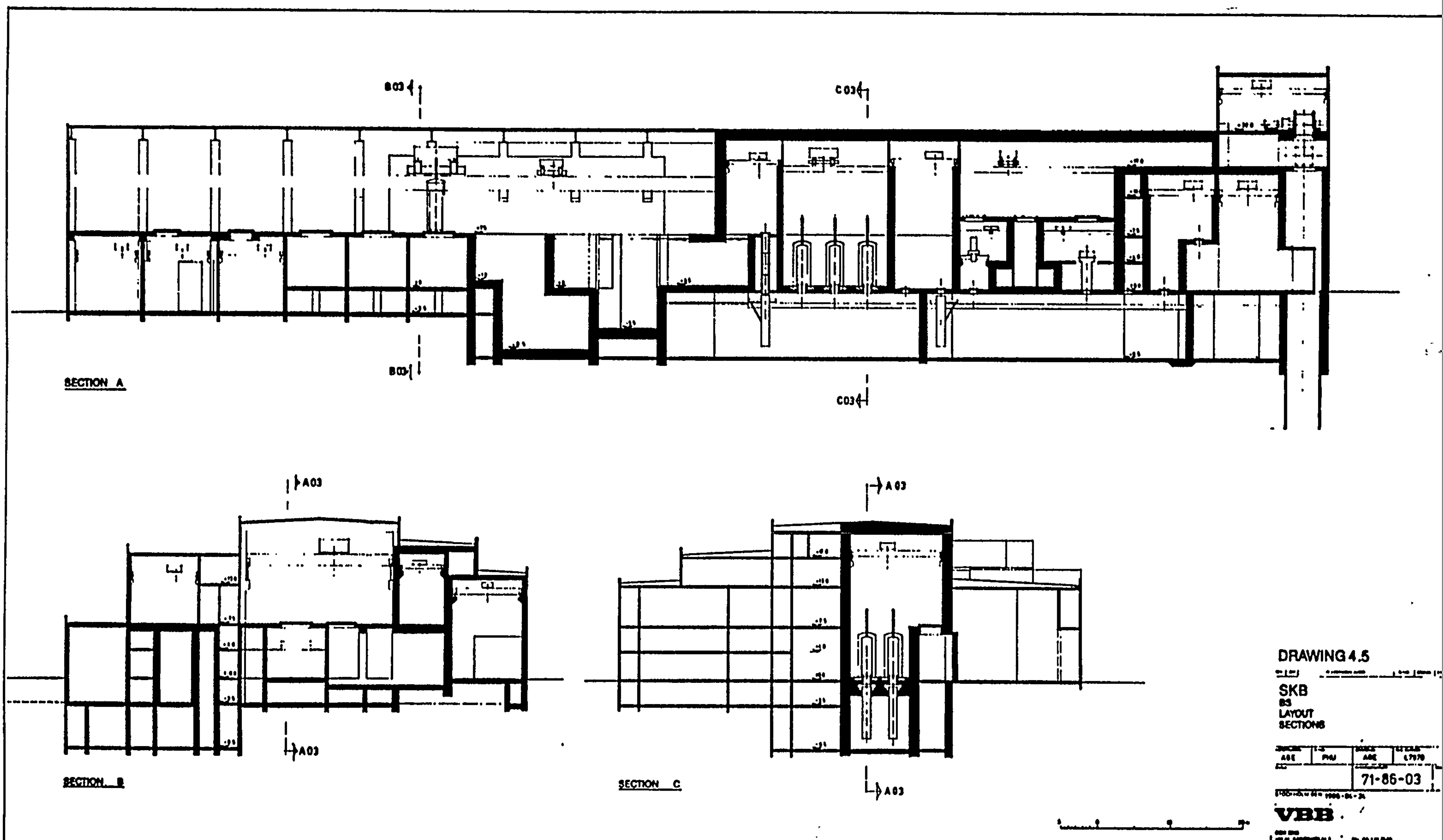

Figure 6.1.21. Encapsulation Station Detail 3

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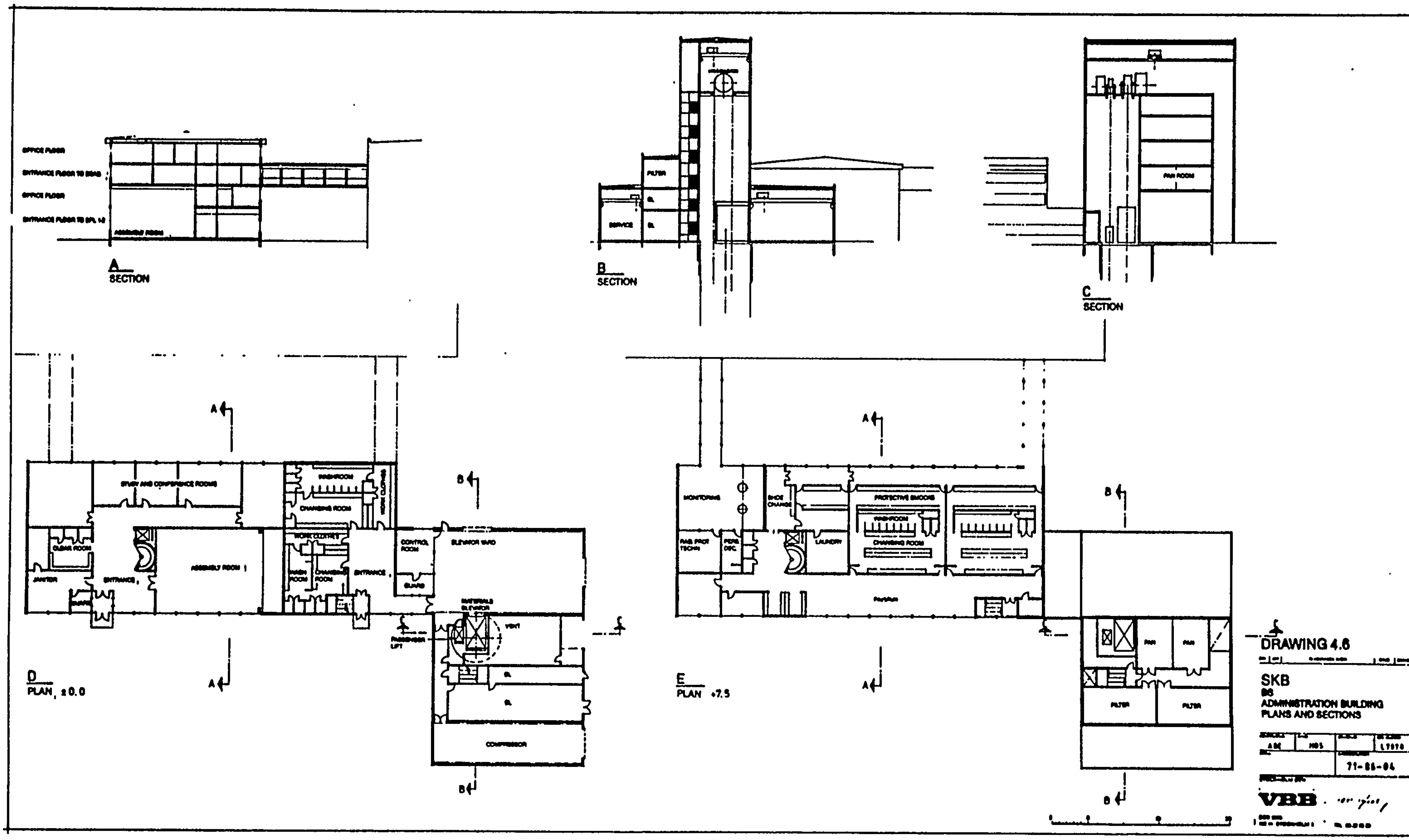

Figure 6.1.22. Encapsulation Station Detail 4

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8. Annexes to the encapsulation station will house the hoist and shaft services buildings as well as the headframe for the repository waste access shaft.

The encapsulation station will house several encapsulation systems, depending on the type of waste that is being handled.

Spent fuel will arrive at the receiving section in transport casks. The casks will be washed down, lifted upright and flushed internally in the washing and cooling cell. The cask will then be placed onto a wagon travelling on rails in the water-filled cask pool. The wagon will be driven under the unloading pool and aligned with a covered hatch in the bottom of the pool. Before removing the cover, a watertight connection will be made between the cask and the opening to prevent contamination of the water in the cask pool. The lid will be removed from the cask, the fuel assemblies transferred to a rack in the buffer pool, and the empty cask resealed and taken away for cleaning and, if needed, repair. While moving the fuel assemblies to the racks, the boiling water reactor fuel boxes or the pressurized water reactor boron glass rods are removed for separate encapsulation in concrete molds. When a rack has been filled with fuel, it is transferred to the encapsulation section where it is allowed to dry and placed in a prefabricated copper canister. At this point, two alternatives are being considered for sealing the canisters. One is filling the cavities around the fuel in the canister with molten lead and then welding on a lid. The other alternative involves filling the cavities with copper powder and, using a hot isostatic pressure (HIP) process, pressing a lid onto the canister and forming a homogeneous copper block with imbedded fuel elements. Both methods are discussed in more detail in Section 6.1.5.3.

In the receiving section, fuel handling will be carried out under water to shield against radiation. In the encapsulation section, fuel will be handled in air in remotely controlled cells with thick concrete walls for radiation protection.

After the canister has been sealed and received final inspection and acceptance in the dispatch section, it will be placed into temporary storage or transported to the waste shaft and taken underground for final emplacement in the SFL-2 repository. Some buffer storage of canisters is provided in the dispatch section to allow for interruptions in underground emplacement process.

The fuel boxes and boron glass rods remaining in the buffer pool after fuel assembly removal will be taken to the concrete casting area where they will be cast into a concrete mold. This is described in more detail in Section 6.1.5.3. The finished mold is then taken by truck or railway to the SFL-5 repository.

Other contaminated components such as decommissioning products, core components, etc., will arrive at the mold loading cell in a transport cask. The components will be unloaded and cast into concrete molds in a similar manner to the fuel boxes and boron glass rods.

6.1.3.2.3 SFL-2 Surface Facilities. The SFL-2 site layout will be similar to that shown in Figure 6.1.23. The largest building will be the encapsulation facility (designated BS). Apart from the encapsulation facility, other service and operational surface facilities at the repository may include the following, depending on the site location and ease of access:

- headframes, hoists, winches, etc., plus buildings at the shafts part form the waste shaft
- ventilation fans and housings at the ventilation shaft(s)
- Mine services such as compressed air plant, concrete mixing plant, electrical service, etc.
- Emergency power generation equipment
- Storage areas for excavated rock
- Backfill and concrete preparation area
- Waste water treatment facility
- Crew change house
- Lamp room
- First aid and mine rescue station
- Shop and warehouse buildings
- Offices
- Underground monitoring instrumentation system
- Sewage treatment facility
- Electrical substation
- Housing for operating personnel
- Railroad yard
- Site roads, fencing, and security facilities.

6.1.3.3 Underground Configuration

The SFL repositories described previously will be divided into two underground facilities. One will be the SFL-2 repository for spent fuel canisters and the other facility will include the SFL 3-5 repositories for reactor and decommissioning wastes. The plan is to locate the two

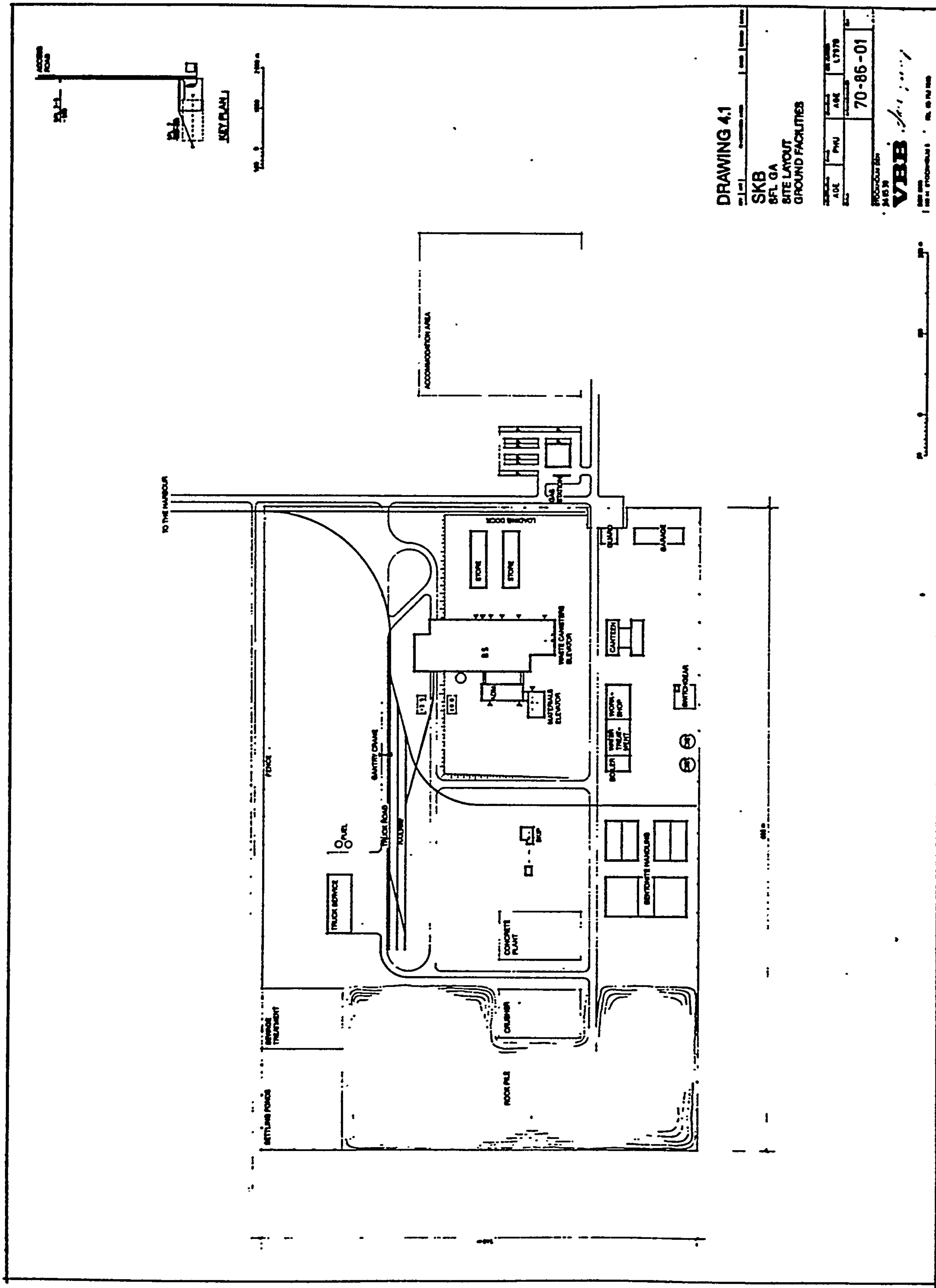

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Figure 6.1.23. Surface Layout

facilities close to one another to allow common use of some of the surface facilities such as the encapsulation station, administration, etc. This section describes the SFL-2 underground configuration. The other repositories are described below in Section 6.1.8.

6.1.3.3.1 The SFL-2 Repository. The final spent fuel repository will be located approximately 500 m (1,640 ft) underground. Access for the spent fuel from the encapsulation station will be via a specially equipped shaft. Additional shafts will be constructed for personnel and material access, excavated rock handling, and ventilation. The repository itself will consist of a network of about 38 km (24 mi) of emplacement tunnels as well as backup facilities such as transport tunnels, service areas, rock handling equipment, etc. The areal extent of the repository is expected to be about 1 km^2 (0.4 sq. mi) although the actual configuration will be determined by the heat generation of the spent fuel as well as in situ geologic conditions. Spent fuel emplacement in the repository may be on a single level or can be divided into two levels at about 500 m (1,640 ft) and 600 m (1,970 ft) below the surface. Figures 6.1.24 to 6.1.26 show examples of configurations that may be used. The waste canisters will be placed in vertical boreholes in the emplacement tunnel floors. A total of about 5,600 holes will be required.

In one configuration studied, the repository will be divided into two levels to allow separation of the emplacement operations from the other repository activities such as the excavation and sealing/backfilling work. The two levels will be connected via spiral drifts and the shafts. Spent fuel emplacement and excavation will be carried out concurrently. The configurations shown in the diagrams are schematic only, as the final layout will be optimized for the existing fracture geometry. This will be determined on the basis of an extensive exploratory drilling program during the site characterization and the excavation phases.

The repository will consist of a central service section connected to the encapsulation station on surface via the shafts, and a repository emplacement section. The central service section will be connected to the surface via three shafts (Figures 6.1.24 to 6.1.27) with the following functions:

1. The central or service shaft will be used as the main access shaft for personnel and materials. The shaft will be equipped with two cages (one for personnel and one for material) and will be used as the ventilation intake shaft. Other services such as compressed air, water, electricity, etc., will also be located in this shaft.

2. The skip or rock shaft will be equipped with a skip hoisting system to hoist the excavated rock to the surface. All excavated rock will be hauled to the 500 m (1,640 ft) level and dumped into the loading pocket/crusher installation. The skip capacity will be about 5 tons. Figure 6.1.28 shows a proposed arrangement. The shaft will also function as emergency egress and exhaust ventilation shaft.

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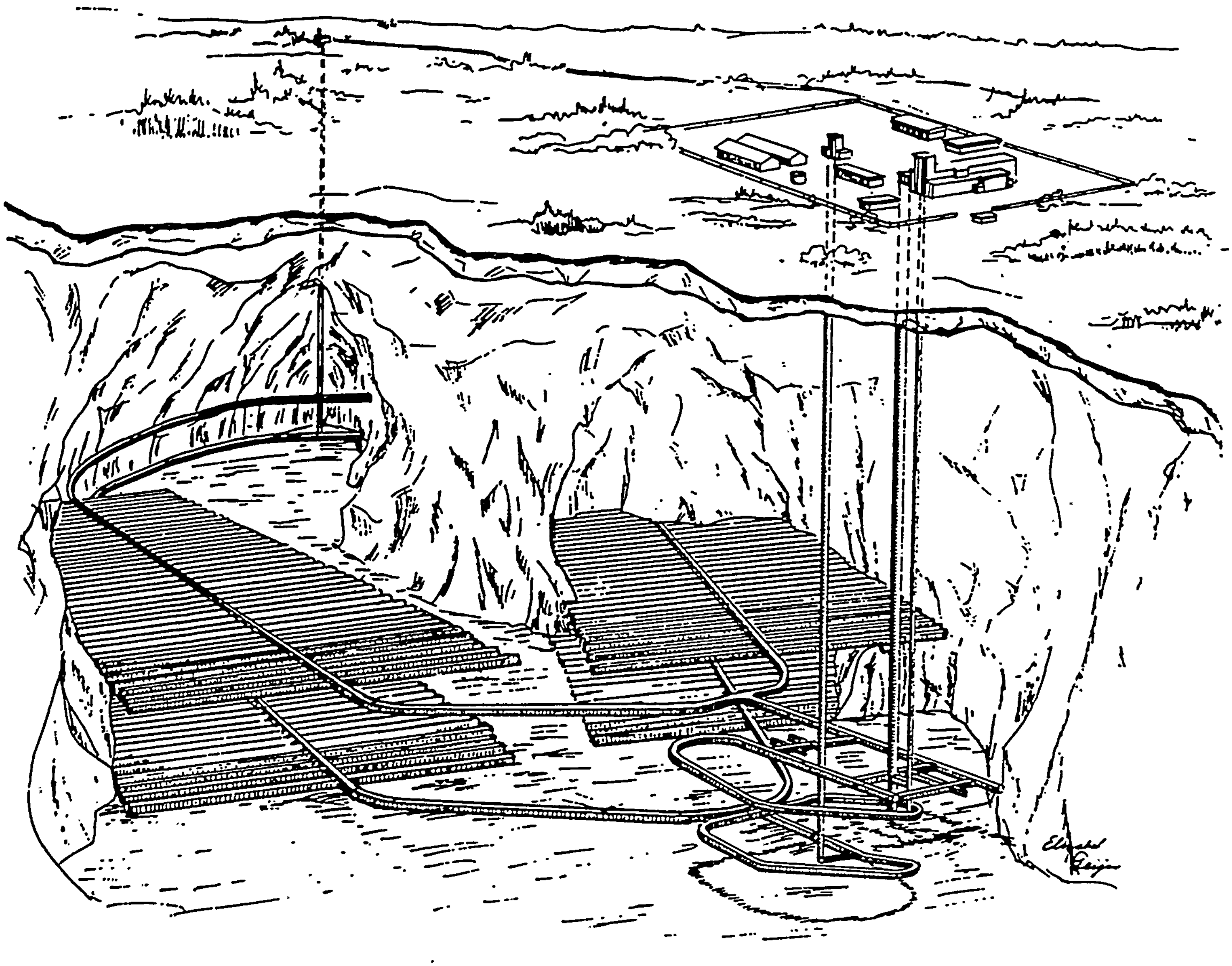

Figure 6.1.24. SFL-2 Repository Configuration

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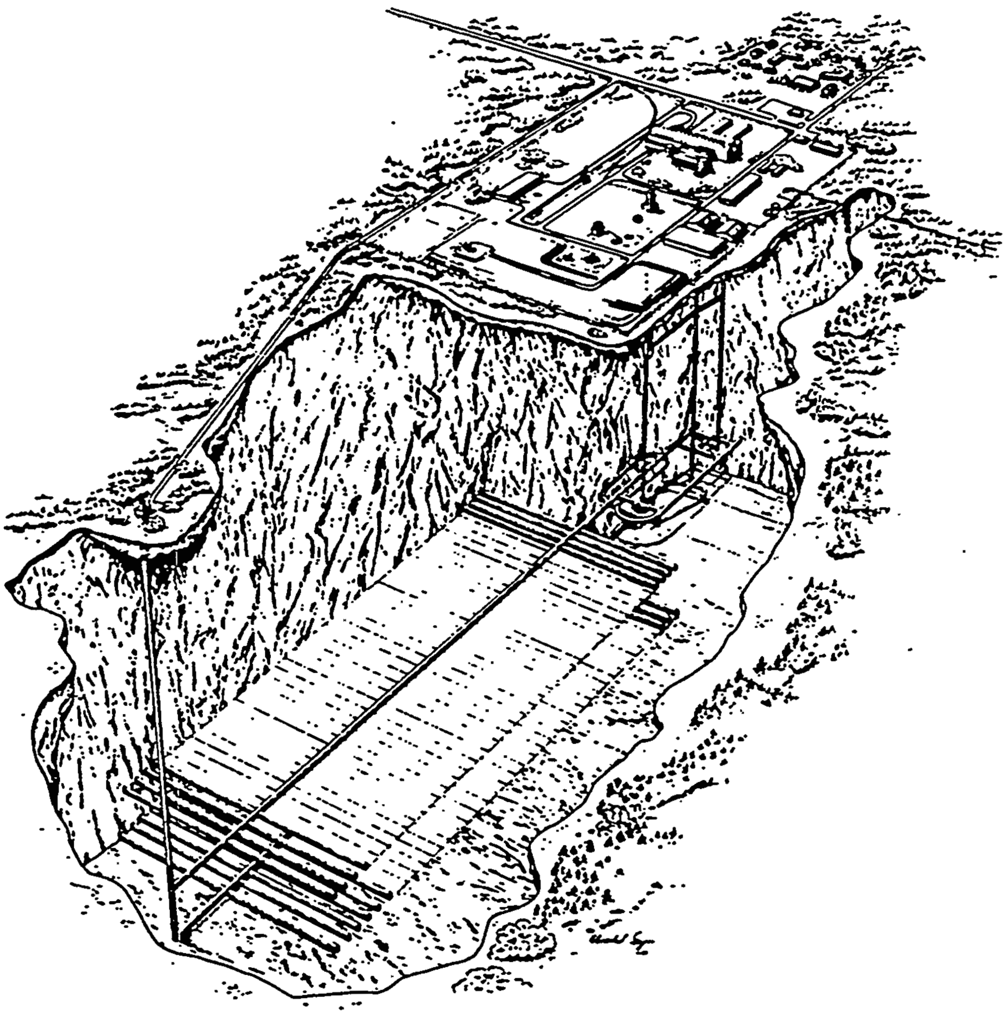

Figure 6.1.25. SFL-2 Repository Configuration

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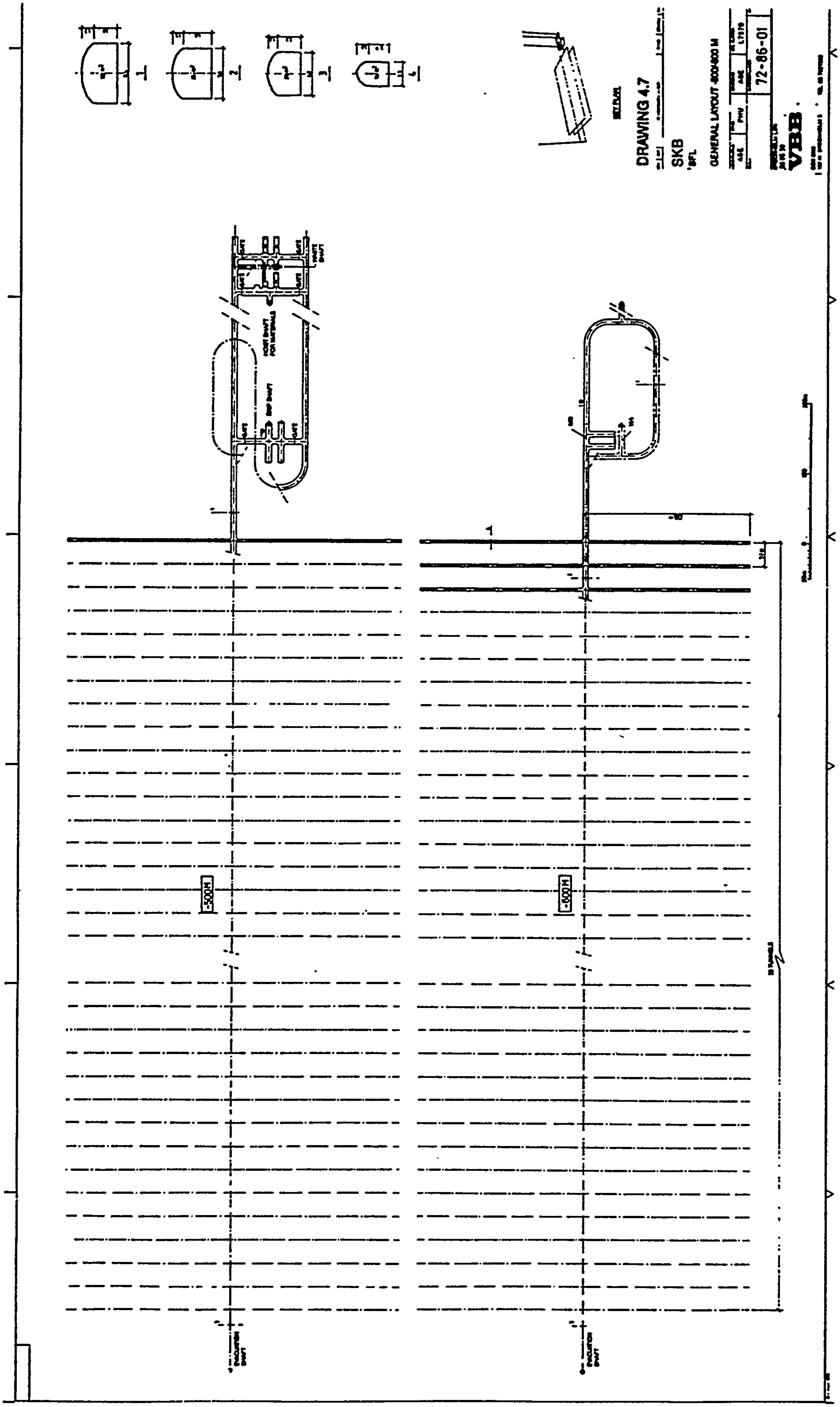

Figure 6.1.26. SFL-2 Underground Layout

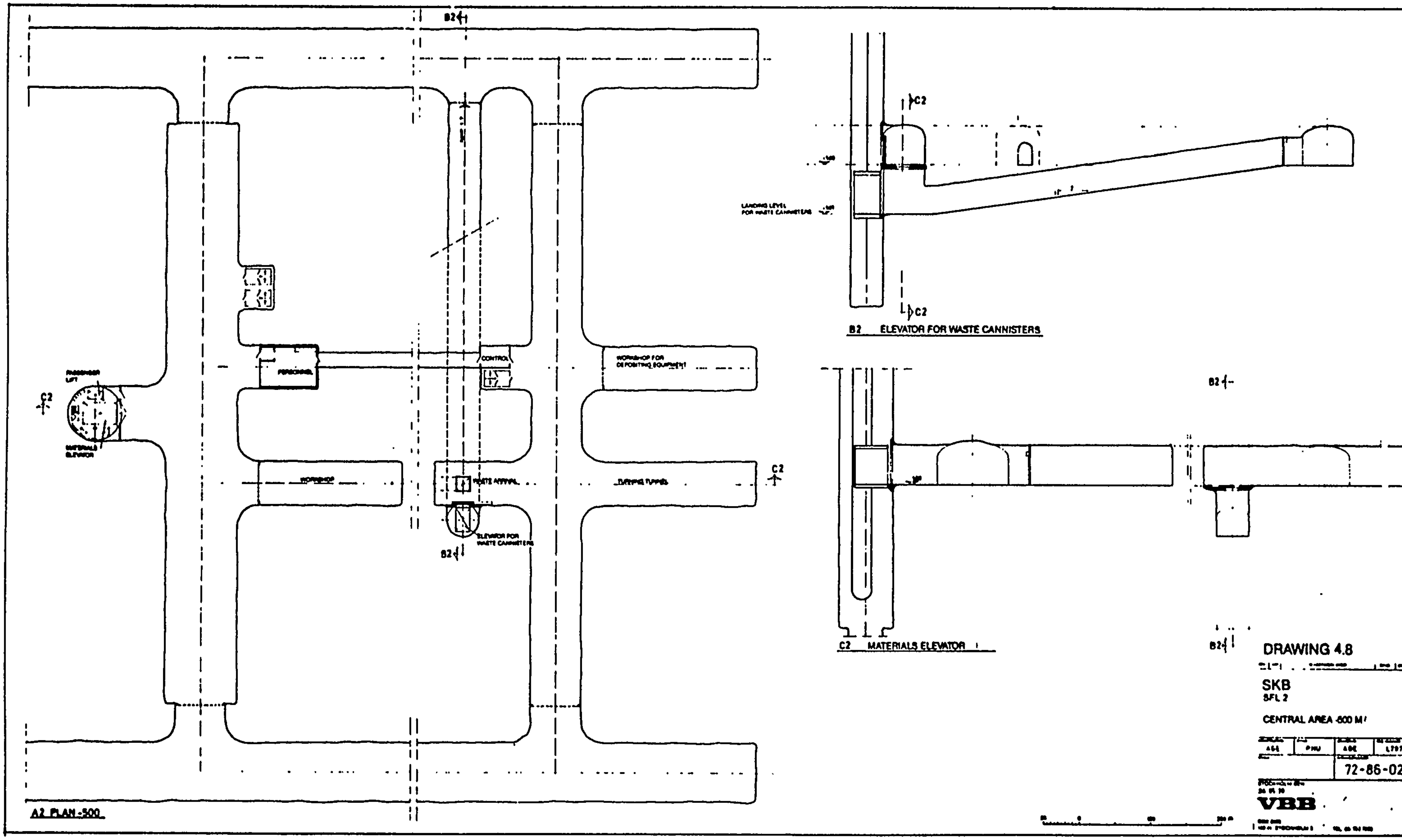

Figure 6.1.27. SFL-2 Shaft Area

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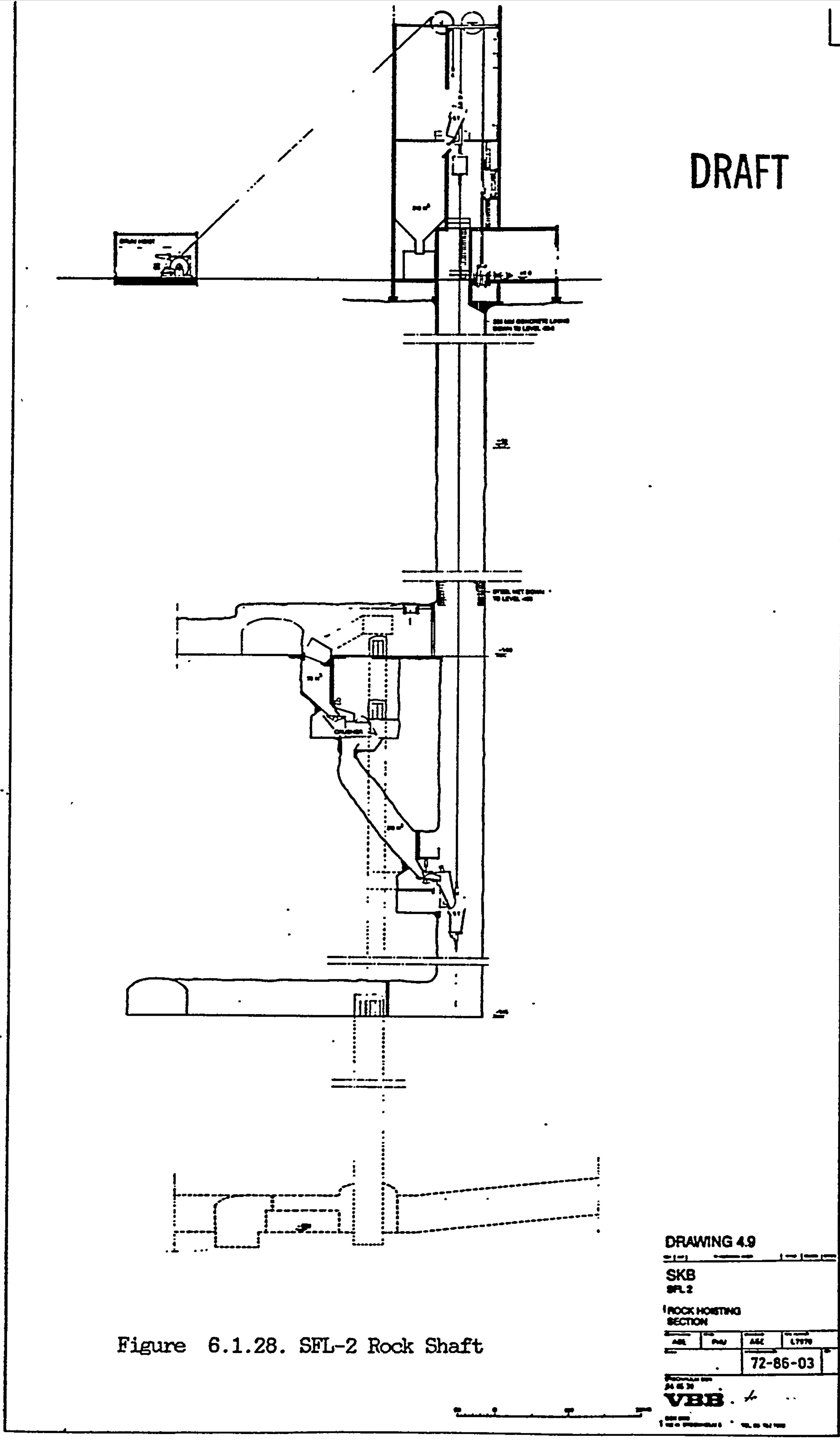

Figure 6.1.28. SFL-2 Rock Shaft

3. The waste shaft will be devoted solely to the lowering of waste canisters to the repository levels and will be equipped with a special heavy-duty hoist and cage to handle the heavy loads associated with this operation.

A fourth shaft will be constructed at the opposite extreme of the repository to the central service section to serve as the ventilation exhaust shaft and as an emergency evacuation shaft.

Due to the competent nature of the crystalline rocks, substantial support of the shaft wall with a concrete lining may only be required near the surface in the more weathered sections or where less competent overburden may be present. In the remainder of the shaft down to the repository level, a lining of rock bolts and wire mesh may be all that is needed to support the shaft wall and provide a safe working environment. Also, employing shaft sinking techniques using controlled blasting or shaft drilling will reduce the damage to the shaft wall and therefore enhance the long-term stability of the shaft structure.

The total excavation volume of the repository will be about 800,000 m^3 (1,050,000 yd^3), of which the emplacement tunnels account for about 500,000 m^3 (654,000 yd^3). The emplacement tunnels will have a minimum cross-sectional area of about 14 m^2 (150 ft^2) to allow passage of the canister transport vehicle. The service and haulage tunnels will vary in cross-sectional area from about 25 to 38 m^2 (270 to 410 ft^2) (Figure 6.1.26).

As was discussed above, the required tunnel support may be very minimal due to the competent structure of the crystalline rocks and the use of machine excavation techniques and controlled blasting will enhance the structural integrity of the openings.

The configuration of the emplacement holes is shown in Figure 6.1.29. The holes will be drilled into the floor of the tunnels. The diameter will be about 1.5 m (5 ft) and the depth about 7.5 m (25 ft). The actual hole spacing will be determined by the in situ conditions, but is expected to be about 6 m (20 ft).

A very important influence on the repository design is the heat generated by the spent fuel. The spent fuel canister generates a lot of heat, which raises the temperature of the surrounding buffer material and rock. A welded canister containing about 1.4 tons of boiling water reactor fuel gives off about 0.8 KW at the time of emplacement (i.e., after 40-50 years of storage). The heat conducting properties, the geometry of the deposition holes and final repository, and the rate at which the heat production declines determines the temperature variations of the material surrounding the canister. The thermal properties of the repository host rock will vary with location, geology, depth and climate, but the repository configuration will be adjusted to keep the maximum temperature below about 80°C (176°F). For example, assuming a virgin rock temperature of about 15°C (58°F) with a canister spacing of 6 m (20 ft) and a tunnel spacing of 25 m (82 ft), the maximum temperature for a single level repository would reach about 80°C (176°F) at the hottest canister. For a two-level

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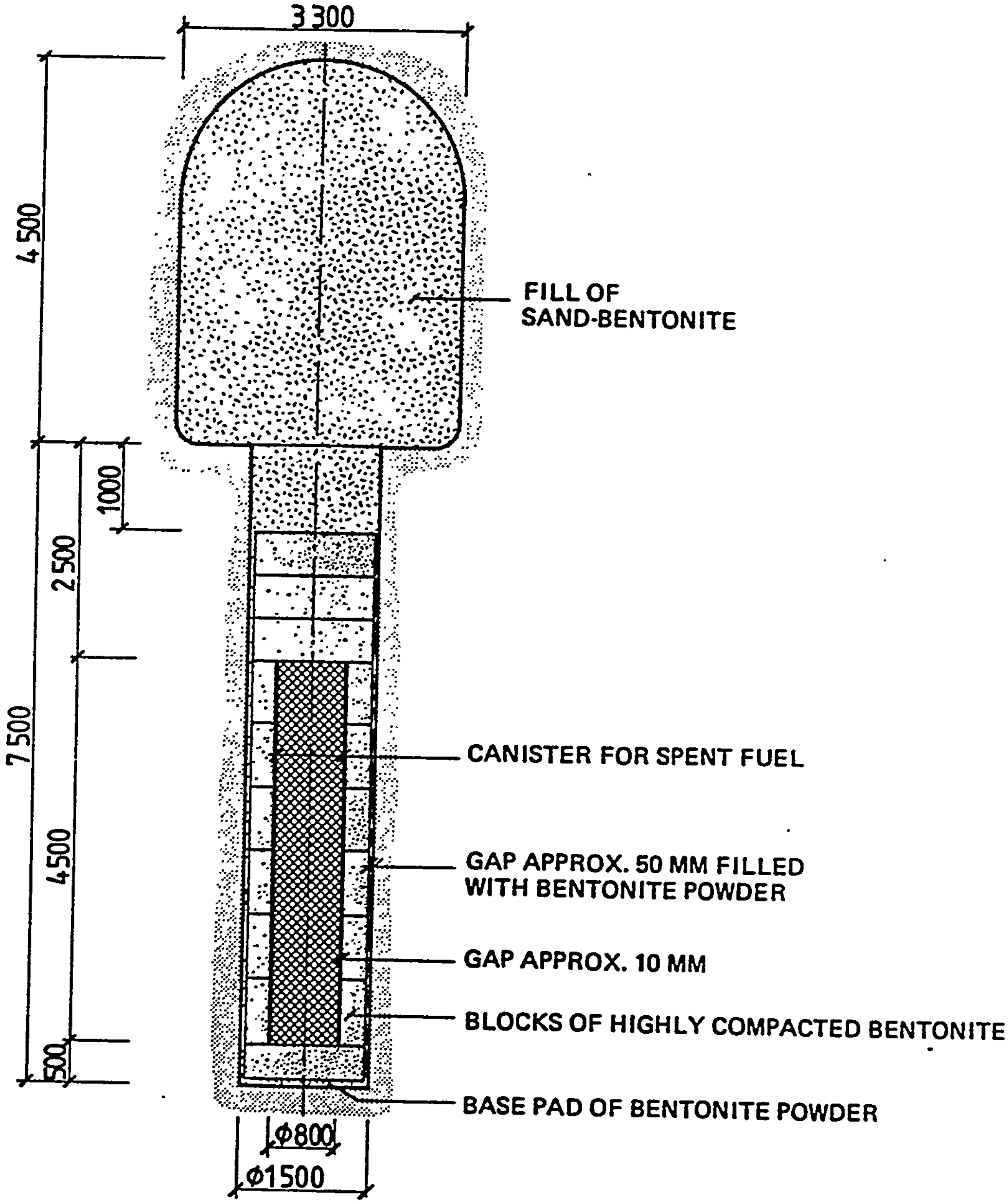

Deposition hole with canister and buffer material and with backfill in storage tunnel

Figure 6.1.29. Emplacement Hole Configuration

repository with about 100 m (330 ft) between levels, the tunnel spacing will have to be increased to about 33 m (110 ft) so that the maximum temperature does not exceed 80°C (176°F) with a virgin rock temperature of about 20°C (68°F) at the lower level. The reasons for limiting the temperature in the repository include operational restrictions during emplacement such as personnel safety and mine ventilation requirements, and as well as changes in buffer and sealing material properties that occur at temperature above 100°C (212°F).

6.1.4 Repository Construction

This section describes the construction methods that are likely to be adopted for the construction of the SFL-2 repository. The descriptions are based on currently available technology and accepted civil and mining construction practices. Extensive experience will be gained during the construction of the CLAB facility and the SFR repositories, which are currently in operation or under construction. The construction and operation of nuclear waste handling facilities (reprocessing plants, interim storage facilities, etc.,) in other countries may also provide valuable input to the Swedish construction program. In addition, Sweden has always had a very active and advanced mining and underground construction industry and thus has available a wealth of experience and technology for the construction of a repository in the crystalline rock environment.

6.1.4.1 Schedule

Figure 6.1.2 shows the current timetable for the construction and operation of the various facilities of the Swedish nuclear waste disposal program. Construction of the SFL-2 repository is scheduled to start around 2010 and will last for about 10 years. Included in this construction phase will be installation and commissioning of all surface facilities, the shafts and underground service excavations and some of the deposition tunnels to allow full spent fuel encapsulation and underground emplacement to start around 2020. The construction of all facilities will therefore have to be scheduled to be completed within the 10 year period such that all facilities will be completed and operational for the start of waste deposition. Surface and underground construction will most likely be concurrent and therefore detailed construction scheduling will be needed to reduce interference between the various activities. Allowance for changes in actual repository configuration must also be included in the underground construction program based on information from underground site investigation.

6.1.4.2 Methods and Equipment

6.1.4.2.1 Surface Facilities. The surface facilities were described in Section 6.1.3.2. A lot of these facilities (headframes, hoists, ventilation system, underground services, etc.,) are normally part of a underground mining operation and thus construction of these will present no unusual requirements. Stringent standards, however, will govern the construction of facilities such as the encapsulation facility to ensure adequate radiation protection during the nuclear waste handling process. Construction experience in this area will be available from construction projects of similar nature (e.g., CLAB) in Sweden and abroad as well as the nuclear power generation industry.

6.1.4.2.2 Shafts. The SFL-2 repository configuration requires construction of four shafts. The sequence of shaft construction will depend on program priorities, construction times and overall project scheduling requirements. However, in order to expedite the underground construction program, the first shaft to be sunk should be the skip or rock shaft. Completion of this shaft will provide the excavated rock hoisting facility needed for the construction of the underground tunnels. As this is the first shaft, sinking will likely be by conventional drill-and-blast methods, although blind shaft drilling technology may have advanced to the stage where it can also be applied. Once the repository horizons have been reached, development of the stations and service area tunnels can commence, using the skip shaft to remove the excavated rock. Figures 6.1.26 and 6.1.27 show a possible configuration of the shaft area and Figure 6.1.28 shows a skip shaft configuration. The skip hoist will most likely be a double-drum arrangement for a two-skip setup in the shaft for hoisting the excavated rock. The arrangement shown in Figure 6.1.28 requires all rock to be transported to the 500 m level, where the 5 ton capacity skips are loaded by a crusher and loading pocket arrangement.

In order to free the skip shaft from any personnel traffic and therefore allow the rock hoisting capacity to be fully utilized, construction of the other shafts should be completed as soon as possible. Several alternative methods can be used for the sinking of these shafts. They can be sunk concurrently with the skip shaft so that all shafts are complete at about the same time. Since access can be established to the bottom of the waste and service shafts shortly after completion of the skip shaft, some type of shaft drilling method may also be applied to construct these shafts. The use of shaft drilling methods will depend on several factors such as shaft diameter, rock strength and shaft depth but a method such as raise boring should be applicable for either drilling of the complete shaft or drilling of a shaft pilot hole for subsequent enlarging using drill-and-blast techniques (slashing). Drilling techniques have many inherent advantages such as enhanced personnel safety and speed. Also, an important advantage of drilling techniques from the viewpoint of repository sealing and long-term integrity is the minimized rock damage around the shaft opening.

Shaft diameters have obviously not been finalized at this point. The sizes will be derived once the final repository configuration and thus ventilation and hoisting requirements have been finalized. As a first approximation, however, based on mines of similar size, the shaft sizes may be in the range of 5 to 8 m (16 to 26 ft) in diameter, depending on their function.

All shafts will be furnished with guided conveyances. The waste shaft will have a specially designed cage and hoisting system to handle the waste canister and will be dedicated solely to canister transport. The service shaft will be equipped with a large material cage and a personnel cage and the skip shaft with two rock skips. The exhaust shaft is planned to be an emergency egress shaft and will thus be equipped with a mancage of sufficient size to allow evacuation of all underground personnel within a specified period. The hoist will most likely have an independent power supply in case of power failure during emergencies.

One or more of the shafts could also be constructed as part of the site exploration and characterization program. This would also expedite the repository construction process since the shafts could be incorporated into the repository concept. Incorporation of the exploratory shafts into the final repository configuration would mean that development of the access drifts and service entries could begin immediately after completion of site characterization and site acceptance.

6.1.4.2.3 Tunnels. Once the service area ramps and skip loading arrangements are complete, construction of the service headings and emplacement tunnels can begin. Figures 6.1.24 and 6.1.25 show possible repository arrangements and Figure 6.1.44 shows examples of adaptation of the repository layout to the fracture geometry of the area. As with the shafts, excavation techniques for the tunnels will be determined by rock strength, tunnel dimensions, schedule requirements, etc., but the same advantages of shaft drilling (personnel safety, speed, and structural integrity of the rock around the tunnel) also apply to mechanical tunnel excavation. Tunnel drivage techniques vary from conventional drill-and-blast methods using hand-held drills to full-face, completely mechanized tunneling methods. Partial mechanization can be achieved by the use of drill jumbos or partial face tunneling machines (e.g., roadheaders). Drill-and-blast techniques provide greater flexibility in terms of direction changes and tunnel profiles since fully mechanized systems are often uneconomic for short tunnels due to machine costs and long set-up times. The muck removal system used with the various excavation techniques can include LHD (Load-Haul-Dump) vehicles, trucks and conveyor belts. The final repository configuration, personnel experience, the state of equipment development in similar rock condition as well as safety and economic requirements will ultimately decide the methods used for construction of the repository.

The connections from the service area to the exhaust shaft should have priority to establish the ventilation system for the repository followed by construction of the emplacement tunnels. The length of the connection headings on each level as shown in Figure 6.1.26 will be approximately 1,500 m (4,900 ft). The length of each emplacement tunnel will be about 255 m (830 ft), with a total of about 76 tunnels per level. The total length of emplacement tunnel excavation will therefore be about 38,700 m (127,000 ft). Thus, the minimum tunnel excavation rate for emplacement of about 210 canisters per year is roughly 1,500 m/year (4,900 ft/year), based on the output of the encapsulation station. Higher excavation rates must, however, be planned for to account for construction delays due to unknown fracture zones and rock conditions determined by the underground investigation program, unsuitability of some emplacement hole locations, unscheduled equipment breakdowns, etc. Tunnel excavation and canister emplacement will take place concurrently. Figure 6.1.30 shows an example where the two operations are be effectively separated with excavation and emplacement taking place on separate levels. A series of gates or bulkheads are installed to separate the two operations from one another with each operation having access to the appropriate shafts and facilities. A change-over every 3 years, say, would require excavation of roughly 5,000 m (16,500 ft) of tunnel and preparation of about 700 holes. Emplacement hole drilling can be carried out as soon as a tunnel has been completed.

Preparation of each emplacement hole (Figure 6.1.29) will start with the drilling of an 150 mm (6 in) diameter cored hole at the center of the prospective emplacement hole location. The core will be used to determine the suitability of the location as an emplacement site based on the rock's structural and permeability properties. If the site is found suitable, large-diameter drilling will commence, after first pouring a thin concrete collar slab around the borehole. The purpose of the concrete slab will be to prevent water from the tunnel floor flowing into the borehole as well as provide a flat support surface for the outriggers of the deposition vehicle.

6.1.4.3 Cost

In 1986, a cost estimate of the Swedish waste disposal program was published (4). A summary of these costs is presented in Table 6.1.2. This shows that roughly 55% ($3,343,000,000) of the overall estimated waste disposal costs of $5,782,000,000 are associated with the spent fuel disposal program (includes transport, CLAB, encapsulation (BS), and SFL-2 costs). Underground costs for the SFL-2 repository break down into approximately $342,000,000 for repository construction, $53,000,000 for operation, $280,000,000 for sealing and decommissioning. Costs are based on a January 1986 exchange rate of 1 U.S. $ = 7.73 SEK.

6.1.5 Waste Emplacement Cycle

This section describes the actual handling of the high-level nuclear waste from the reactor to the final emplacement location in the underground repository.

6.1.5.1 Waste Treatment

The Swedish nuclear high-level waste disposal program does not presently include reprocessing of spent fuel as a treatment option. The program is based on direct disposal of the fuel after an interim storage period. The fuel will therefore be stored for about 40 years at the CLAB facility to allow the heat and radioactivity content to reduce. After this interim storage period, the fuel will be taken to the encapsulation station

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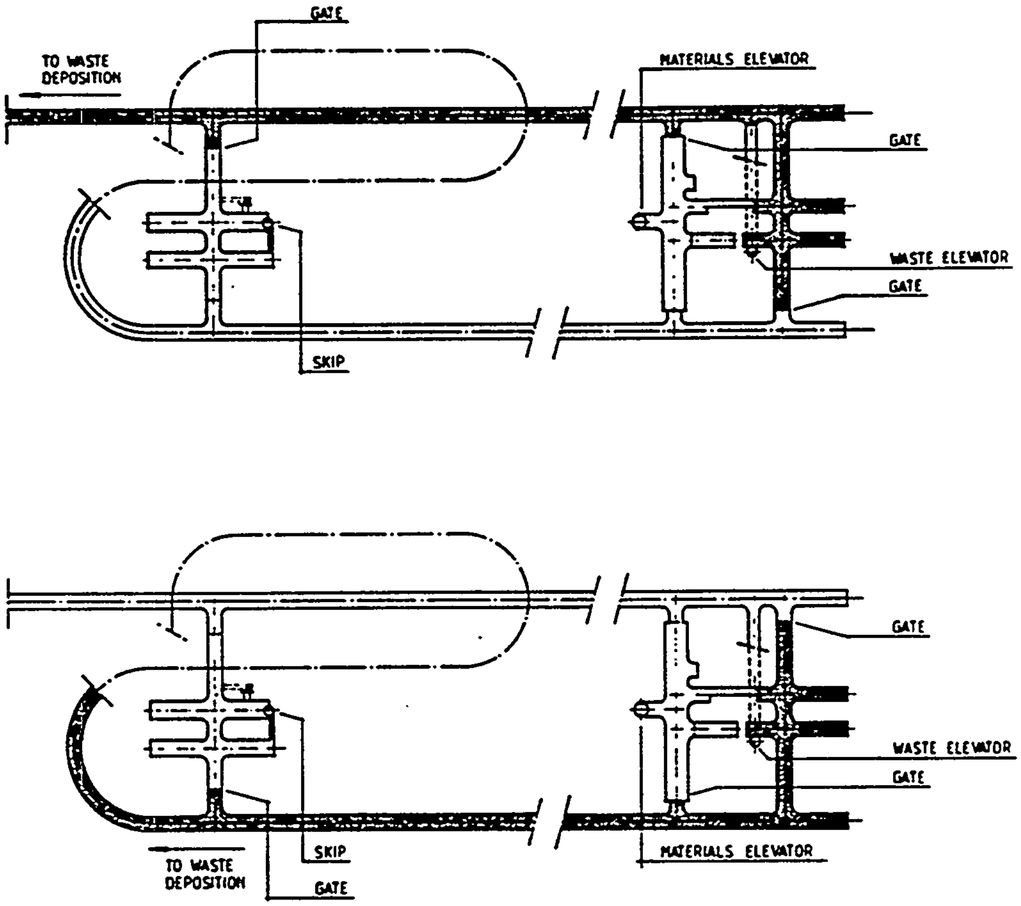

Figure 6.1.3Ø. Emplacement/Excavation Arrangement

Table 6.1.2. Estimated Cost of Waste Disposal Program (after SKB 86-12)

COST ITEM	$ U.S. (millions)
SKB, Admin., R & D	463
Transport	245
Decommissioning Power Plants	978
CLAB	1,078
Encapsulation Facility Administration	457
Encapsulation Facility Construction and Operation	888
SFL-2 Repository	675
SFL 3-5 Repositories Administration	101
SFL-3 Repository	53
SFL-4 Repository	5
SFL-5 Repository	23
SFR Repositories Administration	18
SFR-1 Repository	145
SFR-3 Repository	53
Remaining Reprocessing Contracts	600
TOTAL	5,782

(1 $ U.S. = 7.73 SEK - January 1982)

at the final repository to be encased in copper canisters before being taken underground for final disposal in the SFL-2 repository. The spent fuel assemblies are shown in Figure 6.1.15 and will be either boiling water reactor fuel or pressurized water reactor fuel. The overall disposal strategy is not affected by the fuel type. Both assemblies will be encased in copper canisters and then placed in the disposal holes in the underground repository. Only the number of assemblies per canister will be affected by the fuel type since the heat generation is higher for pressurized water reactor assemblies. However, the encapsulation system will be set up to handle only one type of fuel or a combination and thus the encapsulation process will not be significantly affected by the fuel type.

6.1.5.2 Waste Transport

The transport of nuclear waste involves a considerable effort in moving the waste from the production site to the final repository. Some waste such as spent fuel and core components will also have to be moved to and from the central interim storage facility (CLAB). In Sweden, all existing nuclear facilities are near the coast and thus sea transport can be utilized. The site of the final high-level waste repository SFL-2 has not yet been chosen, but should it be located further inland, the sea transport system will be augmented by a rail link between the SFL-2 and a suitable harbor facility, using existing rail lines as much as possible. The whole transportation system will include the containers and casks, ship, terminal equipment and rail transport system.

Heavy transport containers holding several waste units are used to protect the environment from the radiation and also protect the load against damage during transport. The spent fuel transport cask is made of a cylinder of thick steel and has been provided with a neutron-shielding layer and cooling fins on the surface. Figure 6.1.31 shows the cask currently being used to transport spent fuel. The casks, designation TN17/MK2, are designed according to the International Atomic Energy Association's regulations and are designed to withstand extreme stresses. The casks, approximately 6.15 m (20 ft) long and 1.95 m (6.5 ft) in diameter, can hold 17 boiling water reactor fuel assemblies or 7 pressurized water reactor assemblies and weigh about 80 tons, which includes about 3 tons of uranium. The cask is carried on a specially designed transport frame during transit.

The vehicles used to handle the casks at the terminal are shown in Figure 6.1.32. The vehicle is a low ground-speed, 6-axle unit and the bed can be raised and lowered hydraulically to pick up and off-load the frame with the cask.

Sea transport is carried out using the specially constructed ship, M/S Sigyn, shown in Figure 6.1.33. The ship is a combination roll-on/roll-off and lift-on/lift-off vessel so that the casks can be either driven on or off using the transport vehicle or lifted in or out of the cargo hold using a crane. The ship is about 90 m (295 ft) long, weighs 2,000 tons empty, and has a payload capacity of 1,400 tons. Extensive safety features have been

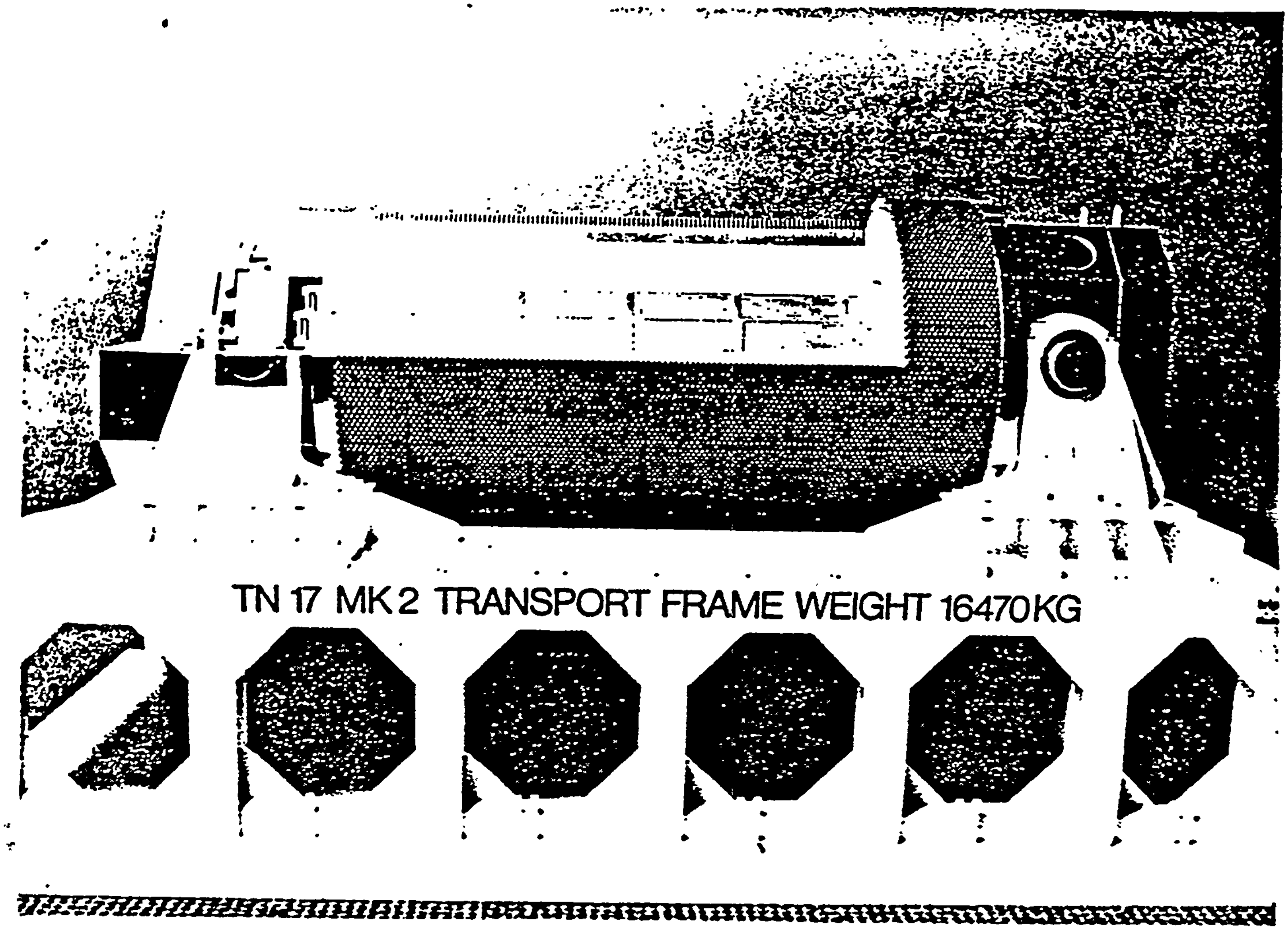

Figure 6.1.31. Spent Fuel Transport Cask

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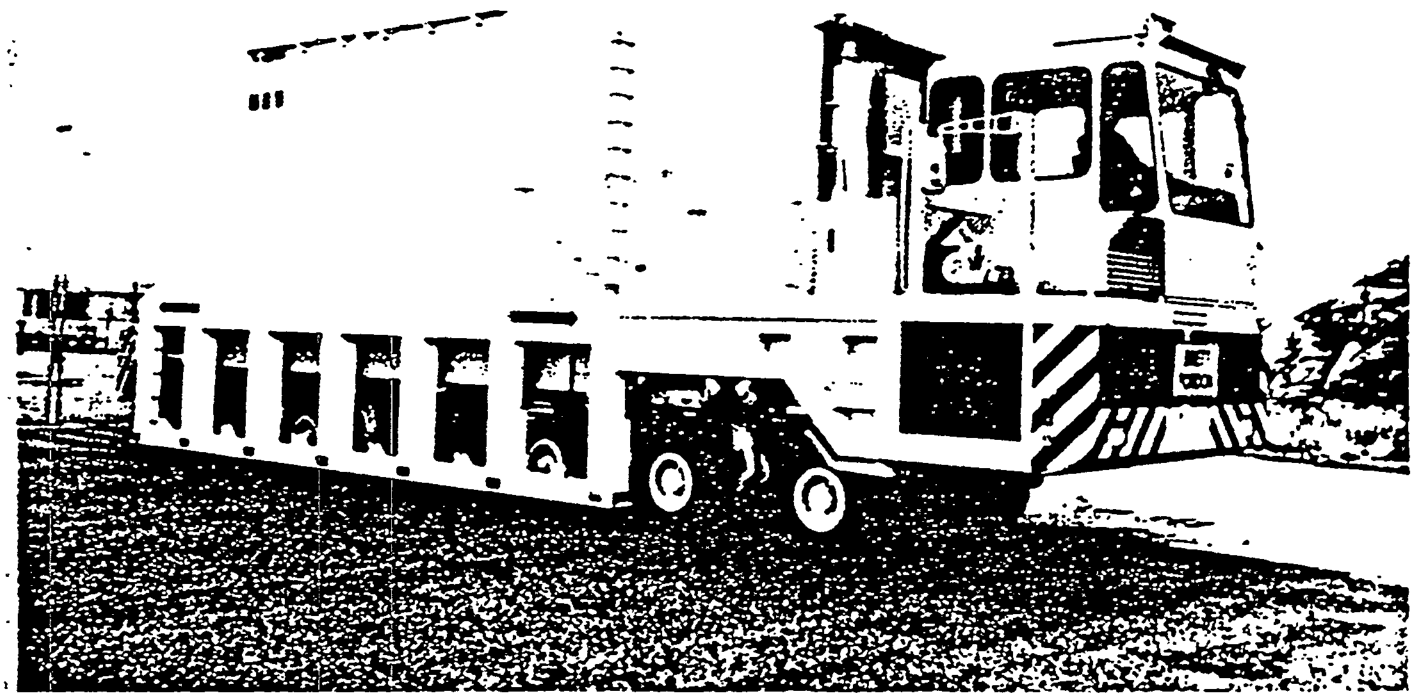

Figure 6.1.32. Transport Vehicle

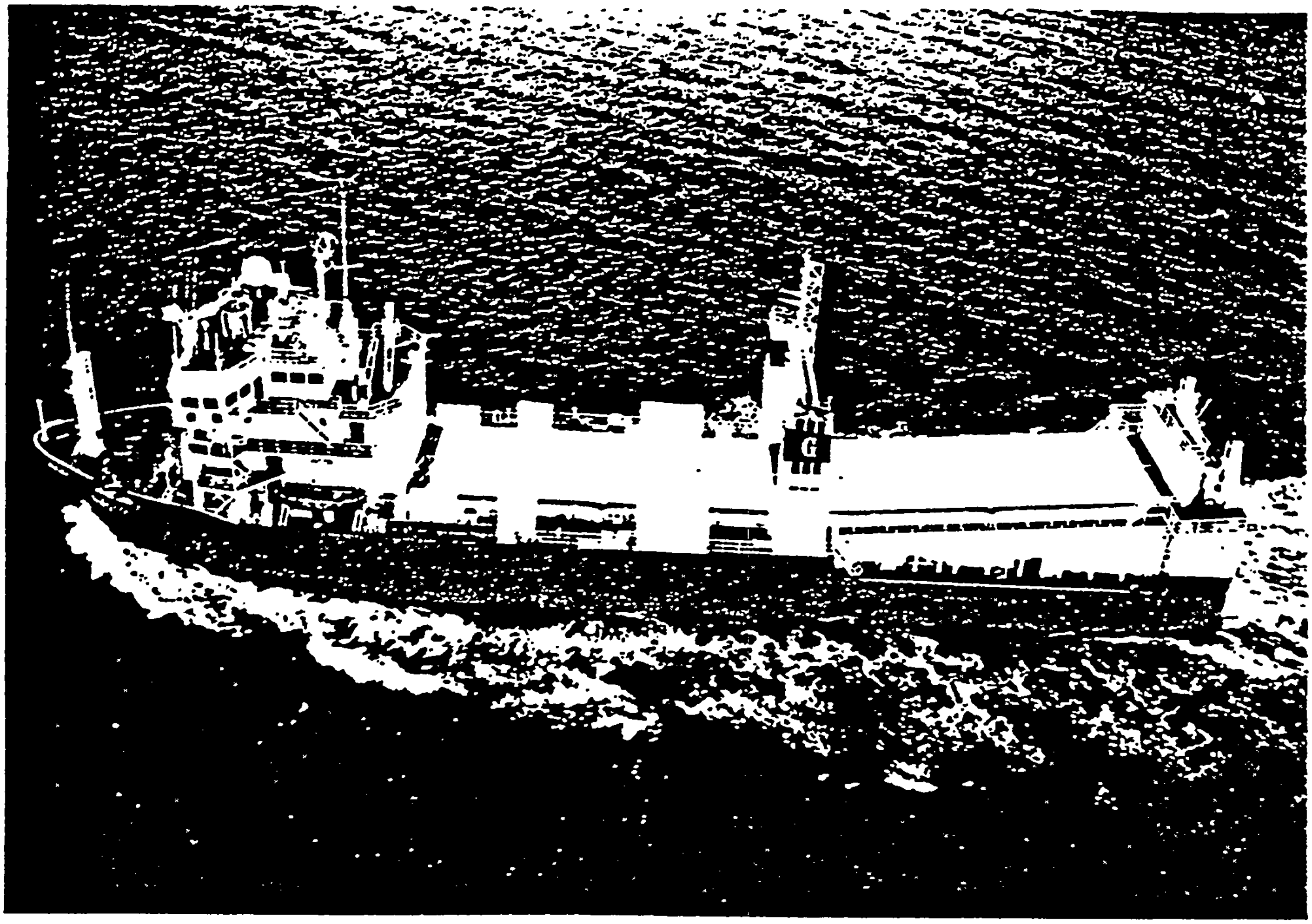

Figure 6.1.33. Transport Ship

incorporated into the ship to protect against radiation and fire as well as systems to aid in the search for and salvage of the ship in case of accidents.

At the CLAB and BS facilities, the casks will be handled in the various stages of loading or unloading by transport wagons travelling on rails at the bottom of the pools and/or overhead cranes.

The concrete castings and molds for the decommissioning wastes and core components will be transported from the encapsulation station to the final repository (SFL 3-5) using truck or rail transport.

The handling of the canisters and concrete molds underground is described in Section 6.1.5.4.

6.1.5.3 Waste Package

Enclosure of high-level waste, whether it is vitrified high-level waste from reprocessing or spent reactor fuel, in a canister is required for two main reasons. The waste must be packaged in an impervious container for handling to prevent radioactive release and, after emplacement, it is desirable to prevent the release of radioactivity into the geologic environment for a certain period of time. Also, the Swedish nuclear waste disposal strategy relies substantially on the presence of engineered barriers around the waste as a means of isolation. These barriers consist of the waste form itself, the canister, and the buffer materials around the canister. All barriers will be designed to isolate the fuel completely from the environment for a long time and to retard and dilute the radioactive substances that can eventually leak out from the repository over an even longer period of time. A canister of copper with a wall thickness of several centimeters is expected to remain leakproof and intact for at least a million years. The bentonite clay in the storage holes comprises a sealing layer as well as a mechanical and chemical buffer between the canisters and the rock mass. Geological investigations have shown that bentonite is a durable natural product and retains its properties for at least some millions of years, provided that the temperature does not exceed 100°C (212°F). Also, at this time, it is easier to predict the radionuclide containment process by engineered barriers of known materials and properties than that when surrounded by only the rock mass of uncertain containment properties. A lot of emphasis is thus placed on the investigation and development of canisters and other barriers that can contain the wastes for a time long enough for the radioactivity to reduce to harmless levels before being released into the accessible environment. The geochemical environment, however, can in no way be ignored since it must always be assumed that an early breach of the engineered barriers occurs and a premature release of radionuclides occurs. The only barrier to the spread of radionuclides in such a case will be the host rock mass and its geochemical properties must be such that there exists reasonable assurance of the retardation of the radionuclide flow to the accessible environment to such an extent that the radioactivity has been reduced to harmless levels.

In the final repository, canister damage will determined by the chemical environment prevailing in the emplacement hole. The composition of the ground water is of primary importance in this respect. Since very high demands will be made on canister life, it will be desirable to use a material for the canister that is thermodynamically stable in the environment expected. The main canister material under consideration in the Swedish program is copper and extensive research programs are being conducted to investigate its properties and behavior in repository conditions. Copper is one of the least reactive of common engineering materials. It is thermodynamically stable in pure water. This means that the corrosion of copper will be determined by the supply of corrosive substances in the ground water. These substances are primarily dissolved oxygen and, for reducing groundwater conditions, dissolved sulphide. Unalloyed, oxygen-free copper has therefore been chosen as the primary canister material. However, continued research will also be carried out with other metallic materials such as stainless steel, steel, aluminum, titanium, or lead, and non-metallics such as ceramics and cement.

Encasing the spent fuel in the copper canisters will take place in the encapsulation facility located directly at the repository site. This allows the canisters to be taken directly underground for final emplacement after encapsulation. The various sections of the facility have been described in Section 6.1.3.2 and the following will discuss the actual encapsulation processes in more detail.

6.1.5.3.1 <u>Welded Copper Canisters</u>. The canisters used for the welded process will be forged from pure copper ingots and machined to the final configuration shown in Figure 6.1.34. The open end of the canister will be prepared to accommodate a cover.

Unloading of the fuel from the transport cask in the receiving section prior to encapsulation was described in Section 6.1.3.2 above. The rack with the fuel assemblies will enter the encapsulation section from the receiving section via an "air lock" pool. The fuel rack will be allowed to dry and placed into a copper canister on a transport wagon which then takes the assembly into the furnace (Figure 6.1.35). Due to the difference in the size and radioactivity between boiling water reactor and pressurized water reactor fuel elements, the fuel assembly content of the canisters may vary. Figure 6.1.36 shows an example of the fuel arrangement in a canister. In the furnace, the canister will be placed inside a vacuum hood, evacuated or filled with nitrogen, and then heated to about 380-400°C (572-752°F). Molten lead will be pumped in and the solidification process controlled to ensure that shrinkage leaves no voids in the lead mass. In the cooling cell, the canister will be allowed to cool for about four days. After cooling, the top of the canister will be machined to allow fitting of the lid. The lid will then be welded onto the canister using a fully automatic electron beam welding process. The canister will be returned to the machining cell for weld quality testing using ultrasonics and final machining. If the weld is accepted, the canister will be given a final inspection, cleaned and either placed in temporary storage or transferred directly to the waste handling cage in the shaft to be taken underground for

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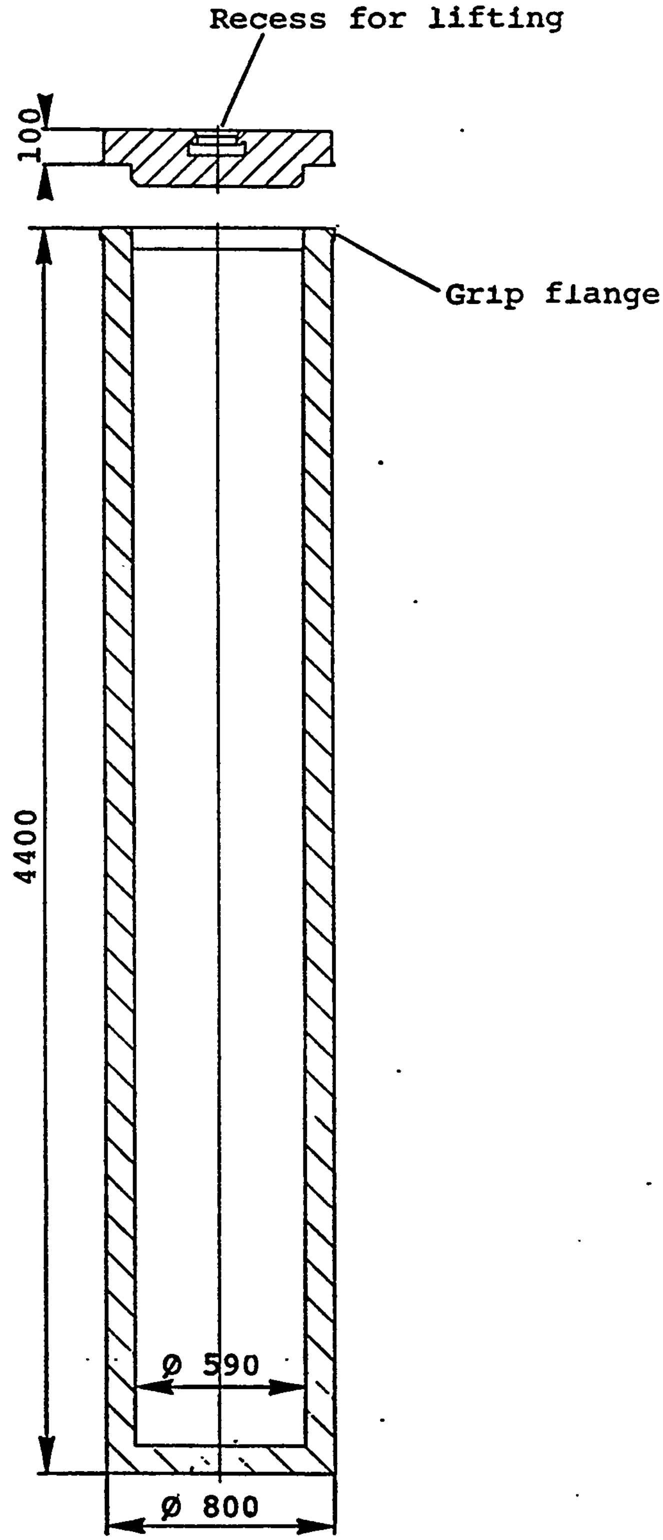

Figure 6.1.34. Welded Copper Canister

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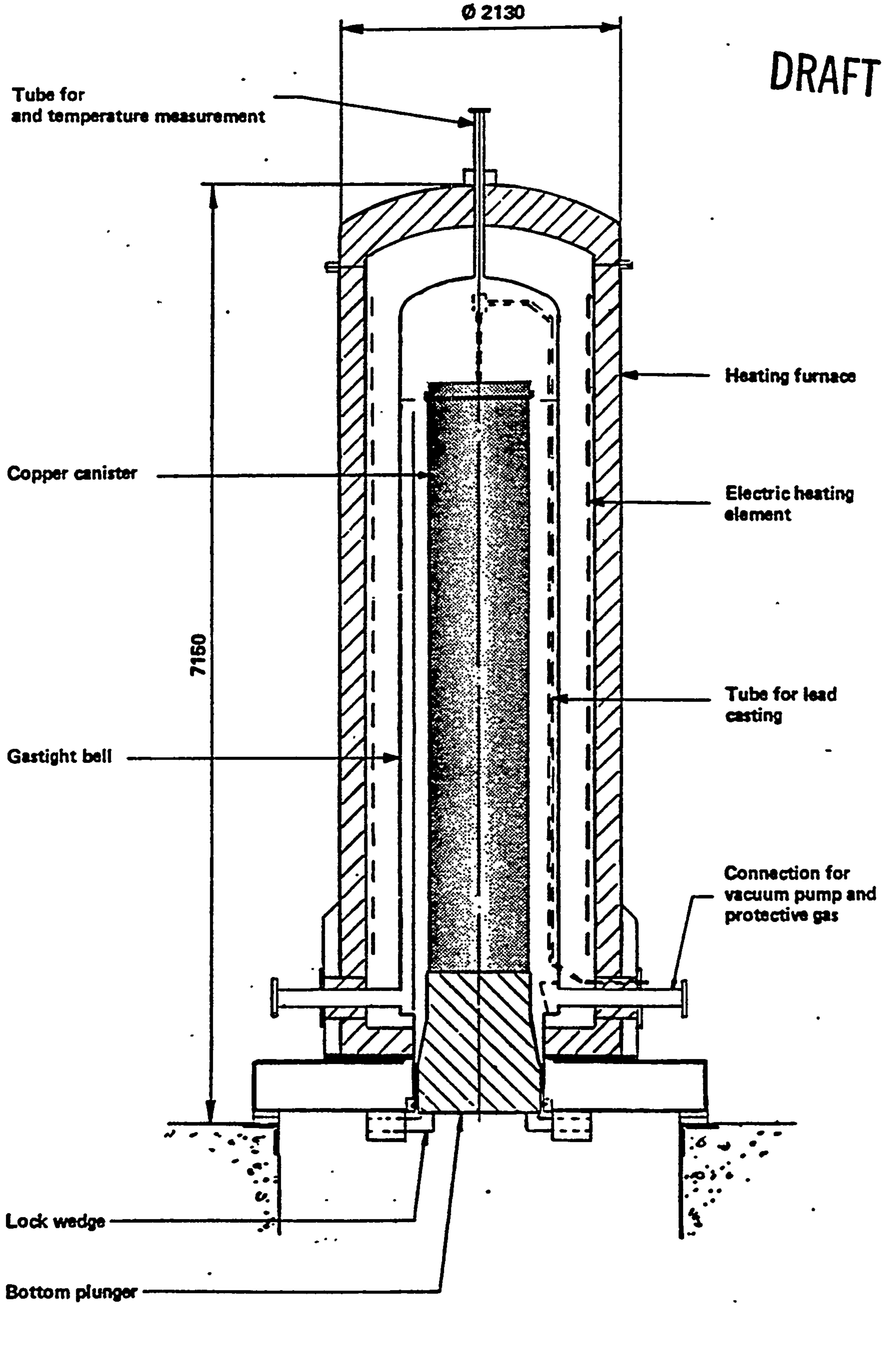

Figure 6.1.35. Furnace for Lead Casting

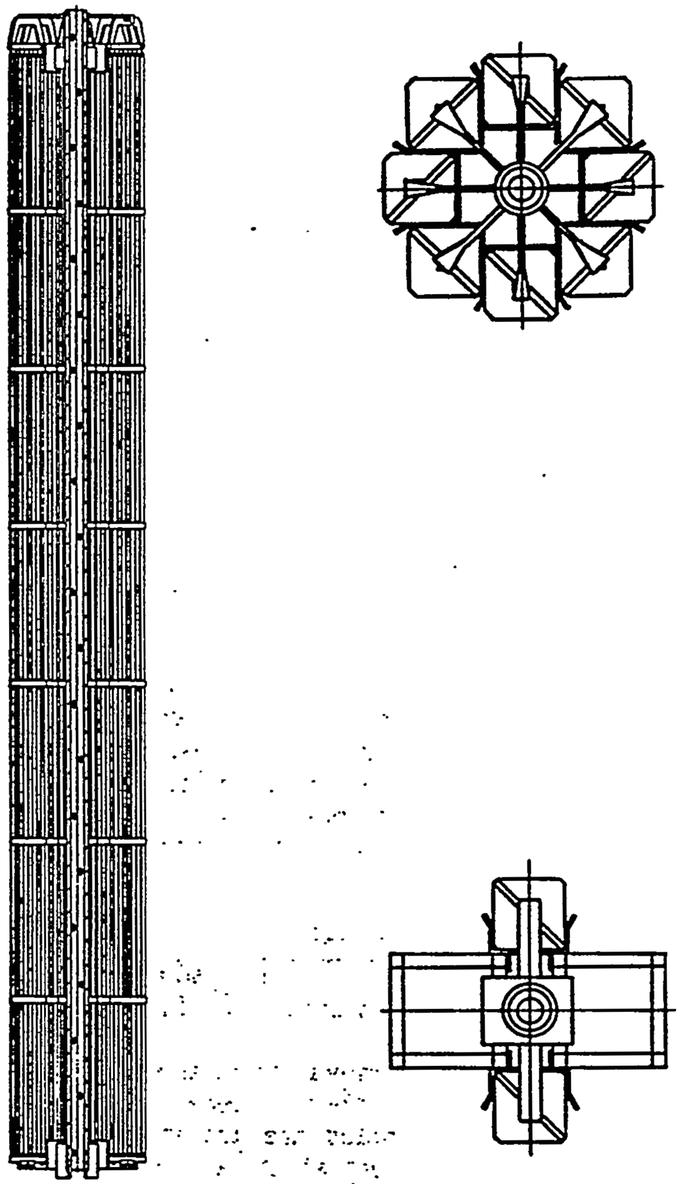

Figure 6.1.36. Fuel Assembly Arrangement

final deposition. If the weld is unacceptable, the lid will be removed and a new one welded on. A finished welded canister is shown in Figure 6.1.17.

6.1.5.3.2 Pressed Copper Canister. A possible design for the copper canister is shown in Figure 6.1.37. This will consist of a cast and forged shell which will be machined to the correct shape. The canister will be sealed with a main cover of about 200 mm (8 in) thickness plus inner and outer seal covers which are welded on during the encapsulation process to allow creation of a vacuum inside the canister. The finished canister will form a solid copper shell with a thickness of at least 10 cm (4 in) covering all parts of the contents.

The handling sequence up to the start of the encapsulation process will be the same as for the welded process. The drying process and the operations up to the actual compressing of the canister are carried out in cells or rooms filled with an inert gas, preferably nitrogen. This is to avoid oxidation of the copper powder and thereby reduce the need for hydrogen conditioning of the powder.

In the drying cell, a fuel rack will be dried thoroughly and placed into a canister. As with the welded canister, the fuel content of the canister will vary depending on the combination of boiling water reactor and pressurized water reactor fuel elements that will be used. The canister will then be filled with copper powder and the inner seal cover welded on to form an air-tight seal. The canister will be heated to about 350°C (660°F) and, after evacuation, filled with hydrogen. The hydrogen will be replaced several times until all the oxides in the copper powder and canister have been removed, after which the canister will be evacuated, the main cover fitted and the outer cover plate welded on. The joint surface between the main cover and the canister will then be treated with hydrogen in the same manner as the copper powder in the canister and evacuated. The canister will be transferred into the press for hot isostatic pressing (HIP). The press will consist of a furnace inside a pressure vessel with water-cooled walls. A suitable pressurizing gas, usually argon, will be pumped into the pressure vessel while the furnace is heated to the operating temperature. The canister will thus be isostatically compressed at a pressure of about 150 MPa (22,000 psi) for several hours at a temperature of about 500°C (930°F). During this process, the copper powder will be compressed into a solid mass and the cover is joined to the canister. The finished canister will be inspected for defects using ultrasonic testing. Acceptable canisters will be transferred into temporary storage or taken underground to the repository. Figure 6.1.17 shows a pressed copper canister. Defective canisters will be returned for re-evacuation and resealing.

The hot isostatic pressing process is based on extensive experience in the pressing of other materials and full-scale tests conducted with copper canisters with simulated fuel.

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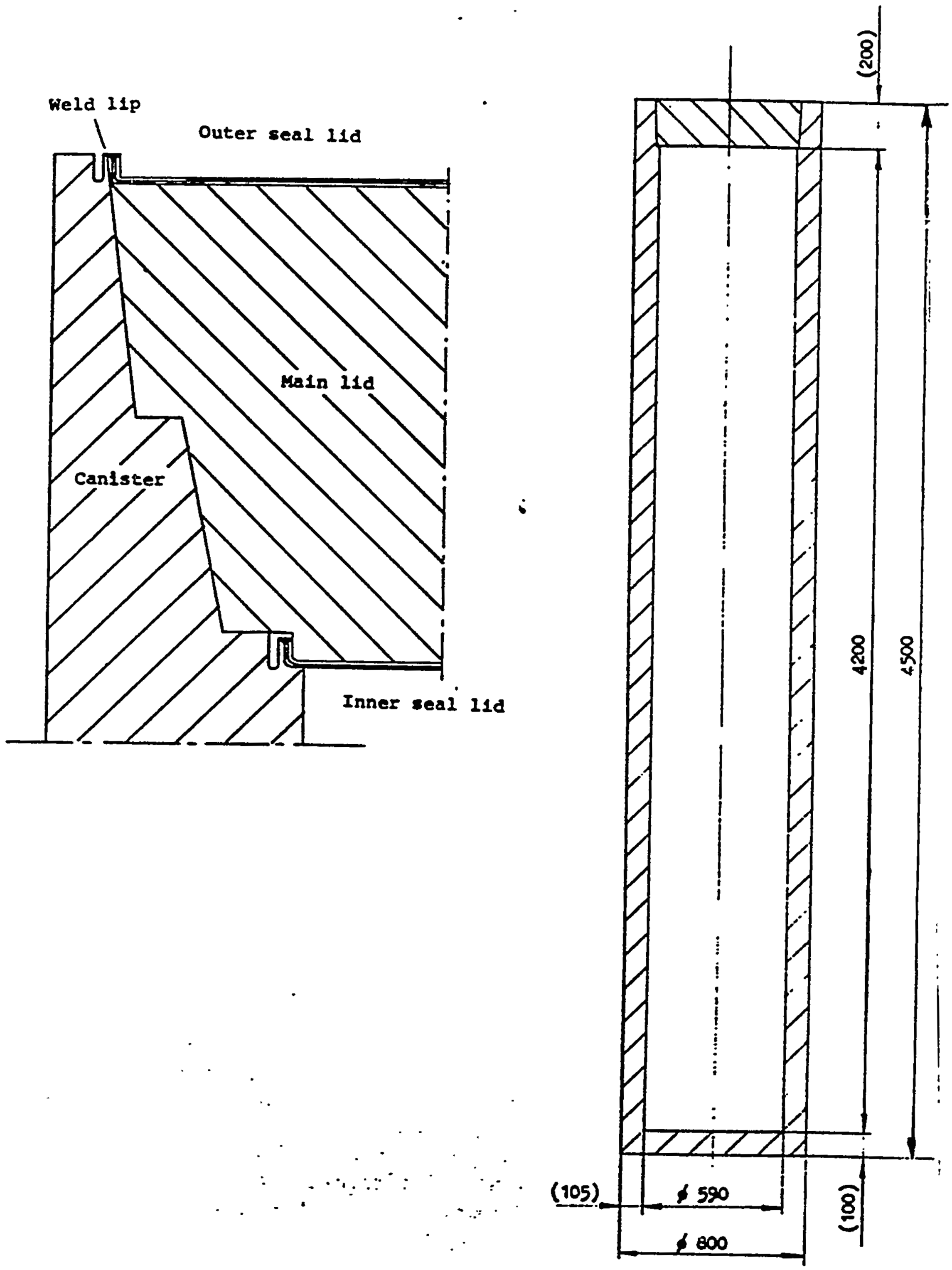

Figure 6.1.37. Canister for Isostatic Pressing

6.1.5.3.3 Encapsulation of Other Wastes. As described previously, the boiling water reactor fuel will be removed from the fuel boxes and the boron glass rod bundles will be removed from the pressurized water reactor fuel. These fuel boxes and glass rods will also require encapsulation and underground disposal.

The boxes and glass rod bundles will be embedded in concrete molds in the concrete encapsulation section (Figure 6.1.18). The molds will be filled with concrete, a lid fitted on the cask and the edge welded to the cask. The gap between the top of the concrete and bottom of the lid will then be backfilled by injection of grout. The thickness of the mold will be at least 100 mm (4 in) to provide sufficient radiation protection against 40 year old fuel boxes. A total of 49 fuel boxes in a 7x7 pattern may be placed in one mold. The boron glass rod bundles, which have higher radiation, will be placed in molds filled with fuel boxes where the central nine slots have been left empty to leave room for four rod bundles. In the filled mold, the rod bundles will therefore be surrounded by an extra thickness of concrete.

Operating wastes and final decommissioning wastes from CLAB and BS will also be encased in concrete casting for final disposal.

The completed molds are transported to the SFL 3-5 underground repositories for emplacement.

6.1.5.4 Underground Waste Emplacement Sequence

This section describes the proposed handling methods at the SFL-2 repository.

A copper canister ready for emplacement will be taken to the waste shaft for transfer underground. The transport wagon and copper canister will be secured in a specially designed cage during lowering to the repository level. The shaft and hoisting system has been described previously in Section 6.1.3.3.

At the repository level, the wagon will travel off the cage and the canister will be collected by a specially designed deposition vehicle (Figure 6.1.38). The deposition vehicle will be a rubber-tired truck with articulated construction for maneuverability. The operator will be enclosed in a radiation shielded cab from where he will control the vehicle while travelling and during canister emplacement. The canister will be manipulated during emplacement using several handling devices at the back of the vehicle:

1. A radiation shield drum will surround the canister during transport and deposition to provide radiation shielding. The base of the drum will be covered during travel and during deposition, the cover will be removed to allow lowering of the canister within into the hole.

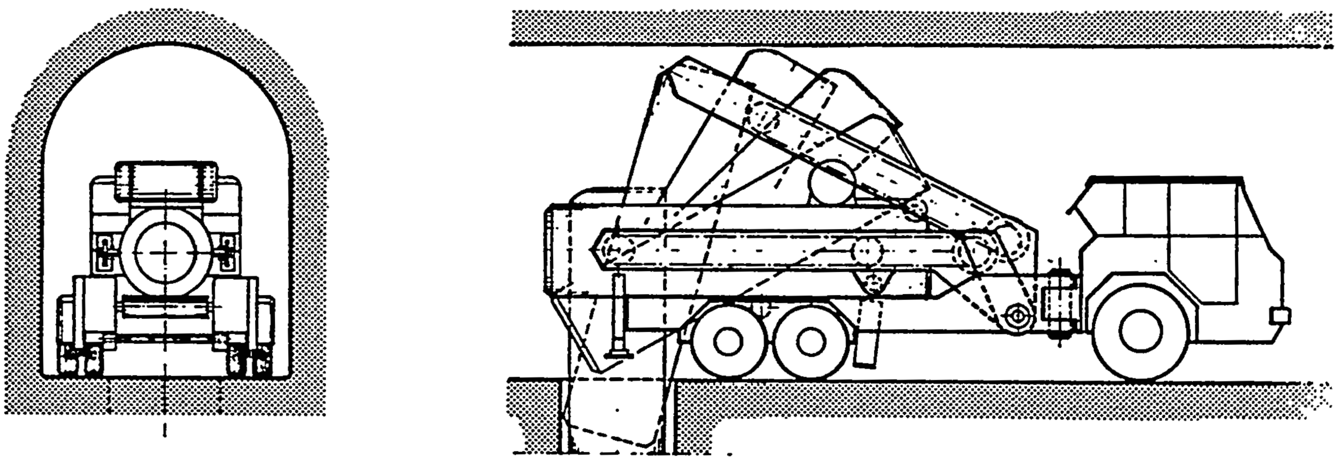

Figure 6.1.38. Deposition Vehicle

2. A hydraulic boom manipulates the drum into the required positions during the emplacement. During travel, the drum will lie horizontal on the truck. For emplacement, the drum will be raised into a vertical position over the emplacement hole so the canister can be lowered through the open base of the drum.

3. Hydraulic outriggers will support the rear of the truck during lowering of the drum into the emplacement hole. A centralizer, incorporated into the structure, will allow horizontal adjustments of the drum position over the canister during collection or in the hole during deposition.

The cage will return to the surface when the deposition vehicle has loaded the canister. For safety reasons, a canister is only lowered to the repository when the emplacement hole is ready and lined with bentonite and the deposition vehicle is at the shaft ready to receive a canister to avoid extended canister standing times underground.

Figure 6.1.39 shows the sequence of emplacement for the canister. First, the emplacement hole will be lined with bentonite blocks which are centered using a steel pipe simulating a canister. The edge of the top bentonite block will have steel edge reinforcement to guide the canister and prevent damage to the block. When the deposition vehicle arrives at the hole, the operator will align the vehicle roughly over the hole before lowering the outriggers onto the concrete collar. The drum with the canister will be lowered into the hole using the boom and the centralizer will adjust the drum to its final position. The canister will then be lowered inside the drum and into the bentonite-lined borehole. After removing the drum, the deposition vehicle will return to the shaft. Another vehicle will be used to place additional bentonite into the hole to cover the canister and allow unrestricted access to the remainder of the tunnel. The top of the borehole is finally capped with a watertight seal to prevent inflow of water from the tunnel. This seal remains in place until the tunnel is ready for backfilling.

6.1.5.5 Emplacement Schedule

Figure 6.1.2 shows the overall schedule for the Swedish nuclear waste program. Emplacement is scheduled to start around 2020 and last for about 30 years until 2051 at which point the repository will be sealed and decommissioned. The total number of canisters to the emplaced in the SFL-2 repository is about 5,600. The expected emplacement rate will be about 210 canisters per year, based on the output capacity of the encapsulation facility which averages out at roughly one canister per day for 10 months.

6.1.6 Buffer and Sealing Materials

This section describes the materials and techniques used to contain and isolate the canisters both in the emplacement hole as well as the repository as a whole.

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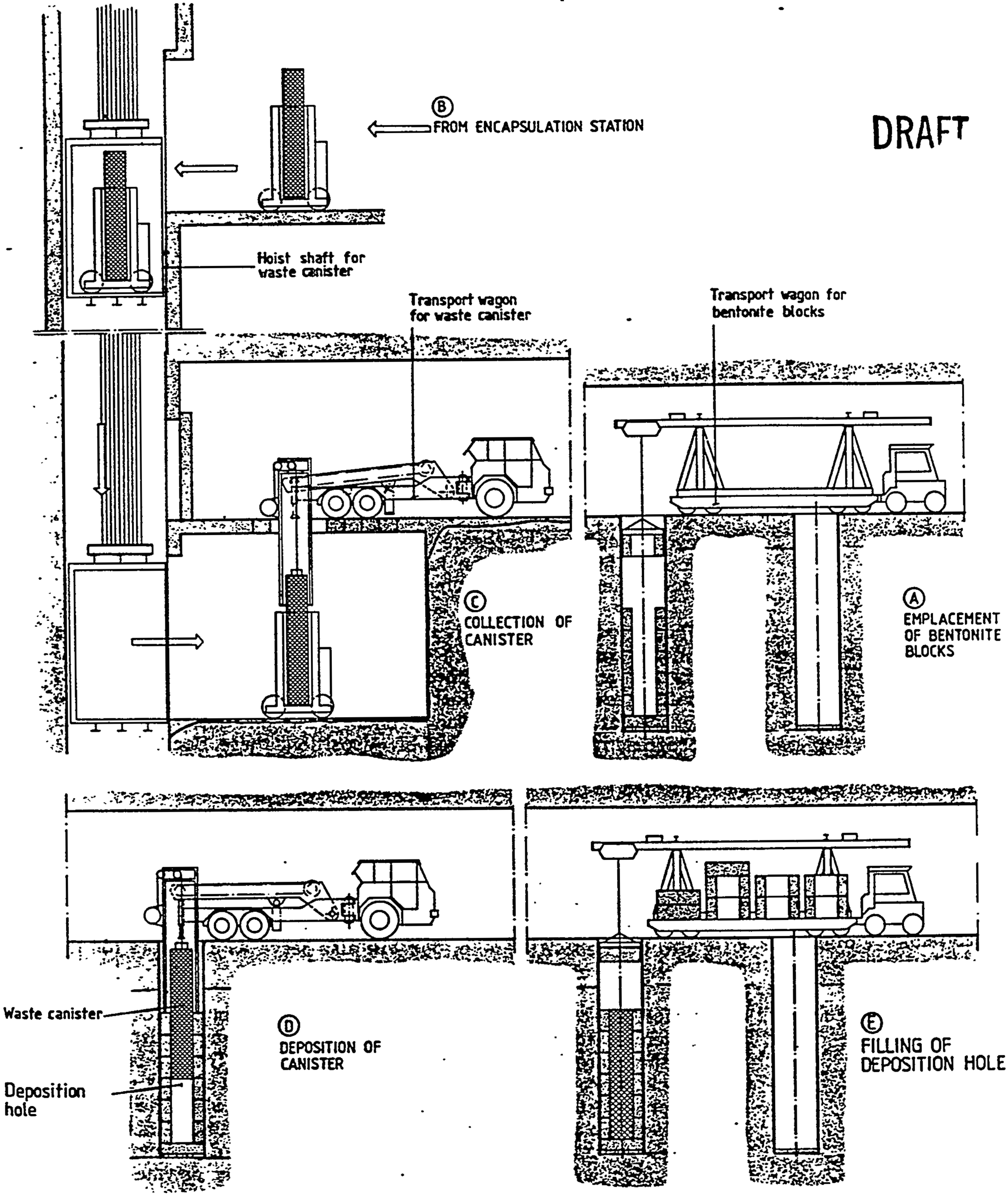

Figure 6.1.39. Emplacement Sequence

6.1.6.1 Buffer Materials

The function of the material around the canister in the emplacement hole (Figure 6.1.40), called buffer material, is to present a mechanical and chemical barrier of protection around the canister and to limit the inward transport of corrosive substances from the ground water to the canister surface. Also, in the later stages when the canister has been breached, the buffer material serves to limit the outward travel of leached radioactive substances from the canister into the surrounding host environment. The buffer material under investigation for the Swedish waste disposal program is pure, high density bentonite, which has the following desirable properties:

- Very low hydraulic conductivity and diffusivity
- Sufficient bearing capacity to hold the canister in the emplacement hole
- Sufficient ductility to compensate for possible ground movements
- Swelling capability when in contact with water prevents creation of water-bearing passages within the material and voids in walls of storage hole are filled
- Good thermal conductivity to prevent overheating of the canister or buffer material
- Long-term chemical and physical stability
- Good ion exchange capacity to retard any positive radionuclides escaping from the canister
- A pH value of about 8-9 in the saturated stage is desirable so corrosion rate of copper is not significantly increased.

Bentonite is the name for natural clays formed from volcanic ashes. These clays are rich in swelling clay minerals, smectities, of which montmorillonite is a common variety. A bentonite named Volclay MX-80, mined in Wyoming and South Dakota, has been chosen as a reference material. MX-80 is a so-called sodium bentonite consisting of about 90% montmorillonite, with sodium as the dominant adsorbed ion. This ion coating ensures the best sealing and swelling properties. The main chemical constituents of Volclay MX-80 are silicon and aluminum with smaller amounts of sodium, magnesium, and iron as well as calcium and potassium. The granulate has a grain size of about 0.07-0.8 mm (0.003-0.3 in).

The bentonite can be isostatically compressed into blocks and then cut to fit into the emplacement holes. The gap that will exist between the blocks and the rock will be filled with bentonite powder (Figure 6.1.40). Water absorption in a hole will take place slowly over a period of years and saturation of the bentonite can be delayed until after tunnel backfilling by preventing inflow of water from the tunnel floor. A raised, concrete collar

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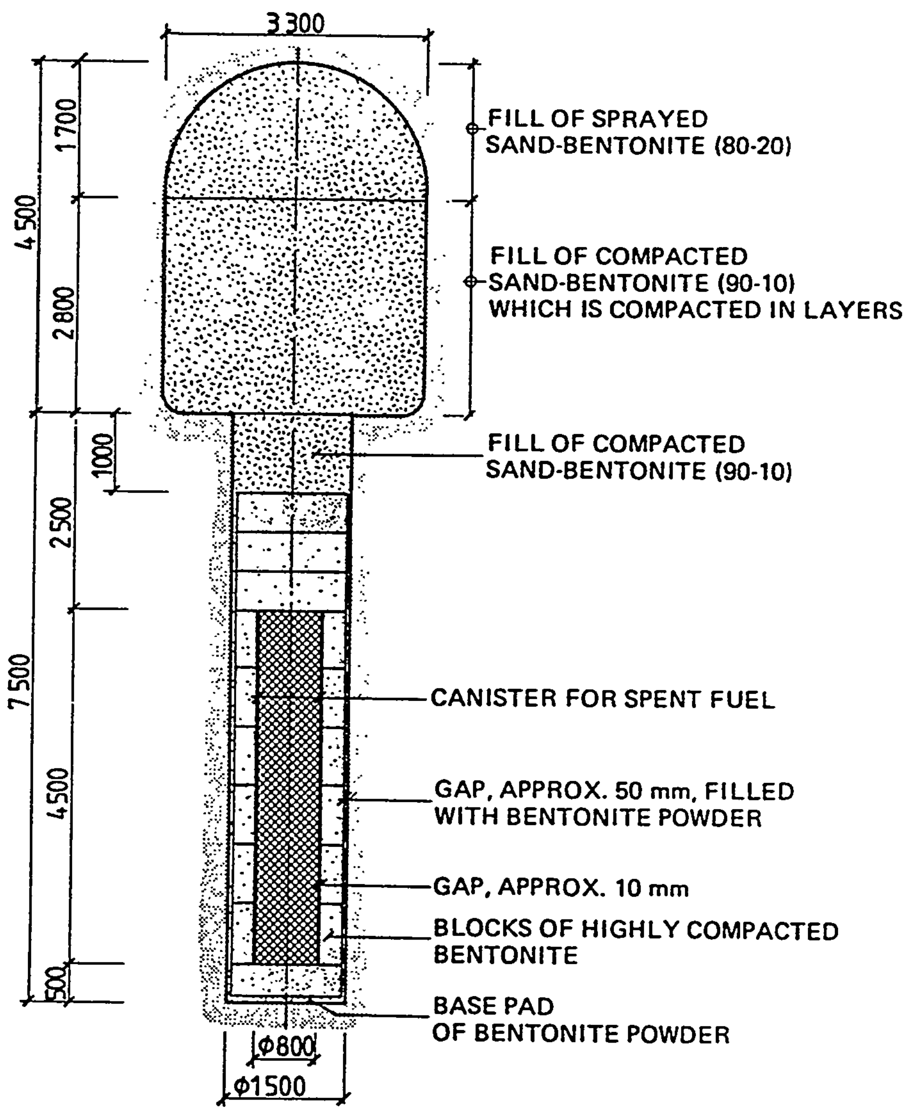

Figure 6.1.4Ø. Emplacement Hole

will poured before drilling the hole (see Section 6.1.4.2 above), which will be covered once the canister and bentonite have been inserted. The collar will be removed just before the tunnels are backfilled with a sand/bentonite mixture.

When the repository has been sealed, the original ground water conditions are gradually restored and the swelling pressure of the bentonite will force it into fractures and seal them. Experiments at the Stripa facility to date have shown that bentonite will be an appropriate material to use as the buffer material and that its important functional characteristics such as hydraulic and thermal conductivities, specific gravity, etc., are not altered by the absorption of ground water such that it would be unsuitable to function as buffer material.

6.1.6.2 Backfill Material.

The deposition tunnels, repository service tunnels, shafts, boreholes, and fracture zones encountered will be backfilled and sealed using bentonite-based materials. The primary function of this material will be to provide a durable support to the rock and restore the hydraulic conditions in the area to as near to their original condition as possible. The backfill will therefore have to have at least as low a hydraulic conductivity as the rock is replaces. The backfill will thus be a mixture of bentonite powder (10-20%) and sand of a suitable grade.

Backfilling in the tunnels will start after the emplacement holes in 8-10 tunnels have been filled with canisters (Figures 6.1.40 and 6.1.41). The hole cover and concrete collar will be removed from each hole and the backfill in the lower part of the tunnel placed in several passes. The backfill will be compacted after each pass to ensure that the mixture has the right density. Near the roof, the tunnel will be filled by spraying the backfill material using shotcreting machines. Tests at Stripa have shown that with the compaction method, a backfill density from 1.8-2.2 t/m^3 (113-138 lb/ft^3) at a water content of 8-13% can be achieved. Using the spraying method, densities in the region of 1.1-1.8 t/m^3 (69-113 lb/ft^3) at a water content of 11-22% can be attained. The hydraulic conductivity of the bentonite-sand mixture amounted to a maximum of 10^{-9} m/s.

Until the main access tunnels are backfilled during repository sealing, steel bulkheads will be used to provisionally seal the emplacement tunnel entrances (Figure 6.1.41). As the bentonite buffer material in the emplacement holes swell, it will expand into the tunnel backfill to a small degree and the provisional bulkheads will therefore yield to prevent excessive pressure buildup.

The repository emplacement tunnels will be placed in rock masses with as few fractures as possible, but at some point a tunnel will inevitably intersect a zone of high ground-water flow. Such sections, although not used for waste emplacement, must be isolated using a method similar to that shown in Figure 6.1.42, where the suspect zone is isolated by highly-compacted bentonite plugs or seals. The shafts will be backfilled with the same bentonite/sand mixture as the tunnels, except that the

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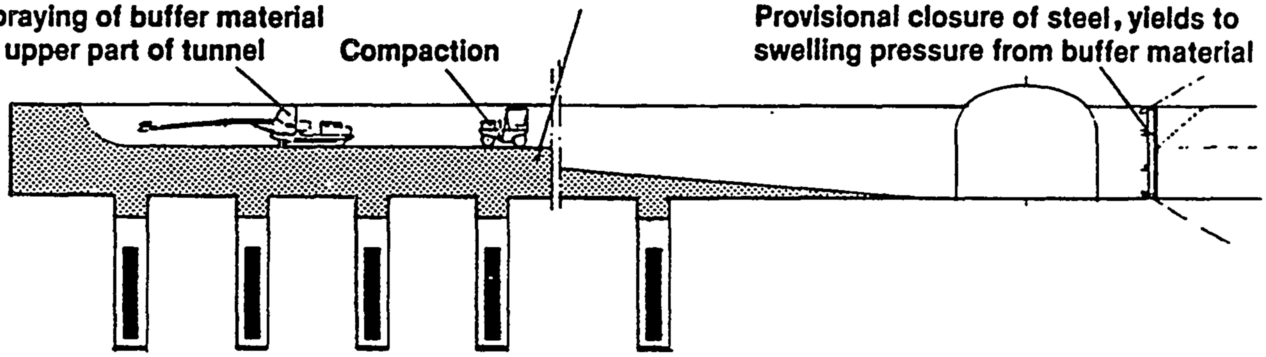

Figure 6.1.41. Tunnel Backfilling

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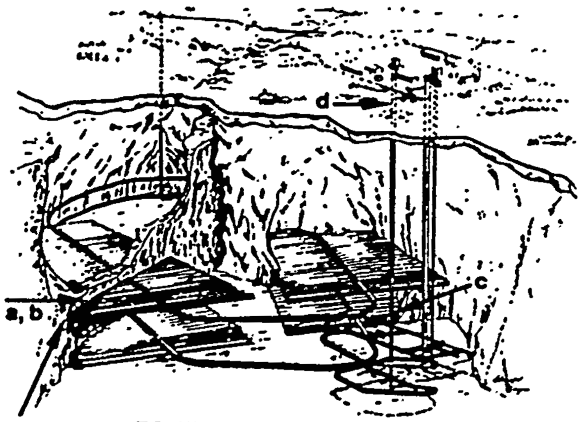

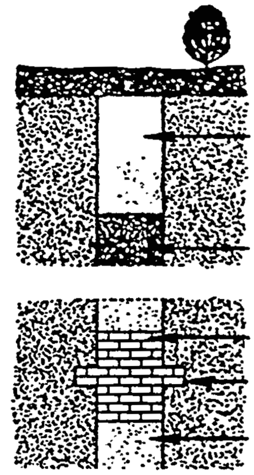

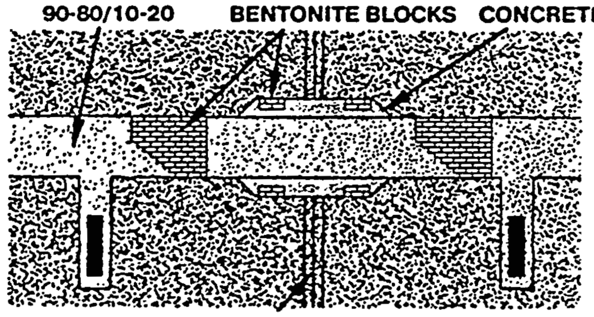

a) Isolation of fracture zone with high hydraulic conductivity from tunnel in final repository. The tunnel permits transports during the deposition period and is backfilled after deposition has been completed in the tunnel.

b) Backfilling with plugs of highly-compacted bentonite blocks that are stacked in connection with layer-by-layer compaction and spraying of sand/bentonite mixture.

c) Backfilling of shafts with plugs of highly-compacted bentonite blocks that are stacked in connection with layer-by-layer compaction of sand/bentonite mixture.

d) Backfilling of upper part of shaft with layer-by-layer compaction of moraine and with plug of concrete.

Figure 6.1.42. Repository Sealing

upper section will be filled with concrete and moraine. At certain locations, the shaft will be overexcavated using non-blasting techniques to reach the zone of undisturbed ground around the shaft that was not affected by sinking operations. These sections will be filled with highly-compacted bentonite rather than the bentonite/sand mixture and the bentonite acts as a tight shaft seal as it swells.

Bentonite will also be used to seal the boreholes that will be drilled as part of the various underground investigation programs. Perforated tubes filled with bentonite pellets will be inserted into the borehole and, on absorbing water, the swelling bentonite will fill the borehole (Figure 6.1.43). In boreholes penetrating highly fractured zones with the risk of material loss, pellets of sand/bentonite or magnesium oxide may be used. Material loss will thus be prevented by the coarser sand grains blocking fracture openings or the magnesium oxide swelling to form a tightly sealing stable plug of $Mg(OH)^2$.

6.1.7 Summary

The following presents a brief summary of the SFL-2 repository concept discussed in the preceding sections. The main items of discussion are some of the major advantages and drawbacks of the concept.

In summary, the basic assumptions for the repository concept presented in this chapter were as follows:

- Spent fuel assemblies will be stored for about 40 years before disposal
- Total inventory of spent fuel (pressurized water reactor and boiling water reactor) amounts to approximately 7,800 MT (8,580 tons)
- Number of canisters is approximately 5,600
- Repository to be located at approximately 500 m (1,640 ft) depth in crystalline rock
- Each canister will contain about 1.4 MT (1.5 tons) with an estimated heat output of 0.8 kW at time of emplacement
- Maximum temperature at the canisters is 80°C (176°F)
- Spent fuel will be encased in copper canisters designed for long-term containment of the radionuclides (approximately 1 million years)
- Disposal of canisters will be in vertical boreholes drilled into tunnel floors

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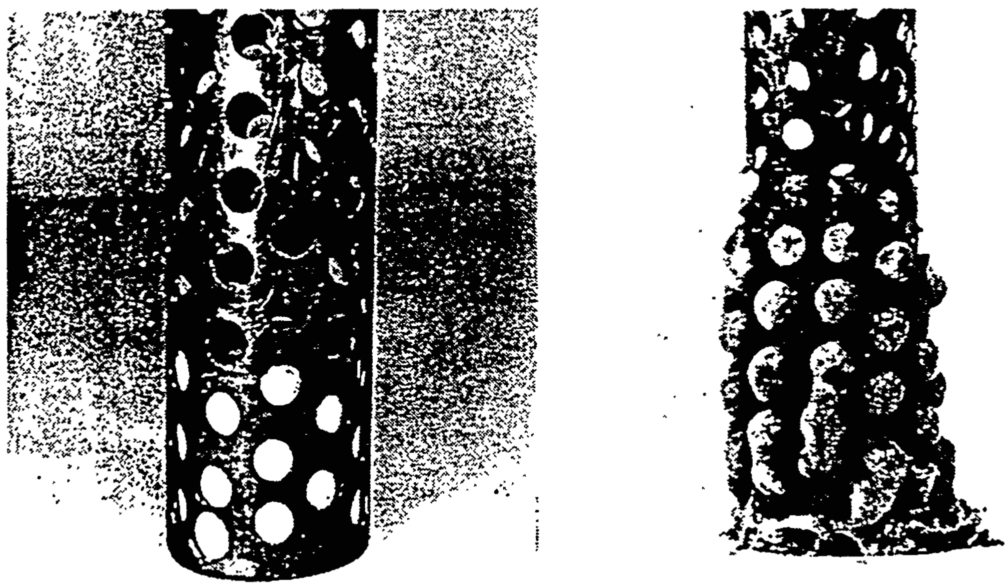

Figure 6.1.43. Borehole Sealing

- Borehole spacing will be about 6 m (20 ft) with a tunnel spacing of about 33 m (108 ft) in a two-level and 25 m (82 ft) in a single level repository
- Emplacement tunnel lengths will be about 255 m (830 ft)
- Bentonite backfill will be used in emplacement holes as buffer material.

6.1.7.1 Advantages

1. Based on site investigations both in Sweden and abroad, the crystalline host rocks can provide a suitable geologic setting for the long-term disposal of high-level wastes. The investigations have underlined the long-term structural integrity of the rock formations as well as that crystalline rock properties such as low permeabilities and good geotechnical characteristics promote the containment of radionuclides.

2. Sweden has extensive experience with major underground mine construction projects similar to the repository in crystalline rock formations. Also, from the shaft sinking and tunneling viewpoint, crystalline rocks are a favorable medium because excavations often require little or no rock support. This is very important since the shafts and access tunnels may have to remain open for as long as 50 years before they are sealed again.

3. The design allows the various repository operations such as site investigation, excavation, emplacement, backfilling, monitoring, etc., to be conducted concurrently. Flexibility also exist to adjust the layout of the repository to avoid unexpected disturbed ground conditions (Figure 6.1.44).

4. Most of the heavy transport between the reactors, the storage facility and the repository will be by sea, thereby avoiding the dangers associated with land transport.

5. Emplacement in vertical rather than horizontal boreholes has advantages from a waste package handling point of view.

6. Shaft drilling and tunnel boring technology may be used for construction of part of the shaft and tunnel network which has long-term structural stability advantages.

7. Retrieval of waste packages, if needed, before final sealing of the repository is technically feasible.

8. The use of the engineered barriers which includes the waste canister and bentonite buffer material provides the primary radionuclide containment in addition to that of the host rock. This is of particular advantage since homogeneity of the rock

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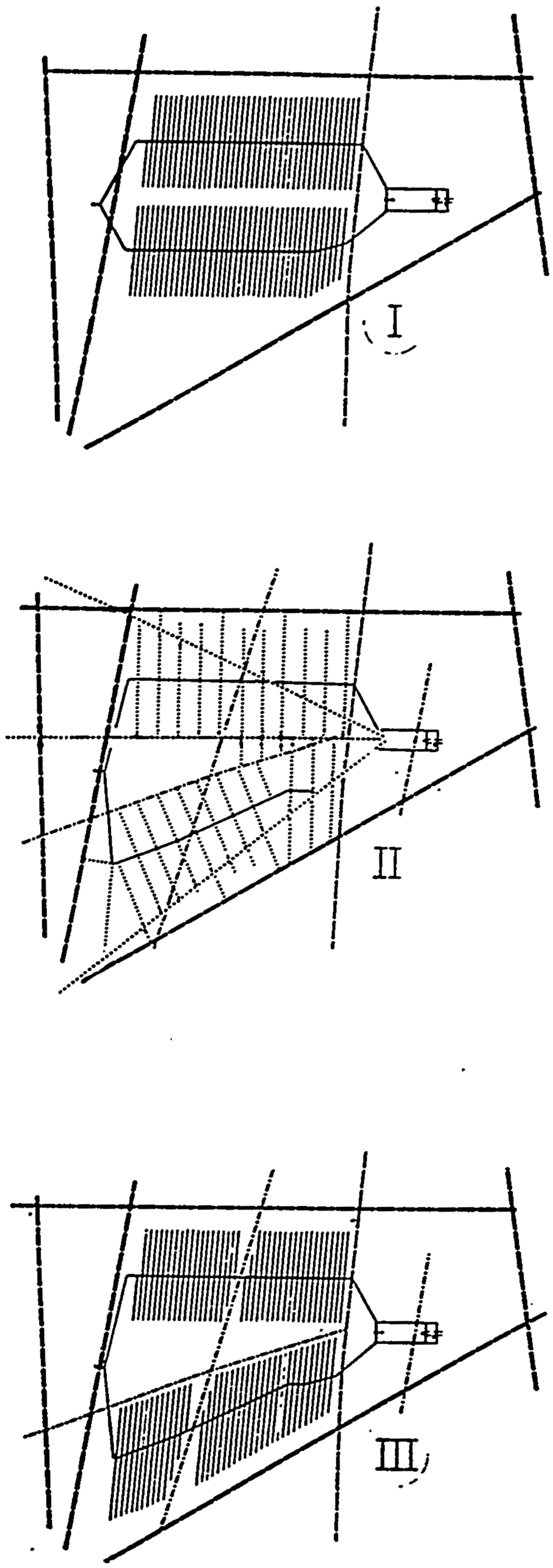

KEY TO SYMBOLS

Central tunnels with shafts

Storage tunnels

Zones with appreciably elevated permeability
Storage holes not closer than 100 m

Zones with moderately elevated permeability
Deposition holes not closer than 25 m

Zones found during construction of repository. Otherwise same as above

Horizontal sounding drillings during construction phase

FIGURES

(I) I Planned configuration on the basis of investigation results from the surface

(II) II Modification of central tunnels to allow for zones found during construction

(III) III Final configuration

Figure 6.1.44. Adjustment of Repository Layout

properties can never be guaranteed for a rock mass the size of a repository. The canister will protected from the corrosive ground waters by the saturated bentonite and the bentonite also retards radionuclide migrations once a canister has been breached. The canister material (copper) was chosen because of its longevity and resistance against corrosion. The life span of the canister is expected to be of the order of a million years by which time a lot of the radioactivity will have decayed to harmless levels.

9. The concept includes interim storage of the spent fuel for about 40 years and therefore, the heat production and radioactivity of the waste package will be lowered substantially. The design restricts the maximum temperature at the canisters surface to about 80°C (176°F). Less heat production from the waste will reduce ventilation requirements during operation.

10. Location of the encapsulation facility at the repository reduces the handling requirements between encapsulation and final deposition.

11. Utilization of the exploratory shafts in the final repository would present economic and technical advantages.

6.1.7.2 Potential Problems

One major drawback for all geologic high-level waste repository concepts throughout the world is that knowledge about the long-term properties of some of the components making up the repository is limited. This would include to a limited extent the engineered barriers and to a larger extent the geochemical, geothermal and hydrogeological characteristics of the repository environment. Extensive investigations of these aspects have been ongoing in many countries and will continue into the future in order to increase the understanding of the processes involved and reduce the uncertainties associated with the disposal of nuclear waste in deep geologic repositories.

6.1.7.3 Summary

Crystalline rock formations are being considered for the U.S. second high-level waste repository program and therefore the Swedish work is of great relevance. The Swedish concept of waste displacement in vertical boreholes drilled into the floors of emplacement drifts is also one of the concepts being considered for the U.S. program. Thus, the work being done in Sweden may have direct application in the design of the U.S. repository. The U.S. repository will be designed to accommodate about 70,000 MT (77,000 tons) or about eight times the amount of waste to be emplaced in the Swedish repository. In spite of the increased size, the concepts described above can be directly applied in siting characterization and construction of a U.S. repository.

6.1.8 The Low- and Intermediate-Level Concepts

This section gives a brief summary of the repository concepts for the low- and intermediate level wastes. These repositories are also to be located in crystalline rock.

6.1.8.1 The SFR-1 and 3 Repositories

The SFR-1 repository is currently under construction at the Forsmark power station. About 90,000 m^3 (118,000 yd^3) of short-lived low- and intermediate-level wastes from reactor operation and non-electricity producing activities, mainly from the Studvik research facility, are to be stored in this repository. Figures 6.1.45 and 6.1.46 show the completed facility. The surface facilities are located in the power station harbor area near the portals of the two access tunnels. The tunnels lead out under the Baltic Sea to the repository, located approximately 1 km (0.6 mi) offshore. The repository will be constructed about 60 m (200 ft) below the seabed in the crystalline bedrock.

The SFR-1 repository will be constructed in two phases (Figure 6.1.45). Phase 1 will consist of a concrete silo constructed inside a cylindrical rock cavern plus four 160 m (525 ft) long disposal galleries. The concrete silo will be filled with intermediate-level waste. Three galleries will be filled with low-level waste which can be handled by radiation-shielded equipment while the fourth gallery will contain intermediate-level waste requiring remote handling. Phase 2 will consist of an additional silo and one or two more disposal galleries.

Each silo will be a free-standing concrete structure erected inside a cylindrical rock cavern about 70 m (230 ft) high and 30 m (98 ft) in diameter. The silo will sit on a 1.5 m (5 ft) thick bed of compacted sand/bentonite and the 1 m (3 ft) gap between the silo and cavern wall filled with bentonite powder. The silo will thus be isolated from the surrounding rock by a buffer material, in a manner similar to the canisters in the SFL-2 repository described previously. Internally, the silo will be divided into 2.6 m (8.5 ft) square cells. This cell structure will act as internal reinforcing for the silo and will facilitate emplacement and grouting of the waste packages. Figure 6.1.47 shows the disposal operation. Containers with the waste packages will be transported to the receiving room adjacent to the silo. A tunnel above this room will connect it to the top of the silo. The waste packages will be taken from the container one at a time by a remote-controlled crane and transported to the top of the silo and lowered into one of the disposal cells. When two layers of packages have been placed in a cell, they will be covered with grout. After all cells are full and backfilled, a concrete cover will be poured over the silo and all associated excavations backfilled with sand/bentonite materials.

The gallery in which the intermediate-level wastes will be placed will also be backfilled with grout. The galleries with low-level waste are not expected to require grouting.

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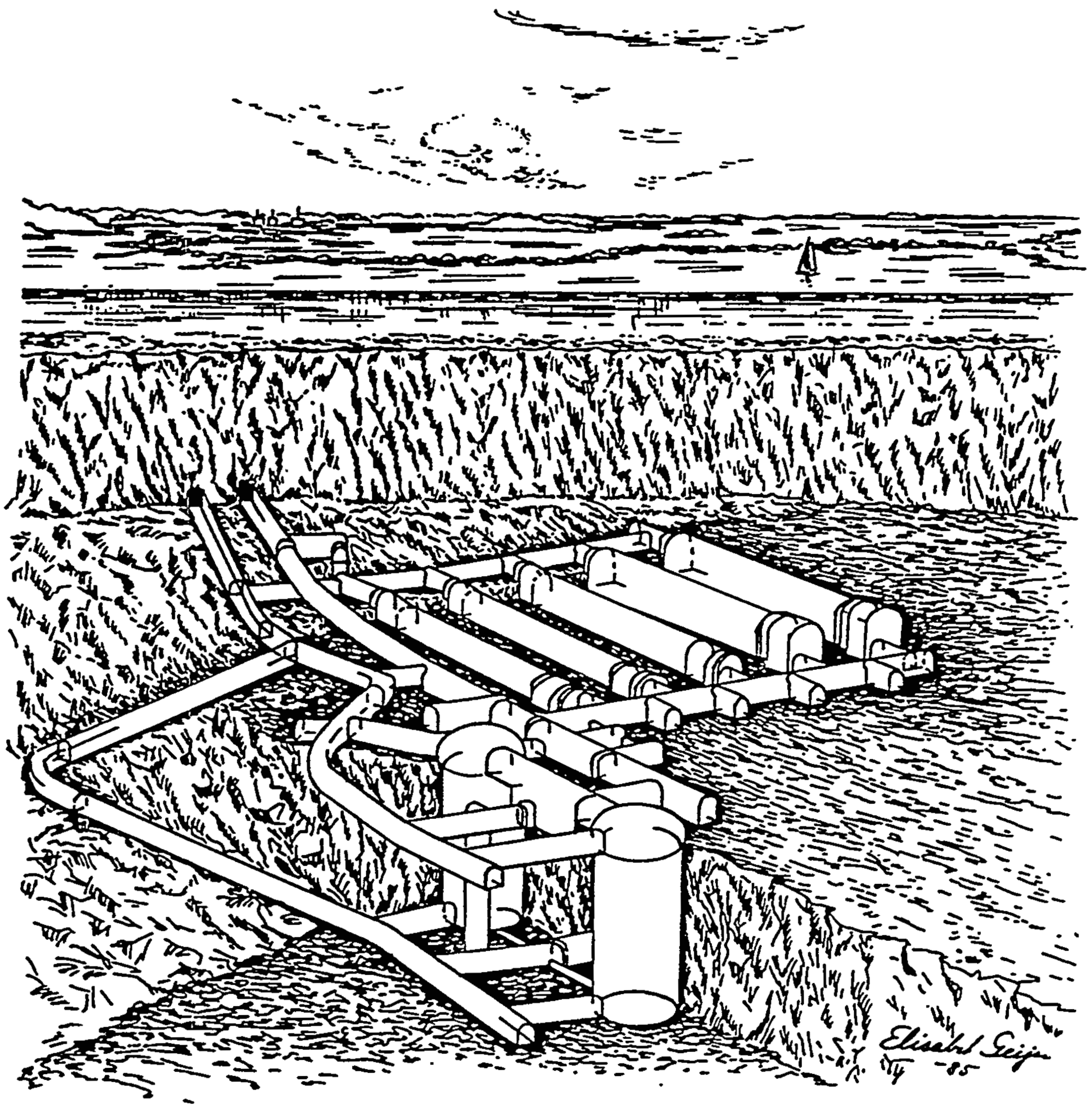

Figure 6.1.45. SFR-1 Repository

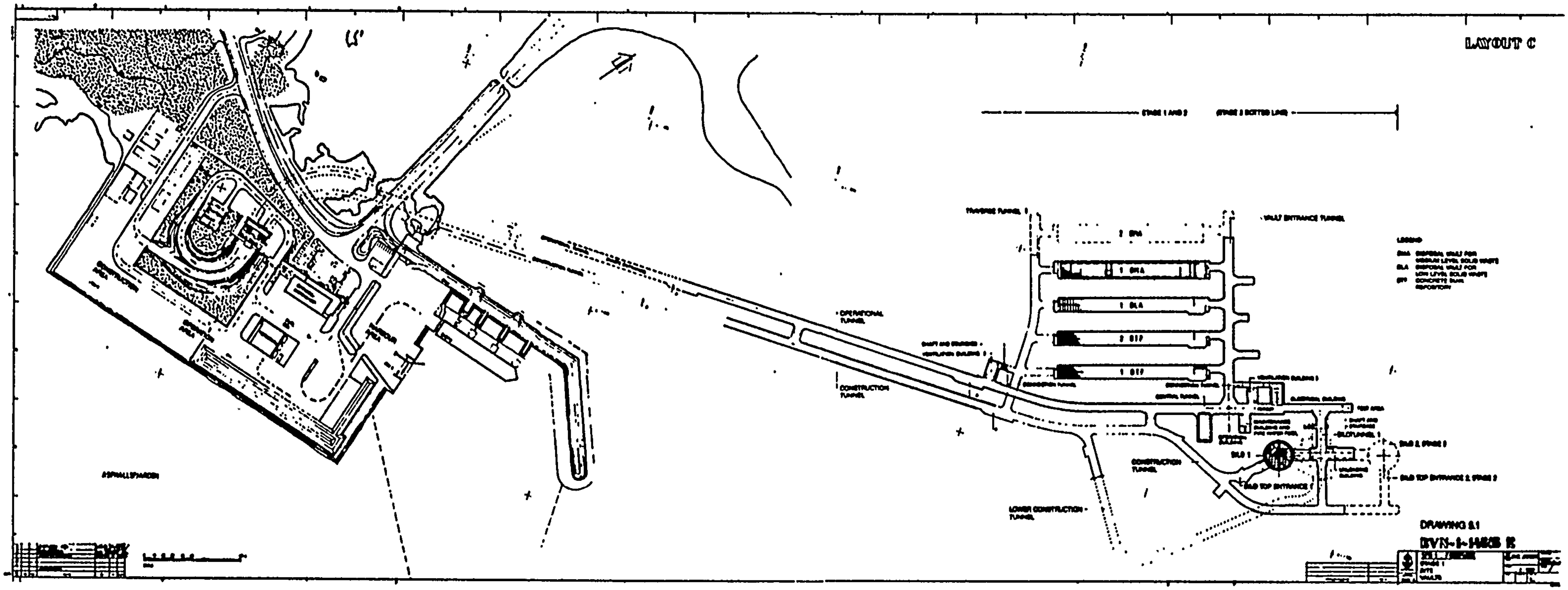

Figure 6.1.46. SFR-1 Repository

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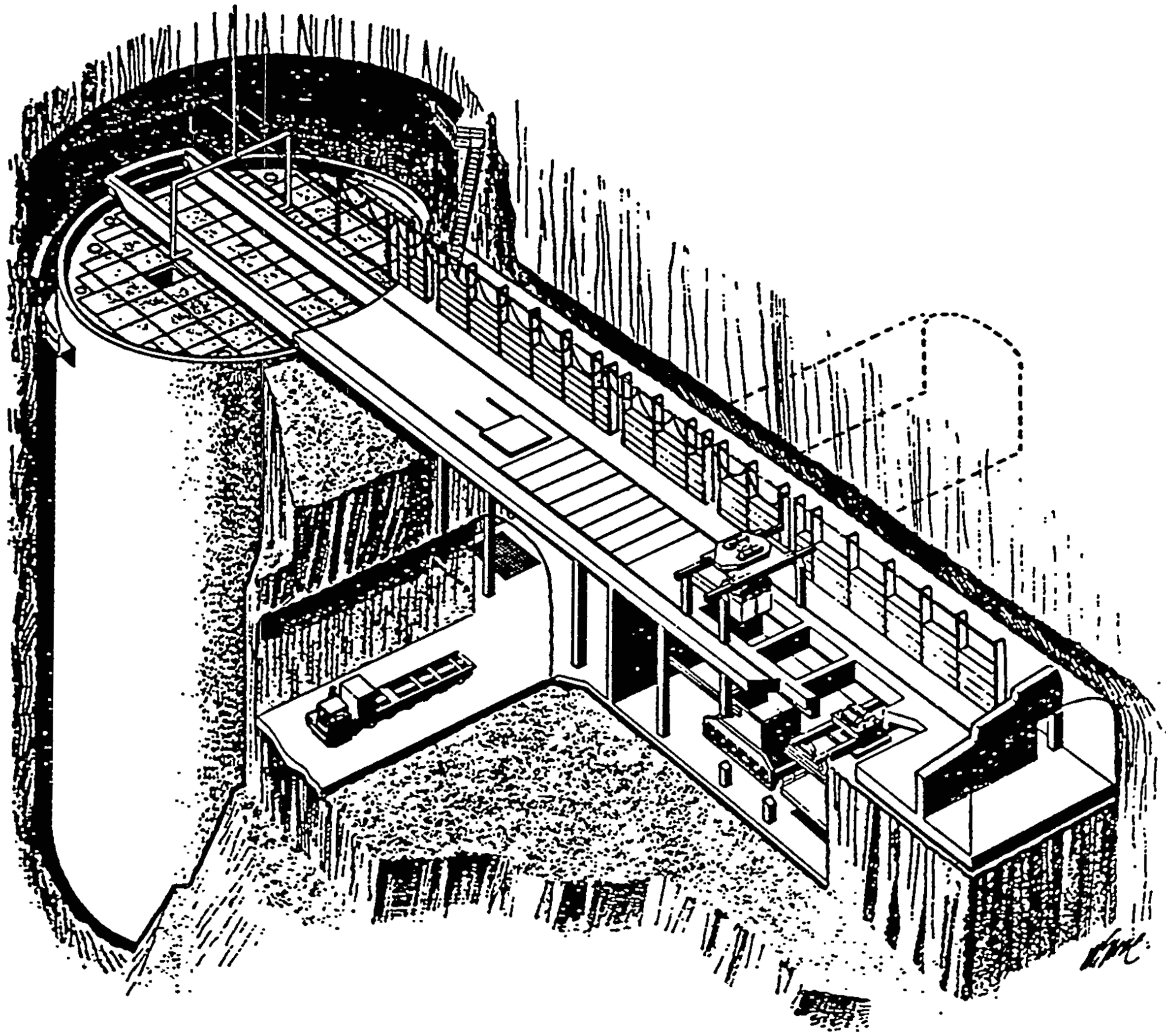

Figure 6.1.47. SFR-1 Repository Silo

Surface facilities at the tunnel portals include the underground ventilation system, the terminal building for temporary container storage, and office and workshop buildings.

Construction of Phase 1 of the SFR-1 repository began in 1983 and waste disposal is planned to start in 1988. Phase 2 expansion is scheduled for the period from 1995 to 1999. The two access tunnels will allow concurrent waste disposal and excavation during future expansion. Excavation is by conventional drill-and-blast methods using drill jumbos for blast drilling and rubber-tired loaders and trucks for rock haulage. Controlled blasting is used to limit damage to the rock formation around the excavations and an extensive monitoring program is being carried out concurrently with excavation to monitor rock deformation, determine rock support requirements, and establish the geohydraulic conditions around the repository.

The SFR-3 repository (Figure 6.1.48) is intended for the nuclear power plant and Studvik decommissioning wastes. A site for this repository has not yet been determined but current plans are to expand the SFR-1 repository. The SFR-3 repository will consist of disposal galleries similar to those in the SFR-1 repository. Low-level wastes will be emplaced in their containers using shielded transport equipment while intermediate-level wastes may be handled remotely.

When all waste has been placed in the SFR repositories, the entrances to the storage chambers will be plugged with concrete and the tunnels backfilled with rock material. The area around the portals will also be restored to its original condition after removal of the surface facilities.

6.1.8.2 The SFL 3-5 Repositories

The SFL-3, 4 and 5 repositories will be combined into a single underground facility and will be constructed about 500 m (1,640 ft) underground in crystalline bedrock. The functions of the repositories are as follows:

1. The SFL-3 repository is intended for the operating wastes from CLAB and the encapsulation station after closure of SFR-1.

2. The SFL-4 repository is intended for the active decommissioning wastes, mainly from CLAB and the encapsulation station, as well as transport casks.

3. The SFL-5 repository is intended for the concrete molds containing the core components (fuel boxes, glass rods).

Figures 6.1.49 and 6.1.50 show the configuration of these repositories. Plans are to locate the SFL 3-5 repositories close to the SFL-2 repository and thus many of the SFL-2 surface facilities can also service the SFL 3-5 repository. These will include the waste encapsulation station, offices,

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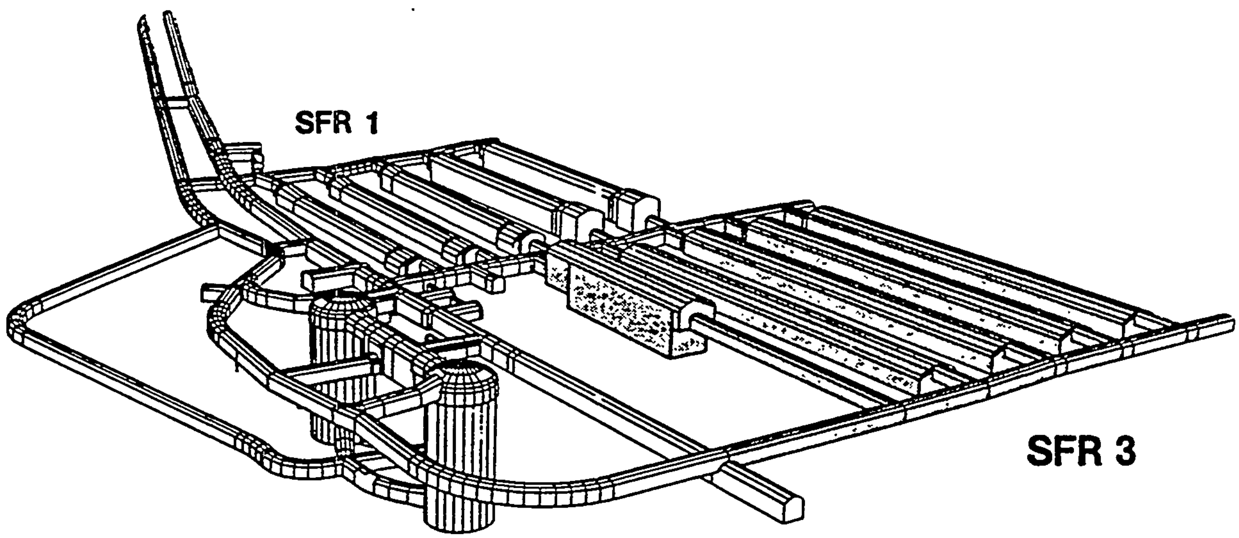

Figure 6.1.48. SFR-3 Repository

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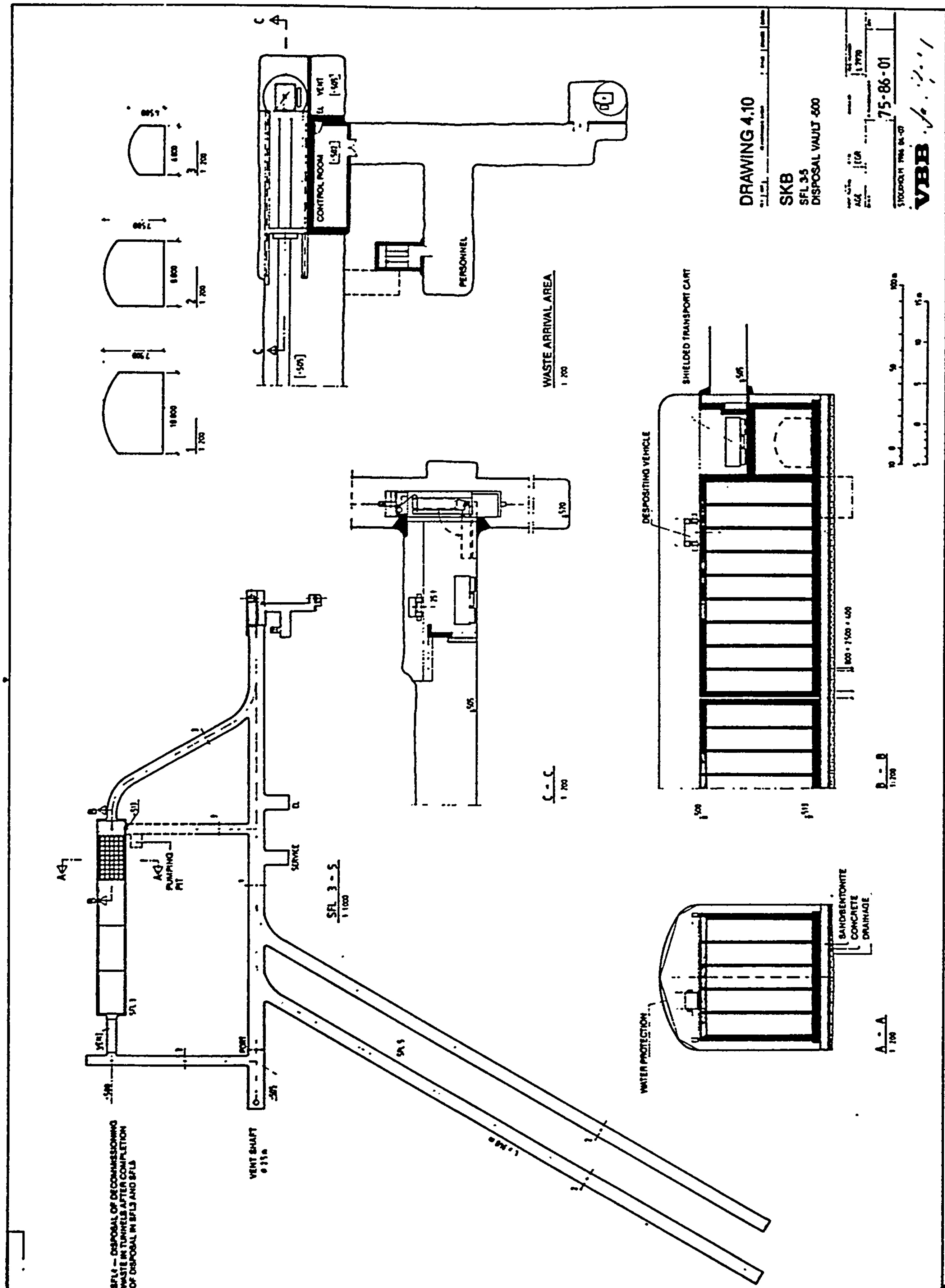

Figure 6.1.49. SFL 3-5 Repositories

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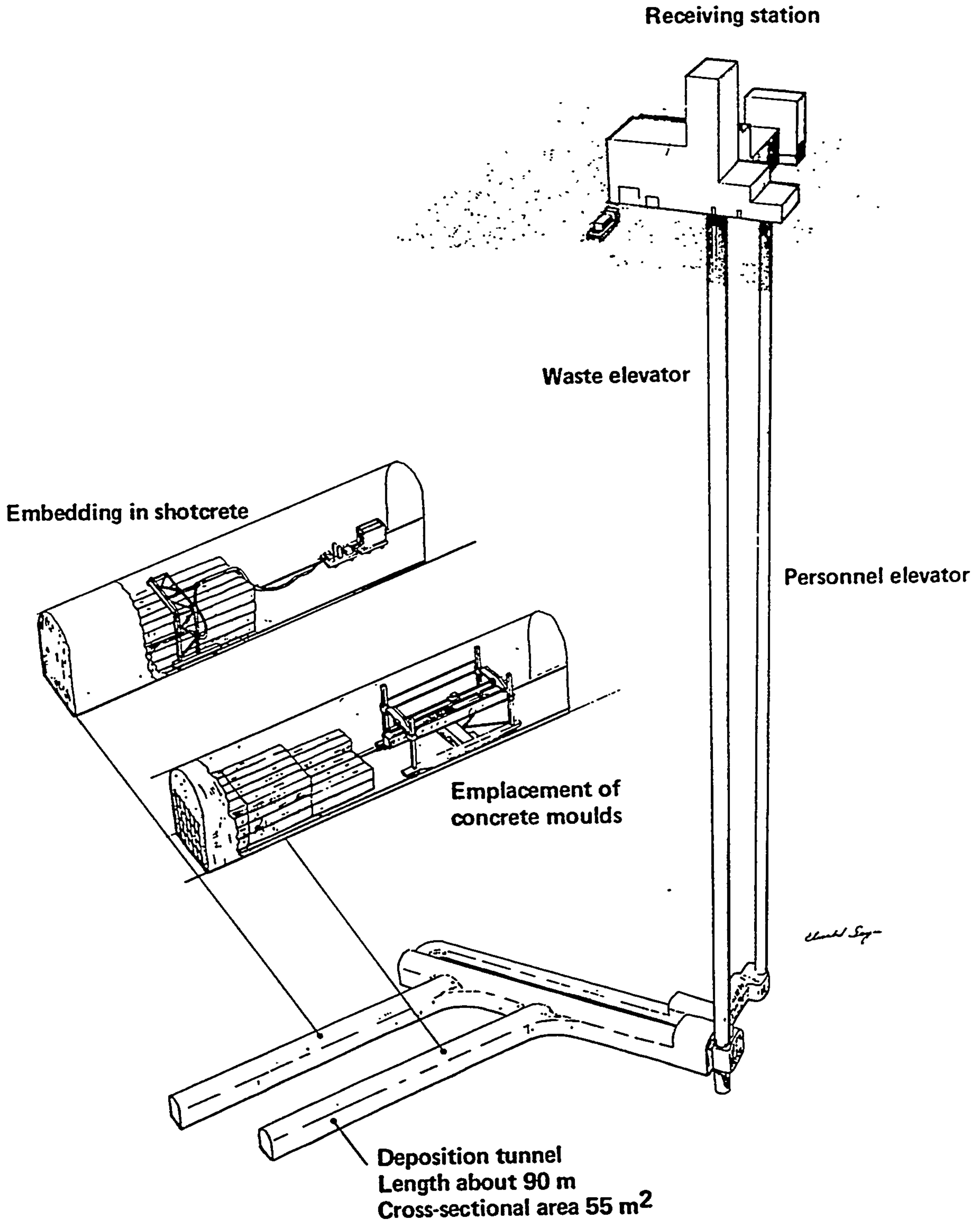


Figure 6.1.5Ø. Repository for Fuel Boxes

housing, change rooms, lamp rooms, and first aid and mine rescue. The facilities at the actual SFL 3-5 site will include the following (Figure 6.1.51):

- Receiving station for unloading the waste containers and transferring them to the waste shaft cage. The receiving station will incorporate the headframe and hoist for the waste shaft plus services such as compressed air, water, electricity, ventilation, etc., for underground. Also included may be minor office space, temporary storage for waste canisters, shop and warehouse facilities, and overhead cranes

- Hoist and headframe for the access and ventilation shafts

- Site fencing, roads and security services.

The repositories will be located about 500 m (1,640 ft) underground in crystalline bedrock and will be accessed via three shafts. One shaft will function as waste transport shaft, the second as ventilation intake and personnel/material access shaft, and the third as exhaust ventilation and emergency egress shaft. The waste disposal areas will consist of caverns and tunnels making up the various repositories (Figure 6.1.49).

Operating waste from CLAB and BS will be encased in concrete and placed in a cavern (SFL-3) approximately 120 m (395 ft) long by 21 m (69 ft) high by 18 m (60 ft) wide. The SFL-3 cavern (Figure 6.1.49) will contain a concrete structure divided into square cells into which the concrete castings will be placed. The structure will be surrounded by a sand/bentonite mixture to isolate the concrete structure from the surrounding rock. A remote controlled overhead crane will manipulate the concrete castings, arriving in a shielded transport cart from the waste shaft, into the cells. When the cells are full, the cavern and service areas will be sealed with concrete backfill. Access tunnels will be sealed with a sand/bentonite mixture.

Two tunnels will serve as the SFL-5 repository for the concrete molds containing the fuel boxes and boron glass rods. The tunnels will be about 340 m (1,115 ft) long with a cross-sectional area of about 55 m^2 (590 ft^2). The concrete molds to be deposited in the tunnels will be handled by a remote controlled crane (Figure 6.1.50). The molds will be stacked into the tunnels and as deposition proceeds, the gap around the molds will be backfilled with concrete. A total of about 2,300 molds will be placed into the SFL-5 repository.

The final wastes, such as the transport casks and decommissioning wastes from CLAB and BS left after operations have ceased, will be placed in the remaining tunnels (SFL-4) following backfilling of the SFL-3 and 5 repository caverns and tunnels. The small steel waste containers will be placed into the tunnels which will be backfilled, possibly with crushed rock material.

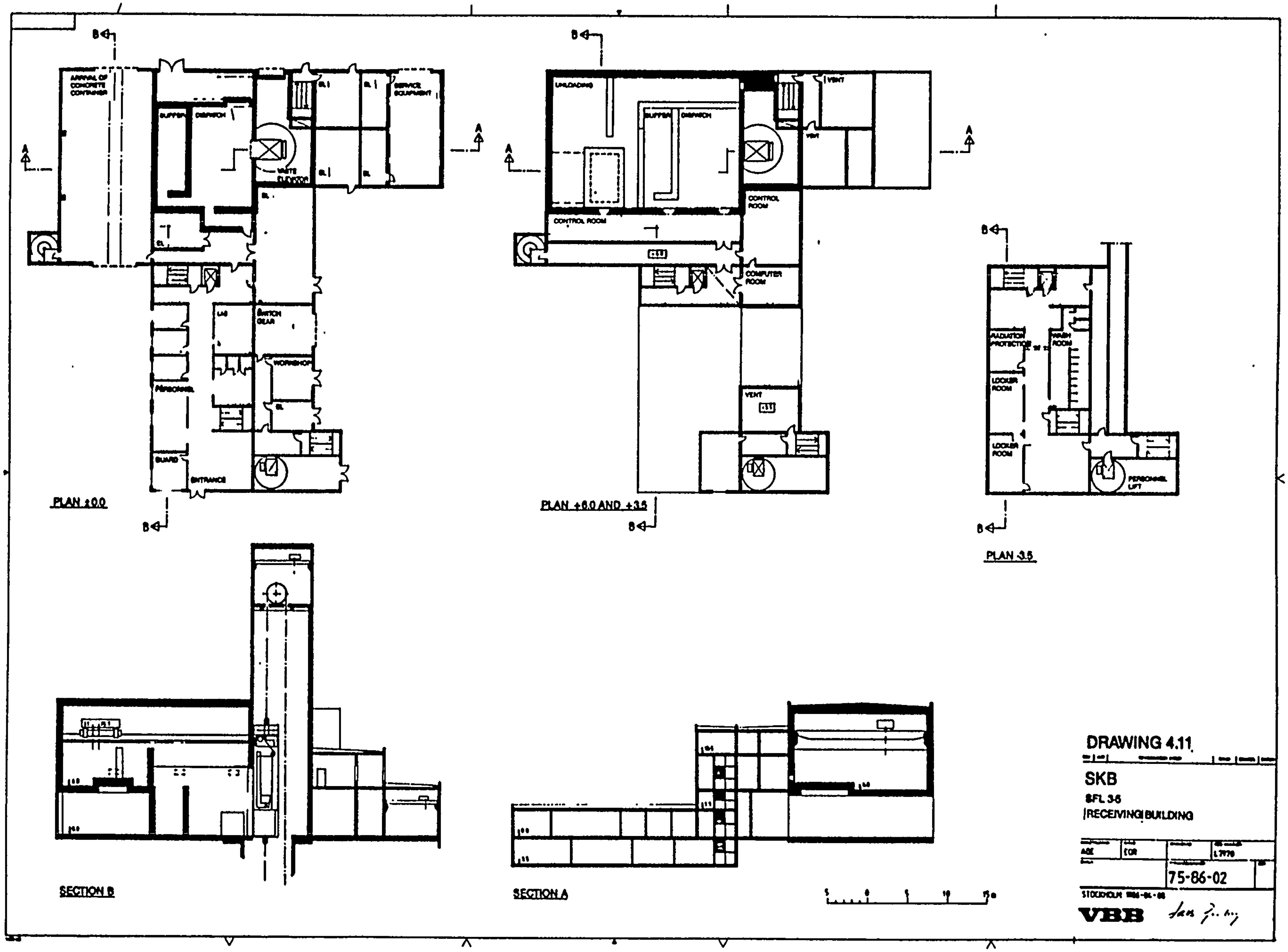


Figure 6.1.51. SFL 3-5 Waste Reception Facility

The final operation will then be to seal the shafts using methods similar to those described for the SFL-2 spent fuel repository.

Figure 6.1.2 shows the schedule for the construction and operation of the SFL 3-5 repositories.

6.2 CRYSTALLINE ROCKS: THE SWISS DISPOSAL CONCEPT

This chapter outlines the Swiss high-level waste disposal concept in crystalline rocks. A technical and safety evaluation has been ongoing for the last decade and in 1985 a series of reports were published as part of the "Gewähr Project" studies to demonstrate the technical feasibility of disposal of nuclear wastes in geologic repositories.

The scope of the Project Gewähr covers the permanent safe management and final disposal of all radioactive wastes and concentrated on:

- The construction, operation, closure and post-operational aspects of the final disposal of all types of radioactive wastes given present-day knowledge

- The preparation of the wastes for final disposal and demonstration of the long-term safety of the disposal option.

A choice of repository site can only be made once a detailed site investigation program has been completed. The Gewähr studies therefore used model data sets, representative of potential repository locations, for the engineering and safety analyses. Current regional investigation programs have confirmed general agreement between the data set and field results. The use of this data base did not, however, prejudice the future selection of the actual location, host rock or repository layout, which will occur around the year 2000 based on an extensive site characterization program.

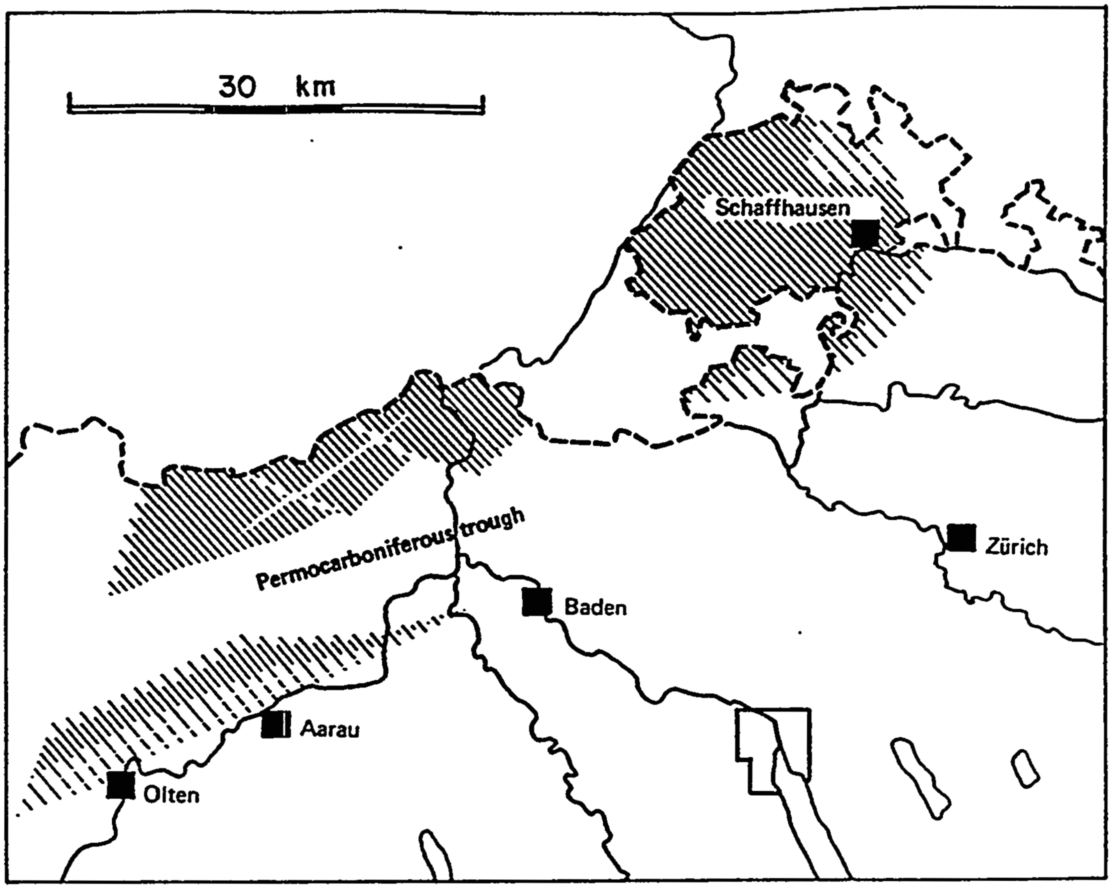
The descriptions in this chapter are based on the results of the Project Gewähr studies.

6.2.1 Geographic Location

The location of the final high-level waste repository has not yet been defined, but for the Gewähr Project an area in northern Switzerland was selected. Figure 6.2.1 shows the potential areas that were identified in the studies.

6.2.2 Summary of Geologic Setting

A regional investigation program of northern Switzerland was started in 1978 and expanded in 1982 to include a series of deep boreholes. This detailed investigation of the crystalline bedrock and the overlying



Overview map showing potential sites for a type C repository in the investigation region in northern Switzerland for the repository concept in Project Gewähr 1985 (for host rocks other than crystalline and other construction concepts, other potential sites exist).

Figure 6.2.1. Potential Repository Areas

sediments has resulted in a wealth of new information and allowed some preliminary assessment of the suitability of the formations in this region as host rocks.

For the Project Gewähr studies, a concept of mined tunnels and silos was selected and the crystalline bedrock in northern Switzerland was chosen as a possible location. A definite site selection was not made due to the conceptual nature of the Gewähr studies but a reference formation was chosen to act as a model data base for the analysis. Within the selected repository concept, the selection of a repository site will be determined by four major factors:

- The exact characteristics of the host rock (in Project Gewähr, Böttstein granite was chosen)
- The permissible rock temperature (around 55°C or 130°F for the Gewähr study) and a resulting maximum repository depth of around 1,200 m (3,930 ft)
- Location of repository within a stable block with adequate distance from major disturbed zones
- Suitable hydrogeologic conditions with low ground-water flows and long travel times

The Böttstein granite, a mostly coarse-grained, porphyritic, biotite granite, from 300 to 1,500 m depth (980 to 4,920 ft) was selected as the representative host rock formation for the Gewähr study. The maximum permissible rock temperature is limited by repository construction and operation considerations and the heat generated by the waste. The temperatures should not exceed 100°C (212°F) in areas where there is bentonite backfill, leading to a maximum ambient rock temperature of about 55°C (130°F) and a maximum depth of about 1,200 m (3,930 ft). Figure 6.2.1 shows the geographic locations of potential sites and the repository is assumed to be located north of the permocarboniferous trough in a several kilometer wide, tectonically stable crystalline formation with low permeabilities and reducing conditions and adequate isolation from larger disturbed zones. It was also assumed that the granite host formation was overlain by about 300 m (980 ft) of sedimentary formations.

6.2.3 Repository Concept Description

This section describes the high-level waste repository concept derived during the Project Gewähr study. The repository concept is based on the disposal of high-level and transuranic reprocessing wastes rather than direct disposal of the spent fuel assemblies and the descriptions in this section therefore focus on this option. The direct disposal option was, however, considered and is also described briefly in Section 6.2.8. For completeness a short summary of the low-level repository concept is also given in Section 6.2.9.

6.2.3.1 Nuclear Waste Disposal Strategy

Two main objectives dictate the Swiss nuclear waste disposal strategy, based on the regulations published by the Swiss reactor safety agencies:

1. The repository designs must be such that closure of the repository can be achieved within a few years at any time and after closure, it must be possible to dispense with safety and supervisory measures.

2. Radionuclide release to the biosphere from a sealed repository resulting from credible processes and events must at no time lead to individual doses which exceed 10 mrem per year.

Before operational licenses will be granted for any more reactors, the nuclear power companies have to demonstrate that the technology exists for the safe and permanent management and disposal of all nuclear wastes in Switzerland. The radioactive waste management program in Switzerland designed to meet these objectives is based on the following assumptions:

1. Final disposal of the radioactive wastes will take place in a deep geological repository which has to provide long-term safety and isolation after closure of the repository.

2. Long-term safety means that release of radionuclides from a sealed repository as a result of a plausible process must not lead to radiation doses exceeding 10 mrem/year.

3. Waste quantities and categories were based on an output from 8 reactors with a total output of 6 GWe and a life span of 40 years. The total amount of vitrified high-level waste produced from reprocessing of all the spent fuel will be about 1,200 m^3 (1,570 yd^3). If the spent fuel is to be disposed of without reprocessing, the volume would be about 3,700 m^3 (4,840 yd^3) of spent fuel without the repository canister.

4. Waste types are categorized according to maximum allowable radionuclide concentrations for the individual repository types designed to meet the radiation protection objectives.

5. All spent fuel is reprocessed abroad with the resulting vitrified high-level waste and conditioned low- and intermediate-level wastes being returned to Switzerland for disposal. A secondary option is the direct disposal of spent fuel elements.

6. Interim storage at a central facility is envisioned for all high-level reprocessing wastes, spent fuel elements and some intermediate-level reprocessing wastes. The storage time is assumed to be at least 40 years.

7. Project Gewähr provides for two types of repositories:

   a. Type C repository for high-level wastes and some alpha-containing intermediate-level wastes.

   b. Type B repository for all remaining low- and intermediate-level wastes (Table 6.2.1 summarizes the waste quantities and the distribution between the various repositories).

   Some consideration is also being given to a Type A repository for low-level waste only which would reduce some of the requirements on the Type B repository. A decision on this option will be made at the time the Type B repository is being finalized.

8. Waste isolation will be achieved by a series of safety barriers. These will include various engineered barriers and geological barriers, depending on the type of waste and its toxicity.

9. The repository projects will be time-phased so that they are operational at the appropriate times. The Type B repository should be ready as soon as technically possible (currently envisioned for the mid-1990s) while the Type C repository has to be operational when the high-level reprocessing wastes are produced after the interim storage period (around 2020).

Figure 6.2.2 summarizes the Swiss nuclear waste disposal strategy as outlined in the Project Gewähr study.

The high-level waste repository (Type C) will receive the following waste types:

- Vitrified high-level wastes from reprocessing, placed in horizontal tunnels

- Alpha-containing intermediate-level wastes, placed in silos.

An alternative to this would be the direct disposal of the spent fuel in the Type C repository (see Section 6.2.8).

The wastes in the tunnels or silos will be isolated from the biosphere by several engineered and geologic barriers. Figures 6.2.3 and 6.2.4 show the barrier concept for the high-level and alpha-containing wastes, respectively. The geologic barriers are the same for both types of waste but the engineered barriers were chosen based on their effectiveness in restricting water penetration into the storage area, limiting the transfer of radionuclides from the waste into the water, retarding the transport of water to the earth's surface, and retarding dissolved radioactive contaminants.

For high-level wastes the barriers will be as follows (Figure 6.2.3):

1. Radionuclides will be dispersed in a leach-resistant glass matrix to limit their release into the ground water.

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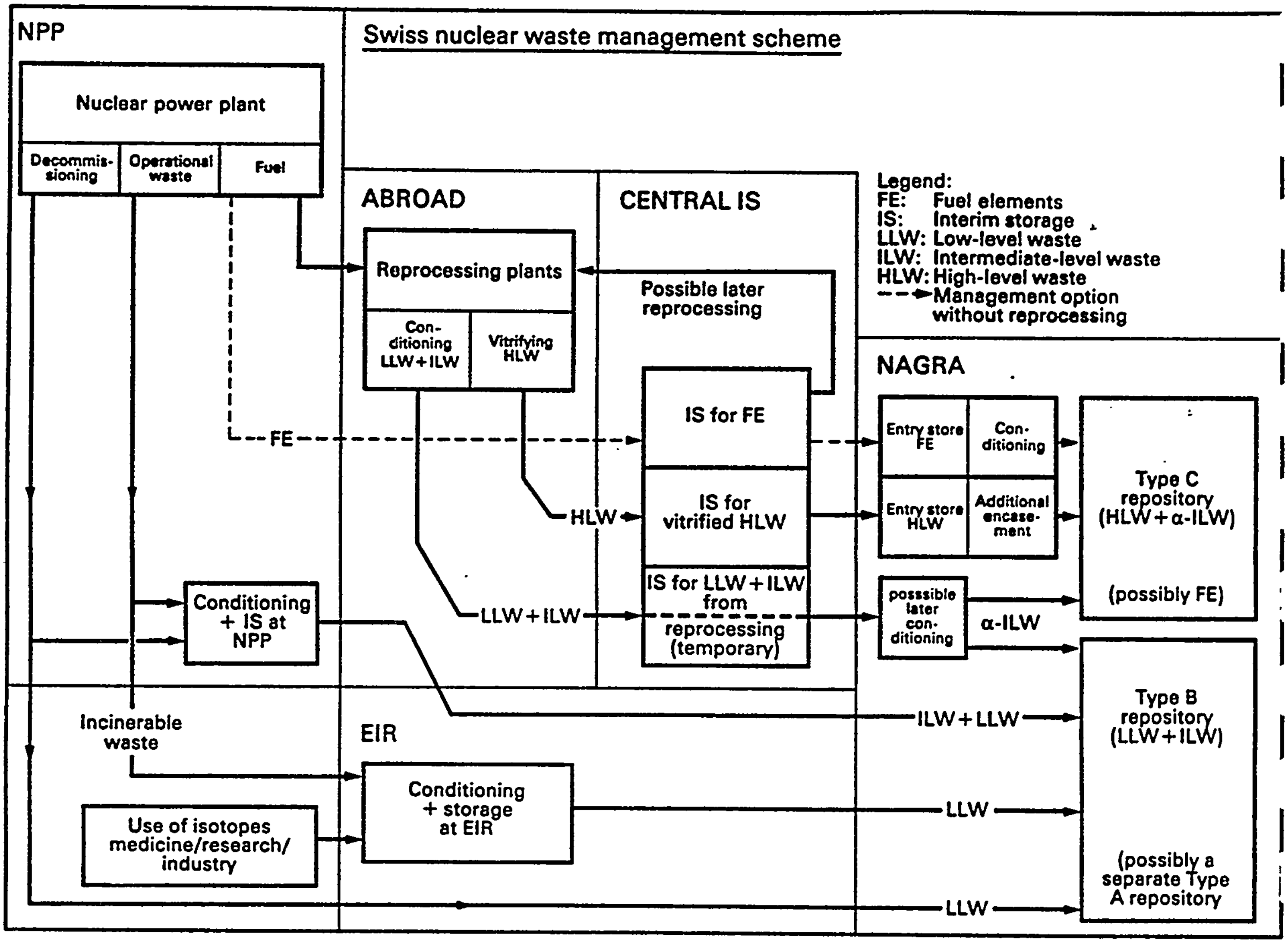

Figure 6.2.2. Nuclear Waste Disposal Strategy

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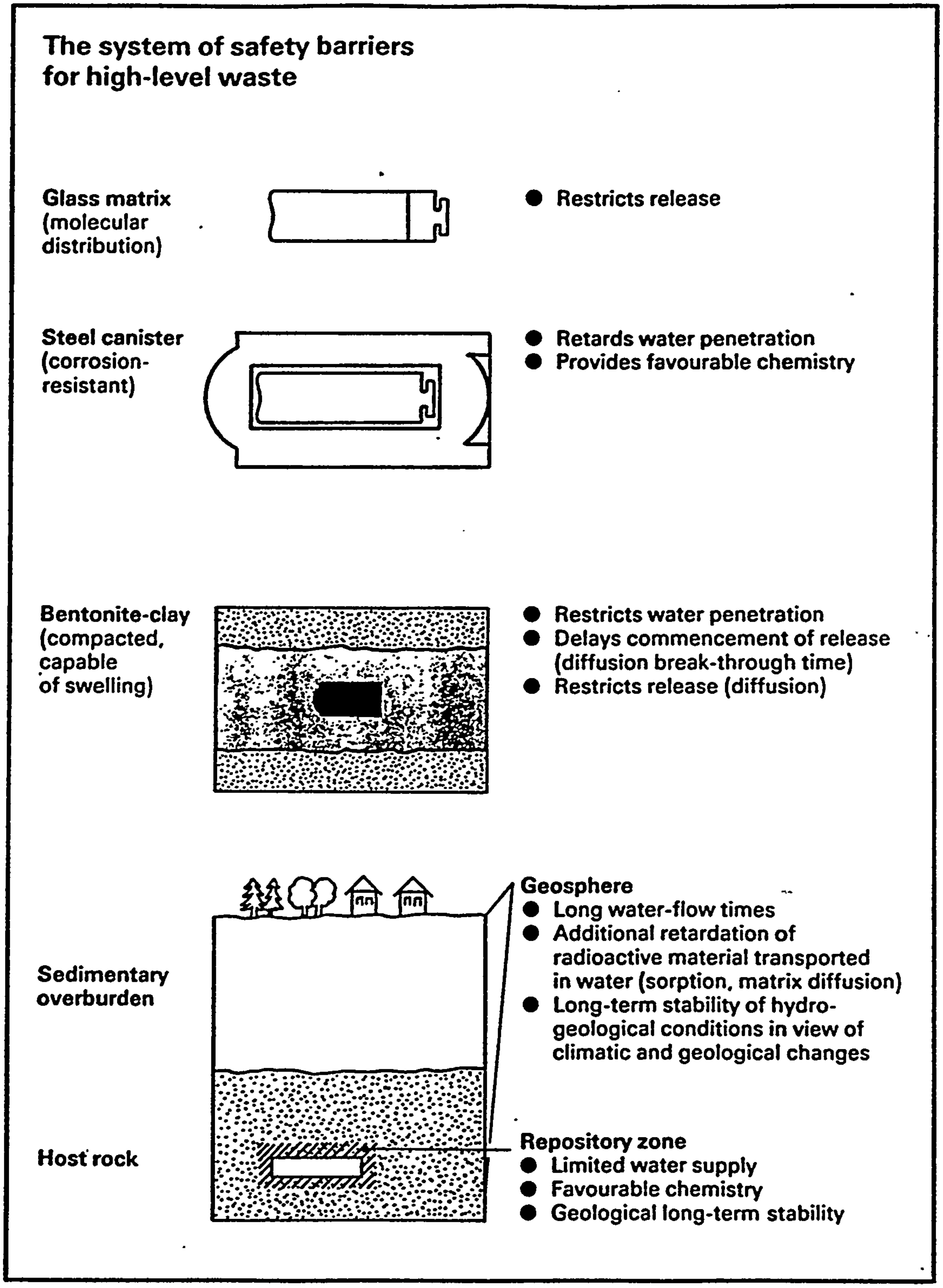

Figure 6.2.3. Multiple Barriers for HLW Waste Disposal

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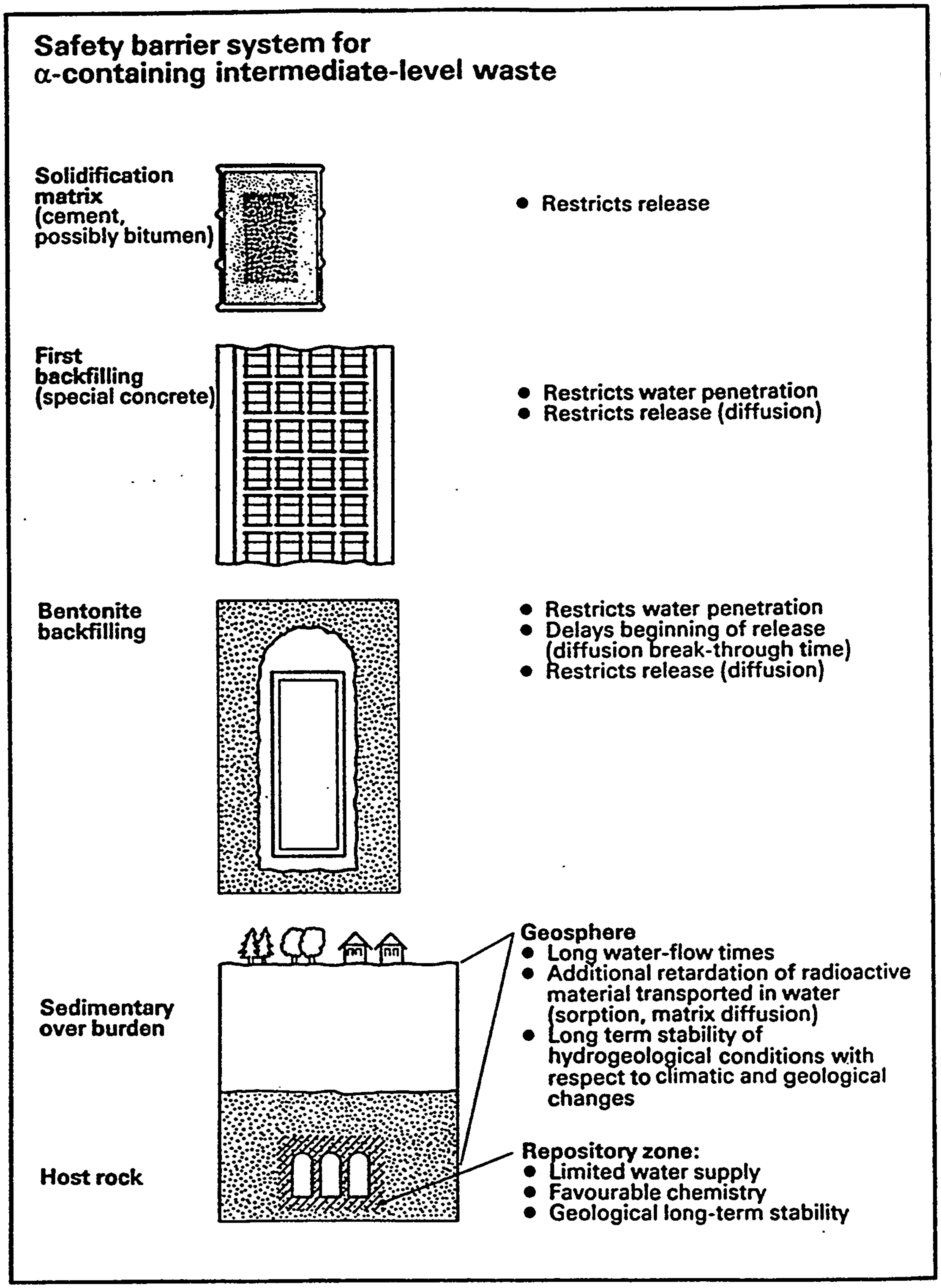

Figure 6.2.4. Barrier for ILW Waste Disposal

: **Total waste quantities for the 240 GWa nuclear energy scenario with reprocessing in terms of the management concept and finalisation in Project Gewähr 1985**

General definition of the management concept (NTB 83–02)							Finalisation of concept in Project Gewähr 1985		
Waste sort (repository type)	Waste composition	Waste features	Waste origin	Weight of radioactive material in fresh waste	Activity of radioactive material (10 years after unloading)	Volume of conditioned waste	Additional reserve for volume-intensive conditioning, packaging, etc.	Distribution of waste types between repositories: Gross volume of conditioned waste	Distribution of waste types between repositories: Repository type
Low-level LLW (Type A)	Low-level short-lived radionuclides (only traces of long-lived radionuclides)	can be manipulated without shielding or cooling	mainly dismantling of NPP's, partly from medicine, research and industry	less than 1 t	less than 0.01 MCi	100,000 m³	40,000 m³	200,000 m³ LLW + ILW	Type B
Low-and intermediate-level LLW + ILW (Type B)	Low-and intermediate-level short lived, limited component of long-lived radionuclides	can be manipulated without cooling but, to some extent, only with additional shielding	operation of NPP's, partly dismantling of NPP's, operation of reprocessing plants, partly from medicine, research and industry	less than 2 t	less than 150 MCi	70,000 m³			
High-level HLW (Type C)	High-level short- and long-lived radionuclides	initially high heat output: can be manipulated only with cooling and heavy shielding	reprocessing plants	290 t	less than 4400 MCi	1000 m³	200 m³	10,000 m³ ILW 1200 m³ HLW (net volume HLW = 884 m³)	Type C

Table 6.2.1. Nuclear Waste Quantities

2. Corrosion-resistant steel canisters around the glass cylinder will delay the contact time between water and waste and initially provide total containment. Later, the metallic iron and iron corrosion products ensure reducing conditions in the near-field, lowering the solubility of some nuclides.

3. The canister will be surrounded by highly compacted bentonite. Convective water flow through the bentonite is negligible and bentonite acts as a chemical buffer, retarding diffusion of many radionuclides. Bentonite also swells on contact with water and therefore is a favorable material for backfilling the storage cavities and can protect the waste package from tectonic movements.

4. The host rock is important from two aspects: it protects the engineered barrier system from climatic or tectonic disturbances and therefore the repository will be located in a host rock which has been stable over long periods; also, the host rock provides near-field conditions of low water supply and favorable geochemical conditions (reducing, near neutral PH water).

5. The geologic setting of host rock and its sedimentary overburden provides long ground-water travel times and radionuclide migration times will be increased by sorption and matrix diffusion. Additionally, it also protects from human intrusion.

For the alpha-containing intermediate-level wastes the following barriers (Figure 6.2.4) apply:

1. The wastes will be solidified in a leach-resistant matrix (cement or bitumen) to restrict radionuclide release into the ground water.

2. The solidified waste will be placed into a concrete lined underground silo and encased in special concrete so that the concrete wall and backfill act as additional barriers to radionuclide release and water infiltration.

3. The gap between the concrete lining and rock wall will be filled with bentonite to provide the same protection as was described for the high-level wastes.

4. The function of the natural barriers will be the same as for the high-level waste repository.

The repository concept chosen to incorporate all the above systems for the Project Gewähr study was a system of mined tunnels and silos at a depth of about 1,200 m (3,930 ft) in the crystalline bedrock. The concept was chosen based on a representative geological situation in northern Switzerland and the repository is assumed to be located in a several kilometer-wide stable granite block between two major faults.

The Type C repository for high-level wastes is based on the following:

1. High-level waste disposal will be in horizontal tunnels and intermediate-level waste disposal in vertical silos. Access to the underground facilities from the surface reception area will be via two shafts. The tunnel and silo system is designed to be adaptable to changes in host rock conditions. The tunnels and silos will be separated to avoid undesirable interactions (e.g., chemical).

2. Extensive site investigation programs will be conducted to locate major disturbed zones so they can be avoided by the repository layout. Minor disturbed zones intersecting the excavations will be backfilled and no waste disposed near them.

3. A final safety evaluation of long-term, in situ experiments, including observations of the materials in the repository for several decades, will take place before final sealing of the repository. If necessary at this time, retrieval of the waste will be possible, though difficult.

4. The safety barriers will be as described previously. The waste will arrive at the repository in solidified form and the remaining engineered barriers will be installed during the construction, operation and closure of the repository.

5. Wastes will be prepared on surface in the reception area with the high-level wastes being encapsulated in the canisters. Transport of the wastes in the shaft and underground will require additional shielding and remote operation will be used in areas of higher local radiation.

6. Quality control measures will be implemented throughout the waste disposal operation and facilities for repair of canisters, decontamination of canister surfaces, etc., are foreseen.

7. An exact inventory of the waste will be kept.

8. Construction and emplacement operations will be carried out simultaneously but all blasting work will be completed before start of waste disposal.

Figure 6.2.5 shows a perspective view of the Type C repository. The following sections describe the facilities in more detail, beginning with the surface facilities, followed by the underground layout.

6.2.3.2 Surface Facilities

6.2.3.2.1 Interim Storage. Before the high-level waste or spent fuel is actually placed into the underground repository, it will be stored in an interim storage facility for 40 to 50 years to allow reduction in heat and

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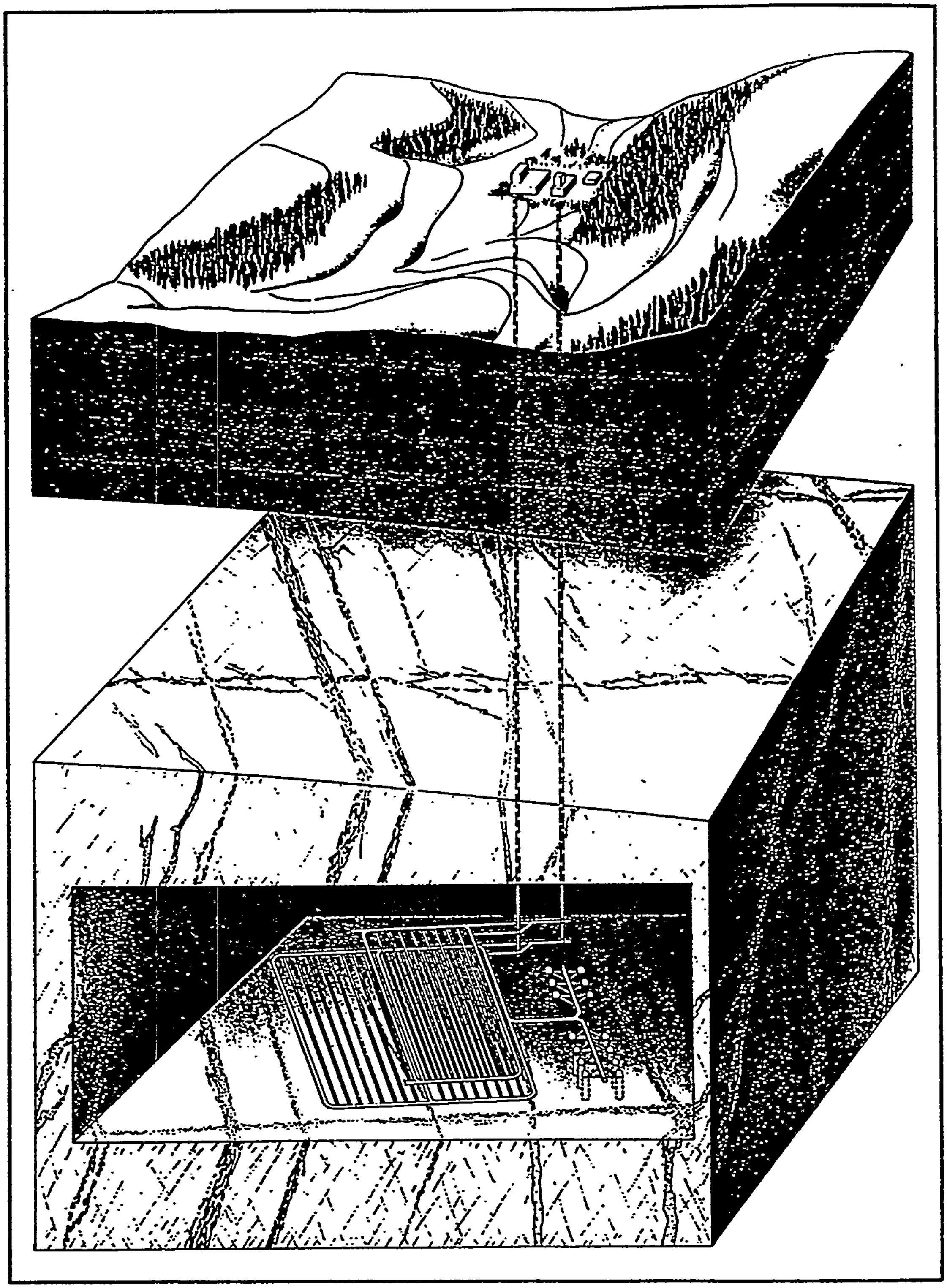

Figure 6.2.5. Type C Repository Configuration

radioactivity emitted by the wastes. Interim storage of spent fuel is now stored initially at the reactor in water-filled pools to allow the radioactivity and heat production to decrease, and to some extent at the receiving station of the reprocessing plant. High-level waste is stored partly in liquid form in high-grade steel tanks, and partly in vitrified form at the reprocessing plant. Transfer from foreign reprocessing facilities of the high-level waste from treatment of Swiss spent fuel to an interim storage facility in Switzerland is foreseen to start, at the earliest, in 1992. At this point, the high-level waste will be put into storage for about 30 years and thus the interim storage facility must be operational at that time (giving a repository start-up date of about 2020).

Evaluation of the methods of interim storage is still ongoing. One concept being proposed is shown in Figure 6.2.6. The storage concept is as follows:

1. Dry storage of fuel element or high-level waste in transport containers (CASTOR-type) in surface halls with a storage capacity for 30 to 50 years.

2. Low- and intermediate-level wastes are stored in separate halls.

3. Construction of the facility would proceed in stages. The first stage would create enough capacity for the first 15 to 20 years of storage.

The proposed interim storage facility will contain the following facilities:

1. A storage hall for 184 transport containers loaded with high-level waste from about 5,000 MT (5,500 tons) of reprocessed spent fuel (approximately 500 m^3 or 655 yd^3 high-level waste net volume) or with 1,555 MT (1,700 tons) non-reprocessed spent fuel elements. This capacity covers the storage requirements until 2020 with the reprocessing option and until 2005 if the spent fuel is not reprocessed.

2. A storage hall for about 4,000 m^3 (5,230 yd^3) of conditioned and packaged low-level waste.

3. A storage hall for about 1,000 m^3 (1,310 yd^3) of conditioned and packaged intermediate-level waste.

4. All necessary back-up installations and facilities for the transfer and control of the wastes, administrative facilities, etc.

6.2.3.2.2 Type C Repository for HLW. Figure 6.2.5 shows a conceptual view of the Type C repository layout. The surface facilities at the repository are likely to include the following:

1. The waste reception area where the waste will be delivered and stored temporarily. The high-level waste will also be encapsulated

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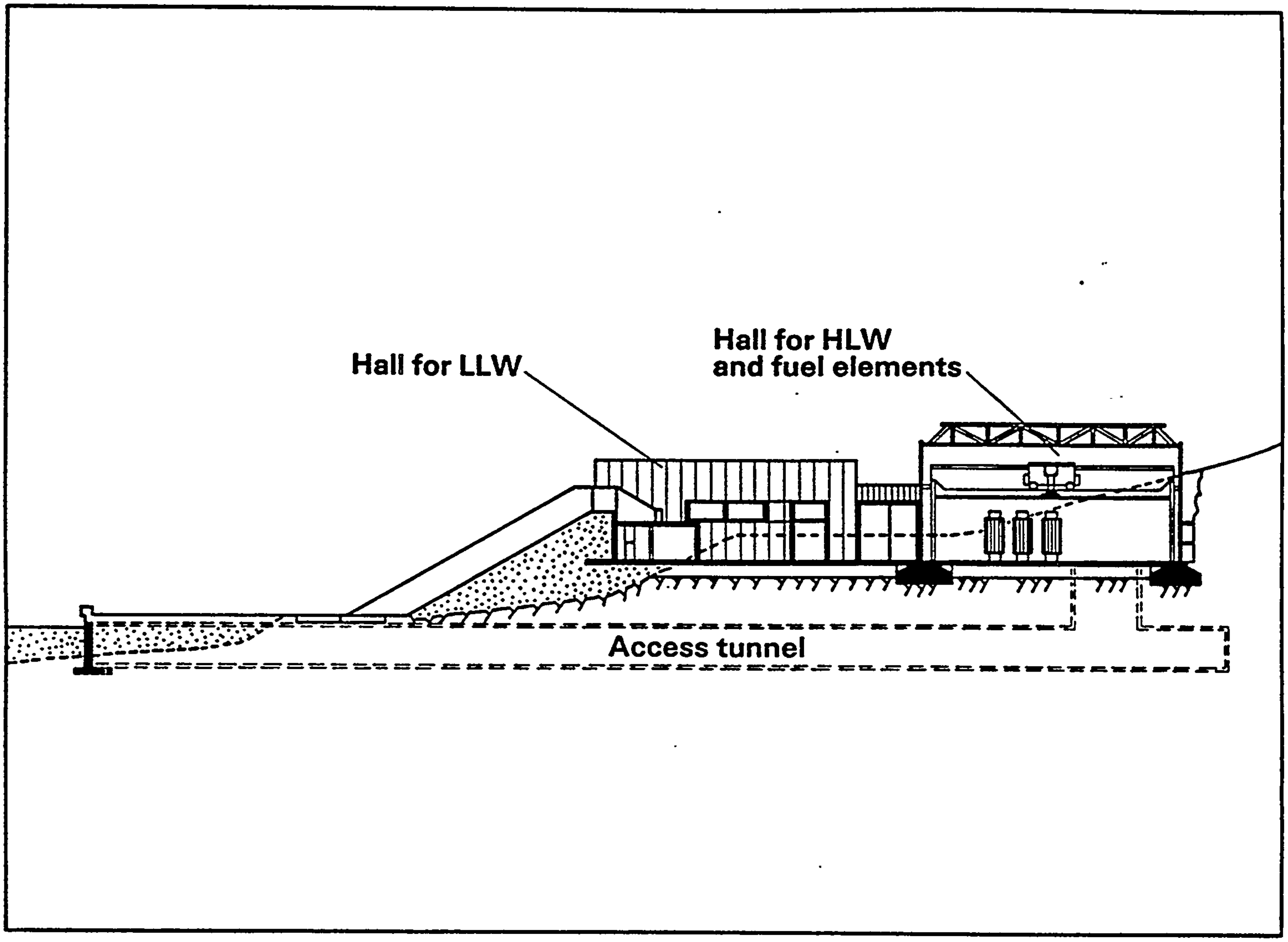

Figure 6.2.6. Interim Storage Concept

in the steel canisters in this facility. Figures 6.2.7 and 6.2.8 show sections of the reception facility.

2. The shaft headframes, shaft houses, hoist and hoist houses at each shaft location. The waste shaft headframe will likely be incorporated into the reception facility to allow direct access with the waste canister to the shaft cage for loading.

3. Underground ventilation facilities (fans, ducts, heater and/or cooler buildings, etc.).

4. Concrete and bentonite preparation/storage areas.

5. Excavated rock storage areas.

6. Sewage treatment/sediment ponds.

7. Warehouse, shops, first aid and mine rescue facility, change houses, offices, security, etc.

Depending on the remoteness of the location, personnel housing may also have to be provided.

6.2.3.3 Underground Configuration

Figures 6.2.5 and 6.2.9 show the proposed underground repository configuration. The repository will consist of a series of parallel entries on one or more levels for the disposal of high-level waste canisters and a number of vertical silos for the disposal of the intermediate-level wastes. The underground facilities will be connected to the surface via two shafts.

The repository will be located about 1,200 m (3,930 ft) underground in the crystalline bedrock. Both shafts will be about 1,250 m (4,100 ft) deep and have a finished diameter of about 6.6 m (22 ft). The shafts will be furnished in the same manner with a hoisting system of approximately 25 tons capacity. One shaft will serve as the personnel and material shaft as well as the ventilation intake. The other shaft will serve as the waste transport shaft and ventilation exhaust. The shafts will be lined with a water-tight steel and concrete lining in the water-bearing sediments overlying the granite to prevent any water inflow into the underground facility. In the granite, the lining will consist of about 50 cm (20 in) of concrete to secure the shaft wall. Figure 6.2.10 shows the lining proposed for the various shaft sections. Both shafts will be equipped with a friction hoist and a two-cage hoisting system for lowering waste, material and personnel to the repository as well as hosting the excavated rock from tunnel and silo excavation operations. Figure 6.2.11 shows the conceptual shaft configuration. In addition to the two main cages, the shafts may also be equipped with a small service hoist. The shafts will also be furnished with all the underground service lines such as compressed air, water, telephone, power, etc. A shaft station will be excavated in each shaft at the repository horizon. This shaft station will allow room the handling of

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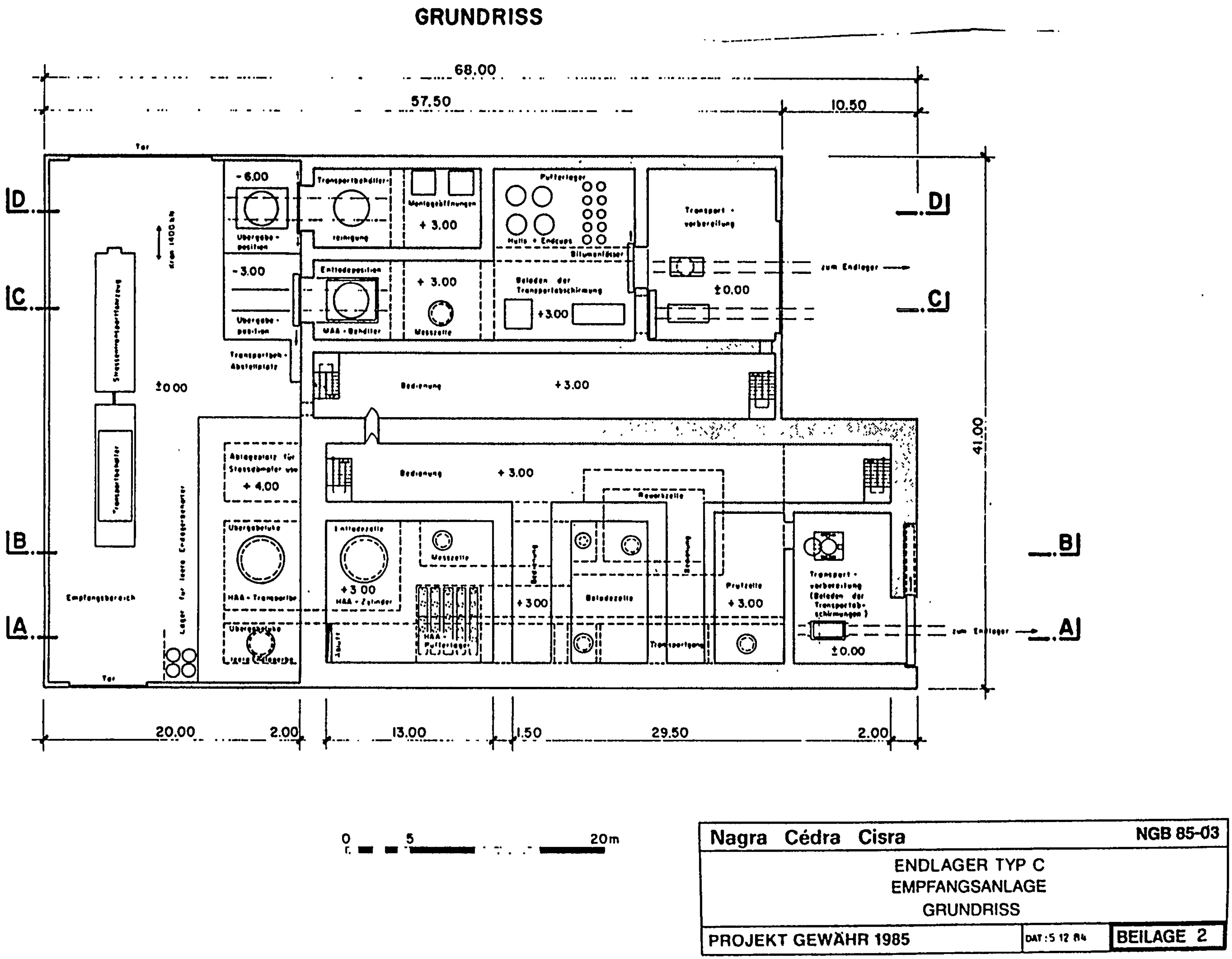

Figure 6.2.7. Reception Facility for Type C Repository

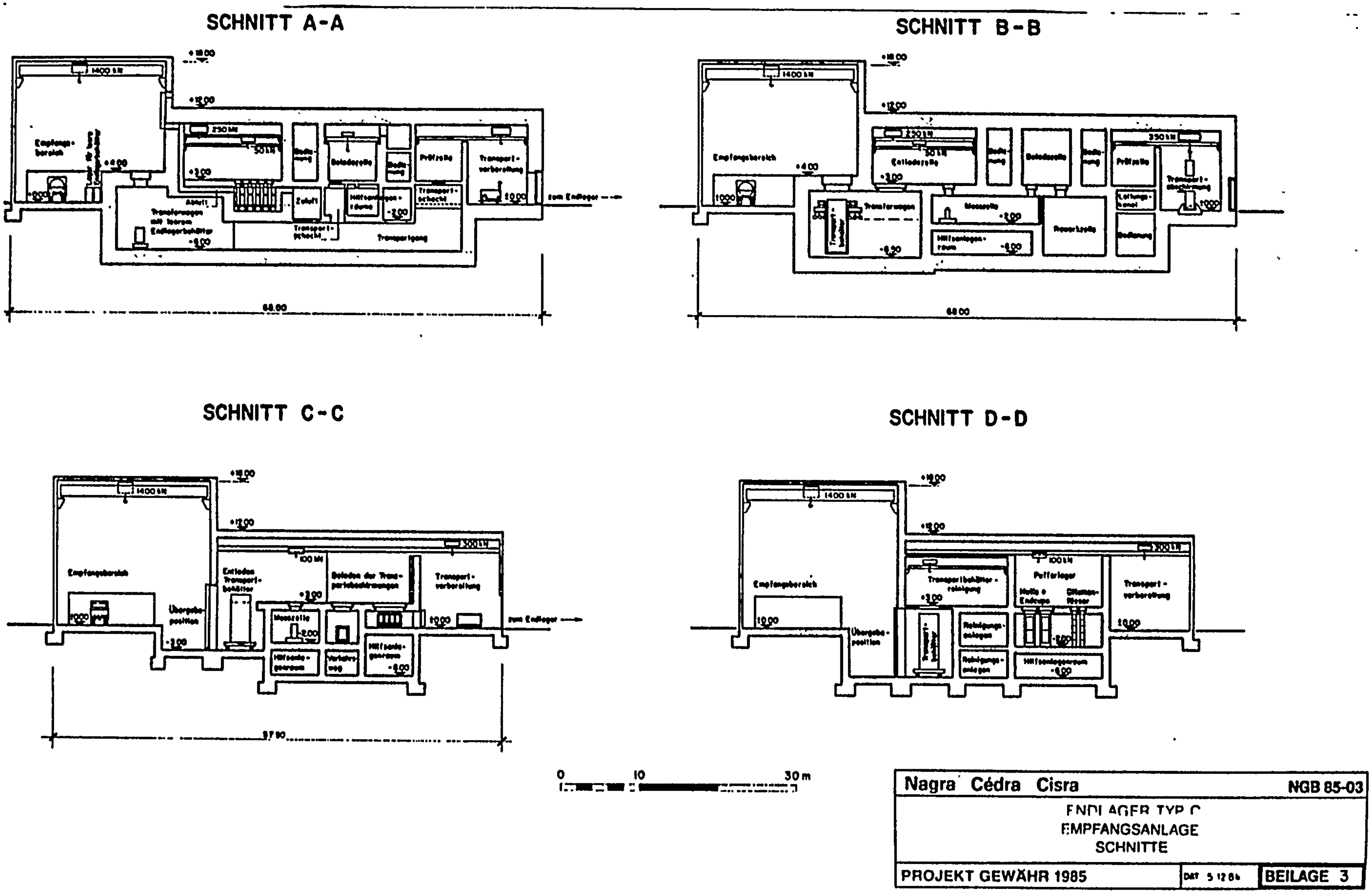

Figure 6.2.8. Reception Facility for Type C Repository

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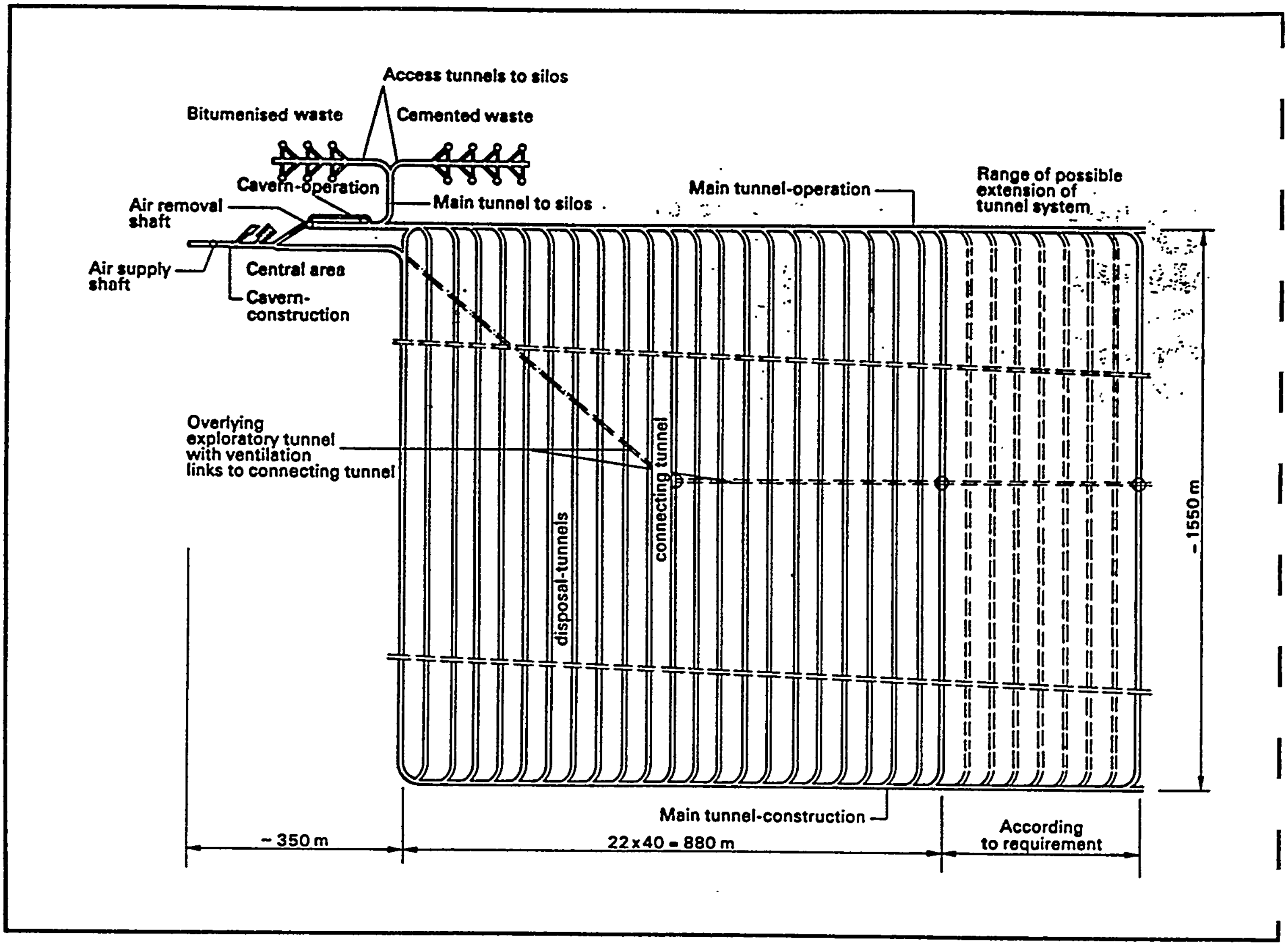

Schematic lay-out of repository area. The arrangement will be adapted to the geometry of the host rock in the storage zone.

Figure 6.2.9. Underground Repository Layout

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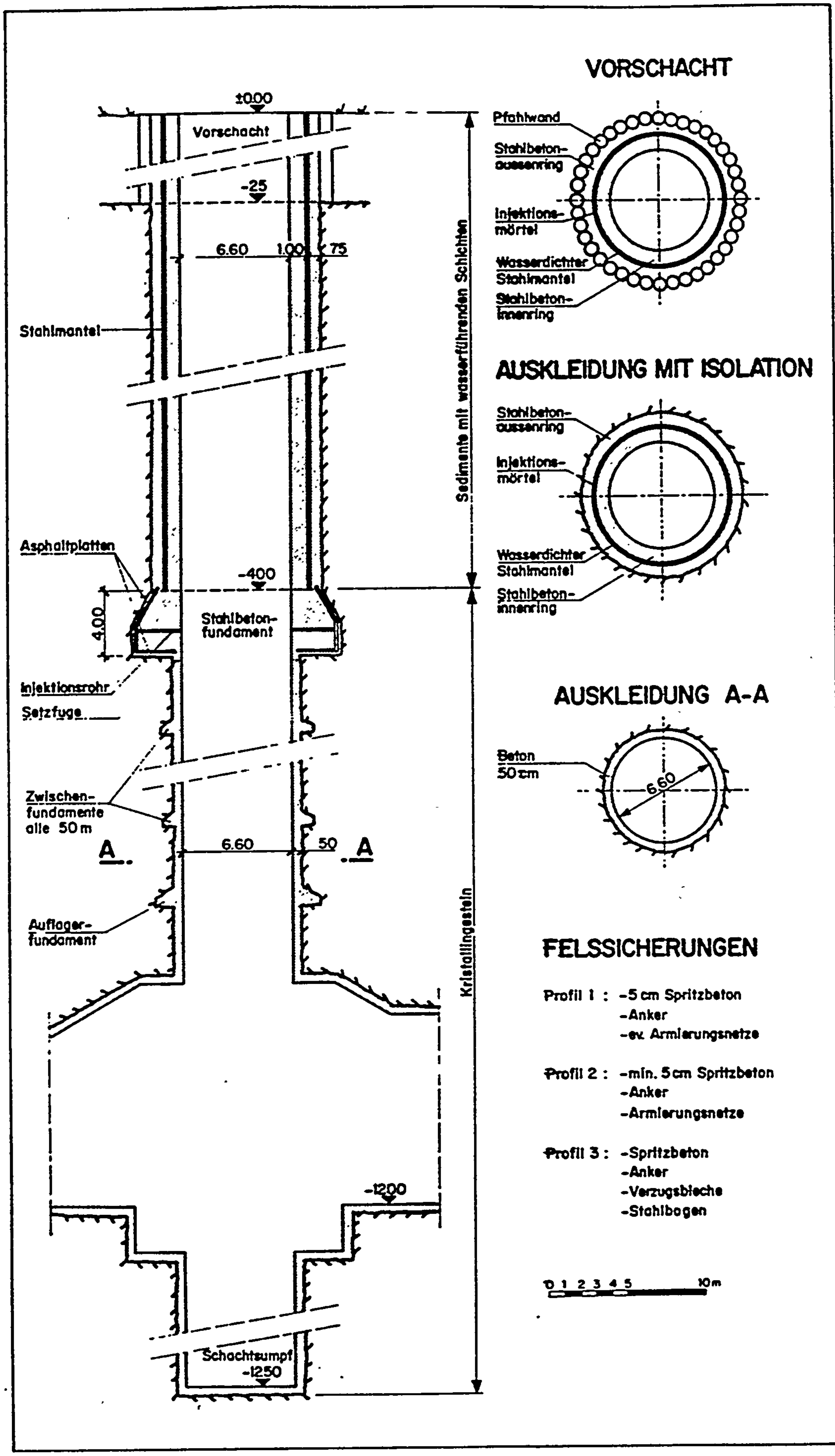

Figure 6.2.10. Upper Shaft Section

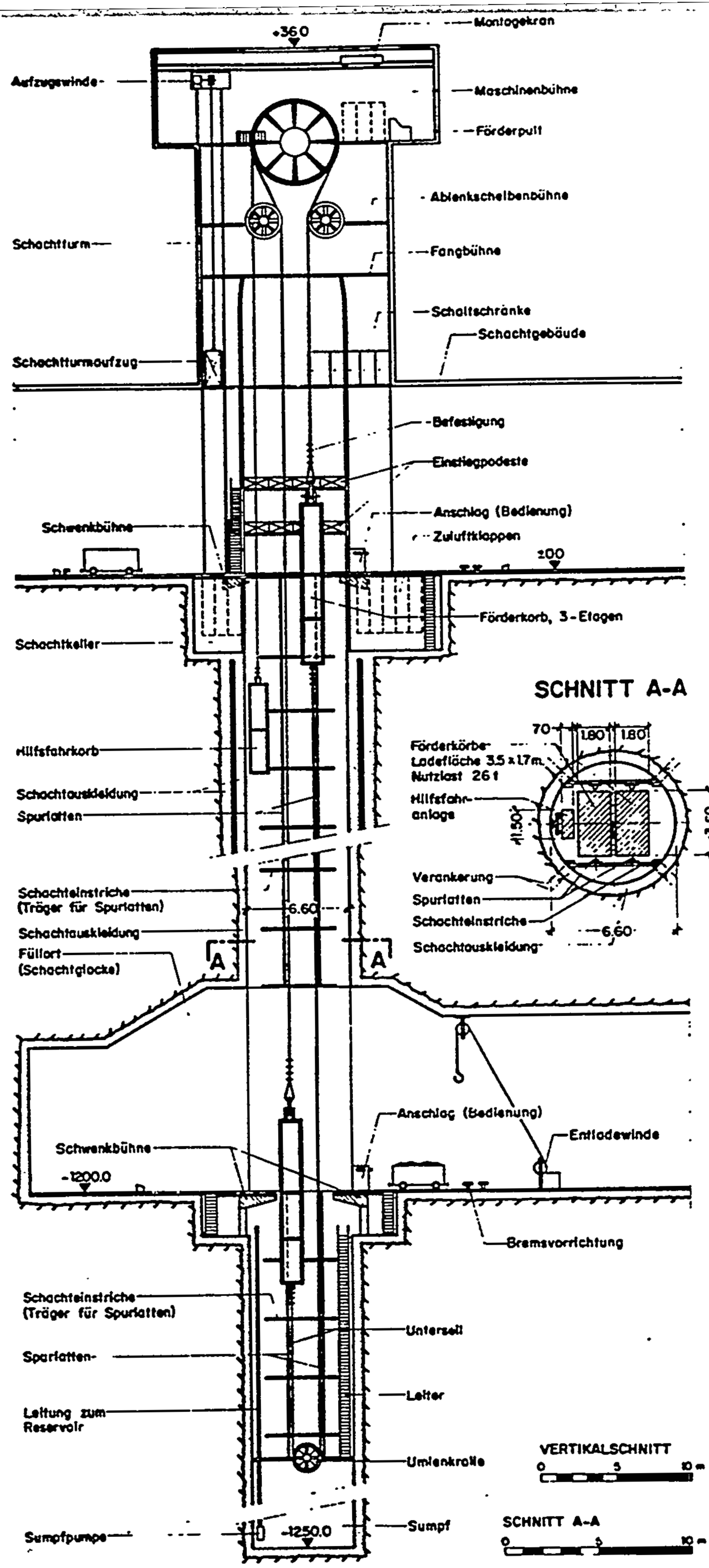

Figure 6.2.11. Lower Shaft Section

waste canisters and construction materials. A central area around the shafts will consist of two caverns (Figure 6.2.9). These caverns, one near each shaft, are connected by a heading containing a bulkhead. This will allow the separation of the excavation operations from the waste emplacement process. Other underground service areas such as shops, electrical distribution station, etc., will also be located in the vicinity of the shaft areas.

The repository area can be divided into two main areas (Figure 6.2.9). One area will be the parallel entries for the disposal of the high-level waste canisters (or direct disposal of spent fuel assemblies) and the other the silo area for the disposal of the intermediate-level wastes immobilized in bitumen or cement. The overall repository dimensions will be approximately 1,500 m by 1,500 m (4,920 ft by 4,920 ft). Separate access headings will be provided for waste transport and excavation operations to each area (Figure 6.2.9).

Two main entries will be excavated around the high-level waste repository area before the start of the disposal operations (Figure 6.2.9). The cross-sectional area of these entries will be approximately 60 m^2 (645 ft^2) to accommodate the waste transport vehicles. The parallel, circular disposal tunnels will be excavated using tunneling machines with a diameter of about 3.7 m (12 ft). About 20 or more tunnels will be required, spaced at about 40 m (131 ft) with a length of approximately 1,500 m (4,920 ft). Waste disposal operations will be conducted from one main heading and excavation from the other (Figure 6.2.9). The high-level wastes will be placed in the tunnels at regular intervals and the surrounding space sealed with bentonite backfill (see Section 6.2.6).

About 14 silos for the intermediate-level wastes will be constructed in an area separate from the tunnel repository. The circular silos will be about 55 m (180 ft) high and 10 m (33 ft) in diameter. Each silo will contain a free-standing concrete structure into which the waste canisters will be placed and immobilized with concrete backfill. The gap between the concrete structure and the silo rock wall will be backfilled with bentonite to completely isolate the emplaced waste (see Section 6.2.6).

6.2.4 Repository Construction

This section describes the proposed construction of the underground facilities for the repository. Systematic investigation of the repository level will ensure that zones of major disturbances will be avoided during the construction of the final repository layout. The plan is therefore to complete the drilling and blasting excavation of the main entries and access drifts around the repository area before the start of machine tunneling of the emplacement tunnels. Minor disturbed zones which cross the disposal tunnels can be isolated using appropriate sealing techniques and no waste will be stored in their vicinity to ensure proper isolation of the waste canisters.

6.2.4.1 Schedule

The current target date for the start-up of the Type C repository is the year 2020 based on the 40-50 years interim storage period included in the high-level waste disposal strategy. The construction period for the complete facility, excluding geological investigations, is estimated to be around 15 years. Start of construction is therefore scheduled for between 2005 and 2010. In the period between now and the start of construction, a detailed site investigation program will be conducted. Phase I consisted basically of the Project Gewähr studies including regional investigations, conceptual studies, siting studies, etc. Phase II of this program will last until about 1995 and will include further geological investigation at one or more potential repository sites. Underground experimentation will also continue at the Grimsel Underground Research Laboratory to supplement site investigations. During Phase III, up to about 2010, an initial site choice will be made and detailed geological investigations will be conducted including an underground laboratory at the potential repository site. After the site has been accepted, start of construction can begin.

6.2.4.2 Methods and Equipment

The actual construction of the Type C repository for high-level and intermediate level wastes will be the final phase of an intensive site investigation program. The investigation program and the initial construction phases will basically form part of the same operation. The intent is to sink a pilot shaft to the repository horizon at the potential site which has been exhaustively characterized using deep boreholes as well as other surface-based investigation techniques. The shaft will be located so that it can later be used as a repository shaft. An underground laboratory for in situ rock investigation will be constructed in what will later be the central service area (Figure 6.2.9). On the basis of the result of this in situ investigation, a preliminary repository configuration will be established (e.g., location of second shaft, layout of tunnels and location of silos). Exploratory drifts will be driven along the lines of the future main headings (Figure 6.2.9) and based on further testing in these, a final layout will be derived. At this point, experiments to investigate the long-term behavior of the canister and backfill materials, radionuclide transport, etc., may also be installed. Each drift and other new excavation will be geologically mapped, both during the experimental and the construction phases as well as continuing with the geological, rock mechanics, etc., experiments. Before final closure of the repository, all data will be analyzed to provide a safety assessment of the overall long-term repository performance.

6.2.4.2.1 Repository Shafts. The two shafts will likely be identical and Figures 6.2.10 and 6.2.11 show a conceptual shaft configuration. The construction techniques and shaft lining configuration will vary, depending on the rock formations. The approximately 400-500 m (1,310 to 1,640 ft) of sediments overlying the granite may be water-bearing and unconsolidated. In this case, shaft construction may require special ground stabilization and lining techniques in order to guarantee exclusion of the ground water from the shaft and therefore repository excavations. As Figure 6.2.10 shows, in the shaft collar area down to about 25 m (82 ft), the ground may be stabilized using piling techniques, while in the region from the bottom of the shaft collar to the top of the granite the application of shaft freezing or grouting techniques may be necessary to retain the ground water and unconsolidated formations long enough for the installation of a suitable lining. The shaft lining shown in Figure 6.2.10 is a completely watertight lining consisting of an inner reinforced concrete lining, a welded steel liner, and an outer reinforced concrete lining with a grout backfill between the steel liner and outer concrete. In some cases, a preliminary safety lining of rock bolts, wiremesh and shotcrete may be applied during excavation to prevent spalling of the shaft wall before installation of the final lining described earlier. The lining column will be supported by a foundation keyed into the competent granite formation below the sediments. A seal made of asphalt or chemicals will be installed at the base of the lining column to prevent water migration down the shaft wall behind the lining. This type of concrete and steel lining has been employed in shafts in several countries (notably West Germany, Canada, Great Britain) when absolute watertightness was required. It is also similar to that proposed for the exploratory shafts at the Texas salt repository site. The excavation diameter in this upper region may be as large as 10-11 m (32-36 ft) to accommodate the lining which can be 1.75 m (5.7 ft) thick and still leave a required finished shaft diameter of about 6.6 m (22 ft). In the lower shaft section in the granite, an approximately 50 cm (20 in) concrete lining may be all that is needed due to the stability of the granite formations.

The shafts will most likely be constructed using conventional drill and blast techniques with the application of controlled blasting to prevent excessive damage to the rock around the shaft. The presence of waterbearing formations may exclude the use of drilling techniques such as raise boring or large hole drilling.

Figure 6.2.11 shows a sketch of the type of shaft furnishing that may be installed in each shaft. The major installation in each shaft will be the two shaft cages for lowering the waste canisters in one shaft and the hoisting and lowering men, material and excavated rock in the other. The capacity of the hoist for each shaft is expected to be in the region of 25 tons. Other installations in the shafts will include a small service/emergency cage, underground services, ventilation doors, etc.

6.2.4.2.2 Service Areas and Haulage Tunnels. Figure 6.2.9 shows the proposed system of service caverns and access headings for the repository. During the site investigation program described above, a large part of these will be excavated. These excavations will include the cavern at the waste shaft for transferring the waste canisters to the transport vehicle and the operational cavern at the service shaft containing the services, materials handling equipment, etc. The two caverns will be connected by a drift which can be sealed, however, by bulkheads to separate the disposal process from the rest of the underground operations. Other excavations include the access drifts to the silos and the disposal tunnel area. Excavation of the openings described here will most likely be by controlled drill-and-blast methods supported by rubber-tired rock haulage equipment. The current proposal is to have all drill-and-blast excavations such as the service areas, main access headings as well as the silos completed before waste disposal operations start. Excavation of the disposal tunnels is then planned to be done using tunneling machines concurrently with waste disposal.

6.2.4.2.3 Repository Tunnels. Approximately 20 disposal tunnels may be required for the disposal of the high-level waste canisters. The tunnels are to be spaced at roughly 40 m (131 ft) intervals and have a length of about 1,500 m (4,920 ft). In order to preserve the integrity of the rock around the tunnels (and thus the waste package), the tunnels are to be drilled using tunneling machines. Figure 6.2.12 shows a full-face tunneling machine of the type applicable for this. The machine consists of a rotating cutter head which chips or "cuts" the rock as it is forced into the tunnel face by massive hydraulic cylinders. The cuttings fall to the tunnel invert where they are gathered by scoops on the cutter head and loaded via a conveyor behind the machine into mine cars. The full mine cars are taken to the service shaft to be hoisted to the surface and emptied. An installation crew continually extends the railway track, ventilation and other services behind the tunneling machine as it advances. At an average advance rate of about 7 m/day (23 ft/day) it will require about 200 days to complete one tunnel. After completion of one tunnel, the tunneling machine and back-up equipment will be moved to the next location and the process begins again. Since the granite is expected to be stable, it is not anticipated that a lining will be required in the tunnels. Since it is likely that during the excavation of the tunnels some disturbed zones will be encountered where no waste canister will be placed, allowances will be included in the layout for the excavation of additional tunnels to compensate for these zones.

6.2.4.2.4 Repository Silos. About 14 silos are expected to be required for the disposal of the intermediate-level waste canisters with the fuel boxes and medium alpha-radiation wastes encased in bitumen or cement. The silos will consist of a free-standing concrete structure inside a cylindrical, vertical cavern. The height of the silos will be about 52 m (170 ft) and the inner diameter about 10 m (33 ft). The cavern will be excavated using drill-and-blast techniques and the concrete structure may be erected by slipforming. Access drifts will be excavated to the top of the silo and a crane will lower the waste packages into the silo (Figure 6.2.13). The packages inside the concrete silo will be secured with concrete backfill and the gap between the rock and concrete structure backfilled with bentonite granulate.

The total excavation volume for the repository will be approximately 1,100,000 m^3 (1,440,000 yd^3).

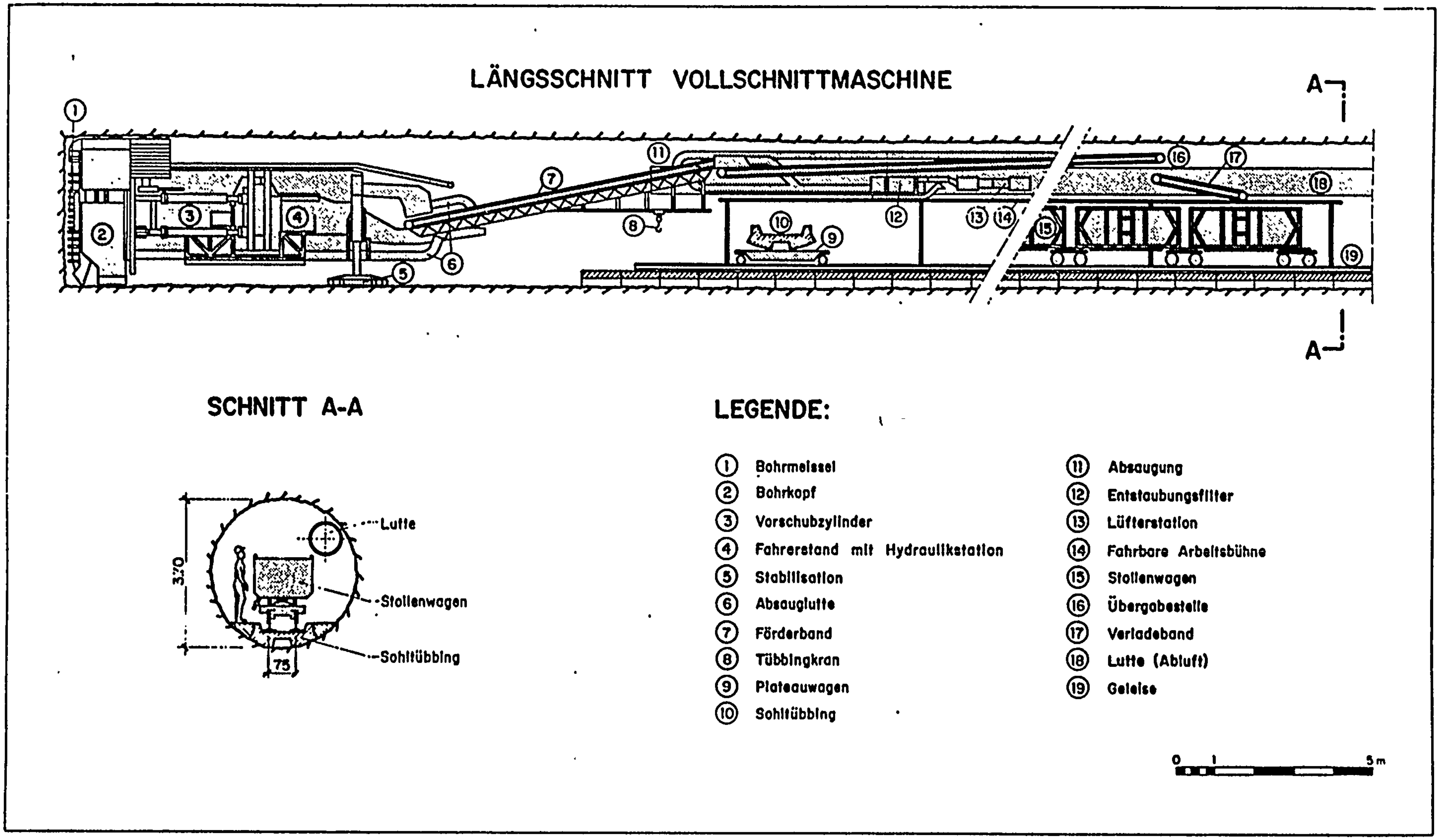

Figure 6.2.12. Full-Face Tunneling Machine

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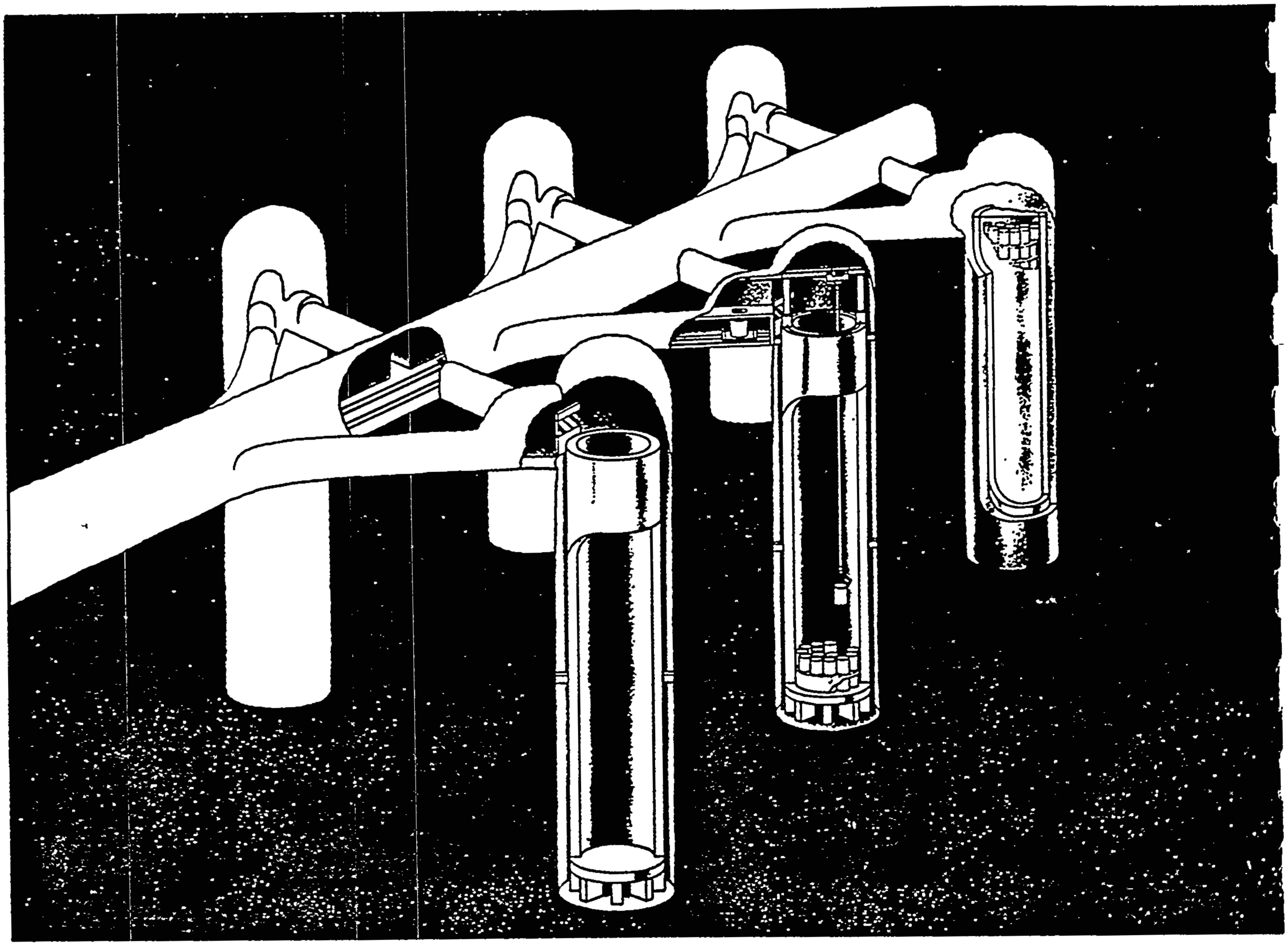

Figure 6.2.13. Repository Silos

6.2.4.3 Cost

The construction costs for the whole facility were estimated to be approximately $500 million.

6.2.5 Waste Emplacement Cycle

This section briefly outlines the waste treatment and packaging concepts derived for the Swiss Project Gewähr studies.

6.2.5.1 Waste Treatment

Figure 6.2.2 shows the overall nuclear waste management strategy in Switzerland. Switzerland has contracts with COGEMA in France and BNFL in the UK to reprocess all the spent fuel produced by the Swiss nuclear power plants after an initial storage period at the reactor. The resulting high-level waste will vitrified with glass-forming additives and allowed to solidify in high-grade steel molds to form compact blocks. These molds will then be shipped back to Switzerland (currently planned to start in 1992) for interim storage for several decades before disposal in an underground repository. Low- and intermediate-level wastes will be conditioned and solidified with cement, bitumen or polymers in steel drums or concrete containers and, after interim storage, disposed of in an underground repository.

6.2.5.2 Waste Transport

Transport of the spent fuel and other nuclear wastes will require the use of specially designed transport containers which can be carried on rail or truck transporters. These containers will be constructed so that the waste package will be completely shielded during transport and be structurally strong enough to maintain this shielding even in case of an accident.

6.2.5.3 Waste Package

Figures 6.2.14 and 6.2.15 show the types of waste canister and containers that are proposed for the various types of wastes.

The high-level wastes will be vitrified in high-grade steel molds. These steel molds will then be encased in a repository canister. The primary functions of this repository canister will be to ensure complete containment of the radioactivity for an initial period of about 1,000 years until the short-lived fission products have decayed and the period of high temperature emission has passed. The canister will also provide for safe handling and transport of the radioactive material during disposal operations. The cast steel canister will have a length of about 2,0 m (6.5 ft) and a diameter of about 0.94 m (3 ft) with a wall thickness of

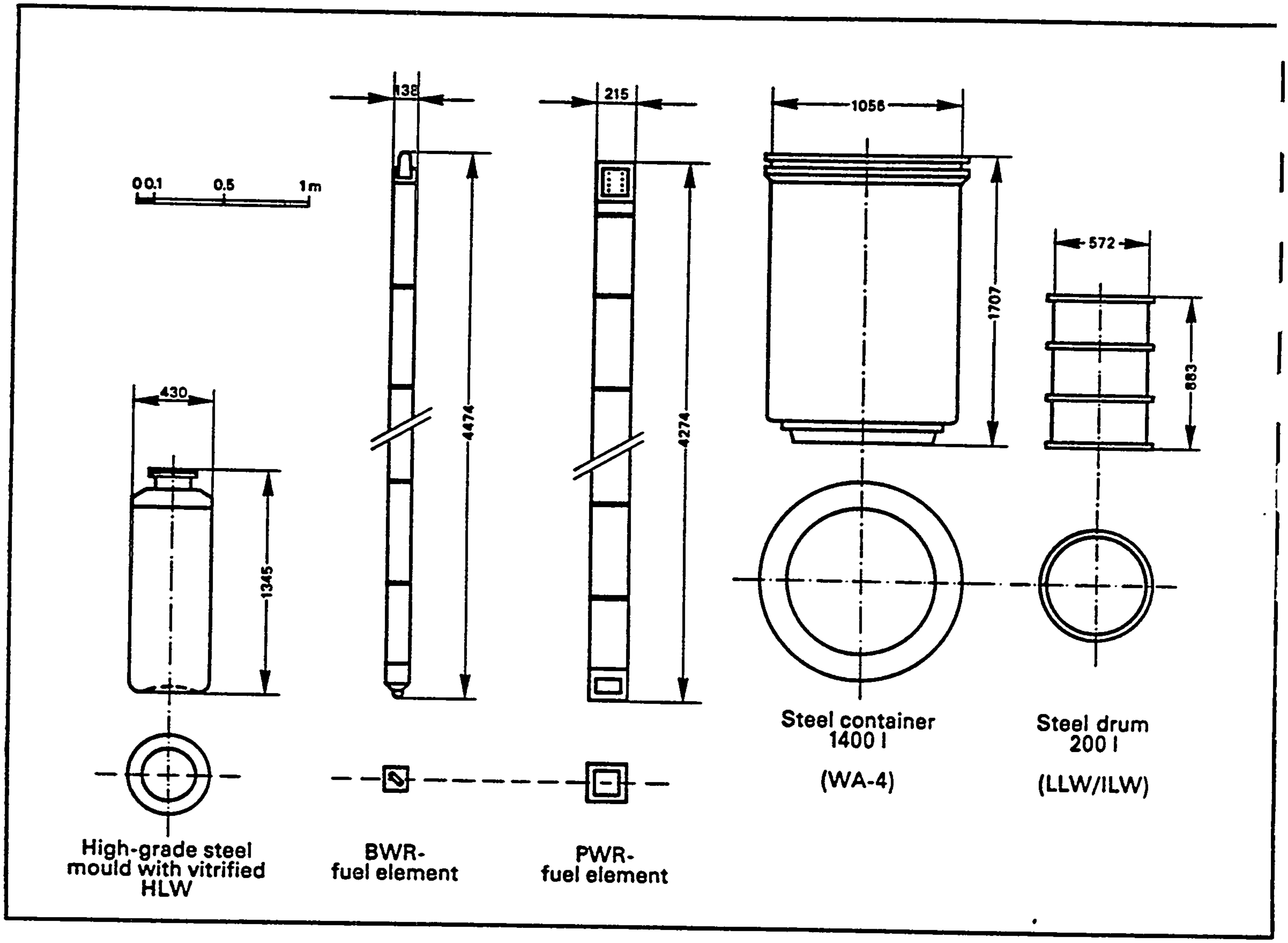

Figure 6.2.14. Waste Containers

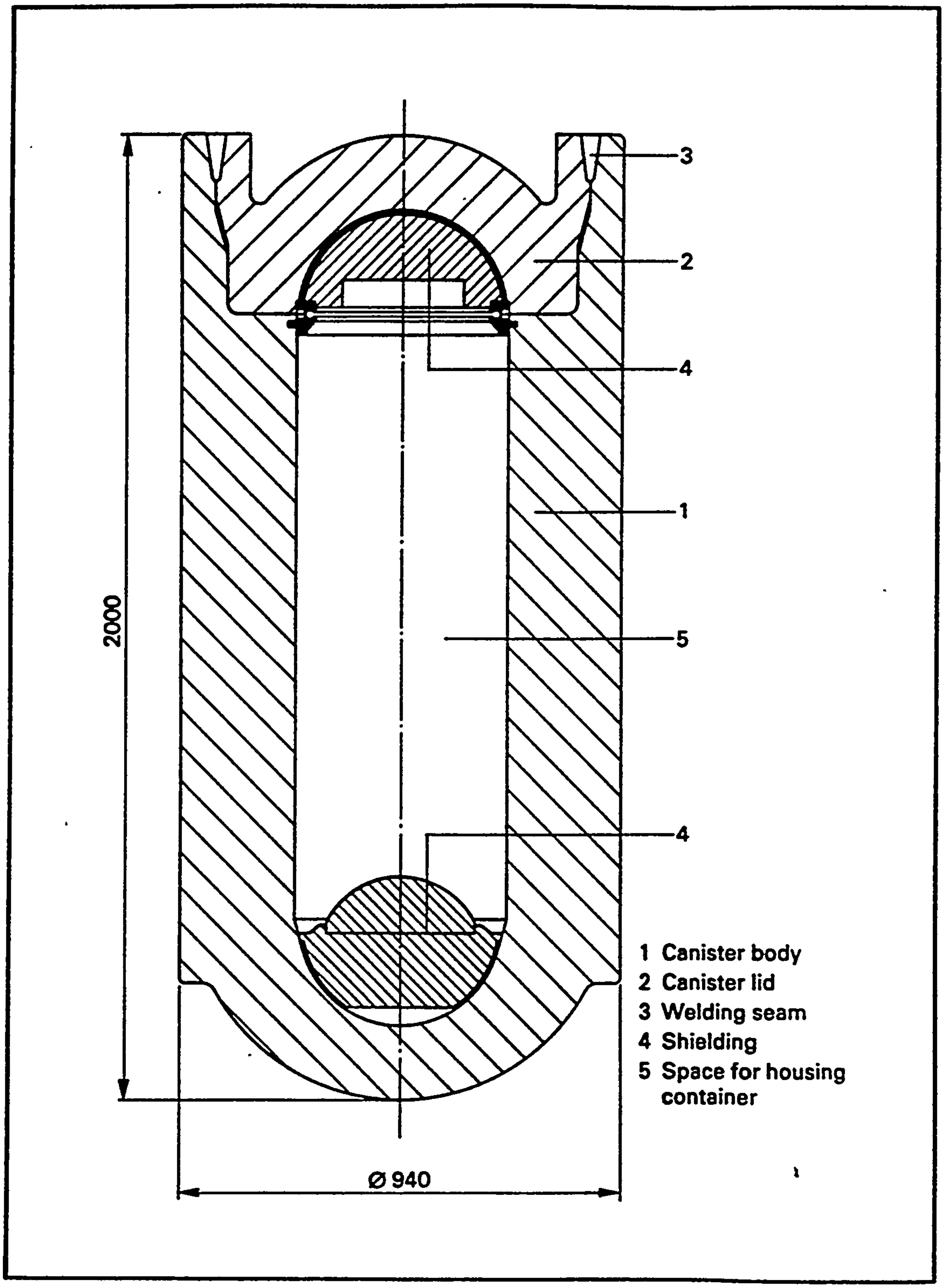

Figure 6.2.15. High-Level Waste Canister

0.15 m (6 in). The total weight including the steel waste mold will be roughly 8.5 tons. The lid will be welded to the canister body to ensure a leak-proof seal (Figure 6.2.15).

The low- and intermediate-level wastes will be solidified in bitumen or cement in steel containers and drums (Figure 6.2.14).

6.2.5.4 Underground Waste Emplacement Sequence

6.2.5.4.1 Tunnel Disposal. The high-level waste in the steel molds will arrive by rail or truck in the shielded transport containers at the surface reception facilities of the repository. There they will be encased in the final repository cast steel canisters plus a transport shield will be fitted before they are lowered underground. At the repository level, the canisters will be transported by rail via the separate disposal cavern and waste transport headings to the correct disposal tunnel (Figure 6.2.18). The repository layout will be such that access to the disposal operation and to the excavation operations will be separated from one another (Figure 6.2.9). Figures 6.2.16 to 6.2.19 show the emplacement method in the disposal tunnels. The canisters will be transported from the shaft on a shielded transport wagon to the disposal location. Disposal will operate in a retreat mode, where first the rail track and concrete segments are cleared from a section of the disposal tunnel invert (Figure 6.2.19). This will allow filling the lower half of the tunnel with the compacted bentonite blocks. When the canister arrives, a bridge crane will place the canister into the prepared slot and the backfill around the canister with bentonite blocks will be completed before preparing the next section of tunnel. The canister spacing will be about 5 m (16.4 ft) based on the allowable heat load created by the waste in the repository. This spacing may, however, increase in some places due to minor disturbed zones which may be encountered. This procedure will be repeated until all tunnels are full.

6.2.5.4.2 Silo Disposal. Unlike the high-level waste canisters, which must be separated because of their high heat output, the intermediate-level waste canisters can be stacked in close proximity. The silos will therefore be constructed to accommodate the alpha-containing intermediate-level wastes. The waste canisters will arrive by rail at the top of a silo and a crane will stack the canisters inside the concrete silo. Figure 6.2.20 shows the transport method for the cement and bitumen encased intermediate-level wastes. The canisters will be stabilized using special concrete backfill and the space between the concrete silo and the excavation will be backfilled with bentonite granulate. Figures 6.2.13 and 6.2.21 show the silo disposal method.

6.2.5.5 Emplacement Schedule

The start of high-level waste disposal in the underground repository is currently scheduled for around the year 2020, based on an interim storage period of about 40 years with projected waste production rates and a nuclear

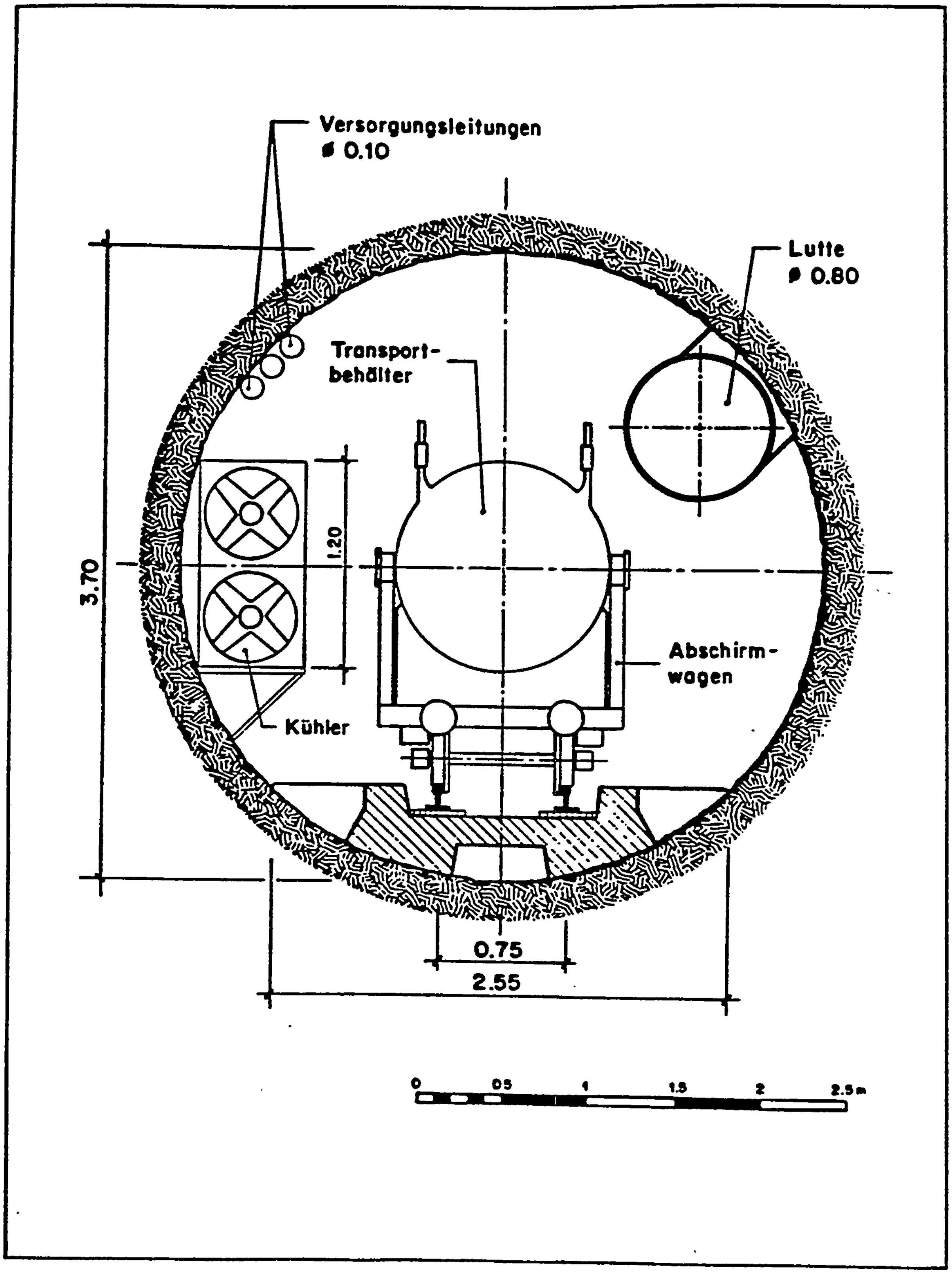

Figure 6.2.16. High-Level Waste Disposal Tunnel

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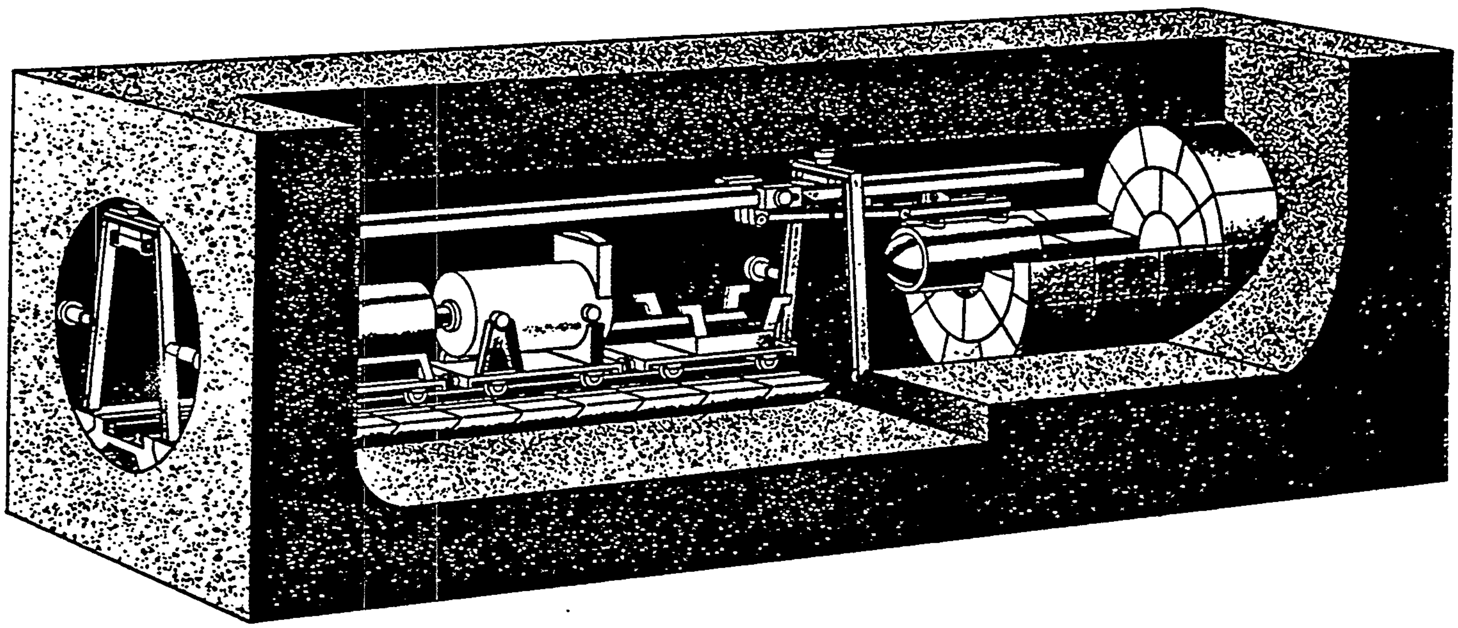

Figure 6.2.17. High-Level Waste Emplacement

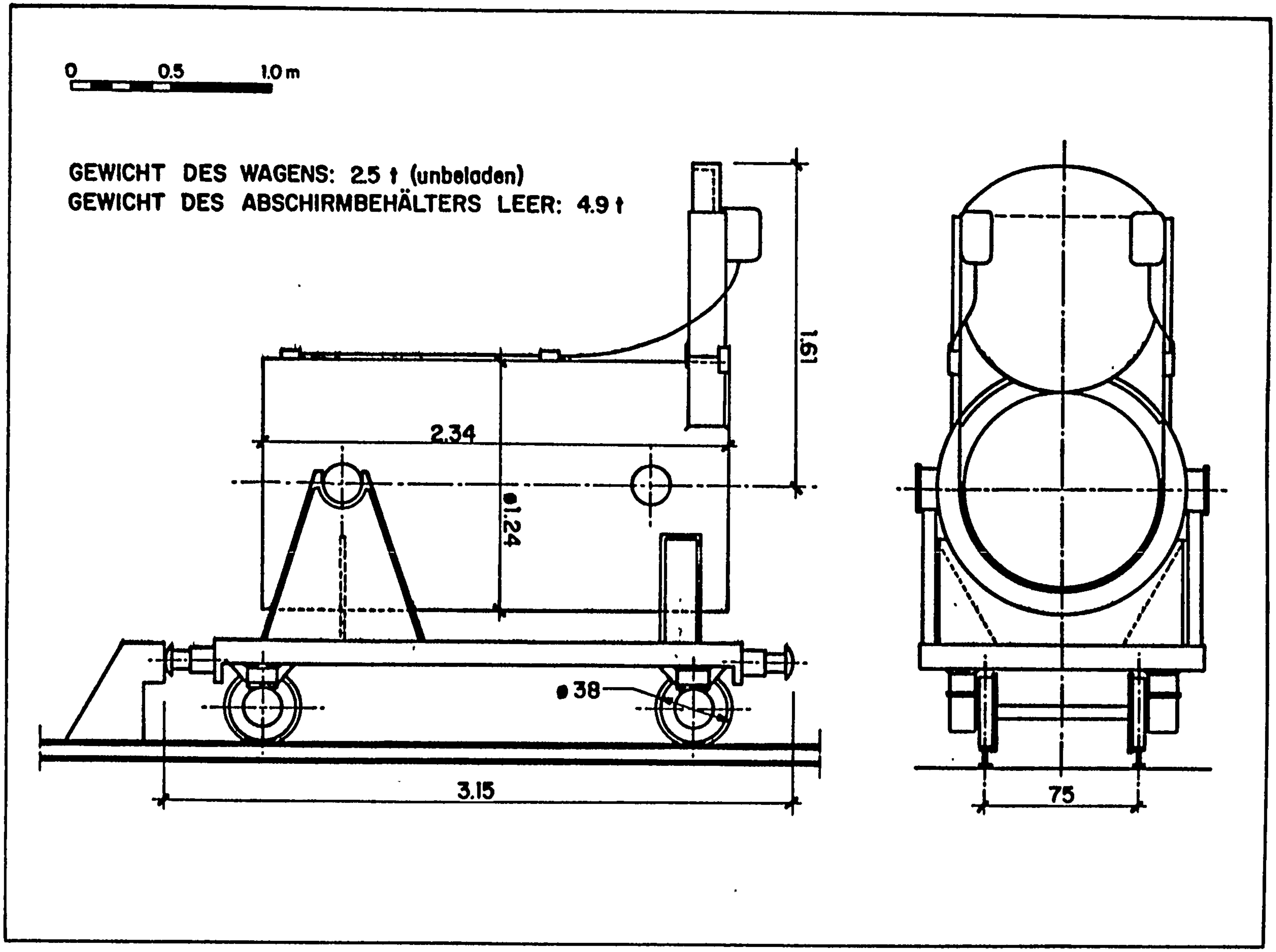

Figure 6.2.18. High-Level Waste Transport Shield

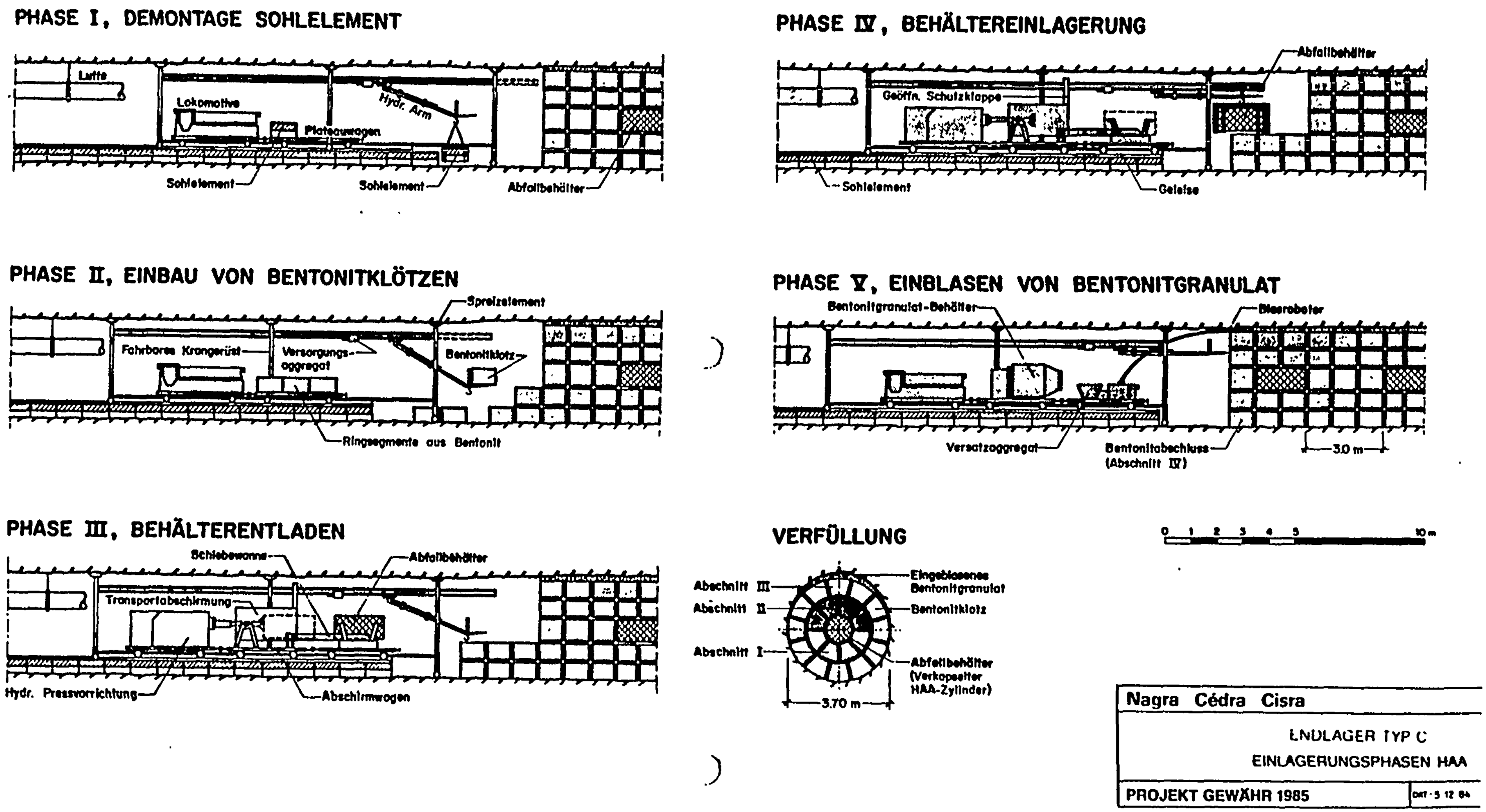

Figure 6.2.19. High-Level Waste Emplacement Sequence

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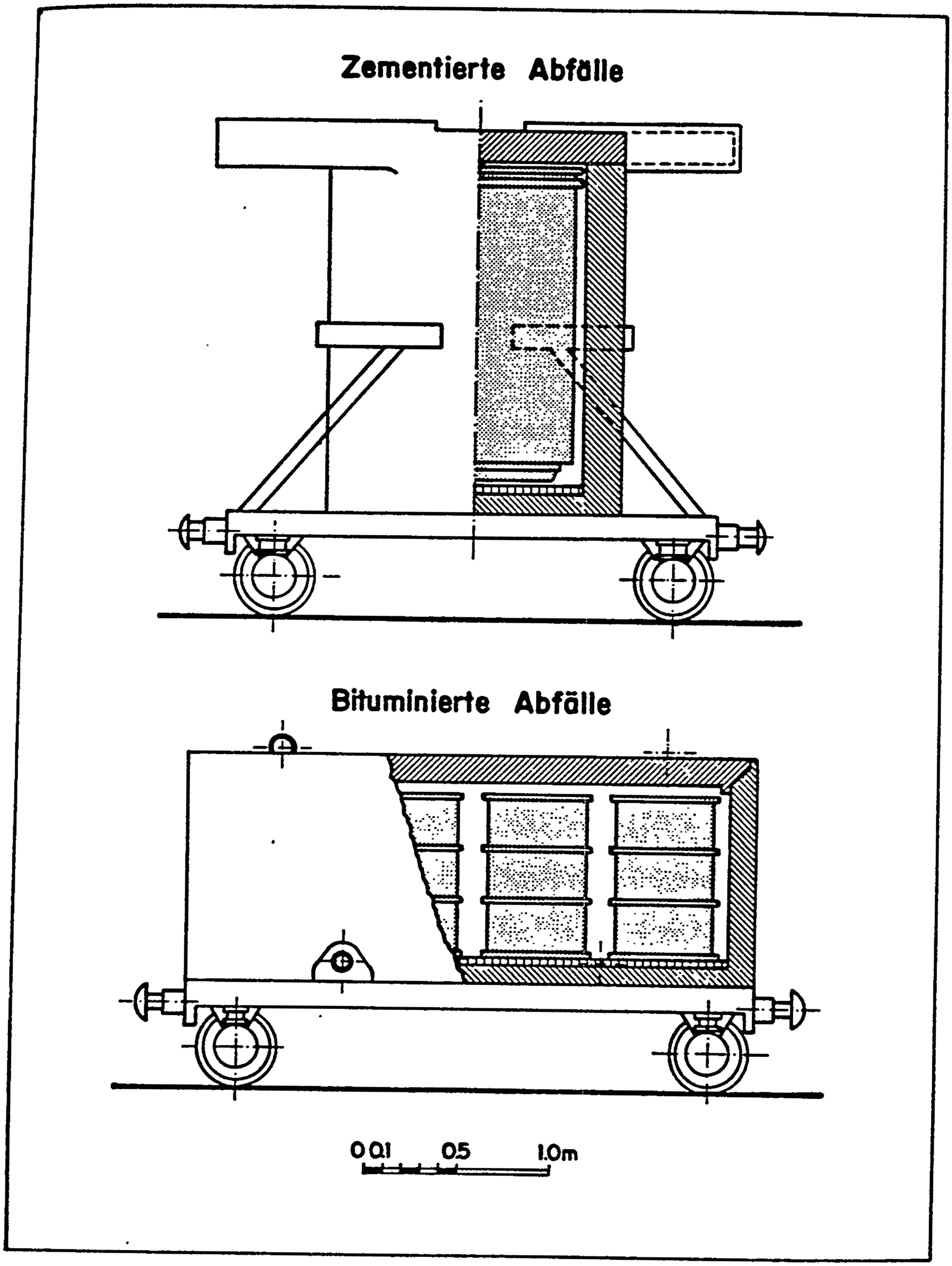

Figure 6.2.20. ILW Transport Shield

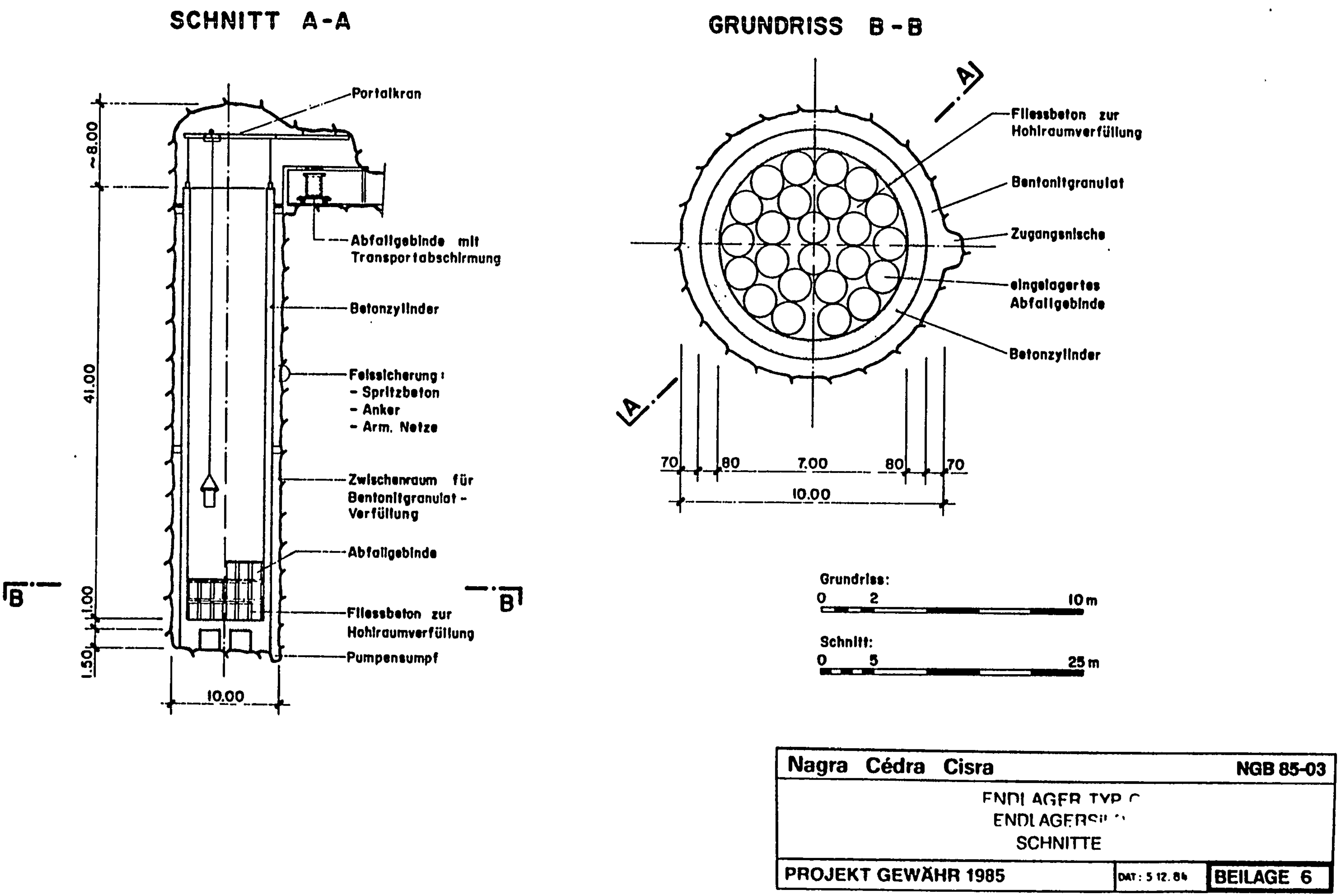

Figure 6.2.21. Silo Waste Disposal

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plant operating period of about 40 years. With the last reactor planned to start operation around the year 2000, the operational period of the repository will thus last until approximately 2080.

6.2.6 Buffer and Sealing Materials

The main buffer and backfill materials being considered for the repository are bentonite, concrete and granite blocks.

In the high-level waste disposal tunnels, the canister will be surrounded completely by compressed blocks of bentonite. High density pure bentonite has a number of favorable properties as a backfill material:

1. Low hydraulic conductivity of the compressed material inhibit ground-water inflow and retard nuclide transport from the canisters.
2. The bentonite swells as it contacts water, thereby self-sealing cavities in the bentonite as well as penetrating fine fissures in the surrounding rock.
3. The plastic properties of the bentonite compensate for rock movement and prevent formation of fissures in the backfill mass.
4. The bentonite has good retardation characteristics.
5. Good thermal conductivity ensures that the heat generated by the waste is conducted away from the waste canister and thus avoid excessive heat build-up.
6. The loading capacity of the bentonite is sufficient to keep the steel canisters located in the center of the disposal tunnel.

The bentonite backfill will be placed around the steel canisters in the tunnel in the form of high-compressed block as shown in Figure 6.2.19.

In the silos for the alpha-containing intermediate-level wastes, the backfill immediately around the canisters in the concrete structure will consist of concrete. This concrete will act not only as an immobilization matrix but also as the structural and sealing material. The space between the concrete structure and the rock wall will also be backfilled with a bentonite granulate for the reasons described above.

Backfilling and sealing of the remaining access tunnels, caverns and shaft after repository operations have been completed will most likely be with a bentonite/sand mixture. Figures 6.2.22 and 6.2.23 show the sealing method proposed for the shafts and tunnels.

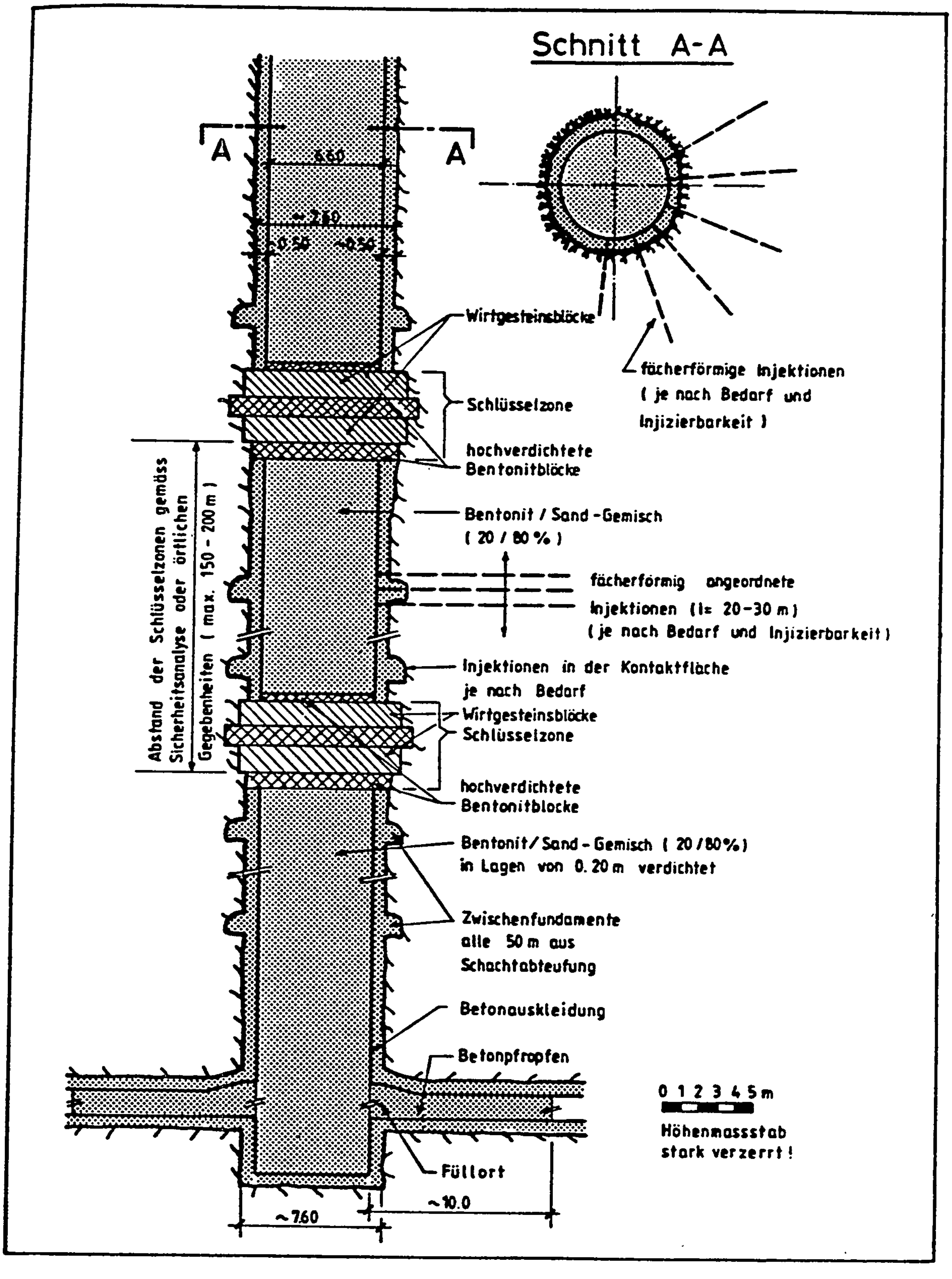

Figure 6.2.22. Shaft Sealing

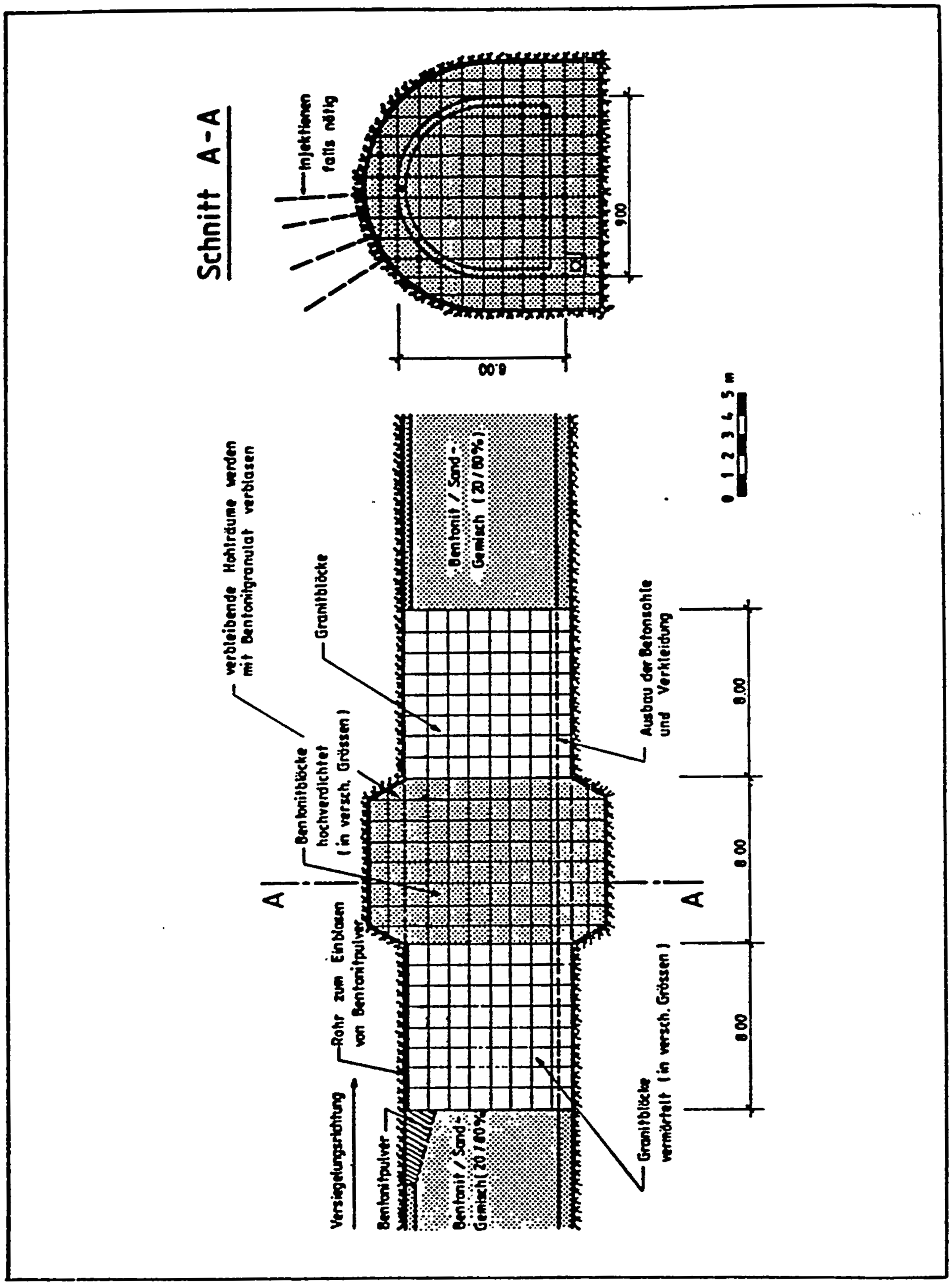

Figure 6.2.23. Tunnel Sealing

6.2.7 Summary

In summary, the basis for the repository concept described above is disposal of reprocessing wastes encased in cast steel canisters placed in horizontal tunnels which are then backfilled with bentonite. The depth of the repository is assumed to be about 1,200 m (3,930 ft) in crystalline rock with overlying sedimentary formations. The radioactive waste will be isolated using a series of engineered and natural barriers. Total high-level waste volume was assumed to be about 1,200 m^3 (1,570 yd^3) and the alpha-containing waste total was estimated to be about 10,000 m^3 (13,100 yd^3). The following sections discuss some of the advantages and potential difficulties associated with this concept and how it might relate to the U.S. repository program.

6.2.7.1 Advantages

1. In addition to the natural barriers, the nuclear wastes will also be encased in several engineered barriers which will add to the overall effectiveness of the isolation system. The use of bentonite as a buffer material is also being considered by other countries in their nuclear waste disposal concepts.

2. Extensive underground construction experience exists in Switzerland in the crystalline rocks from the construction of tunnels, caverns, and other civil underground construction projects and therefore a firm basis exists for the technical evaluation of repository construction methods.

3. Flexibility in the repository layout allows adjustment of the location of the tunnels to avoid or isolate disturbed zones.

4. The application of mechanized tunneling methods reduces the damage to the surrounding rock formation.

5. Retrieval of the waste packages will be possible.

6. Interim storage of the wastes before final disposal in the underground repository will mean that the heat and radioactivity release will be greatly reduced, thereby enhancing the actual disposal operation and long-term safety aspect.

7. By reprocessing the wastes, the amount of high-level waste is greatly reduced over the spent-fuel disposal option.

8. Combining repositories for different wastes has economic and operational advantages.

6.2.7.2 Potential Problems

A major drawback of the deep geologic repository concepts is the limited knowledge of the long-term behavior of the natural and engineered barrier materials and the impact of the construction and disposal methods and the heat and radioactivity on the characteristics of the barriers. Major investigation projects are currently underway in many countries in the world to resolve some of these issues.

6.2.7.3 Summary

Since the crystalline rocks are one of the formations being considered to host a high-level waste repository in the U.S. program, the Swiss work is of great interest. However, the amount of waste in Switzerland will be around 15% of the volume in the U.S. and thus the repository will have to be correspondingly larger. Disposal in a tunnel or drift is one of the options under consideration and thus the Swiss concept may have direct application in the U.S. program.

6.2.8 Optional Direct Disposal of Spent Fuel

If the option to dispose of the spent fuel without reprocessing is selected, the facilities and installations of the repository will be similar to the high-level waste repository with the following exceptions:

1. The total volume of the spent fuel elements requiring disposal will be about three times the high-level waste volume (approximately 3,700 m^3 or 4,940 yd^3 of spent fuel or about 8,000 MT or 8,800 tons).

2. Before disposal, the fuel elements will be enclosed in copper canisters in a similar manner to that described for the Swedish disposal program (see Section 6.1).

3. The total weight of the copper emplacement canister with the spent fuel will be about 24 tons and therefore transport in the shaft with the roughly 26 ton capacity hoist will be without the transport shielding. Surface and underground transport will, however, be shielded as proposed for the high-level wastes.

4. The emplacement operation in the tunnels will be the same as for the high-level wastes but with the associated increase in the capacity handling equipment.

5. The overall repository size will be about the same as the high-level waste repository. However, the higher heat output of the spent fuel will require greater tunnel lengths (larger spacing between canisters) but the need for the intermediate-level waste silos will be reduced or even eliminated.

6.2.9 The Low- and Intermediate-Level Concepts

This section briefly outlines the low- and intermediate-level repository concept derived as part of the Project Gewähr studies.

The concept envisioned was a separate, Type B underground repository for the low- and intermediate-level operational short-lived wastes not assigned to the Type C repository silos. Potential host rock formations include clay, marl, anhydrite, and crystalline rocks. In the Type B repository, the wastes will also be isolated from the environment by both natural and engineered barriers as shown in Figure 6.2.24. The engineered barriers will consist of:

- Waste solidification matrix such as cement, bitumen or polymers
- Waste containers will be encased in concrete casks which are then backfilled with cement
- The concrete containers will be placed in concrete lined tunnels which will also be backfilled with concrete.

Figure 6.2.25 shows the proposed repository layout. Access to the repository will be via a connecting tunnel from an underground reception area where the wastes will be temporarily stored and prepared for disposal. The reception area, accessible from the surface via an adit, will also include other technical installations such as the ventilation equipment and backfill preparation facilities. The waste will be transported to the reception area by road vehicle through the access heading. In the reception area, the drums will be encased in the concrete containers (Figure 6.2.26). About 35 waste drums will be encased in one concrete container. The conditioned repository containers will be transported to the disposal cavern by a conveyor. In the disposal caverns the canisters will be handled by remotely controlled equipment (Figure 6.2.27). The space around the canisters will finally be backfilled with concrete. The last operation when the repository caverns are full, will be backfilling and sealing of the remaining repository access headings and caverns.

The following summarizes the main aspects of the underground repository configuration:

1. The repository will be under about 700-1,200 m (2,300 to 3,940 ft) of overburden.
2. The total volume of waste will be about 200,000 m^3 (262,000 yd^3).
3. The repository container dimensions will be about 4,8 m x 2.2 m x 2 m (15.7 ft x 7 ft 6.5 ft).
4. The access tunnels to the reception area will be about 600 m (1,960 ft) in length and about 8.7 m x 6.5 m (28.5 ft x 21.3 ft) in section. Excavation will probably be by drill and blast.

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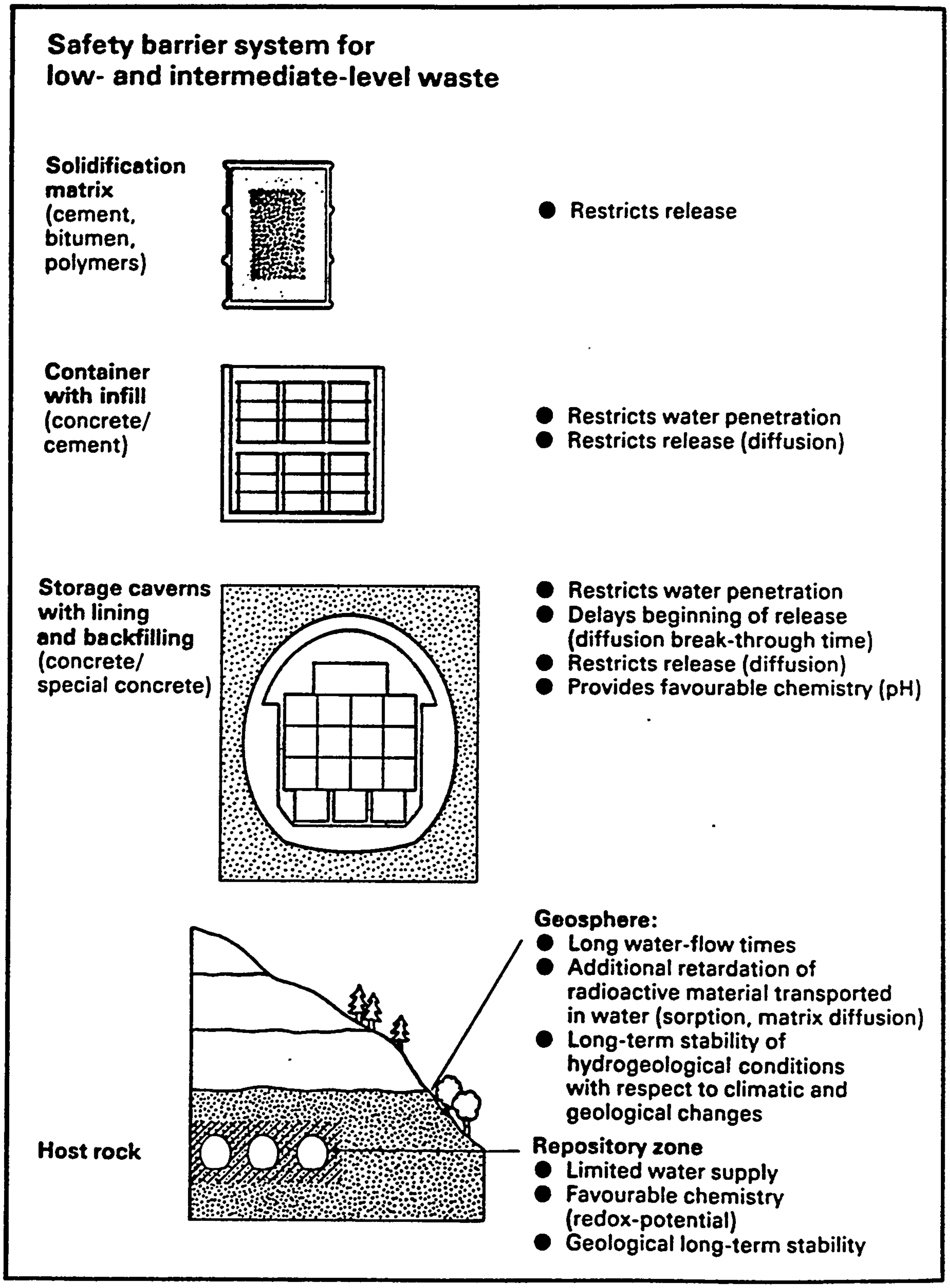

Figure 6.2.24. Type B Repository Barriers

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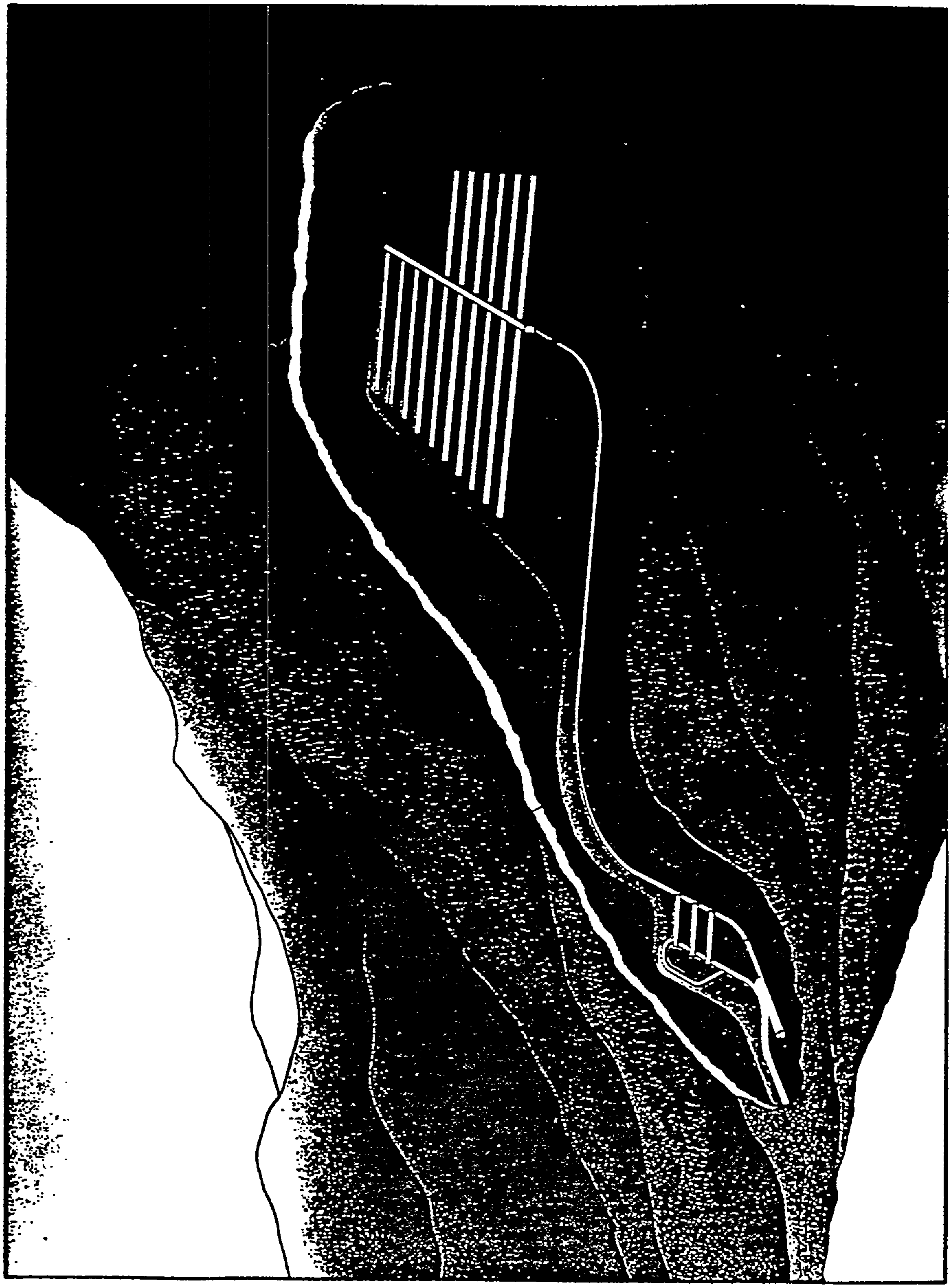

Figure 6.2.25. Type B Repository Layout

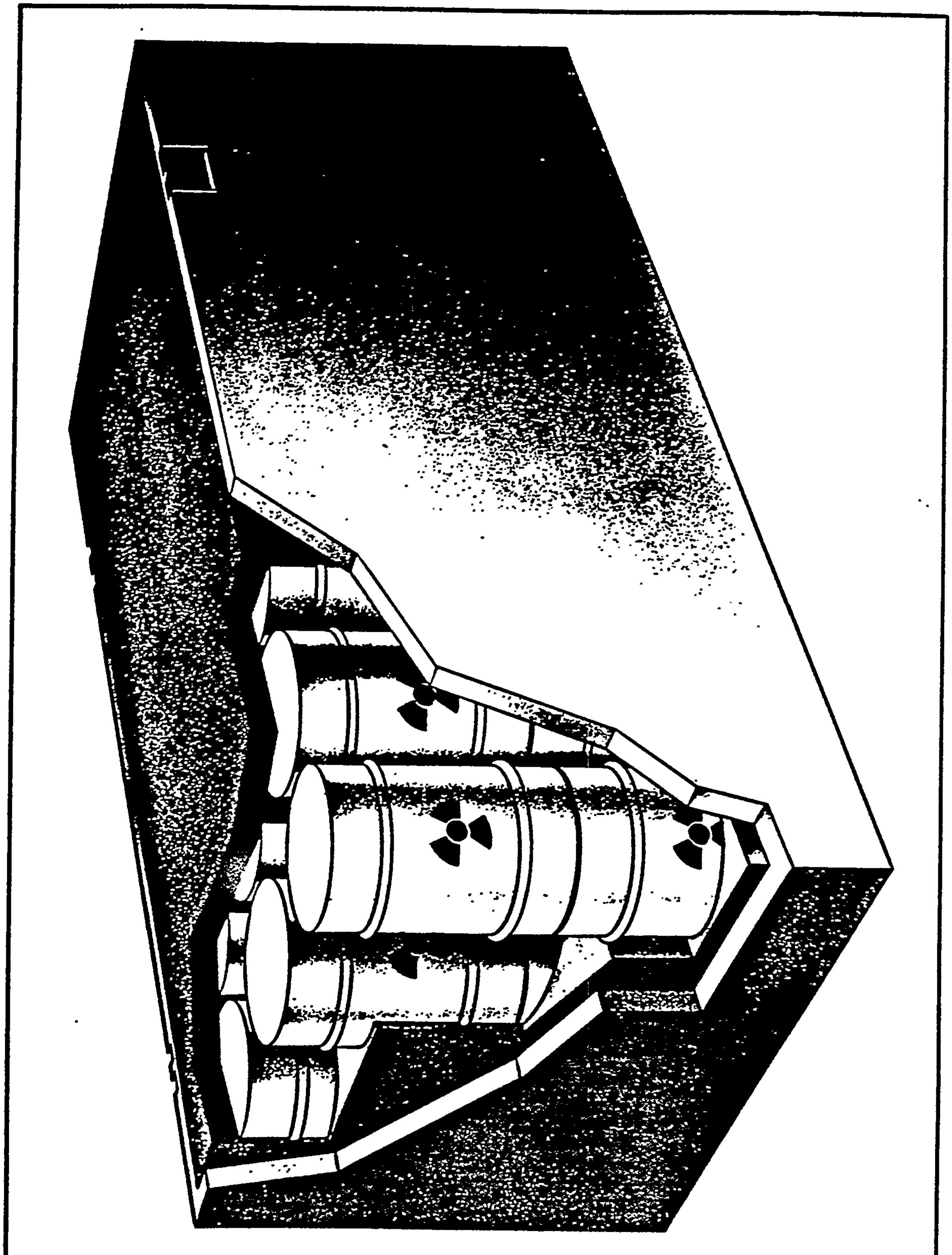

Figure 6.2.26. Type B Repository Waste Containers

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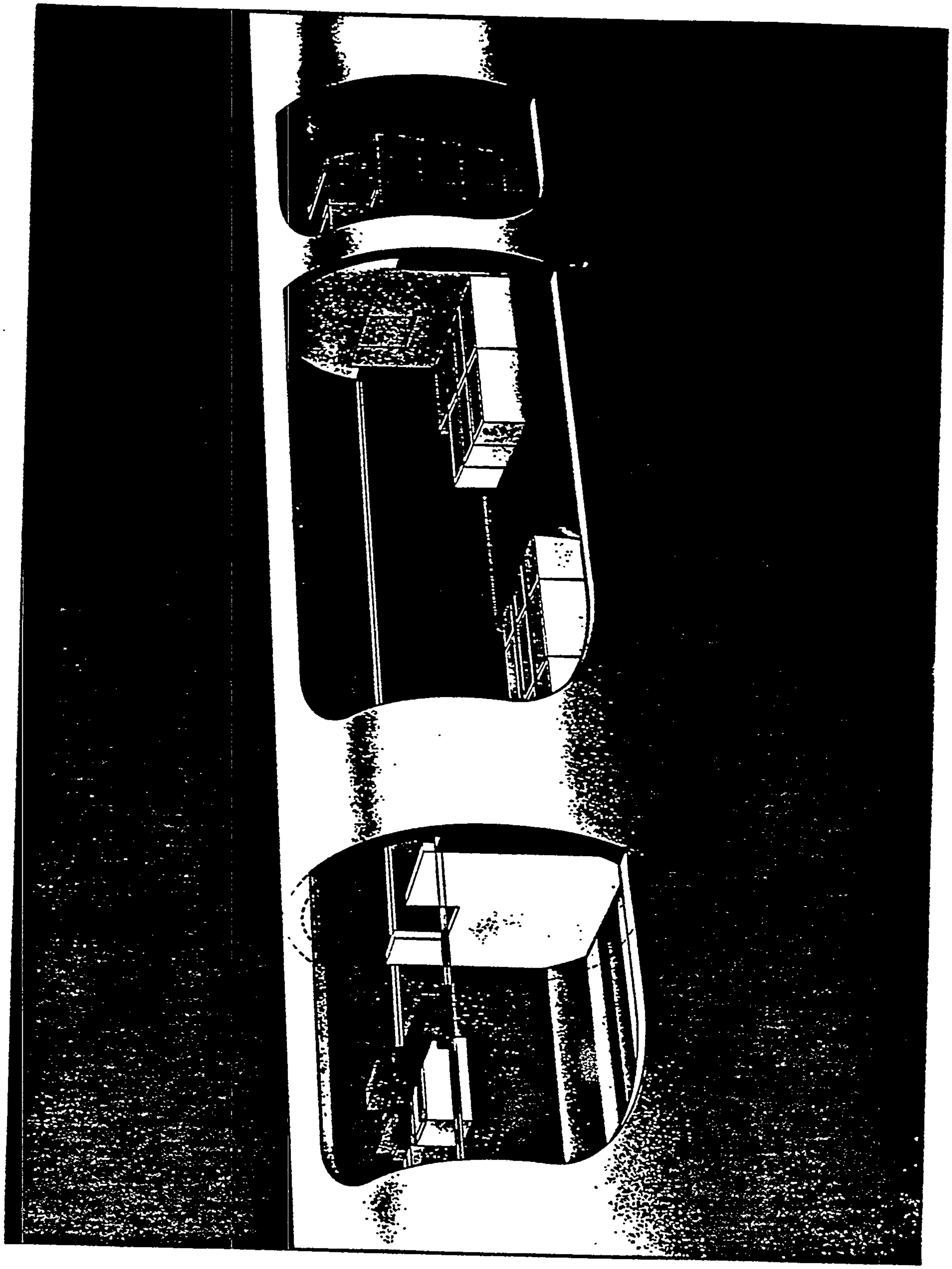

Figure 6.2.27. Type B Repository Waste Emplacement

5. The three reception area caverns will be spaced at roughly 30 m (100 ft) intervals and be 11.5 m wide by 15.4 m high by 53 m long (38 ft x 50 ft x 170 ft). Excavation will probably be by drill and blast.

6. The connecting tunnel will be about 1,600 m (5,250 ft) long with a diameter of 5.2 m (17 ft). Excavation will probably be by tunneling machine.

7. The 16 disposal caverns will be between 210 and 360 m (690 to 1,180 ft) long with a cross-section of about 160 to 180 m^2 (1,720 to 1,940 ft^2). The interval will be about 70 m (230 ft) and the total length will be about 4,660 m (15,280 ft). The caverns will be excavated using partial-face tunneling machines (roadheaders) and lined with concrete about 60 to 90 cm (2-3 ft) thick. See Figure 6.2.28.

8. The total repository excavation volume will be about 1,000,000 m^3 (1,310,000 yd^3) at a construction cost of about $220 million. Operation time will be about 60 years.

6.3 THE SWEDISH WP-CAVE CONCEPT

This chapter outlines the Swedish WP-Cave Concept for the storage and final disposal of high-level nuclear waste. The concept has been under development since about 1975 and was initiated as an alternative idea for the underground isolation of high-level wastes. The descriptions in this chapter are based on the SKN (National Board for Spent Nuclear Fuel) report published in 1985 which details the concept. The WP-Cave Concept is substantially different from the reference KBS III repository described in Section 6.1 and is distinguished by several features:

- Dry, intermediate storage of spent fuel which can be converted to a final repository

- Storage in a compact configuration in an isolated body of rock

- Isolation of the rock body by a man-made, bentonite-sand barrier of low hydraulic conductivity

- Enclosure of the whole system in a hydraulic cage to divert the ground-water flow and reduce the hydraulic gradient across the storage area.

The following sections will describe these features in more detail.

The Swedish nuclear power program and radioactive waste disposal strategy was described in Section 6.1 and will not be repeated here.

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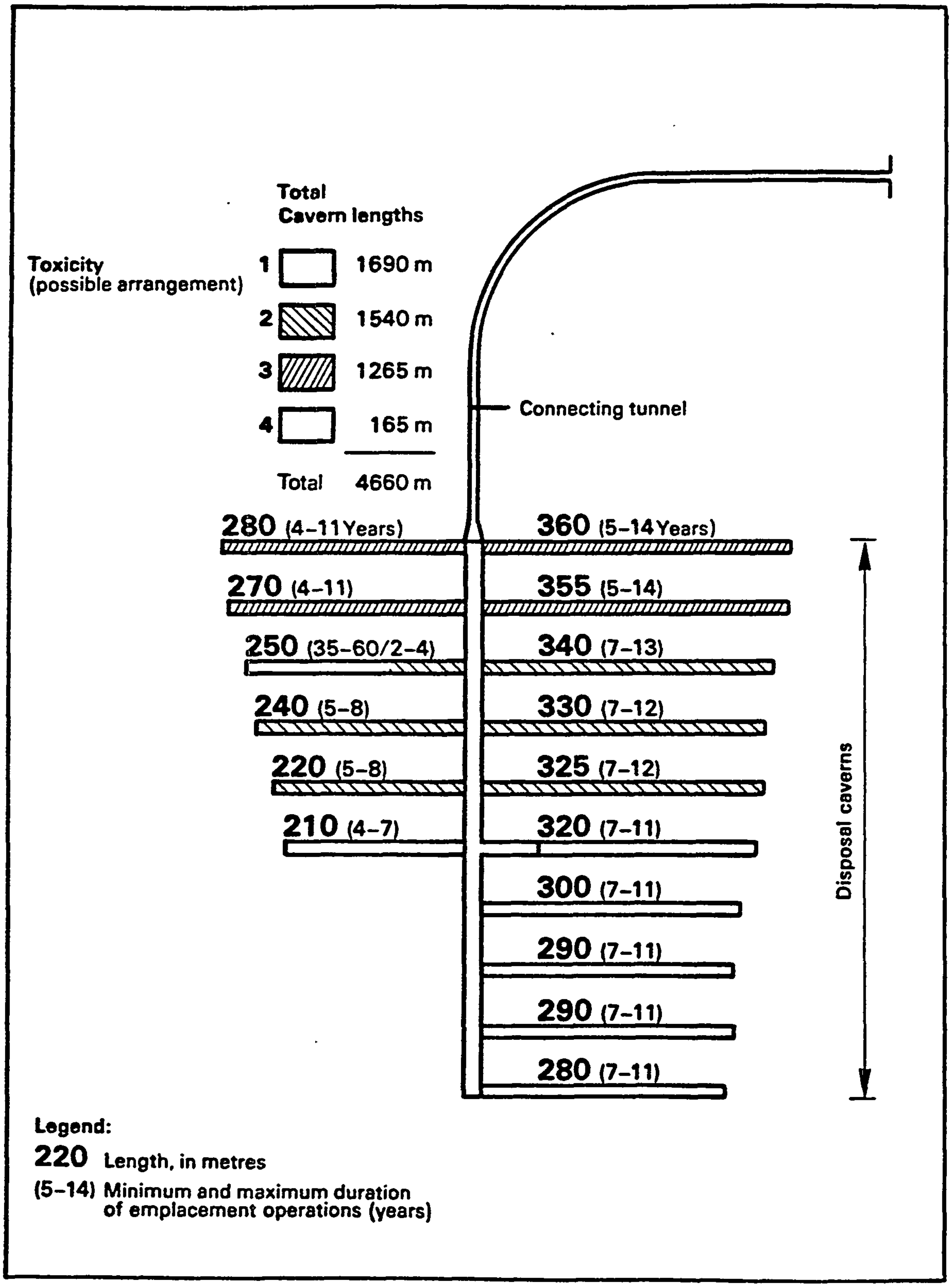

Figure 6.2.28. Type Repository Tunnel Layout

6.3.1 Geographic Location

The areas that have been investigated as potential repository locations for the reference repository were discussed in Section 6.1 and it is assumed that the same areas would also be applicable for the WP-Cave Concept.

6.3.2 Summary of Geologic Setting

A summary of some potential geologic repository environments was given in Section 6.1. The WP-Cave Concept is based on flexible, engineered barriers which can be adapted to various rock qualities. One main requirement is, however, that the rock strength and in situ stress conditions must allow for construction of the inner storage area and surrounding bentonite-sand barrier. Also, the ground-water flow across the repository after excavation must be sufficiently low to keep the radionuclide transport from the WP-Cave to the accessible environment within the required limits. Initial investigations of the crystalline formations in Sweden have shown that it should be possible to meet these requirements.

6.3.3 Repository Concept Description

The Wp-Cave is designed with an oval configuration as shown in Figure 6.3.1. The overall dimensions will be determined by the radioactive waste storage capacity. A WP-Cave designed to hold about 1,600 tons of spent fuel, will have an estimated diameter of about 110 m (360 ft) and an overall height of about 250 m (820 ft). It will need to be located at sufficient depth to exclude the possible effects of glaciation, surface explosions and human intrusion. It was estimated that a depth to the top of the repository of about 200-250 m (650-820 ft) would be sufficient to meet these requirements.

The radioactive waste storage areas are excavated in a rock body isolated by a low water conductivity bentonite-sand barrier with a width of about 5 m (16.4 ft). The waste canister storage areas within this body of rock will consist of a series of inclined openings accessible by several shafts for emplacement and ventilation (Figure 6.3.2). At some distance from the bentonite barrier, a hydraulic cage will be constructed whose purpose is to divert the ground-water flow around the waste storage area and bentonite barrier. In this way, the waste containers can be stored in relatively dry conditions both in the interim and after repository closure. During the proposed interim storage period of about 100 years, the waste containers will be cooled by circulating air. Heat exchanger will be used to cool the air and keep the repository temperature at around 60°C (140°F). Air cooling will be required during the interim storage period since the compact waste canister storage layout means that the heat produced cannot all be dissipated in the rock mass. Once the heat production has reduced sufficiently, the repository can be sealed by backfilling all access shafts and drifts. The Wp-Cave is then left for the ground water conditions to return to their natural condition.

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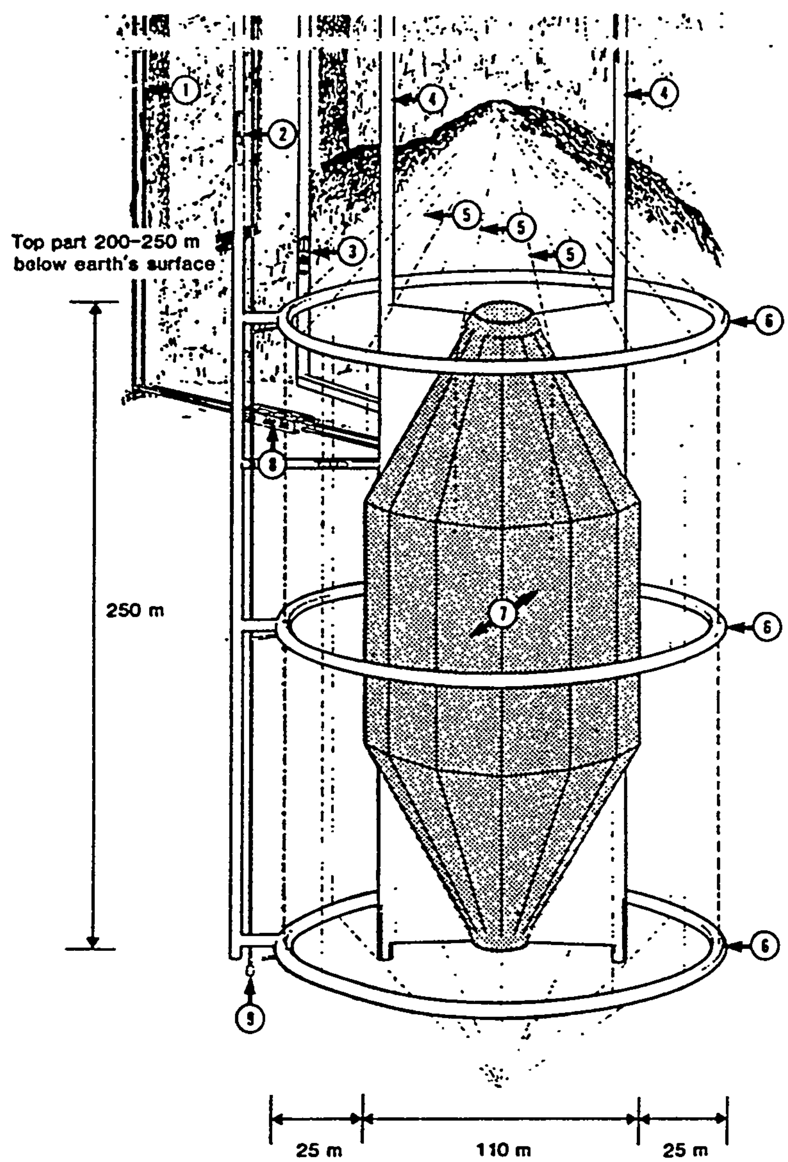

Figure 6.3.1. WP-Cave Design

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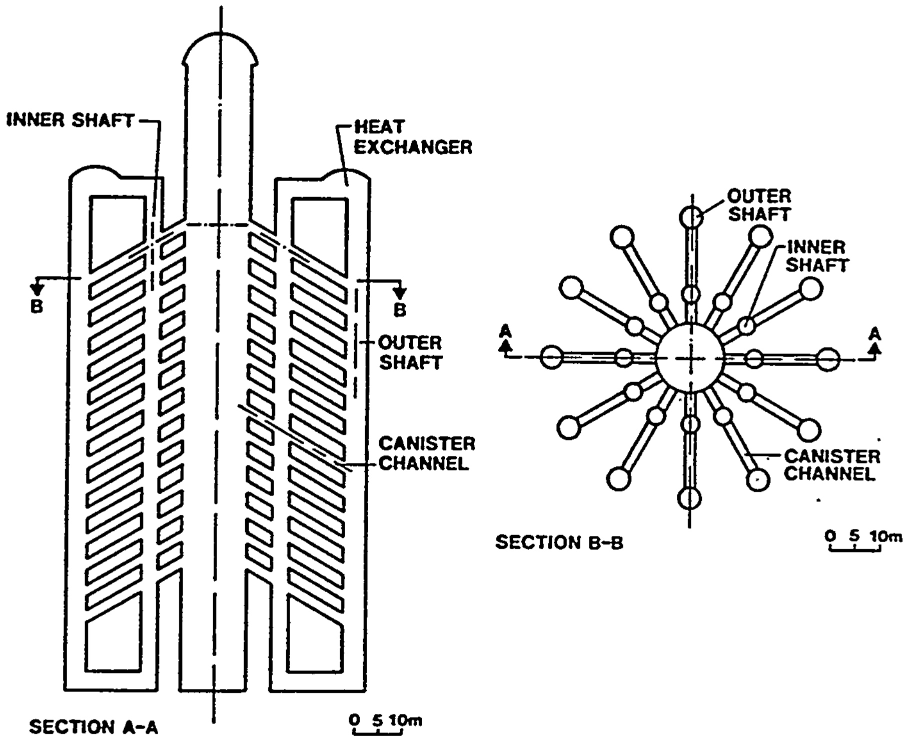

Figure 6.3.2. Waste Canister Storage Area

The WP-Cave is designed for the storage of even fairly young spent fuel on the order of 3-5 years old and thus would probably eliminate the need for an interim surface storage facility. However, even the relatively small Swedish waste disposal program with about 7,800 MT (8,580 tons) of spent fuel (when compared to the US) will likely require several of these WP-Caves since constructional limits will probably not be far above the proposed 1,600 MT (1,760 tons) of spent fuel to be stored in the facility described in this chapter.

6.3.3.1 Surface Facilities

The underground facility will be accessible from the surface via several shafts and drifts. The surface facilities are likely to be similar to those described for the KBS-III repository concept in Section 6.1. These will include:

- Shaft houses and headframes
- Hoisting systems
- Ventilation systems
- Mine service systems
- Waste receiving and preparation facilities
- Offices, work shops, warehouses, change rooms
- Security services
- Rock storage
- Backfill storage and preparation areas.

6.3.3.2 Underground Configuration

Figures 6.3.2 to 6.3.6 show the configuration of the WP-Cave repository concept. The top of the repository will be roughly 200-250 m (650-820 ft) below surface. The WP-Cave will have an overall height of about 250 m (820 ft) and a diameter of 160 m (525 ft) to the hydraulic cage. Access to the facility will be via shafts and drifts (Figure 6.3.4). The shafts will include several ventilation shafts as well as access shafts for construction operations and waste canister transport.

6.3.3.2.1 Waste Storage Area. The actual radioactive waste storage area (Figures 6.3.2 and 6.3.3) will consist of an approximately 14 m (46 ft) diameter central storage access shaft with a ring of outer and inner ventilation shafts. The canister storage channels will be a series of inclined (~ 30°), roughly 1.5 m (5 ft) diameter boreholes radiating from the

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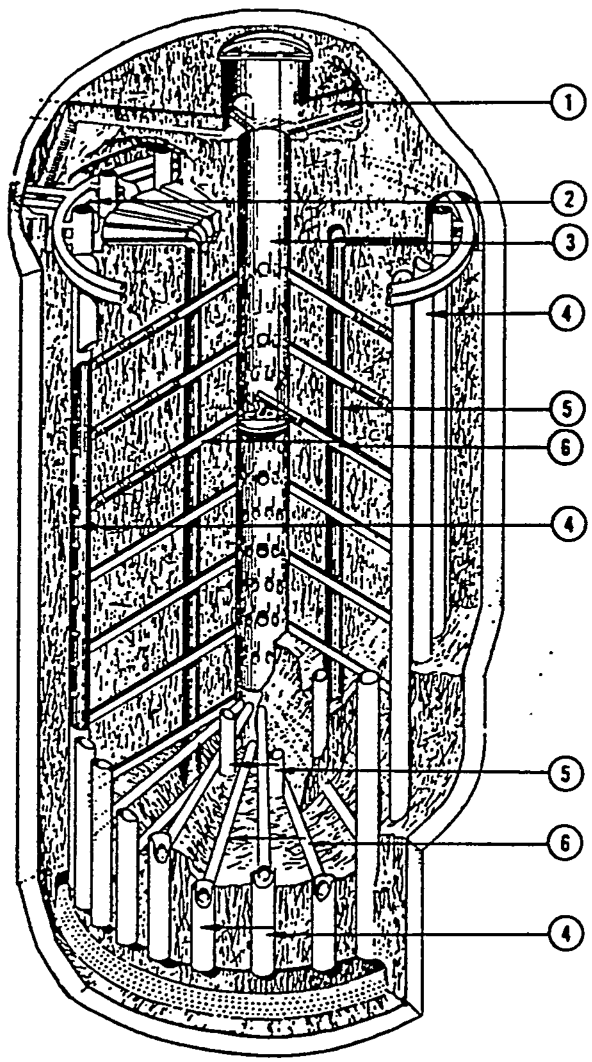

1. Bulk storage room
2. Heat exchange
3. Central storage shaft
4. Outer ventilation shaft
5. Inner ventilation shaft
6. Canister channel

Figure 6.3.3. Artist's View of Storage Area

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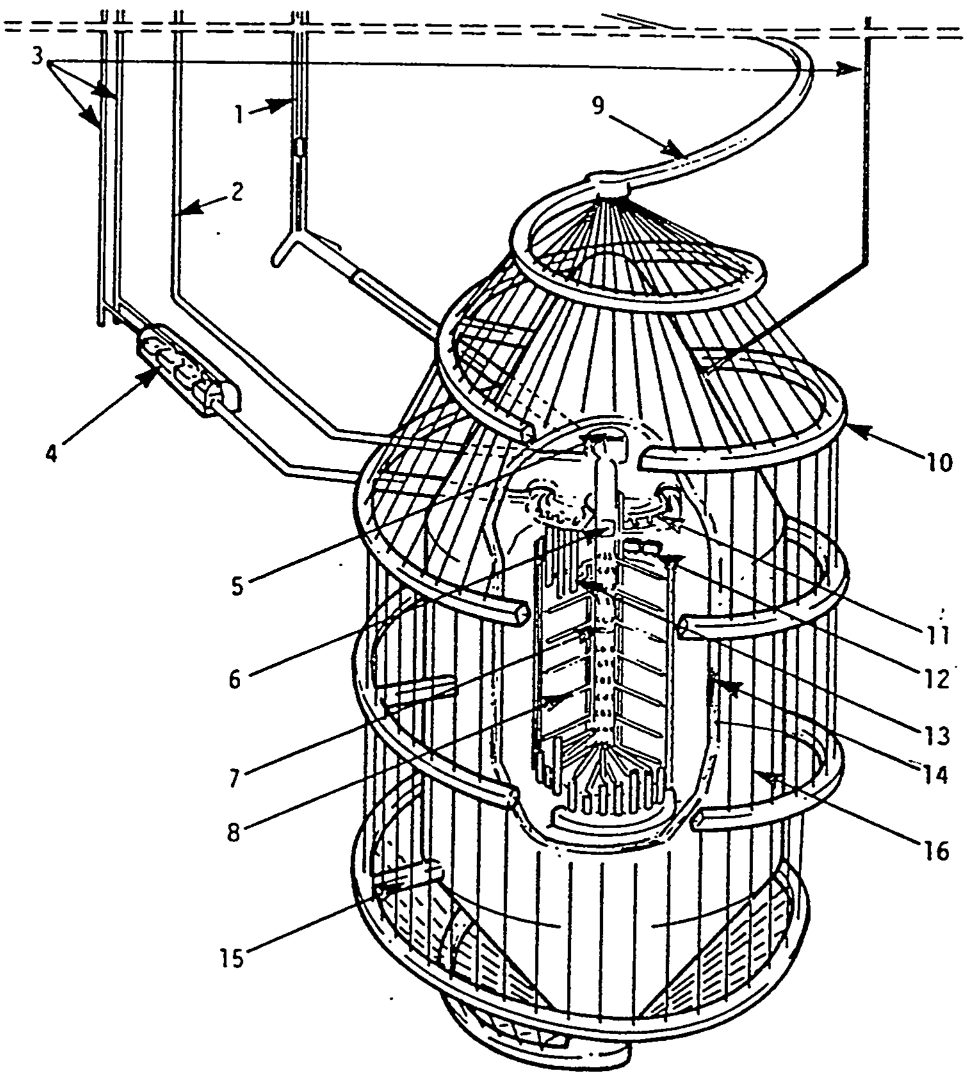

1 SHAFT AND DRIFT FOR TRANSPORTATION OF CANISTERS WITH NUCLEAR FUEL WASTE
2 MANWAY PASSAGE
3 VENTILATION SHAFT
4 HEAT EXCHANGE (1) STATION
5 HOISTING EQUIPMENT
6 MAIN SHAFT FOR STORING OF CANISTERS
7 DEVICE FOR REMOTE CONTROLLED STORING OF CANISTERS
8 CANISTERS IN STORAGE POSITION
9 MAIN ADIT FOR EXCAVATION AND CONSTRUCTION
10 TRANSPORTATION RAMP
11 FAN AND HEAT EXCHANGE (2) AREA
12 STORAGE OF LOW OR MIDDLE ACTIVE MATERIAL WITH THE LONG-LIVED ISOTOPES, NICKEL -59 AND NIOBIUM -94
13 STORAGE VENTILATION SHAFT
14 BENTONITE-QUARTZ BARRIER WITH A THICKNESS OF 5 METER
15 DRIFT TO BENTONITE BARRIER
16 DRILLHOLE FOR HYDRAULIC CAGE

Figure 6.3.4. Excavation of Barrier by Spiral Ramp

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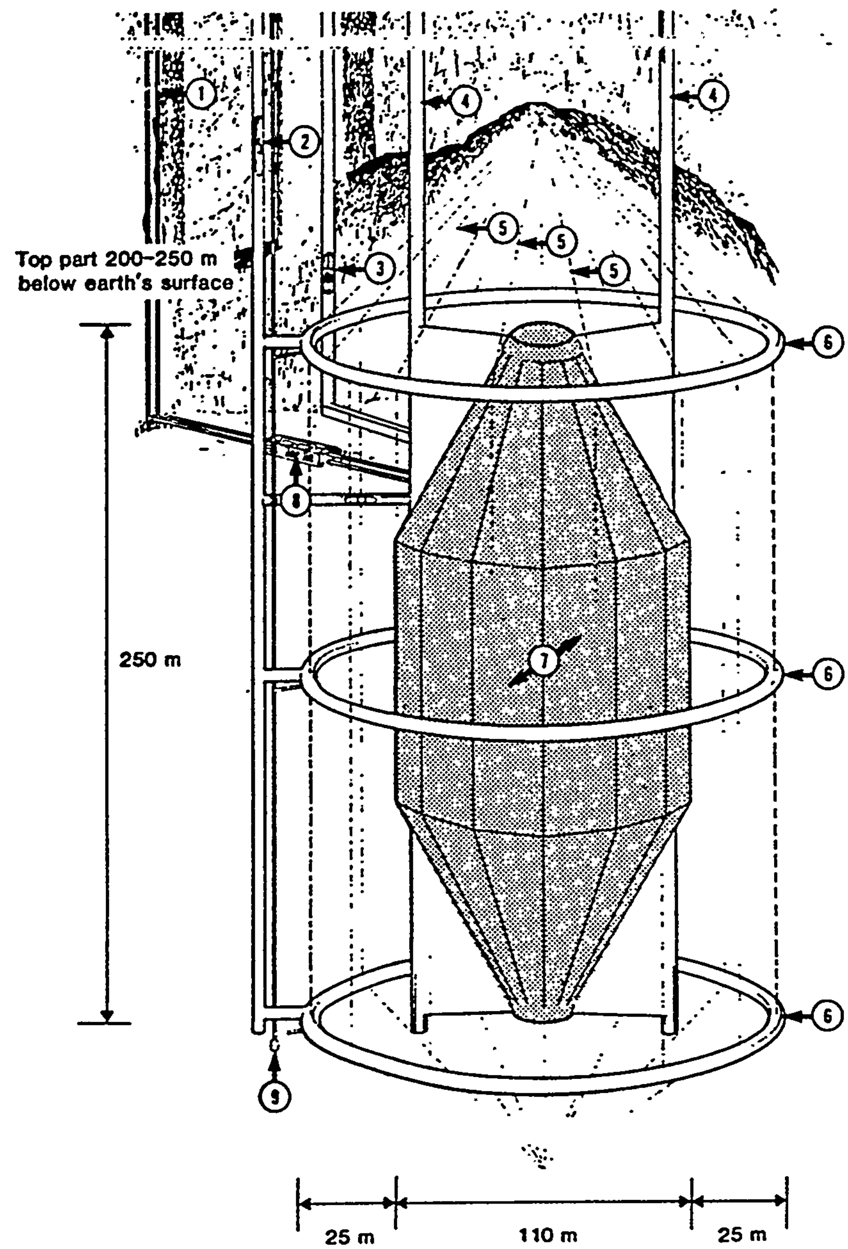

1. Ventilation shaft
2. Shaft for construction of hydraulic cage
3. Transportation of canisters
4. Main shaft for excavation and refilling of slot
5. Drillholes in hydraulic cage
6. Annular tunnels
7. Bentonite-sand barrier with a thickness of 5 m
8. Heat exchange station
9. Pump station

Figure 6.3.5. Excavation of Barrier by Shaft and Tunnel

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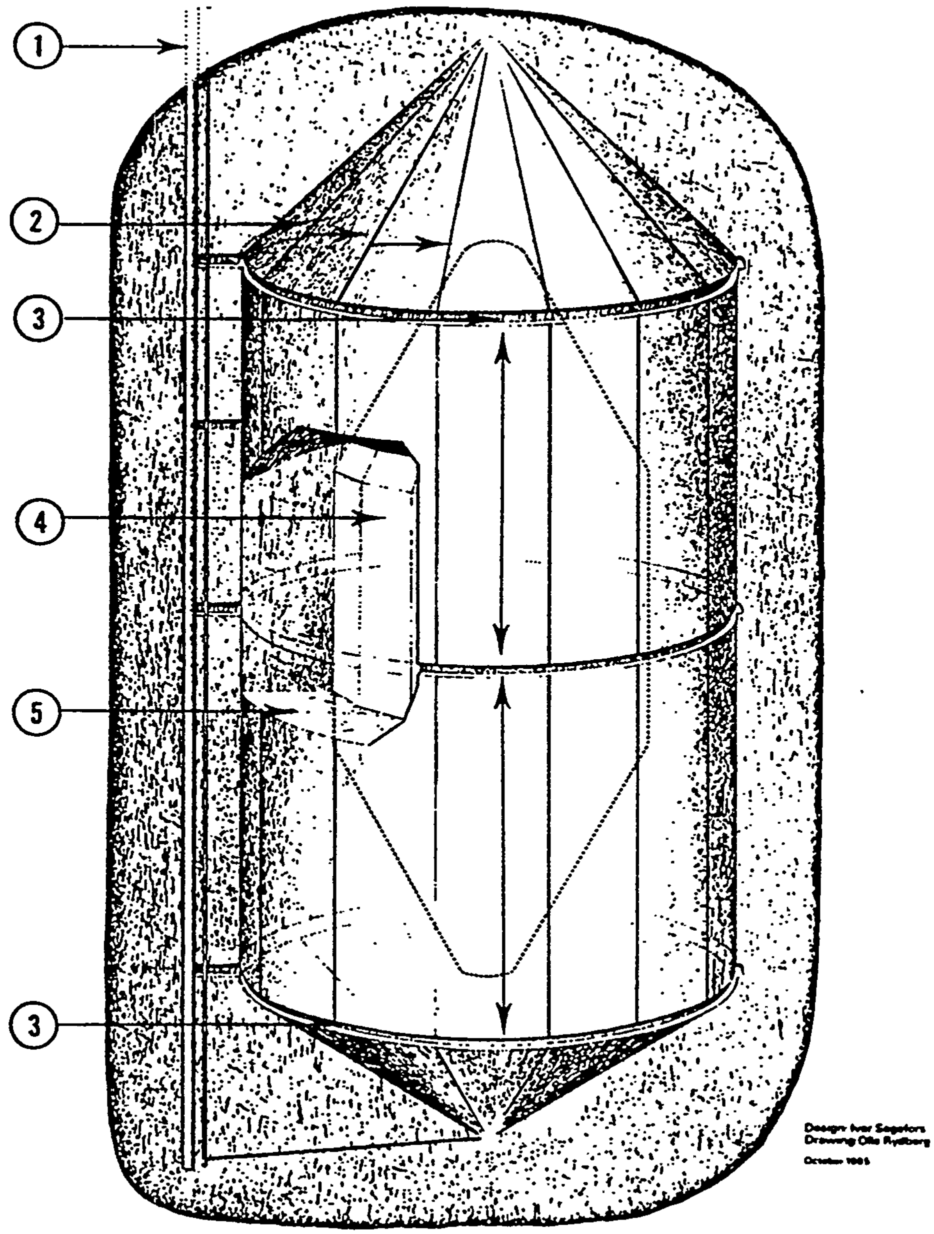

1. Shaft for construction of hydraulic cage
2. Drillhole in hydraulic cage
3. Annular tunnels in hydraulic cage
4. Bentonite-sand barrier
5. Rock mass between hydraulic cage and bentonite-sand barrier

Figure 6.3.6. Hydraulic Cage Construction

central access shaft. The layout of the storage channels will allow storage of the spent fuel canisters such that they are accessible for monitoring and control, and cooling by forced or natural air circulation. The channels will be long enough to accommodate three canisters between the inner and outer ventilation shafts. There will be 14 storage levels spaced at about 4.5 m (15 ft) vertical intervals with 12 channels per level. Storage capacity will thus total 504 canisters containing 1,600 MT (1,760 tons) of spent fuel. Each channel will be equipped with a rail system for handling the canisters. The ventilation shafts will be designed for raise boring and have diameters in the range of 2.4 to 4.0 m (8 to 13 ft), depending on ventilation requirements. Fans and heat exchangers will be installed at the top of each set of inner and outer ventilation shafts to cool the circulating air. The ventilation system will be designed to keep the temperature in the repository area between 40 to 60°C (104 to 140°F) during the monitored storage period.

Total excavation volume was estimated to be about 75,000 m^3 (98,000 yd^3).

6.3.3.2.2 Bentonite-sand Barrier. The storage area described above will be surrounded by a barrier of compressed bentonite clay mixed with a friction material such as sand (Figures 6.3.1 and 6.3.4). The barrier serves two main functions:

- Isolation of the rock mass containing the storage areas from the ground-water flow in the main rock body
- Delay long-term migration of radionuclides from the storage area to the main rock mass ground-water flow.

The barrier thickness required for this study was estimated to be about 5 m (16.4 ft) and the slot for the barrier will be located at about 110 m (360 ft) diameter around the storage area. The barrier slot will be excavated from the bottom to the top using the "cut-and-fill" mining method where several meter high sections of the slot are excavated and then backfilled with the bentonite mixture. Access for the excavation of the barrier can be via a spiral decline from the surface (Figure 6.3.4) or via shafts and drifts (Figure 6.3.5). After completion, the rock mass with the storage areas will be supported in all directions by the bentonite barrier. Near the bottom of the barrier, a higher friction material will be required so the bentonite content may only need to be about 10%. The cylindrical part may need about 20% bentonite and near the top, a 50% bentonite-sand mixture may be needed to reduce hydraulic conductivity. The excavation volume for the barrier slot for the design shown in Figure 6.3.1 was estimated at about 350,000 m^3 (458,000 yd^3) and the total bentonite requirements for the mixtures described above would be about 90,000 m^3 (118,000 yd^3).

6.3.3.2.3 Hydraulic Cage. The hydraulic cage will be constructed around the bentonite barrier at a distance of about 25 m (82 ft). The cage will consist of a series of circular drifts around the repository at various levels and a large number of vertical boreholes connecting these drifts (Figures 6.3.6 and 6.3.7). This so-called cage has completely surrounds the repository and bentonite barrier and has two basic functions:

- Drain the rock mass between the cage and bentonite barrier during construction and operation or until the repository is finally sealed

- Divert ground-water flow when the WP-Cave has filled with water after repository closure and thus reducing ground-water flow through the storage area.

The cage can be constructed either from a shaft and drift access or from a spiral ramp, depending on the method chosen for the construction of the bentonite barrier (Figures 6.3.4 and 6.3.6). Dimensions of the drifts or ramp will depend on the construction equipment and methods. The drill holes were assumed to be 150 mm (6 in) in diameter for the study summarized here. Model studies of the ground-water flow through the cage showed that for a cage diameter of about 150-160 m (490-525 ft) and a hole spacing of about 3 m (10 ft), giving a total of about 150 holes, approximately 97% of the ground-water flow will be diverted. The studies also showed that much higher diversions in ground-water flow can be achieved with the cage than with the bentonite-sand barrier. The 150 hole concept will divert about 97% of the flow compared to 35% for the bentonite-sand barrier with a hydraulic conductivity ten times lower than that of the rock.

6.3.4 <u>Repository Construction</u>

6.3.4.1 Schedule

The schedule and requirements of the Swedish radioactive waste disposal program were described in Section 6.1. The estimated construction time for the WP-Cave is about 5 years.

6.3.4.2 Methods and Equipment

6.3.4.2.1 <u>Shafts and Tunnels</u>. The design of the WP-Cave assumed that the latest shaft drilling and raise boring technology would be applied to a large portion of the construction program. Construction of the WP-Cave is envisioned in two phases:

- Completion of the hydraulic cage

- Excavation of the bentonite barrier and storage areas.

Three shafts will be required for the excavation and backfilling operations if the facility is to be constructed using shafts and drifts for access and hoisting of the excavated rock (Figure 6.3.5). Two shafts would service the excavation and backfilling operations in the barrier slot and

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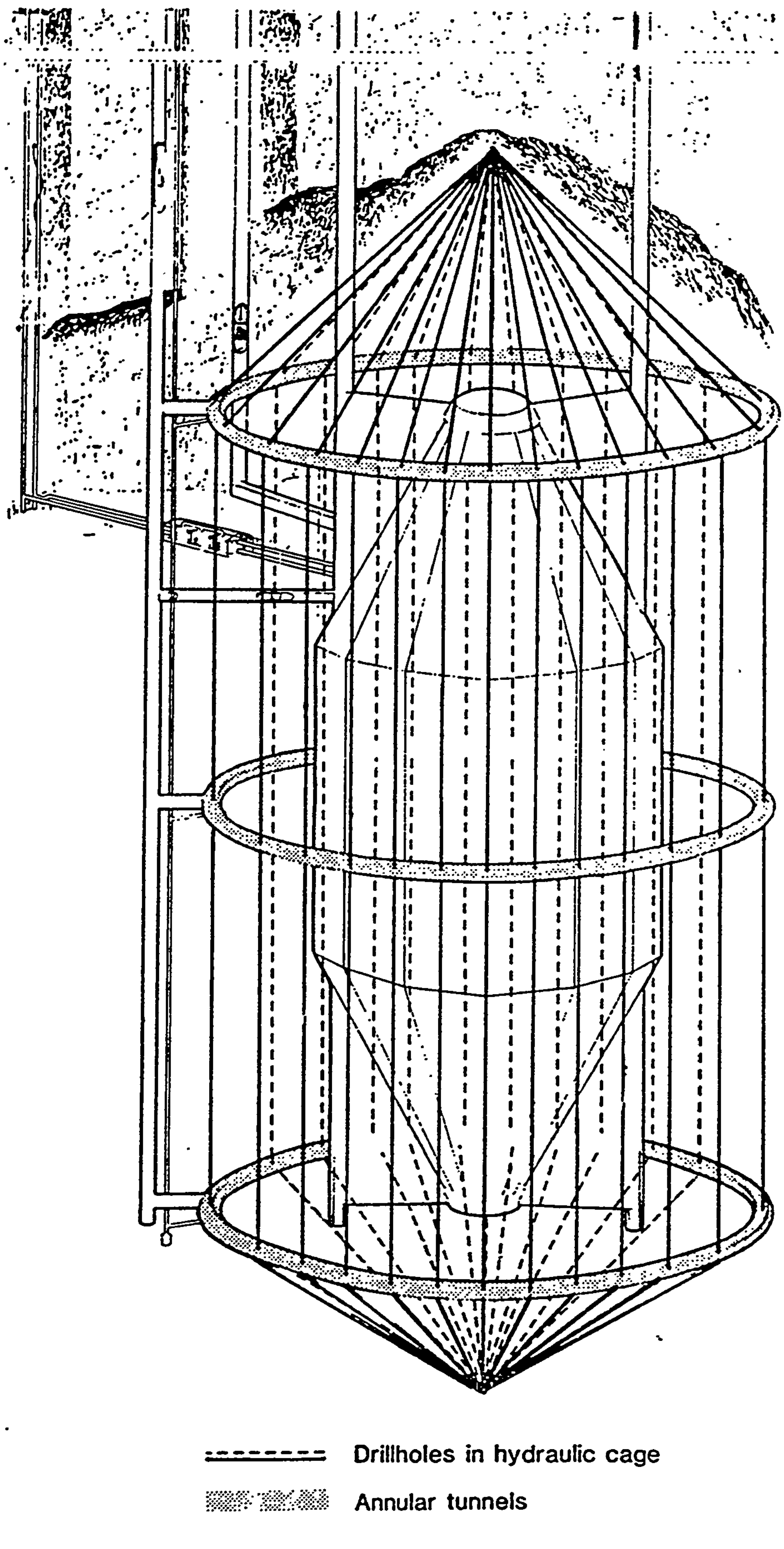

Figure 6.3.7. Hydraulic Cage Configuration

the third shaft would be used for the construction of the annular drifts from which the holes for the hydraulic cage would be drilled. The hydraulic cage drifts and boreholes as well as the pumping station at the base of the repository must be finished before the barrier slot is constructed to divert the ground-water around the slot.

Access for personnel and material during construction of the waste storage areas as well as for ventilation and radioactive waste container transport will be via additional shafts and tunnels (Figure 6.3.5). The inner and outer ventilation shafts in the storage areas will probably be raise-bored while the large central shaft will be excavated using smooth-wall blasting. For the excavation of the storage channels, machine excavation was investigated and several concepts such as multi-purpose borers, blind hole raise drills, and raise drills were suggested. The overall objective of the study of the excavation techniques for the WP-Cave was to apply mechanical excavation techniques as much as possible to preserve the integrity of the surrounding rock mass. The proposed WP-Cave concept will thus be continually revised as mechanical rock excavation methods continue to develop.

6.3.4.2.2 Bentonite Barrier. The slot for the bentonite barrier will be excavated using about 12 raise-bored shafts spaced equally around the line of the proposed slot. The inclined shaft sections will terminate in a chamber at the top (and bottom) and a peripheral drift at the top (and bottom) of the vertical slot section. Excavation of the slot will start using smooth-wall blasting techniques once the shafts have been complete. Figures 6.3.8 and 6.3.9 show the excavation and backfilling of the slot. Excavation and backfilling proceeds in stages starting at the bottom. A certain height of the slot (say, 5 to 10 m or 16 to 33 ft) will be excavated and then backfilled with the bentonite-sand mixture in a manner similar to the cut-and-fill mining method.

6.3.4.2.3 Storage Areas. The storage areas will consist of a number of holes drilled radially from the central shaft and sloping down at about 30° to intersect the raise bored inner and outer ventilation shafts (Figure 6.3.3). The actual storage section of the boreholes lies between the inner and outer ventilation shafts and will be approximately 15-20 m (50-66 ft) long to hold three canisters. The storage boreholes will be excavated using drilling techniques such as raise boring or blind hole raising. The central storage shaft will have a diameter of about 14 m (46 ft) and excavation of this shaft will probably be by the pilot-and-slash method with the application of smooth-wall blasting to protect the rock mass.

6.3.4.3 Cost

Cost estimates for the WP-Cave construction and operation were not found in the literature.

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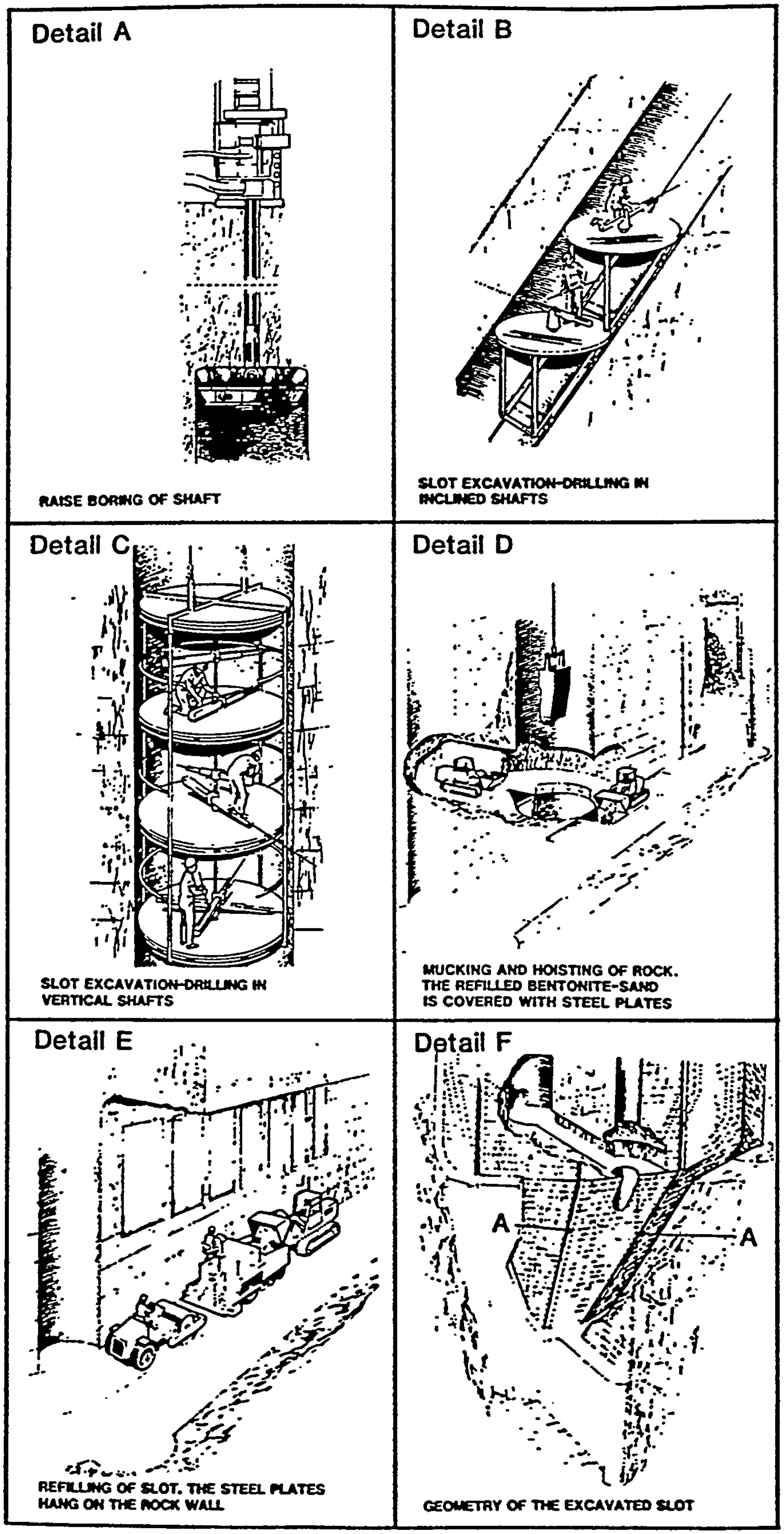

Figure 6.3.8. Bentonite Barrier Construction

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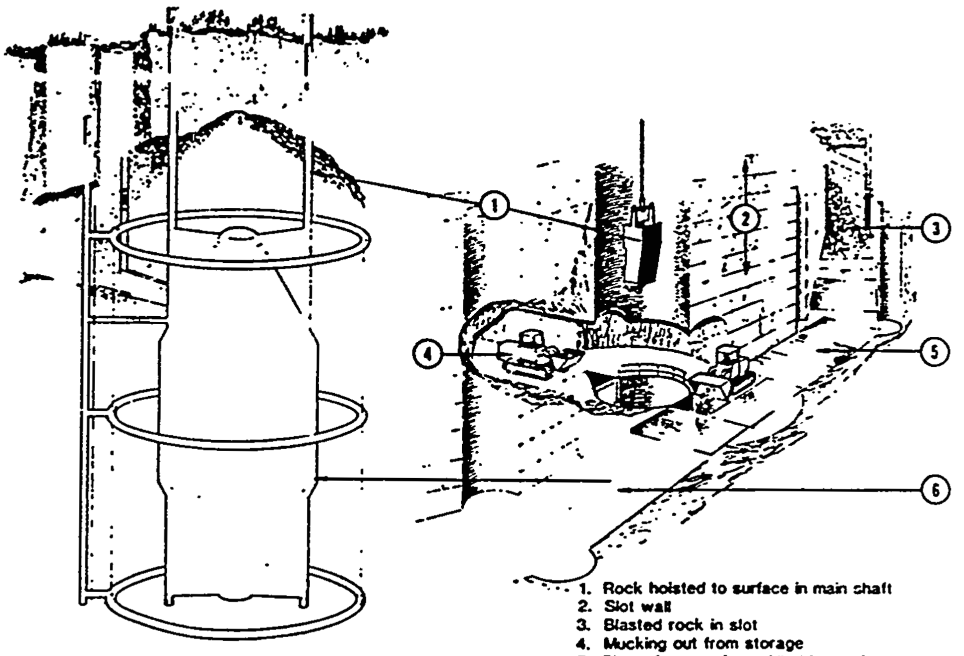

Figure 6.3.9. Barrier Slot Construction

6.3.5 Waste Emplacement Cycle

The Swedish waste disposal strategy was outlined above in Section 6.1 and is based on the direct disposal of spent fuel in a mined, geologic repository after an interim storage period of about 40 years in a near surface monitored storage facility (CLAB). This section describes a proposed spent fuel canister and emplacement method for use in the WP-Cave repository which is designed to operate as an monitored storage facility as well as the final repository.

6.3.5.1 Waste Treatment

The WP-Cave is designed to store even relatively young, 3-5 years old, spent fuel from boiling water reactors as well as pressurized water reactors. The WP-Cave ventilation study, however, assumed that canisters containing 40 year old boiling water reactor spent fuel had been placed in the WP-Cave and that the canisters would be cooled by ventilation for about 100 years.

6.3.5.2 Waste Transport

The waste transport system was assumed to be similar to that described in Section 6.1 which is currently being used and thus the canisters, etc., were designed to utilize this equipment.

6.3.5.3 Waste Package

The waste package for the WP-Cave will basically be a carbon-steel canister containing spent fuel elements which will be in the same condition as when they were removed from the reactors (Figure 6.3.10). The canister was expected to hold about 17 boiling water reactor fuel elements (~ 3.2 MT or 3.5 tons) or about 7 pressurized water reactor fuel elements. The outer barriers of the WP-Cave (bentonite barrier and hydraulic cage) are designed for the long-term isolation of the storage area. The canisters thus do not have to be designed as long-term barriers and canister life was therefore set at about 1,000 years. During the initial 100-200 years, the canisters will have to be accessible and thus may have to withstand handling by the retrieval equipment. Also, some corrosion resistance is desirable after repository sealing to allow short-lived radionuclides to decay and thus decrease the temperature of the spent fuel. A carbon-steel canister was determined to meet these requirements at a fairly reasonable cost. The thickness of the canister was calculated to be about 100-150 mm (4-6 in), based on conservative estimates of mechanical and corrosion resistance requirements. Reevaluation of these requirements is, however, considered necessary since these thicknesses would mean less spent fuel capacity per canister if the canister is to fit currently used transport equipment which limit the canister diameter to about 910 mm (3 ft).

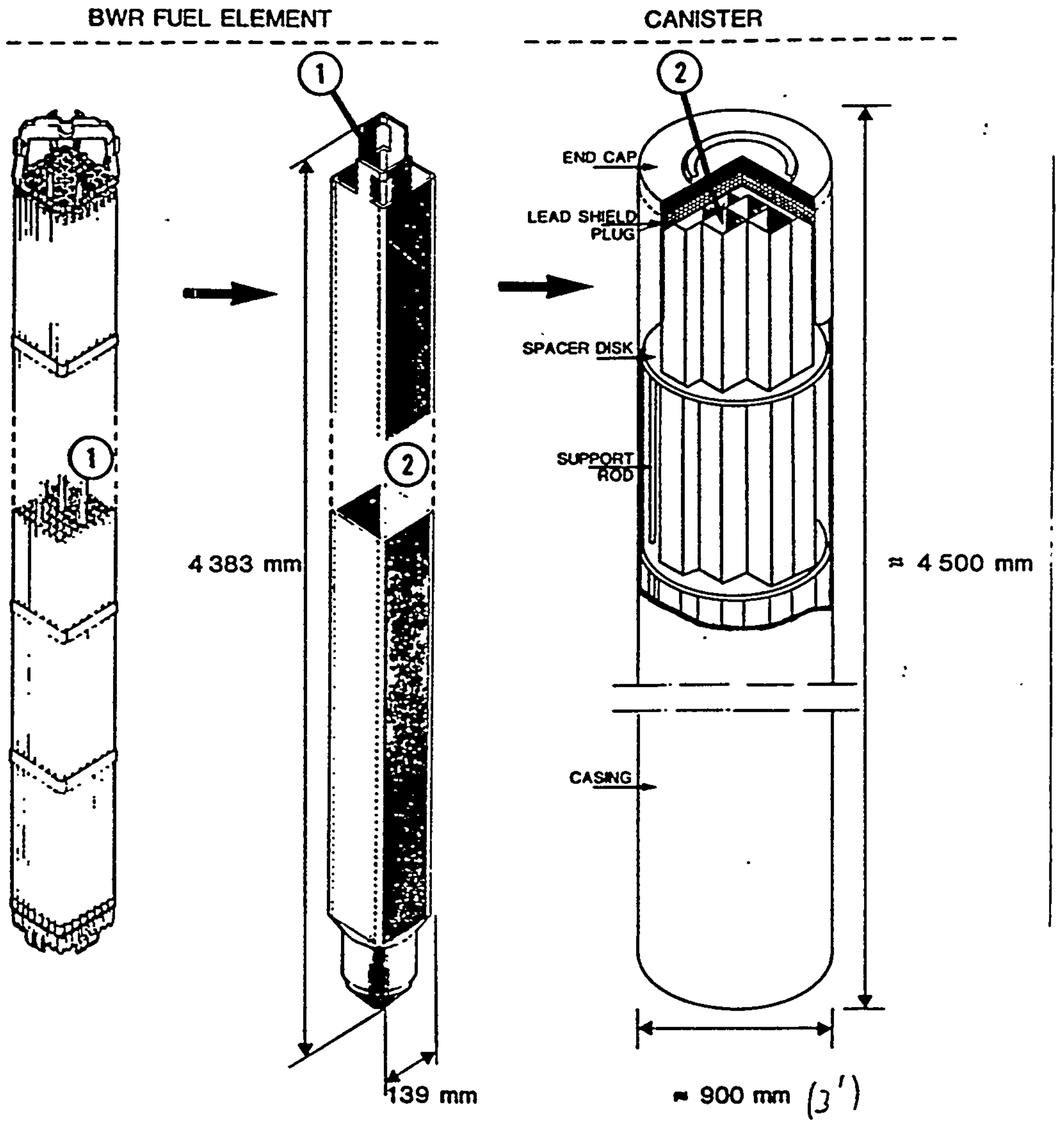

Figure 6.3.10. WP-Cave Waste Canister

6.3.5.4 Underground Waste Emplacement Sequence

The waste canister will be transported from the receiving station at the surface in a specially designed transport cask down the waste shaft to the central shaft of the storage area. The transport in the access drift from the bottom of the shaft to the central storage shaft will probably be by rail. Operations in the central storage shaft will be remotely controlled. The transport cask will be lowered on a special platform and aligned at the correct storage borehole (Figure 6.3.11). The canister will be lowered slowly into position on the rail system. The transport cask will return to the surface once the canister is in position. The same transport cask will be designed so that it can also retrieve the canister should this be necessary at some future time.

The ventilation systems (fans, heat exchangers, etc.,) will operate throughout the waste emplacement period to remove the heat generated by the waste packages. The maximum air temperature in the storage areas will be kept to 60°C (140°F) in case personnel access is required. The supervised storage period is expected to last about 100 years from the time the first waste canister has been stored and during this period, the temperature of the spent fuel must be kept below 350°C (660°F) to prevent conversion of the uranium matrix in to a more water soluble form through chemical oxidation. The exact storage time will be determined by several factors:

- Rate of decay of the spent fuel heat production until the heat dissipation through the rock mass is equal to the heat production

- Maximum temperature to which the bentonite-sand barrier may be exposed.

6.3.5.5 Emplacement Schedule

The estimated schedule for the emplacement of the various radioactive wastes in Sweden was described previously in Section 6.1.

6.3.6 <u>Repository Sealing</u>

When the WP-Cave is to be converted to the final repository, all unnecessary equipment such as fans, heat exchangers, hoists, etc., will be removed. The ventilation shafts in the storage will be left open so that heat transportation can be maintained. Some openings such as the central shaft, will have no function after repository closure and can be backfilled with a suitable sorptive material. It will probably be appropriate to flush the storage areas with an inert gas or to fill it with water before sealing to generate a reducing environment and reduce the corrosion of the waste canisters. All access drifts, ramps and shafts to the storage area (especially through the bentonite barrier) will be properly sealed and the WP-Cave left for the ground water to return to its original level and flow conditions. Over the long term, the rock mass within the bentonite/sand

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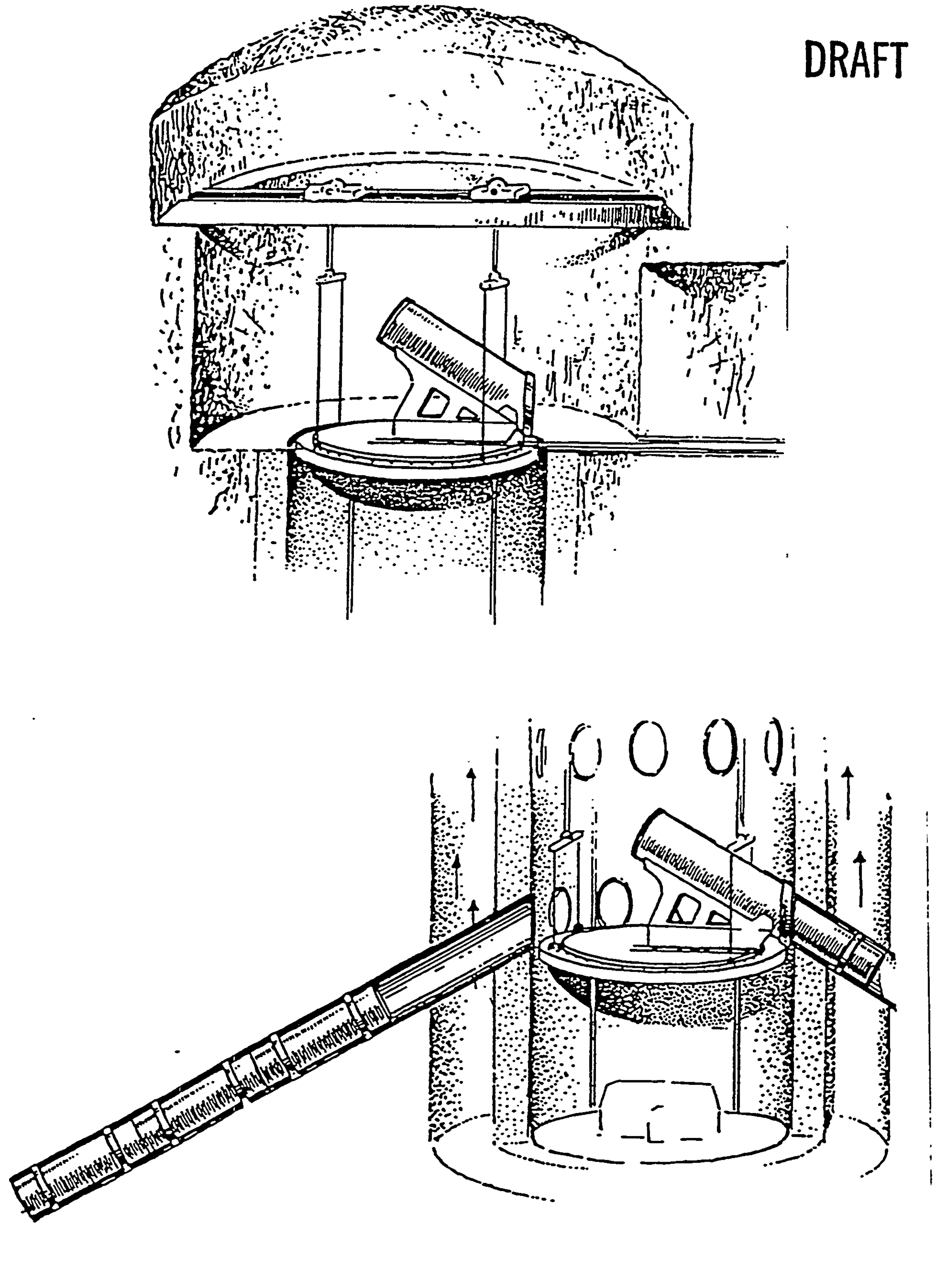

Figure 6.3.11. Waste Canister Emplacement

barrier will gradually refill with water and the canisters are assumed to collapse after their 1,000 year design life. The migration of the radionuclides will then be restricted by:

- Sorption in the inner rock mass, the bentonite-sand barrier and the exterior rock body and,
- The containment of the water inside the bentonite-sand barrier.

The combined effect of the barrier and the hydraulic cage is expected to limit ground-water flow such that diffusion will be the main radionuclide transport mechanism to the exterior rock body.

One subject under consideration is the shape of the bentonite barrier. It may not be required that the barrier surround the storage area completely at the base as shown in Figure 6.3.12. The effects of removing the bottom cone of the barrier on the mechanical, hydrological and thermal properties of the rock mass are, however, still under investigation.

6.3.7 Summary

The WP-Cave has evolved as an alternative radioactive waste disposal concept, possibly incorporating monitored, retrievable storage for many decades with the final disposal of spent fuel or high-level wastes in one mined, underground facility. The following briefly discusses some of the advantages and potential problems of the concept.

6.3.7.1 Advantages

1. Complete enclosure of the waste canisters by a bentonite-sand barrier can be achieved using this concept.
2. The concept may be independent of rock mass structure to some degree since the hydraulic cage and bentonite barrier are used to restrict the ground water flow through the storage areas.
3. Excavation of the shafts and drifts can utilize mechanical methods which will limit the damage to the surrounding rock body.
4. Waste package retrieval will be possible prior to final closure of the repository.
5. Extensive characterization of the storage rock mass will be possible since a lot of shafts and boreholes will be excavated within this area.
6. The repository is relatively small and compact and several of these could, perhaps, be constructed nearer the actual sources of radioactive wastes rather than using one large central repository

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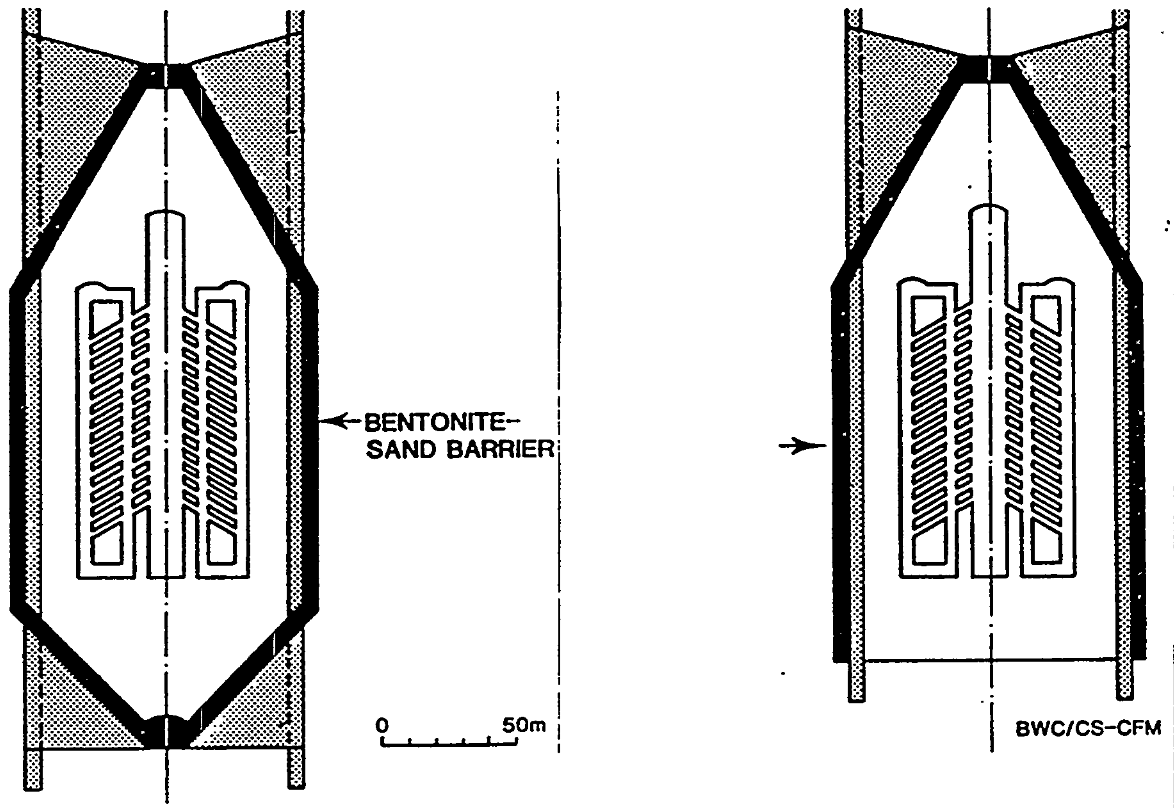

Bentonite-sand barrier with bottom cone

1. Excavation of the interior and the bentonite-sand barrier and refilling and excavation of sloping parts of the barrier.
2. Rock mechanics stability analysis.
3. Rock mass characterization for possible site.
4. Monitoring of all stages of construction.
5. Order of excavation of interior and bentonite-sand barrier.
6. Fracture initiation in bottom cone of bentonite-sand barrier.

Bentonite-sand barrier without bottom cone.

1. Mining of the interior and the bentonite slot.
2. Rock mechanics stability analysis.
3. Rock mass characterization for possible site.
4. Rock mechanics analysis for the ventilation stage.

Figure 6.3.12. Alternative Barrier Configuration

with the associated transport problems and hazards. Also, for countries with small waste programs, the concept might present a viable alternative.

6.3.7.2 Potential Problems

1. A radioactive waste program the size of the U.S. with around 130,000 MT (143,000 tons) of spent fuel would require construction of about 80 WP-Cave repositories of the size described here. This is probably not a cost effective solution compared to two large repositories.

2. Some questions arise with regard to the bentonite-sand mixture:

   a. Can the mixture be compacted tightly enough to avoid slumping and void at the top of the barrier through consolidation?

   b. Will the mixture swell enough to restrict water movement to diffusion alone?

   c. Will the seals along the bentonite-quartz sand mixture prevent water migration?

   d. Will the temperature effects remain negligible?

   e. Will collapse of the waste canisters affect the integrity of the barrier?

   f. What are the long-term properties of the bentonite mixture as a seal?

3. The efficiency and operation of the hydraulic cage under in situ conditions will need to be proven.

4. The feasibility of safely isolating a rock body completely from the host formation within a clay barrier has to be established.

6.3.7.3 Summary

The WP-Cave concept was developed as an alternate concept for the disposal of radioactive wastes. Whether the principles will actually function as proposed will have to be subject of extensive further investigation. For a waste disposal program as large as that of the United States, the WP-Cave system does probably not represent a feasible alternative to the current concepts due to the large number of these repositories that would be needed.

6.4 CRYSTALLINE ROCKS: THE FINNISH CONCEPT

This chapter describes the high-level waste disposal concept for the Finnish nuclear power program. The descriptions are based mainly on a report published in 1982 by the Teollisuuden Voima OY (TVO), the power company operating two of the four Finnish nuclear power plants. The report presented several alternative high-level waste repository configurations, although in general the nuclear waste disposal strategy is based on the Swedish SKB program.

Finland operates four reactors: two Soviet-built pressurized-water reactors (445 MWe each) at the Loviisa power plant and two Swedish-built boiling-water reactors (710 MWe each) at the Olkiluoto power plant. The spent fuel from the Soviet reactors is being returned to the Soviet Union after storage at the reactors for about 5 years, while the low- and intermediate-level wastes from all the reactors require disposal in Finland as well as the spent fuel from the remaining two reactors, if no foreign treatment or final disposal services are acquired.

6.4.1 Geographic Location

Five potential repository locations were identified for further site investigation activities in the spring of 1987. Figure 6.4.1 shows the relative locations of the sites. Investigation activities at the Hyrynsalmi and Kuhmo sites were started in 1987 while those at the Sieve, Konginkangas and Eurajoki sites are scheduled to start in 1988.

6.4.2 Summary of Geologic Setting

The bedrock throughout Finland consists of ancient Precambrian formations. Investigations in these types of formations both in Finland and abroad have shown that suitable repository sites can be found in numerous areas in Finland. Hundreds of potential sites were narrowed down to five by early 1987 and detailed site investigation programs started. The five sites chosen for detailed investigation are all located in the crystalline bedrock. The locations of the sites are shown on Figure 6.4.1 and include Hyrynsalmi (granite), Kuhmo (gneiss), Sievi (granodiorite), Konginkangas (granite) and Eurajoki (migmatite). Site investigation will start with surface reconnaissance and will include drilling of 500 to 1,000 m (1,640 to 3,380 ft) deep boreholes. The activities are expected to be completed in 1992. More sites may be included in the investigation program, depending on results.

6.4.3 Repository Concept Description

6.4.3.1 Nuclear Waste Disposal Strategy

A fully defined nuclear waste disposal program does not yet exist in Finland. The current strategy calls for the interim storage of the spent

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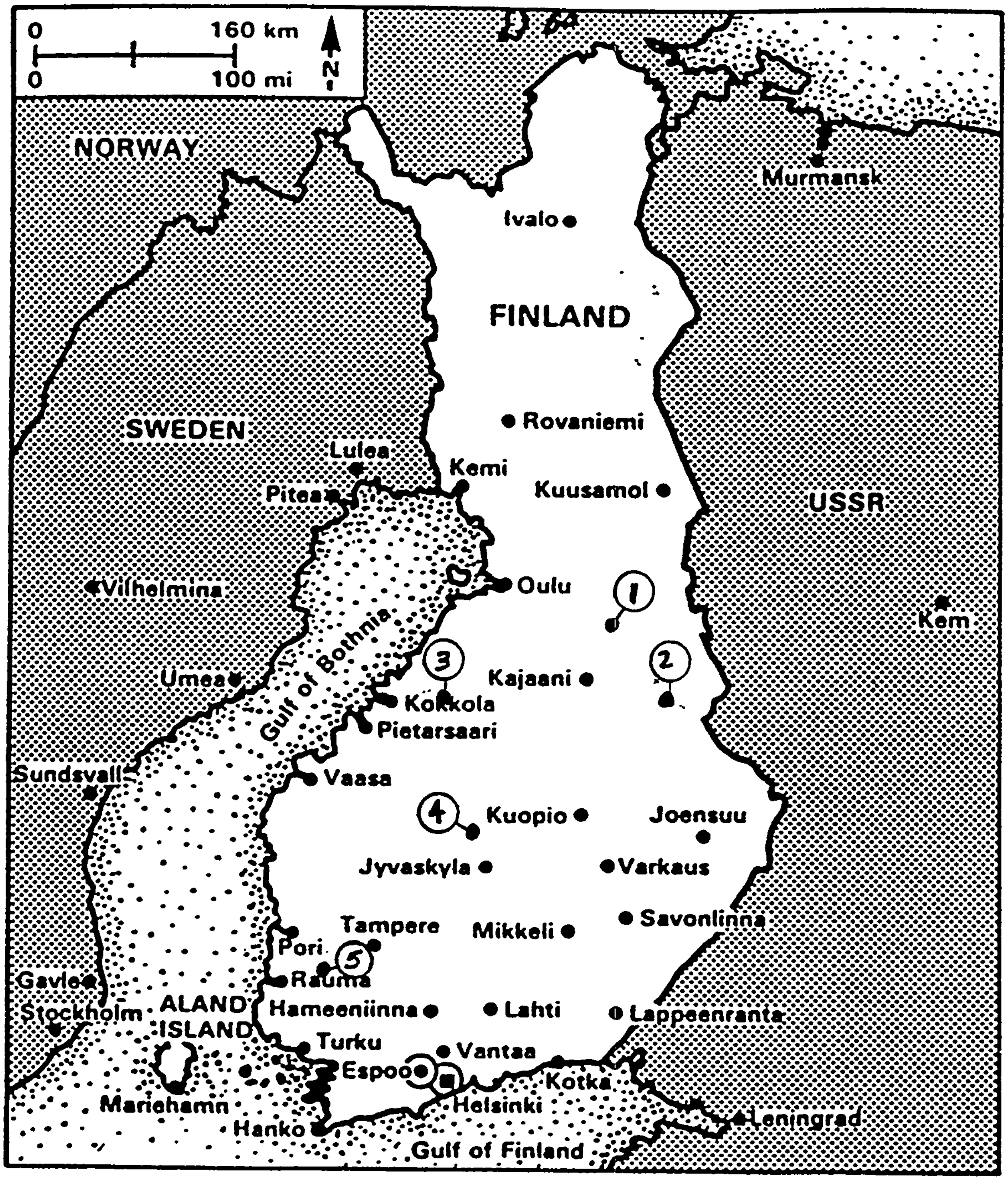

(1)	Hyrynsalmi	Granite
(2)	Kuhmo	Gneis
(3)	Sievi	Granodiorite
(4)	Konginkangas	Granite
(5)	Eurajoki	Migmatite

Figure 6.4.1. Location of Finnish Study Areas

fuel and high-level waste from the non-Soviet reactors for about 40 years, followed by final disposal in a mined, geologic repository in the crystalline bedrock. The final disposal of the spent fuel or high-level waste will become a reality only after several decades. The activity and heat generation of the waste will decrease with time and thus a long storage period is thought to be desirable with regard to encapsulation and final disposal. It is also considered advantageous to be able to learn from the experiences of other countries with larger nuclear waste programs before finalizing the relatively small Finnish program.

The study mentioned above on which these descriptions are based assumed that the spent fuel would not be reprocessed but rather stored for about 40 years before disposal in a mined geologic repository located in the Finnish crystalline bedrock. The spent fuel disposal strategy outlined here is based on the "multiple barrier" approach, where several barriers prevent the release of radionuclides (Figure 6.4.2):

- The fuel matrix
- The canister around the fuel elements
- The buffer material around the canister (excavations are also backfilled and sealed)
- The host rock.

Figure 6.4.3 shows a broad schedule of the major milestones for the proposed program. Site selection is expected to be complete by the year 2000; design and construction of the surface and underground facilities is to be complete in 2020 and after an operational period of 30 years, final closure scheduled for 2060.

The amount of spent fuel arising after an operational period of about 30 years is estimated to be about 1,200 MT (1,320 tons) for the two Swedish reactors at the Olkiluoto power plant in the southwest part of Finland. This is the amount used in the derivation of the repository concepts described here.

The current disposal strategy in short is to store the spent fuel for about 40 years, encapsulate it in copper canisters and then dispose of it permanently in a deep-mined geologic repository constructed about 500 m (1,640 ft) underground. The proposed disposal method is to place the canisters in vertical boreholes drilled into the floor of the access drifts. Final sealing and backfilling of the repository will take place once the repository is full. The following sections outline the major facilities and operations in more detail.

6.4.3.2 Surface Facilities

The main surface facilities that will be required for the storage, preparation and handling of the spent fuel and/or high-level wastes include

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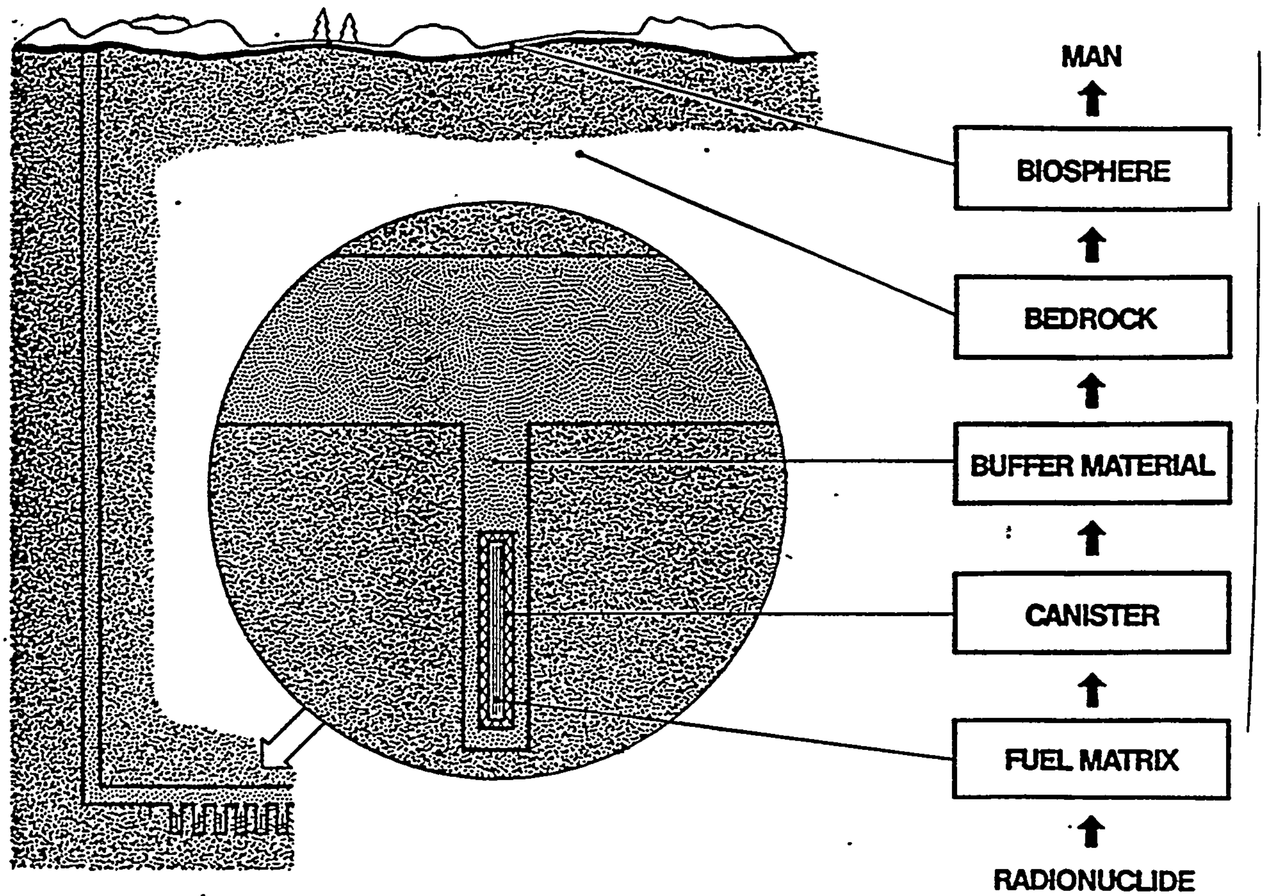

Figure 6.4.2. Multiple Barrier Isolation System

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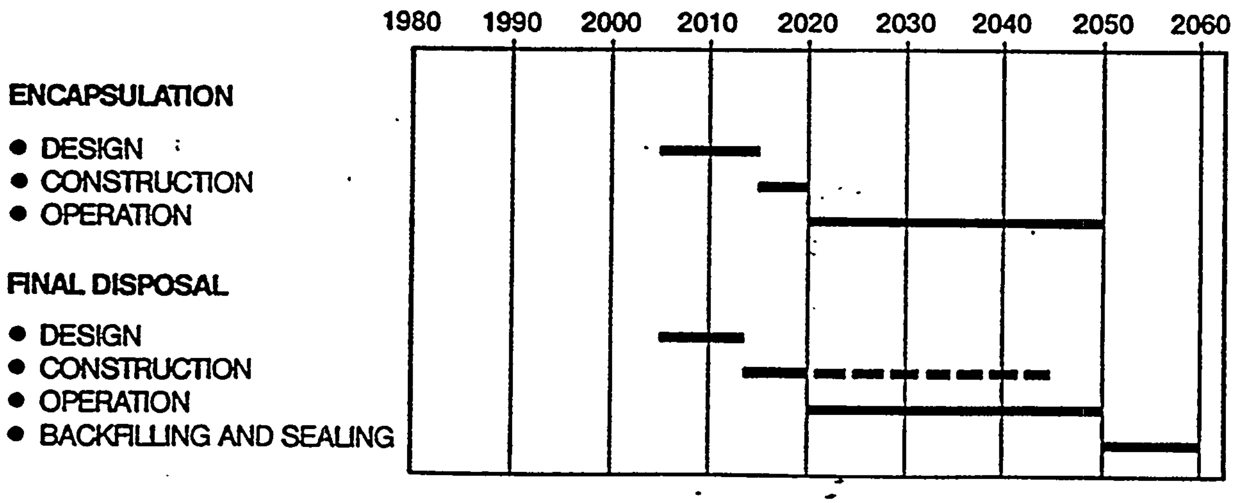


Figure 6.4.3. The Spent Fuel Disposal Program

an interim storage facility, the encapsulation facility for the spent fuel and the surface service facilities associated with the waste transport and repository operations.

Details of an interim storage facility were not available in the literature but it is assumed that the spent fuel would be stored at the reactor or in a specially constructed facility similar to the Swedish CLAB facility. Storage at the reactors is mainly in waterfilled pools while the Swedish CLAB facility is an underground facility where the spent fuel is stored in pools consisting of concrete lined caverns excavated in crystalline formations. Section 6.1 of this report describes the CLAB facility in more detail.

The encapsulation station is based on the design of the Swedish facility described in Section 6.1. Figure 6.4.4 shows the configuration of the facility. It can be located at the repository site near the waste transport shaft so that the canisters can be taken directly underground after the encapsulation process is complete. The spent fuel encapsulation process is shown in Figure 6.4.5. The spent fuel arrives in special transport casks by truck or rail at the reception hall where the casks are unloaded, washed, and checked for radiation leaks. After removal from the cask, the spent fuel assembles will be dismantled and the fuel rods placed into a copper canister (about 500 rods fit into one canister) which will then be filled with lead and sealed with a welded lid. Other fuel assembly components such as the hulls will be cast into concrete blocks and also taken underground for final disposal in the repository.

The other surface facilities will be required for the administration and operation of the repository. These facilities will include:

- Hoist and headframe complexes at each shaft
- Ventilation equipment
- Underground services such as compressed air, water, power, communications, etc.
- Concrete and bentonite preparation and storage area
- Waste rock storage area
- Warehouse, workshop, offices, change house, emergency rescue facility, site security facility, etc.

6.4.3.3 Underground Configuration

The repository for the spent fuel canisters will be constructed at a depth of several hundred meters (500 m or 1,640 ft was chosen as the reference depth for the TVO study) in the crystalline bedrock of Finland. The canisters will be surrounded by buffer materials as part of the multiple-barrier concept of final disposal. Several alternative repository

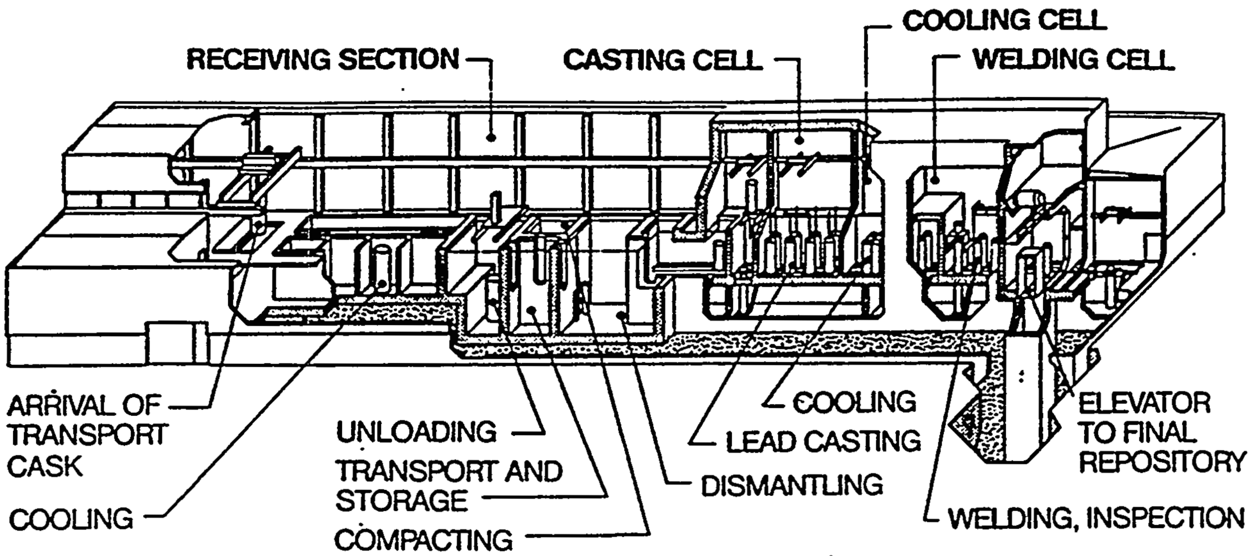
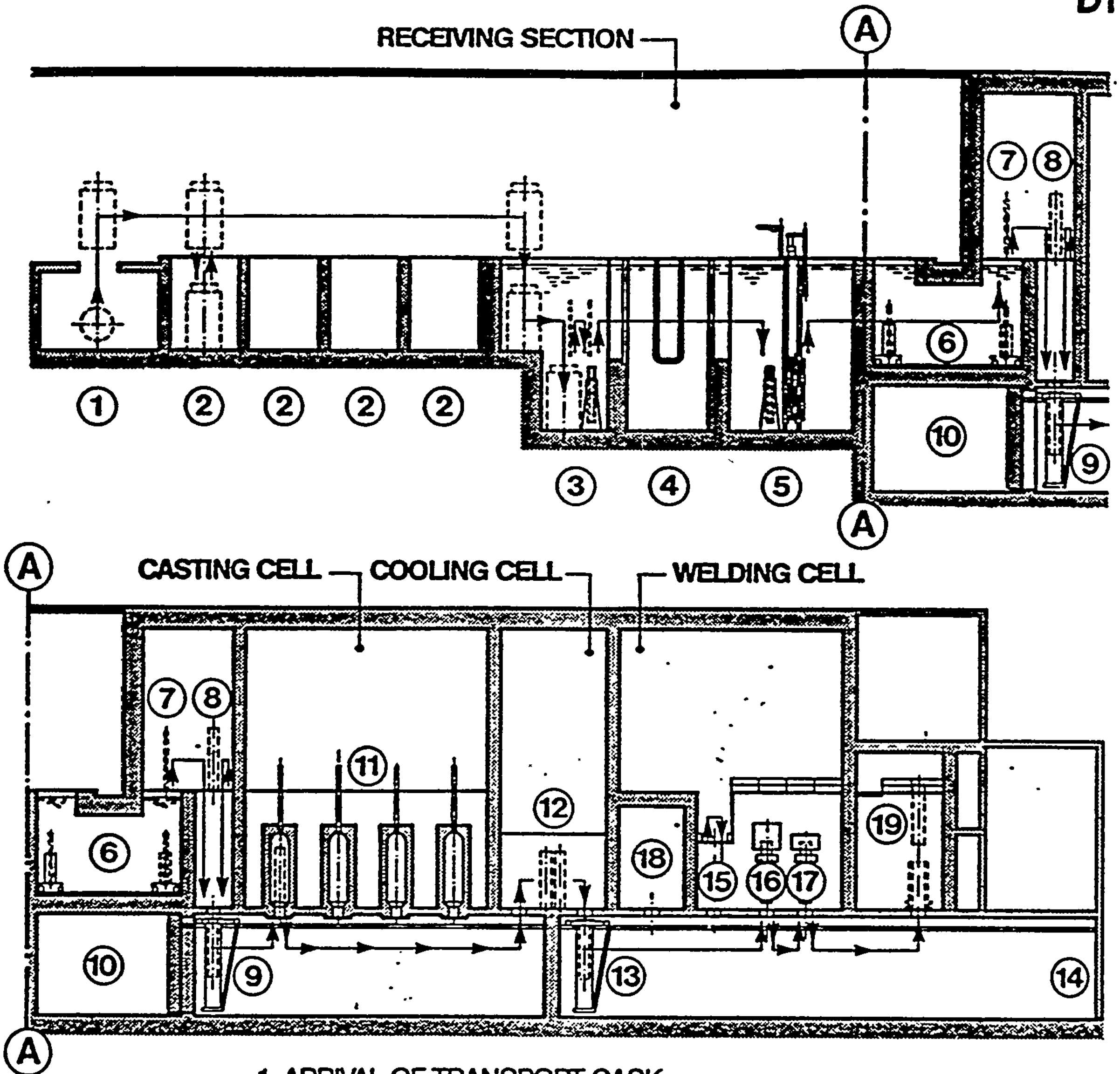


Figure 6.4.4. Spent Fuel Encapsulation Station



1. ARRIVAL OF TRANSPORT CASK
2. COOLING AND WASHING OF TRANSPORT CASK
3. UNLOADING OF TRANSPORT CASK
4. TRANSPORT AND STORAGE POOL
5. DISMANTLING OF FUEL ASSEMBLIES
6. TRANSFER CARRIAGE
7. DRYING OF FUEL RACK
8. TRANSPORT OF COPPER CANISTER FROM STORAGE FACILITY
9. TRANSPORT WAGON
10. SERVICE AREA FOR TRANSPORT WAGON
11. FURNACES FOR FILLING CANISTER WITH LEAD
12. COOLING STATION FOR CANISTER
13. TRANSPORT WAGON
14. SERVICE AREA FOR TRANSPORT WAGON
15. MOUNTING STATION FOR LID
16. WELDING STATION FOR LID
17. INSPECTION STATION FOR CANISTER
18. CELL FOR OPENING SEALED CANISTER
19. DISPATCH STATION

Figure 6.4.5. Spent Fuel Encapsulation Process

configurations were presented in the study report. These are outlined in the following sections. Also presented is the disposal concept for the concrete molds containing the spent fuel hulls, etc., and this was assumed to be the same in all the spent fuel disposal options. The concepts will be refined as more site-specific information becomes available and a better understanding of the disposal operations, requirements and techniques evolves. Disposal will be required for about 850 canisters containing approximately 1,200 MT (1,320 tons) of spent fuel. In addition, about 100 m^3 (130 yd^3) of metal waste (fuel hulls, spacers, etc.,) remaining after encapsulation of the spent fuel in the copper canisters, will require isolation. The metal wastes will be cast into about 200 concrete molds in the shape of 1.6 m (6.5 ft) cubes.

6.4.3.3.1 Repository A. Figure 6.4.6 shows a general view of the repository configuration Alternative A. This configuration is based on the Swedish concept described in Section 6.1. The repository will be located at a depth of about 500 m (1,640 ft) in the crystalline bedrock. The facility will consist of a series of parallel disposal tunnels and the associated access drifts and shafts. An area of about 400 by 500 m (1,310 by 1,640 ft) will be required for the repository. The repository will be designed for about 850 canisters, each of which will be placed into a disposal hole drilled into the floor of the disposal tunnel. The distance between the 1.5 m (5 ft) diameter and 7.7 m (25 ft) deep holes will be about 6 m (19 ft) which means a total of about 5,100 m (16,800 ft) of tunnel will be required. The canister in each disposal hole will be surrounded with a bentonite buffer and the holes, and later the complete tunnel, will also be backfilled to seal in the waste canisters. The repository concept was designed so that the waste emplacement and the development operations can be conducted separately (Figure 6.4.7). Three shafts are planned to serve the underground operations. The main shaft will contain the muck hoist and the personnel and material transport hoist as well as the underground services. The shaft will be constructed first and also serve as the initial exploratory shaft. The other two shafts will be the ventilation intake shaft and the waste transport/main exhaust shaft.

6.4.3.3.2 Repository B. Figure 6.4.8 shows the Alternative B repository concept. This concept is based on a Canadian proposal where the waste canisters are placed in drill holes in the floor of the disposal tunnels similarly to the Swedish proposal but the tunnels are wide enough for two boreholes. The borehole locations will, however, be staggered on either side of the tunnel centerline to the other to maintain at least a 5 m (15 ft) distance between holes (Figure 6.4.8). A total of about 2,600 m (8,530 ft) of disposal tunnel will be required for the 850 spent fuel canisters. Two shafts are proposed for this concept, one for the waste transport and one for personnel, material and muck transport. The disposal and development operations will each be served by different drifts to separate the two activities. The waste canisters will be emplaced in the same manner as in concept A and thus the same multiple-barrier principle applies to the waste isolation.

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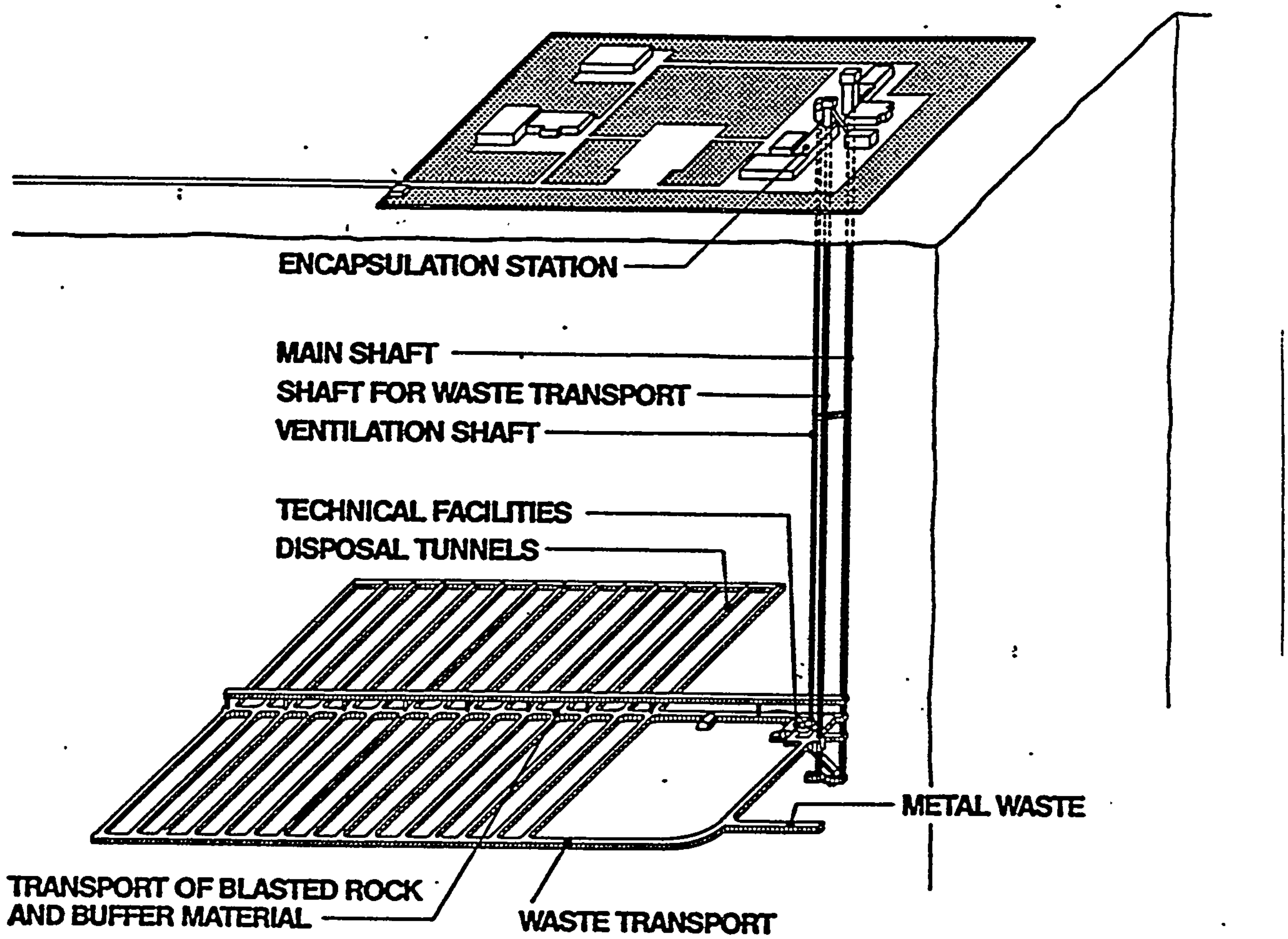

Figure 6.4.6. Repository Configuration A

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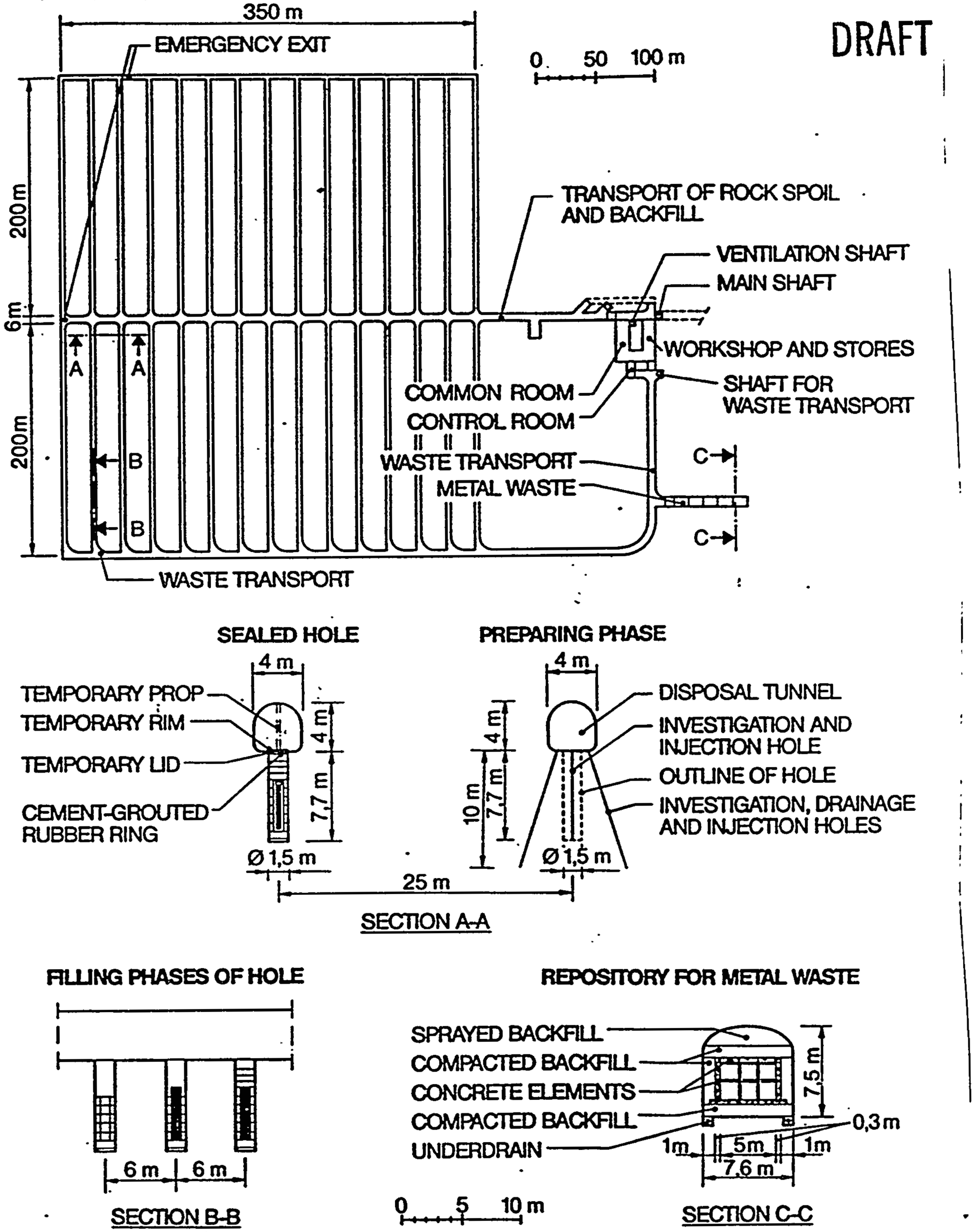

Figure 6.4.7. Repository A Operations

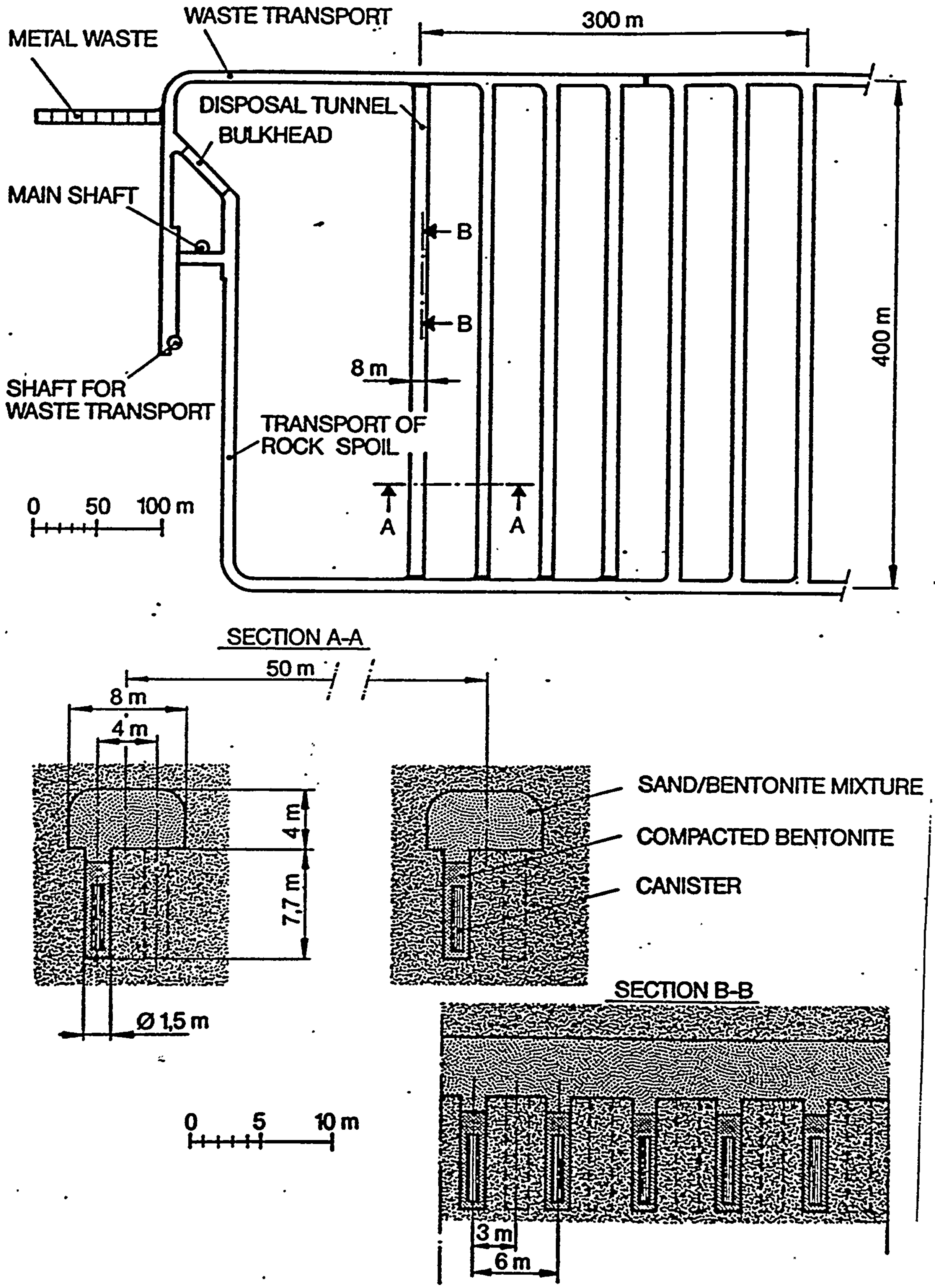

Figure 6.4.8. Repository Configuration B

6.4.3.3.3 Repository C. Figure 6.4.9 shows the repository Concept C, based on a Swiss study of reprocessed high-level waste disposal. In this concept, the waste canisters containing high-level waste or spent fuel will be placed horizontally in the center of the disposal tunnels rather than in vertical boreholes. The disposal tunnels will be completely filled with a sand/bentonite mixture which will provide the buffer around the waste package. The disposal tunnels will be excavated by tunneling machine to preserve the rock mass integrity. To avoid repeated erection and dismantling of the tunneling machine, the tunnel will be excavated in the shape of a spiral. The three shafts and technical facilities will be located in the center of the facility. The shafts will have the same functions as those in Alternative A.

6.4.3.3.4 Repository for Concrete Molds. Figures 6.4.7 to 6.4.9 show the proposed configuration and location of the metal waste repository within the various alternate spent fuel repositories. This repository will basically be a drift of about 100 m (330 ft) in length and 7.6 m by 7.5 m (25 ft by 24.5 ft) in section. The tunnels will be divided into compartments using concrete dividers surrounded by backfill (Figure 6.4.7) and the concrete molds will be placed into these compartments. Gaps around the molds and in the tunnel will be backfilled with a sand/bentonite mixture as the tunnel fills up.

6.4.4 Repository Construction

6.4.4.1 Schedule

Construction of the repository is scheduled to start about 2015 to allow disposal operations to start in 2020. Expansion of the repository will likely continue throughout most the operational period to open up new disposal tunnels as previous ones are filled. Figure 6.4.3 shows the overall schedule for the repository.

6.4.4.2 Methods and Equipment

Several mines are operating in Finland at comparable depths and thus construction experience in this type of rock is available. The use of drilling for the construction of the shafts and tunnels is generally preferred since damage to the surrounding host rock is reduced.

The first shaft to be constructed will most likely be the main shaft using conventional drill-and-blast shaft sinking techniques to open up the repository area, serve as initial exploratory shaft and provide the hoisting capabilities for personnel, materials and muck as the repository is expanded. The shaft will therefore be constructed to its final diameter and fully furnished with the hoisting arrangements and services. The ventilation and waste shafts can then be excavated using raise and/or shaft boring techniques, using the main shaft to remove the cuttings if necessary.

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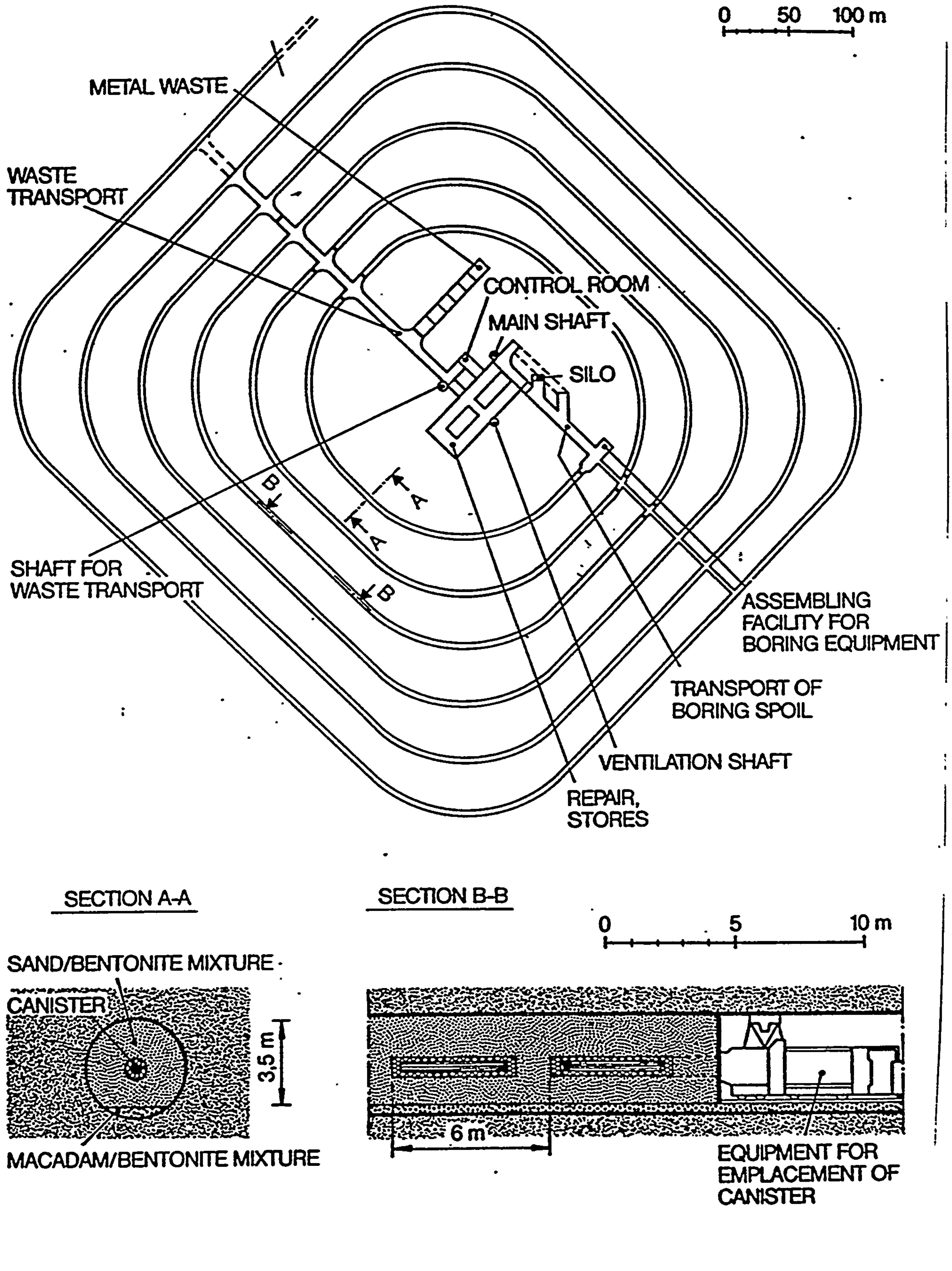

Figure 6.4.9. Repository Configuration C

The use of controlled blasting and pilot-and-slash excavation techniques are proposed for the excavation of the access drifts and disposal tunnels in Alternatives A and B to prevent excessive damage to the tunnel walls. Technology for the efficient drilling of the 1.5 m (5 ft) diameter and 7.7 m (25 ft) deep disposal holes in the tunnel floor is still under investigation. Several alternatives such as full-face shaft boring, slot-drilling, or coring are being tested. It may also be possible to adapt some of the oil-drilling technology to the drilling of the disposal holes.

The tunnels in Alternative C are to be excavated using full-face tunneling machines. This would create similar host rock conditions around the waste packages as in the disposal boreholes in Alternatives A and B without the need for the disposal hole drilling process. The machine tunneling technology has been used for many years under similar rock conditions and has been highly developed over the years. The tunnels would be excavated in the shape of a continuous spiral to avoid repeated assembly and dismantling of the machine.

An extensive site investigation program will also be conducted concurrently with the construction and operation of all the repository alternatives. This will determine the suitability of the actual disposal location or borehole, further characterize the repository host setting and be useful in determining tunnel support requirements to ensure long-term stability.

6.4.4.3 Cost

The TVO study included an estimate of the costs that are going to be incurred during the construction and operation of the encapsulation facility and the repository (reference case is Alternative A). Figure 6.4.10 shows a breakdown of those costs. The construction, operation and decommissioning of the encapsulation facility will cost about $276 million while the same costs for the repository will run about $145 million for a total of about $421 million.

6.4.5 Waste Emplacement Cycle

6.4.5.1 Waste Treatment

A fully defined nuclear waste disposal strategy has not been formulated yet and thus the TVO study was based mainly on the Swedish KBS report. The Swedish work assumed that the spent fuel would be encased in copper canisters without reprocessing before disposal underground. A special encapsulation facility would thus be needed. The encapsulation station assumed for the Finnish reference repository is based on the design of the Swedish facility described in Section 6.1. Figure 6.4.4 shows the configuration of the facility. It can be located at the repository site near the waste transport shaft so that the canisters can be taken directly underground after the encapsulation process is complete. The spent fuel

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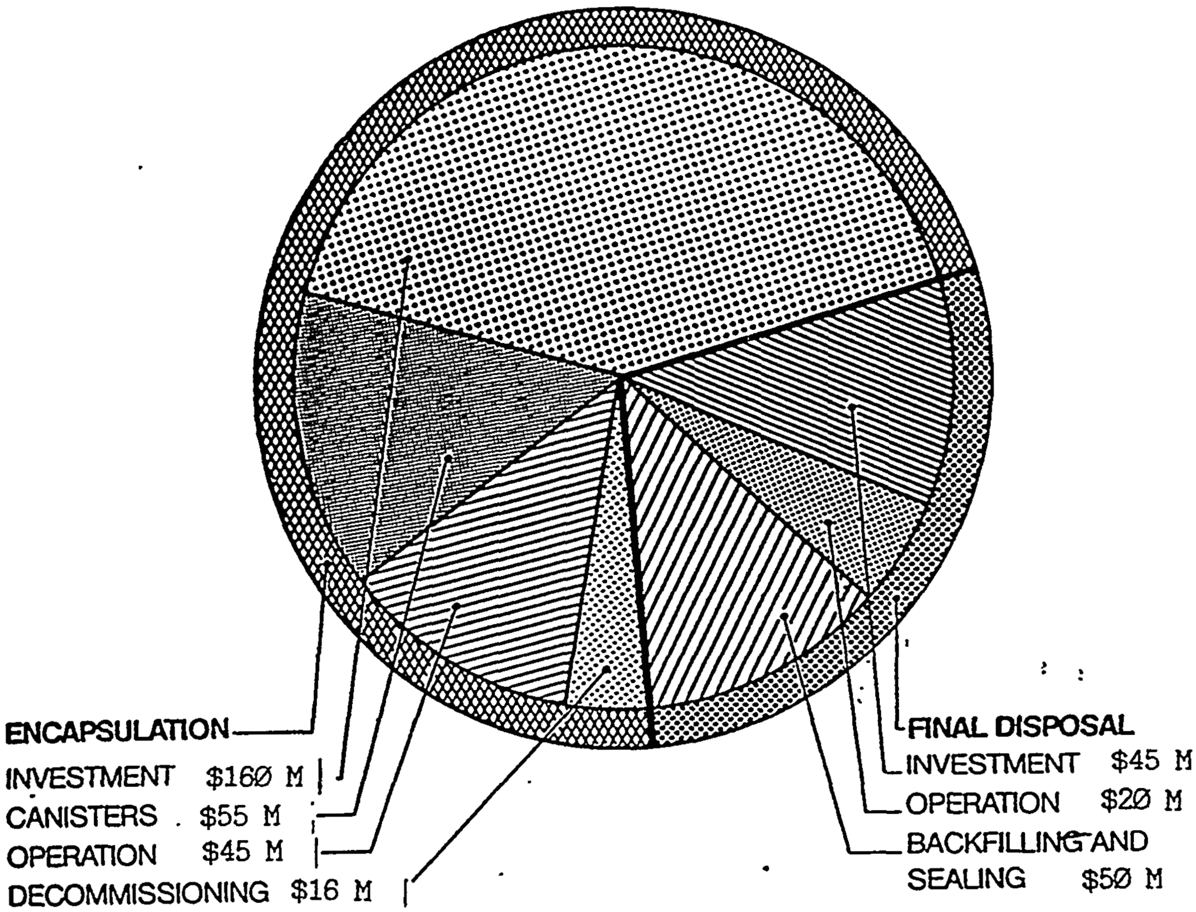

Conversion: 1 $US = 4.6 FIM

Added 2Ø % for inflation, etc. since 1982.

Figure 6.4.1Ø. Spent Fuel Disposal Cost Breakdown encapsulation process is shown in Figure 6.4.5. The spent fuel arrives in special transport casks by truck or rail at the reception hall where the casks are unloaded, washed, and checked for radiation leaks. The cask is then lowered into the unloading pool where the spent fuel assemblies are removed. The assemblies will then be dismantled and the fuel rods placed into a copper canister (about 500 rods fit into one canister). The copper canister is then placed into the furnace area where it is filled with lead. Finally, the canister will be fitted with a welded lid, and, after a thorough inspection, the finished canister will be transported underground to the repository. The spent fuel assembly structural components remaining after the dismantling (fuel hulls or channels, spacers, tiebars, etc.,) will be compacted or cut apart to reduce their volume and transferred to another part of the encapsulation facility containing the concreting equipment. The structural components will be placed in molds and cast into concrete blocks about 0.5 to 1 m^3 (17 to 34 ft^3) in volume. Figure 6.4.11 shows the concrete casting process. It was estimated that about 850 copper canisters will be required for encapsulation of the spent fuel and about 200 concrete molds will contain the structural components.

6.4.5.2 Waste Transport

The transport system of the Finnish waste disposal program for transfer of the spent fuel from the reactor to the interim storage facility and to the repository will most likely be derived from systems in other countries. Transport can be by sea, rail or truck, depending on the location of the various facilities, local conditions, and policies and regulations regarding nuclear waste transport. Special vehicles and transport casks will be used to ensure the highest order of safety.

6.4.5.3 Waste Package

Figure 6.4.12 shows the copper canister which will be used for the encapsulation of the dismantled spent fuel elements. The canister will contain about 500 fuel rods and have a wall thickness of over 200 mm (8 in). The diameter of the canister will be about 800 mm (32 in) and the length 4,700 mm (15.5 ft). Approximately 850 of these units will be required.

The total amount of metal waste (fuel hulls, etc.,) is estimated to be about 100 m^3 (130 yd^3). These parts will be encased on concrete in special cube-shaped molds about 1.6 m (5.5 ft) on a side. Each mold will weigh about 10 tons and approximately 200 will be required. The walls of the molds will be about 30-40 cm (12-16 in) thick to provide adequate radiation shielding.

6.4.5.4 Underground Waste Emplacement Sequence

The waste canisters will be taken underground directly from the encapsulation facility. The waste shaft will be equipped with a special cage and hoist to handle the shielded waste canisters.

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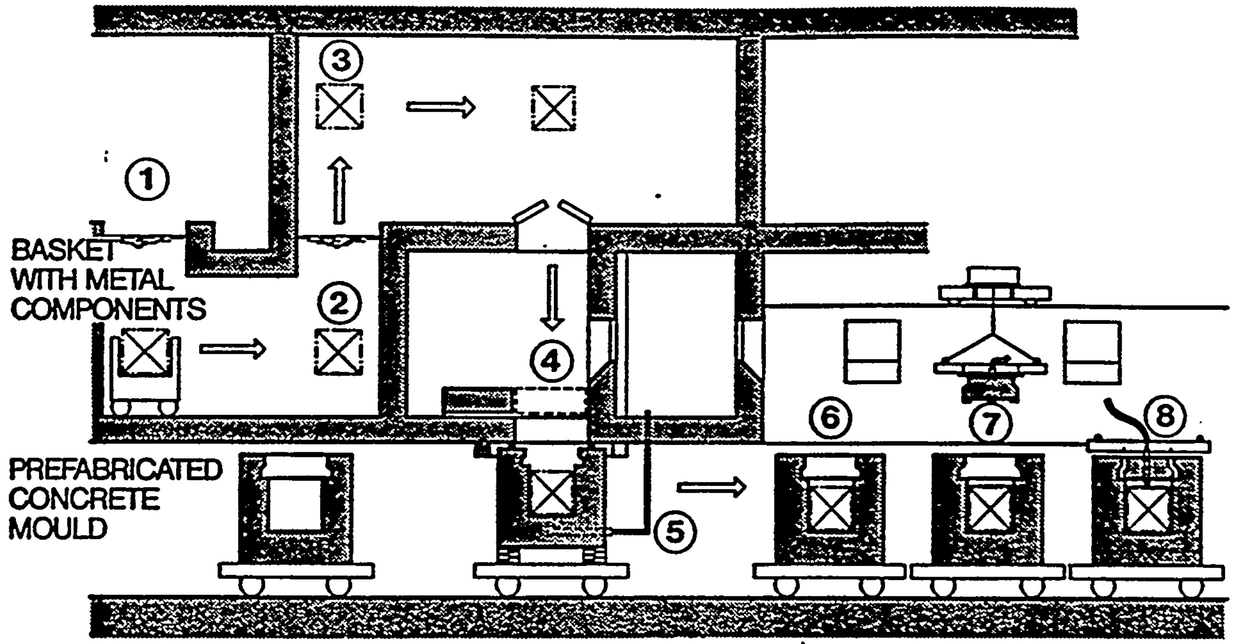

1. STATION FOR RECEPTION, CHOPPING AND COMPACTION OF METAL COMPONENTS
2. TRANSFER VIA WATER POOL
3. DRAINAGE OF TRANSPORT BASKETS
4. PLACEMENT OF TRANSPORT BASKET IN CONCRETE MOULD
5. INJECTION-GROUTING OF TRANSPORT BASKET IN CONCRETE MOULD
6. TRANSPORT TO SEALING STATION
7. MOUNTING OF LID
8. INJECTION-GROUTING OF SPACE AROUND LID

Figure 6.4.11. Casting Metal Components into Concrete Molds

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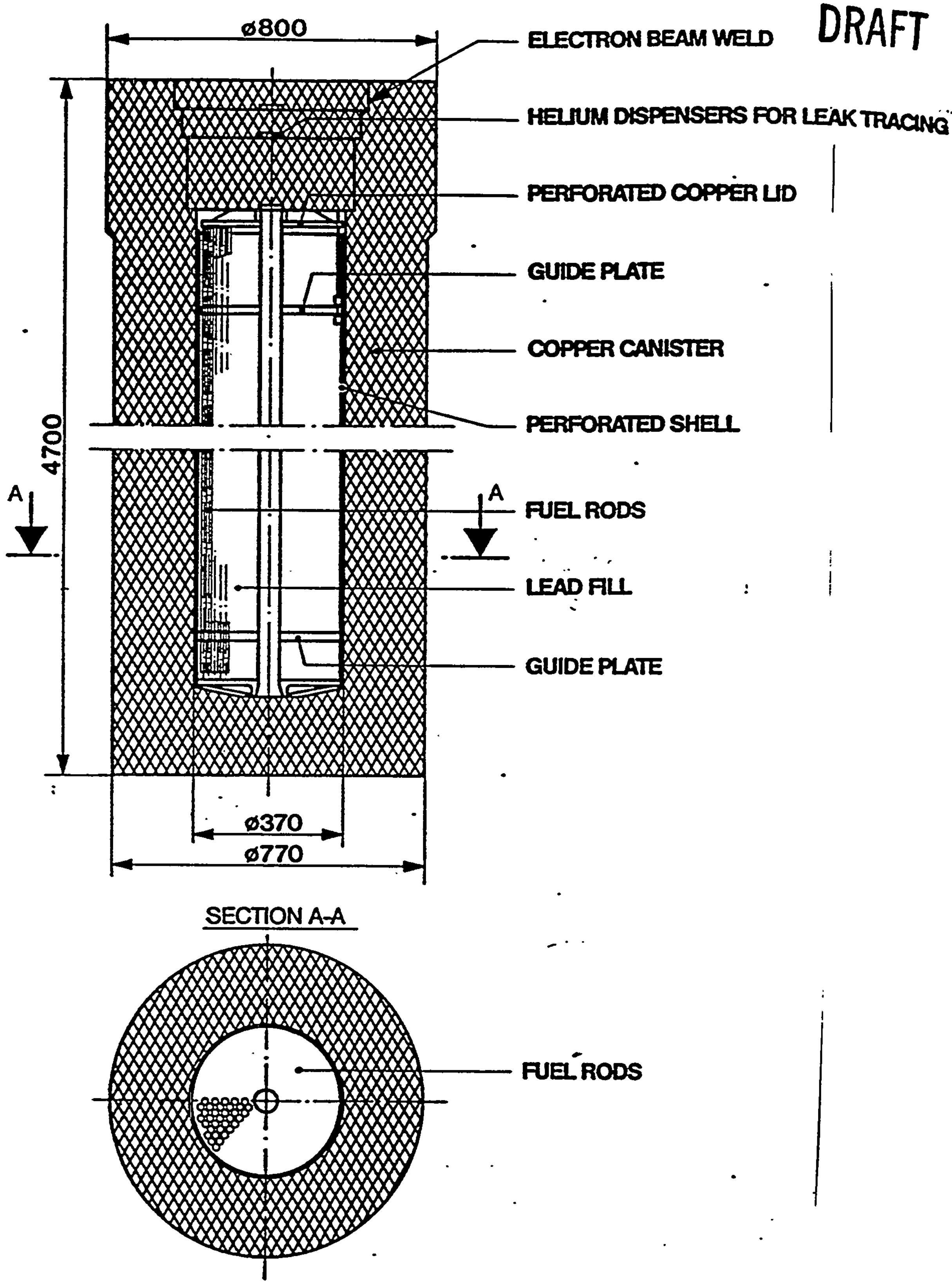

Figure 6.4.12. Copper Spent Fuel Canister

Figure 6.4.13 shows the transport and handling sequence of the waste canisters in repositories A and B. A special transport vehicle will pick up the container at the bottom of the shaft and transport it to the prepared disposal hole. The disposal hole location will have been previously investigated to ensure that it will be suitable and the ground around it grouted to seal any cracks. The hole will also have been filled with donut-shaped compacted bentonite blocks. The transport vehicle will lower the canister into the bentonite-lined borehole and the remaining gaps between the borehole wall, blocks and canister backfilled with bentonite powder. The borehole will then be filled to the top with bentonite blocks and sealed with lid. The tunnel will be backfilled once all the boreholes in it have been filled. Figure 6.4.14 shows the completed disposal of a waste canister.

The emplacement system would be basically the same for repository C as shown in Figure 6.4.9 except that the waste canister would be pushed from the shielded transport carrier into a sleeve embedded in the tunnel in the sand/bentonite mixture in advance. Backfilling would thus proceed as emplacement takes place.

6.4.5.5 Emplacement Schedule

The disposal operations were estimated to start in 2020 and last for about 30 years as shown in Figure 6.4.3. This equates to an annual emplacement rate of about 30 waste canisters.

6.4.6 Buffer and Sealing Materials

The proposed backfill and sealing materials for the repository alternatives discussed above are bentonite and bentonite/sand mixtures. Figures 6.4.9 and 6.4.14 show the bentonite/sand mixture in the tunnels and the bentonite blocks in the boreholes, respectively, as they are used to stabilize and isolate the waste canisters. Bentonite has several favorable properties as buffer material and as backfill and seal material. As a buffer material, bentonite:

- Has the mechanical properties to hold the canister in position
- Has the chemical properties to retard the corrosion of the canister
- Has the thermal properties to transmit the heat generated by the fuel to the surrounding rock mass
- Has the ion exchange capacity to retard the migration of the radionuclides
- Swells as it contacts moisture and thus fills void and seals joints and cracks in the bentonite itself and in the surrounding rock mass.

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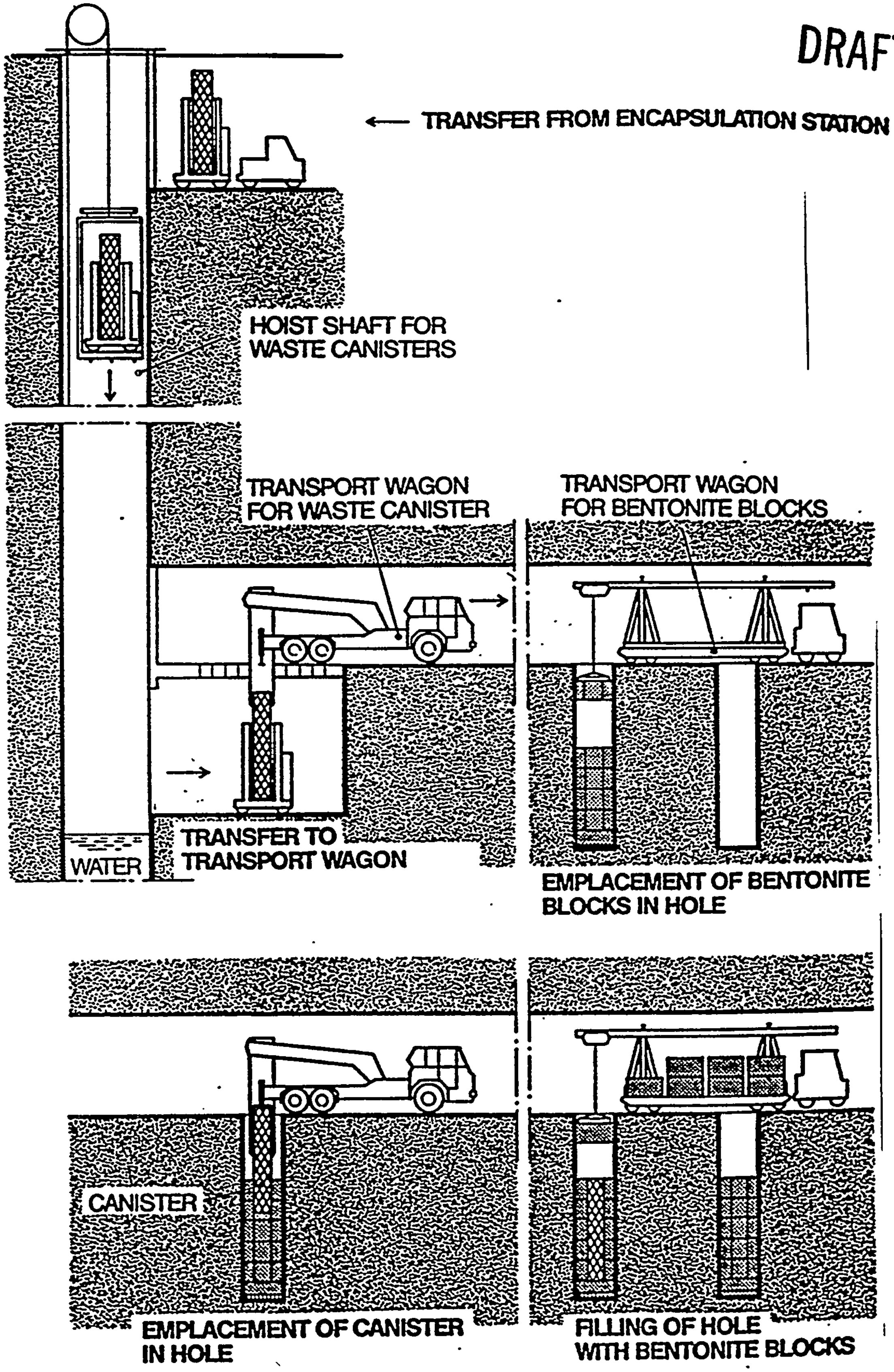

Figure 6.4.13. Disposal of Canisters in Repositories A and B

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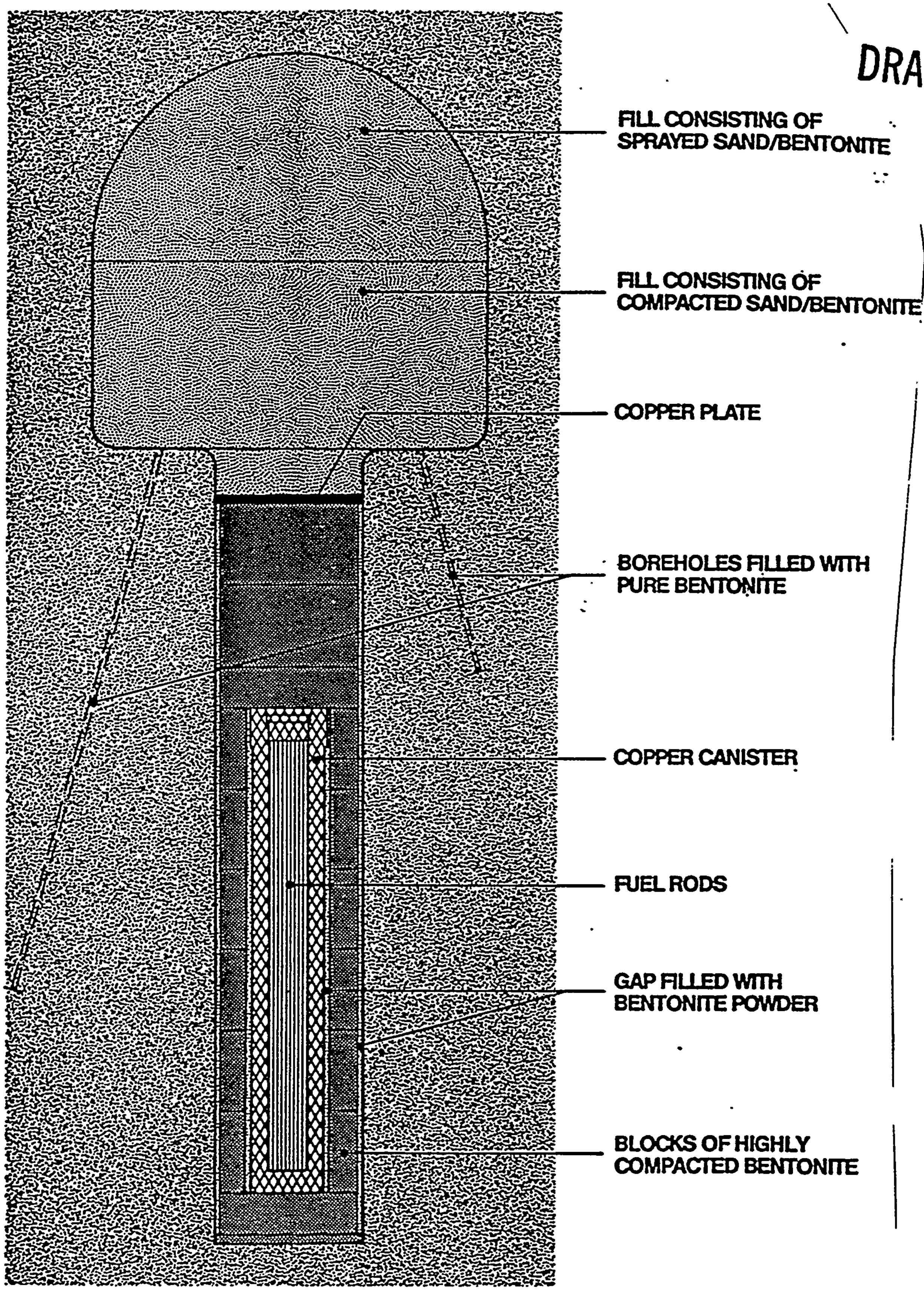

Figure 6.4.14. Disposal Borehole for Repositories A and B

As a backfill and sealing material, bentonite mixed with sand:

- Has a low hydraulic conductivity
- Is plastic so it can seal cracks and joints
- Swells on contact with water and thus acts as a sealant
- Is chemically stable up to about 100°C (212°F)
- Can be used to backfill tunnels completely by compaction or spraying as well as sealing off fractures and shaft and boreholes (see Figures 6.4.15 and 6.4.16).

Once all canisters have been disposed of in the repository, all underground excavations including the shafts will be backfilled. Figures 6.4.15 and 6.4.16 show a possible method using bentonite and bentonite/sand mixtures as sealants and backfill. The properties of bentonite and other materials as buffer materials, seals and backfill materials will continue to be investigated to better understand the properties and also to find the best material for the job.

6.4.7 Summary

6.4.7.1 Advantages

1. The multiple-barrier systems are being considered by several countries (e.g., Sweden, Switzerland, Canada) in their underground disposal systems. The addition of the engineered barriers to the natural ones will add to the overall effectiveness of the isolation system.

2. Crystalline rocks occur in massive formations to great depths and therefore offer a wide choice of sites and environments for the location of the underground repository.

3. A large amount of mining, tunneling and construction expertise in crystalline rocks exists in the world which is essential for the design and construction of the repository.

4. Excavations in crystalline rocks can remain open for long periods of time with little maintenance due to the competent nature of the formations.

5. The application of mechanized excavation methods for shaft and tunnel construction has many advantages such as speed of excavation, minimizing of damaged zone around the excavations, etc.

6. Extended storage of the spent fuel before disposal underground reduces the heat and radioactivity output of the waste and thus reduces the size and waste handling requirements in the repository.

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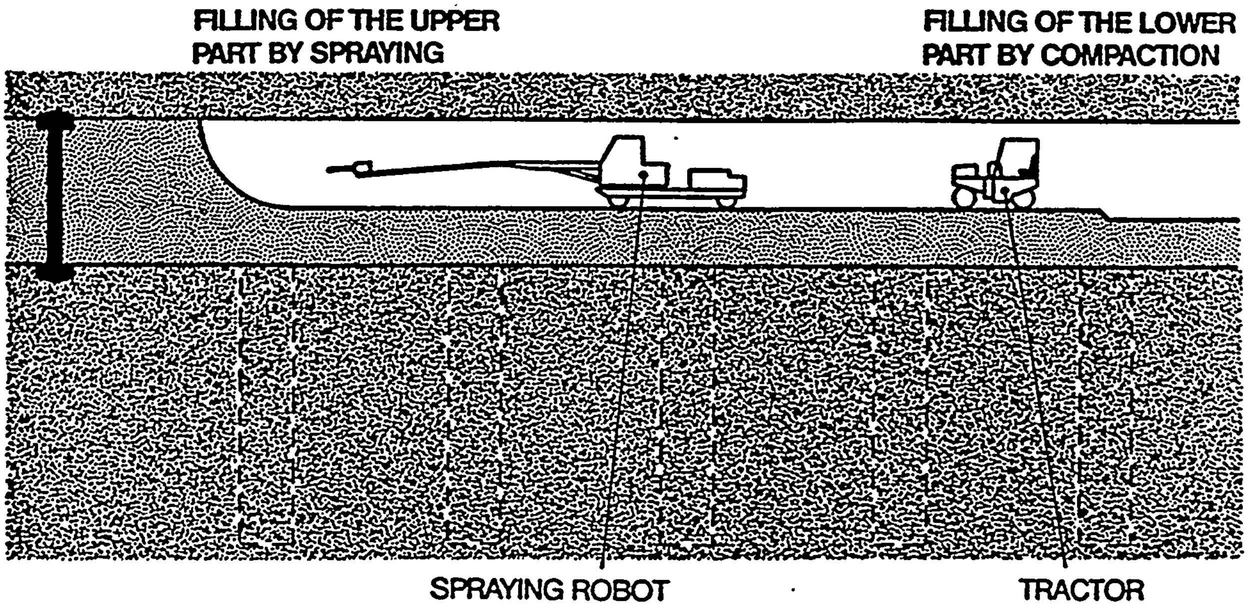

Figure 6.4.15. Backfilling of Tunnels

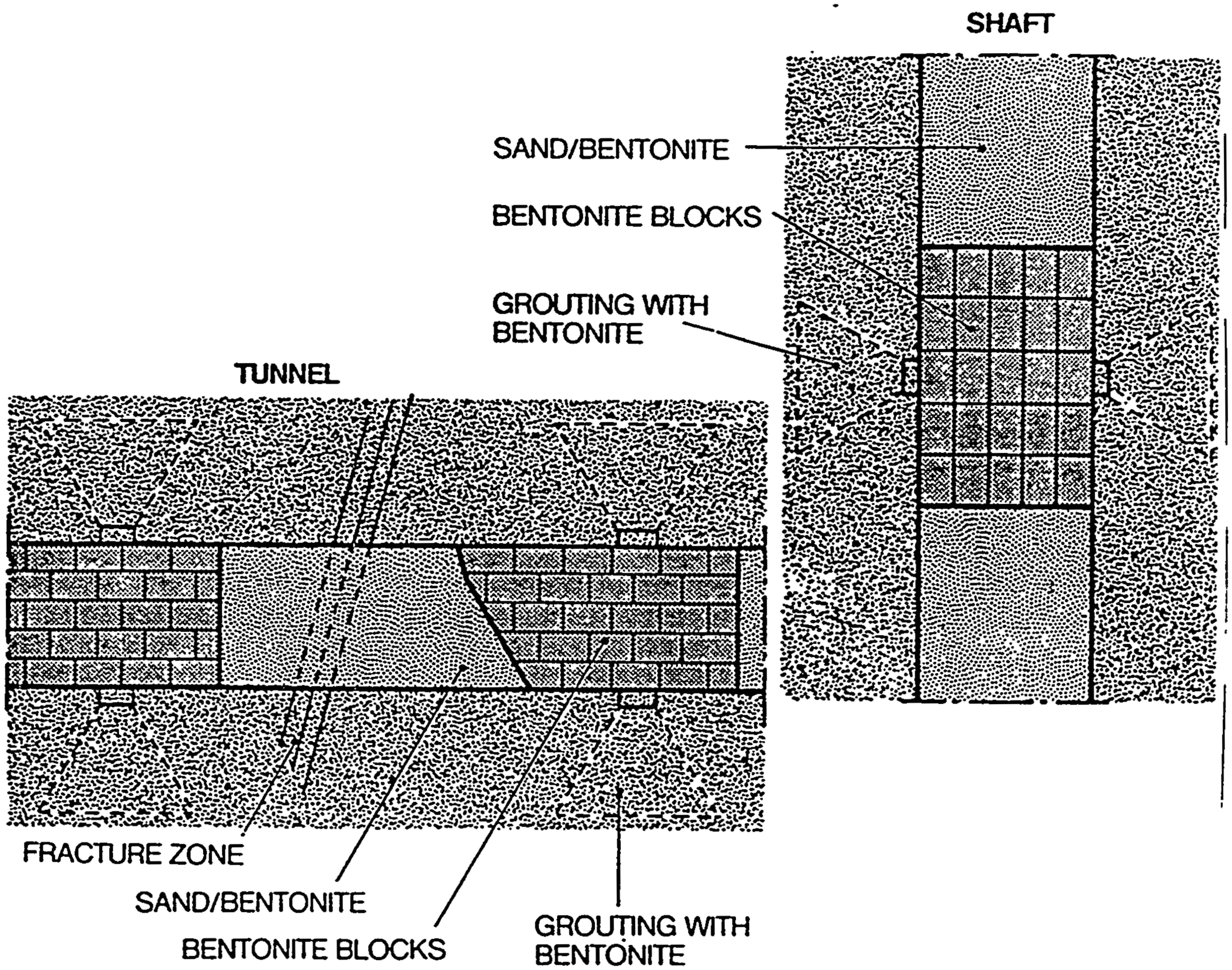

Figure 6.4.16. Sealing of Fracture Zones and Shafts

6.4.7.2 Potential Problems

As with all underground disposal concepts, one of the major drawbacks is the limited knowledge that exists today regarding the long-term behavior of the natural and engineered barriers. Major efforts are underway in many countries to resolve these issues.

6.4.7.3 Summary

Since crystalline formations are also under consideration for the United States high-level waste disposal program, the efforts of the Finnish authorities are also of relevance to the U.S. program.

6.4.8 The Low- and Intermediate-Level Concepts

The low- and intermediate-level wastes from the reactor operations are currently being stored at the two power plant sites. At the Loviisa plant the liquid wastes are stored in liquid form in storage tanks. A solidification plant based on the cementation process has been designed and construction is due to start in 1988 and commissioning in 1992. At the Olkiluoto plant the spent ion exchange resins are solidified with bitumen. The bituminized wastes are stored at the power plant and in a storage facility. At both power plants dry low-level wastes are compacted into 200 l (55 gal) steel drums. The table below summarizes the waste quantities estimated to arise by the year 2010:

		Loviisa	Olkiluoto
Wastes to be disposed of in m^3 by 2010	Dry	2,800	4,300
	Wet (solidified)	7,400	2,600

The current proposal is to construct a repository at each of the power plant sites. The repositories were designed for the disposal of all the low- and intermediate-level wastes produced during the operation of the reactors. The location of the repository at each site is determined by the geological conditions and provisions for possible later enlargement of the repository to accept decommissioning wastes.

The repository at the Loviisa site will be constructed in a zone which is suitable with respect to ground-water flow. It will be constructed in granite bedrock at a depth of about 120 m (400 ft) in a zone of stagnant saline water between two zones of higher permeability. A revised repository layout was prepared in 1985 and is shown in Figure 6.4.17. The repository consists of separate caverns and tunnels for different waste types. A tunnel connects the auxiliary facilities with the waste disposal areas and two shafts will provide the ventilation for construction and operation of the facility. The transport tunnel connects the power plant area with the

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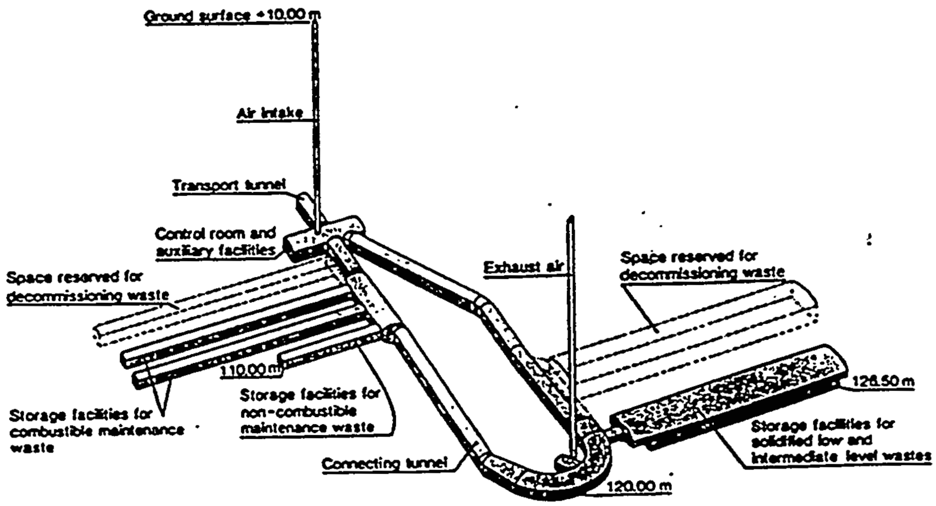

Figure 6.4.17. Configuration of the Loviisa Repository

underground repository. The length of the transport tunnel is about 1,000 m (3,280 ft) and it will have a cross section of about 40 m^2 (430 ft^2). The tunnel will terminate in the control and auxiliary facilities at about 110 m (350 ft) below the surface. Three tunnels with a 25 m^2 (270 ft^2) cross section will house the dry maintenance wastes. Combustible and non-combustible wastes will be stored in separate tunnels. A large waste cavern at about 120 m (400 ft) depth will be used for the disposal of all solidified wastes. The cavern will be about 100 m (330 ft) long and have a cross section of about 300 m^2 (3,230 ft^2).

The Olkiluoto repository will be constructed between 50 and 100 m (165 to 330 ft) underground in a tonalite formation (Figure 6.4.18). The repository will consist of two waste disposal silos plus the auxiliary facilities. One silo will be used for the bituminized waste and the other for the dry low-level wastes. The silos will be about 22 m (72 ft) in diameter and 25 m (82 ft) high. The silo for the bituminized waste will contain an extra concrete barrier of reinforced concrete about 500 mm (1.7 ft) thick. The waste packages will be transported to the repository via an access tunnel while personnel and services access will be via a shaft. Waste packages will be lowered by overhead crane into the silos.

The repository will be backfilled and all access sealed once the repository is full.

6.5 CRYSTALLINE ROCKS: THE CANADIAN CONCEPT

This section outlines the reference concept currently being considered by the Canadian nuclear waste disposal program for the disposal of spent nuclear fuel. Canada currently has 17 operating nuclear power plants: 15 are owned and operated by Ontario Hydro and are located in Ontario while one is located in Quebec (Hydro Quebec) and one in New Brunswick (New Brunswick Electric Power Commission). Four more reactors are also under construction. The reactors are all of the CANDU (CANada Deuterium Uranium) type which use natural uranium as fuel. The current strategy for disposal of nuclear wastes is to dispose of the immobilized spent fuel, after an interim storage period, in a deep mined repository in plutonic rocks in the Canadian Shield. Other options such as reprocessing of the spent fuel and disposal in other rock formations such as shale and salt are also under consideration but at a low level of effort.

Prior to choosing an actual site for site characterization as a possible repository location, a series of studies and reviews will be conducted to evaluate the proposed spent fuel disposal concept. Concept development and evaluation is currently ongoing at the Atomic Energy of Canada Ltd., through generic research and external reviews. A formal concept assessment document is scheduled to be published in 1988 and will outline the latest concept for the disposal of the spent fuel elements and assess the overall impact on man and the environment if it is implemented. Review of this concept will proceed in three stages: Stage 1 will be a review by the Atomic Energy Control Board, in consultation with the appropriate experts and authorities, to ensure that the concept meets regulatory

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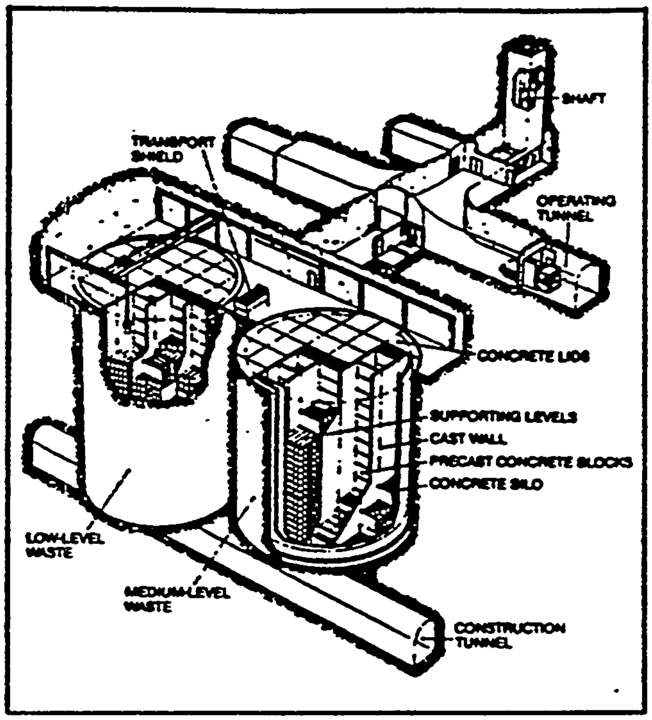

Olkiluoto Repository Waste Vault and Silos

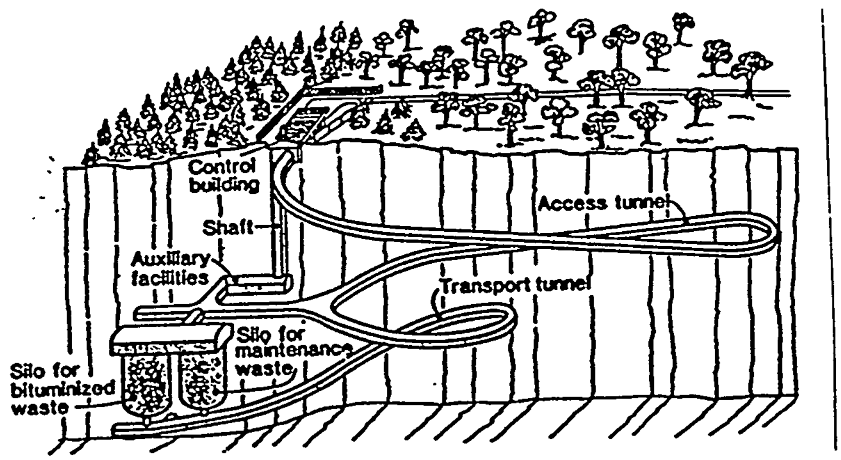

Figure 6.4.18. Configuration of the Olkiluoto Repository

guidelines and criteria, and the results of the analysis will be available to the public as well as the government; Stage 2 will be a public hearing which is to result in recommendations on the acceptability of the concept and the concept assessment; and, Stage 3 will be the decision by the government, based on the recommendations of the Atomic Energy Control Board and the outcome of the public hearing, on the acceptability of the concept. Only if the concept is accepted by the government will actual site selection proceed. A target date for a decision on the concept is 1991.

The Atomic Energy of Canada Ltd. is a crown corporation and is responsible for demonstration of the feasibility of the final disposal concept and has been conducting related research at an Underground Research Laboratory excavated in an exposed batholith in Manitoba since about 1982. The Atomic Energy of Canada Ltd., will prepare the formal concept assessment documents for review and evaluation. The concept described in the following sections was developed by Atomic Energy of Canada Ltd. and experiments are being conducted and planned at the Underground Research Laboratory to demonstrate the viability of this concept. The United States Department of Energy is also involved in the Underground Research Laboratory project since crystalline rocks are one of the rock types under consideration for the U.S. repositories. Therefore, the work at the Underground Research Laboratory is directly related to the U.S. program.

6.5.1 Geographic Location

Potential repository sites will not be selected for site characterization until a disposal concept has been finalized and accepted. The concept described in this section is based on a mined repository located in a plutonic host rock forming part of the Canadian Shield. Figure 6.5.1 shows the extent of the Canadian Shield and the locations of research areas where studies are being conducted to support the concept development program.

6.5.2 Summary of Geologic Setting

Underground disposal of spent fuel will involve the placement of the waste package in a mined repository within a suitable geologic formation to provide long-term radiation shielding, absorb and dissipate heat, and prevent intrusions by man. Studies are being conducted into various geologic formation in Canada, as well as other countries, to find the most suitable environment. The formations being studied in detail in Canada are plutonic rocks of granite or gabbro. Plutonic rocks are large, solidified intrusions of magma into the earth's crust which have great structural strength, resist erosion and normally do not contain significant concentrations of valuable minerals. The Canadian Shield (Figure 6.5.1) contains an abundance of plutonic rocks which are fairly homogeneous and have been stable for millions of years. Nuclear waste disposal in this type of formation is considered technically and economically feasible since construction of the underground facilities could be carried out using the mining technology already existing in Canada. Initial analysis has also

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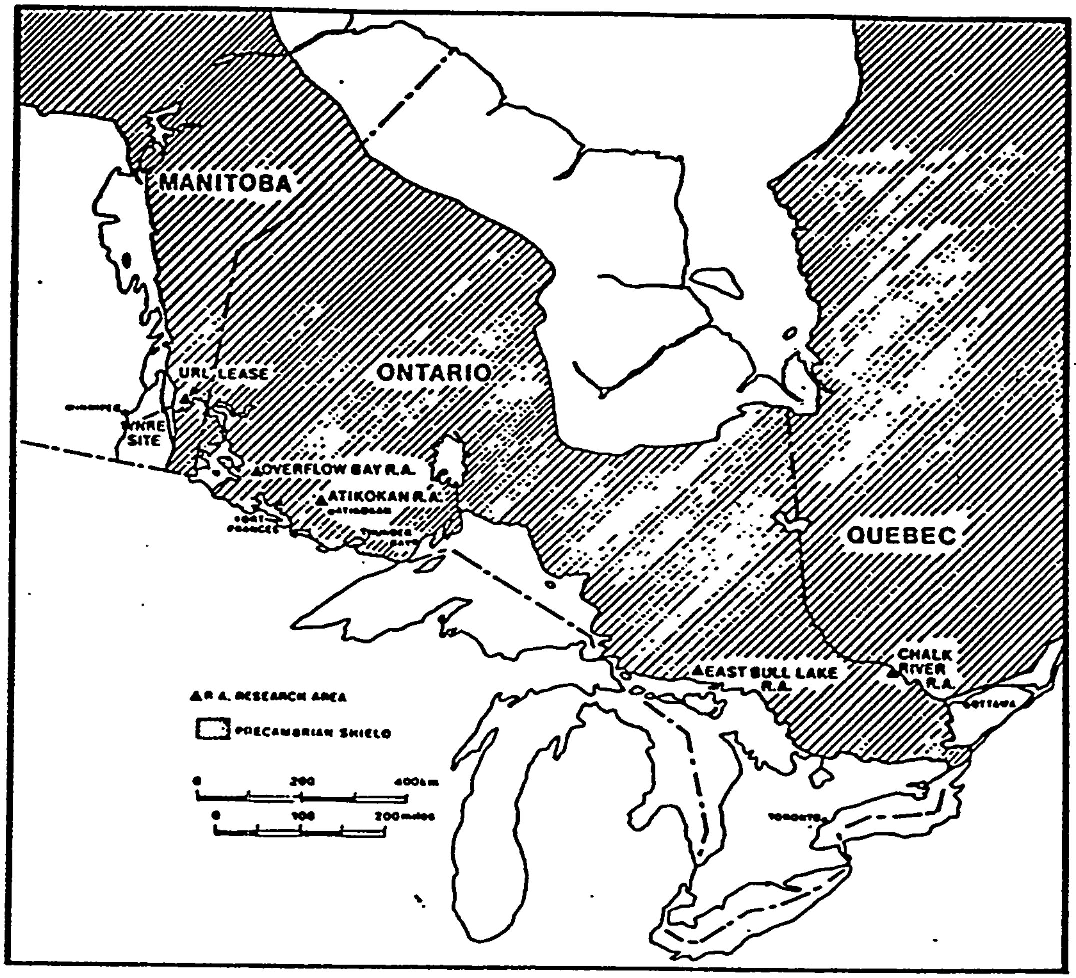

Figure 6.5.1. The Canadian Precambrian Shield

shown that the plutonic rocks would provide the required containment for the nuclear wastes. It is for these reasons that the concept currently under investigation in Canada is the disposal of high-level wastes in plutonic rocks.

6.5.3 Repository Concept Description

6.5.3.1 Nuclear Waste Disposal Strategy

The 17 nuclear power reactors currently in use in Canada are fueled by CANDU fuel elements, using natural uranium as fuel. A CANDU fuel bundle contains about 20 kg (44 lb) of uranium in the form of uranium oxide pellets encapsulated in Zircalloy tubes. Figure 6.5.2 shows the CANDU spent fuel bundles used in the Canadian reactors. Spent fuel elements are stored in water-filled pools at the reactor sites. The water provides cooling as well as radiation shielding and this method has been used safely for over 30 years. The amount of spent fuel expected to arise by the year 2000 is about 33,900 MT (37,300 tons) with a total production of an estimated 191,000 MT (210,000 tons) by the year 2035 based on current estimates of the growth of Canadian nuclear power production. In Canada, spent fuel is currently not being reprocessed and the spent fuel disposal option being considered for long-term isolation is the disposal in an underground geologic repository.

The disposal concept being studied is a multiple-barrier system comprised of a low-solubility waste form, a corrosion-resistant container, a low-permeability clay-based buffer, and a stable surrounding geosphere, separating the disposed waste from the biosphere (Figure 6.5.3). Figure 6.5.4 shows the various engineered barriers and components proposed for the underground repository. The spent fuel bundles will be aged for at least 10 years before being placed in special disposal canisters. The canisters will be taken underground and placed in vertical boreholes drilled into the floor of specially excavated disposal chambers. The boreholes and disposal rooms will be backfilled as they are filled and finally, the access drifts and shafts will be sealed and surface facilities decommissioned and the area rehabilitated.

6.5.3.2 Surface Facilities

The major surface facilities for the handling and disposal of the spent fuel includes the spent fuel storage facilities at the reactors, the transport system and the repository site facilities.

Spent fuel will be transported in special transport casks using rail, truck, or barge transportation.

The primary surface facilities at the repository site will include the fuel immobilization building, the waste shaft headframe facility, the service shaft building and the repository support services. Figure 6.5.5

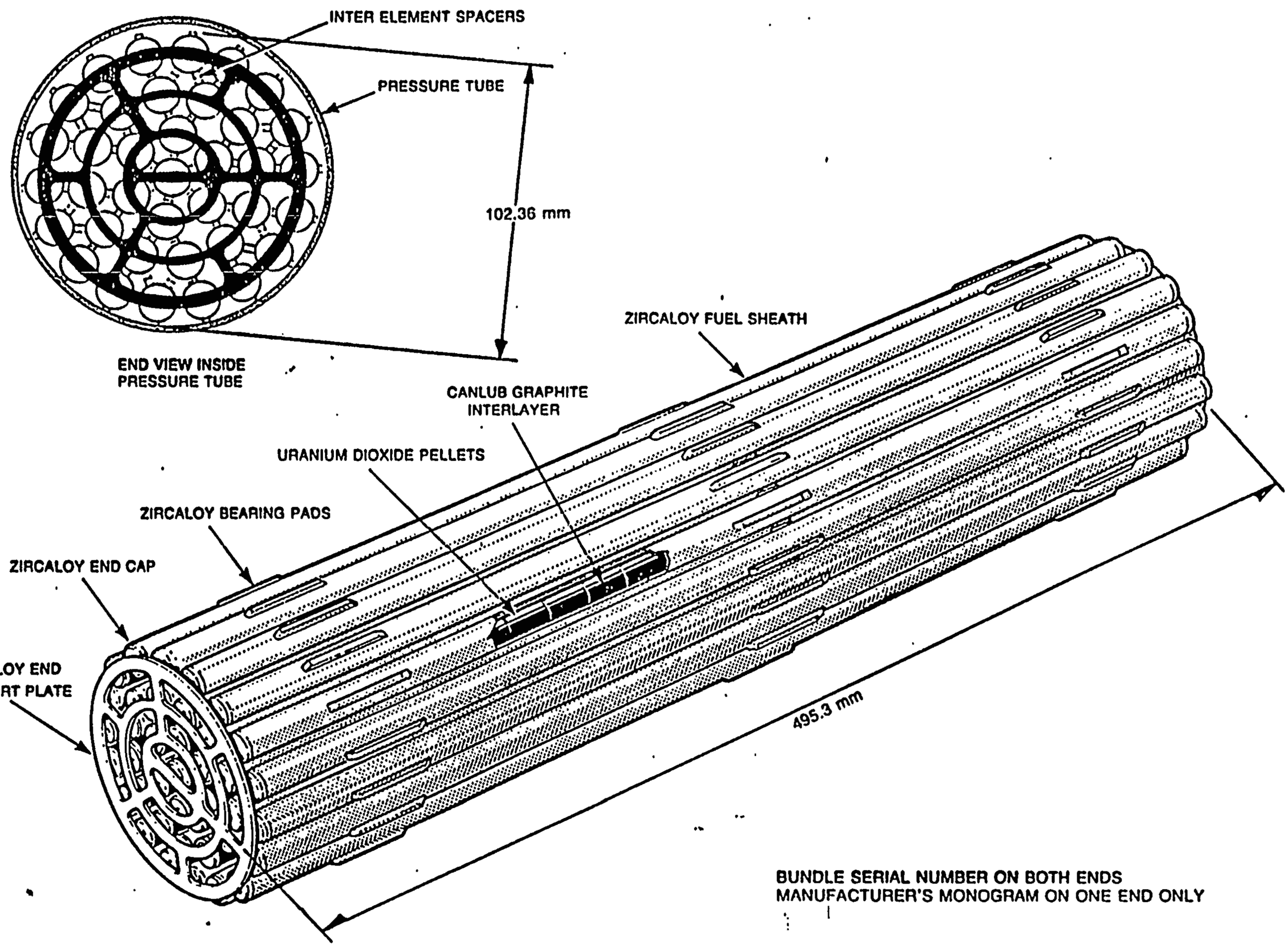

Figure 6.5.2. CANDU Fuel Bundle

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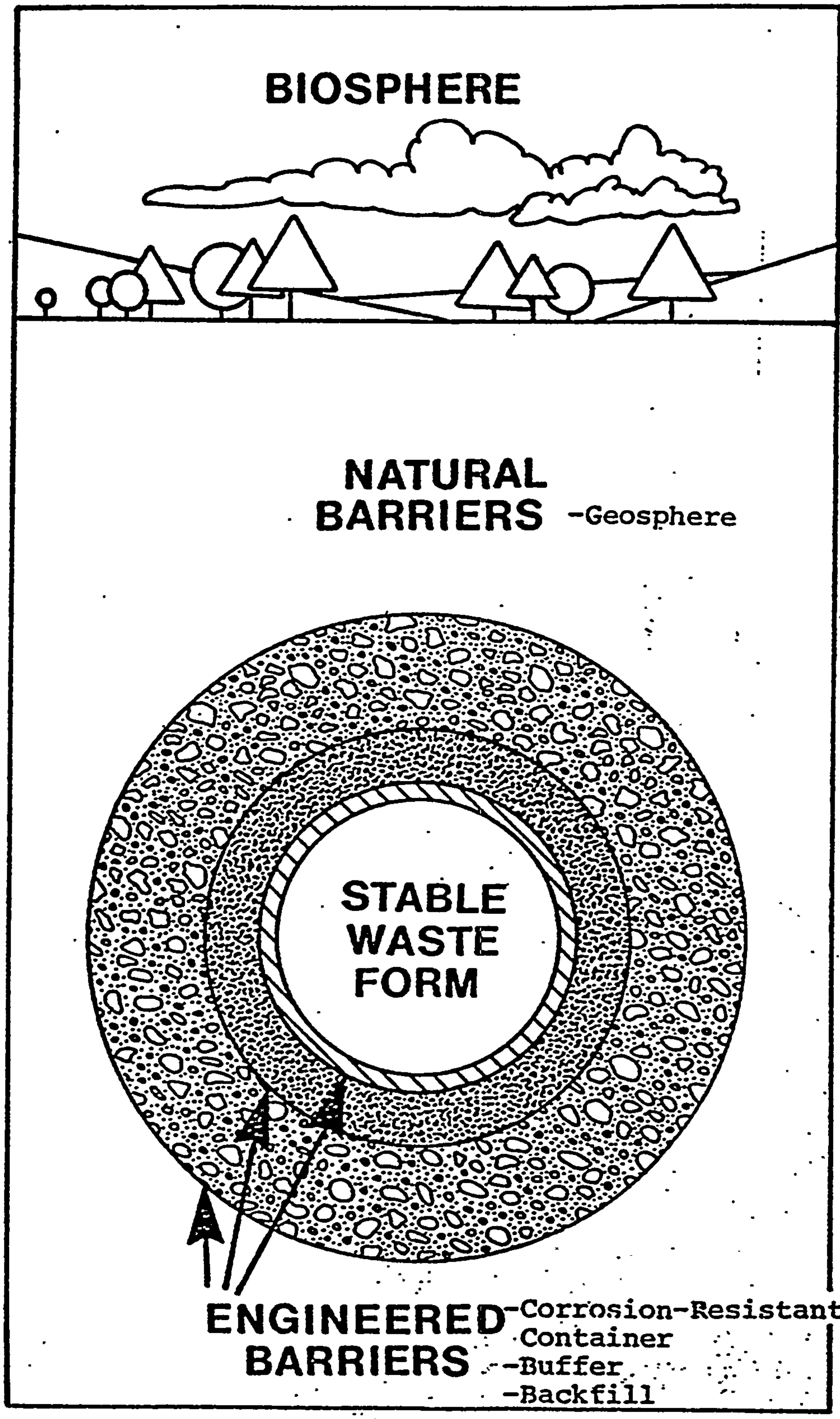

Figure 6.5.3. The Multiple-Barrier System

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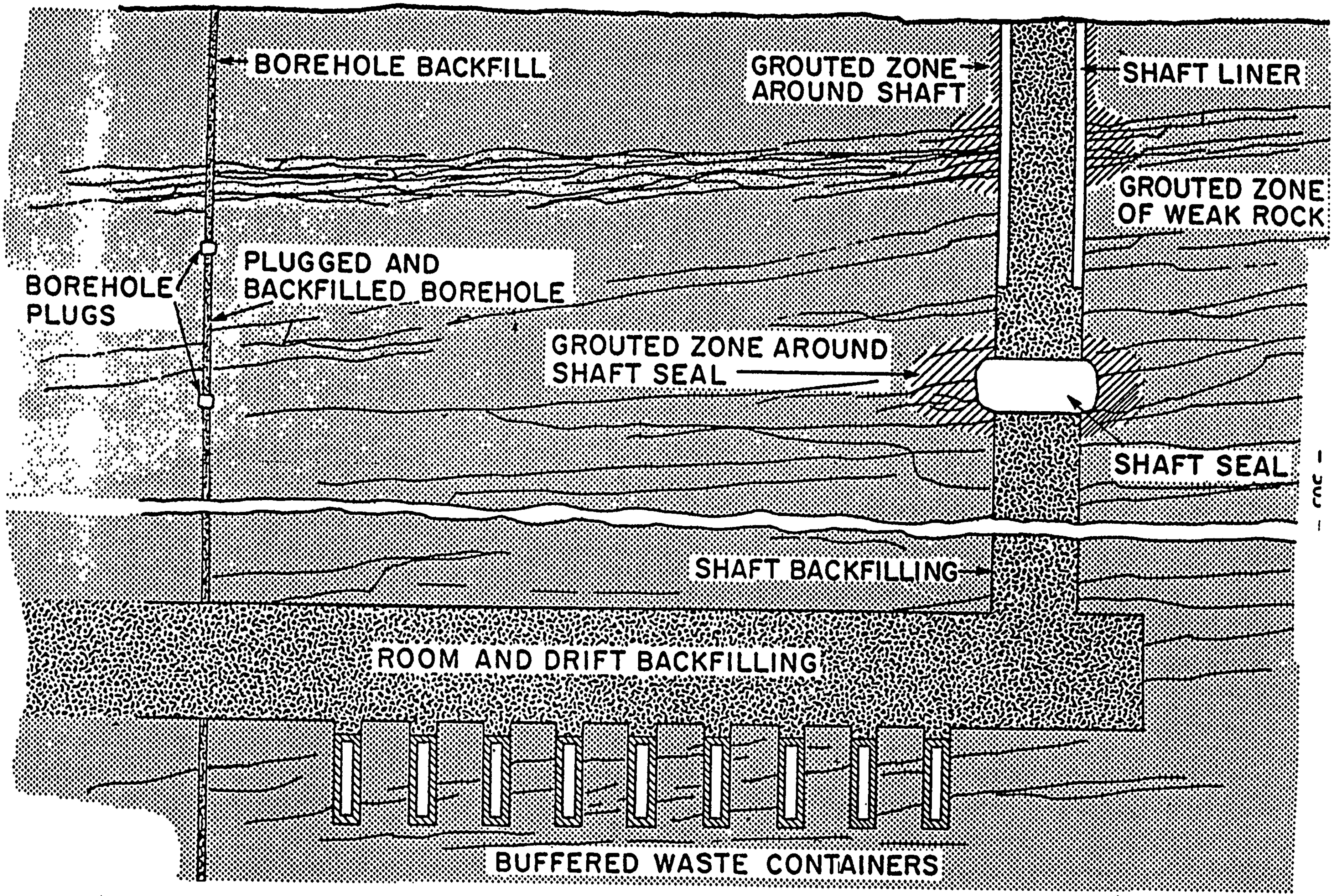

Figure 6.5.4. The Repository Sealing System

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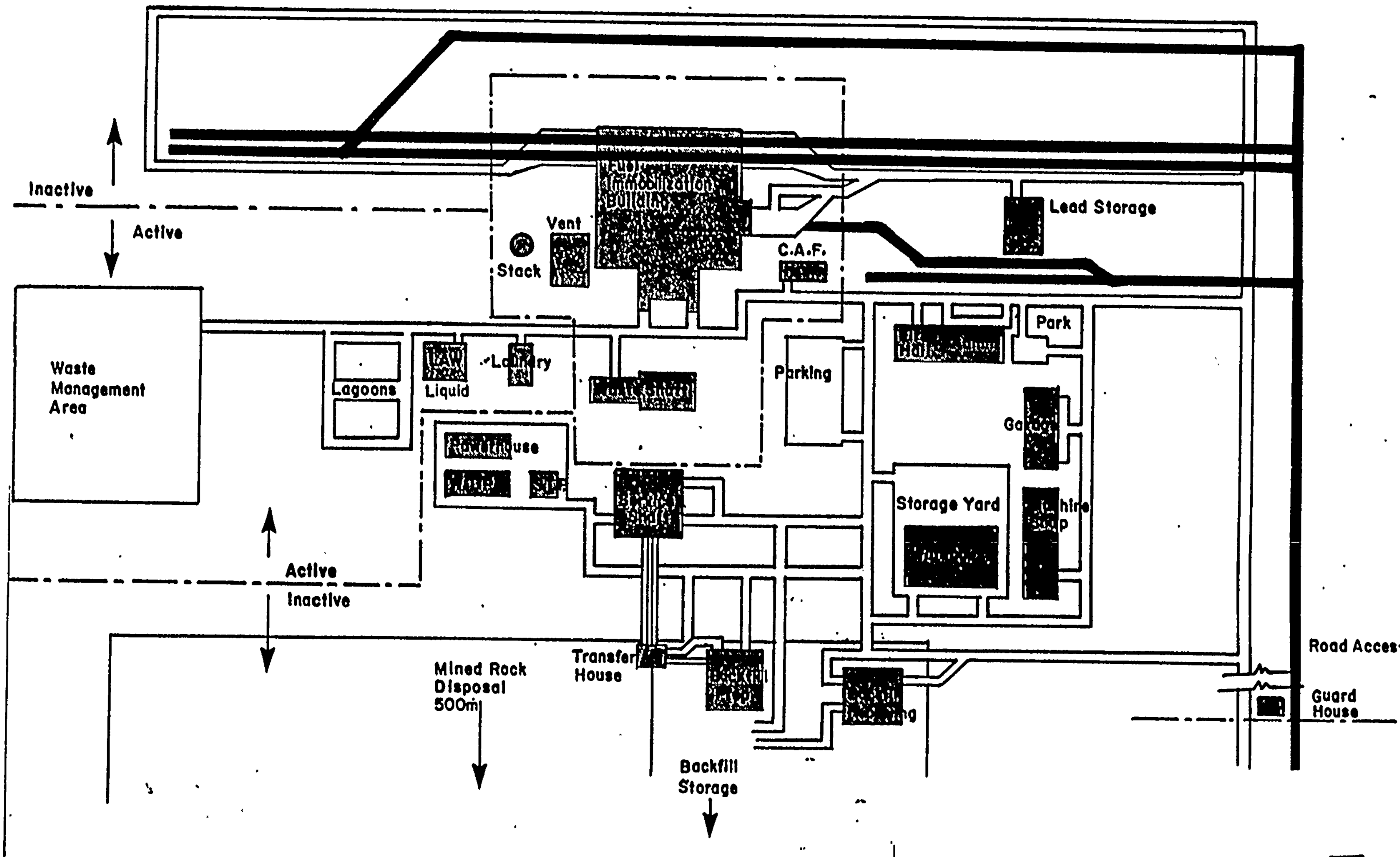

Figure 6.5.5. Repository Site Layout

shows a proposed layout for the surface facilities at the repository. The site area will be about 1 km^2 (250 acres). The buildings have been arranged generally around the waste shaft headframe buildings. The immobilized fuel canisters will be transferred from the fuel immobilization plant to the waste shaft via underground tunnels.

The fuel immobilization building will house two parallel lines of hot-cells and the equipment necessary to receive, package, immobilize and dispatch the spent fuel to the waste shaft headframe, as well as storage for 1 years' throughput of spent fuel. Other facilities included will be the active waste systems, the ventilation systems, shops and offices. The building will be roughly 130 m by 130 m (425 ft by 425 ft) in size and, with a two-level arrangement, have a height of about 25 m (82 ft). The building will have a special ventilation and filtering system to prevent the spread of any radioactivity and ensure a safe working environment for the operating personnel. Figure 6.5.6 shows the floor plan of the proposed facility.

The waste shaft headframe building will be connected to the fuel immobilization facility via tunnels along which the spent fuel canisters will be transported to the waste shaft. The building will contain the hoist/headframe structure, a temporary storage cell, ancillary operations rooms and support services. Figure 6.5.7 shows a section through the waste shaft facility. The canisters arriving from the fuel immobilization facility will be transferred from the transport trolleys by remote controlled overhead crane to the storage area. They will then either remain there temporarily or be taken directly to the shaft for lowering to the repository. The storage cell capacity will be about 810 canisters to compensate for some delays and variances in the operations of the fuel immobilization facility and underground disposal network.

The service shaft building will contain the service hoists and headframe. The service shaft will accommodate the rock skip hoisting system, the man and material service cage and hoist system, an emergency egress system and associated control equipment. A waste rock storage bin will also be provided to act as bunker between the underground and surface operations. Figure 6.5.8 shows the service shaft headframe.

Other support services located at the repository will include:

- Backfill preparation plant and materials storage (the plant may also be placed underground)
- Secondary active waste treatment facility to manage and control radioactive liquid and solid wastes generated at the fuel immobilization facility during the handling of the spent fuel
- Offices
- Change house facilities
- Workshops

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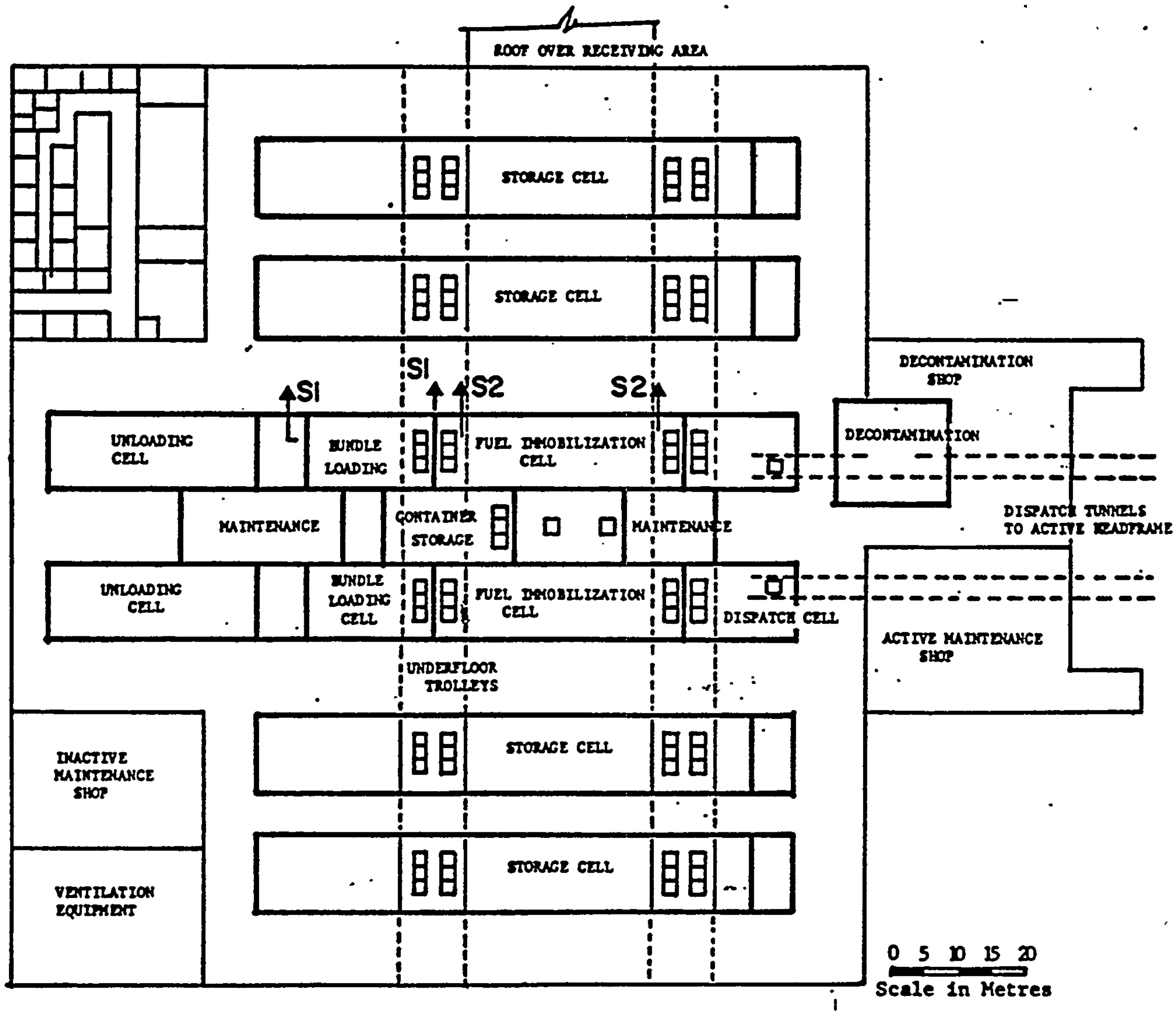

Figure 6.5.6. The Fuel Immobilization Building

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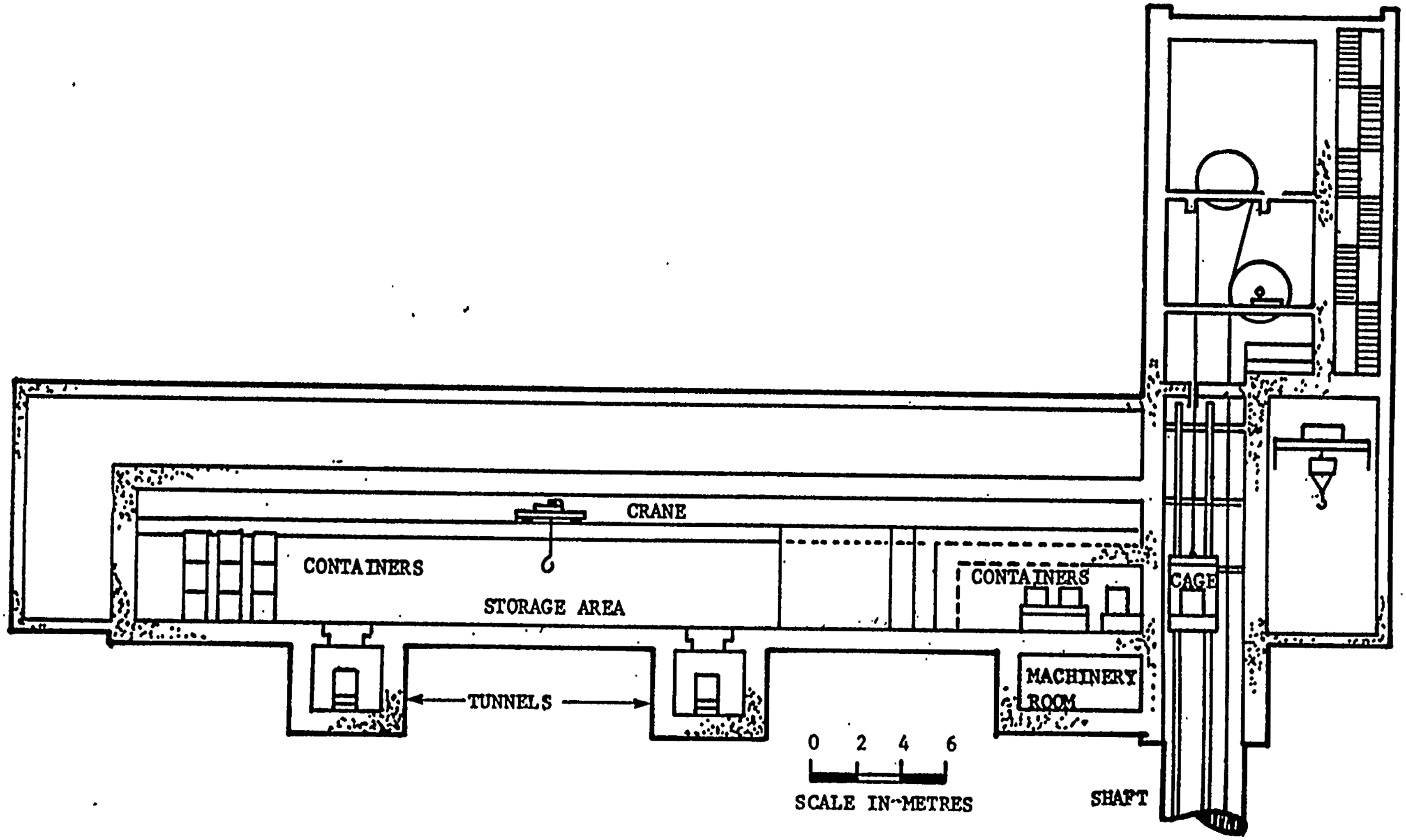

Figure 6.5.7. Waste Shaft Headframe Building

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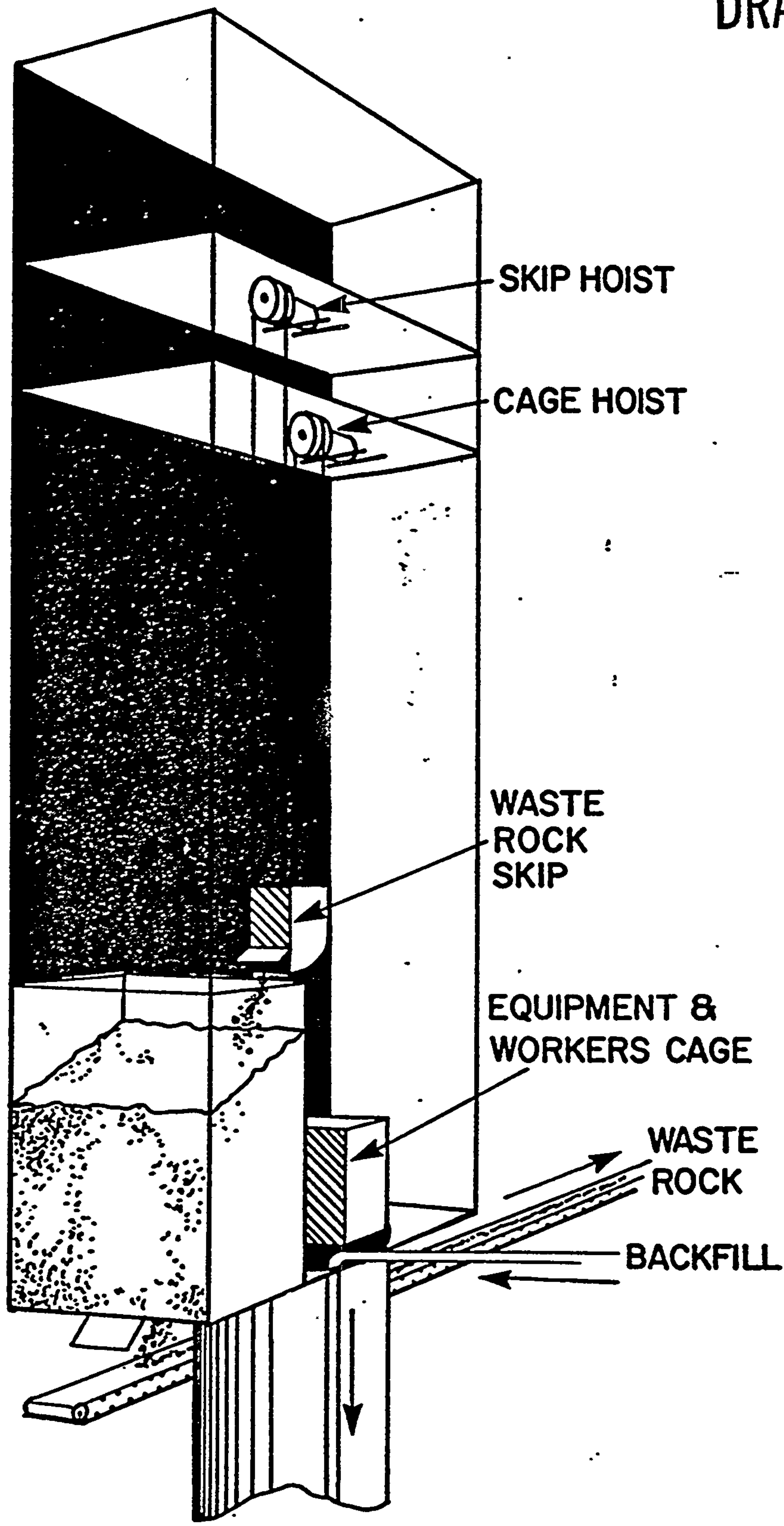

Figure 6.5.8. Service Shaft Headframe

- Underground ventilation systems
- Security, mine-rescue, emergency services
- Storage areas for materials and waste rock
- Compressed air, power, etc., services.

6.5.3.3 Underground Configuration

Figure 6.5.9 shows the general underground arrangement of the repository disposal rooms and access drifts and shafts. The repository covers an area of approximately 2,000 m by 2,200 m (6,560 ft by 7,220 ft) and will be arranged in a roughly square configuration on a single level. The depth of the repository will be between 500 and 1,000 m (1,640 and 3,300 ft) in a suitable plutonic rock formation. Two roughly 2,000 m (6,560 ft) long central access drifts, 6 m wide by 5 m high (20 ft by 17 ft), connect the main shaft complex with the exhaust shaft complex at the other end of the repository. These drifts will serve as the main arteries for the waste canister transport to the disposal area, waste rock removal from the panel development to the rock hoisting (service) shaft, and ventilation intake and return. The main shaft complex (Figure 6.5.10) will encompass the waste shaft, service shaft and intake ventilation shaft plus the support facilities such as service areas, workshops, backfill preparation plant, waste rock handling facilities and waste canister transfer equipment. The exhaust shaft complex (Figure 6.5.11) will encompass two ventilation exhaust shafts. The intake air flow will be divided into two separate flows, one to serve the panel development areas and the other to service the waste emplacement operation. The waste emplacement air stream will be isolated to prevent possible contamination of the rest of the facility. The exhaust shaft complex will therefore include filter and cleanser arrangements for the waste emplacement return air before it is released into the upcast shaft. The other upcast shaft will return air from the rest of the facility to the surface and also act as emergency escapeway.

Development of the waste disposal areas will take place from the central access drifts and the perimeter drifts excavated around the proposed repository (Figure 6.5.12). Two parallel panel drifts about 30 m (100 ft) apart will be excavated between the central access drifts and the perimeter drifts and from these the waste emplacement rooms will be excavated (Figures 6.5.12 and 6.5.13). Approximately 29 emplacement rooms will be excavated from each panel drift to form the panel. The 8 m (26 ft) wide and 5.5 m (18 ft) high rooms (Figure 6.5.13) will be about 250 m (820 ft) long and spaced at about 30 m (100 ft) intervals. The actual disposal area of the emplacement rooms will be about 220 m (720 ft) long (Figure 6.5.13) and the emplacement boreholes will be drilled into the floor of the rooms at a spacing of about 2.1 m (6.9 ft). This means that there will be room for three parallel rows of holes along the axis of the room as shown in Figure 6.5.13. This pattern of holes will yield about 305 holes (i.e., canisters) per emplacement room and thus roughly 18,000 holes per panel, giving a total of about 145,000 holes for the repository as shown in Figure 6.5.9. The

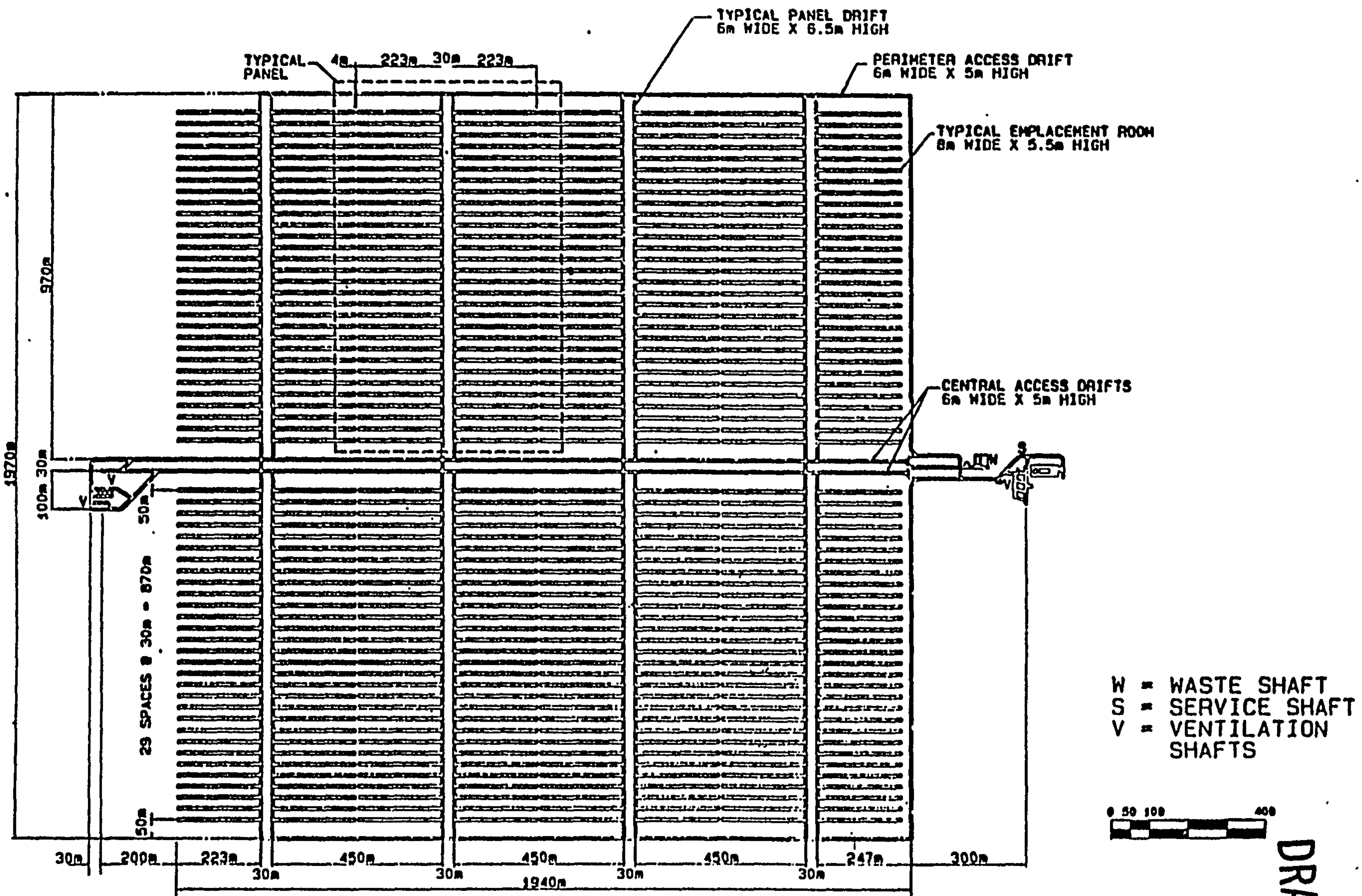

Figure 6.5.9. Repository General Arrangement

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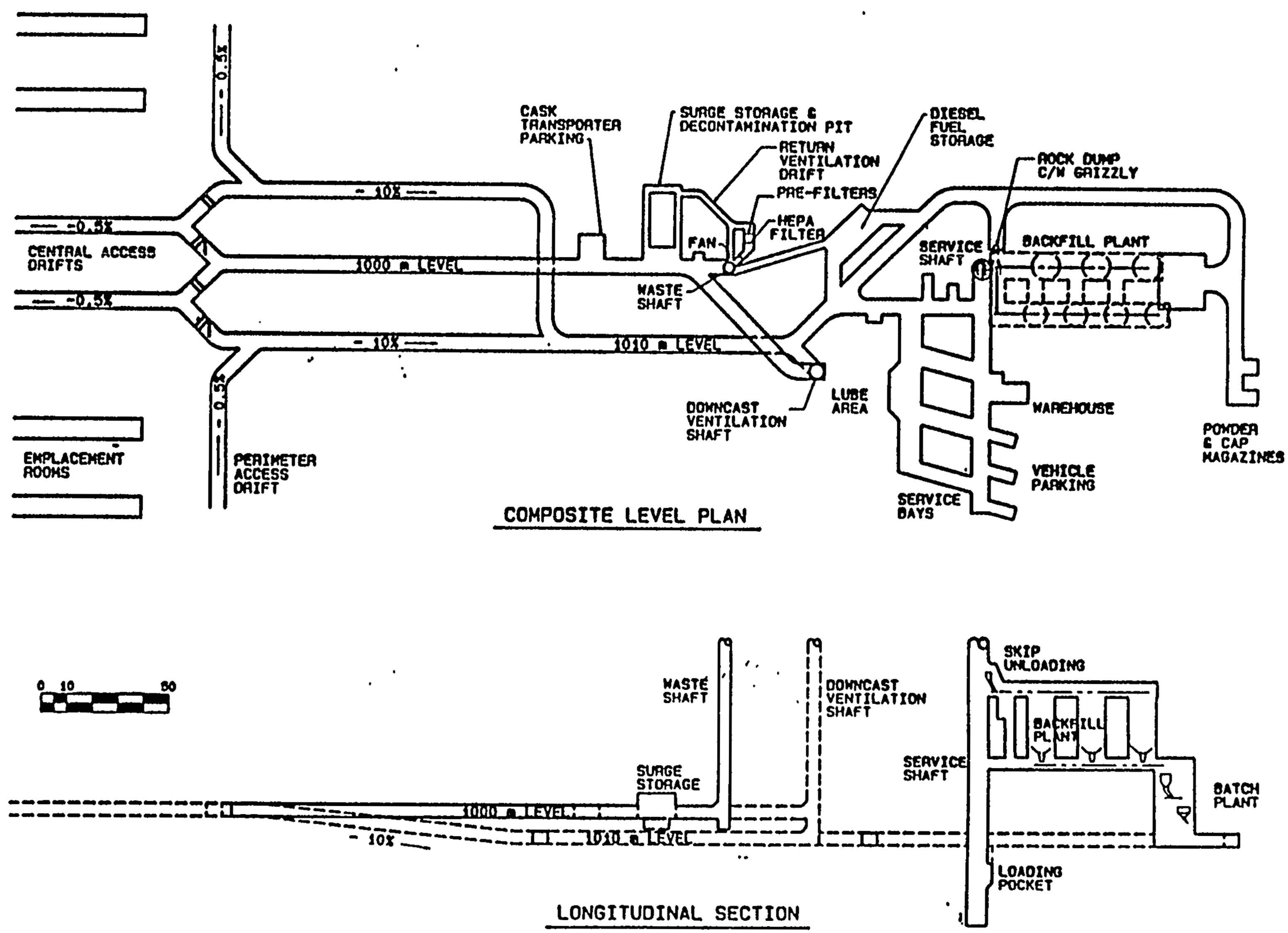


Figure 6.5.1Ø. Main Shaft Complex

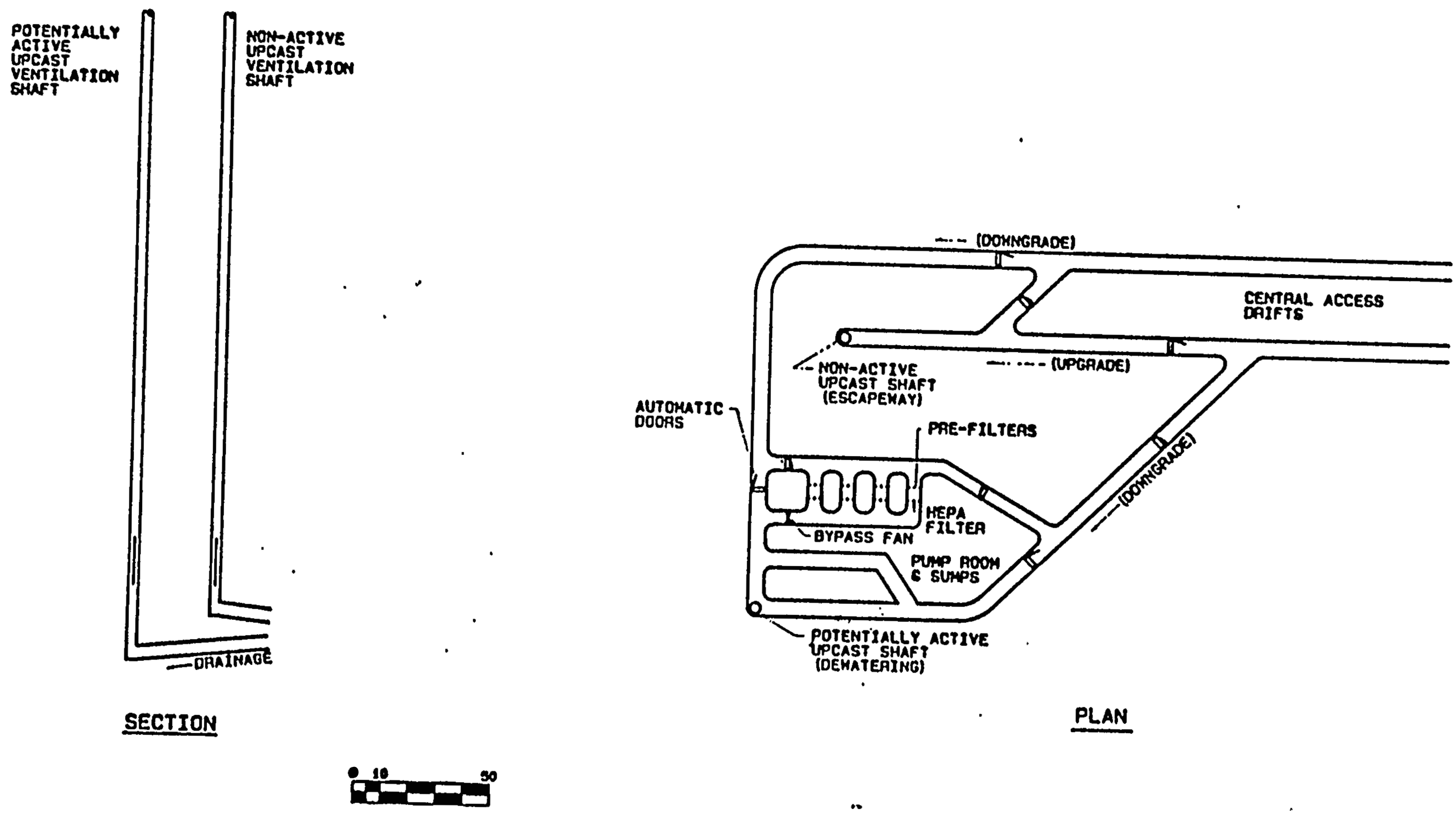
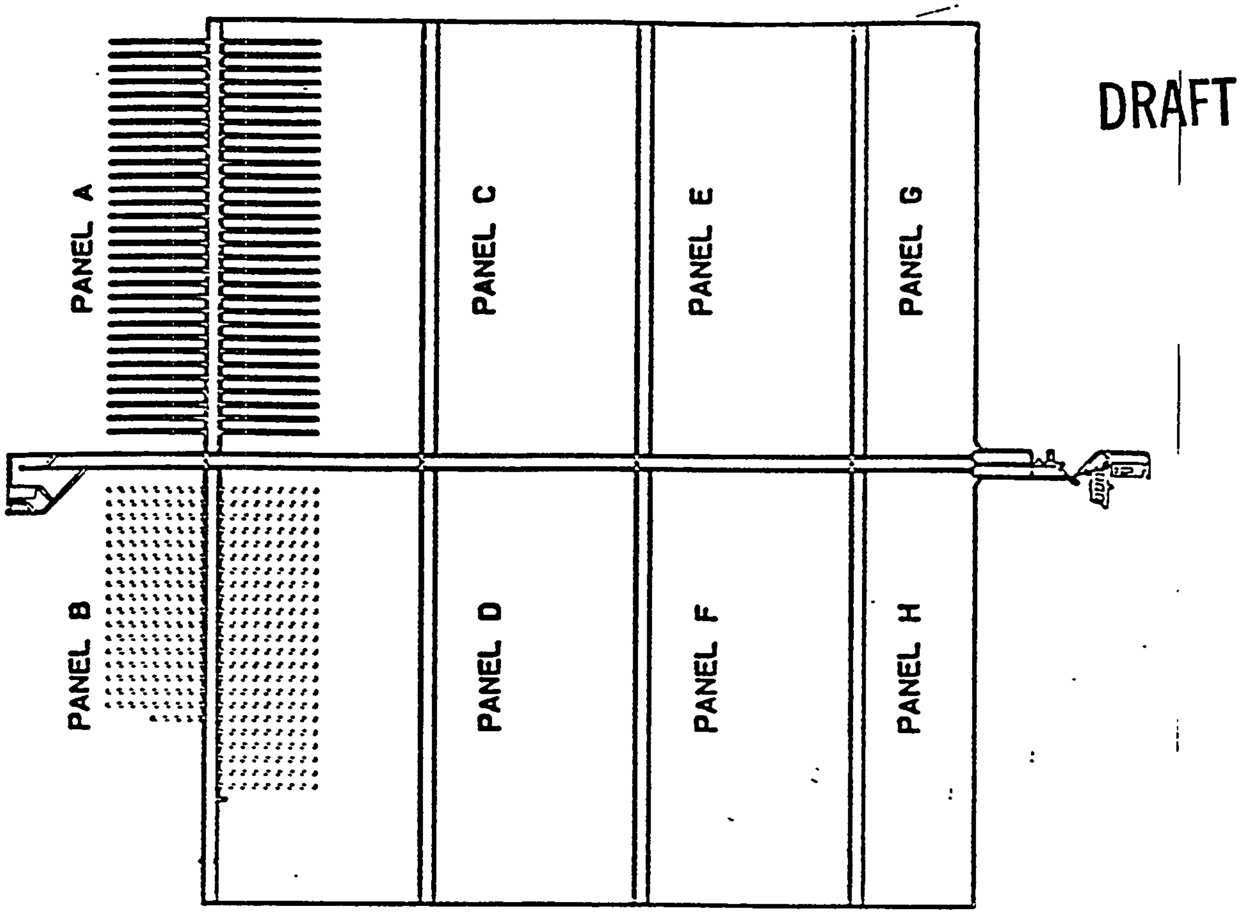


Figure 6.5.11. Exhaust Shaft Complex



VAULT OPERATING	PANEL OPERATIONS	
YEARS	ROOM EXCAVATION	CONTAINER EMPLACEMENT
1 - 5	PANEL B	PANEL A
6 - 10	PANEL C	PANEL B
11 - 15	PANEL D	PANEL C
16 - 20	PANEL E	PANEL D
21 - 25	PANEL F	PANEL E
26 - 30	PANEL G	PANEL F
31 - 35	PANEL H	PANEL G
36 - 40	-	PANEL H

Figure 6.5.12. Repository Development

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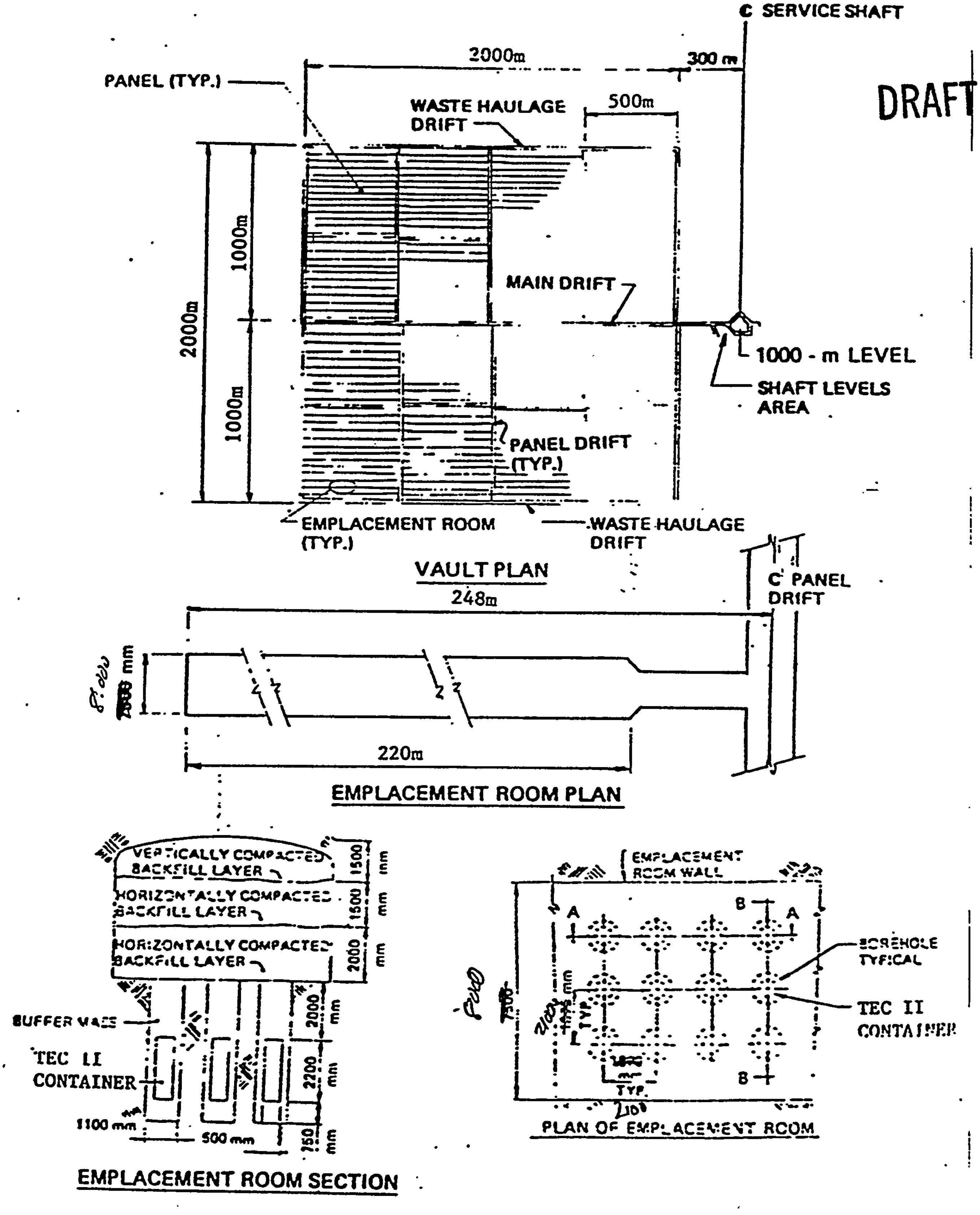

Figure 6.5.13. Emplacement Room Layout

disposal boreholes will be 5 m (16.4 ft) deep and 1.1 m (3.6 ft) in diameter and are designed to hold one waste canister surrounded by a bentonite-clay-based buffer backfill completely filling the hole (Figure 6.5.13). Section 6.5.5 describes the emplacement sequence in detail.

Five shafts will connect the underground facilities to the surface and will function as waste transport shaft, service shafts and ventilation shafts. Figures 6.5.9, 6.5.10, 6.5.11 and 6.5.14 show the location and configuration of the shafts within the repository. The shaft diameter will vary from 3.95 m (13 ft) for the two exhaust shafts to 7.9 m (26 ft) for the large service shaft. The exhaust shafts will be lined with a concrete lining about 15 cm (6 in) thick and the non-contaminated air shaft will contain an emergency escape cage. These shafts will also be used to pump out the drainage water from the repository. These two shafts will be located at the opposite end of the repository to the other three shafts (Figure 6.5.9). The service shaft will be lined with a 30 cm (12 in) concrete lining and will contain the main personnel and materials cage, the skips to hoist excavated rock as well as various underground services. The 4.6 m (15 ft) diameter waste shaft will be lined with a 30 cm (12 in) concrete lining and contain the cage for lowering the waste canisters to the repository. The fifth shaft is the 4.9 m (16 ft) diameter intake shaft which will also have a 15 cm (6 in) concrete lining and will serve as the main air intake.

6.5.4 Repository Construction

6.5.4.1 Schedule

An exact time schedule for the repository program has not been defined. Construction is expected to start early in the next century but this will depend on the outcome of the repository concept review process, site selection and characterization activities, etc. For this concept, it was assumed that disposal operations would start around 2025 and would last for about 35-40 years. From the time of completion of site selection and characterization, the construction, operation, backfilling and sealing, and final closure of the repository is estimated to take about 105 years.

6.5.4.2 Methods and Equipment

Information gathered during the site selection and characterization program will be used to develop a site-specific design for the repository. The repository will be constructed in a manner that allows geological, geophysical, hydrogeological, geochemical and geomechanical data to be collected during facility construction and operation. While the surface facilities are being constructed, the shafts will be excavated at both ends of the repository. As the perimeter drifts are being excavated, geotechnical experiments as well as mapping of the rock formations will be conducted to determine the rock mass properties and confirm the design of all aspects of the repository construction and operation. Revisions to the final designs can be carried out at this time.

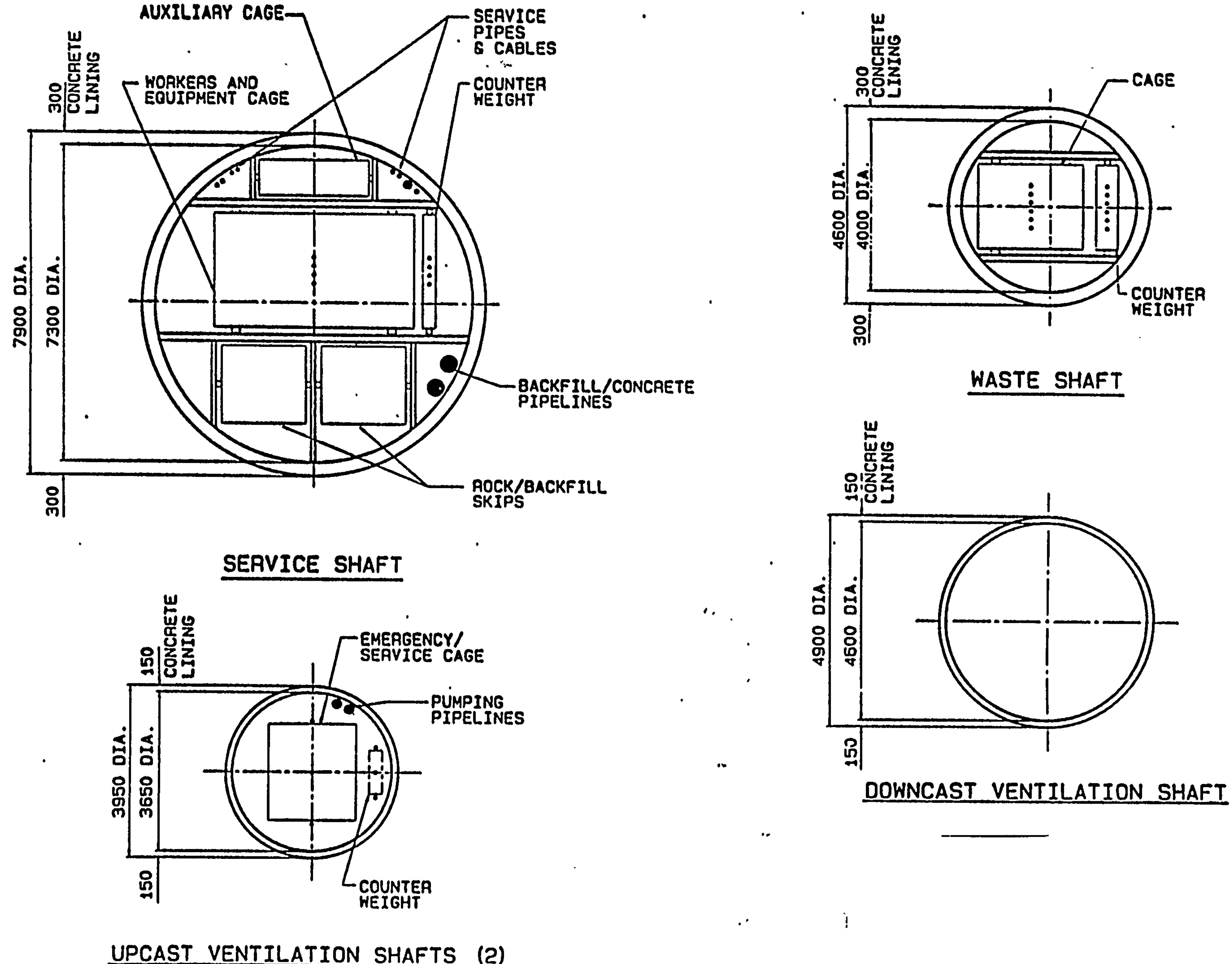

Figure 6.5.14. Shaft Cross Sections

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Excavation of the shafts and underground facilities is expected to be done using drill-and-blast techniques, possibly supplemented by drilling techniques. Some of the shafts can probably be constructed using the pilot and slash technique, where a raise borehole is enlarged to the final shaft diameter and lined with the final lining (Figure 6.5.15). Excavation of the access drifts, perimeter drifts, panel drifts and emplacement rooms will most likely be done using conventional methods as outlined in Figure 6.5.16. Application of controlled blasting to minimize damage around the excavations is currently being tested at the Underground Research Laboratory in Manitoba, Canada, and the results will be included in the future development of the repository construction methods.

The drilling of the emplacement holes is also under further study. Techniques such as large diameter coring and the application of high-pressure water jets are some of the techniques under consideration.

6.5.4.3 Cost

The estimated costs for the construction and operation of the spent fuel disposal concept are summarized in Table 6.5.1. The yearly operating costs for the surface facilities are averaged over a period of 35 years and these costs depend mainly on the cost of the waste canisters and the lead (or other material) used to immobilize the fuel within these canisters. The decommissioning and monitoring annual costs are averaged over a period of 70 years.

Table 6.5.1. Estimated Costs for the Used-Fuel Disposal Concept (Million 1979 $ Canadian). (after Wuschke, et al.)

Component	Capital Cost	Operating Period (Years)	Annual Operating Cost
Surface Facilities	237	35	99-185
Vault	233	35	19-25
Transportation(*)	43-172	35	12-180
Decommissioning & Monitoring	31	70	7

(*) The variation of the transport costs reflects differences in the use of rail, barge, or road transport and varying shipping distances.

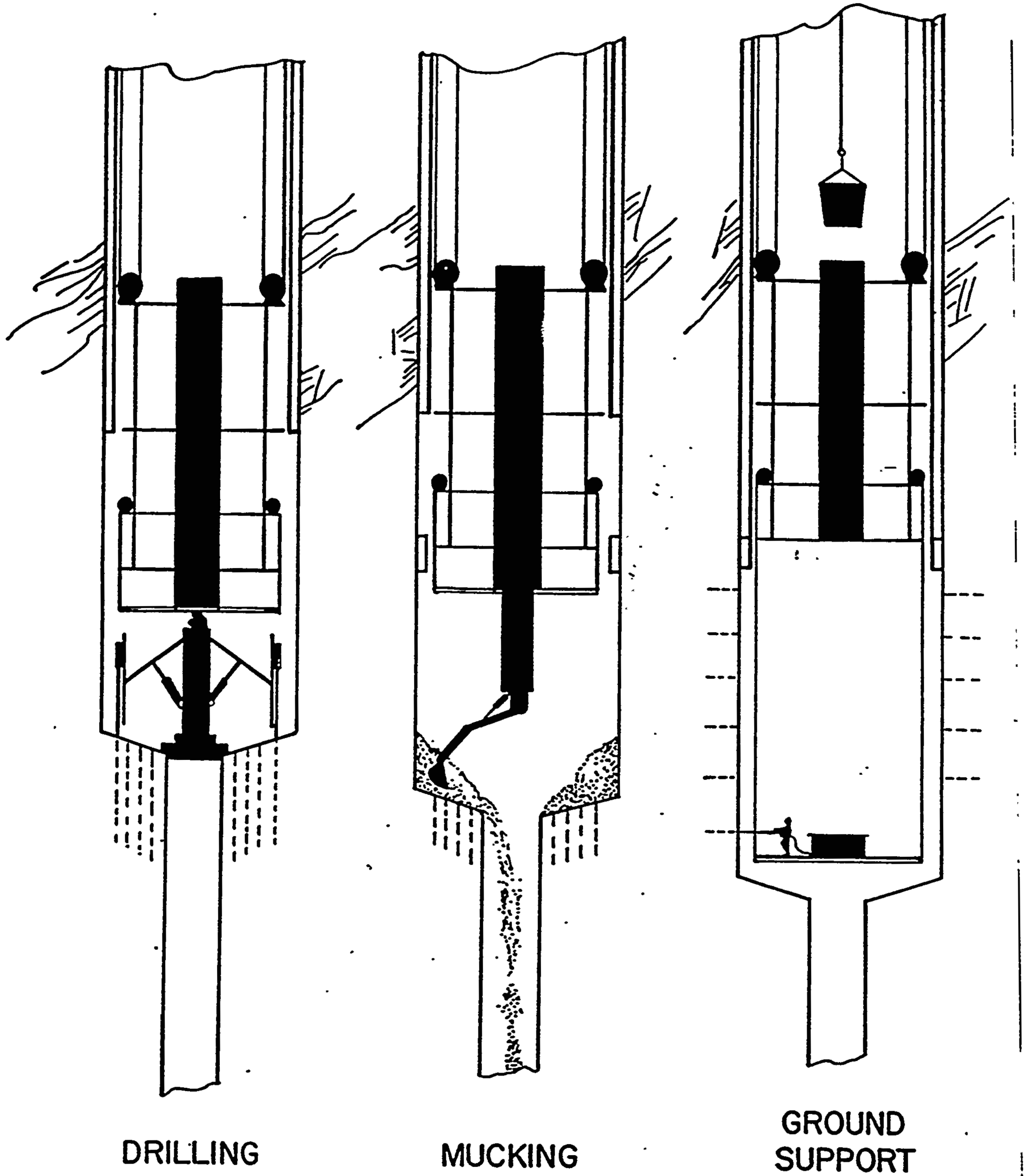

Figure 6.5.15. Repository Shaft Construction

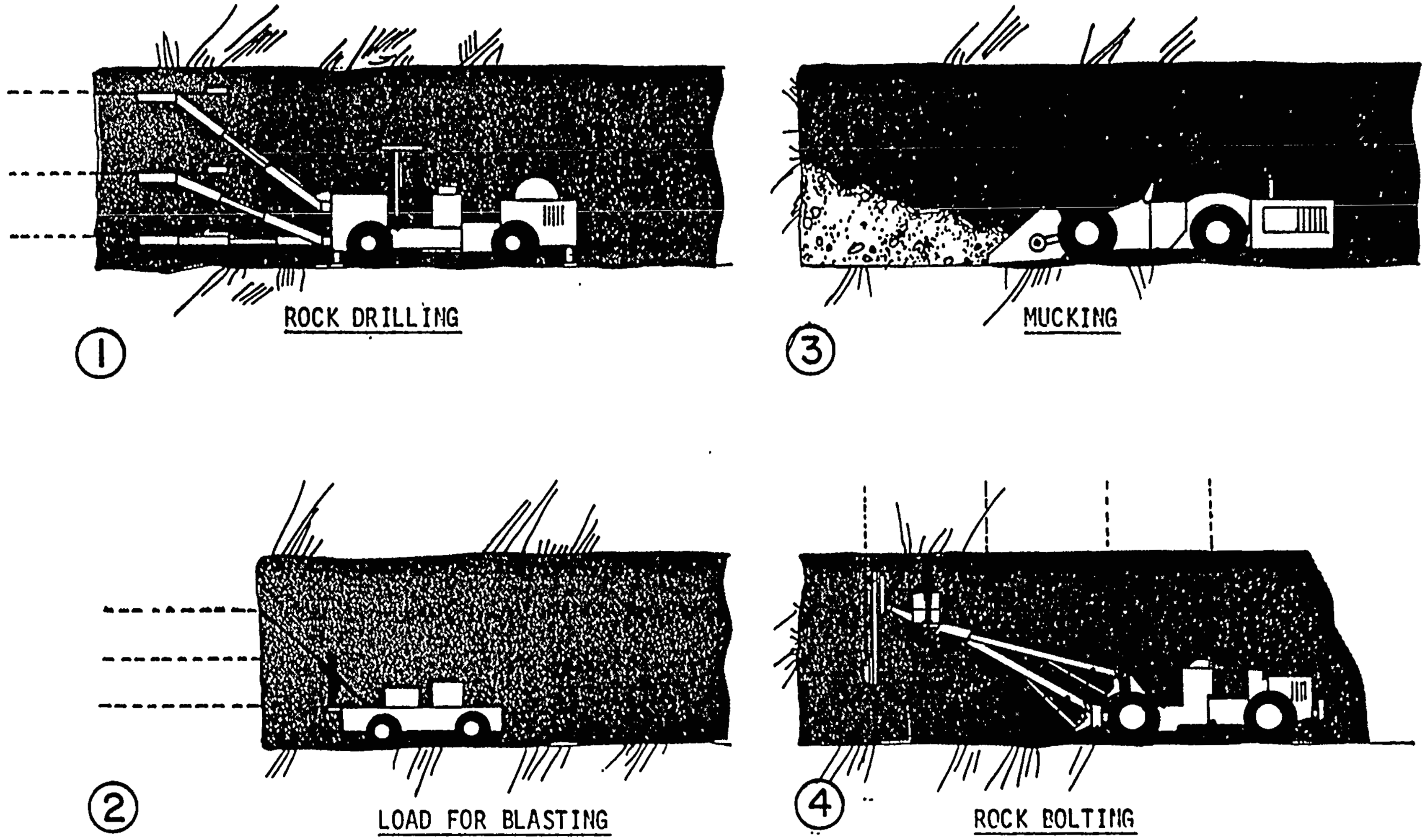

Figure 6.5.16. Repository Excavation Method

6.5.5 Waste Emplacement Cycle

6.5.5.1 Waste Treatment

The spent fuel bundles (Figure 6.5.2) will be stored for at least 10 years before being placed in the special disposal canisters (see Section 6.5.5.3). Reprocessing of the spent fuel is currently not being considered but if it should be adopted in the future, the high-level wastes remaining after recycling would be handled in a manner similar to the spent fuel due to their high radioactivity and heat output.

6.5.5.2 Waste Transport

The spent fuel will be transported from the storage pools at the reactors to the fuel immobilization facility at the repository after a cooling-off period of about 10 years. The main transport methods will most likely be truck and rail transport with the possibility of barge transport in some areas. The road transport cask will contain two shipping modules, each module containing 96 fuel bundles. Figure 6.5.17 shows the Ontario Hydro road transport cask design. The rail cask will contain six shipping modules.

6.5.5.3 Waste Package

The reference CANDU fuel bundle design is the 37-element Bruce fuel bundle shown in Figure 6.5.2. which has been cooled for a period of at least 10 years in a storage pool at the reactor. The spent fuel in the shipping modules will be unloaded from the transport casks at the fuel immobilization facility and temporarily stored in the receiving area (Figure 6.5.6.) before being transferred to the hot-cell areas. In the hot-cell facility, the spent fuel bundles will be transferred from the shipping modules to the disposal canisters.

Several canister designs are being studied for the Canadian program (Figure 6.5.18). The reference design chosen for the study described in this chapter is the packed-particulate supported-shell canister made of titanium (Figure 6.5.19).

In the hot-cell area, the spent fuel bundles will be remotely transferred from the shipping modules into a basket consisting of 19 carbon-steel pipes. The length of the basket assembly is about 2 m (6.5 ft) and the spent fuel bundles will be stacked 4 deep in 18 of the pipes to give a total of 72 bundles per assembly. The full basket will then be installed in the thin-wall titanium canister. The voids will be filled with a particulate material, such as glass beads, which will be vibrationally compacted to the required density to withstand the expected canister loads. Final closure of the canister will be achieved by pressing the lid onto the canister and diffusion bonding the lid to the shell. Final non-destructive

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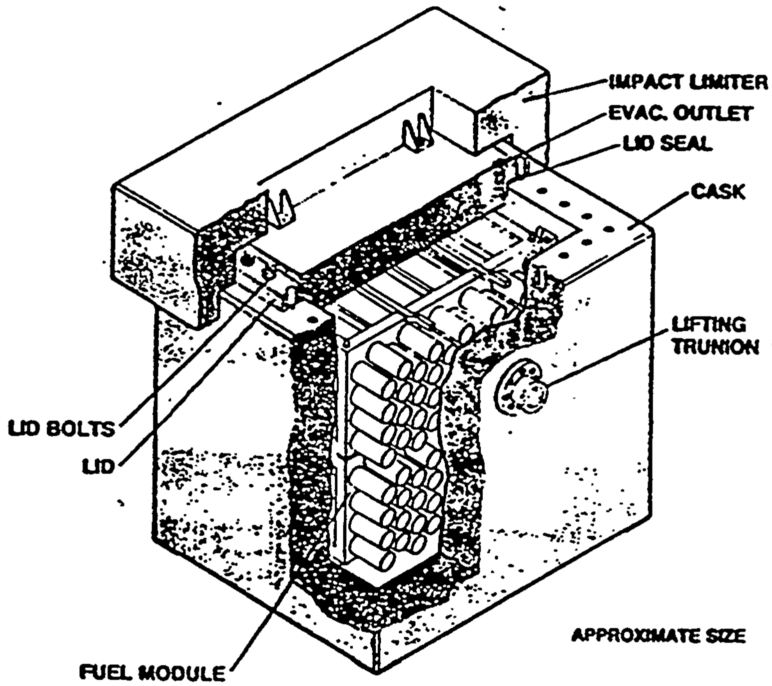

Figure 6.5.17. Road Transport Cask

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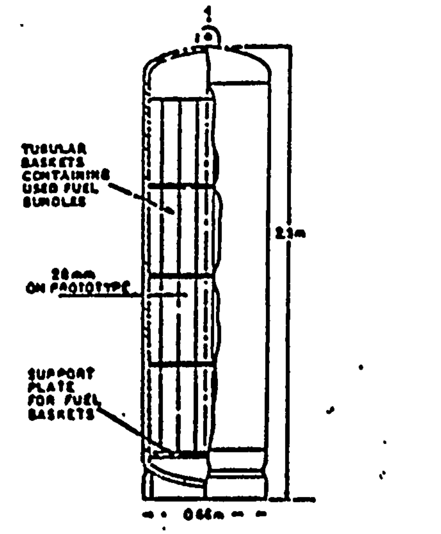
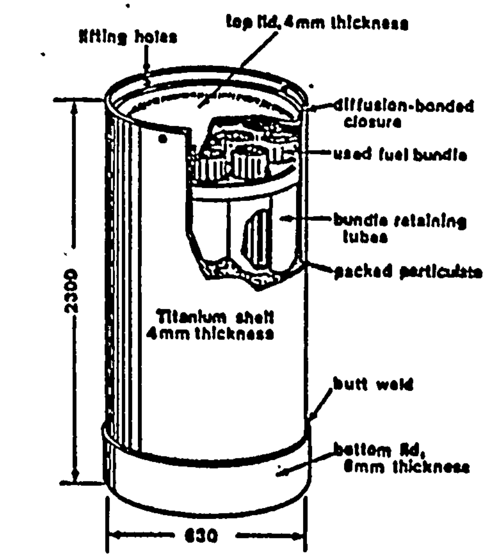
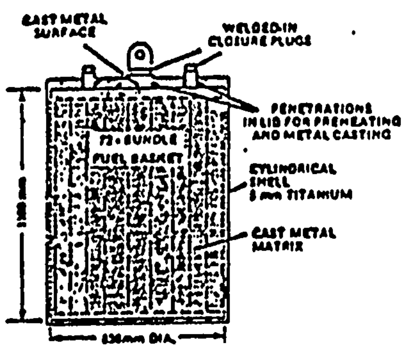


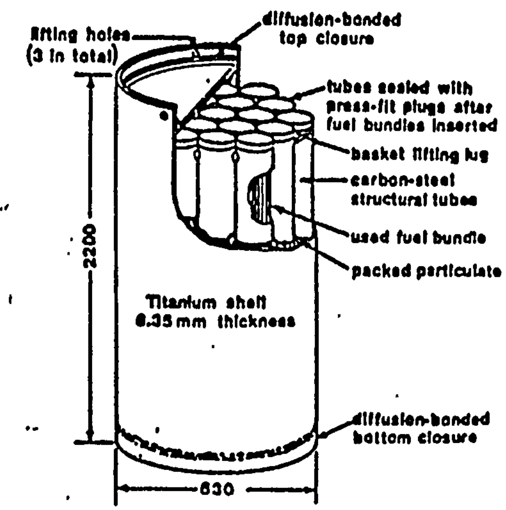
DESIGN DETAILS OF 316 L STAINLESS STEEL STRESSED-SHELL PROTOTYPE CONTAINER



DESIGN DETAILS OF A PACKED-PARTICULATE SUPPORTED-SHELL FUEL ISOLATION CONTAINER



CONCEPTUAL DESIGN OF A METAL-MATRIX SUPPORTED-SHELL CONTAINER



DESIGN DETAILS OF THE PROTOTYPE STRUCTURALLY-SUPPORTED SHELL FUEL ISOLATION CONTAINER

Figure 6.5.18. Waste Canister Designs

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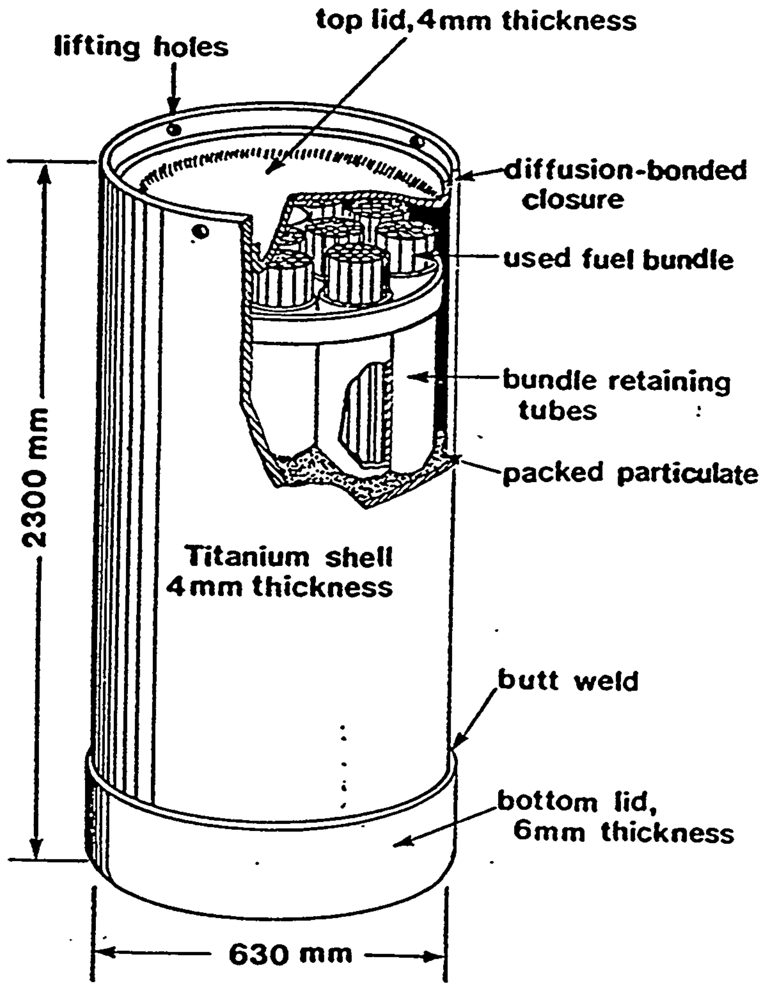

Figure 6.5.19. Reference Spent Fuel Canister

inspection procedures will determine whether the canister has been properly sealed before it is transferred to the waste shaft storage area in preparation for transport to the repository.

6.5.5.4 Underground Waste Emplacement Sequence

When a waste canister is ready to be transported underground, it will be placed into a shielded transport cask (Figure 6.5.20). The shielding cask will be loaded onto the waste shaft cage and taken underground to the repository level (Figure 6.5.21). At the shaft bottom, the cask will be transferred to a special transport vehicle (Figure 6.5.22) and transported to the appropriate emplacement room (Figure 6.5.23).

Before a waste canister can be located into a emplacement room, the room will be prepared in the following manner. The disposal boreholes are drilled into the floor of the room using a special drill rig (Figure 6.5.24). Three holes, 5 m (16.4 ft) deep and 1.1 m (3.6 ft) in diameter, will be drilled in a row across the width of the room at a spacing of about 2 m (6 ft). Spacing between rows will also be about 2 m (6 ft). The completed holes will then be filled with a bentonite-clay based buffer material (Figure 6.5.25). This material will be compacted into the borehole in layers to give a near-homogeneous mass. The buffer mass in the borehole will then be centrally augered (Figure 6.5.26) to a depth of about 4.2 m (13.8 ft) and a diameter of 70 cm (2.3 ft) to accept the spent fuel canister. The emplacement room will also have been prepared with rail tracks for the emplacement platform as shown in Figure 6.5.27.

The shielded waste cask will arrive at the entrance to the emplacement room and transferred to the emplacement platform (Figure 6.5.28). The platform will travel to the appropriately prepared emplacement borehole and center the shielding cask over the hole. A shielding ring will be lowered to prevent radiation leakage and the base of the cask will be opened. The waste canister will then be lowered into the borehole (Figures 6.5.29 and 6.5.30).

The second stage of buffer emplacement follows canister disposal (Figure 6.5.31). During this operation, special radiation shielding measures must be used until the final buffer in the borehole provides adequate protection. The composition and compactions specifications will be the same as for the original material.

Once all the disposal holes in an emplacement room have been filled, the emplacement room and access tunnel backfill operation can commence. The reference backfill material will be a clay and crushed granite mixture which will be placed and compacted in layers in the room to achieve the specified homogeneity. Figures 6.5.32 to 6.5.37 show the backfill operation and the type of equipment that will be used. Permanent concrete bulkheads will be placed at the entrance to the emplacement rooms and the surrounding rock mass sealed with grout (Figure 6.5.38). The bulkheads will provide a means of closing the rooms, allow repository operations to continue in the panel drifts, act as a barrier against ground-water flow and provide radiation shielding.

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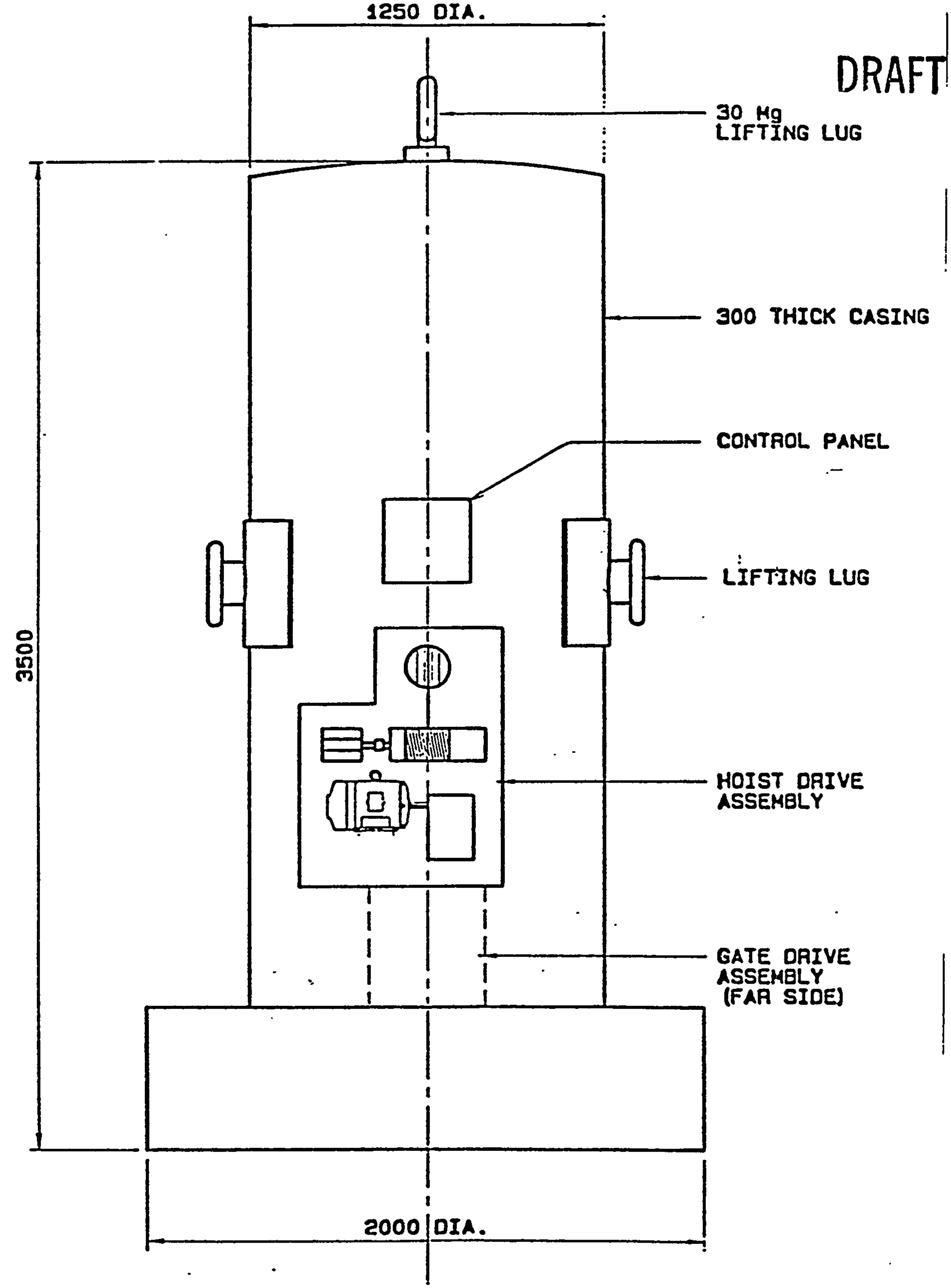

Figure 6.5.20. Shielded Transport Cask

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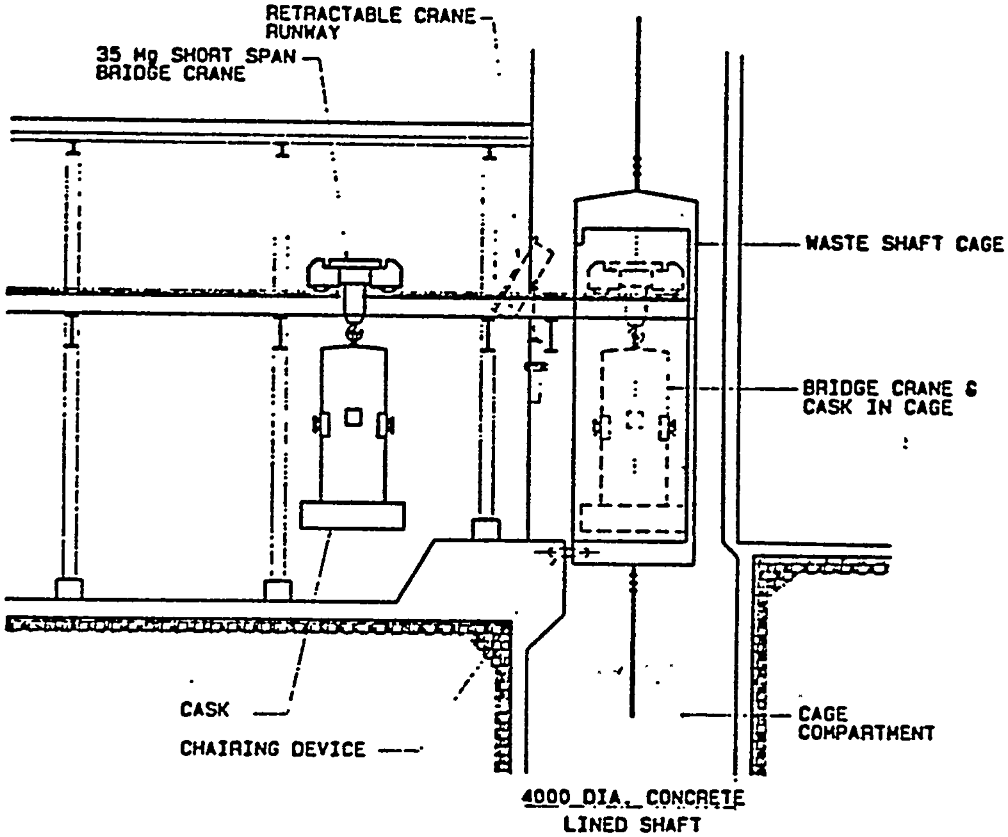

Figure 6.5.21. Cask Handling At Waste Shaft

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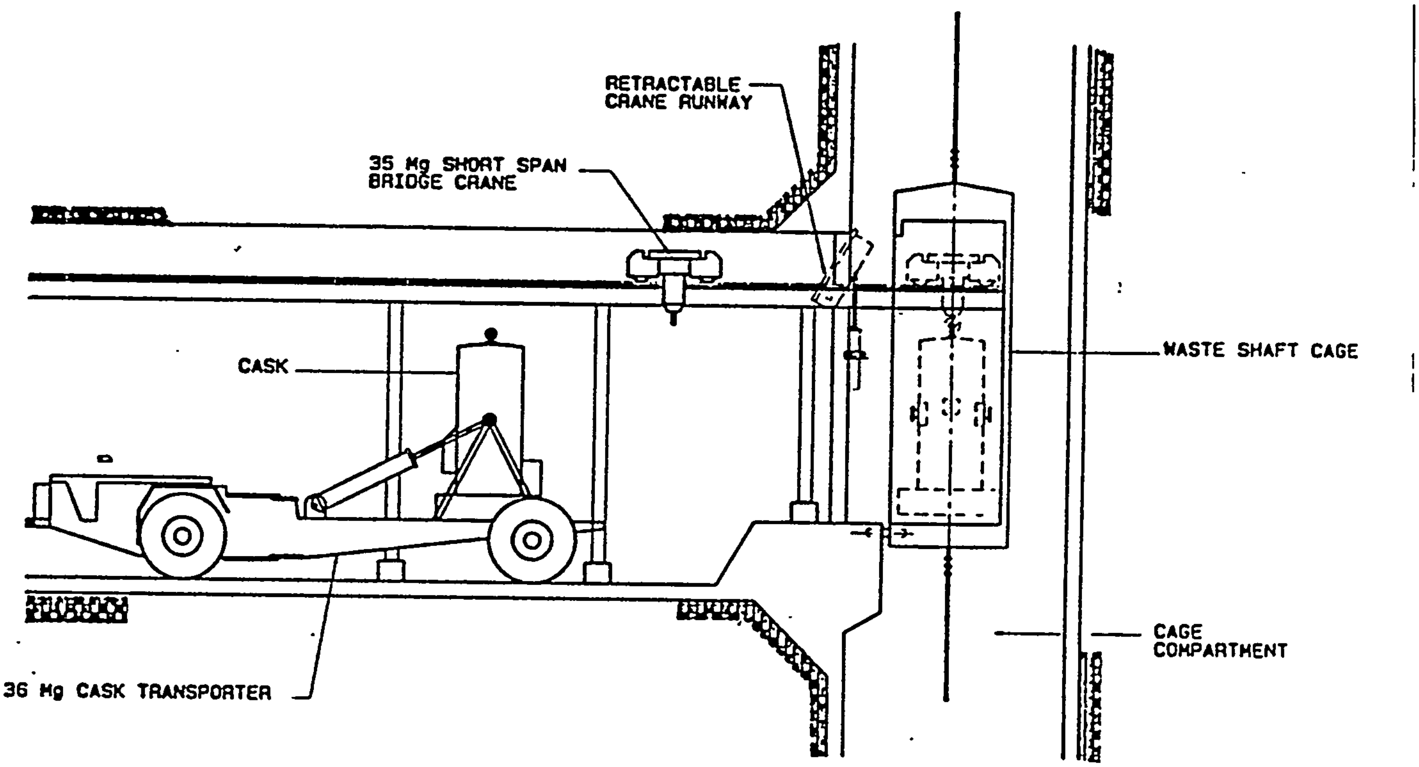

Figure 6.5.22. Cask Handling At Shaft Bottom

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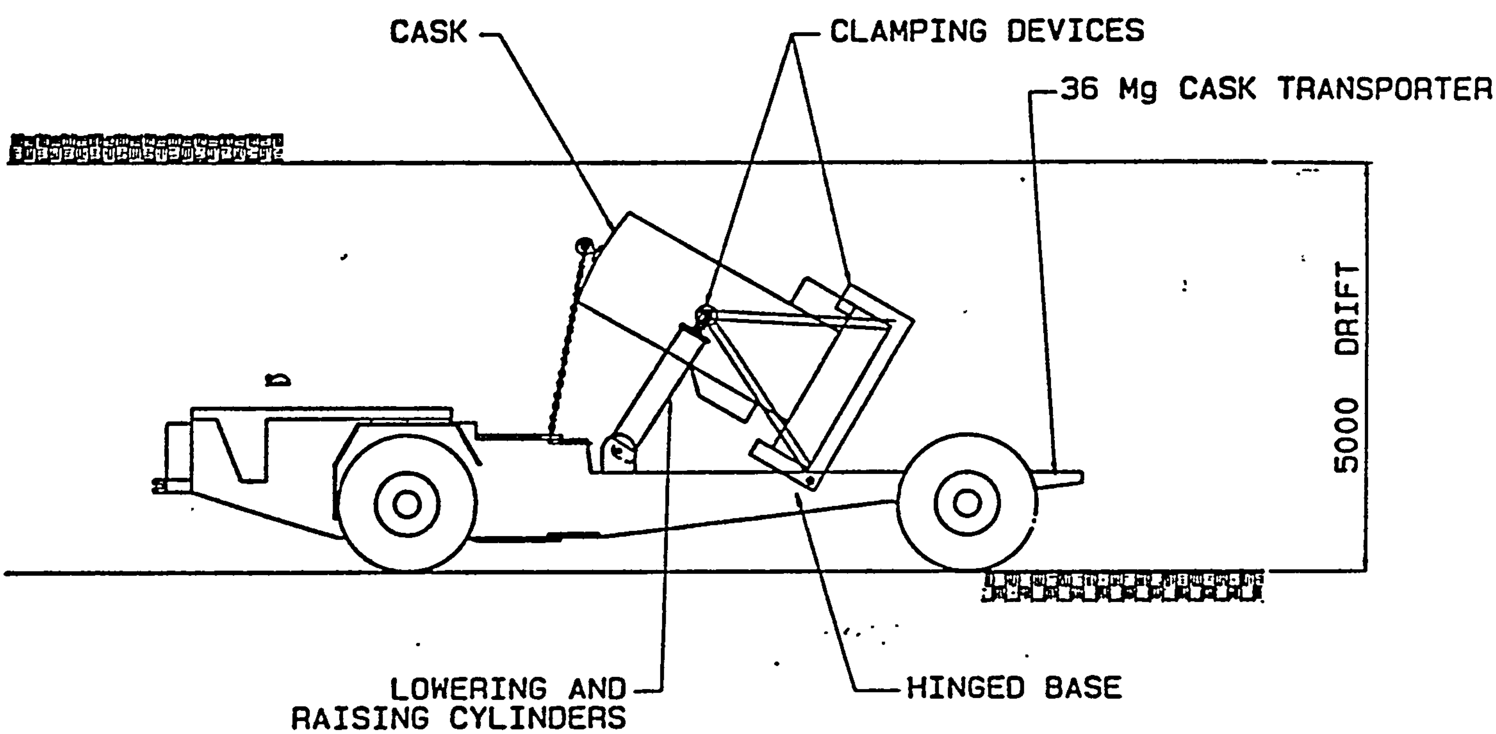

Figure 6.5.23. Underground Cask Transporter

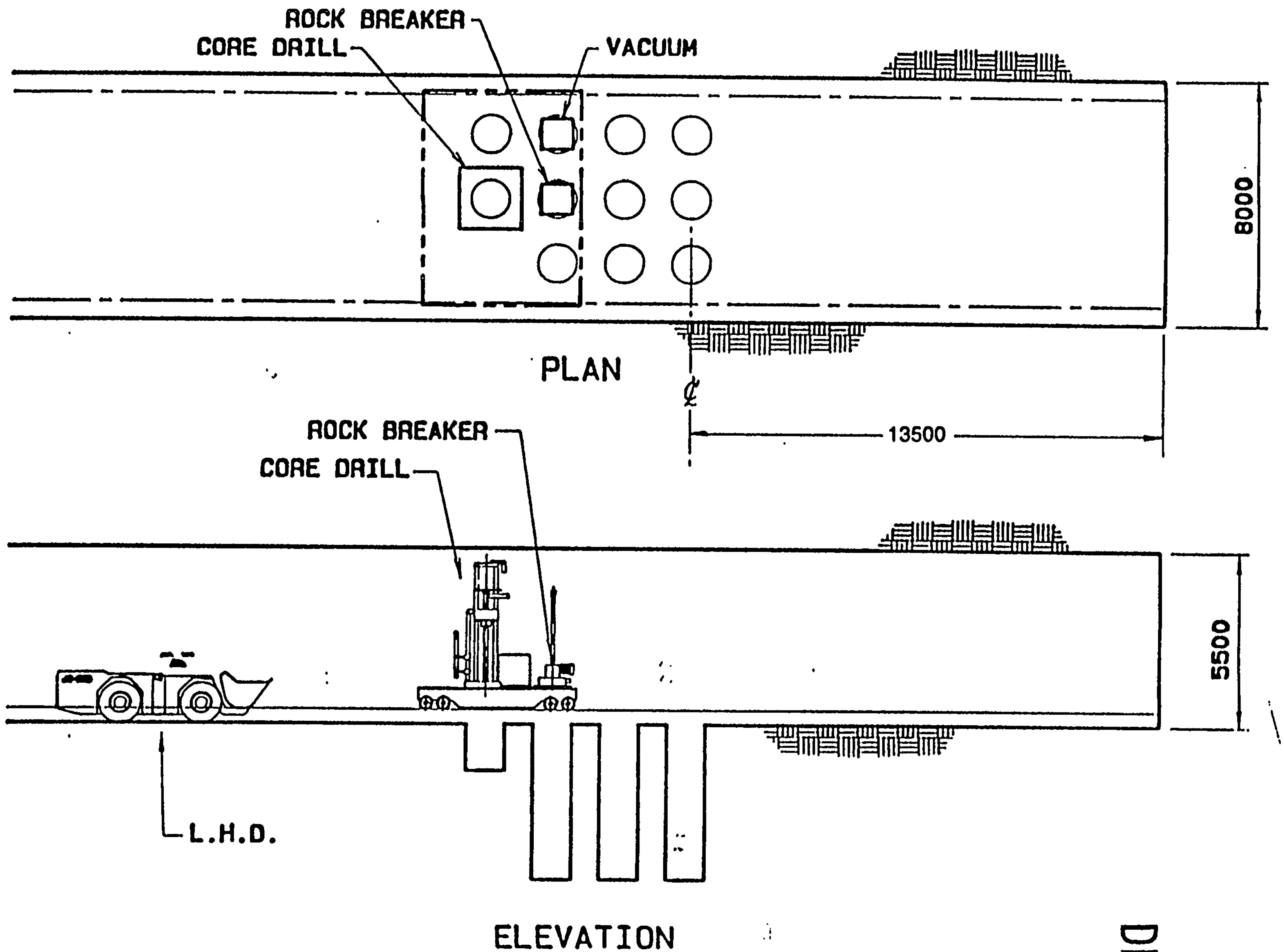

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Figure 6.5.24. Borehole Drilling

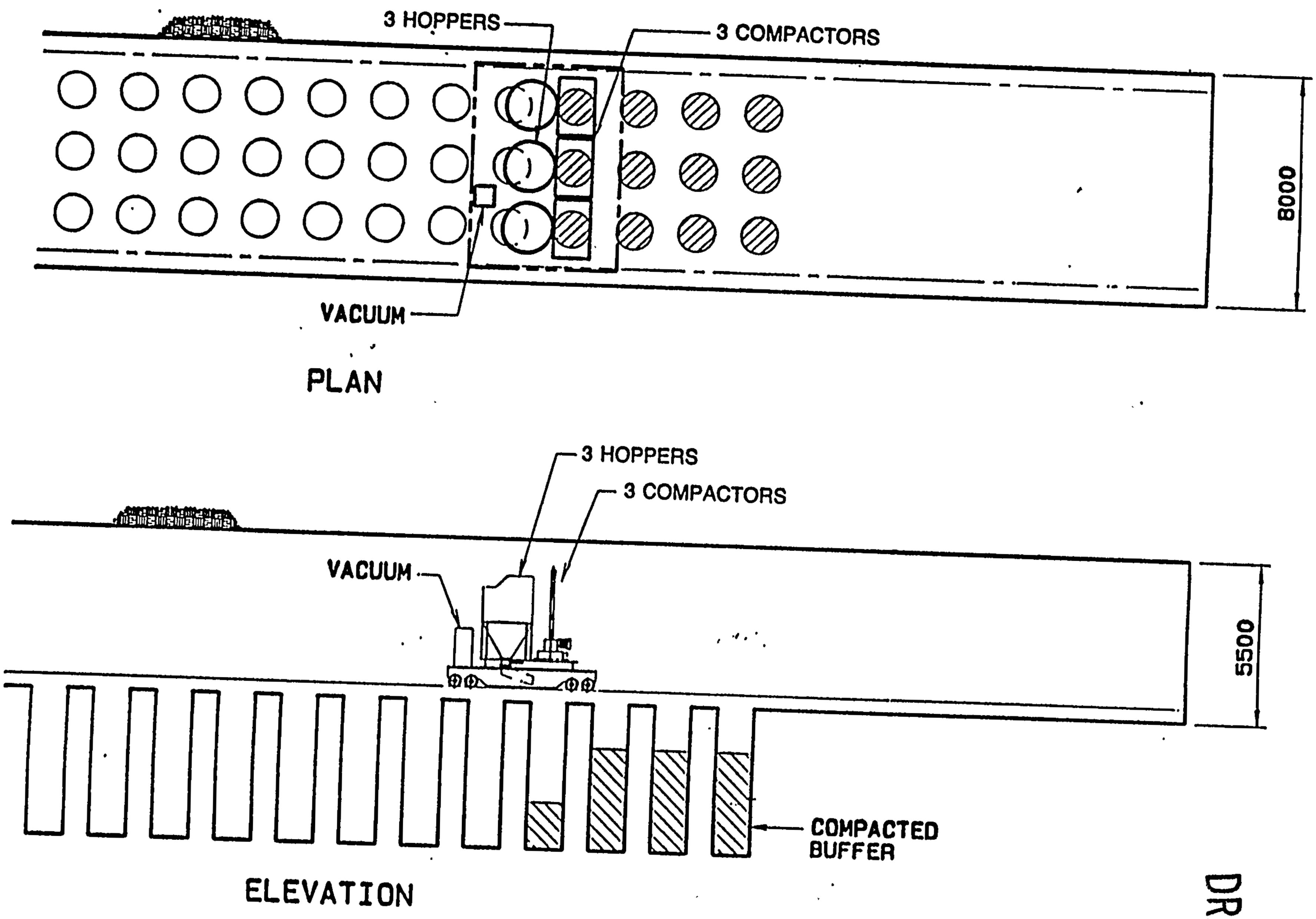

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Figure 6.5.25. Initial Buffer Emplacement

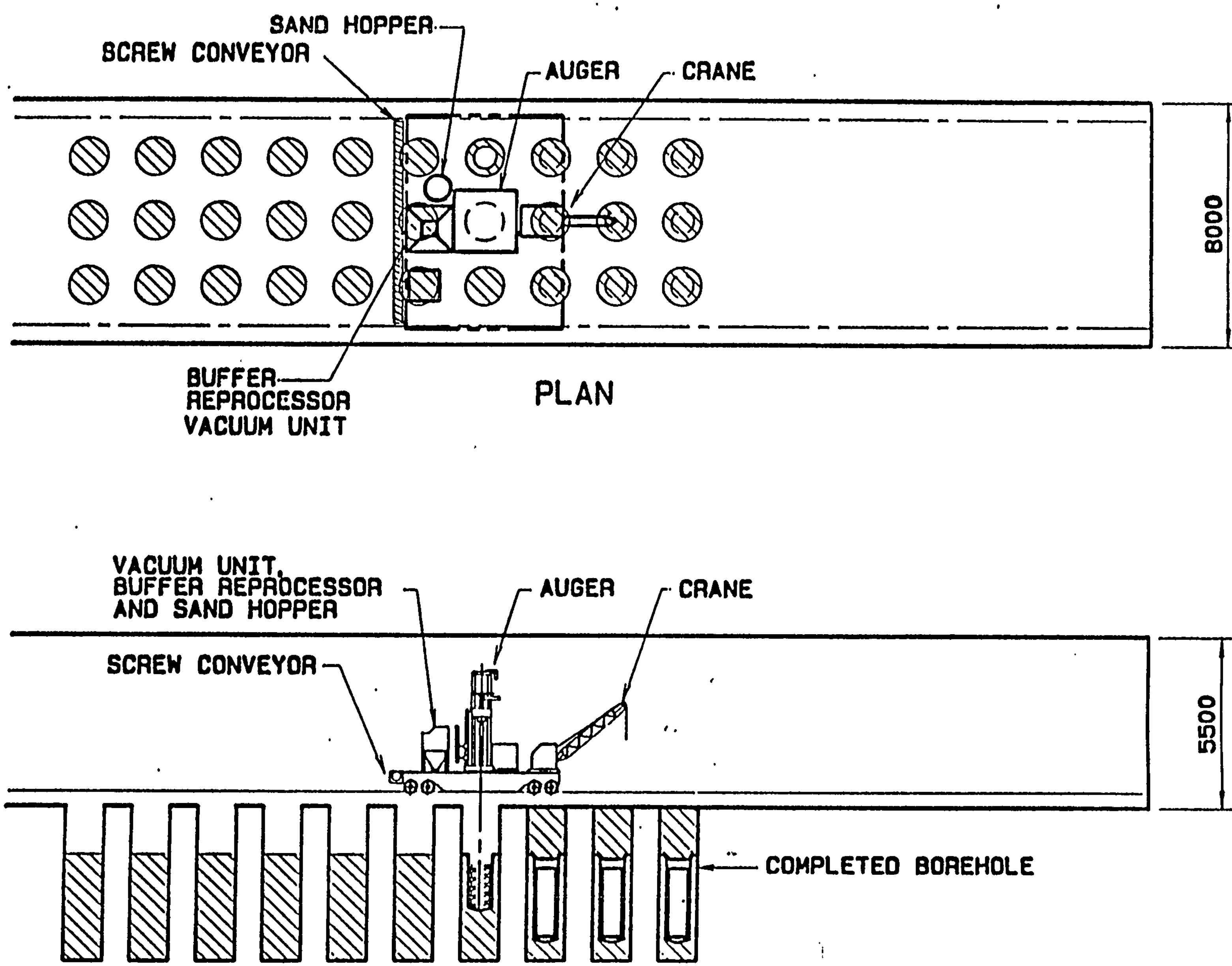

Figure 6.5.26. Augering Buffer Material

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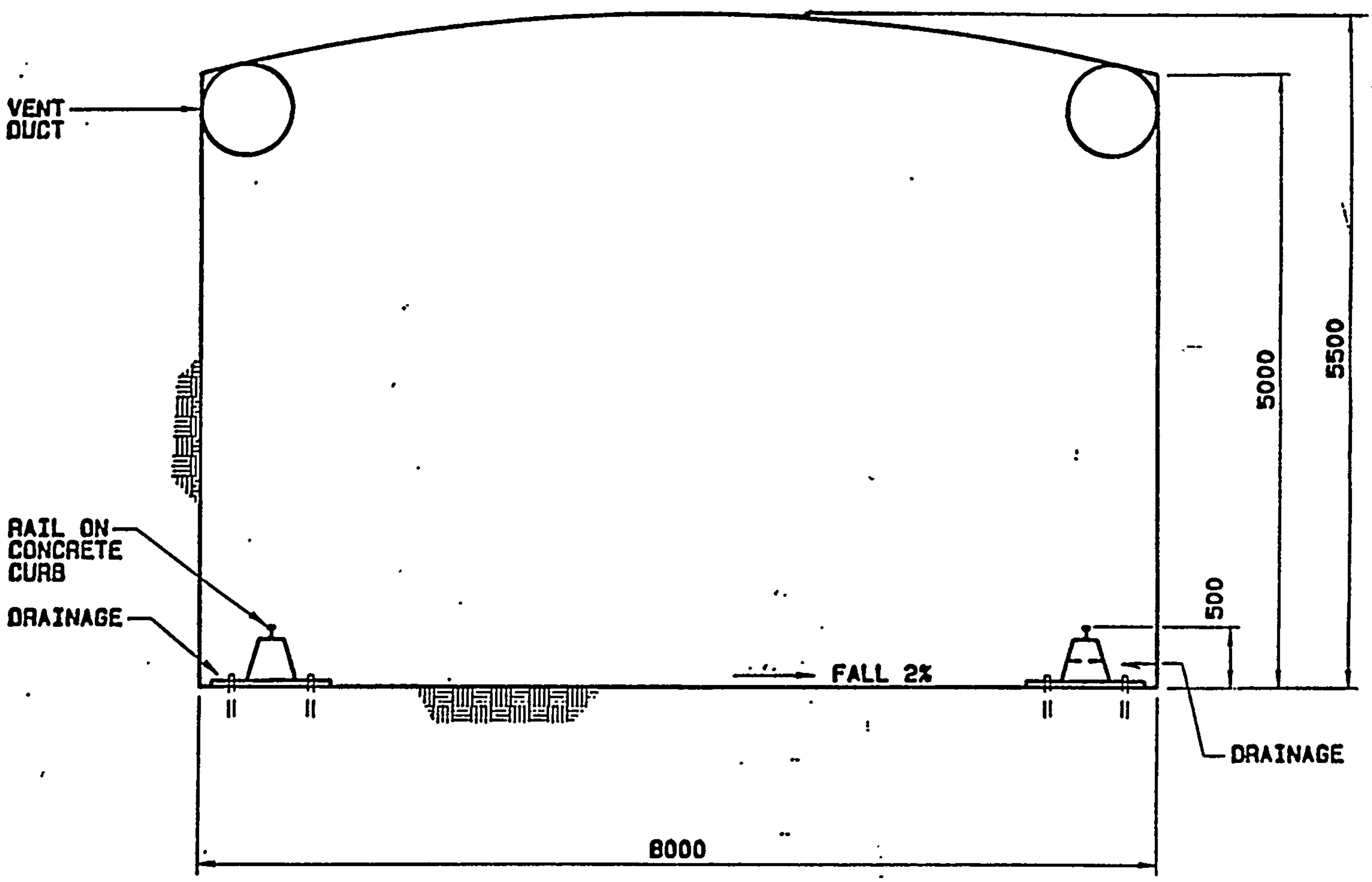

Figure 6.5.27. Emplacement Room Preparation

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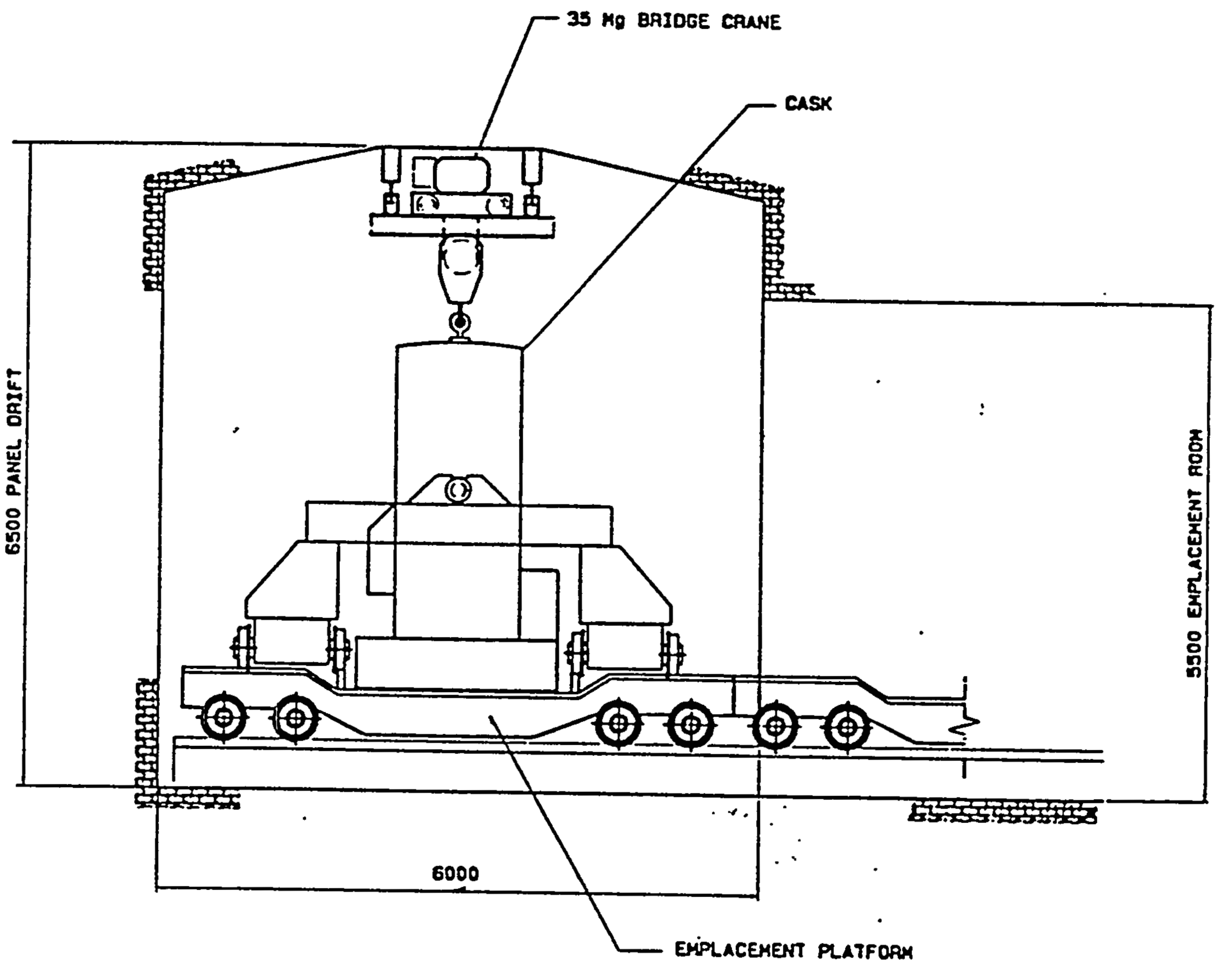

Figure 6.5.28. Shielded Cask Transfer

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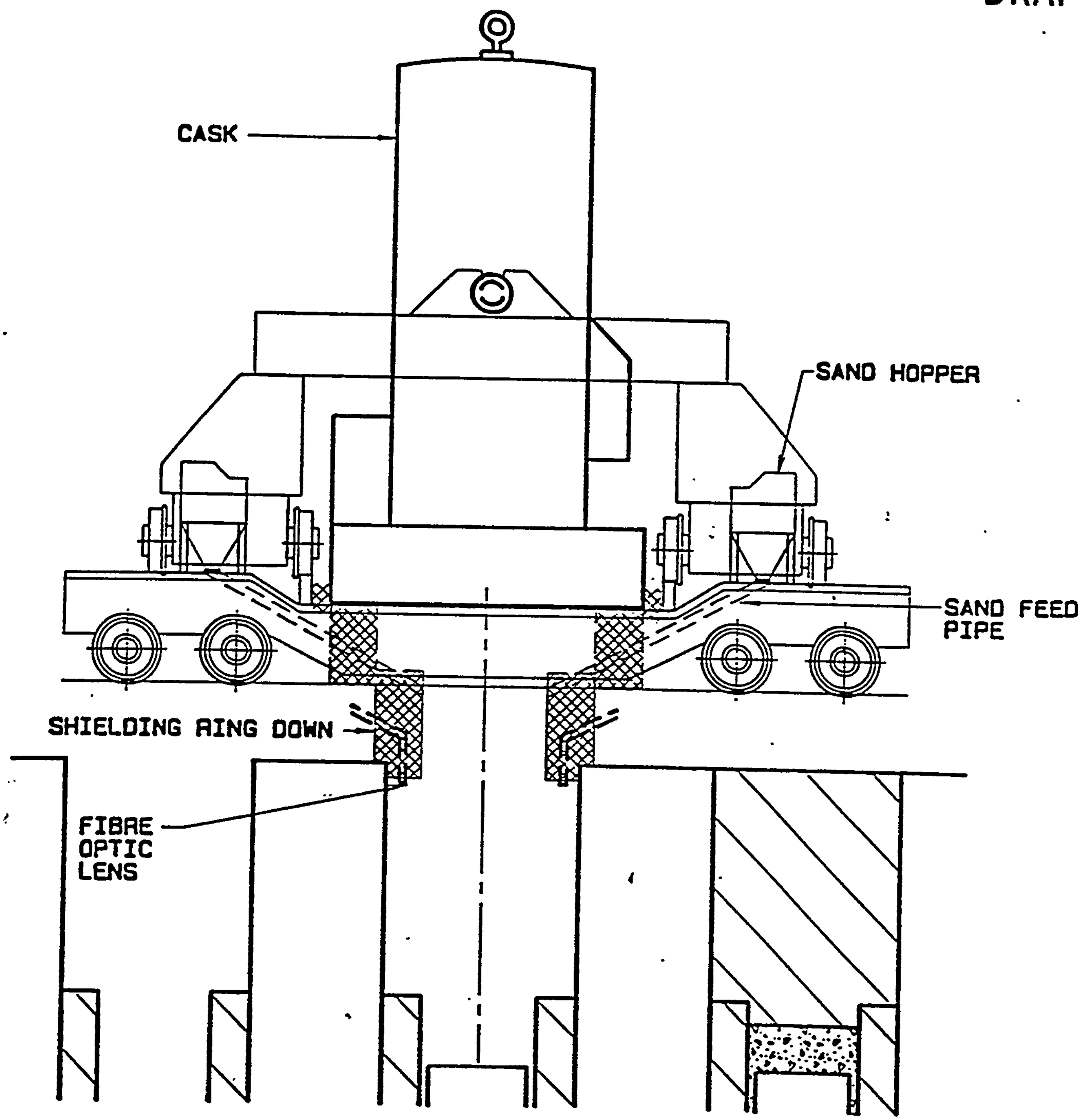

Figure 6.5.29. Waste Emplacement Platform

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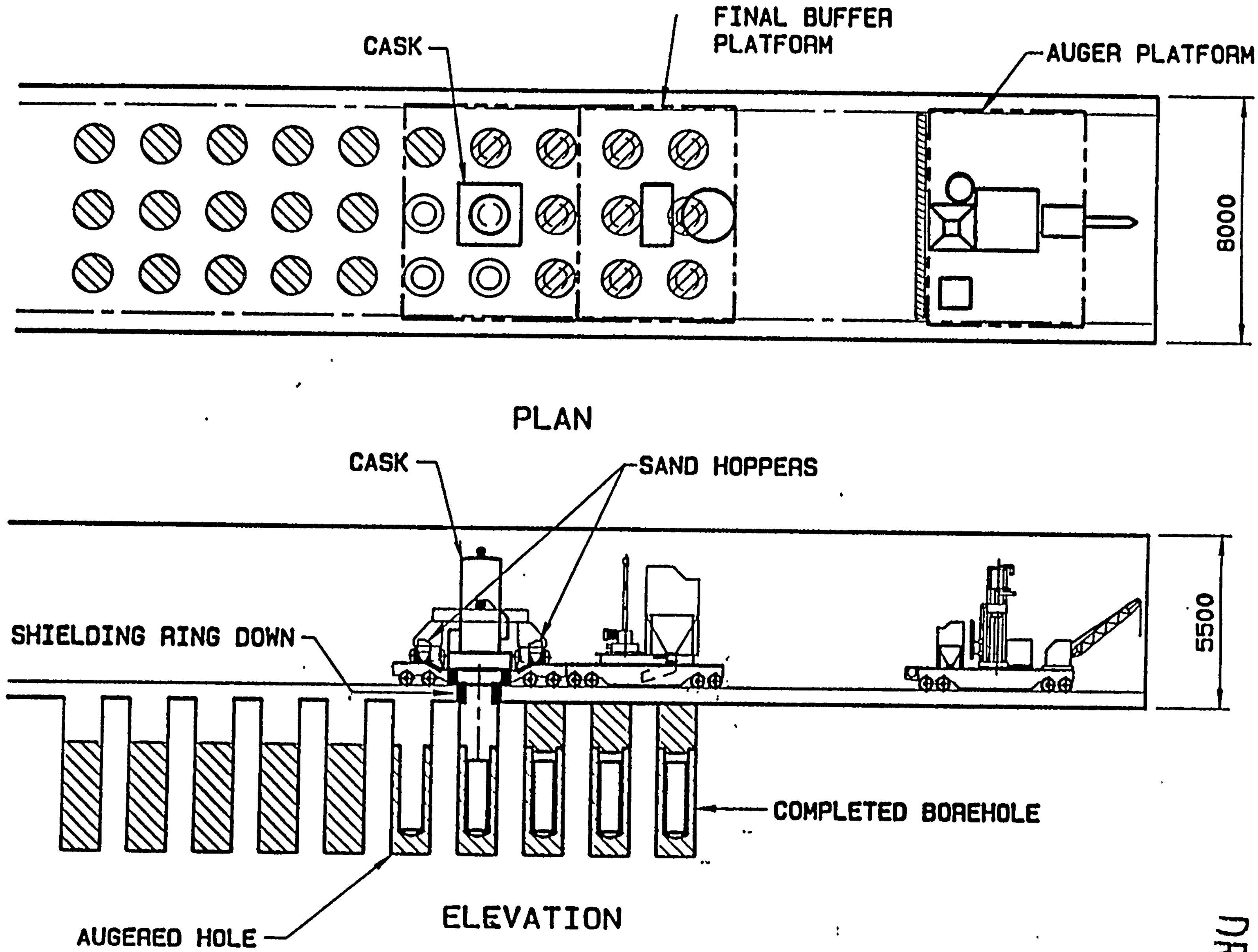

Figure 6.5.30. Waste Canister Emplacement

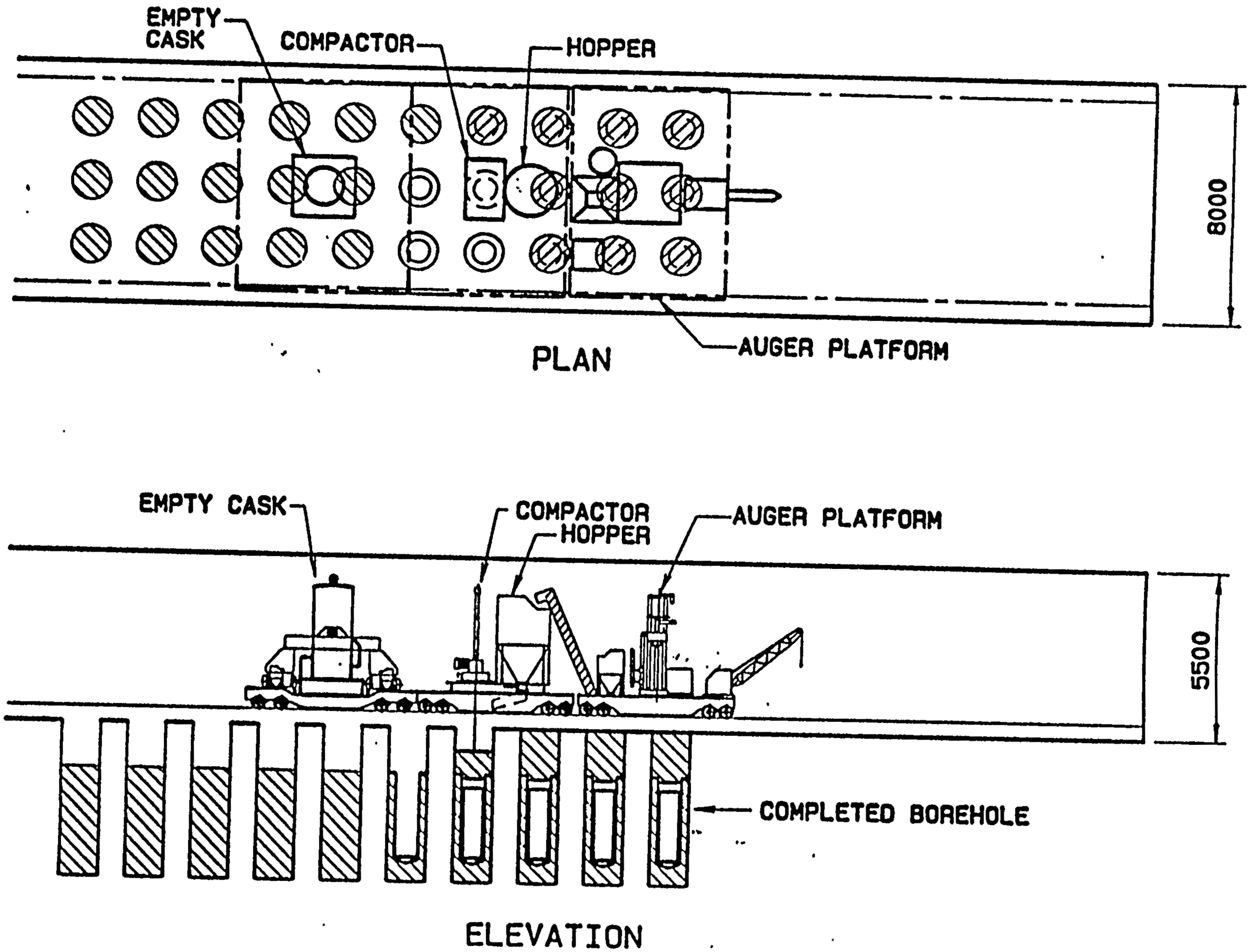

Figure 6.5.31. Final Buffer Emplacement

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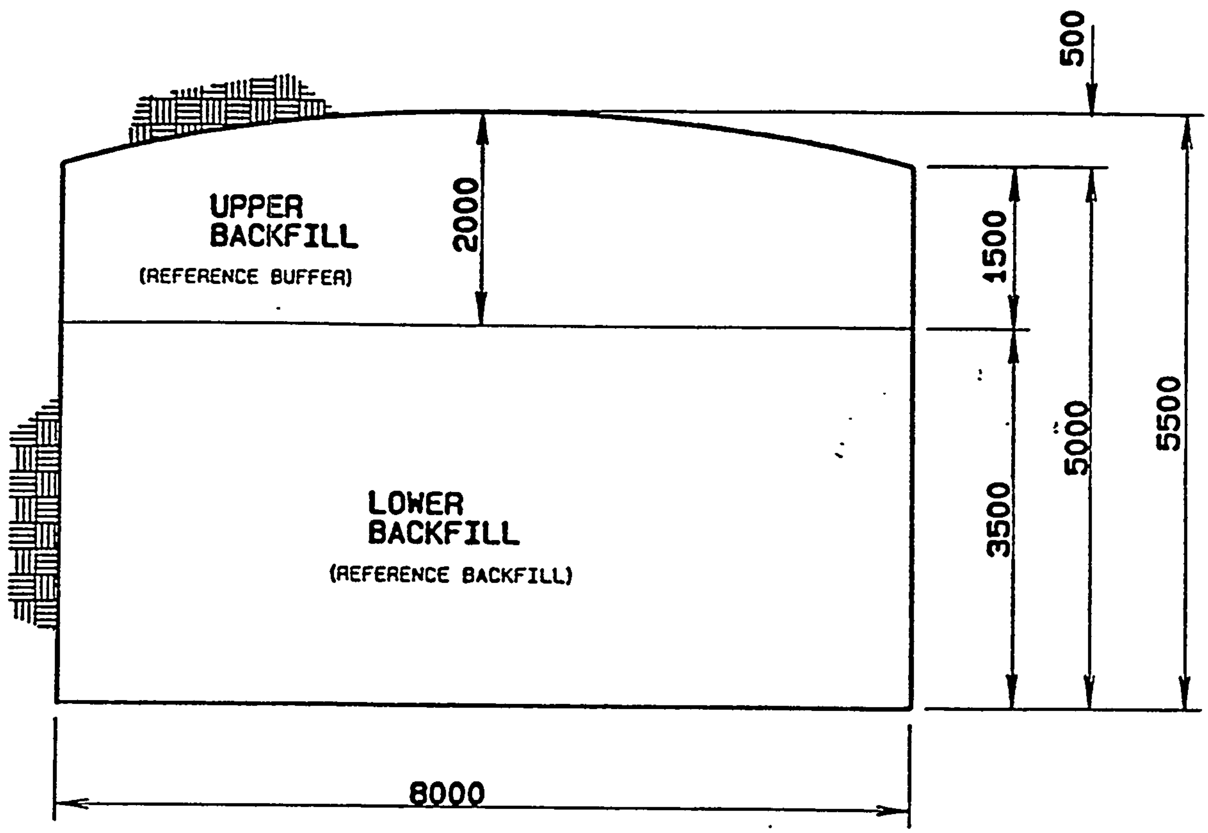

Figure 6.5.32. Room Backfill Distribution

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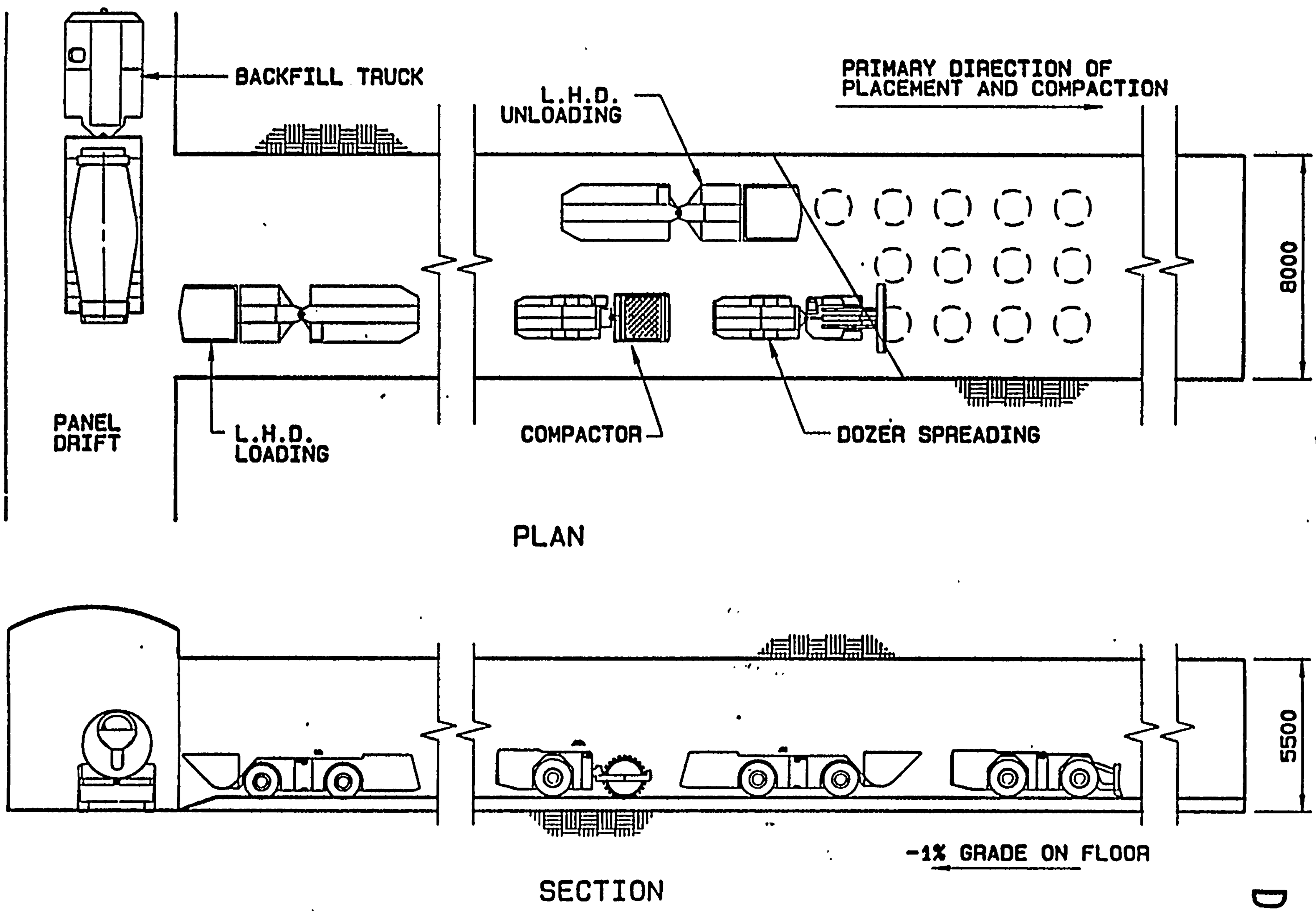

Figure 6.5.33. Lower Backfill Placement (Initial)

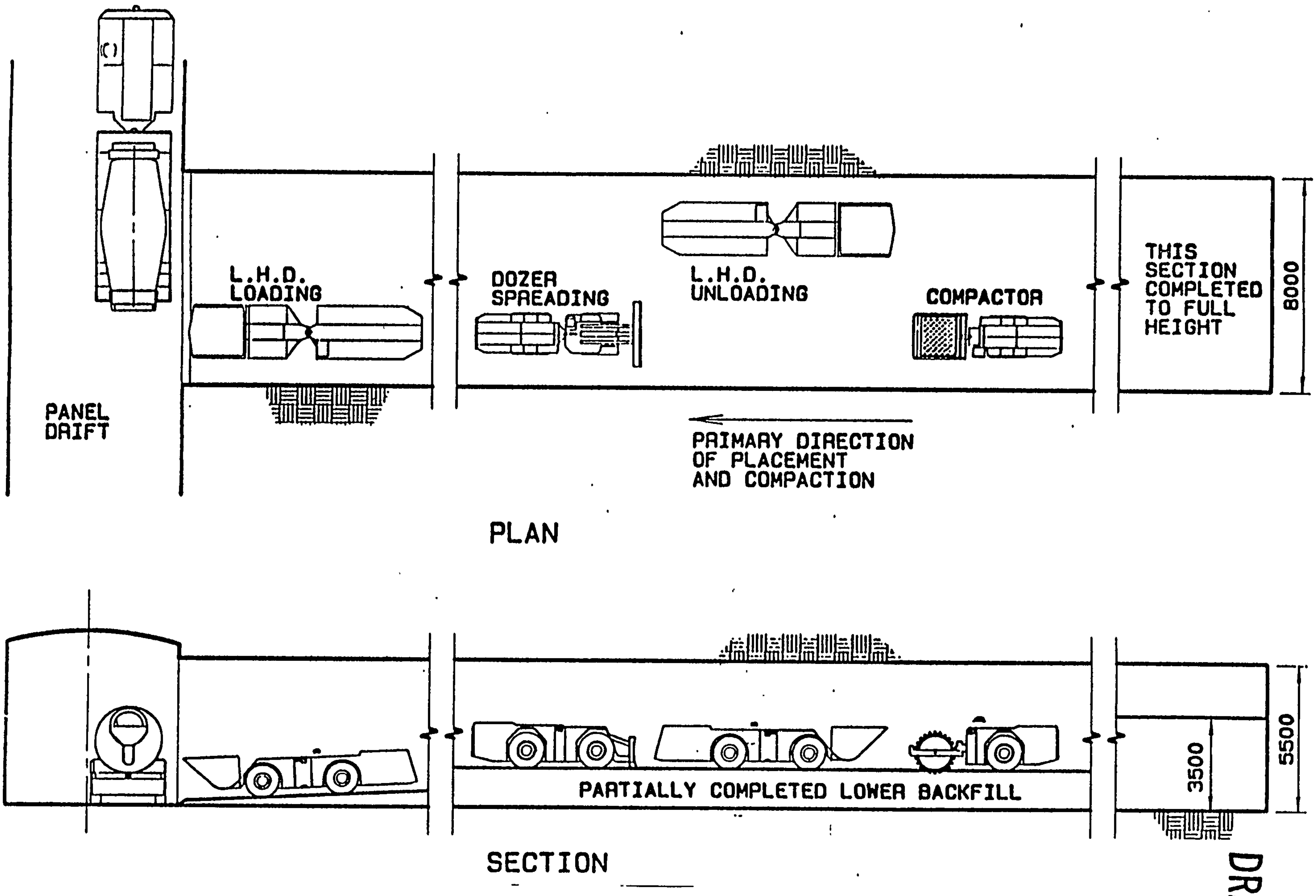

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Figure 6.5.34. Lower Backfill Placement (Intermediate)

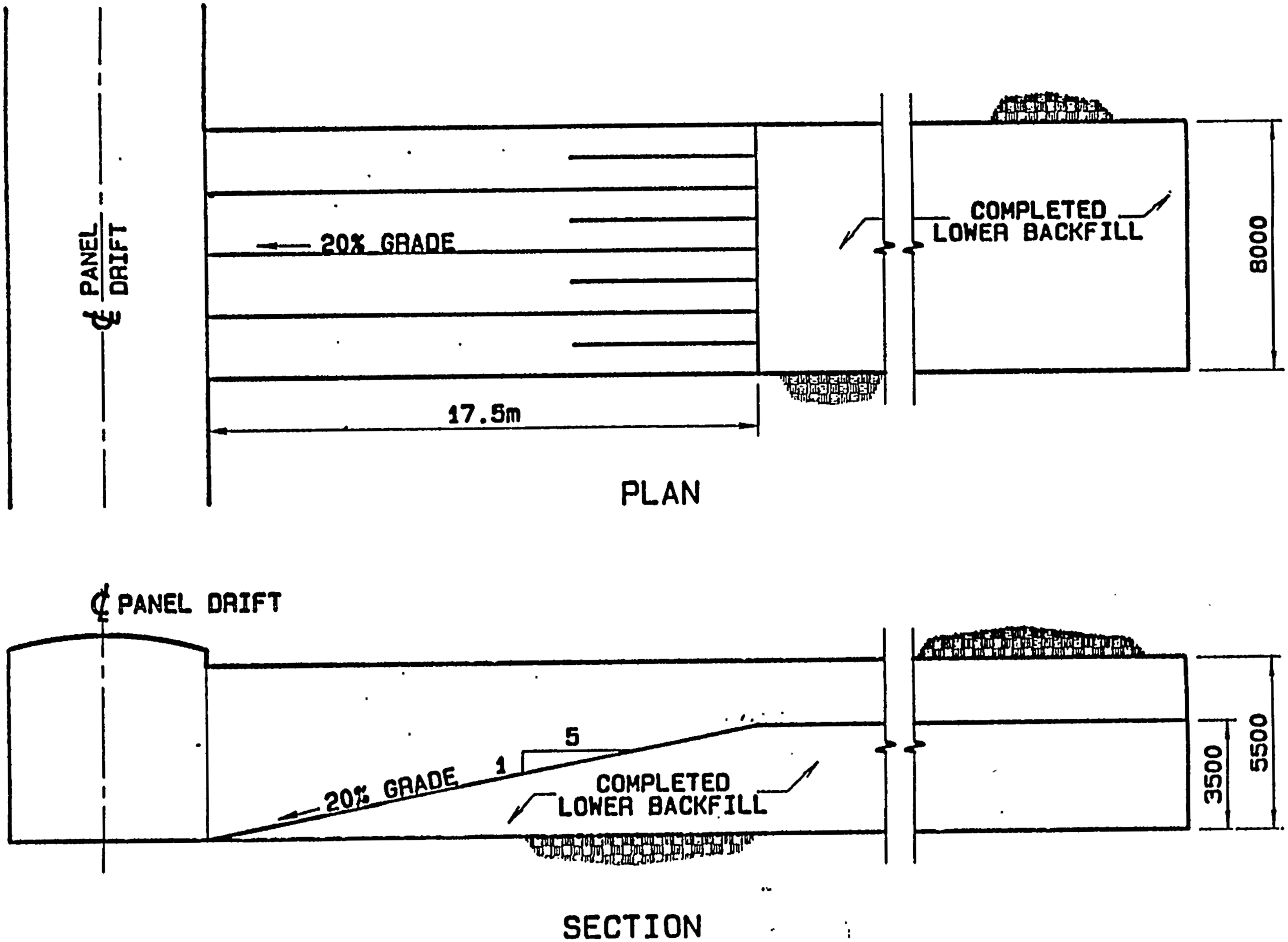

Figure 6.5.35. Lower Backfill Placement (Final)

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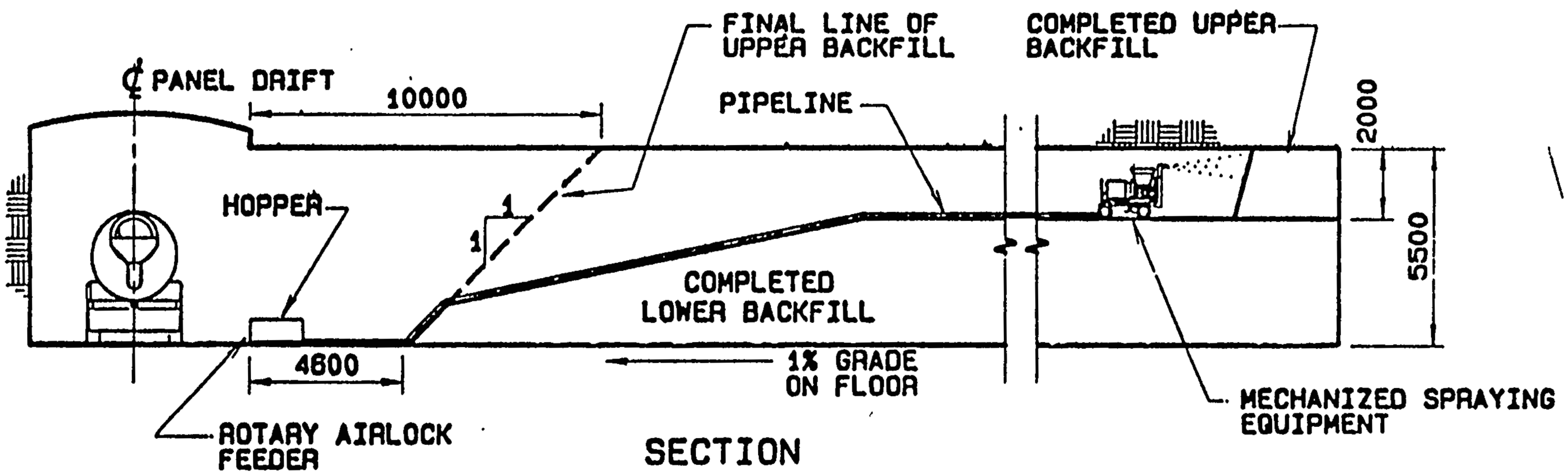


Figure 6.5.36. Upper Backfill Placement and Compaction

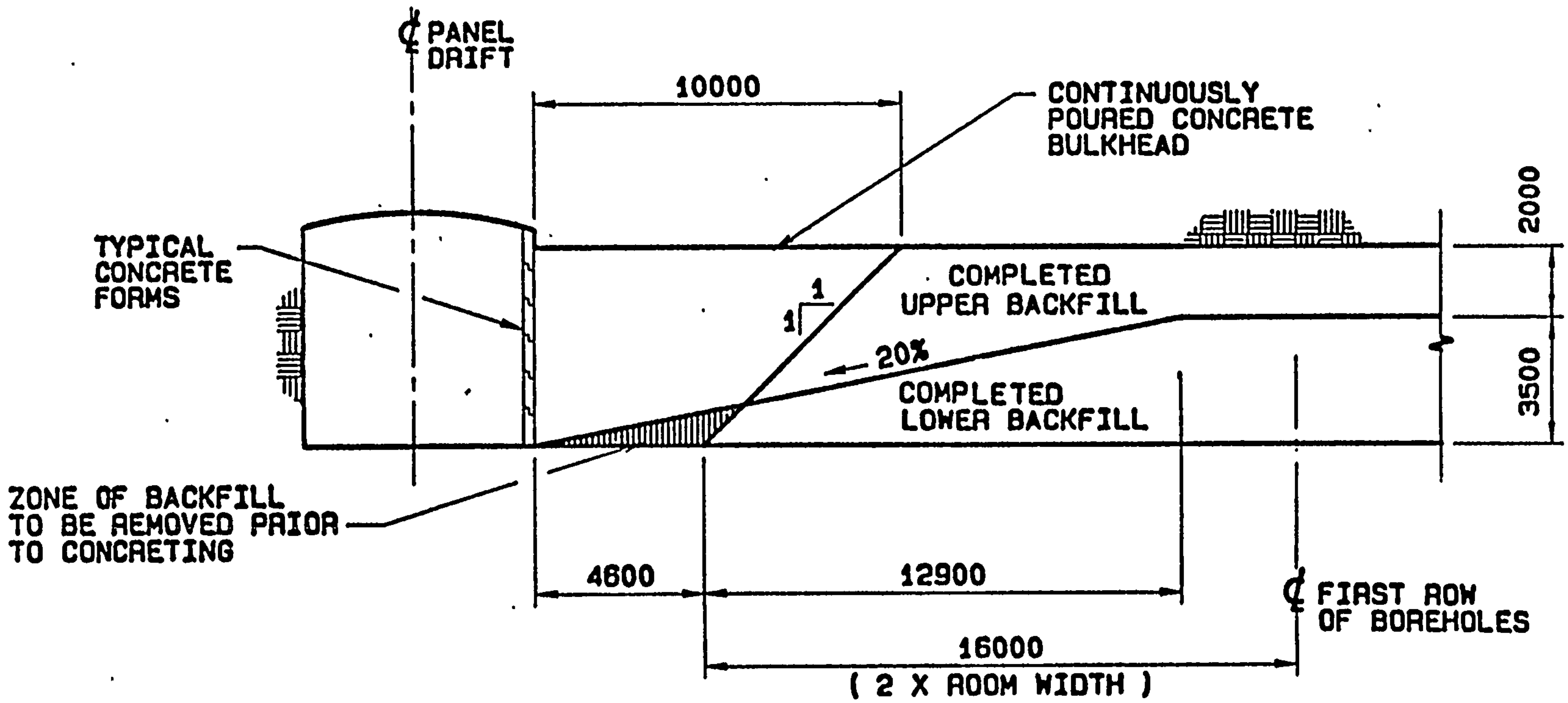
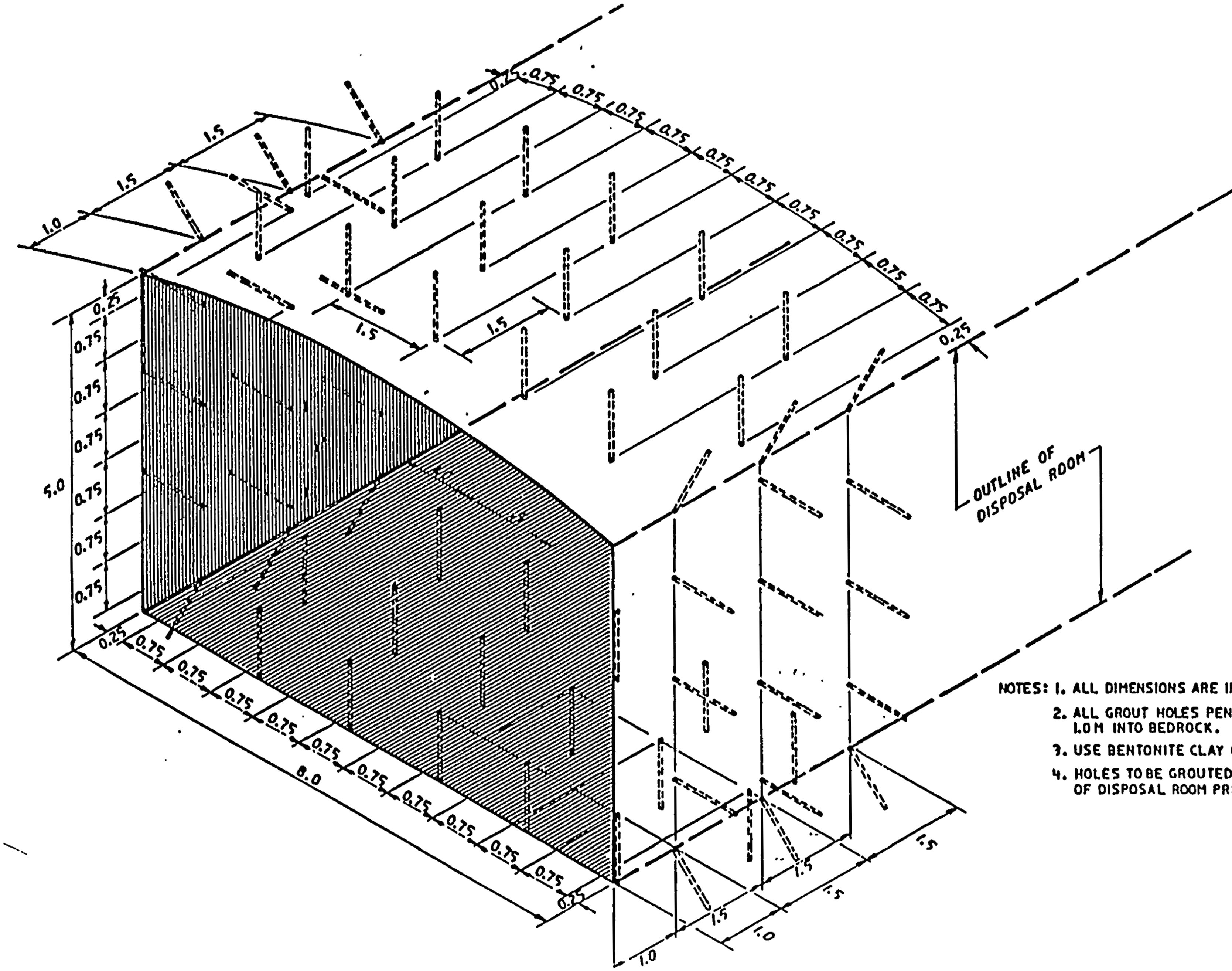


Figure 6.5.37. Final Backfill and Bulkhead



NOTES: 1. ALL DIMENSIONS ARE IN METRES.
2. ALL GROUT HOLES PENETRATE 1.0 M INTO BEDROCK.
3. USE BENTONITE CLAY GROUT.
4. HOLES TO BE GROUTED AS PART OF DISPOSAL ROOM PREPARATION.

Figure 6.5.38. Permanent Emplacement Room Bulkhead

The repository will be operated for its design life with emplacement room development, waste emplacement, and backfilling and sealing operations being conducted in a continuous concurrent operation. Figures 6.5.12 and 6.5.39 show the timing of the various operations. When emplacement operations are completed, the drifts and shafts will be backfilled and sealed (Figure 6.5.4). The surface facilities will be decommissioned and the site rehabilitated for public use.

6.5.5.5 Emplacement Schedule

A detailed schedule for the repository operation has not yet been derived but it is expected that the repository emplacement operations will last for about 40 years (Figure 6.5.12). The overall time for construction of the surface and underground facilities, waste emplacement, and backfilling and decommissioning operations has been estimated at approximately 105 years. For the description of this repository concept it was assumed that emplacement operations would start around 2025 and an annual emplacement rate of 4,730 MT/year (5,200 tons/year) would mean an operating life of about 40 years for the 191,000 total MT (210,000 tons) estimated to arise by the year 2035.

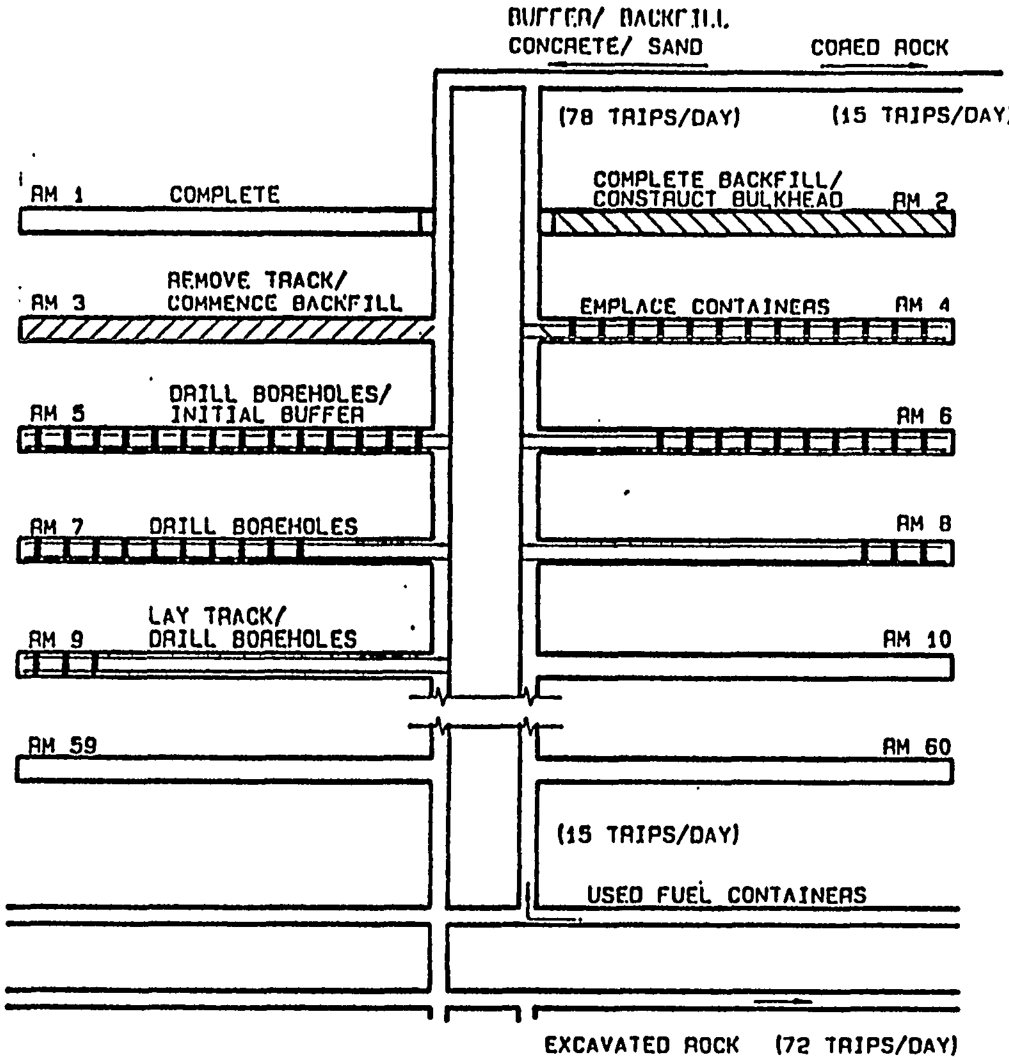
6.5.6 Buffer and Sealing Materials

Buffer and sealing material studies are currently underway and have led to a selection of a reference buffer material having a composition of 50 wt% sodium-bentonite clay and 50 wt% graded silica sand. This material will be compacted into the emplacement boreholes around the spent fuel canister (Section 6.5.5.4). This buffer material has several properties which make it suitable as a buffer material:

- Less than 2% volume decrease after drying
- A hydraulic conductivity of less that 5×10^{-2} m/s
- When exposed to water and confined, it exhibits self-healing properties and a swelling pressure which will seal the surrounding rock by forcing the material into cracks and joints
- A thermal conductivity high enough to allow adequate heat transfer from the spent fuel canister.

Analysis has shown that a vertical buffer thickness of about 1 m (3.3 ft) and a radial buffer of about 25 cm (10 in) thickness will be required for rock with low hydraulic conductivities. These dimensions have been incorporated into the emplacement hole design.

Studies of the backfill material for filling and sealing the emplacement rooms and access drifts derived an engineered material composed of 75 wt% of crushed granite and 25 wt% of lake bed clay. This material



EMPLACEMENT ROOM SEQUENCING

MONTH	TRACK & SERVICES	DRILL BOREHOLES	INITIAL BUFFER	EMPLACE CONTAINERS	REMOVE TRACK	BACKFILL ROOM	CONSTRUCT BULKHEAD
1	RM 1A	RM 1A					
2	2	2					
3	3	1, 3					
4	4	2, 4					
5	5	1, 3, 5	RM 1A				
6	6	2, 4, 6	2	RM 1A			
7	7	3, 5, 7	3	2	RM 1A	RM 1A	
8	8	4, 6, 8	4	3	2	1, 2	RM 1A
9	9	5, 7, 9	5	4	3	2, 3	2
63	3B	59, 1B, 3	59	58	57	56, 57	56
64	4	60, 2, 4	60	59	58	57, 58	57
65	5	1, 3, 5	1B	60	59	58, 59	58
66	6	2, 4, 6	2	1B	60	59, 60	59
67	7	3, 5, 7	3	2	1B	60, 1B	60
68	8	4, 6, 8	4	3	2	1, 2	1B

NOTE: 1. ACTIVITIES IN MONTH 9 SHOWN IN DIAGRAM
2. CONTAINER EMPLACEMENT INCLUDES AUGERING OF BUFFER AND FINAL BUFFER EMPLACEMENT
3. EQUIPMENT IN ROOMS 6 AND 8 NOT OPERATING IN MONTH 9

Figure 6.5.39. Emplacement Panel Sequence

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will also be compacted in place (see Section 6.5.5.4) and have hydraulic conductivity of about 10^{-10} m/s, which is similar to that of the surrounding rock mass.

The shaft backfill and sealing materials will likely require the use of grouts to seal zones of fracturing in the rock mass. The materials will have to be such that, after repository closure, the ground-water flow pattern is controlled by the natural rock conditions. The materials should therefore have a low hydraulic conductivity and some sorption capacity, be physically, chemically and mechanically stable for long time periods and be compatible with each other and the environment. Studies into the material compositions are currently being conducted at the Whiteshell Nuclear Research Establishment and the Underground Research Laboratory in Manitoba, Canada.

6.5.7 Summary

6.5.7.1 Advantages

1. The spent fuel canisters will be encased in engineered barriers in addition to the natural which will add to the overall effectiveness of the isolation system. Use of clay materials is also being considered by other programs (e.g., Sweden and Switzerland) as an addition to the natural barriers.

2. Plutonic rocks present a stable and competent environment for the long-term waste isolation.

3. Extensive mining and civil construction experience in the proposed rock formations exists in Canada and other countries and thus a firm basis exists for the repository construction methods.

6.5.7.2 Potential Problems

As with all underground repository programs, the main drawback is the uncertainty associated with the long-term behavior of the natural and engineered barriers and the impact of the construction methods, heat and radioactivity on the performance of the barriers. Major research programs are underway in Canada as well as many other countries to resolve these issues to ensure a reliable disposal concept is applied.

6.5.7.3 Summary

The Canadian program is of great interest to the United States since the United States program is also considering crystalline rocks as-potential repository host rocks. The United States is in fact monitoring the results of the Underground Research Laboratory project in Manitoba, Canada, very closely and is also providing funding and personnel assistance to the Atomic Energy of Canada Ltd., to support the experimental programs at the

Underground Research Laboratory. The results of these programs and other investigations which are part of the Canadian waste management program will be of great assistance to the United States programs.

6.5.8 The Low- and Intermediate-Level Concepts

Several low- and intermediate-level waste disposal concepts are being used or investigated in Canada. This section presents some examples of these.

1. Augered boreholes are used at various sites in North American and Ontario Hydro has installed them at their Bruce Nuclear Power Development Waste Operations Site-2. The wastes are placed in-ground containers (Figure 6.5.40) which are two concentric steel pipes separated by a shielding ring, if required. The containers are placed into augered boreholes in the ground which are lined with concrete (Armstrong).

2. Engineered trenches are being investigated for low-level wastes in conjunction with augered boreholes for intermediate-level wastes (Bechai, et al.) (Figure 6.5.41).

3. Disposal of wastes in large or small, shallow rock caverns in shales or limestones were investigated by Mansson, et al. Low-level waste drum would be stacked in unlined caverns and the voids backfilled as the caverns are filled. Intermediate-level wastes would be stored in concrete lined engineered caverns to provide adequate radiation shielding (Figure 6.5.42).

4. Disposal in shallow tunnels driven by machine in clayey soils or glacial tills was also investigated (Bechai, et al.). Waste disposal takes place in a series of parallel tunnels driven between two access ramps (Figure 6.5.43).

6.6 CRYSTALLINE ROCKS: THE FRENCH CONCEPT

This chapter summarizes the nuclear waste disposal concepts under investigation in France. France is one of the larger nuclear power producers in the world and currently has approximately 48 operating reactors. It is estimated that by the year 2000 about 20,000 MT (22,000 tons) of spent fuel will have accumulated. Reactor operation and reprocessing will result in the following waste quantities:

1. Category A (low-level):	700,000 to 900,000 m^3
	or 915,000 to 1,180,000 yd^3
2. Category B (alpha-bearing):	60,000 to 80,000 m^3
	or 79,000 to 105,000 yd^3
3. Category C (high-level):	3,000 m^3
	or 4,000 yd^3

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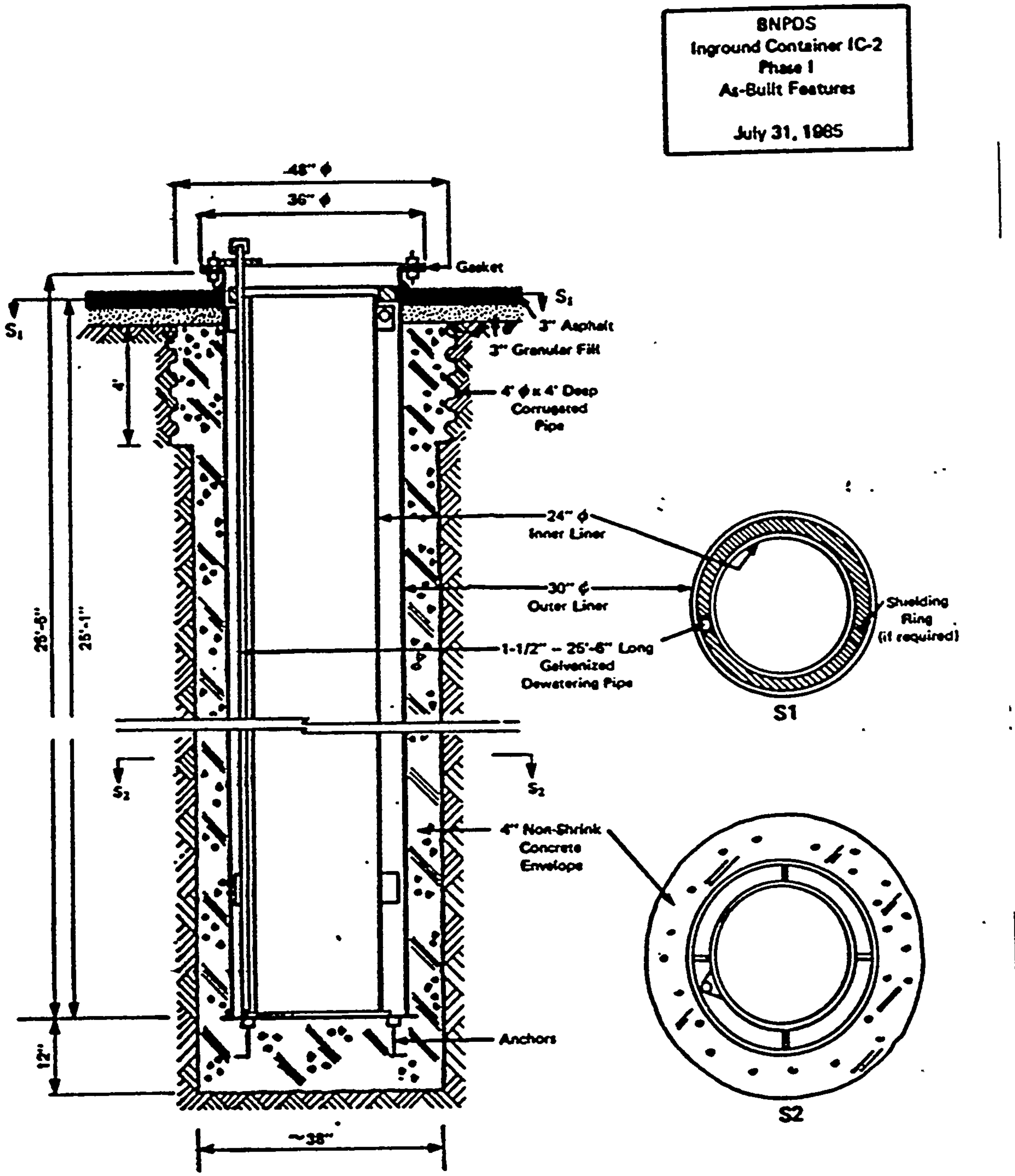
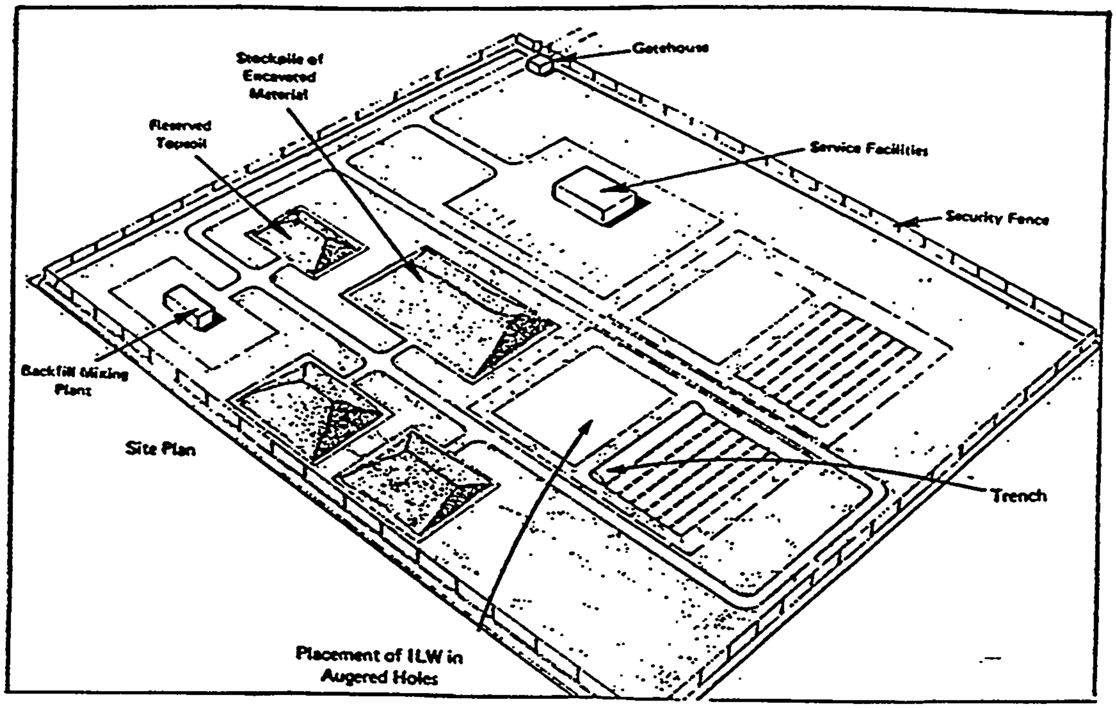
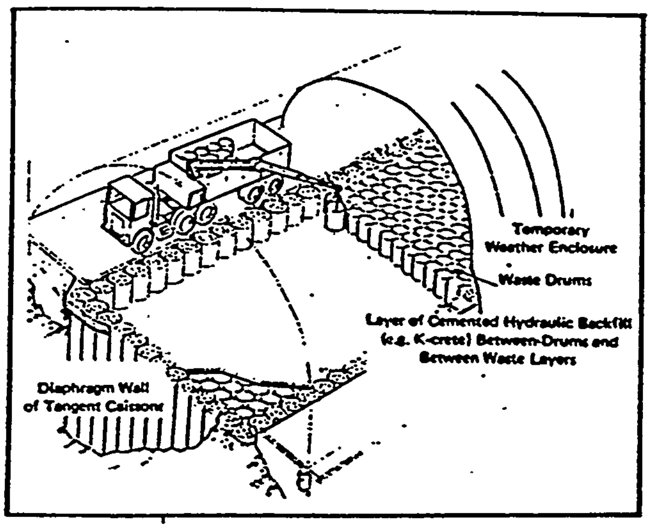
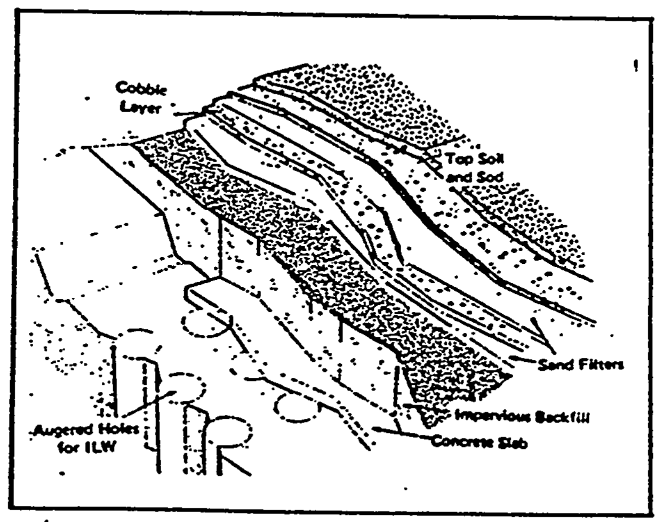


Figure 6.5.4Ø. Augered Borehole Disposal





OPERATION OF AN "ENGINEERED" TRENCH FOR LLW DISPOSAL



AUGERED BOREHOLE AND EARTH COVER FEATURES

Figure 6.5.41. Trench and Borehole Disposal

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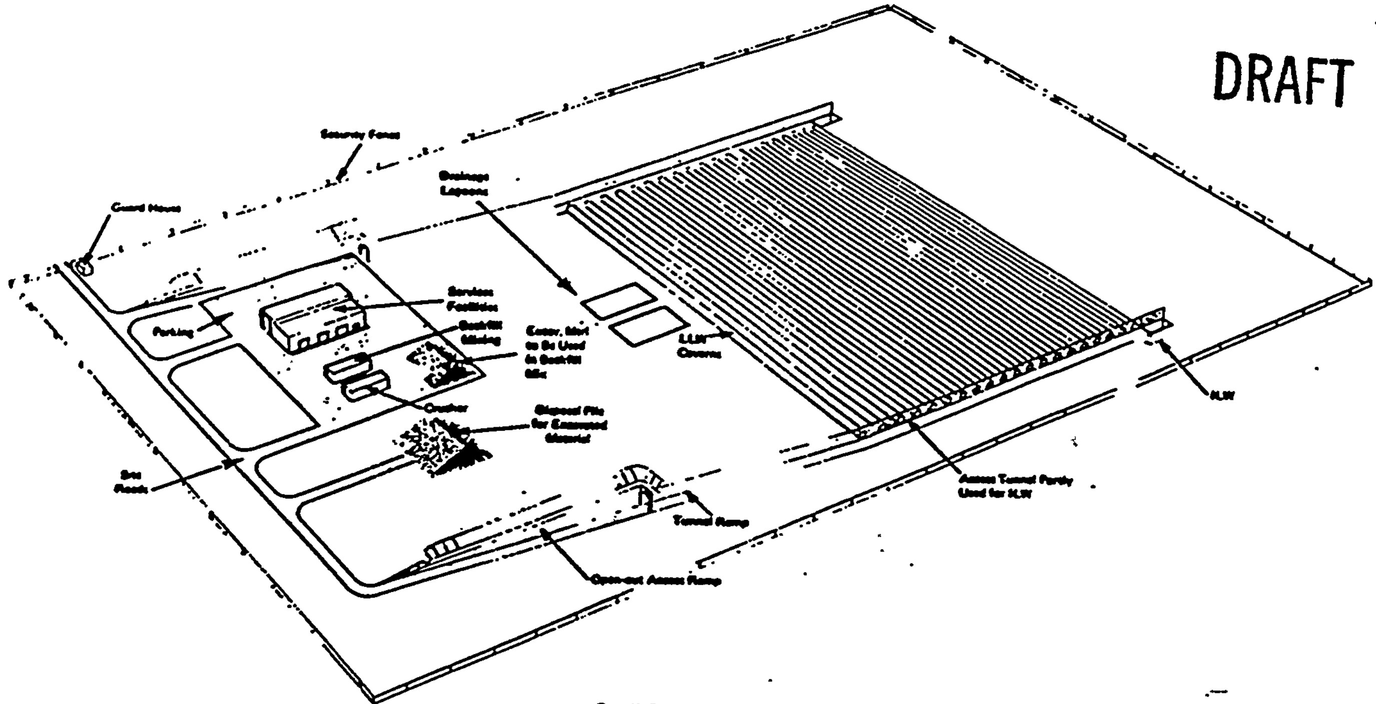
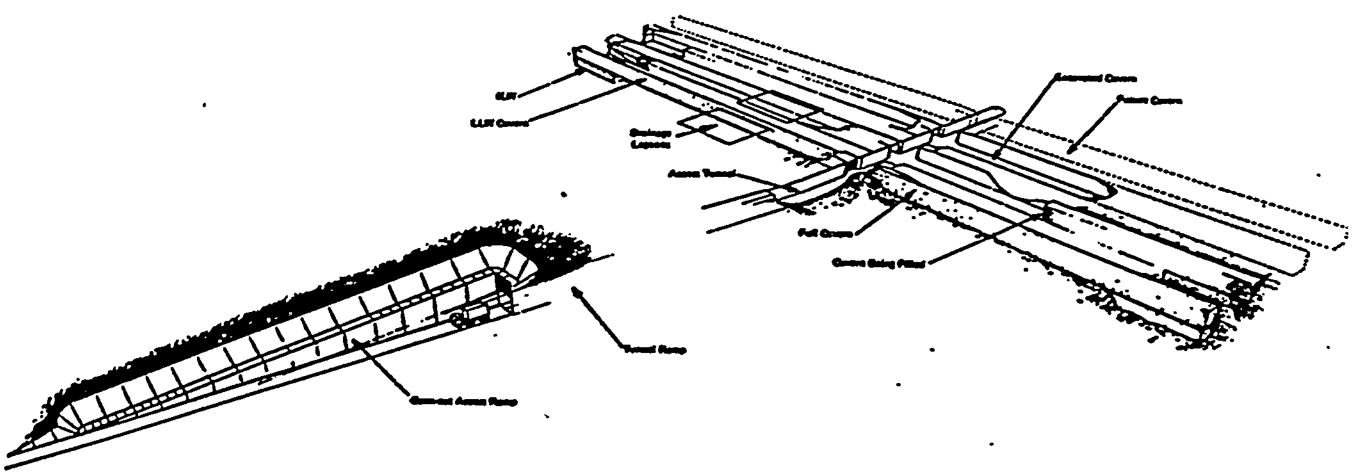


Small Cavern Scheme



Large Cavern Scheme

Figure 6.5.42. Large and Small Cavern Disposal

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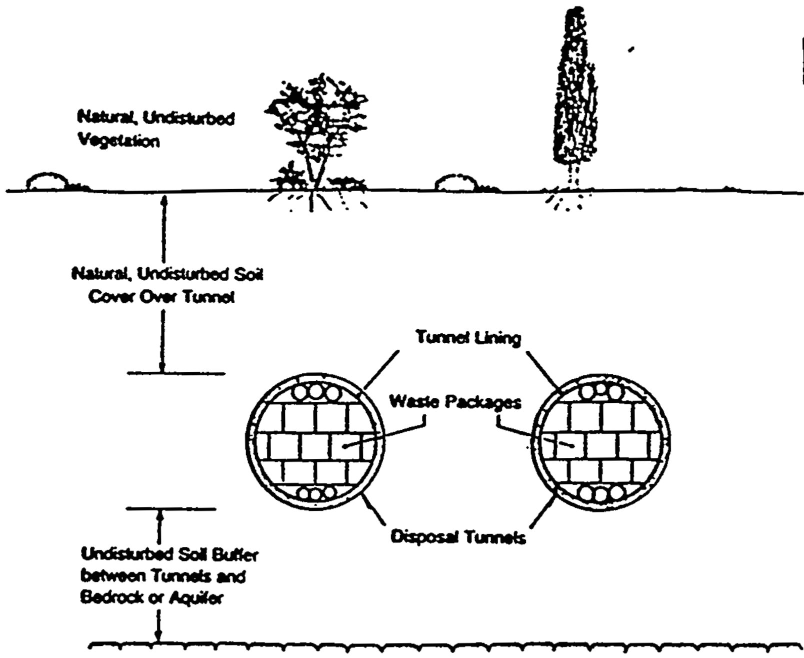
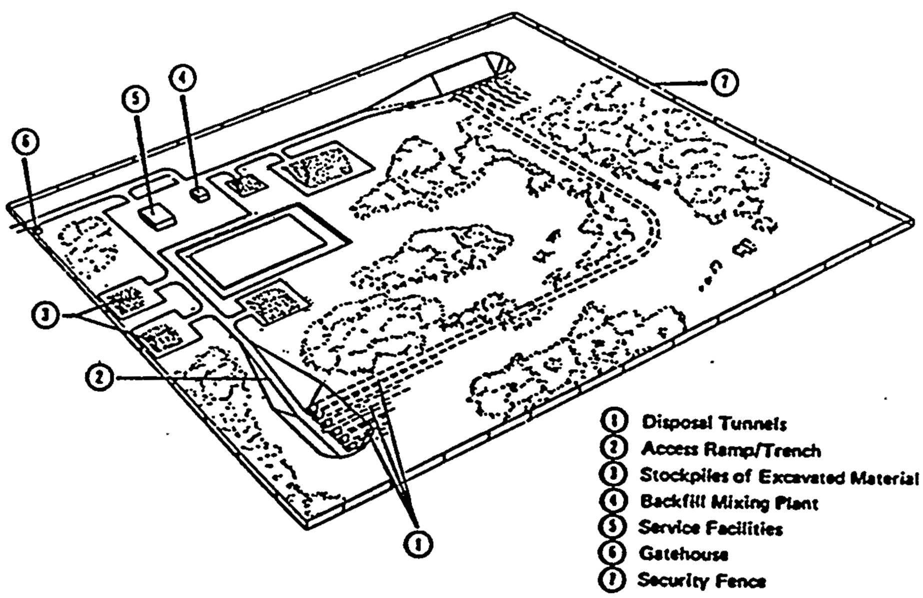


Disposal tunnels in till or clayey soil.



Waste management facility based on the shallow tunnel concept.

Figure 6.5.43. Shallow Tunnel Disposal

Burial in deep underground formations has been adopted in France for long-term isolation. No site has been selected as yet but site investigation work is planned to start at four locations in various rock types in the near future.

6.6.1 Geographic Location

The granites under investigation are located west of the city of Poitiers in the Departement Deux-Sèvres (Figure 6.6.1). Recently, the clay formations near Sissonne, north of Reims, the salt formations near Saint-Julien-sur-Reyssouze (northeast of Mâcon), and the Segré schists near Angers have been added as potential repository locations (Figure 6.6.1). In the next 3 years deep boreholes and seismic investigations are to be conducted at all four locations. A shaft will then be sunk at the best location and an underground laboratory established.

6.6.2 Summary of Geologic Setting

The final repository host rock formation has not yet been determined but investigations have until recently concentrated mainly on granites due to the French commitment within the European Community Cooperative research program to study the granites. Germany and Holland, for example, are studying salt formations, Belgium and Italy are concentrating on clays within this cooperative program. The first drilling was carried out in 1980 and several granite formations appear to have the necessary characteristics for a repository host rock. Recently, however, salt, clay and schist formations have also been added to the list rocks in France suitable for further investigation (Figure 6.6.1).

6.6.3 Repository Concept Description

This section describes the various disposal concepts under consideration for the higher activity nuclear wastes. Several repository configurations are under investigation. The final repository configuration will depend on the waste category, the host rock properties, waste canister handling procedures, etc. The following presents a summary of some of the variations published in the literature.

6.6.3.1 Nuclear Waste Disposal Strategy

The quantities of wastes that are estimated to accumulate in France by the year 2,000 were summarized previously. The wastes are categorized into three groups, depending on their level of radioactivity and heat production:

1. Category A (low-level wastes) wastes are produced during the operation of nuclear power plants, research facilities, medical facilities, etc.

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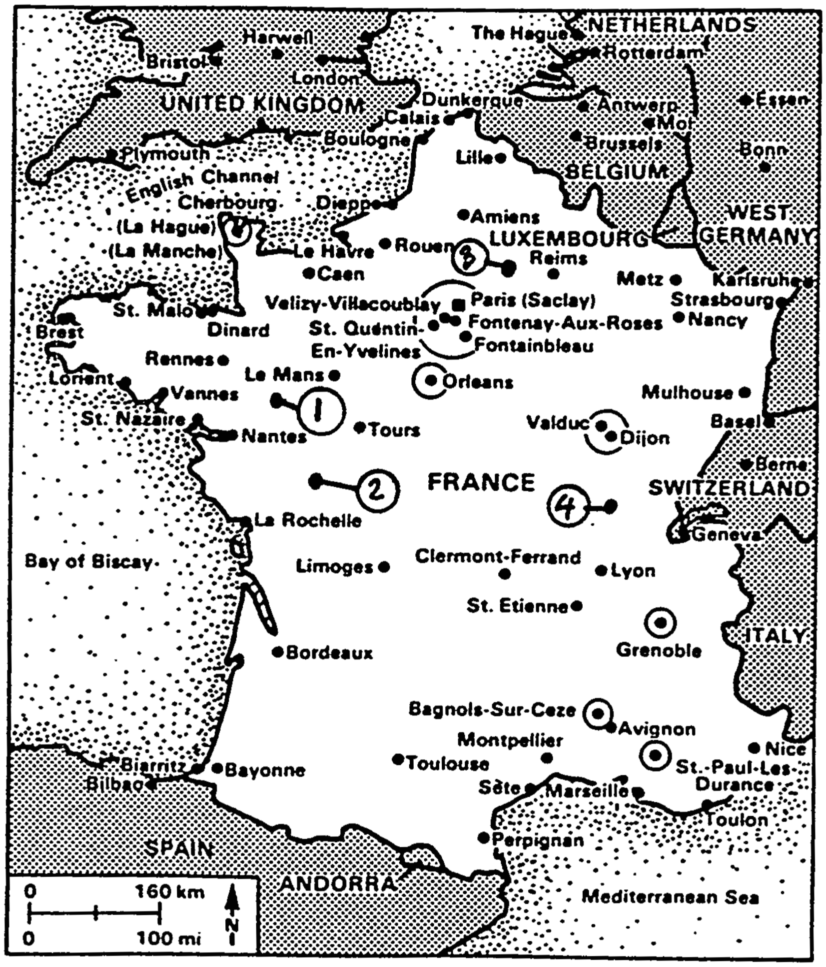

(1) - Schist
(2) - Granite
(3) - Clay
(4) - Salt

Figure 6.6.1. Site Investigation Areas

2. Category B (alpha wastes) are produced mostly (80%) during the spent fuel reprocessing operations while the remainder is produced during the fabrication of plutonium at various nuclear research centers.

3. Category C (high-level wastes) originate during the vitrification of the fission product and actinides produced during reprocessing of the spent fuel elements. These contain the longest living nuclides and produce a large amount of heat.

The disposal concepts described in the following sections deal only with Category B and C wastes but a brief synopsis of the Category A waste disposal concept is given in Section 6.6.8.

Two solutions are available for the highly radioactive spent fuel elements, namely the direct storage of the elements themselves and the reprocessing of the fuel and storage of the resulting wastes. France is heavily committed to the reprocessing option for several reasons:

1. For uranium supply reasons, it is advantageous to recover the residual uranium and plutonium in the spent fuel. The plutonium can then also be used to fuel fast breeder reactors.

2. Extraction of the plutonium from the spent fuel greatly reduces the hazards of the wastes and reduces the long-term shielding problems.

3. The irradiated spent fuel elements represent a much greater volume than the high-level wastes remaining after reprocessing and are not designed for long-term storage in pools.

4. Interim storage of the vitrified high-level wastes is achieved more easily due to the reduced volume of waste.

France is therefore operating and constructing industrial scale reprocessing and vitrification plants at various locations and also operates an interim storage facility where the vitrified wastes canisters are stored in air cooled shafts (Figure 6.6.2). Construction of additional storage facilities is planned. As most other countries, deep geological repositories are being investigated as the final, long-term disposal method for the alpha and high-level wastes.

As part of the European community research program, identification and investigation of final repository sites in granite formations in France has been undertaken and has recently expanded to include other formations such as salt, clay and schist. The objective of the deep geological research and development program concerns the characterization of, and the specification for, the various barriers around the waste package. The barriers include the geological and engineered barriers as well as the waste form itself. Since these barriers do not function independently of one another, the urgency for establishing an underground demonstration facility is dictated by technical rather than waste volume constraints. A disposal center for the low heat alpha wastes is expected to be ready by 1992 while a

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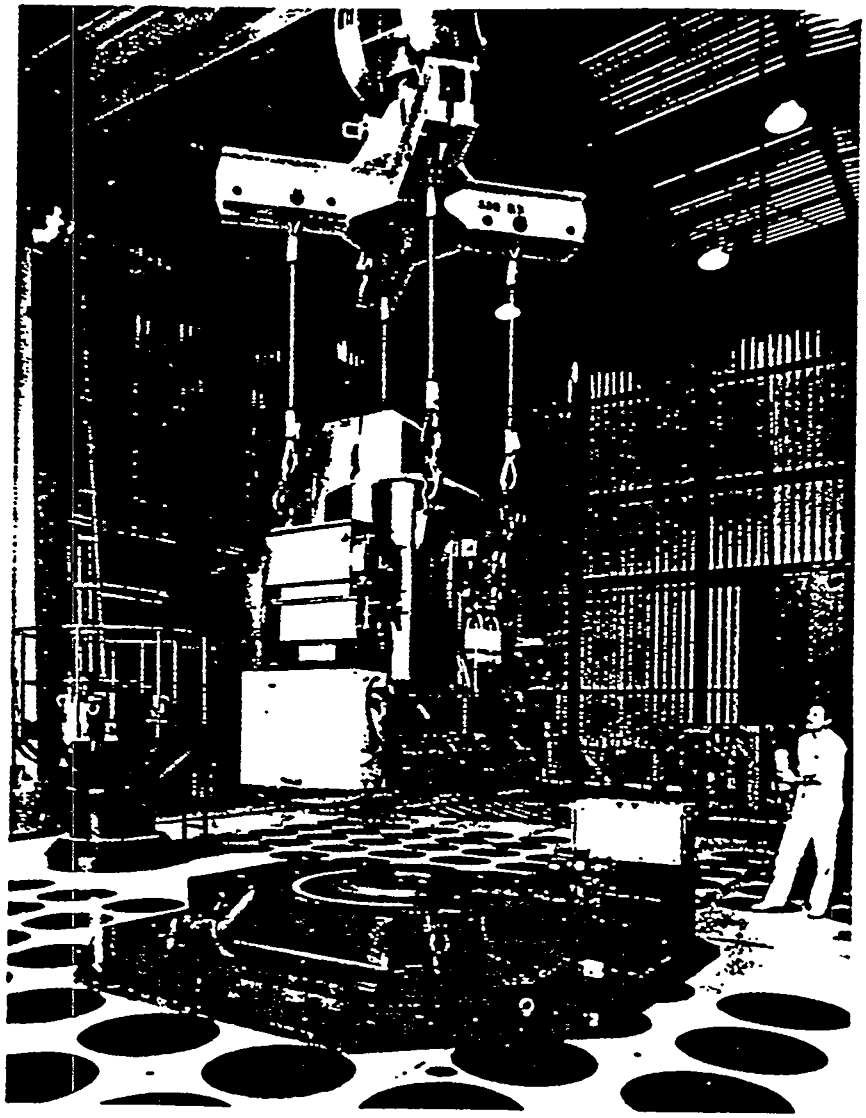

Figure 6.6.2. High-level Waste Storage Facility at Marcoule

demonstration facility will be used to demonstrate the feasibility of high-level waste disposal in geologic formations before final disposal actually takes place. This facility is expected to be operational by about 1993. If the site chosen for this facility is demonstrated to be suitable for waste disposal, the facility may be expanded into a full scale repository. Design of the final repository itself will depend on the interim storage policy adopted. Three options are being considered:

1. Cool the packages on surface for about 150 years and then bury them, leading to a more compact facility.

2. Cool the packages partly on the surface (about 30 years) and then bury them. This would lead to a large repository since the residual heat production is high and the packages must be separated more to prevent overheating.

3. Construct a compact repository and place the waste packages into the disposal areas a few years after vitrification. The packages would then be cooled in situ for a number of years until the temperature has dropped enough to allow final closure of the repository.

Evaluation of these options will be a high priority in the waste disposal program to allow start up of disposal operations by one method or other around the year 2010. Capacity of the repository will have to be about 30,000 high-level waste canisters with an emplacement rate of about 1,000 canisters/year. Repository depths up to 1,000 m (3,280 ft) are being investigated. The following sections describe some of the repository concepts currently under investigation.

6.6.3.2 Surface Facilities

Surface facilities will include the usual installations associated with underground mining operations such as hoists, headframes, ventilation equipment, underground services, shops, offices, change houses, mine rescue, security, etc. In addition, special reception facilities will have to be provided for the reception of the waste packages and preparation of them for transport and disposal underground. It is assumed these will be similar to facilities described in other sections of this report.

Several facilities for reprocessing, vitrification and interim storage of wastes are already in operation and others are under construction. These will have to be expanded as the waste inventory increases.

6.6.3.3 Underground Configuration

Different underground disposal concepts are being investigated, depending on the waste category. The following sections briefly outline the concepts that are being studied. The concepts basically break down into the following categories:

1. Category B wastes:

   a. Disposal in galleries with small cross sections (< 50 m^2 or 540 ft^2)

   b. Disposal in galleries with larger cross sections (>100 m^2 or 1,070 ft^2)

   c. Disposal in silos.

2. Category C wastes:

   a. Disposal in boreholes with or without in situ cooling of the waste packages by air convection

   b. Disposal in galleries with or without in situ air cooling of the waste packages.

Depth of the repository can be up to 1,000 m (3,300 ft).

6.6.3.3.1 Category B Wastes - Small Galleries. The disposal galleries or headings in this concept have a cross section between 15 to 40 m^2 (160 to 430 ft^2), depending on the method used to shield the waste canisters during handling and the mechanical properties of the host rock. The disposal galleries are excavated on either side of an access heading and have a length of approximately 300 m (980 ft) as shown in Figure 6.6.3. The repository may have a single or multiple level arrangement of access headings excavated in a rectangular pattern (Figure 6.6.3) from the shafts. The waste canisters are stacked such that the cross section of the gallery is utilized to the fullest extent. Several shafts will link the repository to the surface and will be equipped to serve as access for the waste canisters, personnel, material and services, and intake and exhaust ventilation. A service and waste canister reception area will be excavated at the base of the shafts. This area will also serve as a staging point for the development of the access drifts and emplacement galleries and for site investigation activities. This type of repository gives a great deal of control over the underground operations and the rate of emplacement room development, canister emplacement and backfilling can be adapted to the anticipated delivery rate of the waste canisters.

6.6.3.3.2 Category B Wastes - Large Galleries. In this concept, a network of one or more sets of parallel double entries is driven in a circle from the shafts to the boundaries of the repository as shown in Figure 6.6.4. The entries will have a rectangular profile of greater than 100 m^2 (1,070 ft^2) and one entry will serve for waste handling and the other for ventilation. The access and waste handling shafts will be grouped together in a underground repository service area. The waste storage galleries will be outfitted with concrete storage cells in the lower half of

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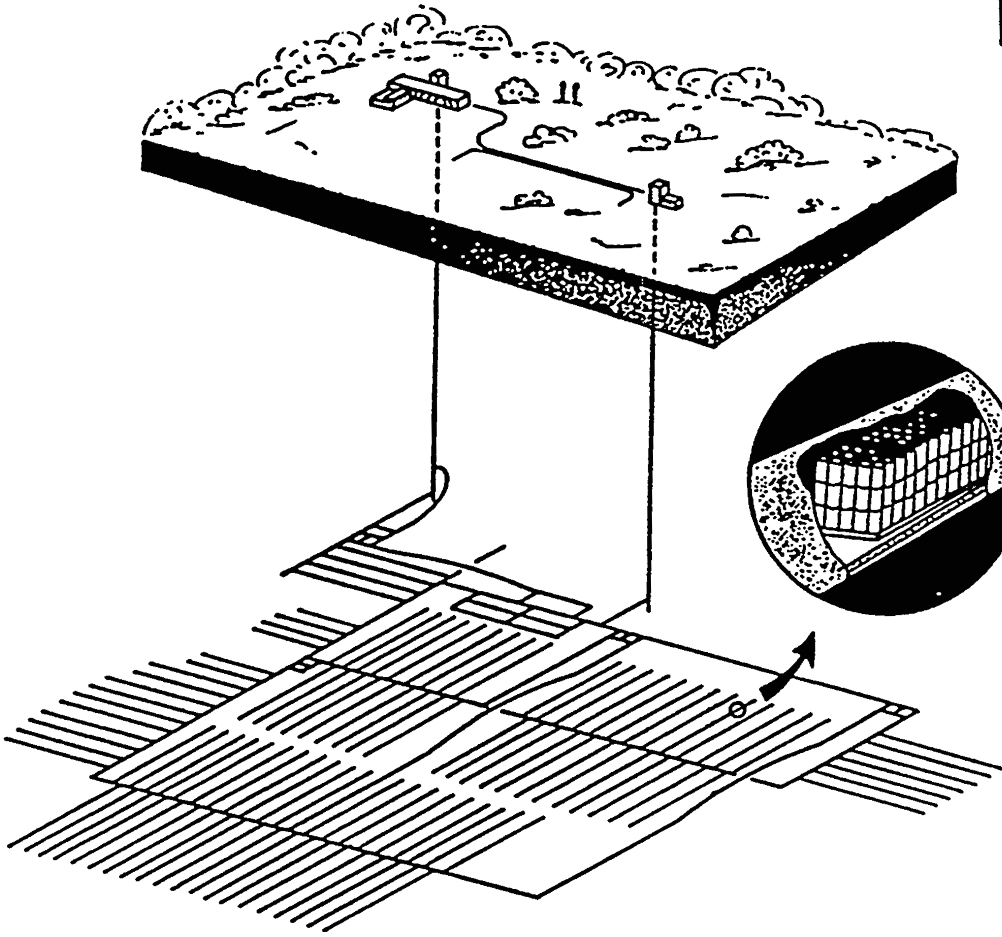

Figure 6.6.3. Category B Repository: Small Galleries

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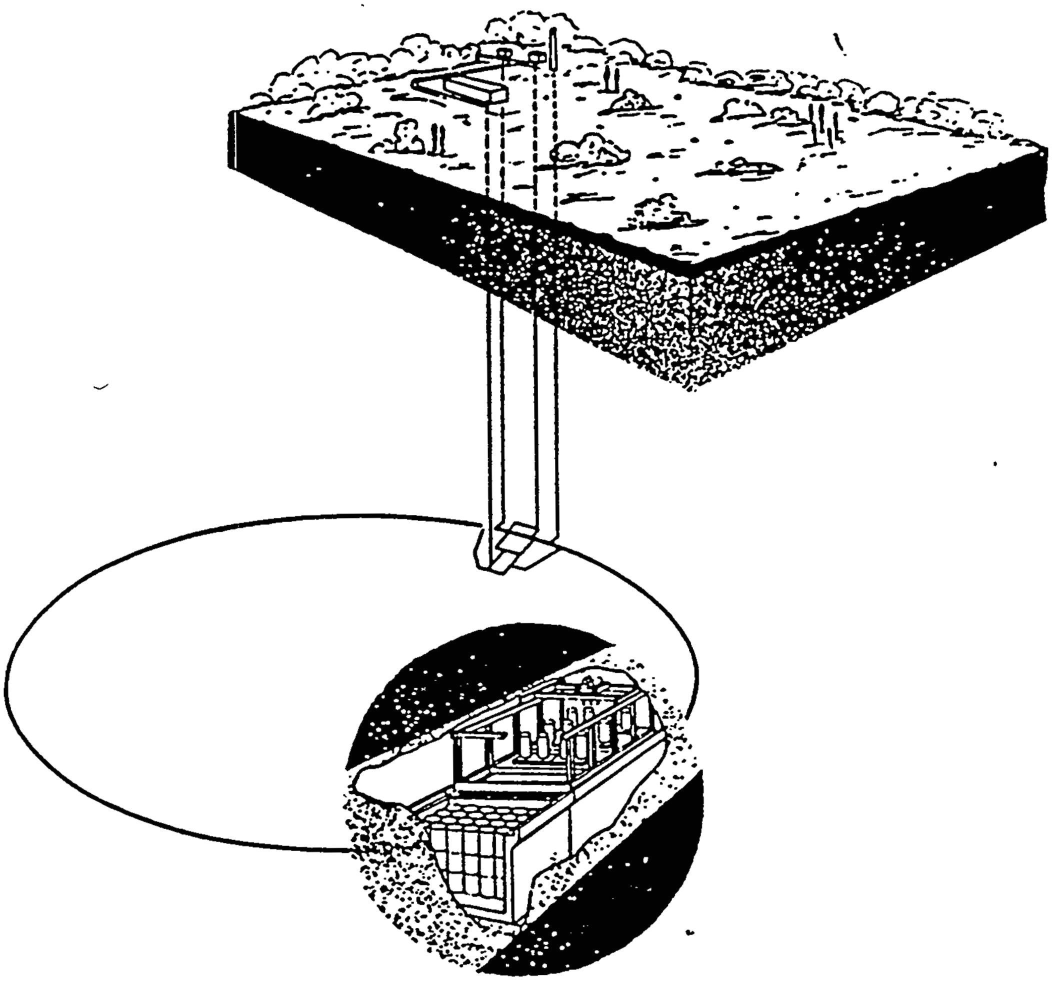

Figure 6.6.4. Category B Repository: Large Galleries

the entry. An overhead crane will manipulate the waste canisters in the tunnel and place them in their designated storage locations in the concrete storage cells. Monitoring of the waste canisters in the repository is thus possible for long periods of time. This type of repository could also act as a long-term storage facility if final disposal of the wastes has not been determined and has the advantage of relocating the interim storage operations underground from the surface. This concept does, however, require greater initial capital outlay than the small section gallery concept described in the previous section.

6.6.3.3.3 Category B Wastes - Silos. The waste canisters will be placed in cylindrical silos with a diameter from 6 to 30 m (20 to 100 ft) and a height of about 30 m (100 ft). A series of access and transport drifts will link the silos to the shafts. The waste canisters will be manipulated in the silos with overhead cranes which can place the canister in any part of the silo.

6.6.3.3.4 Category C Wastes - Non-Ventilated Boreholes. A number of blind shafts or large diameter boreholes are drilled at regular intervals into the floor of a network of parallel headings as shown in Figure 6.6.5. A number of waste canisters will be placed in each borehole, in stacks one or more canisters high, depending on the residual heat production of the canisters and the host rock properties (Figure 6.6.5). Two main shafts will be used for waste transport, personnel and material transport, ventilation and removal of excavated rock. Additional ventilation shafts will be located as needed.

6.6.3.3.5 Category C Wastes - Ventilated Boreholes. The bottoms of the disposal shafts or boreholes described in the previous repository concept are connected to the ventilation shafts by an additional, lower network of drifts. This lower ventilation network allows cooling of the waste canisters actually in the boreholes, which can reduce the interim storage period on surface. A network of air ducts on the lower level directs fresh air to the base of each borehole. The boreholes are equipped with a series of air ducts throughout their entire length, each duct enclosing a stack of waste canisters. Fresh air passes continually through these ducts from the bottom of the borehole to the top, withdrawing heat from the waste canisters. At the top of the boreholes, the hot air is routed to the exhaust shaft via a second network of ducts. After the cooling period, the repository will be finally backfilled and sealed.

6.6.3.3.6 Category C Wastes - Non-Ventilated Galleries. In this concept the waste canisters are placed horizontally or vertically on the floor of the disposal gallery (Figure 6.6.6). The disposal galleries are excavated horizontally on either side of a main service heading in a manner similar to the repository described for the Category B wastes above. This

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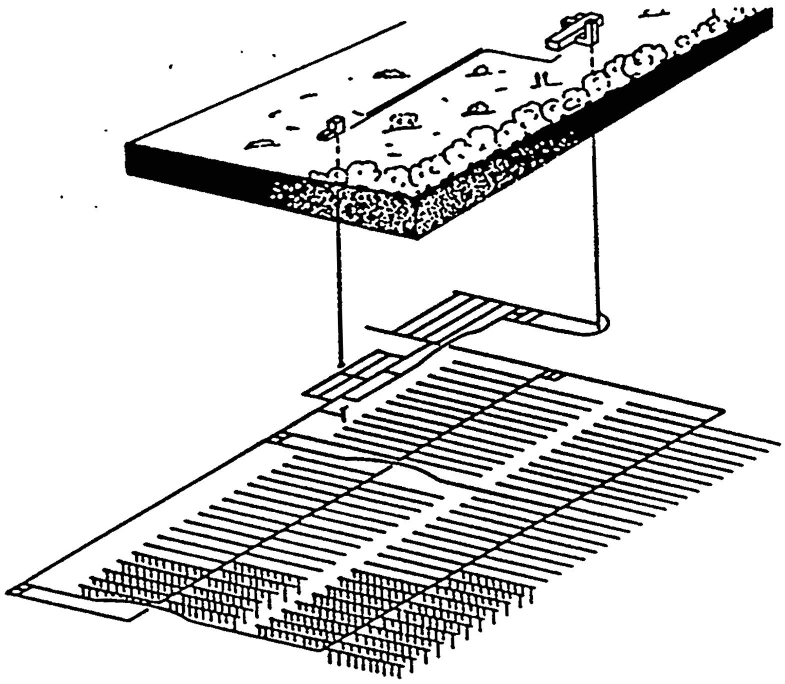

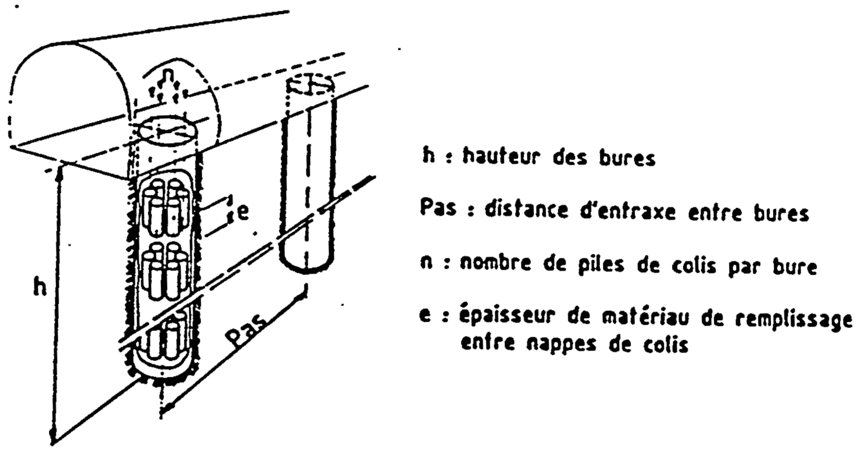

Figure 6.6.5. Category C Repository: Large Boreholes

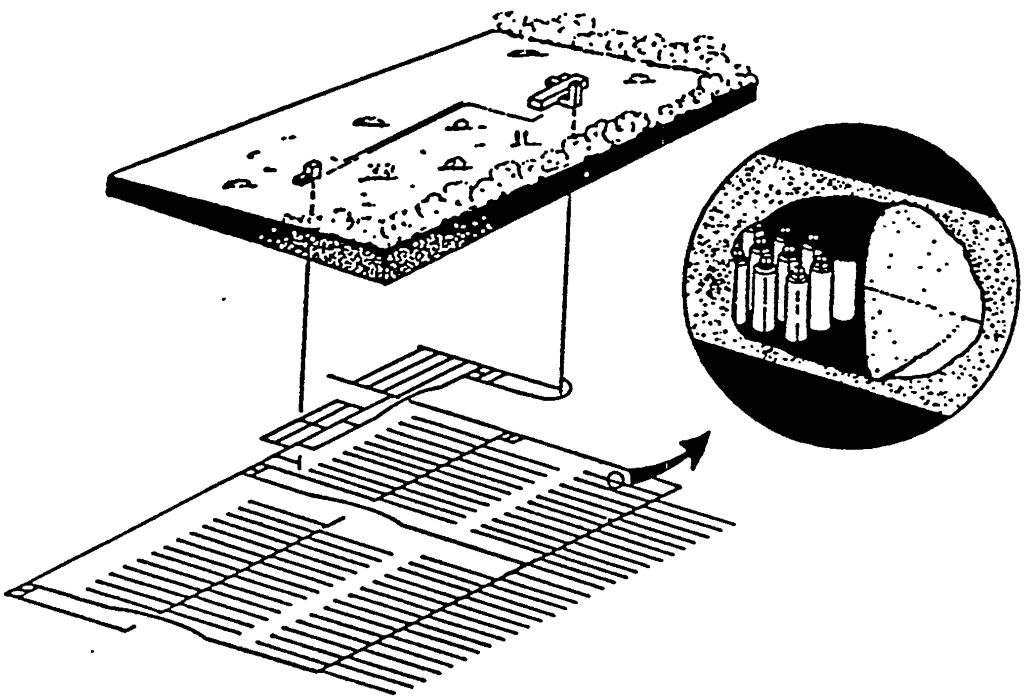

Figure 6.6.6. Category C Repository: Non-ventilated Galleries

concept does not afford as great a degree of control over the waste canisters as the borehole concepts but may be more suitable for lower strength rock formations.

6.6.3.3.7 Category C Wastes - Ventilated Galleries. This repository consists of a series of horizontal disposal galleries with access headings to serve as waste handling and ventilation routes. A concrete or metal structure is constructed along the entire length of the disposal gallery as shown in Figure 6.6.7. This structure will contain an air compartment in its base for the entire length. Cold air is introduced into this compartment and will pass from the compartment through vent holes into the upper part of the gallery, containing the waste canisters. The air picks up heat from the canisters as it travels through the disposal drift to the return airway which routes it to the exhaust shaft.

6.6.4 Repository Construction

This section outlines some of the construction techniques that are under investigation to construct the repository. The final techniques will be determined by the available technology, rock properties, repository configuration, etc.

6.6.4.1 Schedule

The French Atomic Energy Commission plans to select a potential site by the end of 1989 and will build an underground research laboratory there. Construction of the underground laboratory is to begin in 1989 and site evaluation test are expected to start in 1992. Depending on the results, the facility may be expanded into a full-scale repository. Operation of the high-level repository is not, however, expected to start before 2010. Disposal will take place over a 30-50 year period.

6.6.4.2 Methods and Equipment

The repository construction methods are generally dictated by the host rock type in which the repository is to be constructed. Granite is the main rock type under consideration but salt and argillaceous and metamorphic formations are also being studied.

In granite, two main excavation methods are being considered:

1. Drill-and-blast methods have few restrictions as regards the configuration of the excavations but may cause severe damage to the surrounding host rock. To limit this damage, application of pilot-and-slash methods and controlled blasting techniques will be necessary. Other methods such as high-pressure water jet cutting to slot the rock are also under investigation. The requirement for controlled blasting and reduced rock damage greatly reduces advance rates.

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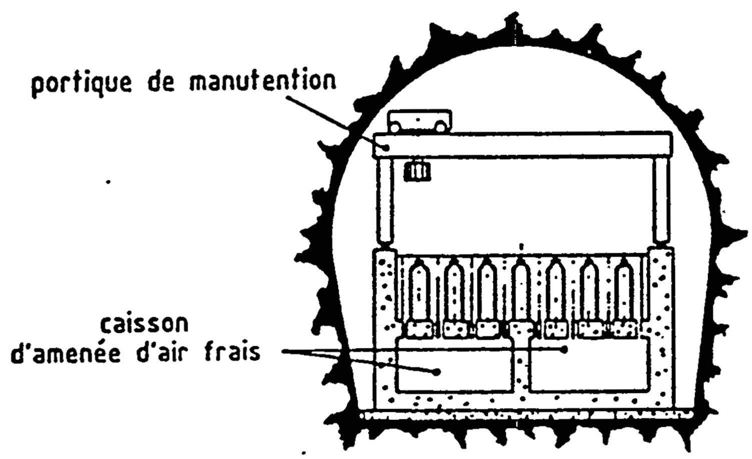

Figure 6.6.7. Category C Repository: Ventilated Galleries

2. Excavation with full face tunnelling machines or partial face machines such as roadheaders for disposal gallery construction and application of raise boring or blind hole drilling techniques for shaft and borehole construction ensures minimum damage to the rock mass surrounding the excavation. However, these methods do have several constraints:

   a. Limited to circular configurations and limited excavation diameters (10 to 11 m or 33 to 37 ft for full face tunnelling machines; 3.5 to 5 m or 11 to 16 ft reach for roadheaders; 3 to 4 m or 10 to 13 ft for raise borers)

   b. Full-face tunneling machines are not economical for short excavations since assembly and dismantling is expensive and time consuming

   c. Full-face tunneling machines are not suitable for excavating sharp turns

   d. Application of raise borers requires access to the bottom of the borehole.

Mechanical methods are also not applicable for construction of the silos, other than for drilling a small pilot hole.

6.6.5 Waste Emplacement Cycle

6.6.5.1 Waste Treatment

The French nuclear waste disposal program is based on reprocessing of the spent fuel with disposal of the alpha and high-level wastes in geologic repositories. The process chosen for the solidification of the high-level waste solutions produced by reprocessing is vitrification into borosilicate glass. This process has been studied intensively in France for more than 20 years and industrial scale reprocessing and vitrification facilities are currently operating or under construction at Marcoule and La Hague. An interim storage facility for the vitrified wastes is also operating at Marcoule. The vitrified reprocessing wastes will be stored in this facility in sealed reinforced concrete shafts for many decades until the repository is ready or an alternative interim storage solution has been found.

Several alternatives are being used or investigated for the Category B waste treatment:

- Embedding in bitumen
- Embedding in thermosetting resins
- Incorporation into concrete.

6.6.5.2 Waste Transport

The French nuclear industry has developed an effective and safe transport system to transport the spent fuel from the reactors to the reprocessing facilities. Special transport shielding casks have been developed for overland transport by truck or rail. It is assumed that the same system would be used to transport the waste canister to the repositories.

6.6.5.3 Waste Package

Interim storage is intended to maintain the glass containing the high-level wastes at a temperature below a specific limit and to cool the glass containers long enough to allow disposal in a geologic repository. At the interim storage facility, the waste canisters are stacked inside vertical metal tubes embedded in a concrete vault. The glass canisters are made of a special stainless steel, each holding about 150 l (40 gal) of glass (Figure 6.6.8). The containers are force-air cooled in the storage facility. Although current plans are not to use an overpack, ongoing studies are evaluating the possibility of using different ceramic or metal overpack materials. In addition, investigations into the use of engineered barriers or buffer materials in the repository are in progress.

6.6.5.4 Underground Waste Emplacement

The underground waste canister handling procedures are still under investigation and will depend on the various forms of waste packages:

- Canisters may be handled singly

- Canisters may be handled in groups on pallets (Category B waste) or in shielded or unshielded containers (Category C wastes) to reduce some of the emplacement operations

- Encase the category B waste canisters in concrete, for example, which will also act as a buffer material.

The hoisting and underground transport of the waste packages can be done with or without shielding. Unshielded transport is being studied in particular because of the possibility of using remote controlled operation and the lower transport loads of the unshielded waste package. A loader could be used to transport the waste canisters in the case of the smaller gallery disposal of the Category B or C wastes (Figure 6.6.9). For the large section Category B waste galleries and the Category C waste boreholes a large crane and winch arrangement, adapted for the excavation configuration, could be used.

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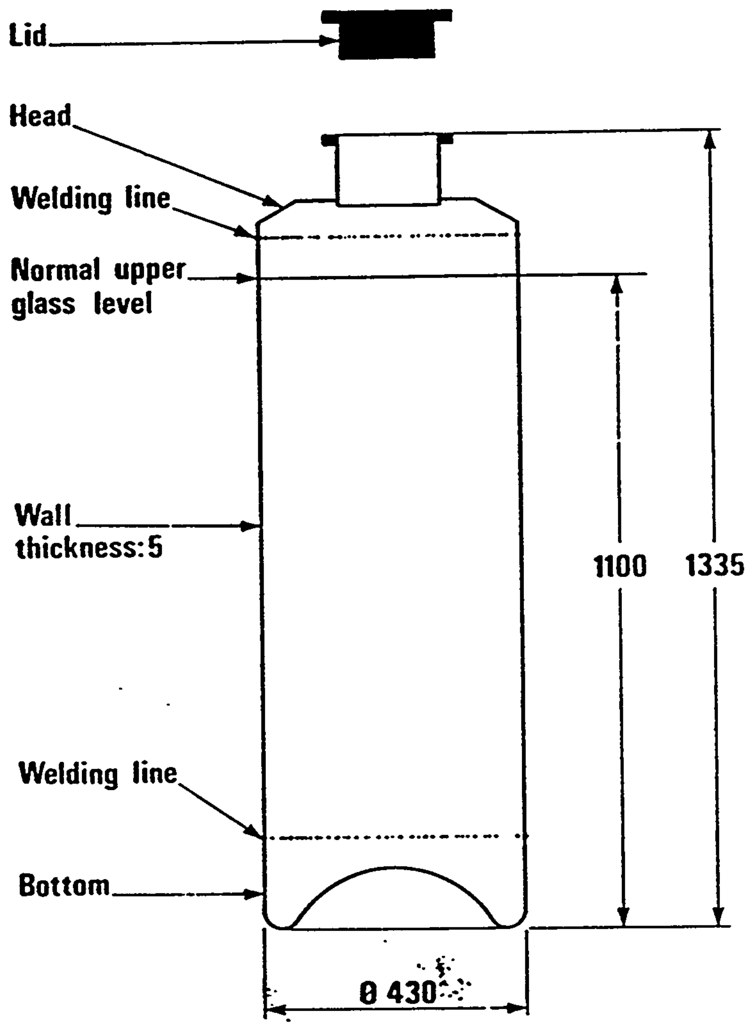

Figure 6.6.8. Vitrified High-Level Waste Canister (R7/T7)

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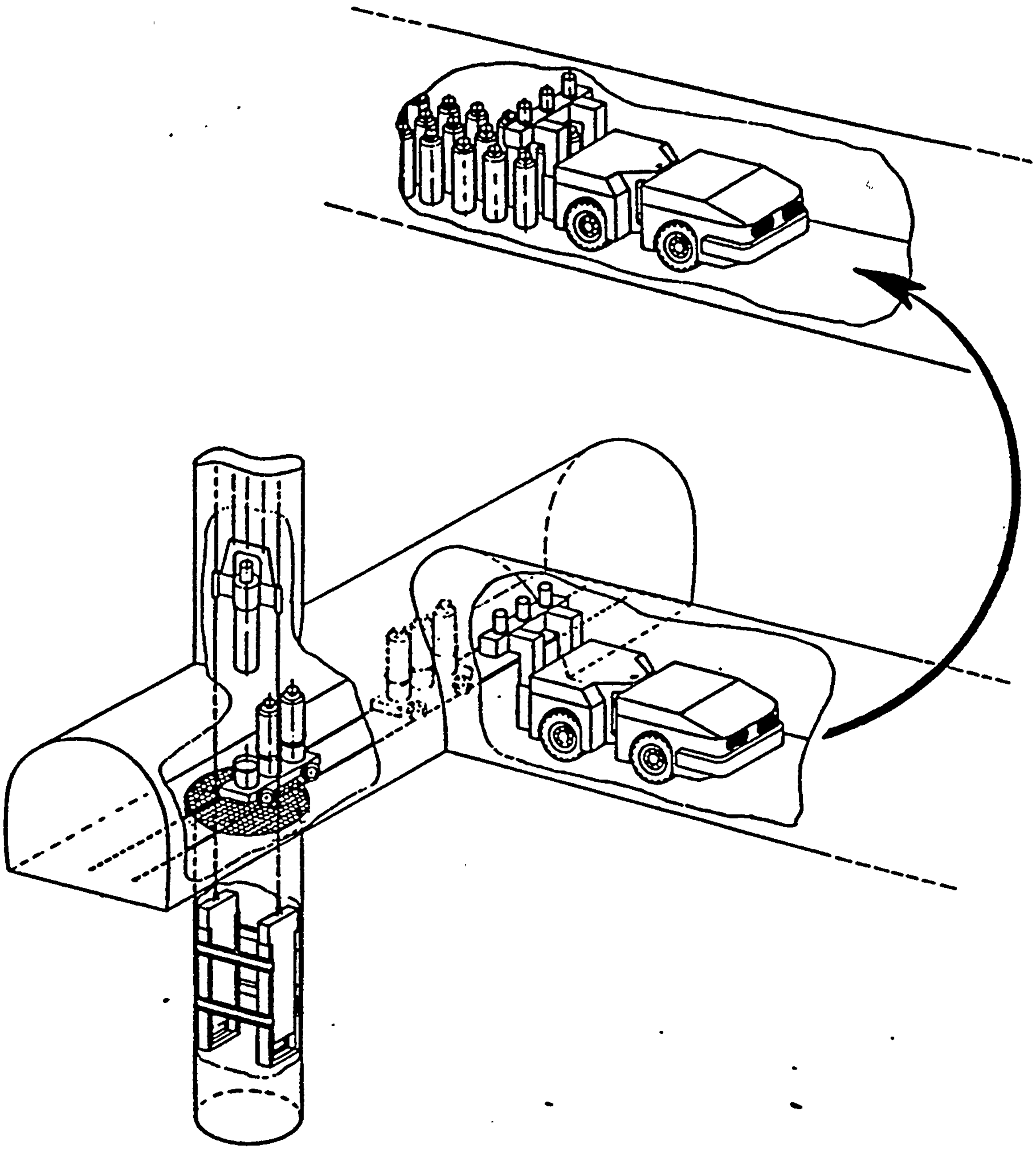

Figure 6.6.9. Unshielded Handling of High-Level Wastes in Galleries

6.6.6 Buffer and Sealing Materials

The backfill and sealing materials have the function, apart from conduction of heat away from the waste, to act as barriers against the intrusion of ground water, retention of the radionuclides and control the physio-chemical environment. In the granite repositories, large openings remain to be filled after disposal is complete and it is vital that these are fully backfilled to achieve a tight contact between the rock and the backfill material. Unlike salt, granite converges very little to compensate for inadequacies in the compaction of the backfill material. It is also almost impossible to guarantee a complete backfill in the granite due to the inherent fractures in the rock mass and therefore the addition of a swelling material such as bentonite clay to the backfill will be required. This will also effect some sealing of the fractures and reduce losses of backfill material due to washing away. Clay materials as well as waterproof concretes are thus being investigated as backfill and sealing materials.

Backfill placing techniques being investigated include:

- Pneumatic stowing
- Placement of prefabricated blocks
- Pumping
- Gravity feeding for silos and boreholes.

The techniques used will be determined mainly by the configuration of the openings to be backfilled and the radiation environment in the disposal area.

6.6.7 Summary

6.6.7.1 Advantages

1. The long interim storage requirement means less heat and radioactivity output of the waste in the repository.

2. Combining the interim storage facility with the repository by adding cooling to the emplaced waste packages has economic and operational advantages and may remove the need for a large surface storage facility. Also, closer spacing of the waste packages is possible.

3. Reprocessing of the spent fuel reduces the size of the repository required.

6.6.7.2 Potential Problems

1. Monitoring of the wastes and operation of the repository will be required for a long period of time.

2. High costs are associated with long-term storage and monitoring.

6.6.7.3 Summary

The French nuclear program is one of the most advanced in the world and will thus very likely also produce many advancements in the field of nuclear waste disposal. Several interesting concepts are being considered in France such as long-term, ventilated storage of the waste packages in underground facilities before final sealing.

6.6.8 <u>The Low- and Intermediate-Level Concepts</u>

The basic principles governing the disposal of low- and intermediate-level wastes in France is that confinement and isolation of the radioactive materials must be guaranteed for a long period of time (200 to 300 years). This highlights three stages in the life of a disposal facility for the low- and intermediate-level wastes:

- Operational phase during which the wastes are emplaced and the isolation barriers installed

- Surveillance phase after the operational phase during which the site is monitored and access is restricted (about 200 to 300 years)

- Harmless phase during which no surveillance takes place and access to the site may be restricted (time limits defined by activity and volume of wastes disposed of at the site).

In order to meet these criteria, a facility was constructed and is currently operating as part of the French nuclear program.

The low- and intermediate-level wastes are the only ones for which final disposal has been authorized in France to date. These wastes have been disposed of since 1969 in the near-surface repository in the Centre de la Manche near Cherbourg, Normandie. Depending on their form of conditioning and nature, two basic methods are used at the repository to dispose of the wastes:

1. Higher activity wastes are disposed of in concrete trenches which are filled up with cement and covered with bitumen.

2. Lower activity wastes are disposed of as a tumulus on a prepared ground area and covered with clay.

Figure 6.6.10 shows the disposal methods being used at the La Manche repository. The capacity of the repository is about 400,000 m^3 (523,000 yd^3) and is expected to be full by about 1990. Studies have been conducted of the last few years to find an alternative site. In 1985, a site, underlain by impermeable clay, was chosen at Soulaines-Dhuys (east of Troyes). Construction of the repository which will be similar to the La Manche facility, was planned to start in 1987 and disposal operations are expected to start in 1990. The new repository is estimated to cost approximately $36,000,000 (Nagra Aktuell, April 1987) and have a capacity of 1,000,000 m^3 (1,310,000 yd^3) of wastes.

6.7 CRYSTALLINE ROCKS: THE JAPANESE CONCEPT

This chapter summarizes the radioactive waste disposal program in Japan. Japan has an extensive nuclear power program with 33 working rectors and more under construction. By the year 2000, approximately 12,400 MT (13,640 tons) of spent fuel will have accumulated and will require disposal. The Japan Atomic Energy Commission established the following policy in 1976 concerning the management of high-level wastes:

1. High-level wastes will be solidified into a stable form and disposed of after an appropriate interim storage period.

2. Reprocessing industries shall be responsible for the treatment (solidification and directly associated storage) of high-level wastes using techniques demonstrated by the government.

3. The government shall be responsible for the permanent disposal, the costs being borne on the "polluter-pays" principle.

Extensive research and development programs as well as safety analyses are now underway to investigate all aspects of the high-level waste disposal cycle. Trial disposal of high-level waste packages is expected to start in 2015 by which time disposal methods will have to be demonstrated in an underground laboratory using both simulated and actual waste packages. Table 6.7.1 shows the schedule of activities of the Japanese high-level waste disposal program and Figure 6.7.1 shows the phases of the program.

6.7.1 Geographic Location

In Phase 1 (Figure 6.7.1), different geologic media in Japan were classified according to hardness, fracture characteristics and rock mass formation, and their siting possibilities assessed. Formations under study were mainly crystalline and sedimentary rocks which are widely spread throughout Japan (Figure 6.7.2).

6.7.2 Summary of Geologic Setting

A great variety of rocks exist in Japan and can be divided into two rough groups: igneous and metamorphic (hard) rocks, and sedimentary (soft to

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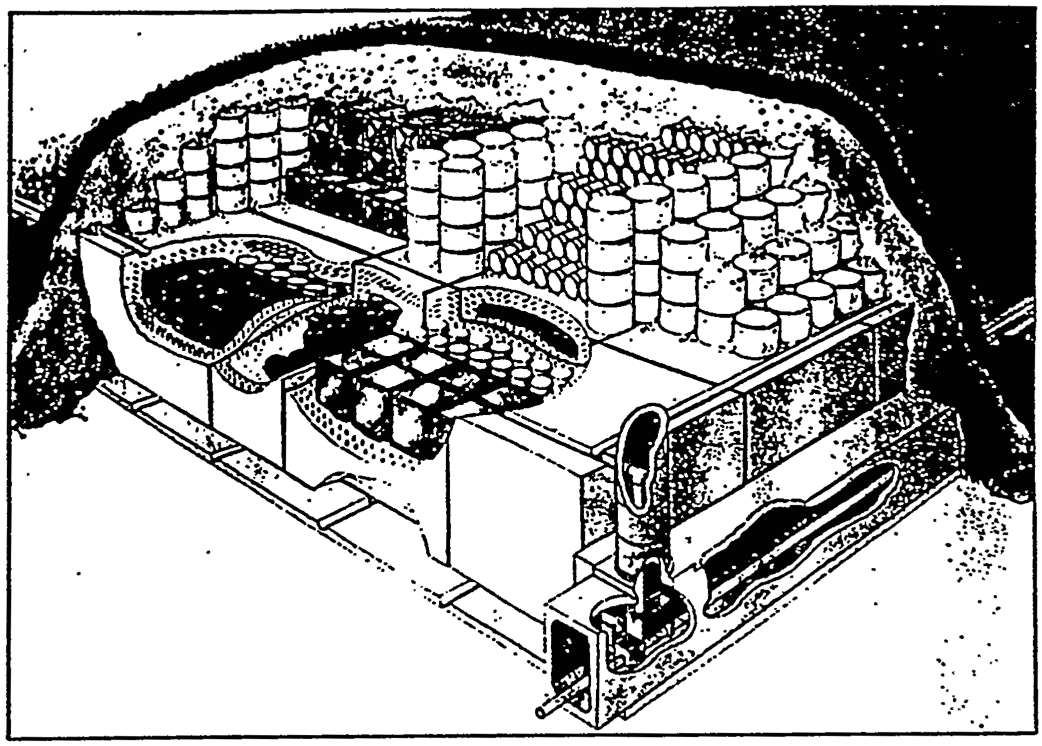

Figure 6.6.1Ø. Low- and Intermediate-Level Waste Repository (La Manche)

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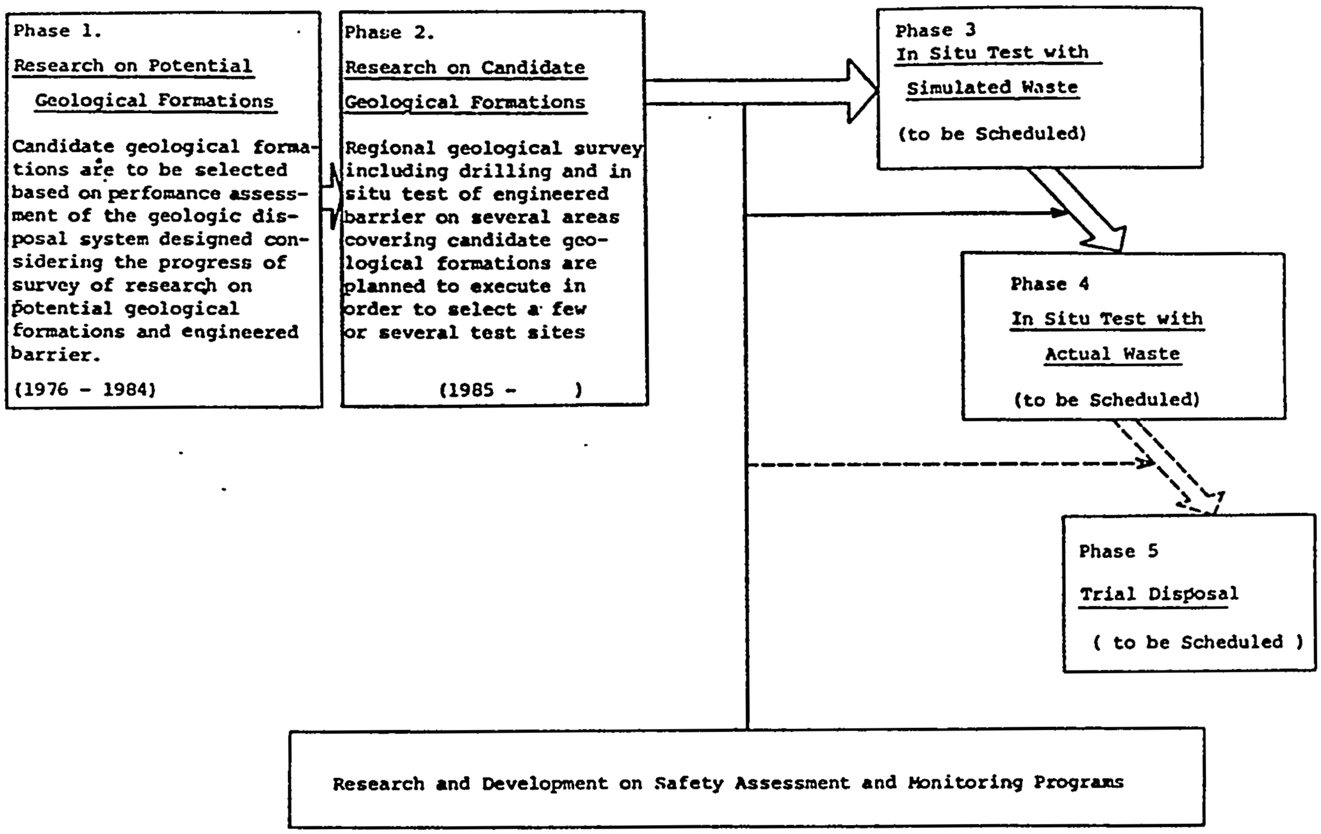

Figure 6.7.1. Schedule of High-level Waste Disposal Research Program

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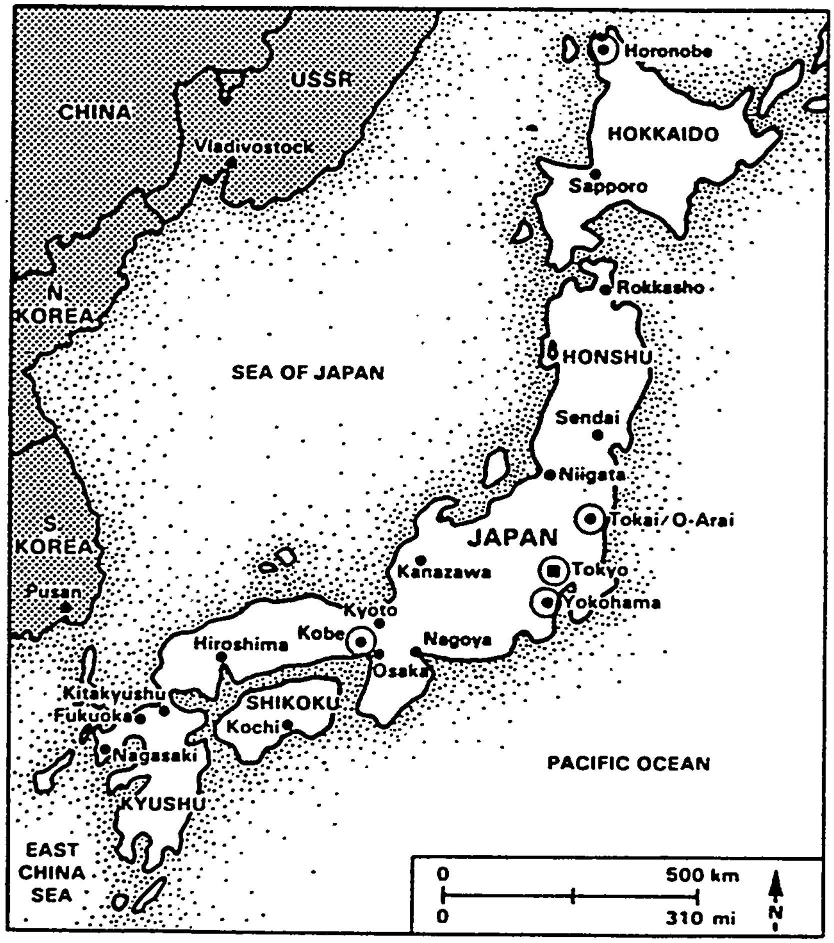

Figure 6.7.2. The Japanese Islands

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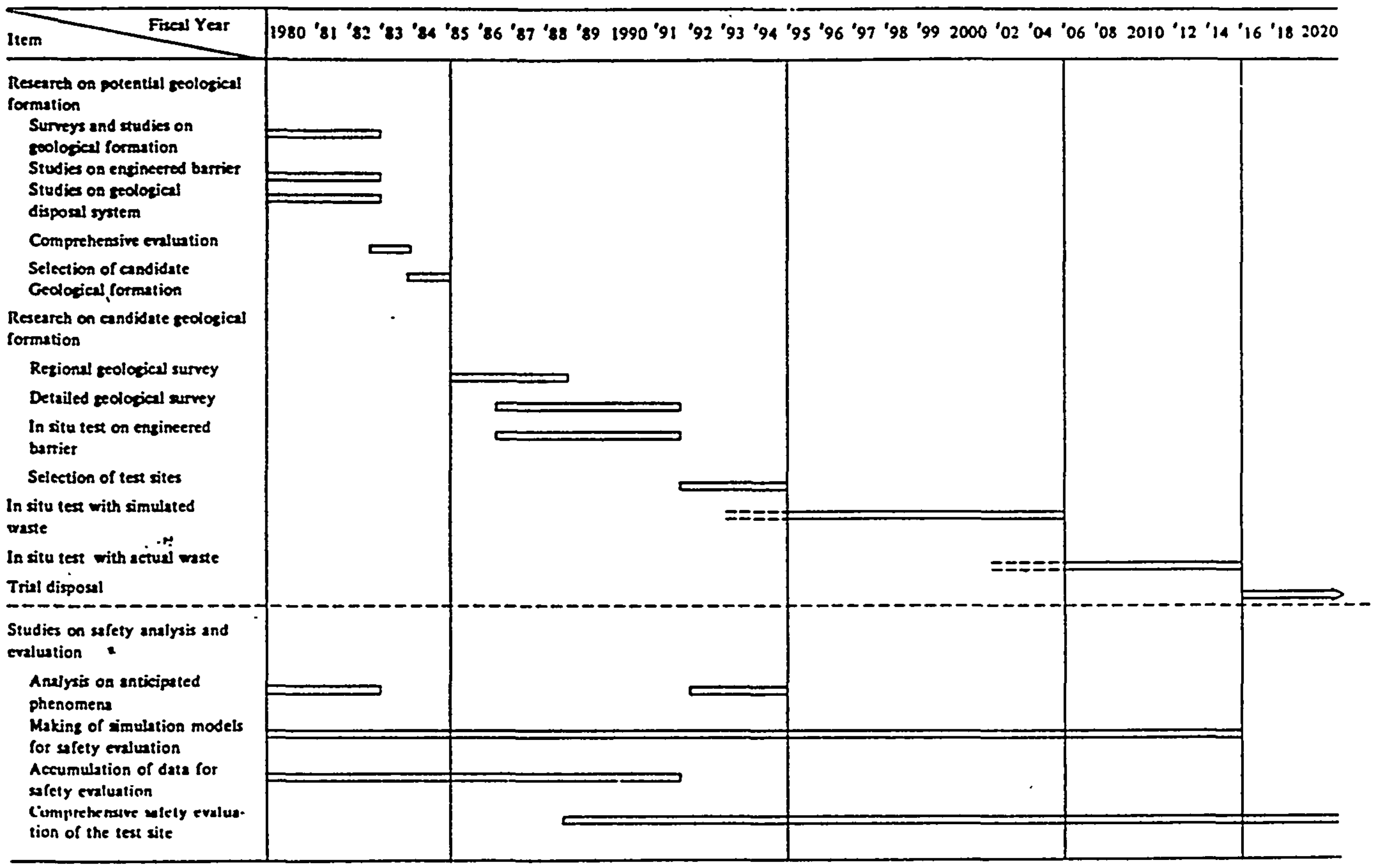

Table 6.7.1. Schedule of HLW Geological Disposal

medium-hard) rocks. Formations under investigation included granite, diabase, tuff, metamorphic rocks and sedimentary rocks.

Granite is widely distributed and represents the most common rock in Japan. The rock masses tend to be relatively simple and homogeneous.

Diabase rocks under study include diabasic intrusive rocks and basaltic andesitic effused rocks. They occur in slate, phyllite and highly graded metamorphic rock.

Tuffs are widely distributed and consist of tuff breccia, lappilli tuff and fine tuff.

Limestones are the sedimentary rocks also included in the studies.

An underground laboratory is planned to start operation in Honorobe in 1992.

6.7.3 Repository Concept Description

6.7.3.1 Nuclear Waste Disposal Strategy

The basis for the Japanese high-level waste disposal program is the reprocessing of the spent reactor fuel. The basic strategy, based on the policies described previously, for the disposal of high-level waste is as follows:

- Storage of the high-level wastes until radioactivity and heat have decayed enough to allow solidification
- Solidification of the high-level wastes
- Storage of the solidified high-level wastes until they have cooled enough to allow final disposal
- Final disposal of the high-level wastes.

Disposal in a deep geologic repository was chosen as the preferred method for the long-term isolation of the high-level wastes. The wastes will be stored for a minimum of 30 years in an interim storage facility to reduce the heat and radioactivity output.

A small scale reprocessing plant has been operating at Tokai since 1977 and approximately 350 - 400 MT (385 - 440 tons) of spent fuel have been reprocessed. The high-level wastes produced are currently being stored in tanks at the Tokai plant. These wastes are to be solidified and disposed of after an interim storage period. The remaining spent fuel is being reprocessed and vitrified in France and the United Kingdom and will presumably be returned to Japan for disposal. A large capacity [about 1,200 MT (1,320 tons) per year] reprocessing plant is planned to start operation around 1990. Research and development work has also been ongoing into the solidification process. Vitrification in borosilicate glass is the method preferred and a pilot vitrification plant is planned to start operation in 1991. Other investigations are also being conducted into:

- Development of waste canister configurations
- Development of interim storage methods
- Repository site evaluation
- Performance assessment and safety studies
- Repository construction methods
- Repository configurations
- Waste package handling methods
- Waste emplacement methods
- Ground-water movement within the rock mass
- Engineered barriers
- Repository backfilling and sealing methods.

In situ investigations will be conducted in underground facilities using simulated and actual waste packages.

6.7.3.2 Surface Facilities

Surface facilities other than the reprocessing and vitrification facilities will include:

- Interim storage facility for long-term monitored storage of the vitrified waste canisters before they are placed in the underground repository
- Reception facilities at the repository for the unloading, inspection and preparation for transfer underground of the waste packages
- Facilities required for the normal operation of underground mines such as hoists, headframes, services, administration buildings, security, emergency services, etc.
- Facilities for the preparation of the backfill and sealing materials.

6.7.3.3 Underground Configuration

The long-term performance in a geologic disposal system relies on the natural barriers and thus it is desirable to bury the high-level wastes as deep as possible. Experiences in the Japanese mining industry have shown that ventilation of repositories at depths up to 1,000 m (3,280 ft) is possible. The optimum repository depth will depend on the presence of disturbed zones in the rock mass and will be determined by safety assessments based on the mechanical properties of the rock and by evaluating the isolation performance of the rock as a natural barrier. For the repository concept described in this chapter it was assumed that the depth is around 1,000 m (3,280 ft).

Figure 6.7.3 shows the proposed repository configuration in the hard rock formations such as granite. Four shafts connect the surface facilities to the service area of the repository. The shaft will function as waste transport shaft, personnel and material transport shaft, ventilation exhaust shaft and emergency exit shaft. From the shaft bottom service area, a series of main access headings will be excavated along the center and around the repository. The disposal drifts will be excavated between these. The studies showed that in granite rock support would probably not be required in the drifts. The repository layout was based on the following considerations:

- Allow adequate ventilation
- Allow good drainage
- Allow supply of uncontaminated air to areas where personnel is handling waste packages
- Allow backfilling of disposal drifts in stages as waste emplacement proceeds
- Flexibility in layout to avoid fracture zones
- Keep repository area as small as possible
- Vitrified wastes are stored for a minimum of 30 years (alternatively, waste may be stored for 100 years in a repository before final disposal)
- Minimum capacity is for 10,000 canisters with expansion capability to 20,000 to 40,000.

Dimensions of the shafts and main access drifts were not given but details of the disposal drifts are given in Figure 6.7.4. The drifts will be roughly 3.5 m wide by 3.5 m high (11.5 ft by 11.5 ft) and the disposal holes will be 4 m (13 ft) deep with a diameter of 0.8 m (2.6 ft). The disposal holes will be drilled into the floor of the drifts and one waste canister will be placed in each borehole. Thermal analysis showed that the distance between boreholes for 30 year old waste would have to be about 8 m

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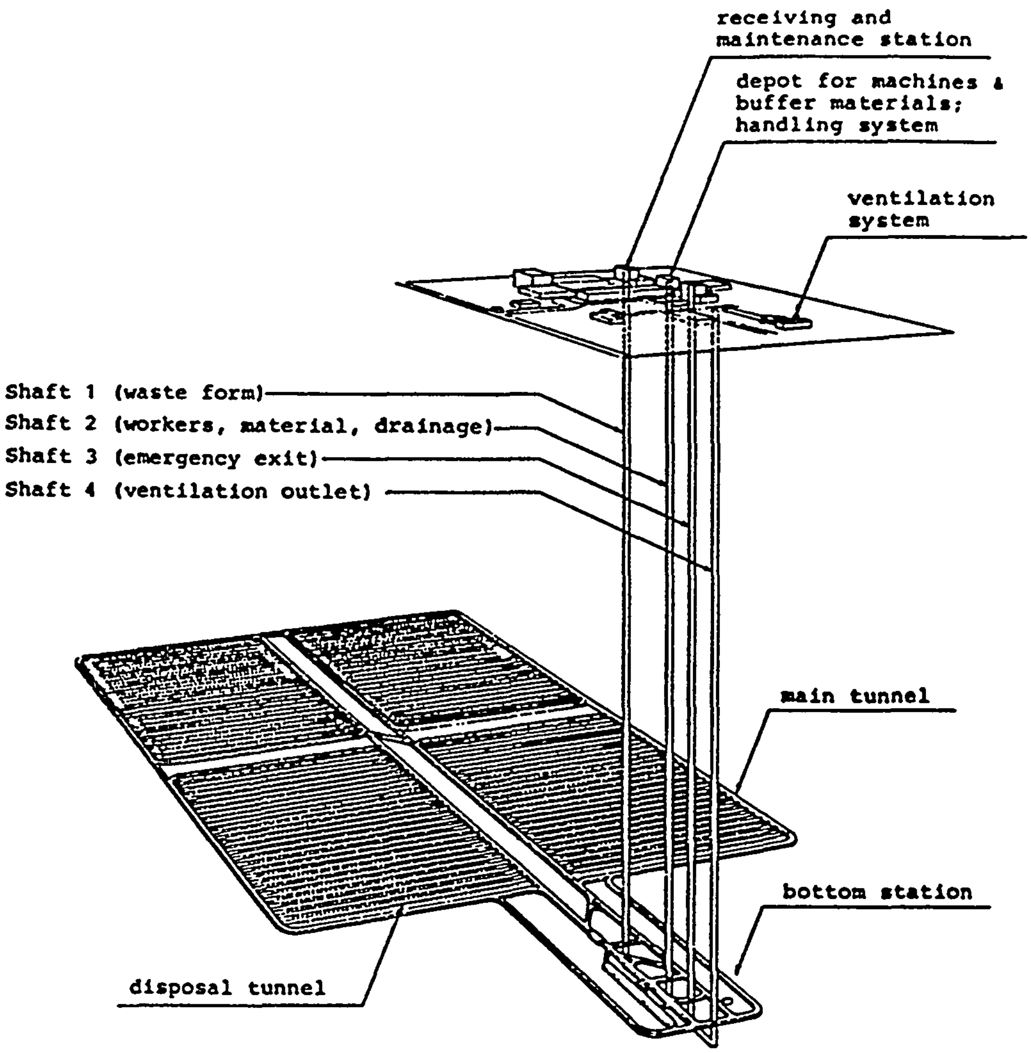

Figure 6.7.3. Conceptual Repository Layout

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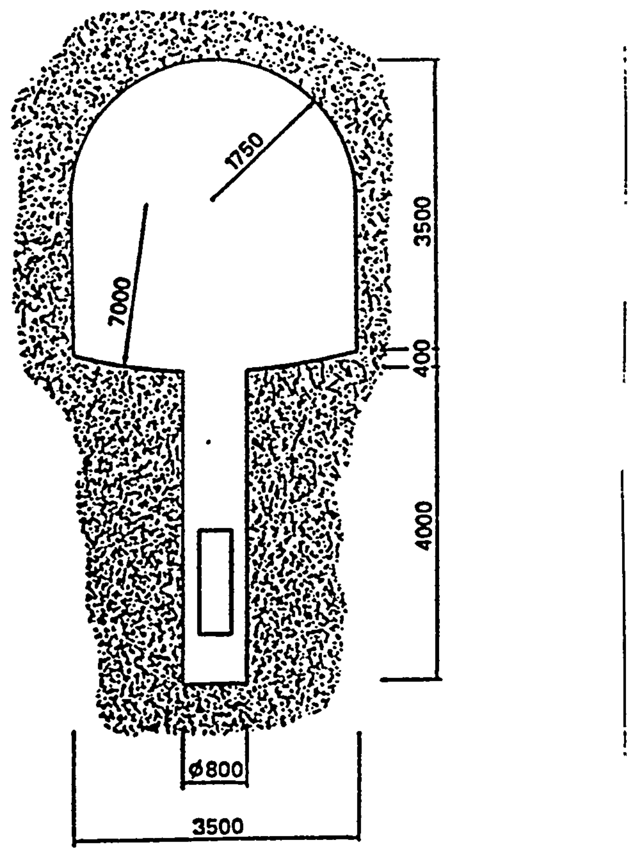

Figure 6.7.4. Disposal Drift and Borehole Configuration

(26 ft). Assuming a repository capacity of 10,000 canisters, a total of about 80,000 m (260,000 ft) of disposal tunnels would be needed. The spacing between tunnels is calculated to be 10.5 m (35 ft). Assuming a disposal tunnel length of about 500 m (1,640 ft), a total of 160 disposal tunnels will be needed. With the layout shown in Figure 6.7.3, the overall repository dimensions would be approximately 1,000 m by 1,000 m (3,280 ft by 3,280 ft).

6.7.4 Repository Construction

6.7.4.1 Schedule

Detailed schedules for the construction and operation of the repository were not available but trial emplacement operations are currently planned to start in 2015 (Table 6.7.1). Assuming an emplacement rate of about 5-10 canisters per day or 250 to 500 per year, disposal operations can be expected last for about 20 to 40 years.

6.7.4.2 Methods and Equipment

Descriptions of the construction methods were not available in the literature. It can be assumed, however, that as in other countries programs, the techniques that will be used will have to ensure minimum damage to the surrounding rock mass and will represent the latest developments in economic and efficient excavation technology. The application of mechanical excavation techniques such as shaft boring, raise boring and tunnelling machines over drill-and-blast methods are preferred for obvious reasons. Extensive mining and civil construction experience exists in Japan and thus the resources needed for construction of the repository are available within the country.

6.7.5 Waste Emplacement Cycle

6.7.5.1 Waste Treatment

Current plans are to reprocess all spent fuel. Spent fuel is being reprocessed at the Tokai reprocessing plant and in France and the United Kingdom after about 3 years of cooling. The liquid high-level wastes are stored in tanks at the reprocessing facility and are to be solidified in borosilicate glass (vitrified). The vitrified wastes are then to be stored for a minimum of 30 years before they are disposed of in a geologic repository. Alternatively, the wastes may be stored and air cooled in a repository for about 100 years before final burial to reduce the size of the repository. Studies of the vitrification process and storage techniques are currently proceeding.

Research is also being conducted into volume reduction and immobilization techniques for the high-level solid wastes such as the spent fuel hulls and other hardware. Methods such as hot isostatic pressing are being investigated to improve storage efficiency and convert the waste to a suitable form for long-term storage. Similar research is also underway for the transuranic wastes as well as the low-level waste forms.

6.7.5.2 Waste Transport

As in other countries, all forms of waste transport are under investigation to find the optimum method for the existing conditions.

6.7.5.3 Waste Package

Specifications for the waste package have not been established yet but for the high-level wastes from reprocessing about 1 MT (1.1 tons) of spent fuel, a cylindrical, stainless steel canister containing 0.11 m^3 (0.15 ft^3) is envisioned. The diameter of the canister will be about 400 mm (16 in) and the height about 1.4 m (4.6 ft).

6.7.5.4 Underground Waste Emplacement

The waste packages will be placed into boreholes drilled into the floor of the disposal drifts. The borehole diameter will be about 0.8 m (2.6 ft) and the depth about 4 m (13 ft). Each borehole will contain one waste canister. The boreholes will be backfilled as they are filled and the drifts sealed once all boreholes have been filled. Disposal room excavation, waste emplacement and backfilling and sealing operations will probably be conducted concurrently. The technology for drilling of the boreholes, backfilling and sealing, and waste canister handling and shielding is still under investigation.

6.7.6 Buffer and Sealing Materials

The waste isolation system is based on the multiple barrier system and investigations are being conducted into backfilling and sealing materials. Grouting technology is being developed to repair and reinforce disturbed zones or fractures in the repository host rock to prevent ground-water seepage and stabilize the excavations. Experiments are being conducted in granite to test the effects of injecting super-fine cements. Bentonite-based mixtures are being researched as buffer materials around the waste packages.

6.7.7 Summary

6.7.7.1 Advantages

1. The repository will be more compact due to the volume reduction during reprocessing and long interim storage times.

2. The concept is based on the multiple barrier system.

3. Detailed, long-term research and demonstration programs are being conducted to study all aspect of the high-level waste disposal concept.

6.7.7.2 Potential Problems

Uncertainties about the long-term behavior of the repository system are inherent in all concepts but, like all other countries, Japan is conducting a comprehensive investigation program until well into the next century to demonstrate the feasibility of the repository concept and reduce the risk associated with underground high-level waste disposal.

6.7.8 The Low- and Intermediate-Level Concepts

The accumulated amount of low-level waste is estimated to be about 1.1 million 200 L (55 gal) drums by the year 1990. The low-level wastes are treated by using volume-reduction techniques and solidified. Disposal techniques being investigated in Japan are sea disposal and on-land repositories.

6.8 CRYSTALLINE ROCKS: THE BRITISH CONCEPT

There are 38 operating reactors in the United Kingdom producing roughly 20% of the nation's electricity. The high-level waste disposal strategy calls for the reprocessing of the spent fuel and final disposal in a deep, geologic repository after an interim storage period of at least 30 years. Britain currently has two operating reprocessing plant, one each at Dounray and Sellafield, which have been operating for about 30 years. Construction of an additional reprocessing facility is scheduled to be complete by 1990 which will provide the capability of reprocessing the uranium oxide fuel from British gas-cooled reactors as well as the fuel from European and Japanese light-water reactors. Plans also exist for the construction of a reprocessing facility for the spent fuel from fast-breeder reactors in Britain, Germany and France. Currently the high-level reprocessing wastes are stored in tanks at the reactor sites and reprocessing plants until a final disposal solution has been defined. All these wastes are to be eventually vitrified before disposal and small-scale vitrification plants have been in operation at Harwell. Another plant is under construction and is scheduled to be complete in 1989.

Although the storage of the high-level wastes in tanks is a safe method, it is not a long-term solution. Until about 3 years ago, Britain had an extensive research program investigating the disposal of high-level wastes. Potential repository sites for the deep, geologic disposal were identified in granite, some exploratory drilling was carried out near the Dounray, Scotland, research facility, and plans for site investigation activities, repository configuration studies, safety analyses, etc., had been prepared. It was then decided to store the high-level wastes for at least 50 years in an interim storage facility and all high-level repository siting activities suspended. The repository locations that had been identified before the suspension of activities were mainly in granite, although studies were also conducted into salt and clay formations. The preliminary disposal concept that had been derived as part of European Community study and was the disposal of the high-level waste canisters in boreholes drilled into the floors of disposal drifts. These drifts were to be at a depth up 1,000 m (3,280 ft) in a granite batholith. The disposal holes were to be about 100 m (330 ft) long and could accommodate 5 high-level waste canisters. Capacity of the repository was to be about 30,000 canisters, placed at the rate of about 1,000 canisters per year.

Britain is also investigating sub-seabed disposal. Until 1983, low- and intermediate-level wastes were dumped at sea and stored in near-surface repositories. Disposal in engineered trenches was used at the near-surface facilities. Investigations are currently underway to find additional near-surface repository sites for the low-and intermediate-level wastes. Clay sites are being studied at Elstow, near Bedford, Bradwell in Essex, Killingholme in South Humberside, and Fullbeck in Lincolnshire.

Figure 6.8.1 shows a summary of the disposal concepts being investigated for the various types of nuclear wastes.

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TABLE 1. National Disposal Facilities.

FACILITY	DESCRIPTION Approx. Depth	PROVISIONAL ACTIVITY LIMITS	WHEN AVAILABLE
Land 1	Shallow Burial at Drigg	α 20 mCi/m^3 β/γ 60 mCi/m^3 With other stipulations	Available
Land 2	Engineered Trench 20 - 30 metres	α 3 Ci/m^3 βγ Higher than Land 1	Late 1980's
Land 3	Mine or Cavity 100 metres	α 50 Ci/m^3 βγ Unrestricted	1990
Land 4	Repository 300 metres	αβγ Unrestricted Low heat output	1991
Land 5	Repository 300+ metres	αβγ Unrestricted Heat generating waste	After 2010
Sea 1	Ocean Disposal Conventional Packages	α 1 Ci/te β/γ 100 Ci/te With other stipulations	Now
Sea 2	Emplacement on/under Deep Ocean Bed	αβγ Unrestricted	After 2010

TABLE 2. Indicative Summary of UK Conditioned Waste Arisings to Year 2000.

WASTE CATEGORY	VOLUME OF CONDITIONED WASTE TO YEAR 2000 (m^3)	POSSIBLE DISPOSAL ROUTE (a)
High-level (heat generating)		
BNFL	900	Land 5
UKAEA	200	
Intermediate-level: αβγ (b)	26000	Land 3 or 4
Intermediate-level: βγ	64000	Land 2 and sea disposal
Low-level	520000	Land 1

Notes:- (a) As defined in Table 1.

(b) Includes low βγ, high α (TRU) Waste.

Figure 6.8.1. Nuclear Waste Disposal in the United Kingdom

7.0 SEDIMENTARY ROCK FORMATIONS

7.1 CLAY: THE BELGIAN DISPOSAL CONCEPT

The Belgian geological radioactive waste disposal program was started in the early 1970s after the first nuclear power plant went into operation. Following a general survey which highlighted the absence of suitable salt or crystalline formations at a reasonable depths in Belgium, it was decided to concentrate the repository research efforts on the clay and argillaceous rocks. In particular, a potential site was selected for further study at the nuclear facility in Mol in the northern part of Belgium. An underground research laboratory has been constructed at the site and investigations into the suitability of the clay formation (Boom clay) as a repository host rock are planned for well into the 1990s. If the site is found suitable, the research facility may be expanded into a full repository.

The Belgian nuclear waste program is based on the waste production from an installed capacity of about 10 GWe with a life time of 30 years. Table 7.1.1 shows an estimate of the production of high-level, intermediate-level, and alpha-bearing wastes. The amount of about 900 m^3 (1,180 yd^3) of high-level waste stems from the reprocessing of approximately 6,000 MT (6,600 tons) of spent fuel elements.

Table 7.1.1. Estimated Waste Quantities Arising From Belgian Nuclear Power Production (Bonne, et al., 1986)

	HLW	CLW[1]	ILW(S)[2]	Alpha-W
No. of Packages	6,000	4,500	30,000	30,000
Dimensions				
H (m)	1.3-1.5	1.5-1.7	0.9	1.1
Dia (m)	0.3-0.43	0.3-1.06	0.6	0.72
Waste Volume (m^3)	0.150	1.3	0.18	0.40
Total Volume (m^3)	900	5,850	5,400	12,000

1 Cladding Waste

2 Intermediate-level waste (Slurries)

The Belgian waste disposal strategy is basically as follows:

- Reprocessing of all its spent fuel elements in domestic or foreign reprocessing facilities

- Interim storage for about 50 years of the high-level wastes to allow the radiation and heat production to diminish

- Final disposal of the wastes in boreholes in an underground repository in clay.

Investigations of the Boom clay are to continue well into the 1990s with expansion of the current facility for large-scale testing of waste emplacement and repository construction techniques. The final high-level waste repository is expected to be operational during the first decades of the next century. This chapter describes the Belgian concept for the geologic disposal of high-level wastes in the Boom clay formations in more detail.

7.1.1 Geographic Location

The clay formations under the Mol nuclear site were chosen as the most promising setting for a geologic nuclear waste disposal program. The site is located in the northern part of Belgium. Figure 7.1.1 shows the approximate location.

7.1.2 Summary of Geologic Setting

The site selected for further study at Mol has an area of about 700 ha (1,730 acres). The Boom clay formation in this region is about 100 to 120 m (330 to 390 ft) thick and the thickness of the overburden is about 160 to 180 m (520 to 590 ft). The clay is a very compact plastic clay from the Oligocene period and is over- and underlain by a succession of sandy and clayey layers. Figure 7.1.2 shows a simplified profile of the geology at the Mol site. The formations above and below the Boom clay contain aquifers, some of which are used as drinking water sources. The clay is composed of clay minerals (illite, smectite and vermiculite), particles of iron oxide, quartz, some organic matter, pyrite and calcite. The formation is saturated and water content varies around 18%. The overall permeability for the Boom clay formation has been estimated at 10^{-10} m/s from studies conducted in the underground laboratory facility.

7.1.3 Repository Concept Description

This section outlines the reference geologic disposal concept that was developed for the high-level and alpha-containing wastes in the Boom clay formation.

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Figure 7.1.1. Location of Mol Site

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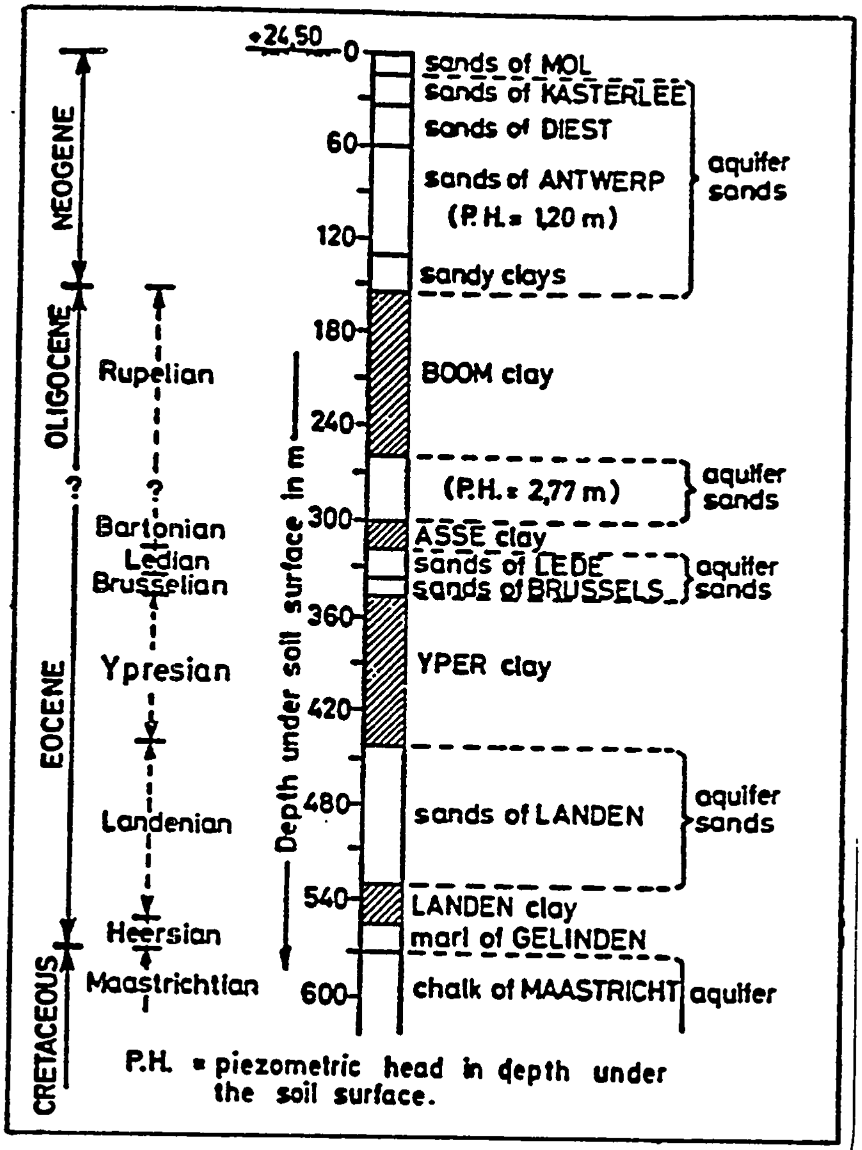

Figure 7.1.2. Stratigraphy at the Mol Site

7.1.3.1 Nuclear Waste Disposal Strategy

The Belgian nuclear waste disposal program is based on an assumed 10 GWe power production with a 30 year lifetime. The estimated waste quantities resulting from this have been summarized in Table 7.1.1. The program is based on the reprocessing of all spent fuel produced during the power production. Reprocessing is currently being done at the La Hague facility in France, although reopening of the closed-down Eurochemic reprocessing plant at Mol is under consideration. The vitrified high-level wastes and alpha-bearing wastes will be stored in an interim storage facility for about 50 years. The wastes will then be taken to an underground disposal facility for final deposition.

Direct disposal of spent fuel has not yet been considered as an option although it is recognized that despite reprocessing of spent fuel at domestic or foreign facilities, some non-reprocessed fuel may also require disposal.

7.1.3.2 Surface Facilities

The surface facilities for the management of the high-level wastes will consist of the interim storage facility and the facilities at the repository for receiving and preparing the waste packages for underground disposal.

For interim storage, two concepts have been studied. One concept proposed storing the vitrified high-level waste containers in water-filled pools and the other proposed storage in air-cooled cells.

The surface facilities at the repository will have to include a reception building to receive the waste packages and prepare them for transport underground. Other installations will include the shaft headframes, hoists, ventilation, and underground utilities such as compressed air, water, power, etc. Ancillary installations will include offices, emergency and security facilities, excavated spoil handling and concrete/backfill storage and preparation areas.

7.1.3.3 Underground Configuration

Figures 7.1.3 and 7.1.4 show the reference concept for the underground disposal of the high-level and alpha-containing wastes in the Boom clay formation. The underground facility will be located about 220 m (720 ft) below surface. Three or four shafts will connect the underground excavations with the surface. Two shafts will function as main access to the repository for the waste packages and men and materials as well as acting as ventilation intake shafts. The finished diameter of these shafts will need to be about 4.5 m (15 ft). One or more smaller diameter shafts (2.3 m or 7.5 ft) will function as ventilation exhaust shafts and emergency egress. Because of the limited thickness of the clay formation, the repository will most likely be excavated on one level only. A series of parallel disposal galleries or headings will be excavated off the main

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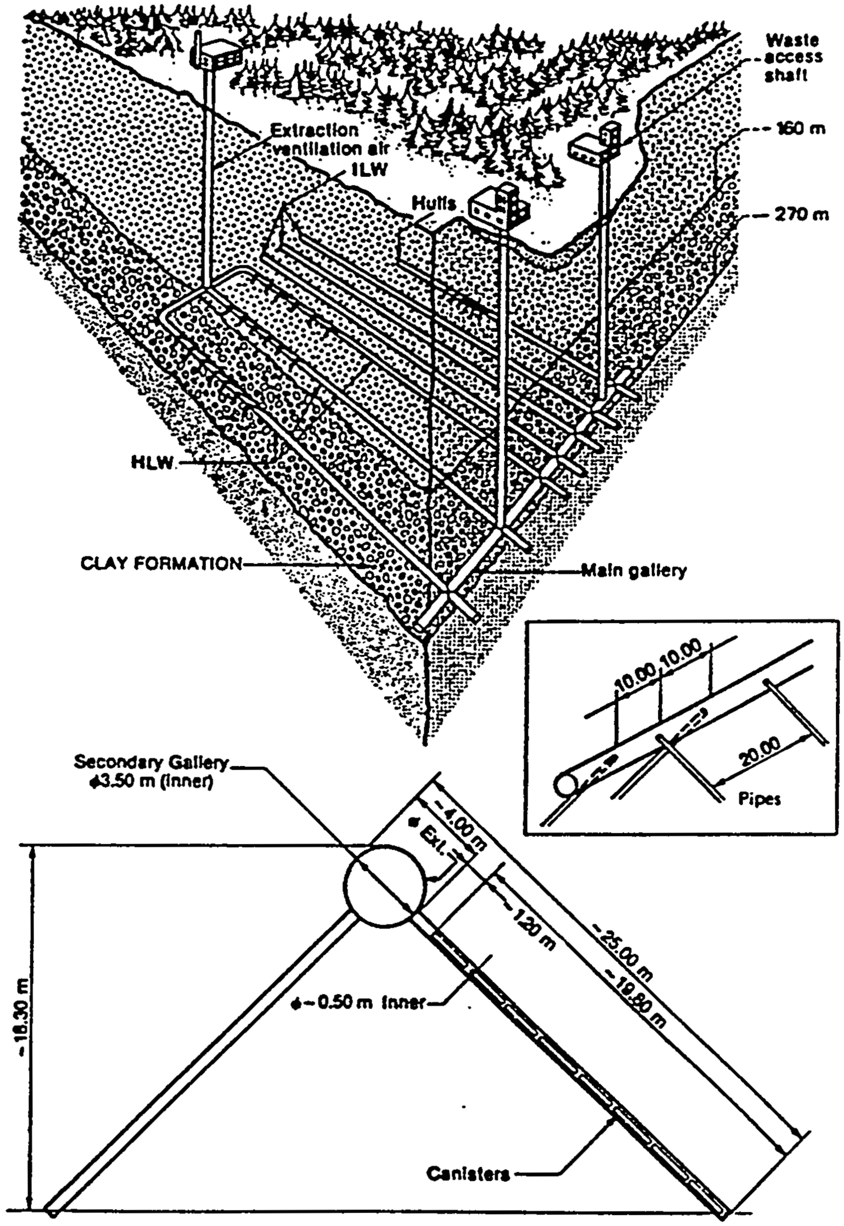

Figure 7.1.3. Reference Disposal Concept

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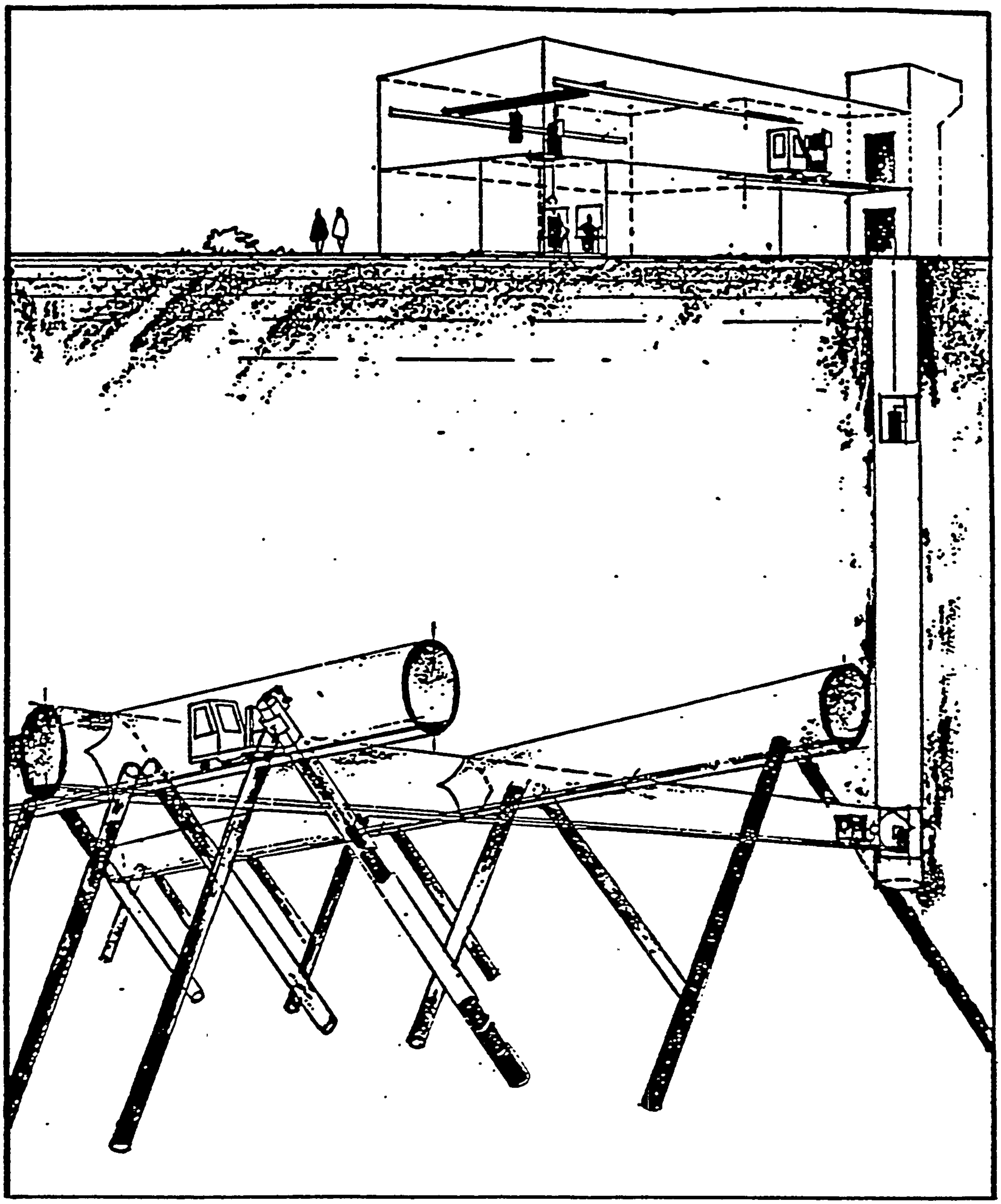

Figure 7.1.4. High-Level Waste Disposal Concept

connecting drift between the two access shafts (Figure 7.1.3). The 1,250 m (4,100 ft) long disposal galleries will have a finished diameter of about 3.5 m (11.5 ft) and the main heading will have a diameter of about 4.5 m (15 ft). The three types of nuclear waste (high-level, fuel cladding, and intermediate and alpha-containing wastes) will be disposed of in separate galleries. The distance between the high-level waste galleries will be about 200 m (660 ft) and the separation of the entries for the other types of wastes will be from 35 to 45 m (115 to 150 ft). Canisters with the high-level wastes will be stacked in inclined (approximately 45°) boreholes drilled from the disposal galleries. The approximately 0.5 m (1.6 ft) diameter boreholes will be spaced at 10 m (33 ft) intervals and have a length of 15 to 20 m (50 to 65 ft). The canisters containing the fuel cladding will be placed in similar boreholes while the intermediate-level waste canister will be stacked in the appropriate galleries.

Other concepts are also under investigation. These include stacking the high-level and intermediate-level wastes in horizontal entries rather than inclined boreholes.

The shafts and underground openings will probably all be circular due to the pressure exerted on the plastic clay and all will require some type of support. The linings under investigation include precast concrete segments, cast-in-place concrete and bolted cast-iron segments. One of the purposes of the underground research facility at Mol is to allow installation and testing of various linings and to study various excavation techniques.

The areal extent of the repository will be roughly 2 to 2.5 km^2 (500 to 620 acres).

7.1.4 Repository Construction

Due to the plastic nature of the clay, special excavation and lining techniques will have to be used for the construction of the repository. This section outlines some of the technology under investigation.

7.1.4.1 Schedule

The operation of the repository is envisioned to start between 2005 and 2025. Before that, extensive in situ testing at the Mol underground laboratory will take place. Plans are to expand the current facility to include larger galleries excavated using tunneling machines and construction of another shaft as part of a program to demonstrate the feasibility of constructing large underground openings in the clay formation. This expansion of the test facility is scheduled to occur in the 1990s.

7.1.4.2 Methods and Equipment

The following describes some of the techniques that may be used to excavate the repository. Special techniques and materials will be required due to the plastic behavior of the clay material. Since knowledge about the construction in this type of material is relatively limited, the technology will be tested and investigated in the underground laboratory before repository construction begins.

For the construction of the shafts, the freezing technique may be applicable. The overlying potable aquifers as well as the clay formation will have to be isolated and protected during construction and operation of the repository and, therefore, sinking using the freezing technique with the installation of a water-tight lining is a possible solution. Another shaft sinking method would be the large hole drilling technique where drilling would take place while the shaft is filled with a bentonite fluid which supports the shaft wall and prevents water inflows during drilling. A prefabricated steel liner would then be cemented into the hole to leave a shaft with a water-tight lining.

The main heading connecting the two access shafts may be mechanically excavated with the freezing technique for temporary ground support. The lining will likely consist of a prefabricated lining such as cast-iron or precast concrete segments. These prefabricated segments are bolted into rings in the tunnel and the joints sealed to form a water-tight lining.

The long disposal galleries may be excavated using a soft-ground tunneling machines or shields. If required, machines which can also support the tunnel face during excavation using bentonite slurry may also be used. The segmental cast-iron or concrete lining is installed right behind the machine as it advances.

One proposal for the excavation of the disposal holes is to use a form of pipe-jacking with simultaneous drilling at the bottom of the steel pipe column as it is forced into the borehole.

7.1.4.3 Cost

A first cost estimate for the construction and operation of this type of repository was made in 1978 and an approximate cost of 130 million Belgian Francs ($5 million) per GW year of nuclear electrical power was derived.

7.1.5 Waste Emplacement Cycle

This section outlines briefly the waste emplacement concept that will probably be used. Information is limited at this point since investigations are still ongoing and the disposal concept is always subject to revision.

7.1.5.1 Waste Treatment

The Belgian disposal concept is based on the reprocessing of all spent fuel produced. Quantities and dimensions are given in Table 7.1.1. Three types of wastes are being considered for disposal in the clay formation:

- Vitrified high-level wastes
- Fuel cladding (hulls) wastes
- Intermediate-level and alpha-containing wastes.

7.1.5.2 Waste Transport

Transport of the wastes will be by truck or rail in specially designed transport containers similar to those described for other disposal concepts.

7.1.5.3 Waste Package

The use of containers for individual canisters of waste is not planned. Instead, canisters of vitrified HLW are to be emplaced in inclined boreholes lined with a permanent steel casing that radiate out from the emplacement tunnel. Canisters 1.335 m (4.4 ft) long and 0.43 m (1.4 ft) in diameter, would be emplaced in each borehole liner, and the space backfilled with uniformly graded sand. The arrangement is illustrated in Figure 7.1.3.

For the purpose of radioactive waste disposal, clay formations have the desirable characteristics of high sorption capacity for radionuclides, and a very low permeability. Reliance is placed on these characteristics, in the Belgian design, to assure required isolation of the waste.

Clay ground water analyses indicate an undisturbed alkaline, reducing chemistry. However, there is a potential for development of locally acidic and oxidizing conditions due to the natural presence of pyrite and organic material in the clay and exposure to air during repository construction and operation. The oxidation of these substances under very near-field temperatures also may result in the formation of corrosive sulfurous oxides and a lowering of pH.

Consequently, corrosion resistant canister materials are being evaluated for potential use in the clay environment with titanium and nickel-base alloys the most promising performers. However, while canister materials are being investigated for corrosion resistance in the clay environment, an extended period of containment is not critical to the isolation concept.

In order to preserve the desirable characteristics of the clay, the maximum permissible temperature rise in the clay is 100°C. This limitation, along with the low thermal conductivity of clay (1.67 W/m-K) and an ambient temperature at repository depth of 16°C (60°F), restricts the repository thermal loading to 2.4 W/m^2 with waste that is aged 50 to 75 yrs prior to disposal.

The maximum projected lithostatic pressure at repository depth is 4 MPa (580 psi), although experimental evidence suggests that actual pressure on the borehole and tunnel liners is substantially lower. The borehole steel casing is designed to resist the overburden pressure, and the sand, within the casing, to transport pressure uniformly to the waste canisters.

7.1.5.4 Underground Waste Emplacement Sequence

Several disposal concepts are still under investigation but the reference concept envisions placing several high-level waste canisters in inclined boreholes. The canisters would thus have to be transported in the shaft and underground in shielded containers. Transport from the shaft to the borehole would require a special vehicle which is capable of receiving the canisters at the shaft, transporting them to the borehole and lowering them into their final location. Each borehole will be backfilled with a suitable material such as concrete, sand or clay. The fuel cladding wastes may be handled in a similar manner to the high-level wastes or, alternately, be stacked in disposal galleries as is envisioned for the intermediate-level waste canisters. Backfill around the canisters and drums stacked in the headings may be concrete or clay, injected as the heading is filled with waste. When the repository is full, the remaining excavations and shafts will be backfilled with clay excavated during the construction of the repository.

7.1.5.5 Emplacement Schedule

Repository operation is scheduled to start sometime between 2005 and 2025, and operations are likely to last about 50 to 60 years including the interim storage period.

7.1.6 Buffer and Sealing Materials

Backfill and sealing materials are still under investigation in the laboratory and at the Mol underground experimental facility. Suggested materials include concrete backfill for the disposal headings and boreholes, and the excavated clay material for backfilling and sealing of the remaining underground service excavations and shafts.

7.1.7 Summary

7.1.7.1 Advantages

1. The chemical properties of the clay make it a suitable medium for the long-term containment and isolation of radioactive wastes.

2. The clay is located at a relatively shallow depth which is an advantage from the construction and operation viewpoint.

3. No blasting should be necessary for excavation of the repository so that rock damage is limited. However, the plastic properties of the clay material present other technical difficulties as far as the stability of the openings during construction and operation is concerned.

7.1.7.2 Potential Problems

1. Experience with construction of openings in non-frozen Boom Clay is still limited to a small diameter shaft and drift. The technical and economical feasibility of construction of a full-size repository in the clay, therefore, require demonstration on a large scale.

2. All openings in Boom Clay are likely to require substantial linings to prevent premature closure due to the plastic nature of the clay.

3. Excavation of the shafts will require sophisticated ground control measures such as ground freezing and installation of a watertight lining.

4. The repository will be located between water-bearing formations.

5. The low thermal conductivity of the clay limits the accumulation of high-level wastes in the repository. This problem can, however, be overcome by sufficient aging of the wastes before disposal to limit the temperatures in the repository to a maximum of 100°C (212°F).

6. Construction of the repository in the clay will require special excavation and lining techniques to ensure that the openings remain stable during the life of the repository.

7.1.7.3 Summary

Studies of the Boom Clay to date have shown that despite some of the technical problems associated with the construction of the repository, the clay will provide adequate isolation of the buried waste. Due to the large size of the U.S. repository, it seems unlikely that this technique could be adapted for the U.S. repository due to the construction difficulties and availability of the massive clay formation needed.

7.1.8 The Low- and Intermediate-Level Waste Concepts

Until it was stopped in 1982, Belgium has been dumping its low-level wastes at sea, but since then has been investigating the possibility of placing the waste packages in engineered trenches and near-surface, underground caverns.

7.2 CLAY: THE ITALIAN CONCEPT

This chapter describes the high-level nuclear waste disposal options under consideration in Italy. Three reactors are currently operating and approximately 2,000 MT (2,200 tons) spent fuel are expected to accumulate by the year 2000. Current plans for the management of the spent fuel involve reprocessing and vitrification under contract with the United Kingdom, storage of the returned wastes in an interim storage facility in Italy, and final disposal in a repository in a suitable clay formation on the Italian mainland or in Sicily. Although other formations such as salt may be suitable, the main focus of the current studies is on the clays which are present in considerable thicknesses throughout Italy.

Two basic repository concepts were assessed in a feasibility study as part of the European Community Cooperation research program. These concepts were disposal in a mined repository, and in deep boreholes drilled from the surface (Figure 7.2.1). This chapter outlines the concepts that were presented in this feasibility study.

7.2.1 Geographic Location

No potential sites have as yet been identified, but in general the deeper and broader clay formations are in basins located on the eastern side of Italy and in Sicily, while smaller basinal structures occur in the western part of Italy (Figure 7.2.2). All areas are thought to contain adequate clay deposits, suitable for a potential repository.

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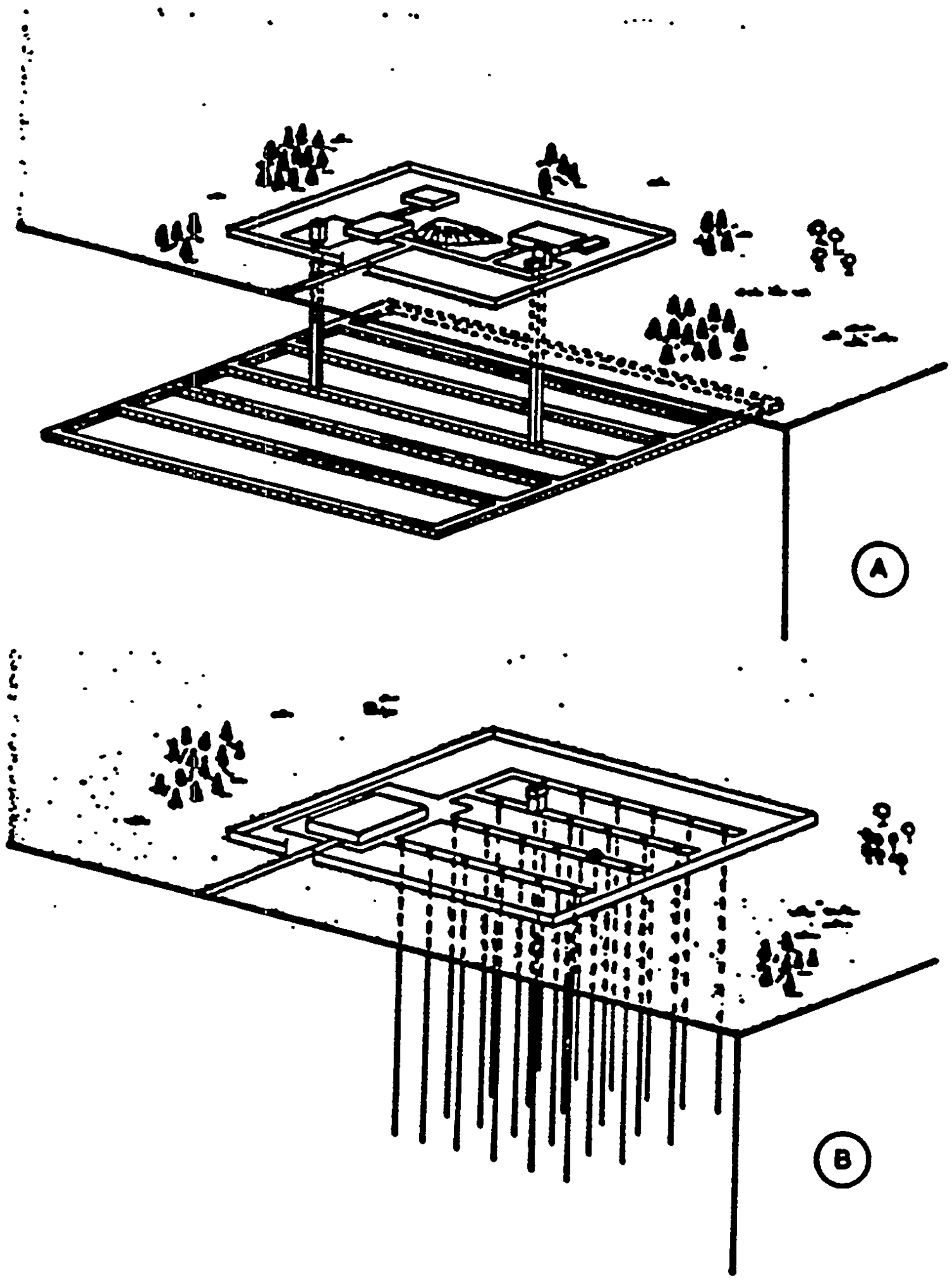

Schematic illustration of the two types of disposal facility considered in this report: (A) mined repository (B) deep borehole facility - DBF.

Figure 7.2.1. Basic Repository Configurations

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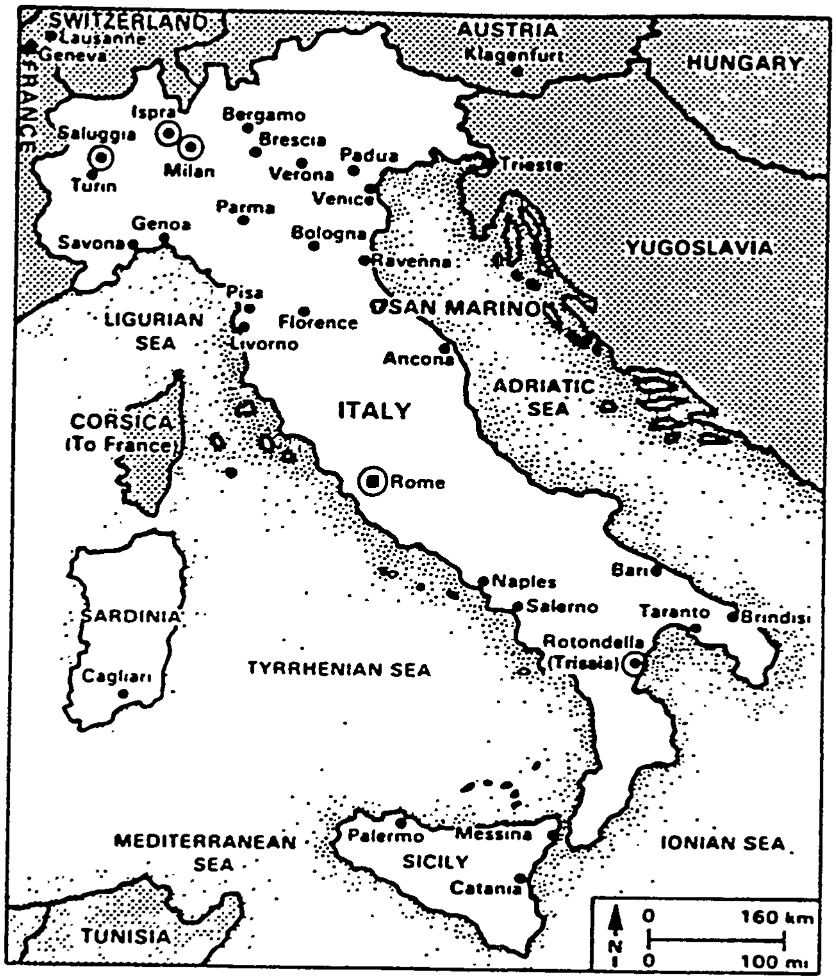

Figure 7.2.2. Map of Italy

7.2.2 Summary of Geologic Setting

The Italian clays have all the well known waste containment advantages of an almost impermeable semi-plastic medium, and consequently the associated equally well known engineering problems of difficult and costly tunnel construction. Only Belgium and the United Kingdom have considered nuclear waste disposal in clays apart from Italy. Belgium is well advanced with its studies in the underground laboratory at Mol while the United Kingdom has basically suspended its investigations. Only limited knowledge exists regarding underground construction in clays at depth of several hundred meters due to the lack of major construction projects in clays.

For the study of the reference repository concepts summarized in this chapter, a generic environment was selected in which a sequence of 150 to 400 m (490 to 1,310 ft) homogenous clay is assumed to be overlain by a between 100 and 300 m (330 to 980 ft) of poorly consolidated aquiferous sands and underlain by a thick limestone unit. This sequence is roughly equivalent to those in some eastern basinal areas. This section was used in the thermal analysis, migrational studies, etc., that were carried out as part of the study. The actual clay property values that were used were based on the characteristics of the Boom clay in Belgium and the Oxford clay in the United Kingdom.

7.2.3 Repository Concept Description

7.2.3.1 Nuclear Waste Disposal Strategy

About 2,500 tons of vitrified wastes were estimated to arise from the reference program used for this study. The spent fuel will be reprocessed and vitrified in borosilicate glass. Interim storage of the wastes is to last about 50 years to reduce the heat and radioactivity output of the wastes. This reference program was planned to run to completion in 2025. Additionally, there will be a further 5,200 tons of cladding wastes which were assumed to require storage with the high-level wastes in the same facility. The possibility of also using the facility for a further 27,000 drums of alpha wastes was also assessed. However, the major problem was the disposal of the high-level wastes and cladding hull wastes which arise in roughly the same volumes. Estimates showed that about 18,000 cylindrical canisters would be used of which 9,000 would contain the high-level wastes and 9,000 for the cladding hull wastes. The canisters are 0.3 m (1 ft) in diameter and 1 m (3 ft) long (Figure 7.2.3).

Two basic repository concepts were derived from an analysis of various alternatives shown in Figures 7.2.4 to 7.3.8. Six mined repository concepts were evaluated as well as three deep-borehole concepts. Thermal

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Characteristics of conditioned HLW resulting from the reprocessing of one tonne of spent fuel

PACKAGING		
- system	sealed cylindrical vessel	
- dimensions	ext. diam. 0.3 m; height 1.5 m	
- weight	320 kg	
PRODUCTION	1 canister/tonne of spent fuel	
WASTE PROPERTIES		
- nature	borosilicate glass	
- quantity	volume: 0.1 m^3; weight: 280 kg	
- density	2.8 g/cm^3	
- composition (wt%)	Fission Products Oxides	12.7
	Actinides Oxides	4.4
	Inert Oxides	0.9
	Glass forming Oxides	82.0
CONTAINER PROPERTIES		
- nature	high corrosion resisting S.S.	
- weight	40 kg	

Characteristics of conditioned CHW arising from one ton of spent fuel

PACKAGING		
- system	sealed cylindrical vessel	
- dimensions	ext. diam. 0.3 m; height 1.5 m	
- weight	619 kg	
PRODUCTION	1 canister/tonne of spent fuel	
WASTE PROPERTIES		
- nature	compacted metallic fragments fixed by a mead-alloy matrix	
- quantity	volume: 0.07 m^3; weight: 582 kg	
- density	8.31 g/cm^3 (apparent)	
- composition (wt%)	metallic fragments	53.10
	Pb-Sb Alloy	46.05
	actinides (*)	0.83
	fission products (*)	0.03
CONTAINER PROPERTIES		
- nature	high corrosion resisting S.S.	
- weight	40 kg	

(*) It is assumed that 0.5% of the fuel will remain undissolved and fixed in the cladding hulls.

Figure 7.2.3. High-Level and Cladding Hull Waste Canister Configuration

TABLE 11

Approximate Dimensions for the "Tunnel and Borehole" Repository Models and "In-Tunnel" Repository (F)

MODEL	INTER-BOREHOLE* SPACING (m)	STACK LENGTH (m)	BOREHOLES/TUNNEL AND CLEARANCE (m)*	DISPOSAL TUNNEL* SPACING (m)	DISPOSAL TUNNEL LENGTH(m)	NUMBER OF TUNNELS AND TOTAL LENGTH (km)	MAIN DRIVE* LENGTH (m)
A	50	60	19 (25)	50	1000	24 (24)	1300
B	50	90	16 (25)	50	850	19 (16.2)	1050
C	35	60	20 (25)	35	750	23 (17.3)	900
D	35	90	16 (20)	35	600	19 (11.4)	760
E	3.6	3	200 (5)	17	730	45 (32.9)	865
EII	6.2	6	120 (5)	20	755	38 (28.3)	860
F	N.A.	N.A.	N.A.	50	810	17 (13.8)	950

* Inter-axial spacing.
* Allowing 50 m operating space at each end.
* Clearance at each end of tunnel from nearest disposal hole.

Suggested tunnel diameters: main drive - 6 or 6.5 m; disposal tunnels - 3 m.

N.A. = not applicable

Figure 7.2.4. Mined Repository Models

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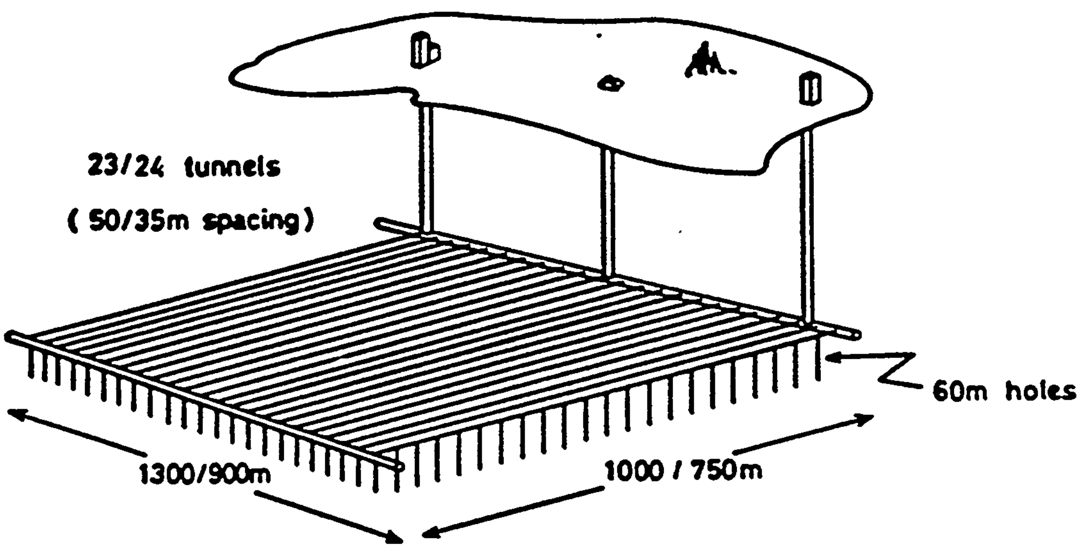
REPOSITORIES A and C



REPOSITORIES B and D

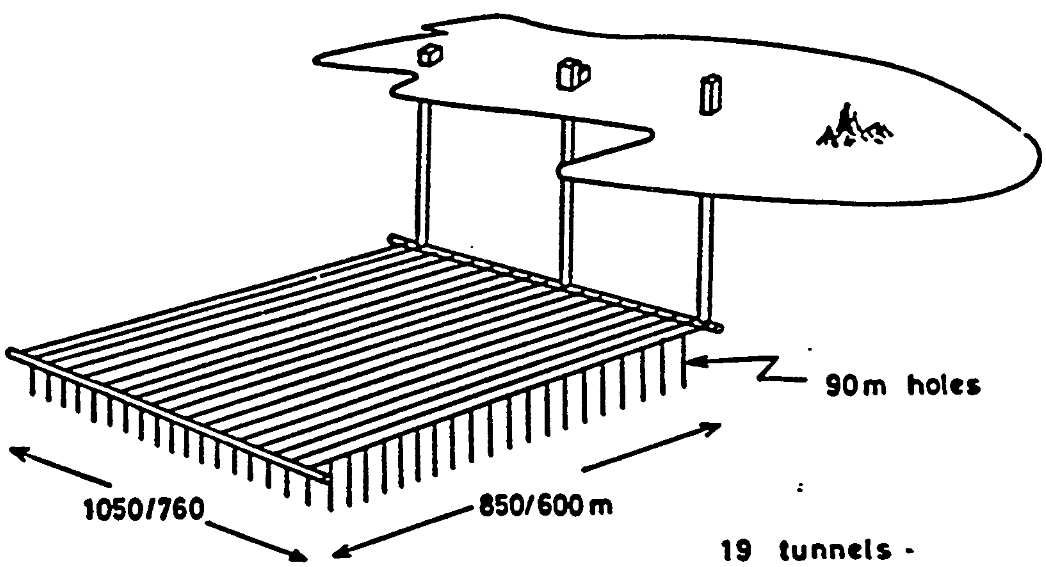


Figure 7.2.5. Mined Repository Facility Designs (1)

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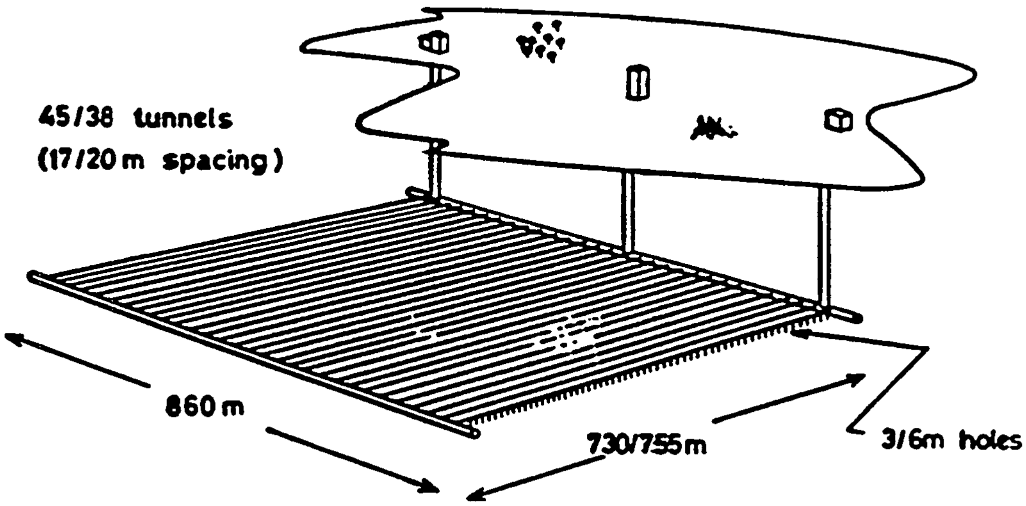
REPOSITORY E(EII)



REPOSITORY F

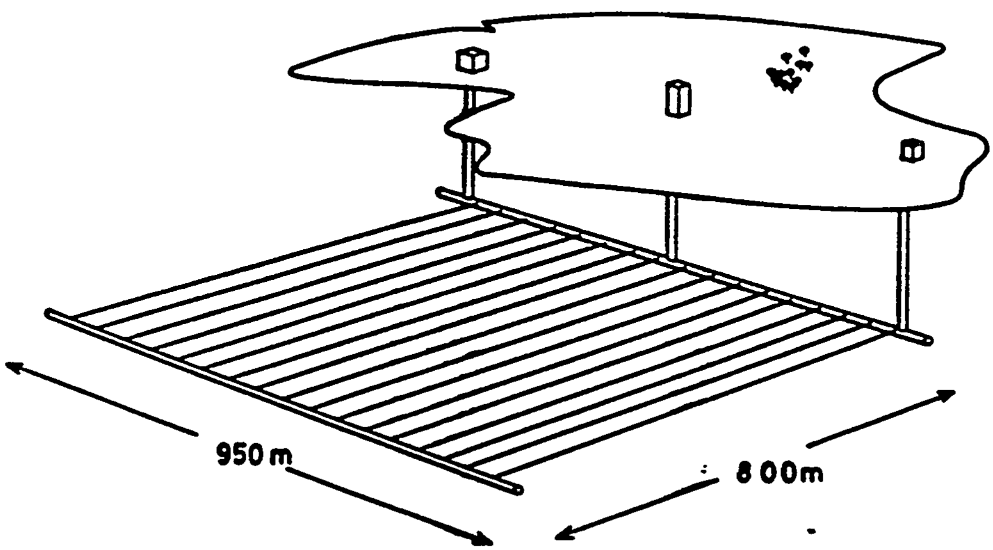


Figure 7.2.6. Mined Repository Facility Designs (2)

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Approximate Dimensions of the Deep Borehole Facility Models

Model	Stack length (m)	Borehole spacing (m)+	Number of holes	Land area occupied (km^2) by boreholes *
DB1	60	50	450	0.97
DB2	90	50	300	0.65
DB3	225	50	120	0.26

* Assumes no buffer zone around site and takes no account of other site facilities. Calculated on the basis of a hexagonal disposition of holes.

+ A 50 m spacing is used to obviate problems with off-vertical holes. In the thermal analysis of the far-field, a 35 m spacing is assumed.

Figure 7.2.7. Deep-Borehole Repository Models

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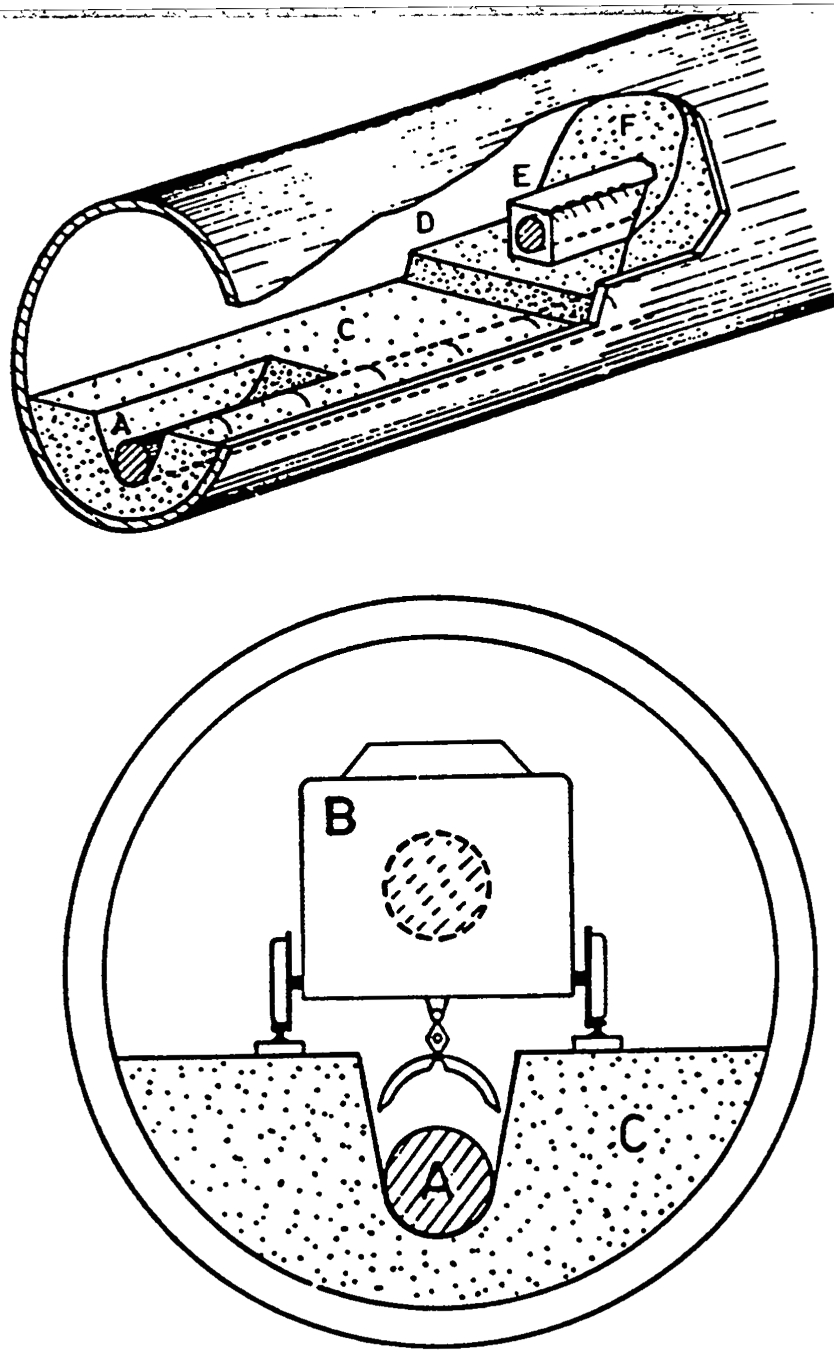

FIG. 26 Schematic representation of "in-tunnel" disposal.

A: HLW packages.
B: Remote control transport and emplacement vehicle (partial shielding) - rail system.
C: Moulded GEOROC bed for HLW. Trench filled with sprayed GEOROC after waste emplacement, using remote vehicle.
D: 50 cm bench: sprayed GEOROC to provide radiation shielding.
E: Cement or GEOROC inverted CHW.
F: Backfill: either GEOROC, cement or compacted clay.

Figure 7.2.8. In-Tunnel Disposal

considerations, constructional aspects and economic factors were used in the evaluation and led to the choices described in the following sections. However, none of the alternatives have been totally rejected since the accumulation of actual site specific data will lead to a re-evaluation of the disposal concepts.

In both concepts, engineered barriers are not considered part of the high-level and cladding hull waste containment system. No credit is given to the waste containers, overpacks and borehole buffer materials. The emphasis is on restoring the hydraulic properties of the clay formation, with the clay acting as principal barrier against radionuclide release. Highly corrosion resistant waste canisters are therefore not considered necessary for the disposal concepts.

7.2.3.2 Surface Facilities

Surface facility details were not part of the study, but as described for other concepts, will include the facilities for the reception and preparation of the waste packages and the normal underground service installations in the case of the mined repository.

7.2.3.3 Repository Configurations

Two disposal methods were assessed, either emplacement in a mined repository or sealing into deep boreholes drilled from the surface.

7.2.3.3.1 Mined Repository. A reference design concept for a mined repository in the clay was chosen on the basis of thermal factors and spatial requirements for the wastes under consideration. Repository depth was assumed to be around 300 m (980 ft). Figure 7.2.9 shows the reference concept design. Disposal of the waste canisters takes place in shallow boreholes, 4 m (13 ft) deep, containing one canister of high-level waste and one of cladding hull waste. These disposal boreholes are spaced about 2.5 m (8 ft) apart along the floor of 3 m (9.8 ft) diameter circular disposal tunnels. A total of about 22 tunnels with a length of 1,000 m (3,280 ft), spaced 20 m (65 ft) apart, will be required. These parallel disposal tunnels will connect to 6 m (20 ft) diameter service drifts at an angle of about 30° (Figure 7.2.9). The service drifts will be about 880 m (2,890 ft) long with extensions, cross-cuts and spurs to accommodate work areas, etc. The main service drift is connected to the waste reception area (Figure 7.2.9) which in turn is connected to the surface via two shafts. One shaft will serve as main access shaft and the other as the waste handling shaft. One or more additional ventilation and emergency exit shafts will be located in other parts of the repository. Transport of the waste canisters underground and remote controlled emplacement takes place

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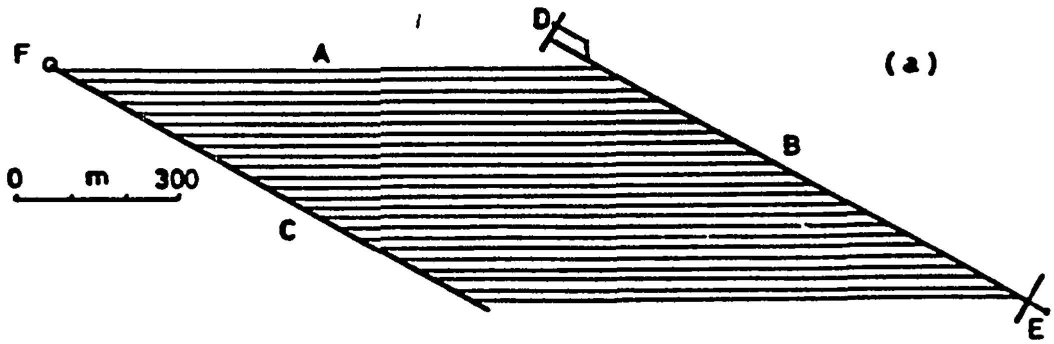
A Disposal tunnels (3·0 m diameter)
B Main service gallery (6·0 m)
C Secondary service gallery
D Waste shaft
E Man shaft
F Ventilation shaft
G Rail turntables



(a) plan view

(b) detail of waste reception area

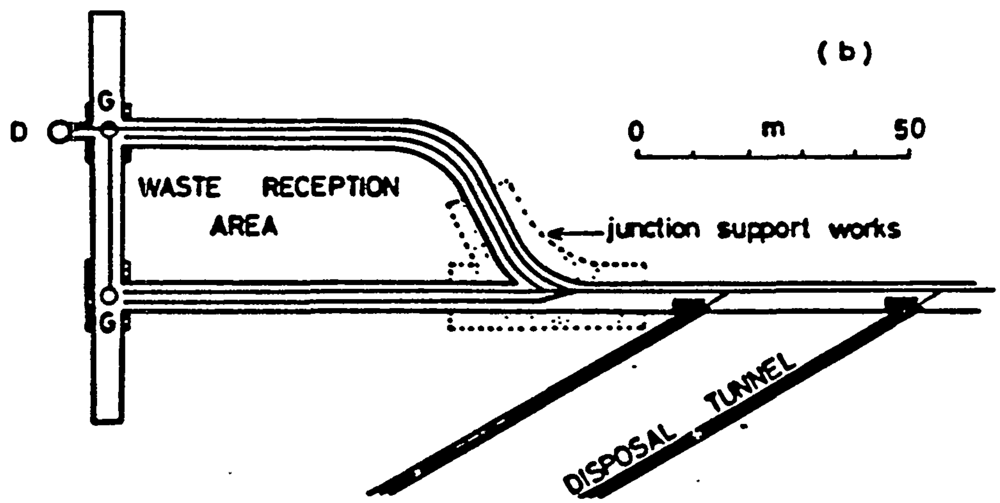


Figure 7.2.9. Reference Mined Repository Design

on a railway system. Figure 7.2.10 shows a possible configuration for the service and disposal tunnels.

Figures 7.2.11 and 7.2.12 show the configuration of the disposal holes. The diameter of the boreholes will be about 0.7 m (2.3 ft) and the depth about 4 m (13 ft). It was assumed that the holes would stay open long enough to allow placement of the waste canisters since the waste canister diameter is much smaller than the borehole. However, liners may be installed if the holes do not stay open. The boreholes can be bored through ports in the tunnel invert lining using small drill rigs.

7.2.3.3.2 Deep-Borehole Repository. The borehole locations within the deep-borehole facility as well as their depths can be varied to suit local geologic conditions. One possible layout of the facility is shown in Figure 7.2.13. The hexagonal pattern is made up of 120 boreholes, spaced about 100 m (330 ft) apart. The holes are about 300 m (980 ft) deep to accommodate 150 waste canisters. High-level and cladding hull waste canisters are lowered alternately into the borehole. Two drill rigs would be used, one for drilling of the holes and one for backfilling. A rail network would be installed to move the rigs and waste canisters around the site as shown in Figure 7.2.14.

7.2.4 Repository Construction

7.2.4.1 Schedule and Costs

Emplacement operations for both repository concepts are estimated to last for about 10-20 years and are planned to start in the early part of the next century, depending on the interim storage period requirements.

Preliminary cost estimates were prepared based on the above reference concepts. The mined repository was estimated to cost about $770 million which includes construction, backfilling, surface facilities and operation. The deep-borehole facility costs came to about $185 million for the same operations.

7.2.4.2 Methods and Equipment

7.2.4.2.1 Mined Repository. The major problem that will probably be encountered in the construction of the repository will be excavation and support of the tunnels in the clay. Construction experience in deep clay

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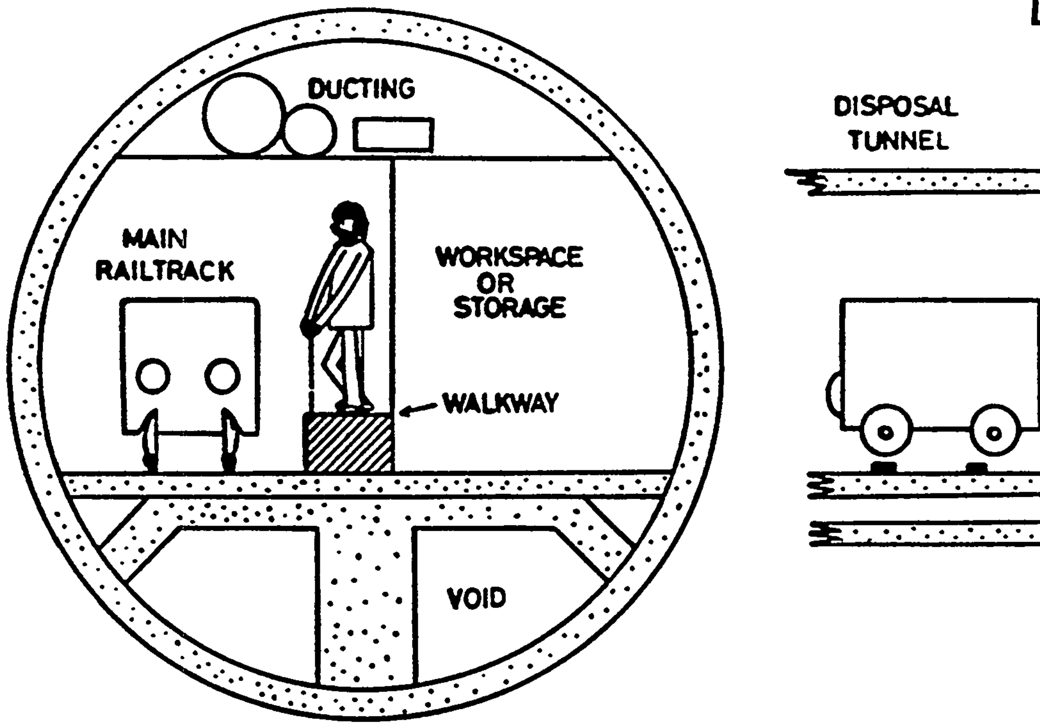

Possible layout in main service gallery. The main railway and a footpath occupy one side while the remaining space is free for workshops or storage. A large void exisxts beneath the floor, which is modular and removable along with the support joists.

Figure 7.2.1Ø. Service Drift Configuration

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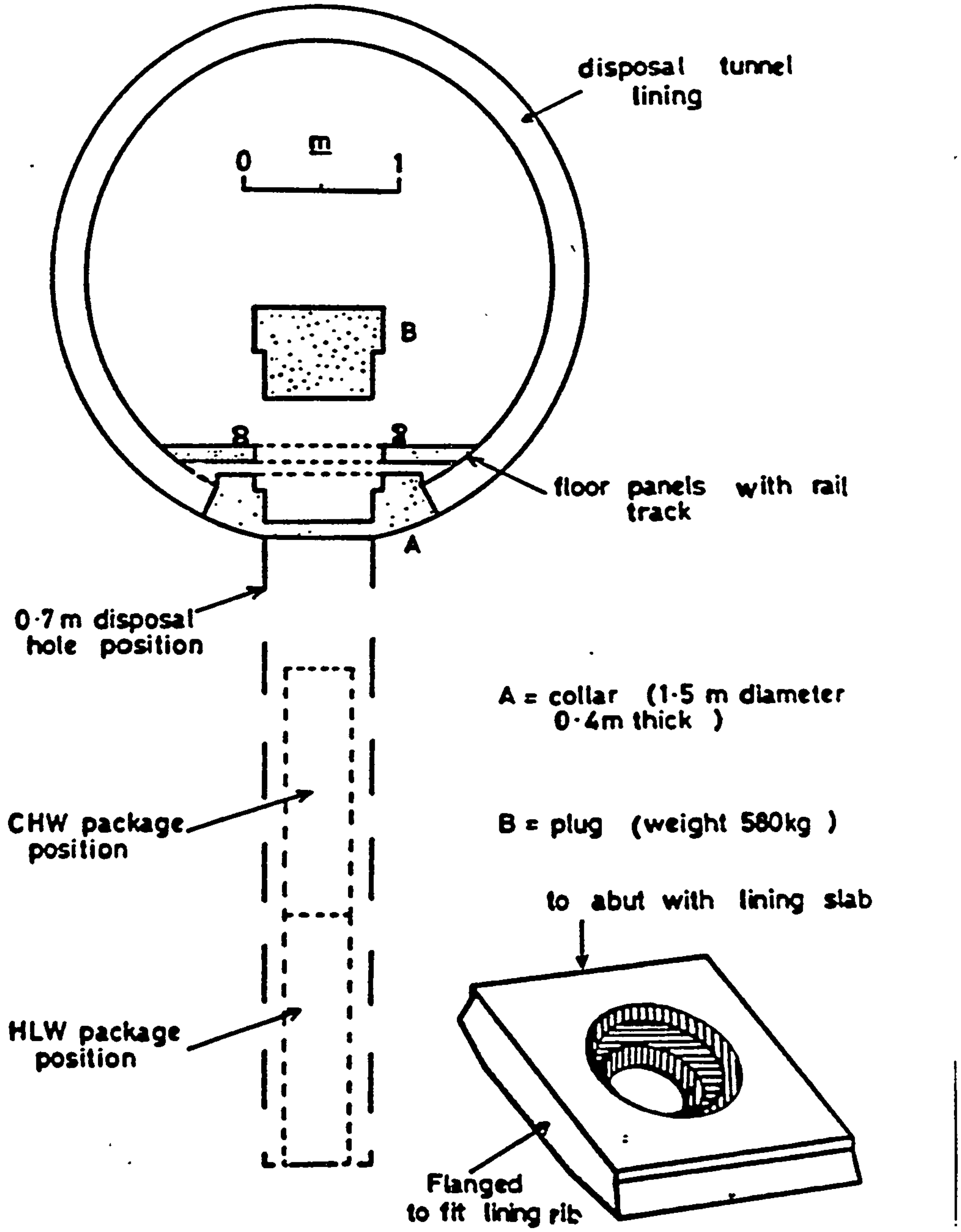

Figure 7.2.11. Disposal Hole Collar System

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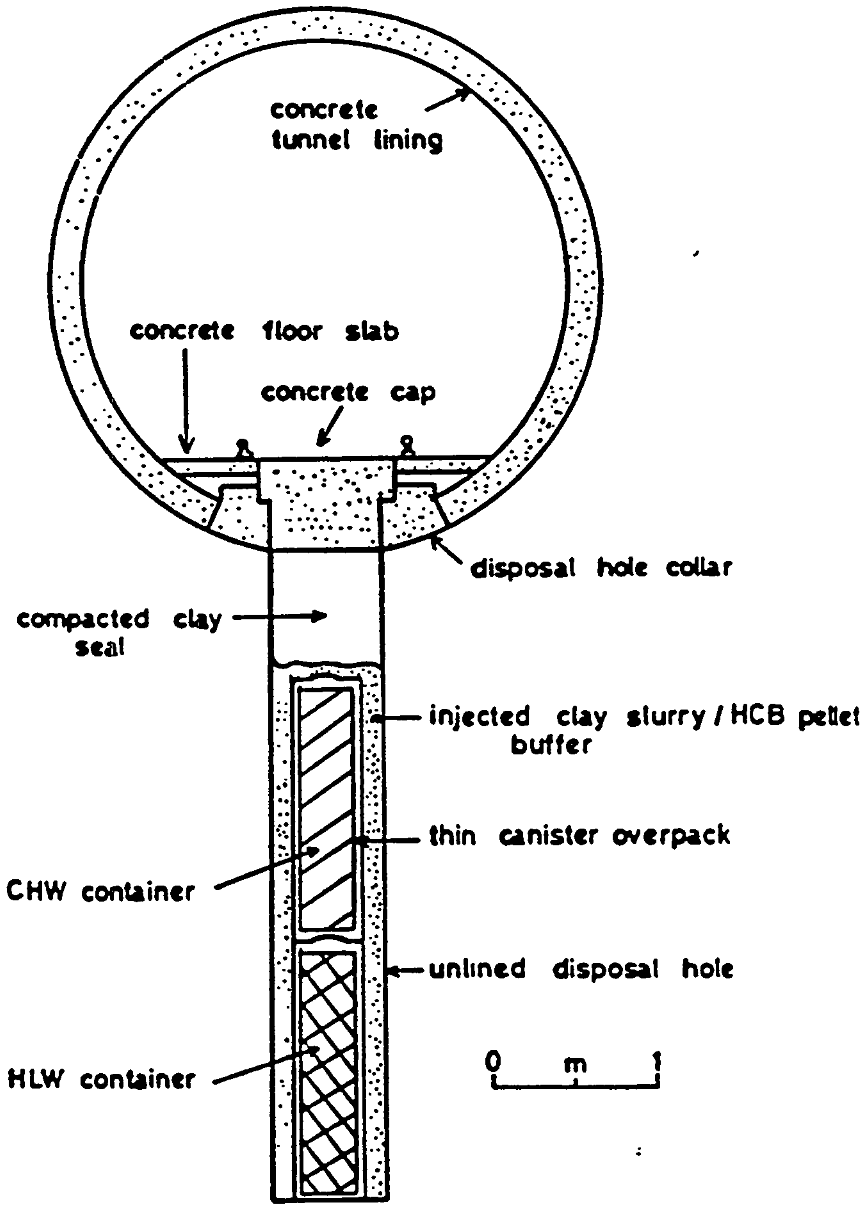

Figure 7.2.12. Completed Disposal Hole

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· Possible layout of a DBF using 120 boreholes with 225 m long disposal stacks, and a hexagonal equidistant hole disposition. Apart from the ring road and site facilities, only a small area of the site need be occupied at any one time.

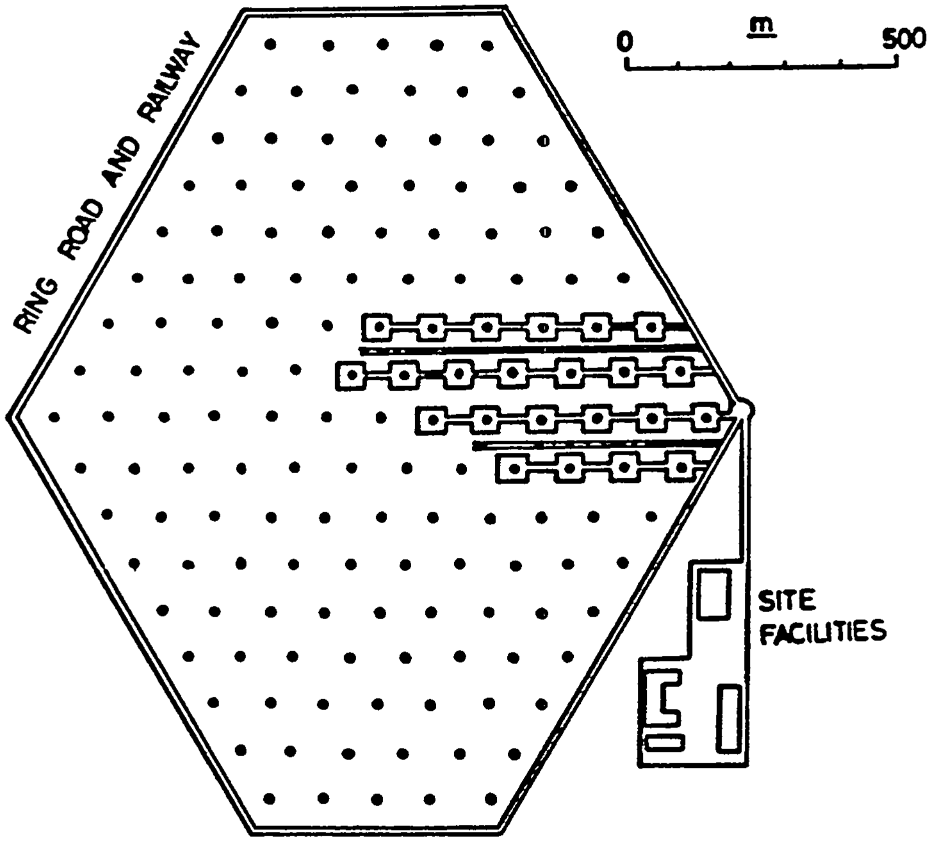

Figure 7.2.13. Surface Layout of Deep-Borehole Repository

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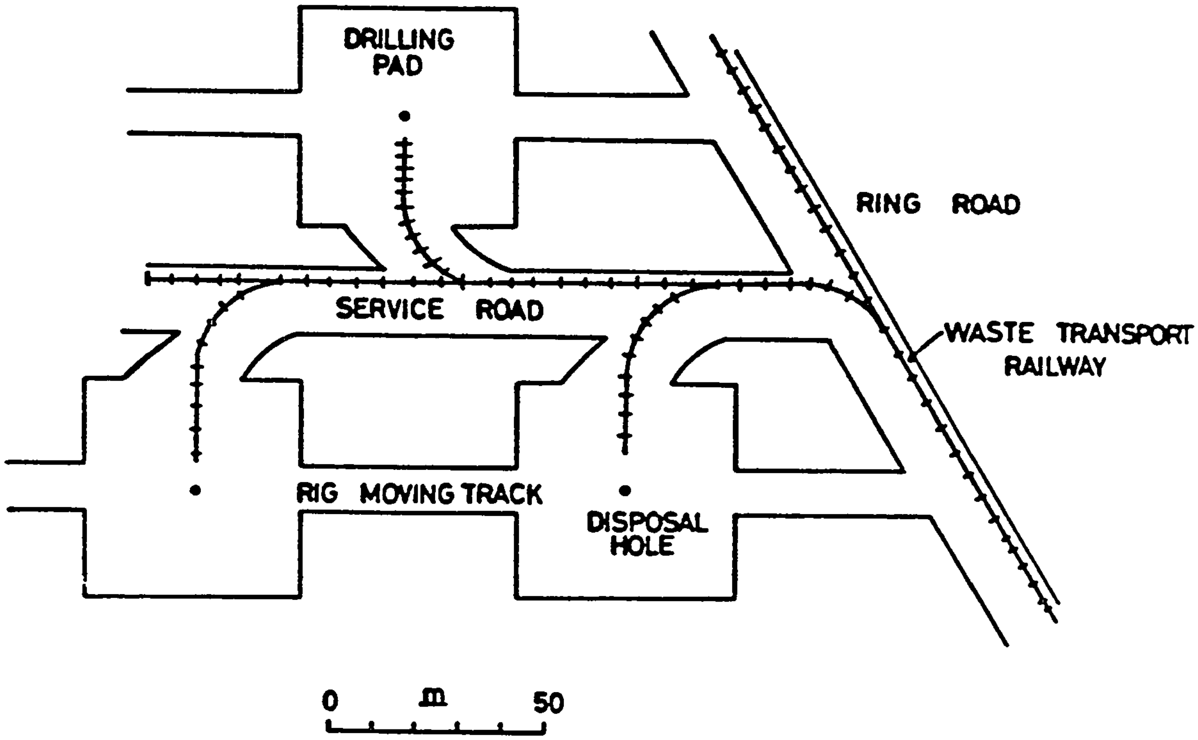

Detail of operational section of a DBF. All concrete pads, road and rail systems within the ring-road are temporary and can be torn up as emplacement moves to another part of the site.

Figure 7.2.14. Deep-Borehole Repository Surface Operations

formations is limited and therefore experimental facilities similar to the Mol underground laboratory in Belgium will be needed to evaluate construction and lining techniques.

For shaft construction the use of blind hole drilling techniques is proposed to avoid the possible need for ground freezing techniques. Figure 7.2.15 shows a comparison of some of the shaft construction methods investigated.

Estimates of the expected tunnel convergence in the generic environment used in the study showed that concrete lining thicknesses of between 20 to 40 cm (8 to 16 in) would be required to support the tunnel walls. Circular tunnel profiles were proposed due to their greater stability in compression. Figures 7.2.16 and 7.2.17 shows alternate layouts for the repository which were derived to allow the use of full-face tunnelling machines (Figure 7.2.18). However, due to problems involving accurate steering requirements, variability in rock conditions, lining emplacement, turning radius limitations, difficulties in moving the machines from one tunnel to another, etc., the full-face system was ruled out in favor of more flexible partial face tunnelling systems (Figure 7.2.18). The selection of the tunnel lining systems was strongly influenced by the eventual backfilling and sealing requirements. Consequently, a removable precast concrete lining system of circular segments was proposed which allows removal of some of the segments during backfilling so that the backfill material can bond with the host rock. This removable lining requirement is one of the reasons the in-tunnel disposal system shown in Figure 7.2.8 was not considered further in the study since elaborate technology would be needed to allow proper backfilling and the risk of radiation exposure for operating personnel would be higher.

The disposal tunnels will connect to the service drifts at an angle of about 30° as shown in Figure 7.2.9. This will allow easier start-up for the tunnelling machines, easier turn-off for the rail tracks and allow a stable junction construction.

7.2.4.2.2 Deep-Borehole Repository. Figure 7.2.19 shows the sequence of operation for the deep-borehole repository. Construction and disposal will take place as follows:

1. An oversize hole is drilled to the required depth with standard drilling technology using drilling mud to keep the hole open.

2. A casing or lining is placed into the hole and cemented from the bottom of the hole to the top of the disposal zone.

3. The drill rig is removed from the site and starts drilling on the next hole. A temporary building is erected over the borehole and

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Comparison of three appropriate shaft-sinking techniques for use in clay/mixed sediments

	MANUAL EXCAVATION	BLIND ROTARY DRILLING	REAMING FROM A PILOT HOLE
1	Work starts at surface	Work starts at surface	Pilot hole must enter existing deep tunnel
2	Cannot handle large aquifers without ground freezing	Aquifers no problem	Cannot handle any large water inflows
3	Lining not a problem	Lining straight forward	Must stand unsupported before lining whole shaft
4	Slow and dangerous operation	Rapid with no safety problems	Fairly quick, no safety problems until lining
5	Little formation damage	-ditto-	-ditto-
6	Possible sealing problems	Easy to seal	Possible sealing problems
7	Inspection and testing easy during construction	Impossible to inspect or test rock manually	Inspection possible while lining
8	Costs predictable and relatively low	High cost of lining and initial capital	Low lining costs, otherwise unpredictable
9	Can be sunk blind	Can be sunk blind	Requires prior data on rock properties
	CONCLUSION		
	Applicable and cheap, but slow, with poor safety and may require ground freezing	Expensive but quick, safe and reliable. Best option	Only appropriate for second/third shafts in a totally clay section

Figure 7.2.15. Comparison of Shaft Construction Methods

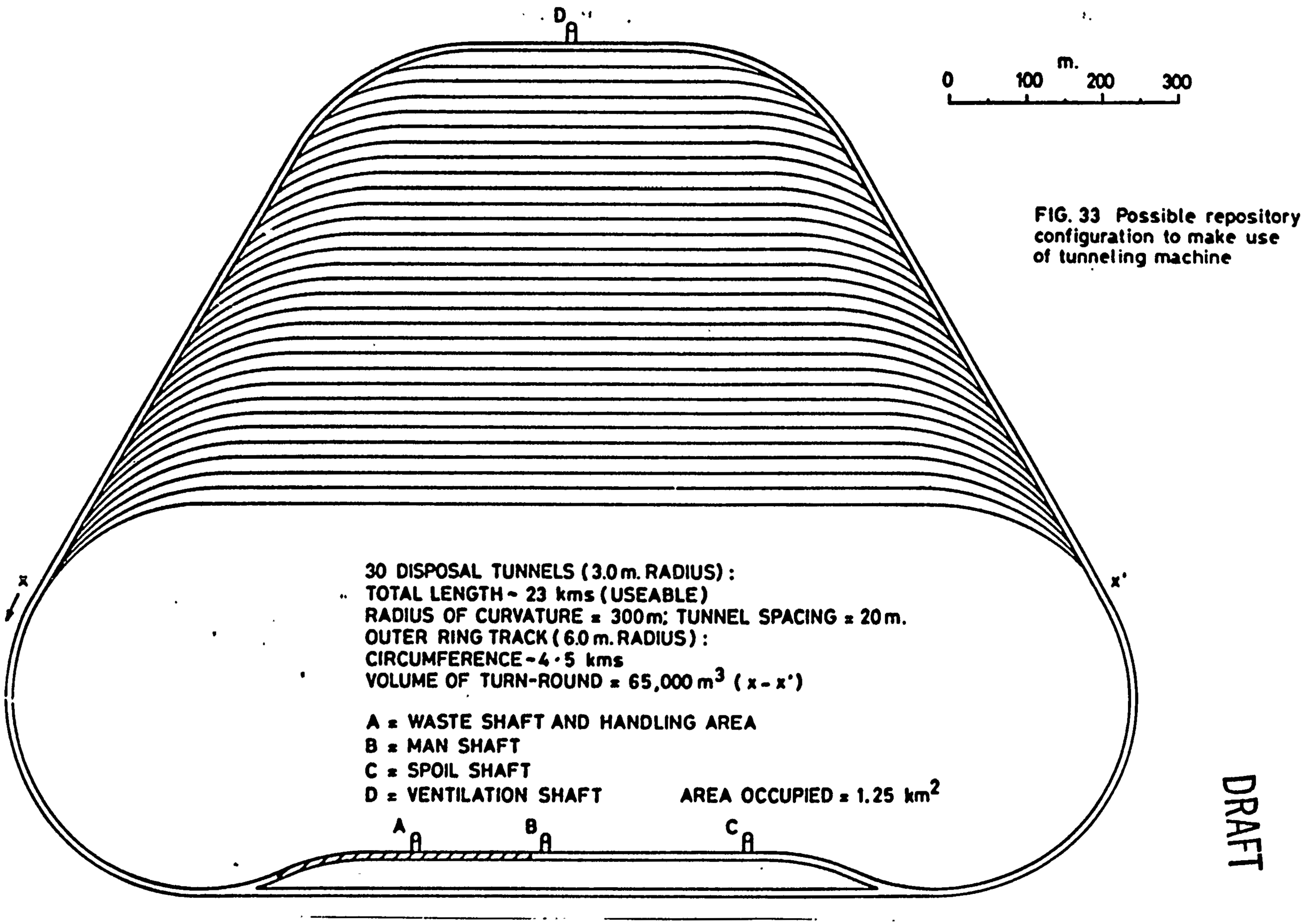

Figure 7.2.16. Alternate Mined Repository Layout (1)

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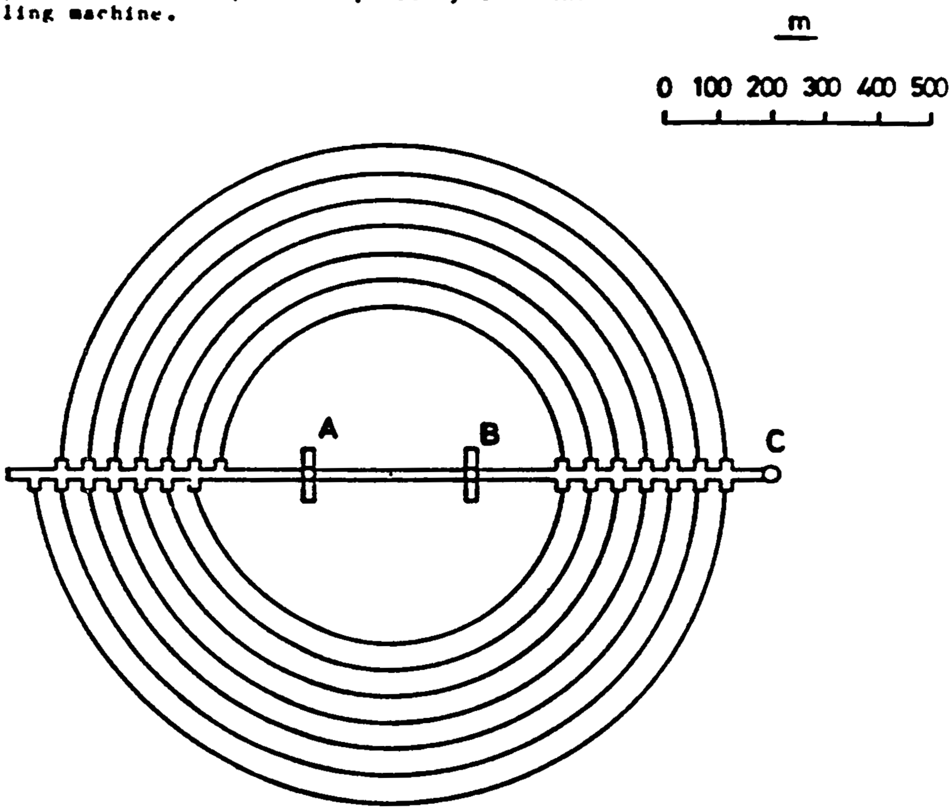
Spiral tunnel design for repository to make use of tunneling machine.



14 disposal tunnels: length 20km
Minimum radius = 300 m

Spine service tunnel = 1.4 km
15 parking/service areas for tunneling machine

A = Waste shaft
B = Man shaft
C = Spoil shaft
Total area = 1,15 km^2

Figure 7.2.17. Alternate Mined Repository Layout (2)

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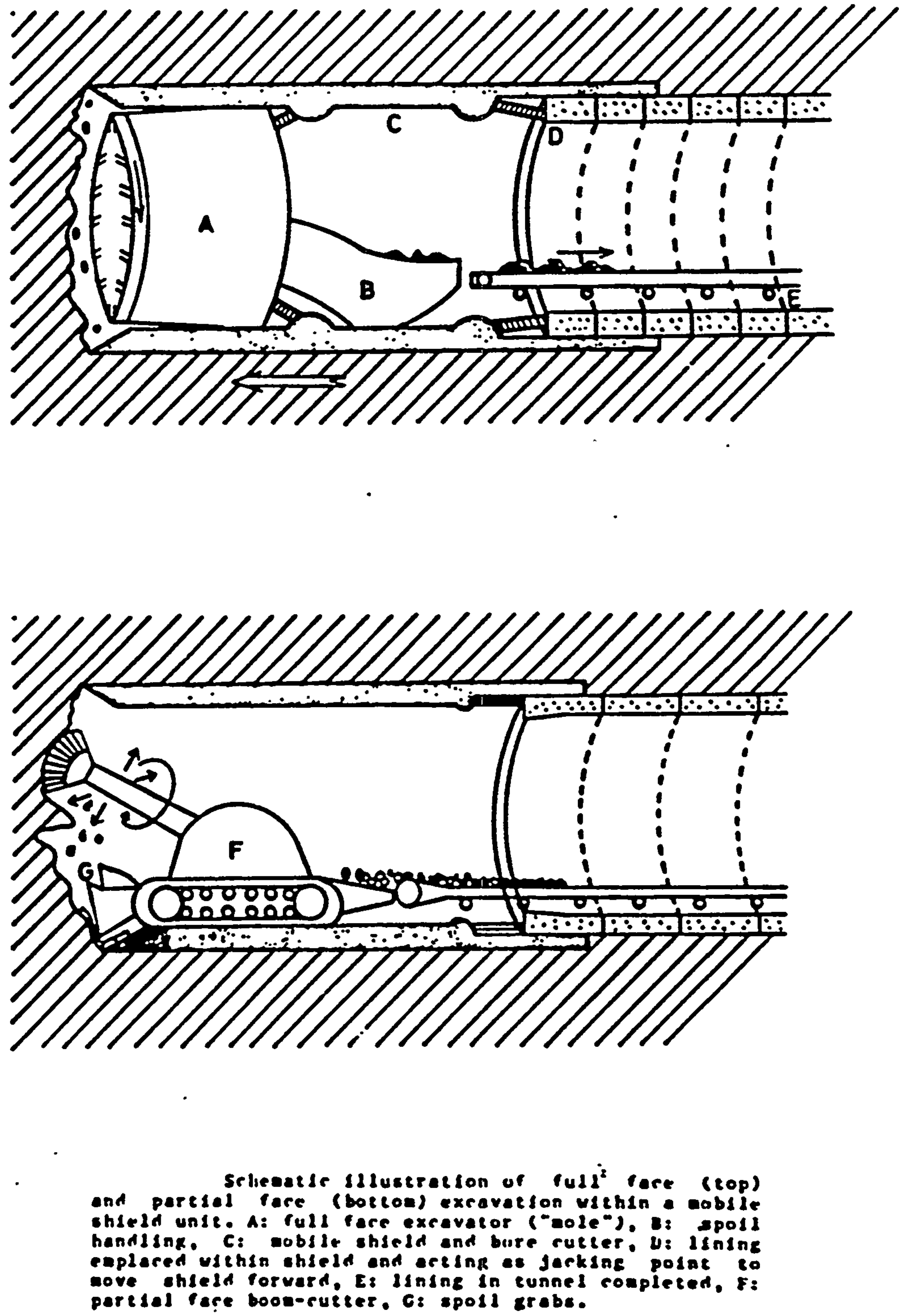

Schematic illustration of full face (top) and partial face (bottom) excavation within a mobile shield unit. A: full face excavator ("mole"), B: spoil handling, C: mobile shield and bore cutter, D: lining emplaced within shield and acting as jacking point to move shield forward, E: lining in tunnel completed, F: partial face boom-cutter, G: spoil grabs.

Figure 7.2.18. Tunnel Excavation Methods

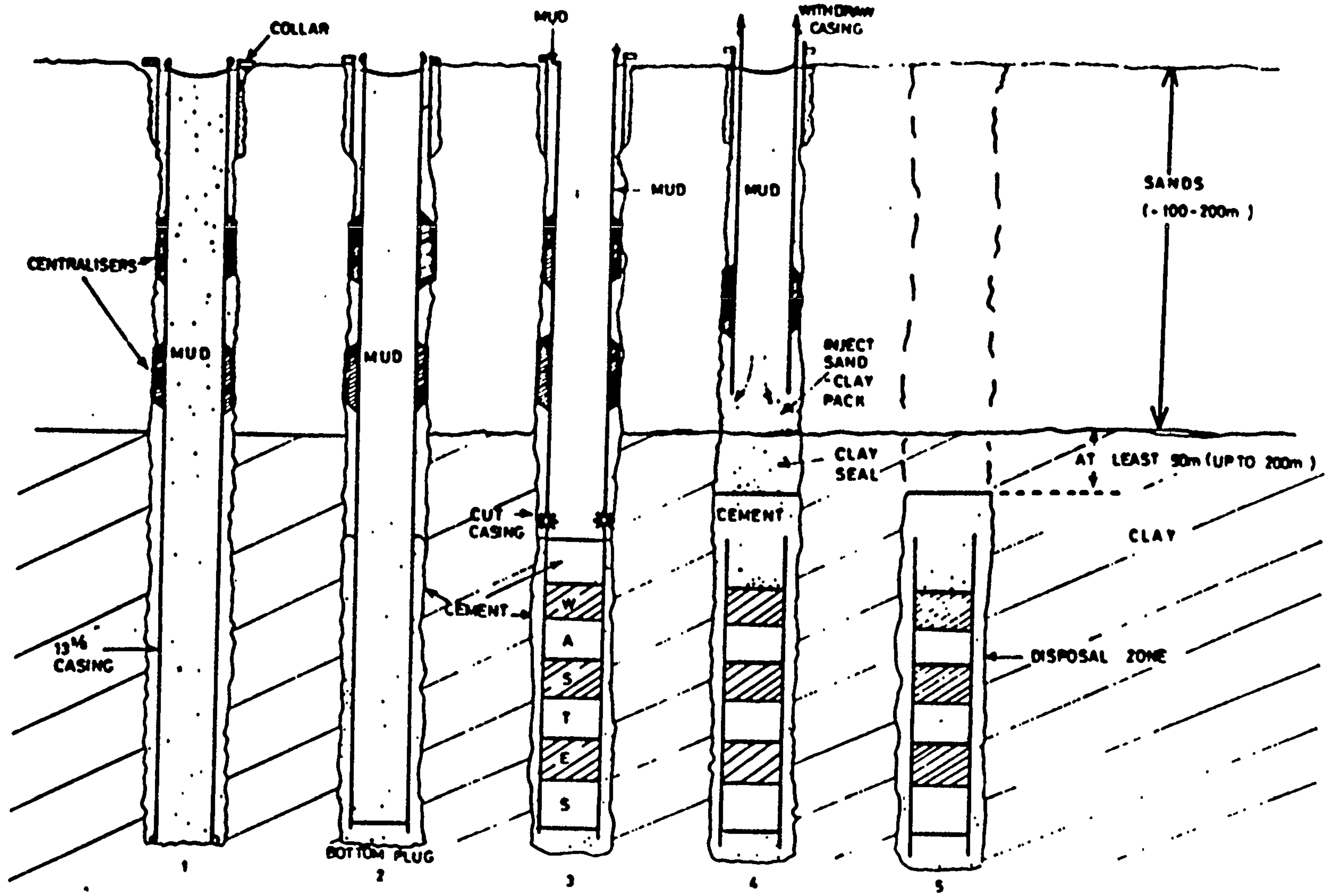

Figure 7.2.19. Deep-Borehole Disposal System

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a rail spur laid to the top of the hole. A shielded rail truck will haul the waste canisters to the borehole and the canisters are lowered into the hole (Figure 7.2.20). High-level and cladding hull canisters will be lowered alternately. After 150 canisters have been lowered, a concrete plug is placed on top of the last canister.

4. The building is removed from the borehole and the backfill rig is moved into place. The casing is cut above the disposal zone and removed and the hole is backfilled to the surface.

5. The removal of the borehole liner and the injection of the backfill and sealing materials is intended to restore the original conditions as much as possible. The disposal zone will be completely isolated and the cement and remaining casing around the waste canisters is not intended as a barrier and thus this concept, as the mined repository, relies on the clay formation for containment and isolation of the radioactive waste materials.

7.2.5 Waste Emplacement Cycle

7.2.5.1 Waste Treatment

Current plans call for the reprocessing of all spent fuel and interim storage for about 50 years before final disposal in a repository in clay. Spent fuel is currently being reprocessed in the United Kingdom and the vitrified wastes will be returned to Italy for storage and disposal. Research and development programs are studying the construction and operation of full-scale reprocessing and vitrification plants in Italy.

7.2.5.2 Waste Transport

Transport systems for the high-level wastes were not included in the study but it is assumed that similar systems to those described for other countries will also be used. Rail, road or sea transport should be feasible in Italy.

7.2.5.3 Waste Package

Figures 7.2.3 and 7.2.21 summarize the waste package concepts that were used in the reference disposal concepts. The high-level and cladding hull wastes will be stored in similar stainless steel canisters while the alpha

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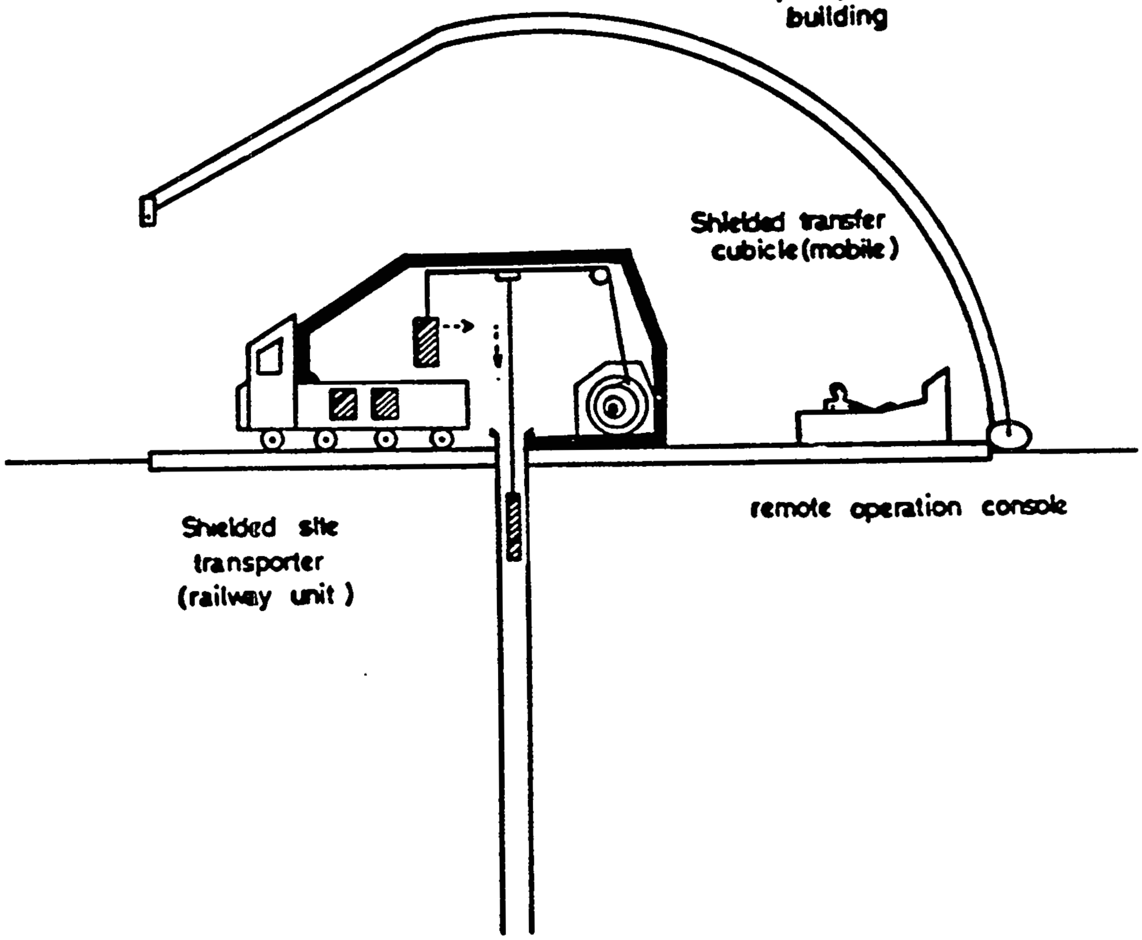

Figure 7.2.20. Deep-Borehole Waste Handling

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Characteristics of the conditioned alpha wastes resulting from the reprocessing of one tonne of spent fuel

PACKAGING - system : - dimensions : - weight :	sealed cylindrical vessel (drum) ext. diam. 0.56 m· height 0.86 m 525 Kg
PRODUCTION :	3 drums / ton of spent fuel
WASTE PROPERTIES - nature : - composition (wt:) : - quantity : - density :	granule from high temperature incineration and melting embedded in Portland cement granules 70 Portland cement 20 water 10 volume: 0.200 m^3/drum weight: 500 kg/drum 2.5 g/cm^3
CONTAINER PROPERTIES - nature - weight	 mild steel eventually pretreated to increase the resistance to corrosion 25 kg

Characteristics of the iodine wastes resulting from the reprocessing of one tonne of spent fuel

PACKAGING - system : - dimensions : - weight :	sealed cylindrical vessel (drum) ext. diam. 0.56 m; height 0.86 m 400 Kg
PRODUCTION :	8×10^{-3} drums / ton of spent fuel
WASTE PROPERTIES - nature : - composition (w:) : - quantity : - density :	barium iodate immobilized in Portland cement Portland cement 43.30 water 37.70 iodine 9.05 barium 10.00 volume: 0.200 m^3/drum weight: 364 kg/drum 1.87 g/cm^3
CONTAINER PROPERTIES - nature : - weight :	mild steel eventually pretreated to increase the resistance to corrosion 25 kg

Figure 7.2.21. Alpha and Iodine Waste Canister Configurations

wastes and iodine wastes will be encased in mild steel canisters. All canisters will be corrosion resistant although a high degree of corrosion resistance is not considered necessary as discussed above.

7.2.5.4 Underground Waste Emplacement

7.2.5.4.1 Mined Repository. In the mined repository, the waste emplacement operation will be a remotely controlled operation to avoid radiation exposure of the operating personnel. Figure 7.2.22 shows a schematic of the waste emplacement operation. A shielded transport container carrying one high-level and one cladding hull waste canister will be moved by locomotive to the appropriate disposal borehole where the concrete cover in the tunnel invert has been removed. The high-level waste canister would be lowered into the borehole first, followed by the cladding hull canister. All operations would be monitored using cameras and sensors. After removal of the waste transporter, a backfill unit would fill the annulus around the canisters with a buffer material of bentonite pellets and clay slurry. Preformed blocks of clay material would then be placed on top of the cladding hull canister and compacted into the borehole to ensure a good bond with the host rock. The concrete plug will be replaced and the canisters in the borehole will then be fully shielded. The final configuration of the borehole is shown in Figure 7.2.12. During emplacement operations, access to the waste transport drifts and disposal tunnels will be restricted to ensure operator safety.

Backfilling of the disposal tunnel can commence as soon as waste emplacement has moved to the next tunnel. The exact methods and backfill compositions are still under investigation. Backfill operations are also closely related to the type of lining that has been installed in the disposal tunnels. One possible backfill scenario would begin with the construction of a bulkhead at one end of the disposal tunnel to act as a barrier to the service drift. Removal of the rail track, concrete invert segments and borehole plugs would then proceed in intervals of perhaps 10 m (30 ft). A predetermined number of lining segments would then be removed to expose the clay formation. Backfilling, most likely with a swelling clay material, could then commence in stages. Compaction of the backfill material will be required to ensure proper filling of the voids. Figure 7.2.23 shows the final configuration of the backfilled disposal tunnel.

Disposal of the alpha waste canisters (Figure 7.2.21) in the mined repository is also being considered. The most likely option would be to dispose of them in the large service tunnels since the introduction of additional foreign materials such as waste canisters into the backfill of the disposal tunnels may jeopardize the proper function of the isolation

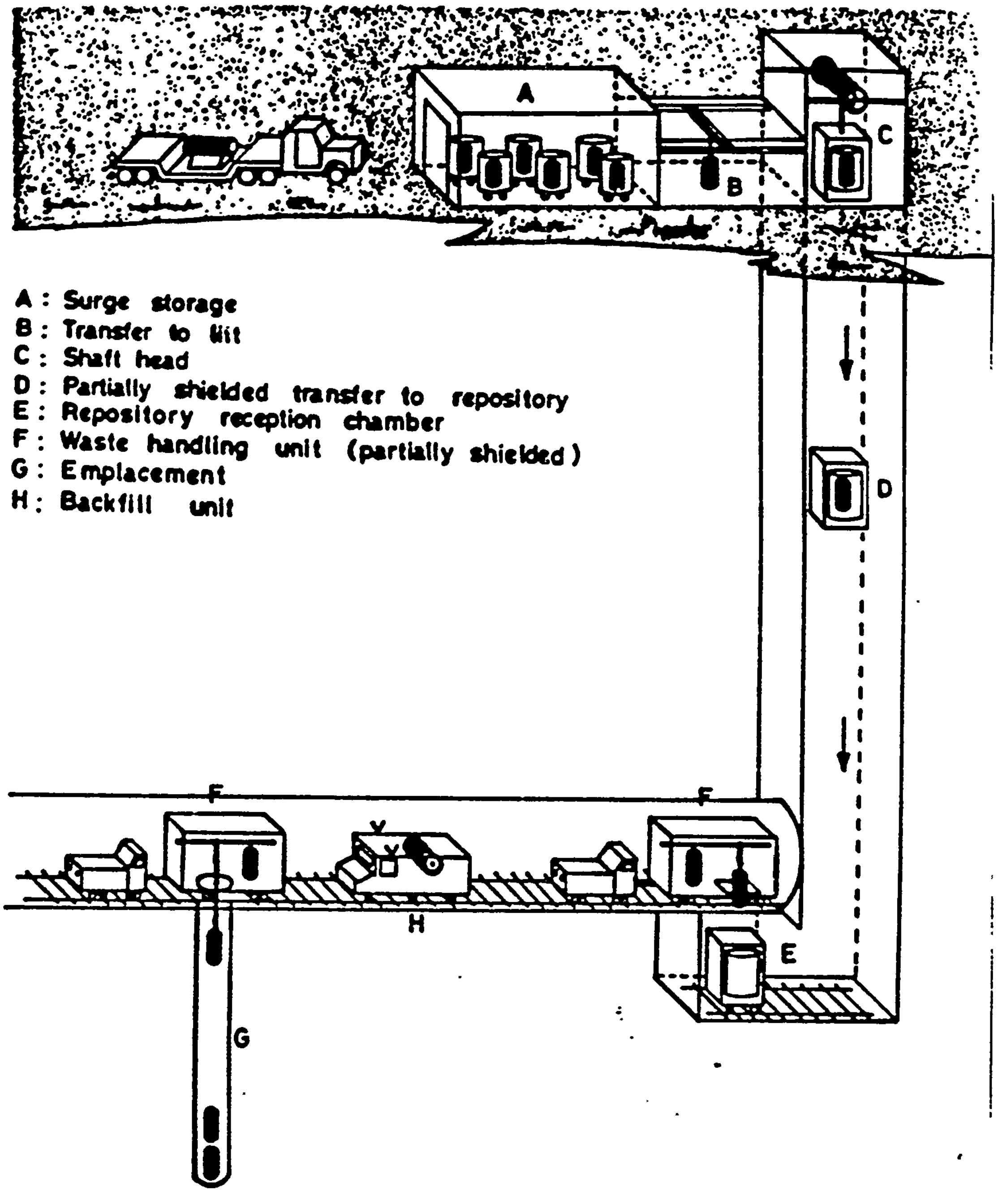

Figure 7.2.22. Mined Repository Waste Handling

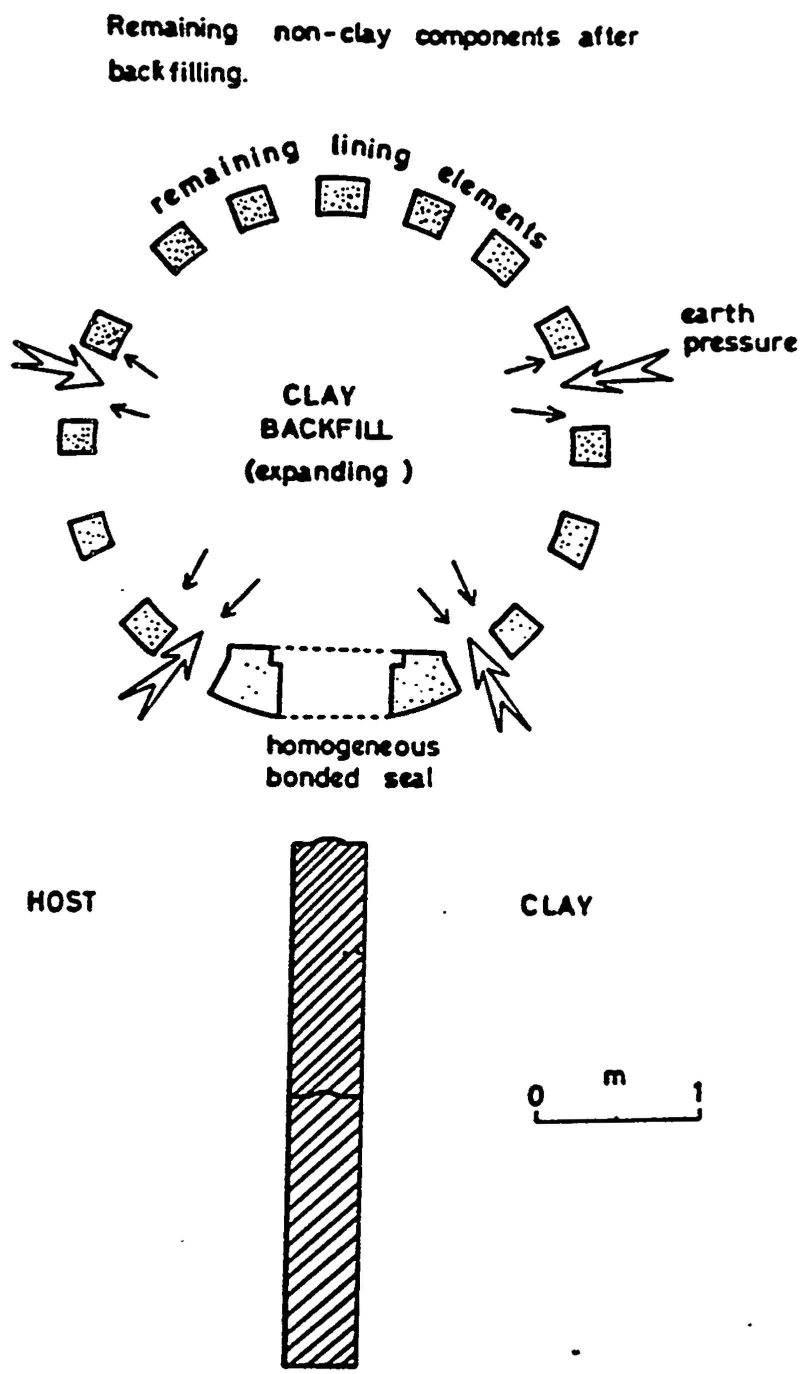

Figure 7.2.23. Completed Disposal Tunnel Backfilling

system around the boreholes. Further studies will be required on the integration of the emplacement of the canisters with the removal of the tunnel lining and backfilling operations.

The final operation will be removal of all services from the repository drifts and shafts, and sealing all openings using clay material as backfill and concrete/cement and bitumen as seals.

7.2.5.4.2 Deep-Borehole Repository. Figures 7.2.19 and 7.2.24 show the waste emplacement and backfilling procedures in the deep-borehole concept. The waste packages are lowered into the borehole (high-level and cladding hull wastes alternate) using a shielded winch. Overpacks may be used (Figure 7.2.24) to allow use of spring locked grapples and allow stacking of the canisters in the borehole. About 150 canisters will be stacked in each borehole and every 30 m (100 ft) it may be advisable to place a cement and clay plug as additional water barriers and to relieve the load on the lower canisters. Emplacement takes place to the top of the disposal zone where a 3 m (10 ft) plug is injected. A bentonite clay seal is then installed to the top of the clay formation as the casing is removed from the borehole. The bentonite will swell and will ensure complete resealing of the clay above the canisters and thus provide the required isolation. The remainder of the hole to the surface may contain water-bearing sands and would thus most likely be backfilled with a sand/gravel material to restore the original ground conditions as much as possible. The final operation will be the removal of the conductor casing and borehole collar and reclamation of the borehole site.

7.2.6 Buffer and Sealing Materials

Buffer and sealing materials being considered are mainly bentonite and other clay materials. These materials will be used as backfill material in the boreholes and tunnels in the repository clay formation to reinstate the original conditions as much as possible. In the isolation system, no credit is given to the waste package and borehole buffer materials as only the clay formation is relied on for radionuclide retention. For shaft sealing, concrete/cement and bitumen are being considered as sealing materials.

7.2.7 Summary

Two concepts are being considered for the disposal of high-level radioactive wastes in Italian clay formations. One is the mined repository concept where disposal tunnels are excavated at depth and the waste canisters are emplaced in boreholes in the floors of these tunnels. The other concept is the deep-borehole concept where the waste canisters are

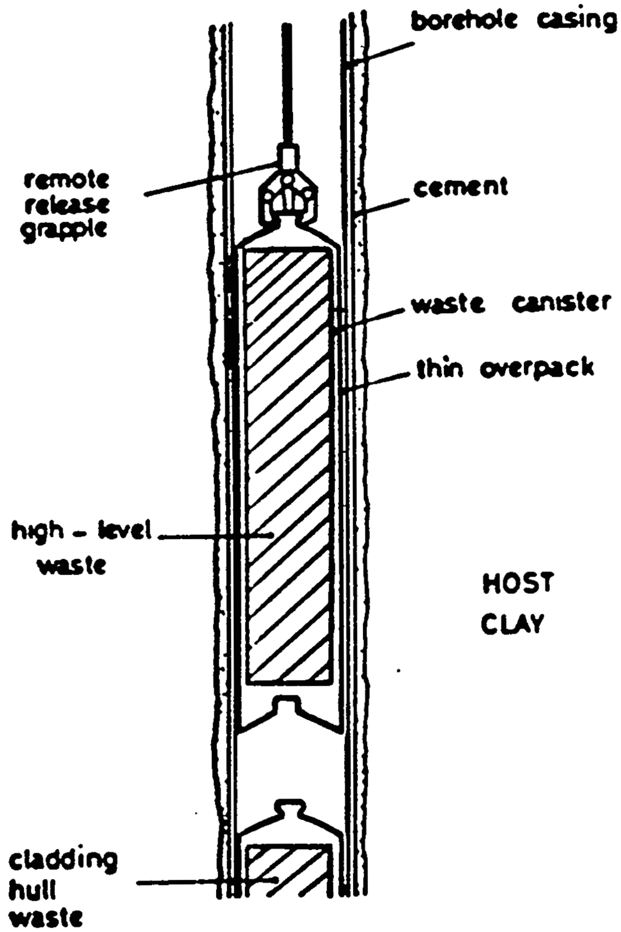

Figure 7.2.24. Deep-Borehole Waste Emplacement

placed in deep boreholes drilled from the surface into the clay formation. Each system has associated advantages and problems which are outlined in the following sections.

7.2.7.1 Advantages

Advantages of the mined repository include:

- Supervised/monitored waste emplacement
- In situ inspection of the disposal areas
- Manual installation of seals and backfillers
- Site characterization of the repository area from underground locations
- Capability to accept alpha wastes in addition to high-level and cladding hull wastes.

Advantages of the deep-borehole system include:

- Better cost effectiveness
- Less excavation volume per m^3 of waste
- Less disturbance of the clay zones
- Disposal operations can be handled from the surface
- Greater flexibility in locating disposal holes
- Less surface construction impact.

7.2.7.2 Potential Problems

Some potential problems may include:

- Low structural strength of clay
- Limited experience with large excavations at depth
- Association with valuable resources
- Limited knowledge of the effects of heat and time on the behavior of the clays as an isolation medium.

- Extended storage of the alpha wastes until the closure of the mined repository appears unacceptable and options such as modifying an existing mine to accept these wastes as well as low-level wastes should be considered.

7.2.7.3 Summary

The deep-borehole repository is much more economical, more flexible and less intrusive on the surface than a mined repository and was recommended as the primary disposal concept. In both repository concepts, no isolation credit is given to the waste package and borehole buffer materials and the emphasis is on restoring the hydraulic properties as much as possible and rely on the clay to act as release barrier.

8.0 EVAPORITE FORMATIONS

8.1 SALT: THE DUTCH DISPOSAL CONCEPT

The Dutch radioactive waste disposal program envisions extended interim storage of the high-level wastes and spent fuel followed by one of several final geologic isolation options. The options include onshore geologic disposal, disposal in a North Sea salt dome and sub-seabed disposal. This chapter describes the Dutch disposal system proposed in the early 1980s for the disposal of high-level waste in an underground repository in an onshore salt dome (Hamstra). The descriptions are based on model salt dome assumptions as no site specific data was available at the time of the study.

8.1.1 Geographic Location

The descriptions are based on model salt dome data since no site specific data was available at the time of the study. Some potentially suitable locations have, however, been identified among the salt domes in the northern region of the Netherlands.

8.1.2 Summary of Geologic Setting

No detailed site specific data is available at this time but salt domes are large intrusive formations often covering many square kilometers in area. A high-level waste disposal system requires to be fully surrounded by a buffer of salt several hundreds of meters in thickness to provide adequate isolation of the wastes and thus the massive salt domes can provide this protection. Salt also has favorable properties for the isolation of high-level wastes: absence of water, relatively high thermal conductivity, lower yield value of creep (self-sealing of openings), and the excavation of caverns and drifts in salt is relatively easy from a construction viewpoint.

8.1.3 Repository Concept Description

The total design of the repository will consist of the surface facilities for the storage, preparation and dispatch of the wastes and the underground facilities for the final isolation of the wastes. This section outlines a proposed repository configuration for a repository in a salt dome. One basic assumption for the concept was that an at least 200 m (660 ft) thick, relatively undisturbed salt buffer must exist around the whole repository. This will be one requirement that may limit the location of the actual repository and the associated surface facilities.

8.1.3.1 Nuclear Waste Disposal Strategy

The basis of the Dutch nuclear waste disposal strategy is the reprocessing of all wastes. Spent fuel accumulation from a projected nuclear power plant capacity of 2.5 GWe has been estimated at 420 MT (460 tons) by the year 2000. The reprocessing, operating and decommissioning wastes will be collected in a single centralized facility. All wastes will be stored for an extended period of time to allow decay of the radioactivity and heat production. Final disposal of the wastes is to take place in an underground repository. Investigations have been conducted into the feasibility of constructing the repository in the salt domes in northern Holland. Other methods such as disposal in the sub-seabed and salt domes in the North Sea are also under consideration.

8.1.3.2 Surface Facilities

Figure 8.1.1 shows a conceptual view of the repository proposed in a salt dome. The facilities will consist basically of the surface facilities to receive and prepare the wastes for underground disposal, the shafts, and the underground complex of headings and boreholes for the disposal of the wastes.

The surface facilities which may or may not be located at the repository include the central receiving and dispatch station and the interim storage facility. Details of these facilities are still being investigated but they are likely to be similar in nature to those described for other waste disposal programs (e.g., Sweden).

Facilities at the repository will include the waste receiving, temporary storage and preparation facilities. In these facilities, the wastes will be transferred from the surface transport system and prepared for transport and disposal underground. Other surface installations will include those normally required for the operation of a mine such as the hoists, headframes, ventilation systems, shops, etc.

8.1.3.3 Underground Configuration

Figures 8.1.1 and 8.1.2 show the salt dome repository concept. Two basic disposal methods are proposed, depending on the type of waste. For the low- and intermediate-level wastes, a series of disposal caverns are to be excavated between two levels, and the high-level wastes are to be placed in boreholes drilled into the floor of the disposal drifts.

With the 200 m (660 ft) thick protective salt barrier around the repository and with the assumed depth to the top of the salt of about

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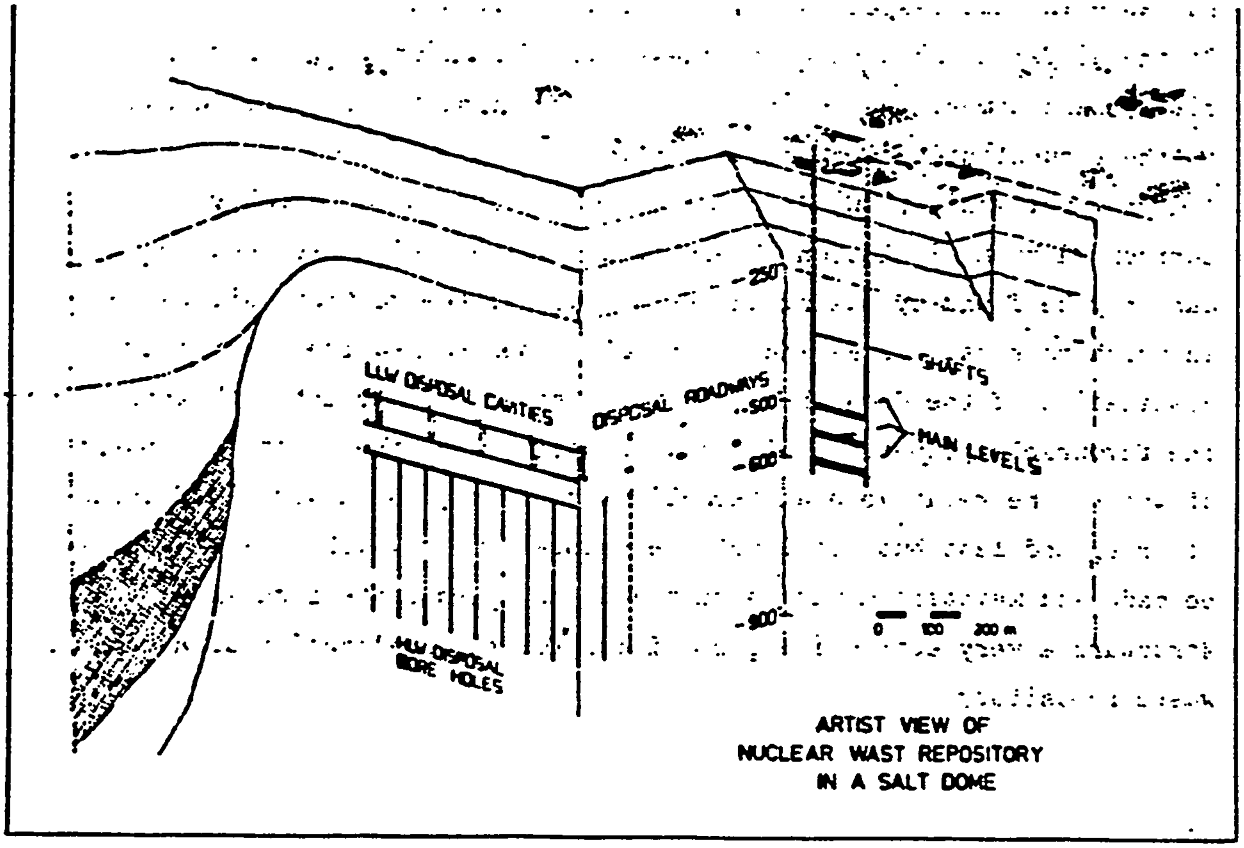

Figure 8.1.1. Salt Dome Repository

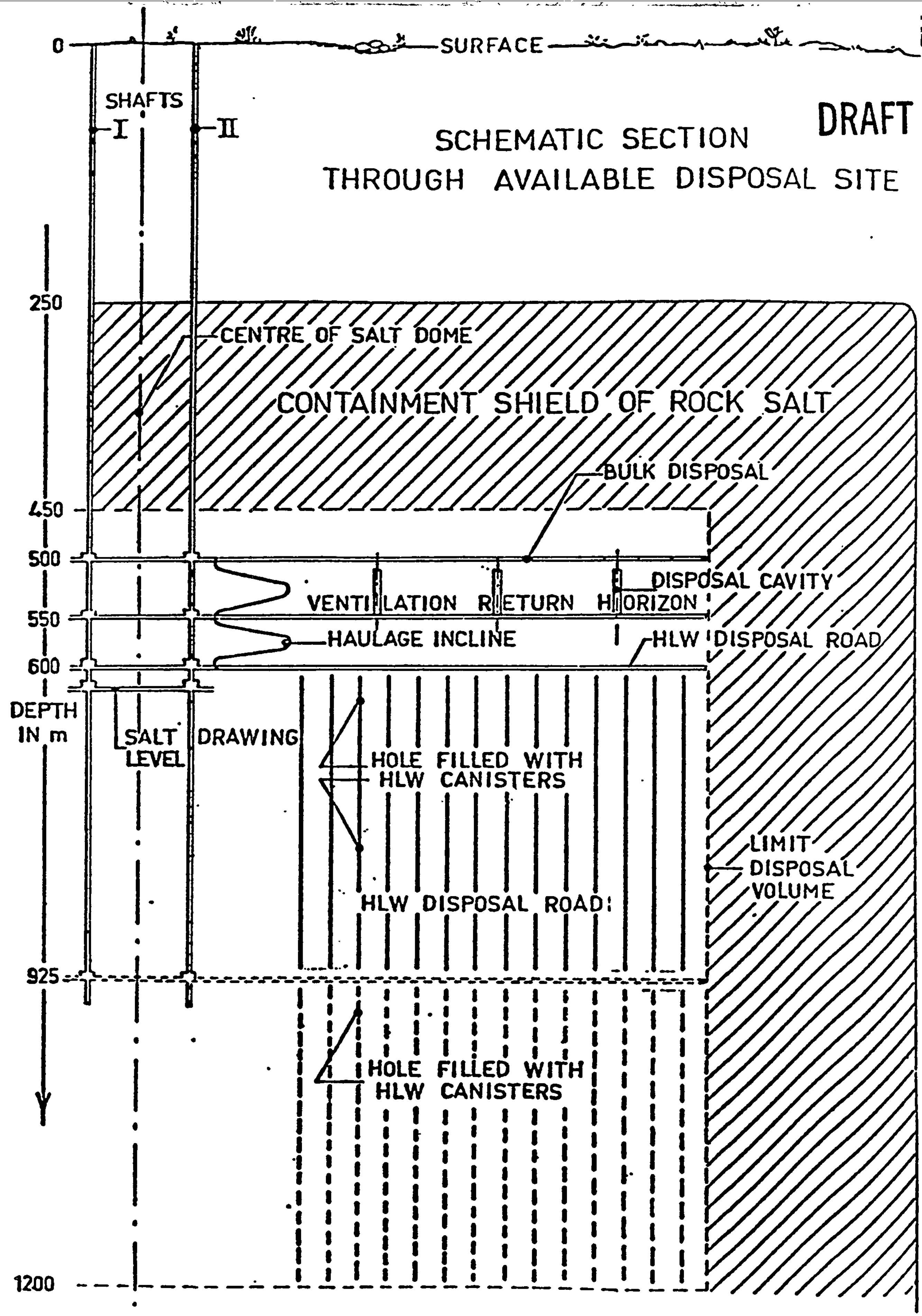

Figure 8.1.2. Section through Repository

250-300 m (820-980 ft), the top level of the repository was located at a depth of about -500 m (-1,640 ft). The disposal caverns for the low- and intermediate-level wastes will be excavated between the -500 and -550 m (-1,640 and -1,800 ft) levels and the -600 m (-1,970 ft) level will provide the access to the top of the vertical disposal boreholes. A small salt handling level will be placed at the -625 m (-2,050 ft) level for the removal of the salt from the excavation and drilling operations. The depth of the disposal boreholes is still under investigation and depends on the technical and geothermal considerations. For this concept, the depth was chosen as approximately 300 m (980 ft). If disposal boreholes are also to be drilled from a second level, the -925 m (-3,030 ft) level will be excavated for this (Figure 8.1.2).

8.1.3.3.1 Shafts. Two shafts will be required for the repository. Construction of the shafts will require special excavation and lining techniques and is described in Section 8.1.4.2. Both shafts will be constructed with a finished diameter of about 5 m (16.4 ft). Shaft 1 will be the intake ventilation shaft and main hoisting shaft for men, material, nuclear waste casks and excavated salt. Shaft 2 will be the exhaust ventilation and emergency egress shaft. Shaft 1 will be equipped with a large cage and counterweight as shown in Figure 8.1.3. The hoist will be a tower mounted Koepe hoist with capacity large enough to handle the 60 ton high-level waste transport cask. Shaft II will be equipped only with an inspection/emergency cage and underground services (power, water, compressed air, etc.).

8.1.3.3.2 Disposal Levels. Figure 8.1.4 shows the configuration near the two shafts. Access to each disposal level will be provided from the shafts. The levels will be connected vertically by a spiral drift and several ore passes for removal of the excavated salt. All excavated salt will be transferred to bunkers at the -625 m (-2,050 ft) level and hoisted from there to the surface. Workshop, offices, etc., will also be provided in the shaft vicinity.

Figures 8.1.5 to 8.1.7 show the layout and configuration of the -500 m (-1,640 ft) level where the low- and intermediate-level waste disposal caverns will be constructed. The disposal caverns will be eight to 12 m (26 to 40 ft) diameter and 50 m (160 ft) high excavations into which the waste containers are dumped. Boreholes will be provided at the top of the cavern for dumping of the canisters, extraction of dust produced during the dumping and backfilling operations. The distance between caverns will be roughly 100 m (330 ft).

Figures 8.1.8 and 8.1.9 show the configuration of the -600 m (-1,970 ft) level where the high-level waste canisters will be placed into boreholes

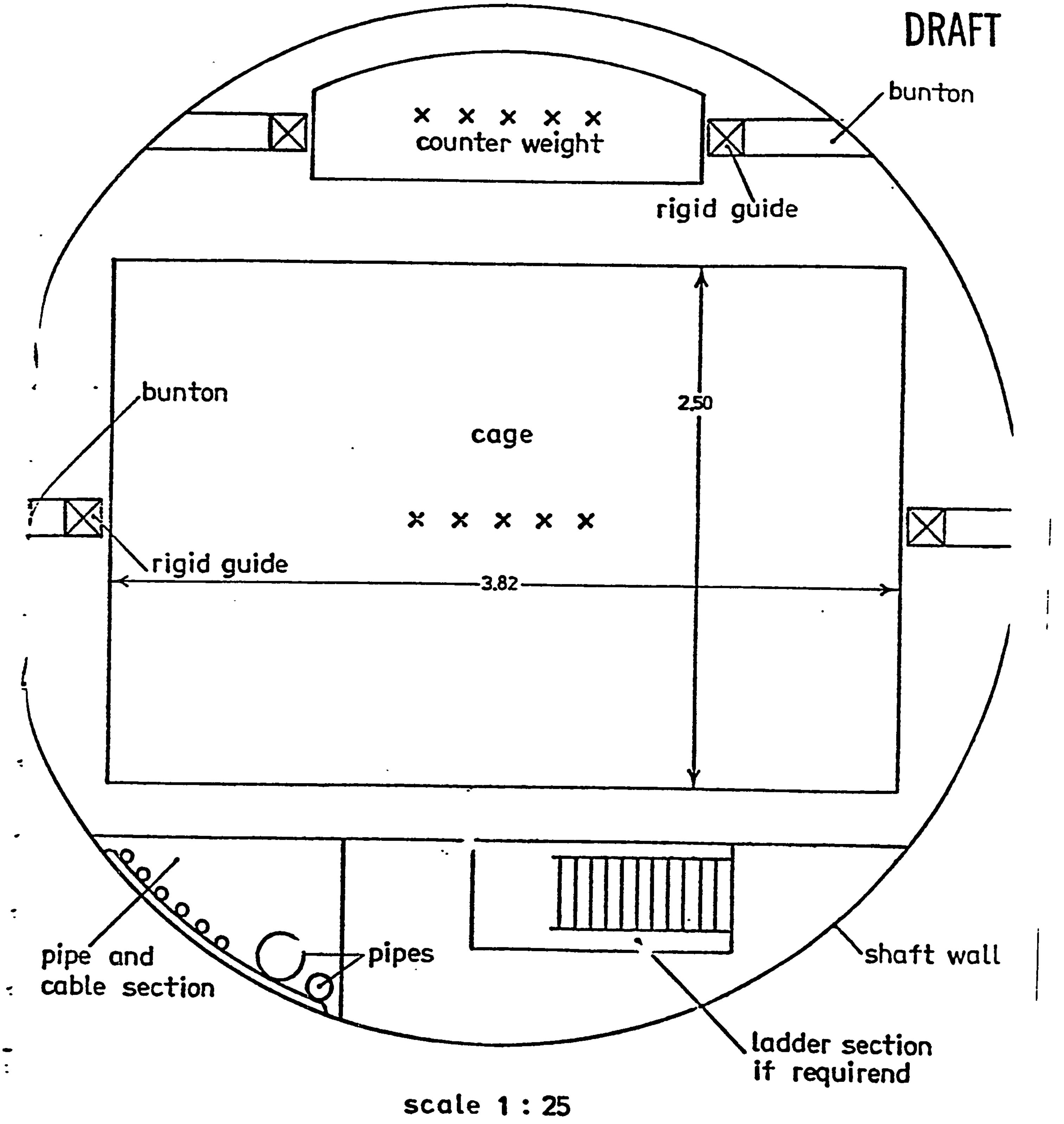

Figure 8.1.3. Shaft Section

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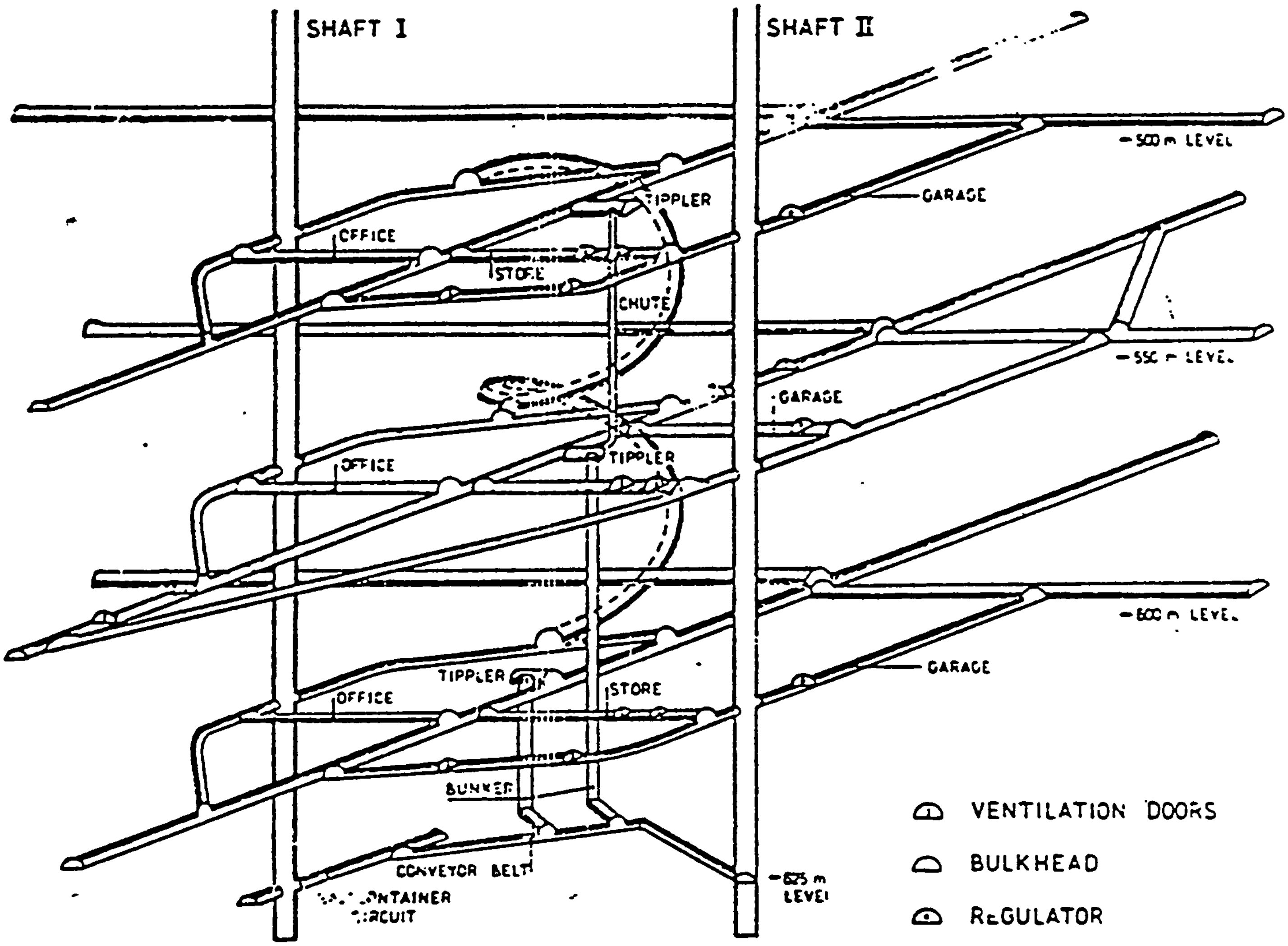

Figure 8.1.4. Shaft Bottom Layout

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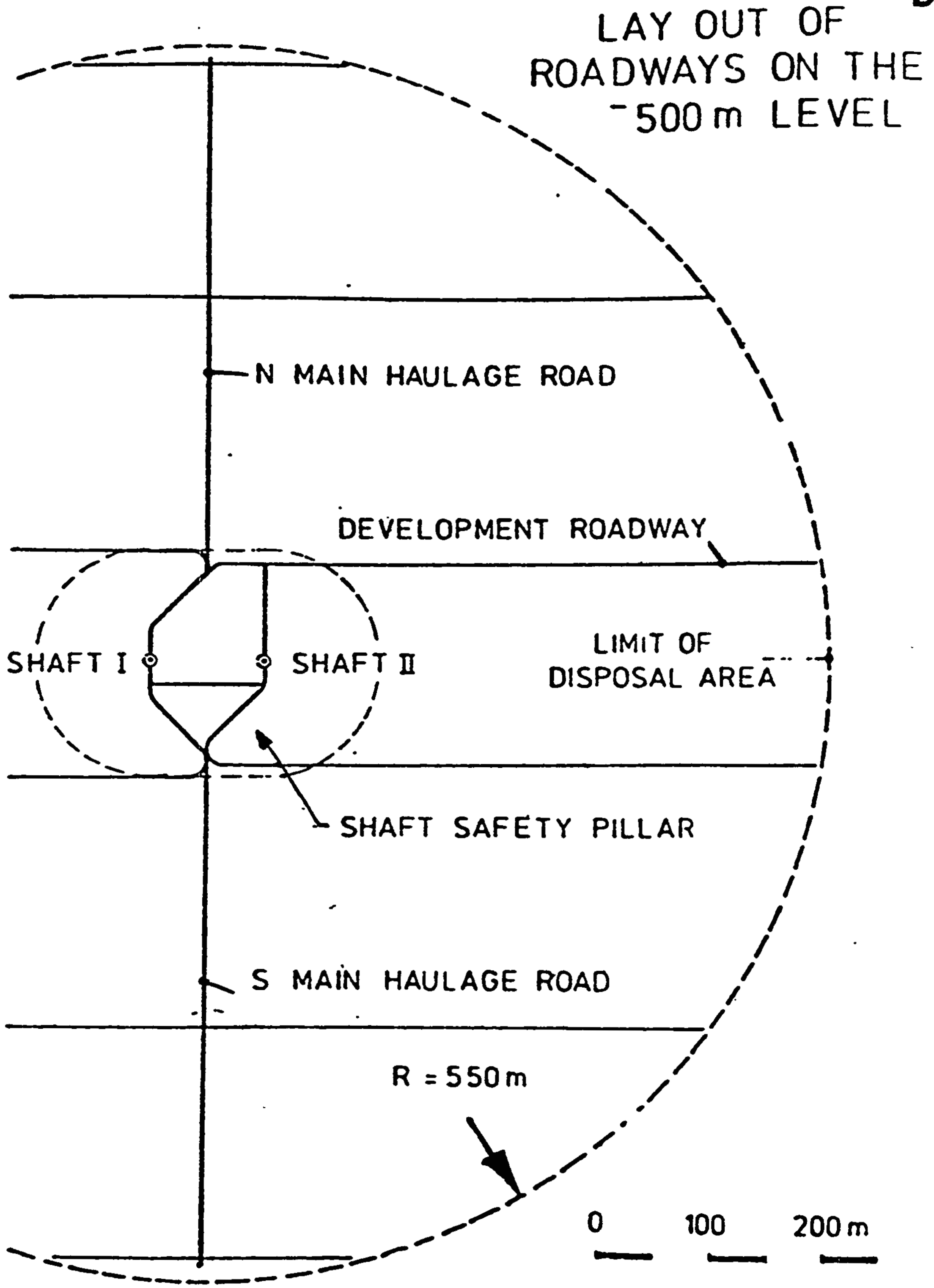

Figure 8.1.5. 500 M Level Layout

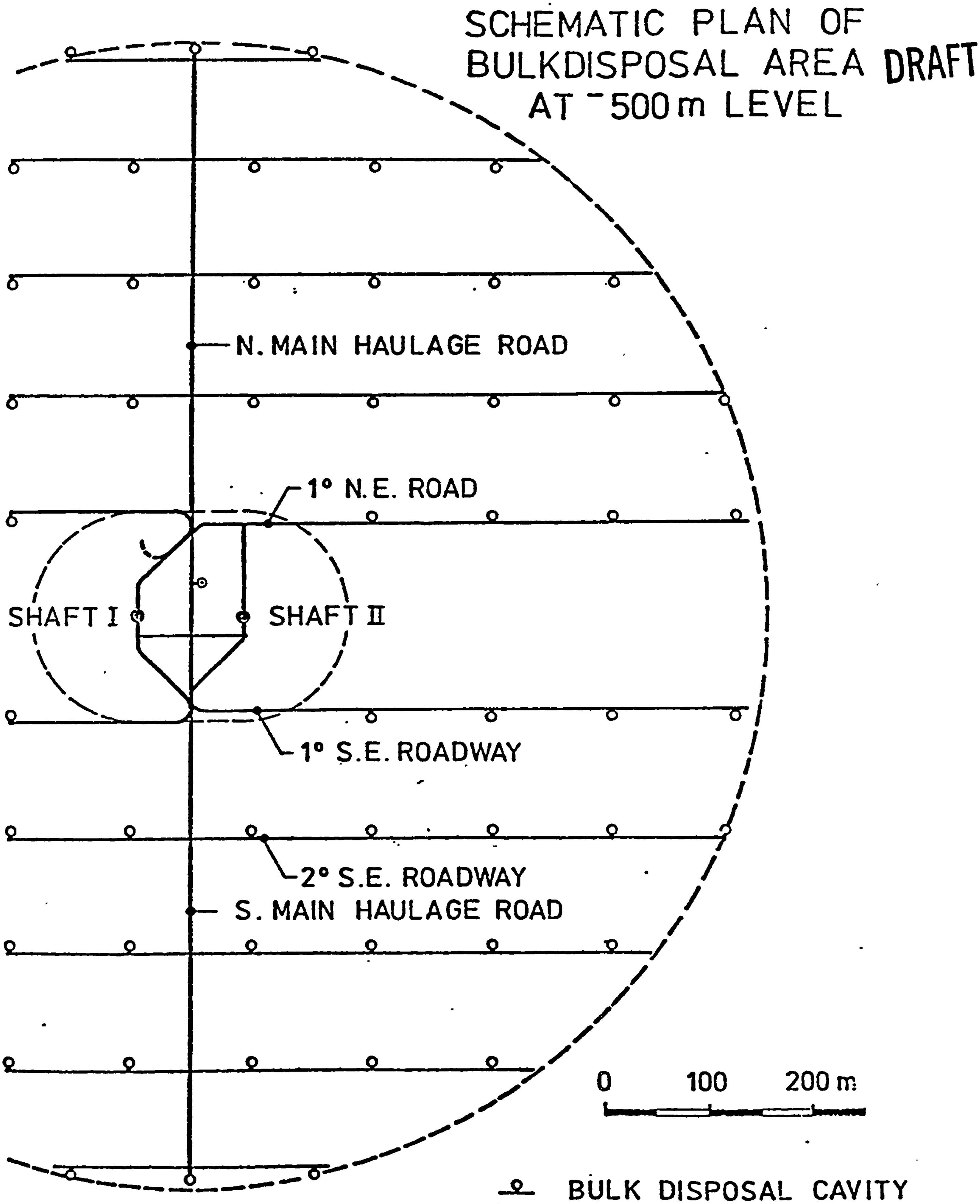

Figure 8.1.6. Bulk Disposal Area

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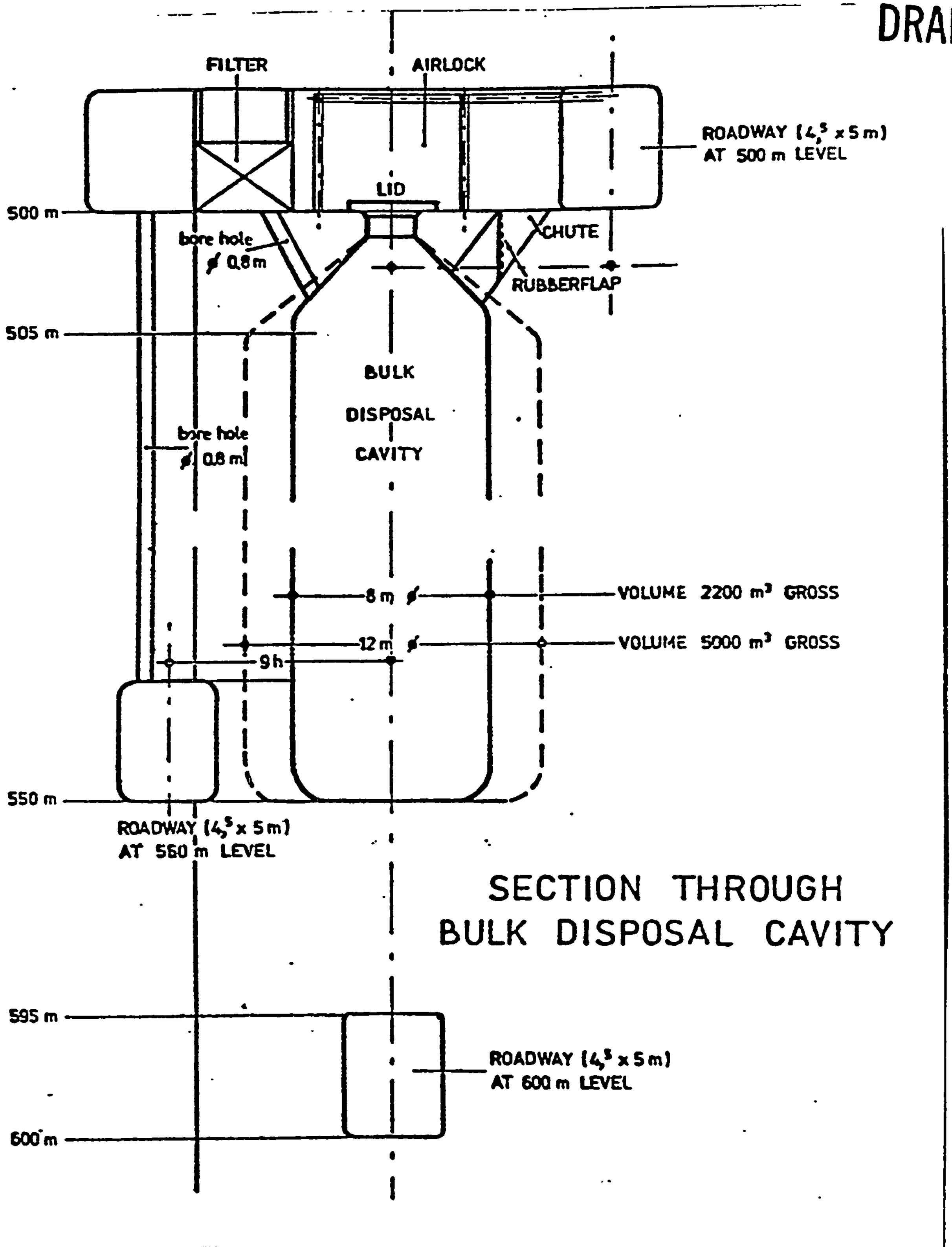

Figure 8.1.7. Bulk Disposal Bunker

SCHEMATIC GROUPING OF HLW DISPOSAL LEVEL

DISPOSAL HOLES CTC 34.5 m

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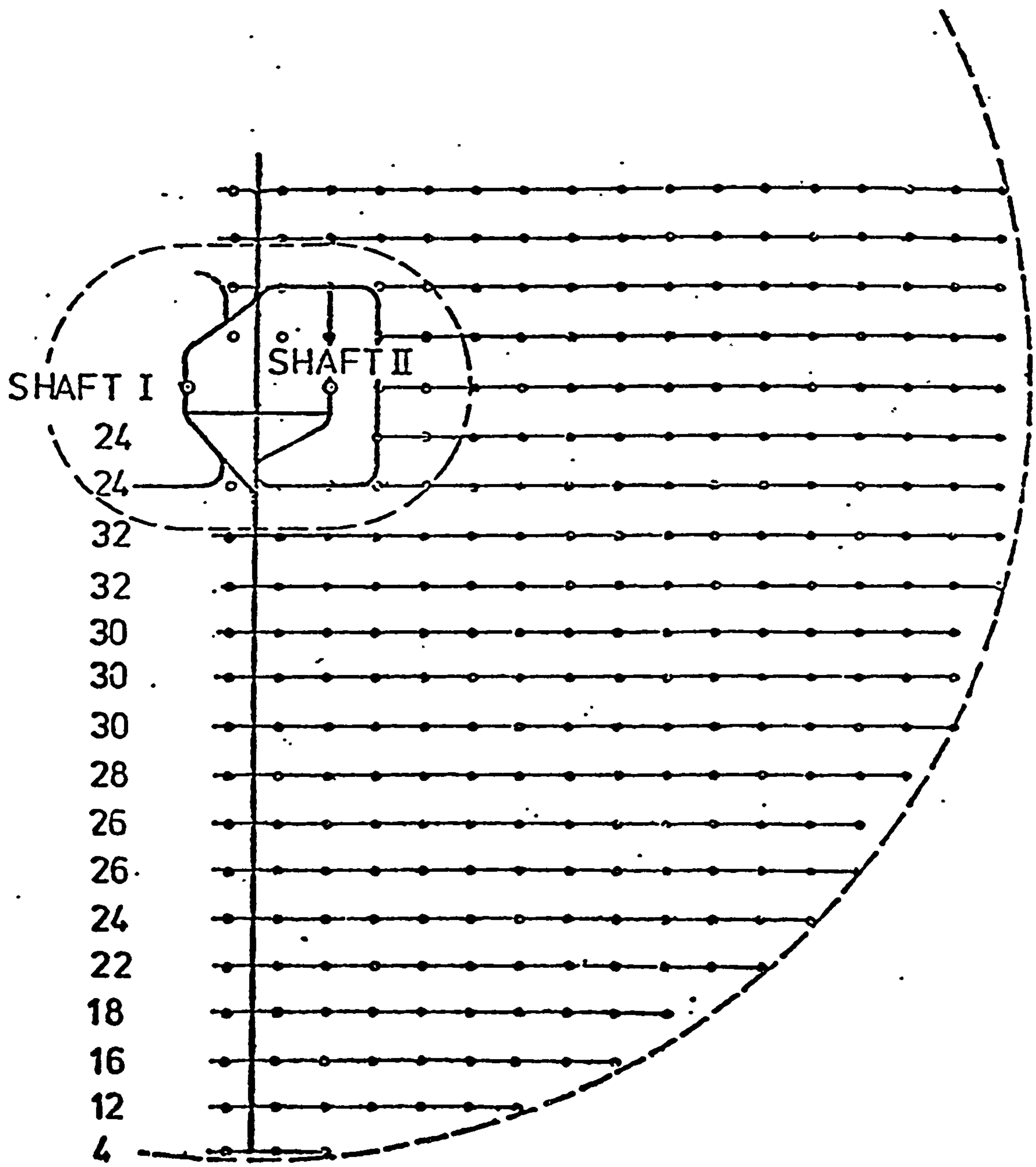

Figure 8.1.8. Disposal Bore Hole Layout

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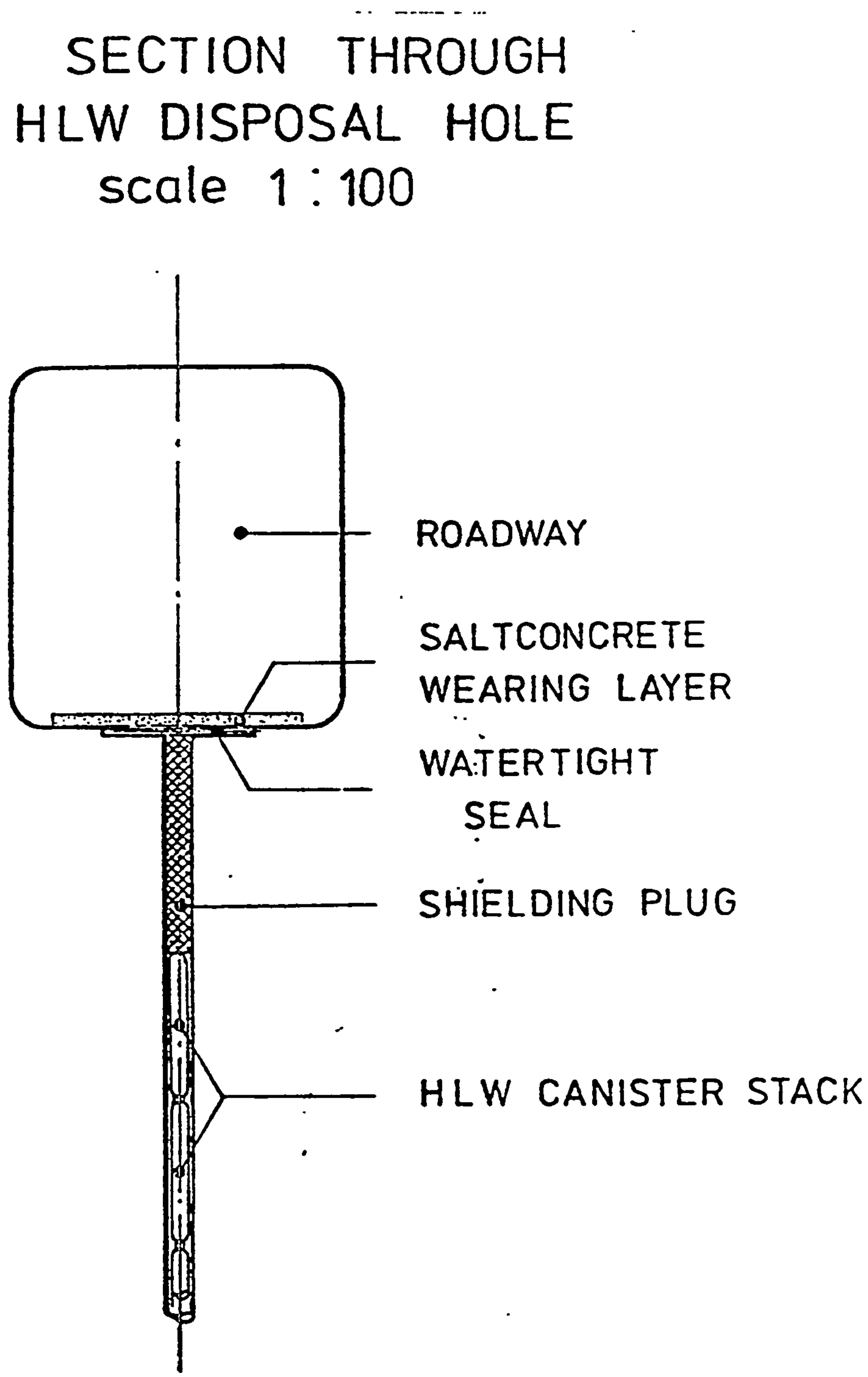

Figure 8.1.9. High-Level Waste Disposal Hole

in the floor of the access drifts. Depth of the holes is not yet defined but was assumed to be about 300 m (980 ft) for this concept. The borehole diameter will be about 300 mm (12 in) and the spacing between holes will be roughly 50 m (160 ft). The repository configuration was assumed to be circular with an overall radius of about 550 m (1,800 ft).

8.1.4 Repository Construction

8.1.4.1 Schedule

An exact schedule for the construction and operation of the repository has not been established as the details of the waste disposal program have not been determined, but it can be expected that the repository will be required in the early part of the next century.

8.1.4.2 Methods and Equipment

This section outlines the construction of the repository shafts and underground development.

8.1.4.2.1 Shafts. Construction and lining of the shafts for the repository is especially critical in the salt environment to prevent any inflows of water in the long-term. As the formations overlying the salt are likely to contain water, the application of the freezing method or the large hole drilling method will be required. Of these, the freezing method is the more favorable since access for inspection and installation of the final, watertight lining is possible. With this method, shaft excavation will take place inside a column of frozen ground to prevent water inflow before the installation of the final shaft lining is complete. Figure 8.1.10 shows the lining for the shafts. Excavation of the shaft will proceed from the surface through the frozen overburden into the competent salt at the top of the dome. The excavation method will depend on the ground conditions although the use of non-blasting techniques is generally preferred to prevent damage to the surrounding shaft wall and also the freeze wall. During sinking, a preliminary lining such as concrete blocks or cast-in-place concrete will be installed to provide shaft wall support until the final lining is in place. Below the frozen zone, in the competent salt, a foundation will be constructed upon which the final watertight lining of welded steel and concrete will be erected. This lining will extend all the way back to the surface. The gap between the preliminary and final linings will be backfilled with bitumen. The density of this bitumen will be controlled such that the pressure of the bitumen column is greater than that of the ground water and thus prevent water migration behind the lining.

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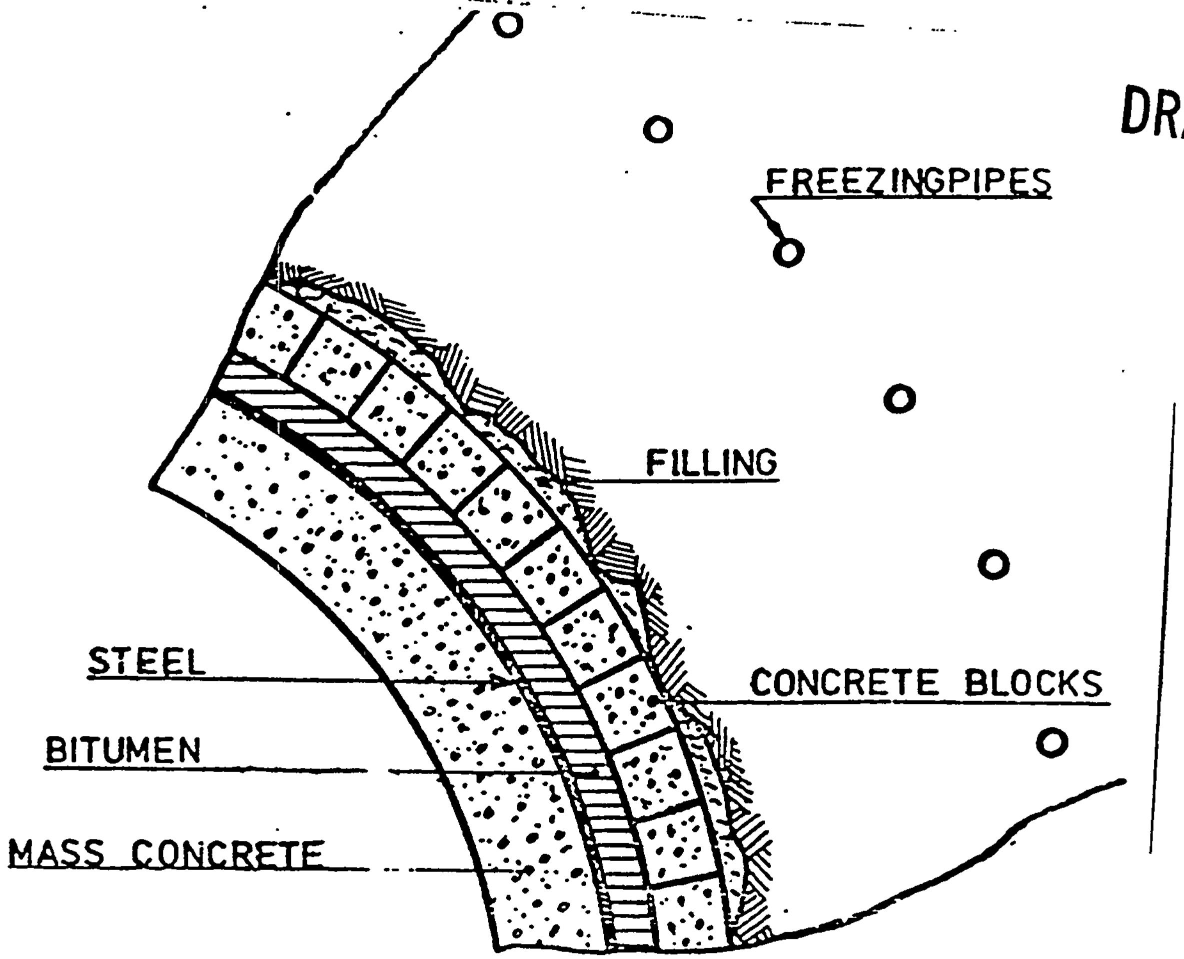
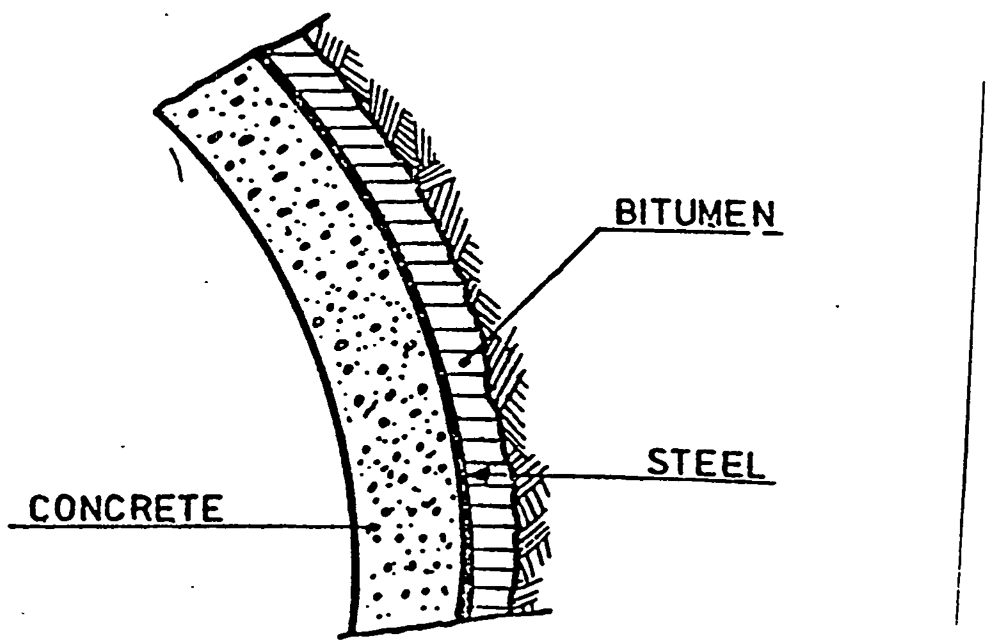


SHAFT LINING FREEZINGSHAFT



SHAFT LINING DRILLED SHAFT

Figure 8.1.10. Shaft Lining

Additional sealing with grout injection may be required in the area of the foundation to ensure a completely watertight system. Once the upper shaft section has been completed and the lining found to be watertight, sinking can continue in the lower salt section to the bottom of the shaft. Below the foundation, in the salt, shaft sinking will continue with the normal drill-and-blast methods. The shaft lining in this lower shaft section will most likely consist of a cast-in-place concrete lining with a thickness of approximately 30 cm (12 in). The shaft stations at the various repository levels will be constructed of reinforced concrete and probably extend about 10 m (33 ft) from the shaft. At the completion of shaft sinking, the shafts will be furnished with the required conveyances and services.

8.1.4.2.2 Disposal Levels. Figures 8.1.4 to 8.1.9 show the proposed development in the shaft vicinity and on the various disposal levels. The access drifts to the boundary of the repository will be excavated using continuous mining machinery such as a Marietta Miner. The 4 m by 5 m high (13 by 16.5 ft) drifts will most likely be excavated in two stages with the top half being excavated first. During the excavation of this section, precautionary measures will be used to detect abnormalities such as brine or gas pockets and changes in the rock properties ahead of the excavation face. These measures may include exploratory drilling in stages ahead of the face, core drilling and seismic investigation of the salt mass. Once the repository area has been characterized and excavated with these exploratory methods, the drifts can be enlarged to their final size without the use of the precautionary measures. Haulage of the excavated salt to the shaft or ore pass may be done using diesel equipment such as loaders or trucks. It is proposed to use haulage equipment for the development of the levels which can also be used for the haulage of the radioactive waste containers. The salt loading pocket will be located at the -625 m (-2,050 ft) level and will be filled via ore passes from the various levels.

8.1.4.2.3 Disposal Caverns. The disposal caverns or bunkers for the low- and intermediate-level wastes will be excavated between the -500 m and -550 m (-1,640 and -1,800 ft) levels (Figure 8.1.7). A short access drift will be excavated from the -550 m (-1,800 ft) level to the base of the proposed cavern. A raise borehole will connect the two levels and will be enlarged to form the disposal bunker as shown in Figure 8.1.7. Short boreholes will be drilled for the dump chute and the air extraction system. After construction of the disposal bunker, the drift at the base will be sealed again to form the closed disposal cavern. It is expected that one or two bunkers will be in use for disposal and one or two will be under construction at the same time.

8.1.4.2.4 Disposal Boreholes. The high-level wastes will be placed in vertical boreholes drilled into the floor of the drifts on the -600 m (-1,970 ft) level (Figures 8.1.8 and 8.1.9). If required, the same configuration will be repeated at the -900 m (-2,950 ft) level to give a second high-level waste disposal level. The boreholes are expected to be about 300 mm (12 in) in diameter and 300 m (980 ft) deep. A steel liner will be concreted into each borehole and bonded to a steel and concrete collar at the top of the hole. The holes will be drilled into the floor in the center of the drifts about 50 m (160 ft) apart. The exact configuration, depth and spacing of the holes and drifts will be determined by the allowable heat load in the salt body and the salt properties. The drilling of some of the initial boreholes such as those near the periphery of the repository where exploration to the depth of the boreholes has not been carried out may require the use of safety precautions such as small pilot holes and blowout preventers.

8.1.4.3 Cost

Information on the cost of the repository construction and operation is not available in the literature at this point.

8.1.5 Waste Emplacement Cycle

8.1.5.1 Waste Treatment

The Netherlands currently have reprocessing contracts with France and England and expect to dispose of the high-level wastes in Holland. The option of direct disposal of the spent fuel is also under consideration.

8.1.5.2 Waste Transport

Transport of the radioactive wastes will very likely be similar to the systems described for other waste programs (e.g., Sweden) using special shielded transport containers and vehicles.

8.1.5.3 Waste Package

Detailed information on the waste packages was not available at the time of review.

8.1.5.4 Underground Waste Emplacement Sequence

A container system is proposed for the haulage of the waste packages. These containers will be transported using the same vehicles as those used to haul the salt transport containers. The low- and intermediate-level waste packages will be transported in containers weighing about 16 tons. The packages will be dumped into the disposal bunkers and the voids backfilled with salt as much as possible. When the bunker is full, all access to the bunker at the top is backfilled with salt concrete. Disposal will start with the bunkers furthest from the shafts and retreat back towards the shafts, sealing and backfilling the access drifts as the bunkers are filled.

The high-level waste packages will be transported in heavy shielding containers (about 60 tons) to the borehole. About seven high-level waste packages will be transported in a container. Once at the borehole, the container will be located over the hole and radiation shields will be put up to allow lowering of the waste packages one at a time. The waste packages can be lowered into the borehole by simple free fall or using a wire line. With the free fall method, the small annulus between the package and the borehole wall is expected to produce an air buffer below the waste package which should retard the speed of descent. However, as an additional buffer, a layer of salt between the packages or a deformation cap on the packages may be needed to minimize the effect of the falling package. The addition of these buffers will reduce the storage capacity of the borehole but emplacement time will be reduced over the use of the wire line for lowering the packages. The application of a particular disposal method will have to be determined during final design stages. At a suitable distance from the top of the borehole, waste package disposal will cease and the disposal hole sealed to the top with salt, clay dust and fly ash and a final plug of granulate bentonite. A final cover of salt concrete and steel will then be placed over the borehole.

Some of the intermediate-level wastes with high radiation activity may also be placed in boreholes in a manner similar to the high-level wastes. These boreholes could be drilled as part of the high-level waste drilling program and could be interspersed among the high-level boreholes as the heat production from these intermediate-level wastes is negligible.

8.1.5.5 Emplacement Schedule

No detailed schedule was found in the literature for the waste disposal program but it is likely that the emplacement of the wastes will last for several decades. Based on a 2.5 GWe nuclear power capacity, the waste production was estimated at about 4,000 m^3 (5,200 yd^3) per year in one report.

8.1.6 Buffer and Sealing Materials

Buffer and sealing material details will be determined during final design stages but for the concept described in this section, the following materials were proposed as buffer and sealing materials:

1. In the low- and intermediate-level bunkers, salt mixed with fly ash may be used to seal voids as much as possible. The ventilation borehole and dump chute will be filled with salt concrete as well as the access drifts. Clay dust may be added to the salt concrete in the disposal areas to ensure retardation of the radionuclides in the unlikely event that water would find its way into the disposal area.

2. The top section of the high-level disposal boreholes will be backfilled and sealed with salt, clay dust and fly ash, with a plug of bentonite and a slat concrete and steel cover. Backfilling the annulus around the canisters with bentonite powder to provide protection against brine migration is also under consideration.

3. After filling all the disposal bunkers and boreholes, the access drifts will be backfilled and sealed using salt concrete and clays dust, possibly also mixed with fly ash. The vertical ventilation borehole, spiral access drifts, salt bunkers, etc., will have a watertight bulkhead at each end to prevent water flow along these paths in the unlikely event of water inflows. The bulkheads will be constructed of reinforced concrete with bitumen and grouting used for sealing.

4. The shafts will be filled up to the caprock with a compacted salt, fly ash and clay dust mixture with bitumen for filling the voids. In the caprock area and transition zone to the overburden, a asphalt concrete plug topped by a layer of bitumen may be used. The rest of the shaft will then be filled to the surface with a sand and clay mixture. All shaft furnishings will be removed before backfilling.

8.1.7 Summary

8.1.7.1 Advantages

1. Salt is considered by many countries to be one of the better host rocks for nuclear waste disposal due to its low water content and hydraulic conductivity, its great abundance in relatively homogeneous deposits, its high thermal conductivity, its self-healing properties and low mining costs.

2. A large amount of expertise in the mining of salt exists in the world and the mining technology is highly developed.

3. The massive salt domes can provide a large barrier around the whole repository.

4. Many research programs in salt (such as Asse or WIPP) have greatly expanded the understanding of possible the behavior of salt under repository conditions.

8.1.7.2 Potential Problems

1. Although salt is generally considered a favorable repository host rock, several negative factors are associated with disposal in salt. The salt is highly soluble, has poor sorptive characteristics and brine solutions are very corrosive.

2. Salt deposits are often overlain by massive aquifers (e.g., sites in Gorleben, West Germany and Deaf Smith County, Texas) and, therefore, require highly specialized shaft sinking techniques. Ground freezing will be used at the two examples quotes above. Also, the shaft lining will have to be watertight and prevent water migration behind the lining into the salt body. The lining proposed at the two sites above will consist of welded steel and concrete with a bitumen backfill to seal out the ground water. The technology to sink shafts with these difficult ground conditions has been used successfully in several countries (Canada, England, West Germany), but a certain amount of risk is always associated with this application.

3. Postclosure seals will have to ensure that no water can penetrate the salt body for a long period after disposal. The location and construction of these will require careful evaluation.

4. The heat build-up in the repository may cause some movement of the salt dome formation. Additional aging of the heat producing wastes may reduce this problem.

5. Salt formations are often associated with valuable deposits of hydrocarbons, potash, and other exploitable resources.

8.1.7.3 Summary

The salt formations are considered one of the best repository host rock formations despite the potential problems noted above. In fact, most countries with suitable salt formations are proposing them as potential host rocks rather than other formations. The United States is currently actively investigating the bedded salt formations in the Deaf Smith County, Texas, area. A proposed repository concept for this area is described in Section 8.4.

8.2 SALT: THE DANISH DISPOSAL CONCEPT

There are no nuclear power plants operating in Denmark at present, but plans are to bring six 1,000 MWe reactors on line in the 1990s and operate them for about 25 years. A two-phase study was initiated in the 1970s to study the feasibility of high-level radioactive waste disposal in Denmark. Phase 1 investigated the waste practices and programs. Phase 2 was completed in 1981 and included investigation of a number of sites in northern Jutland and some site investigation work at two of the sites. Two disposal concepts are being investigated - the deep borehole concept and the mined repository concept. Both concepts are to be located in a salt dome. This chapter briefly outlines these high-level waste disposal concepts proposed for the Danish nuclear power program.

8.2.1 Geographic Location

The salt domes investigated in the waste disposal study are located in the northern part of Denmark. Figure 8.2.1 shows the location of the salt deposits. The salt domes at Mors and Linde were investigated in some detail and formed the basis for the high-level waste repository design study.

8.2.2 Summary of Geologic Setting

The salt domes were created by the uplift and penetration of salt formations into the overlying strata. The Mors salt dome (Figure 8.2.1) was used as the reference formation and is mushroom-shaped and almost circular with a diameter of about 8 km (5 mi). The base of the salt dome lies at a depth of about 5,500 m (18,000 ft) and the top of the dome is about 600 m (1,970 ft) below surface. The sediments above the anhydrite caprock consists of clays, sands and limestones. Two deep holes were drilled to about 3,400 m (11,150 ft) and they, as well as geophysical investigations of the area, confirmed the existence of a large area of pure sodium chloride with only traces of anhydrite at the repository depth of about 2,500 m (8,200 ft). This was the area used in the repository safety and design study.

8.2.3 Repository Concept Description

The concepts described in this section are based on a waste output from a reactor program consisting of six reactors of about 1,000 MWe each to

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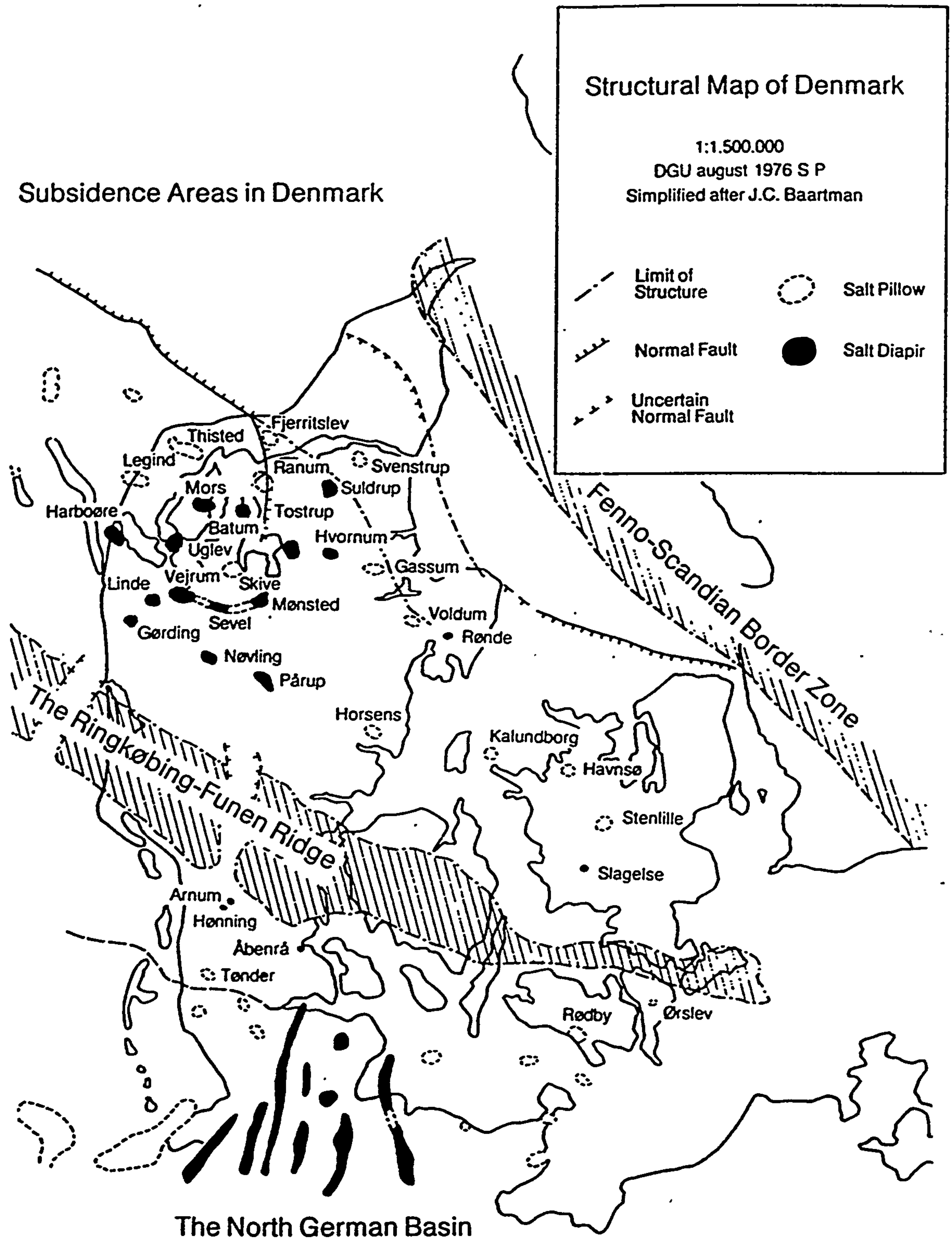

Figure 8.2.1. Salt Domes in Denmark

become operational at roughly two-year intervals in the 1990s. The reactor life was assumed to be about 25 years with a total fuel consumption of about 4,800 MT (5,280 tons).

8.2.3.1 Nuclear Waste Disposal Strategy

For the high-level waste disposal study it was assumed that all spent fuel would be reprocessed at facilities in other countries with the high-level wastes being returned to Denmark for final disposal. The spent fuel is expected to be reprocessed 10 years after removal from the reactor. The high-level wastes will be vitrified in borosilicate glass and the glass cast in stainless steel canisters. Prior to their disposal in the repository, the waste canisters will be stored in an interim storage facility for about 30 years to allow the heat and radioactivity output to diminish. Disposal of the wastes would therefore not be required until about 40 years after removal of the spent fuel from the reactor. Consequently, disposal is not expected to commence before about 2040 at the earliest, by which time the disposal technology should have been demonstrated on a large scale by programs in other countries. It was thus assumed that the repository would not have to be designed to allow retrieval of the waste canisters.

Two concepts were developed for the repository. One is the deep-borehole concept where the canisters are placed in boreholes from the surface to a depth of about 2,500 m (8,200 ft), and the other concept was based on a mined underground facility at a depth of about 1,000 m (3,300 ft) where the waste canisters would be placed in horizontal disposal drifts. Both concepts are described in the following sections.

8.2.3.2 Surface Facilities

The main surface facilities will include the interim storage facility and the surface installations at the repository site. The interim storage facility has not yet been detailed, but various alternatives such as surface or underground storage with water or air-cooling are being considered in other countries and one of these will most likely be adapted to suit the Danish program.

Some of the surface installations at the repository will vary, depending on which disposal method will be chosen. For the mined repository, the usual installations such as headframes, hoists, ventilation fans, underground services, salt and rock storage areas, etc., will be required. For the deep-borehole concept, these permanent installations would not be necessary, but facilities such as mud pits, settling ponds and a large hole drill rig would be installed for the drilling phase. During the emplacement phase, the drill rig or a winch and derrick system will be used to lower canisters into the boreholes. The waste receiving and preparation facilities may also vary since the canisters will have to be encased in shielded transport containers before being taken underground at the mined facility. Receiving and handling at the deep-borehole disposal facility would probably be simpler since the waste canisters do not require such extensive handling.

8.2.3.3 Underground Configuration

8.2.3.3.1 Mined Underground Facility. The earliest concepts proposed disposal in boreholes drilled into the floor of the drifts but drilling trials at the Asse Salt Mine in Germany highlighted some problems with this method, and so it was changed to the system described here where the canisters are placed in horizontal disposal drifts. The problems associated with the drilling of the disposal holes related mainly to lack of room in the access drifts for the drill rig, dust prevention, and removal of the cuttings from the drill bit and subsequent transport up the borehole during drilling.

The mined underground repository consists of two shafts and the underground access and disposal drifts. Figure 8.2.2 shows the proposed facility. The main access shaft will be about 5 m (16.4 ft) and the ventilation shaft about 4.5 m (14.7 ft) in diameter. The shafts will be spaced about 80 m (260 ft) apart. The depth of the shafts will be about 1,000 m (3,300 ft). The main access drifts will be about 1,100 m (3,600 ft) long and roughly 7 m wide by 3.5 m high (23 ft by 11.5 ft). The disposal drifts excavated off the main access drifts will be about 500 m (1,640 ft) long and 5 m wide by 3.5 m high (16.4 ft by 11.5 ft). About 24 drifts will be required and the spacing between drifts will be about 100 m (330 ft). The waste canisters will be placed in pipes installed in the disposal drifts.

8.2.3.3.2 Deep Borehole Facility. The deep borehole facility (Figure 8.2.3) consists of eight holes drilled from the surface to a depth of about 2,500 m (8,200 ft). The holes will be located on a 500 m (1,640 ft) diameter circle and spaced about 200 m (660 ft) apart. The hole diameter will be about 75 cm (2.5 ft). The circular pattern for the hole locations was chosen to simplify the calculations of the movement of the salt in the dome. The disposal zone in each hole will extend from a depth of about 1,200 m (3,940 ft) to about 2,500 m (8,200 ft).

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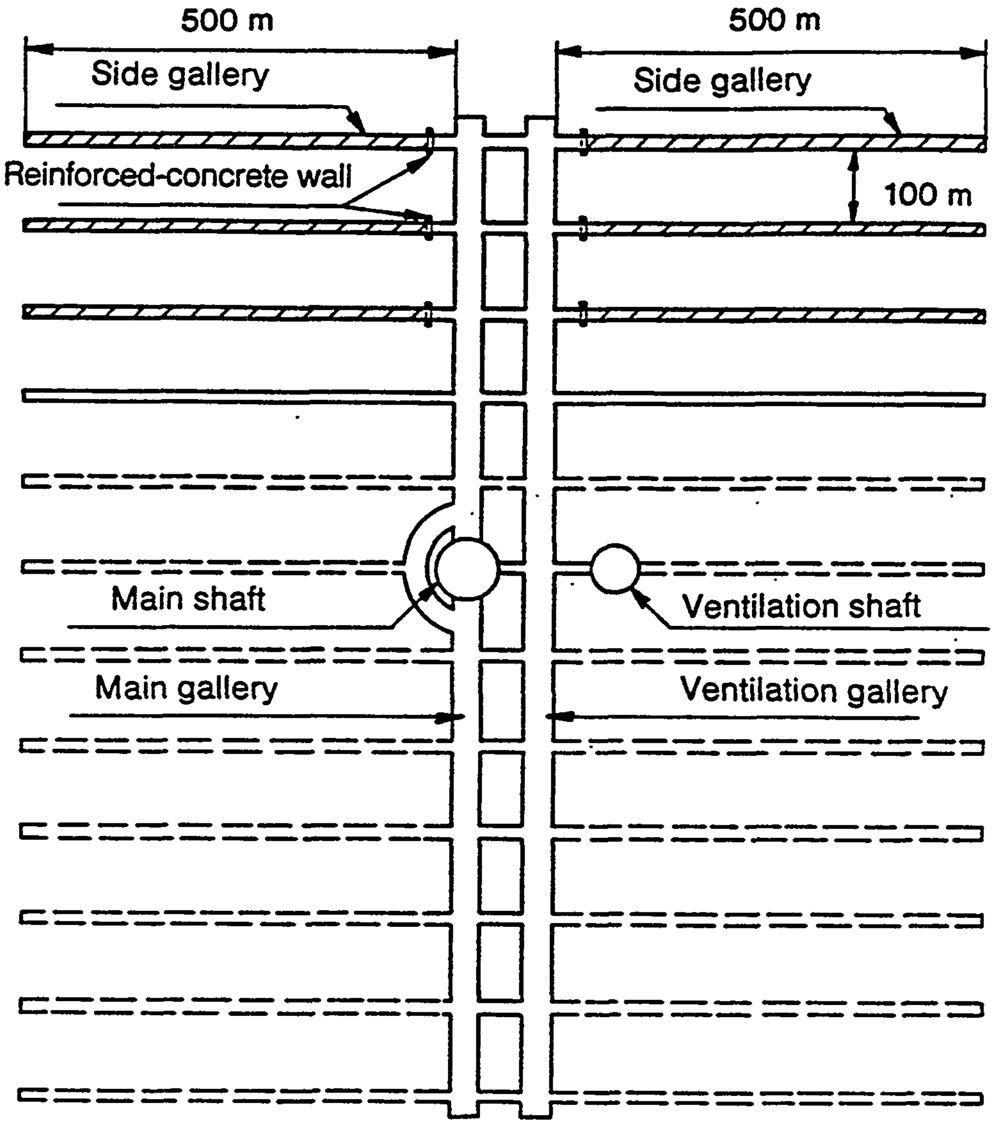

Figure 8.2.2. Repository Layout

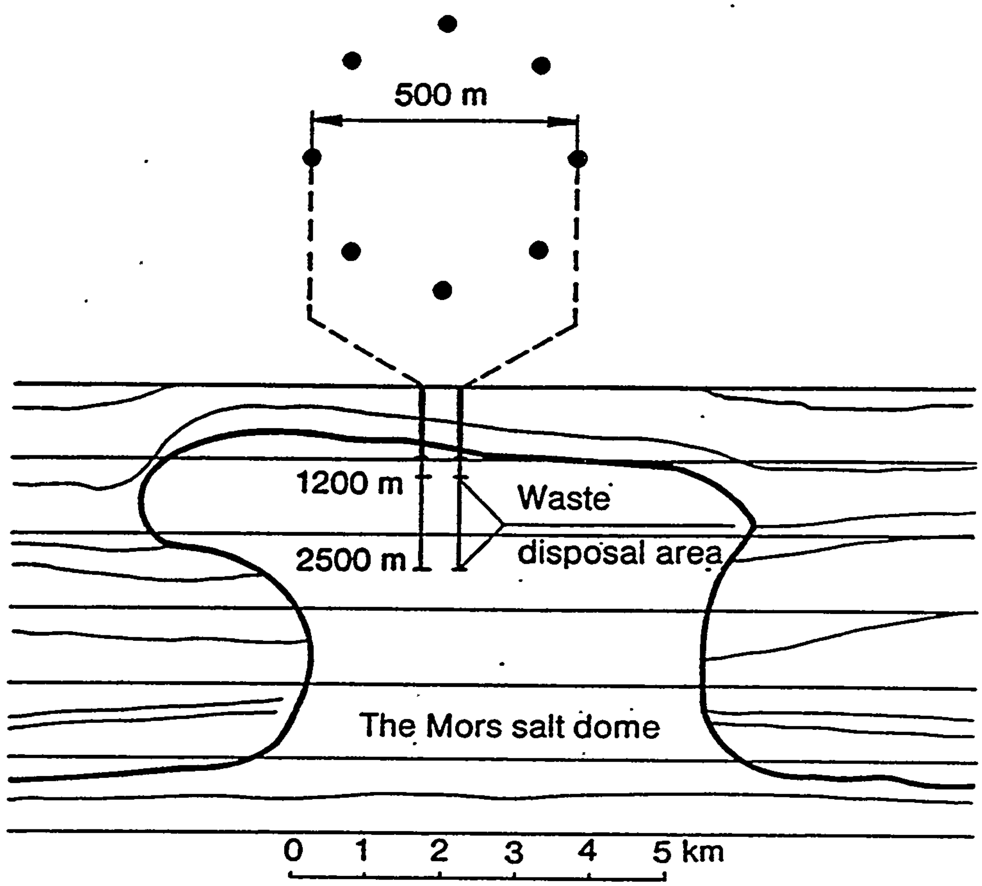

Figure 8.2.3. Deep Hole Disposal Facility

8.2.4 Repository Construction

8.2.4.1 Schedule

Waste disposal is envisioned to start around the year 2040 and thus site investigation and repository construction activities will have to start at least 10 to 20 years before then to ensure that the repository will be ready to accept waste as scheduled. Waste disposal is expected to last for about 25 years.

8.2.4.2 Methods and Equipment

8.2.4.2.1 Mined Underground Facility. Since the upper 735 m (2,410 ft) of the shafts have to penetrate water-bearing and unconsolidated formations before reaching the top of the salt, special shaft sinking and lining techniques will be required. The method proposed for the repository shaft sinking is the ground freezing method, followed by the installation of a watertight steel and concrete lining. Figure 8.2.4 shows the proposed shaft lining for the upper 750 m (2,460 ft). The ground around the shafts will be frozen before excavation and final lining takes place. During excavation a preliminary concrete lining will be placed to support the shaft wall during sinking. Sinking will stop when the shaft reaches the top of the Cretaceous formations and installation of the final lining back to the surface will commence. The watertight lining will be installed on top of a foundation near the top of the Cretaceous strata and will consist of a welded steel and concrete construction with chemical seals installed at strategic locations to prevent water migration behind the lining. Backfill grout will be injected into the gap between the steel lining and preliminary concrete. A similar sequence will be used to excavate and line the shaft to the 750 m (2,400 ft) level to the top of the salt with seals above the lining foundation and in the transition zone between the limestone and caprock to seal the water from the salt formation. A similar lining system is also proposed for the Dutch, German and United States salt repository programs and are described in Sections 8.1, 8.3 and 8.4, respectively. These programs have, however, also propose a preliminary lining of concrete blocks rather than cast-in-place concrete and a backfill of bitumen rather than grout to act as water seal behind the steel liner. Future revisions to the Danish program may also consider this option. The lower shaft section from 750 m to 1,000 m (2,400 to 3,300 ft) in the salt will probably be lined only with rock bolts and wire mesh as experience in other countries has shown that salt has good mechanical properties.

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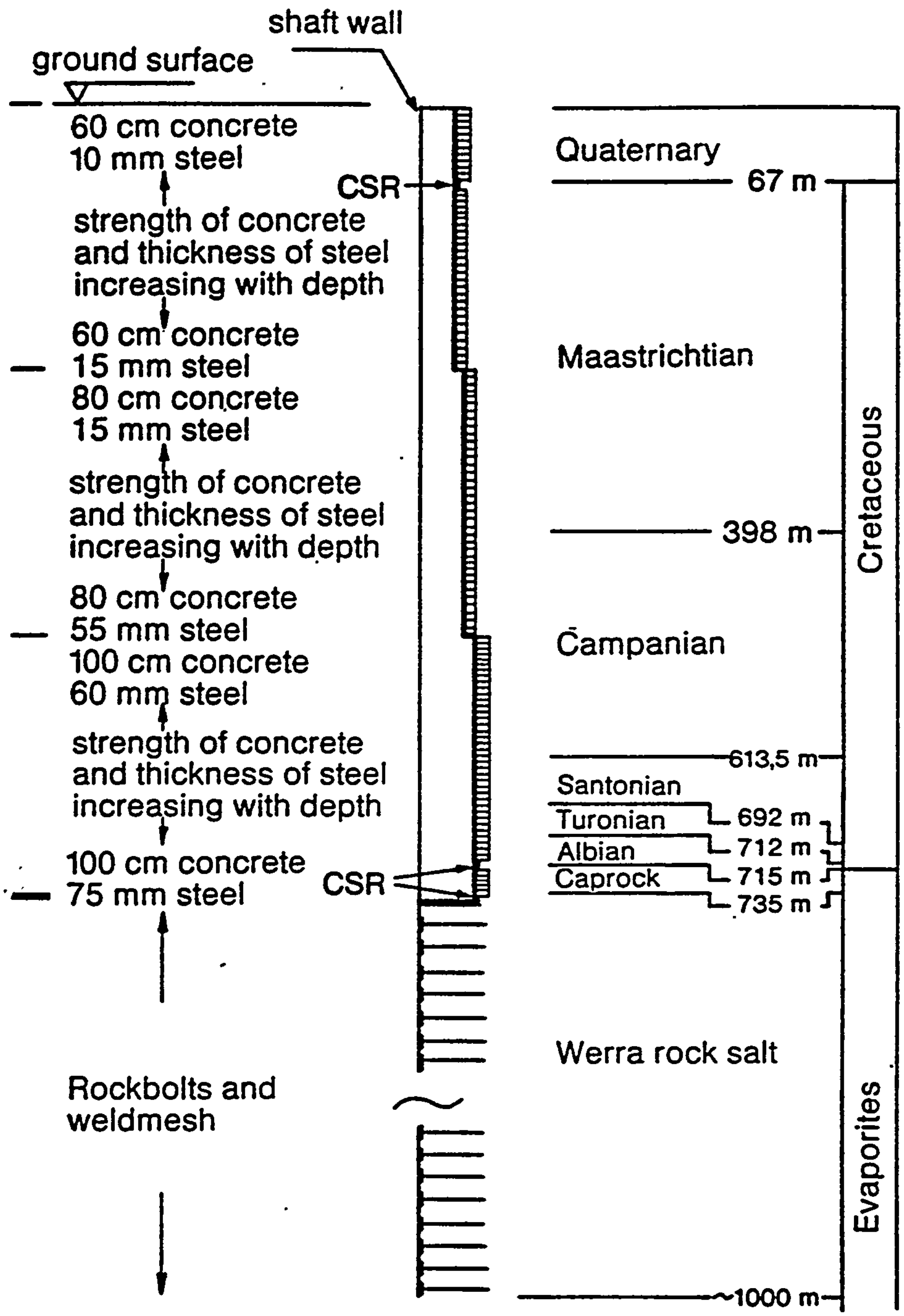

Figure 8.2.4. Shaft Construction

Excavation of the repository access and disposal drifts will be carried out using both drill-and-blast and mechanical methods. The disposal drifts would be suitable for excavation with a roadheader with loaders and trucks for loading and haulage of the excavated salt.

8.2.4.2.2 Deep Borehole Facility. Each borehole will have to penetrate the same water-bearing and unconsolidated formations described above for the shafts and thus special drilling and casing techniques will also be required for this type of facility. Figures 8.2.5 to 8.2.7 show the drilling and casing methods proposed for the deep borehole construction. The first stage will be the installation of the conductor casing to prevent collapse of the near-surface formations. The depth of this casing will be about 30 m (100 ft) with a diameter of about 75.5 in. The casing can be installed in an augered hole if the ground is stable enough or it can be pressed into the ground while removing the earth from the inside. A 73 in diameter hole will then be drilled inside this conductor casing to about 200 m (660 ft) depth in one pass using the reverse circulation method illustrated in Figure 8.2.6. The borehole will be full of drilling mud to transport cuttings up to the surface and also to support the borehole walls and prevent water inflows. A 52 in casing will then be installed and the gap around the casing backfilled with cement. Figure 8.2.7 illustrates a method of cementing a casing where the cement is injected at the base of the casing and then eventually displaces all the drilling mud as the gap fills to the surface. A similar process will be used to construct the next section of borehole down to 950 m (3,100 ft). A 32 in casing will be installed in a 48 in drill hole so that a completely cased borehole will exist about 250 m (800 ft) into the salt body and thereby isolate the upper water-bearing formations from the salt. The lower section of the disposal hole down to 2,500 m (8,200 ft) will then be drilled by reaming a 17.5 in pilot hole to the final borehole diameter of 29.5 in. No casing will be installed in this lower portion. Finally, the drill mud will be replaced with brine to facilitate lowering of the waste containers (see Section 8.2.5.4.2).

8.2.4.3 Cost

A cost estimate was prepared in 1981 for the construction of the mined underground repository and the deep borehole facility. The cost for construction and sealing of the mined repository was estimated to about $150 million with an operational cost of about $5 million per year for a total cost of approximately $275 million. The construction of each borehole for the deep hole facility was estimated to cost $20 million with a sealing and operating cost of about $3 million for a total cost of about $190 million.

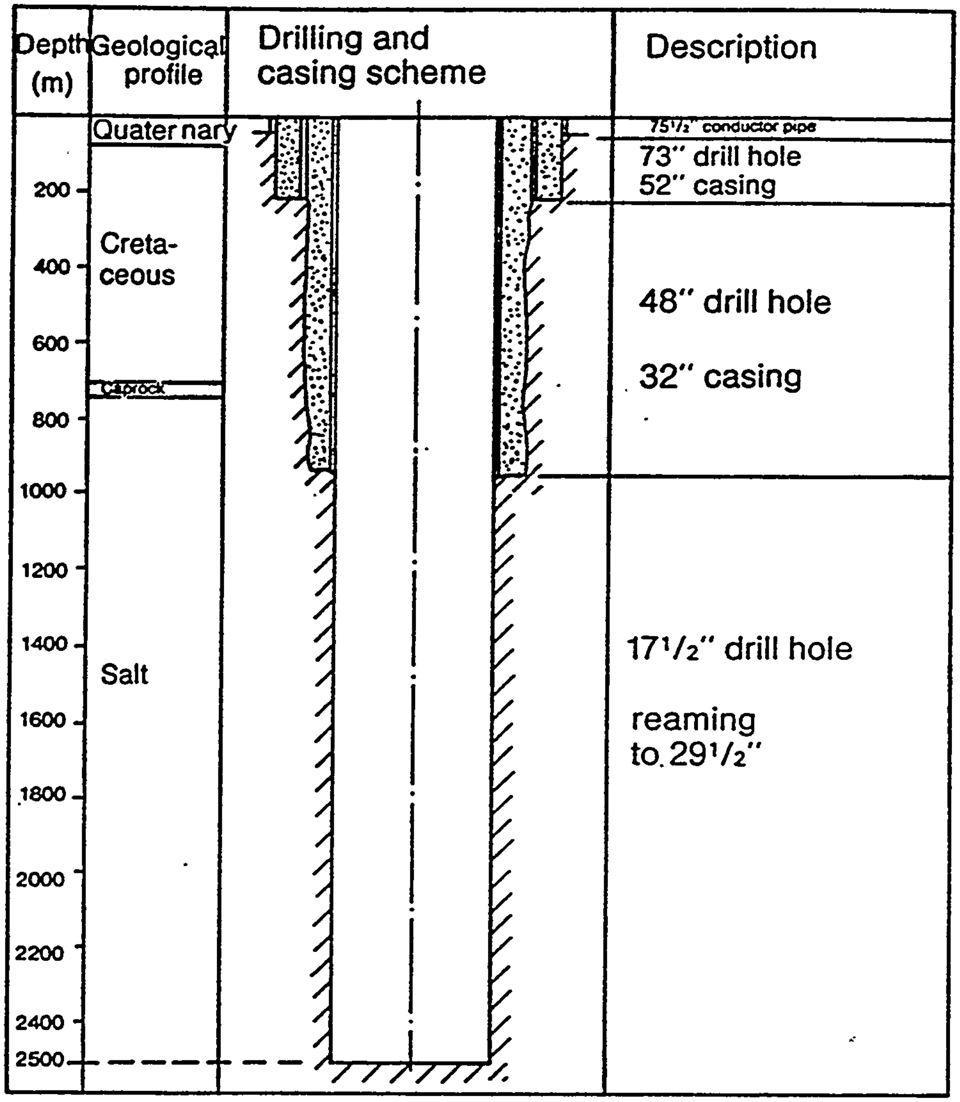

Figure 8.2.5. Deep Hole Construction

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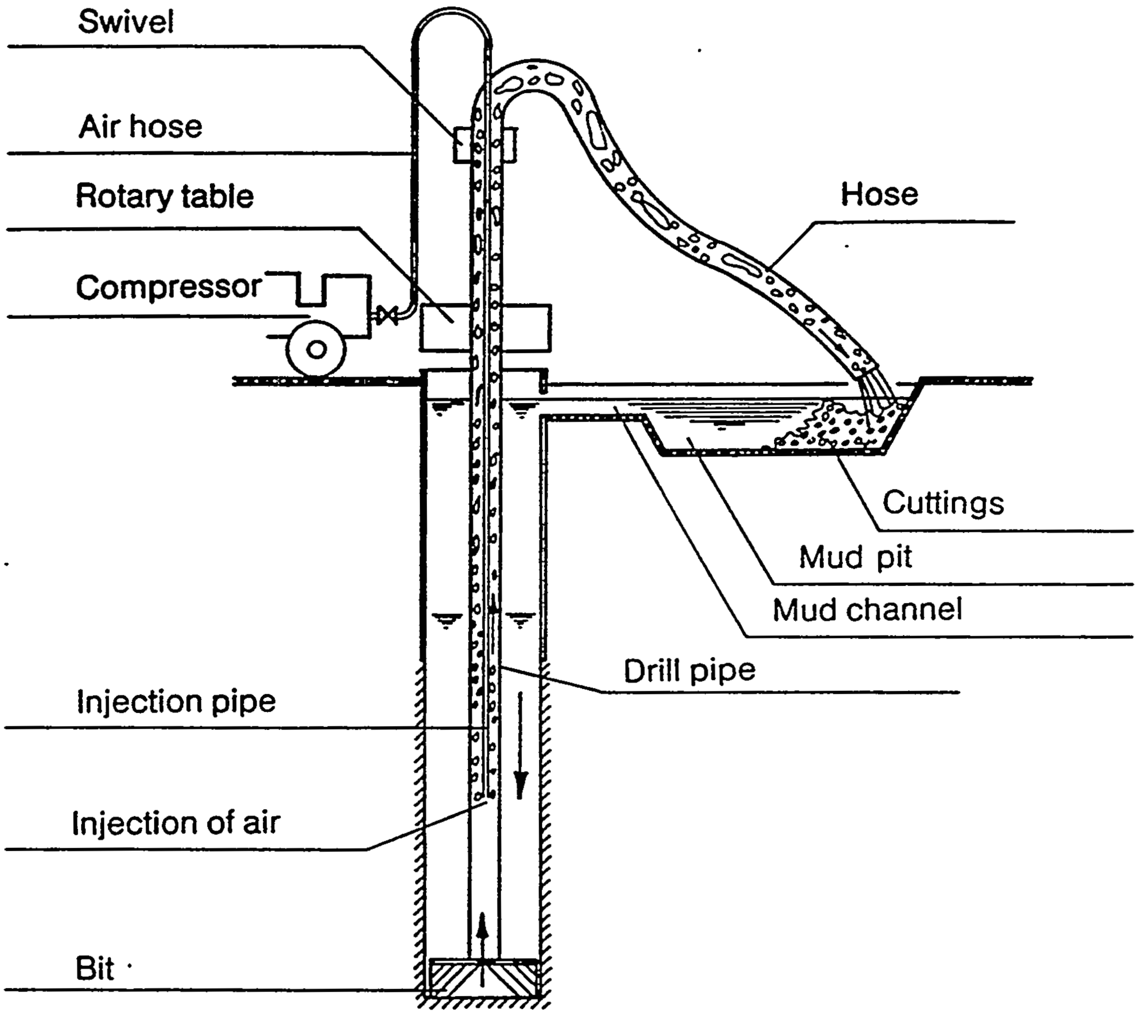

Figure 8.2.6. Air Lift System

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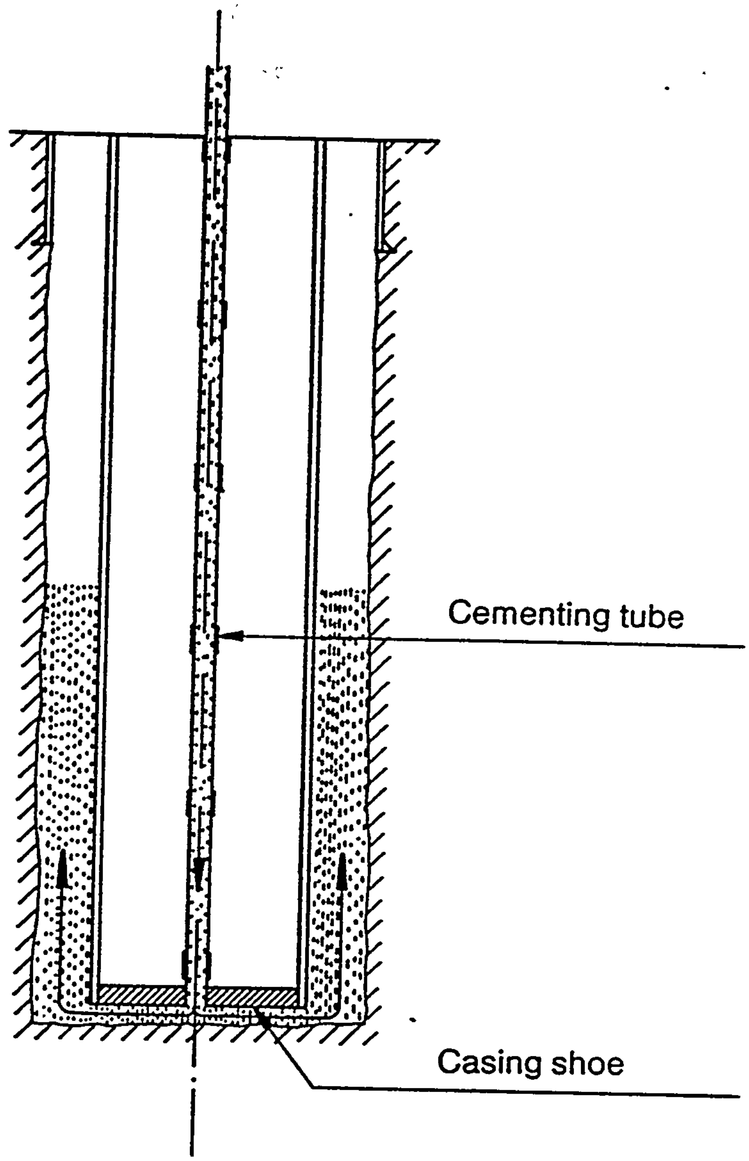

Figure 8.2.7. Cementation of Casing

8.2.5 Waste Emplacement Cycle

8.2.5.1 Waste Treatment

The total fuel consumption for the Danish nuclear power program was estimated to be about 4,800 MT (5,280 tons). The spent fuel is expected to be reprocessed about 10 years after removal from the reactor and the high-level wastes vitrified in borosilicate glass. The glass will be cast in canisters of stainless steel with a capacity of about 150 l (40 gal). Each canister will hold high-level waste from about 1 MT (1.1 tons). The total number of canisters will therefore be about 4,800. Before disposal, the waste canisters will be stored for about 30 years at an interim storage facility to allow the heat and radiation to diminish.

8.2.5.2 Waste Transport

Details of the waste transport system were not described in the literature but it can be assumed that the systems will be similar to those described for other countries.

8.2.5.3 Waste Package

The high-level waste canister described above is about 35 cm (14 in) in diameter and 190 cm (6.2 ft) in length with a total weight of about 500 kg (1,100 lbs). Figure 8.2.8 shows the waste canister. Before disposal in the repository, the canisters will be encapsulated in a waste container for additional shielding and mechanical protection. This container will have a wall thickness of about 15 cm (6 in) and will be constructed of mild steel. Mild steel was chosen since investigations showed that only very small amounts of brine are present in the salt dome at Mors and thus containers will corrode only an insignificant amount. Figure 8.2.9 shows the waste canister configurations. For the deep borehole concept, the container will hold three waste canisters and for the mined repository, the container will hold one waste canister. The weights of the containers, including waste canisters, will be about 13 tons and 5 tons for the deep borehole and mined facilities, respectively. The total number of containers will be about 1,600 for the deep borehole repository and 4,800 for the mined repository.

8.2.5.4 Underground Waste Emplacement Sequence

8.2.5.4.1 Mined Underground Facility. The waste containers will be transported down the shaft in the mine cage and hauled to the disposal drift with a special underground truck. Before the first canister is actually

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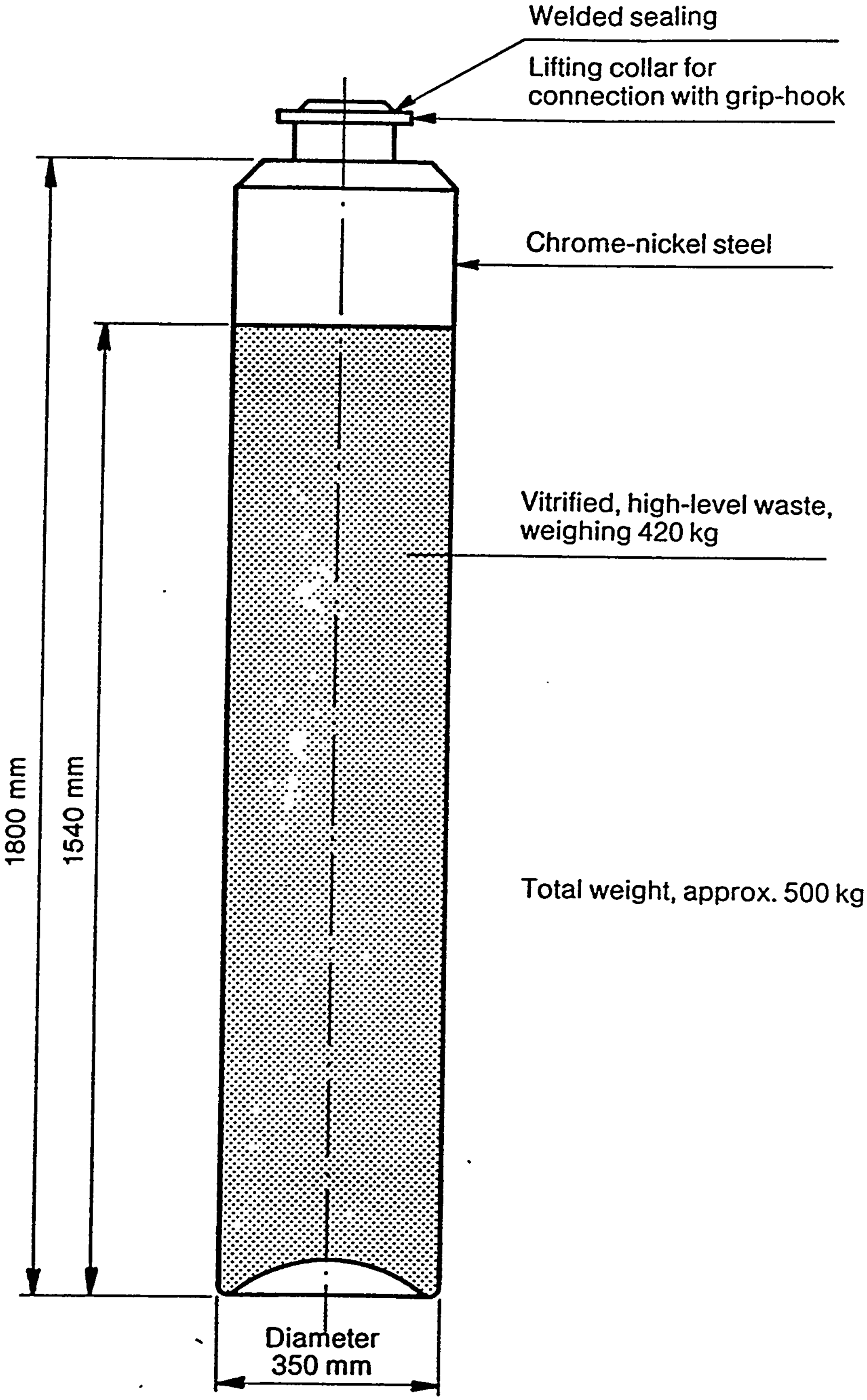

Figure 8.2.8. Waste Canister

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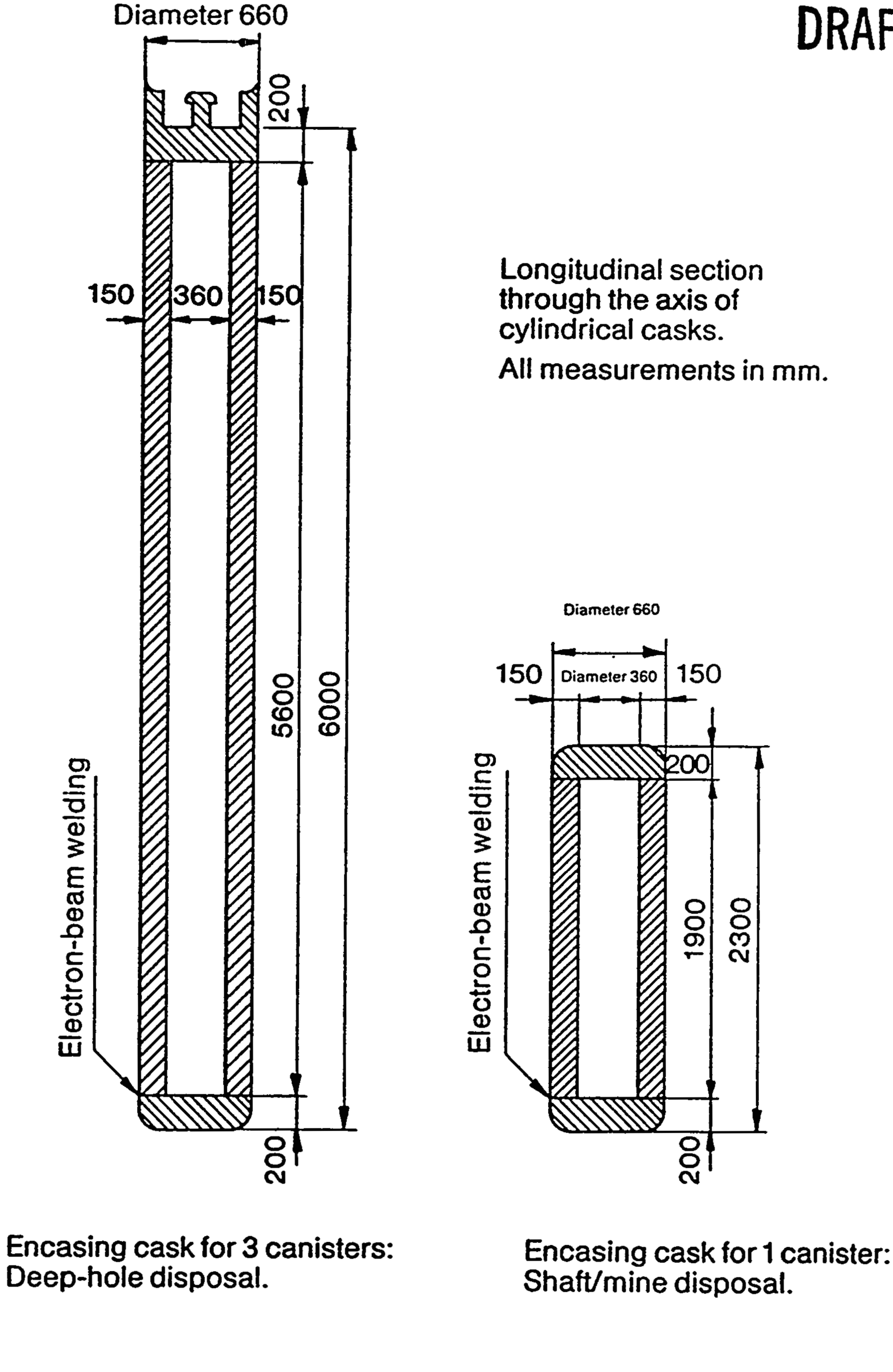

Figure 8.2.9. Encasing of Waste Canisters

placed in its final location in a disposal drift, a 4.6 m (15 ft) long pipe will be placed at the end of the drift and the annulus between the pipe and drift filled with crushed salt. The first container will then be pushed into the pipe, followed by the placement of the next pipe section in the same manner as before. The next two containers will then be inserted into the pipe, followed by installation of the next pipe section, and so on until about 200 containers are placed in the drift. The method of placement will ensure that a distance of at least 2.3 m (7.5 ft), which corresponds to the length of one container, is maintained to the container inserted last. The salt for the backfill around the pipes will most likely be provided by the excavation of the next disposal drift. Emplacement of the containers and excavation of the disposal drifts will take place simultaneously. After the placement of about 200 containers, the drift will be isolated by a 20 m (66 ft) thick bulkhead of reinforced concrete to prevent any water inflows from reaching the containers. The repository and shafts will be sealed when all disposal drifts are full.

8.2.5.4.2 Deep Borehole Facility. When drilling is complete, the hole will still be filled with drilling mud. This will be replaced with brine since the mud is a thixotropic fluid and would make lowering of the containers difficult. The waste containers will be lowered into the borehole using the drill rig or a winch and derrick system. The ends of the containers will be rounded so that they can be lowered easily and stacked in the borehole (Figure 8.2.9). Centralizers will aid in the stacking of the containers. As the containers are lowered into the borehole, they are also cemented into place. The cement will be pumped into the borehole below the brine at suitable intervals so that the containers will be lowered into the cement during the last part of the emplacement process. Retarders will keep the cement fluid for about 50 hours to allow placement of about five to 10 containers. After about 200 containers are placed in a borehole, the upper 1,200 m (3,940 ft) will be backfilled and sealed. Figure 8.2.10 shows the completed deep borehole repository.

8.2.5.5 Emplacement Schedule

It was estimated that one disposal drift would be required per year in the mined repository which is an emplacement rate of about 200 canisters per year. This rate would mean that a new borehole would be required every three year with the deep borehole repository.

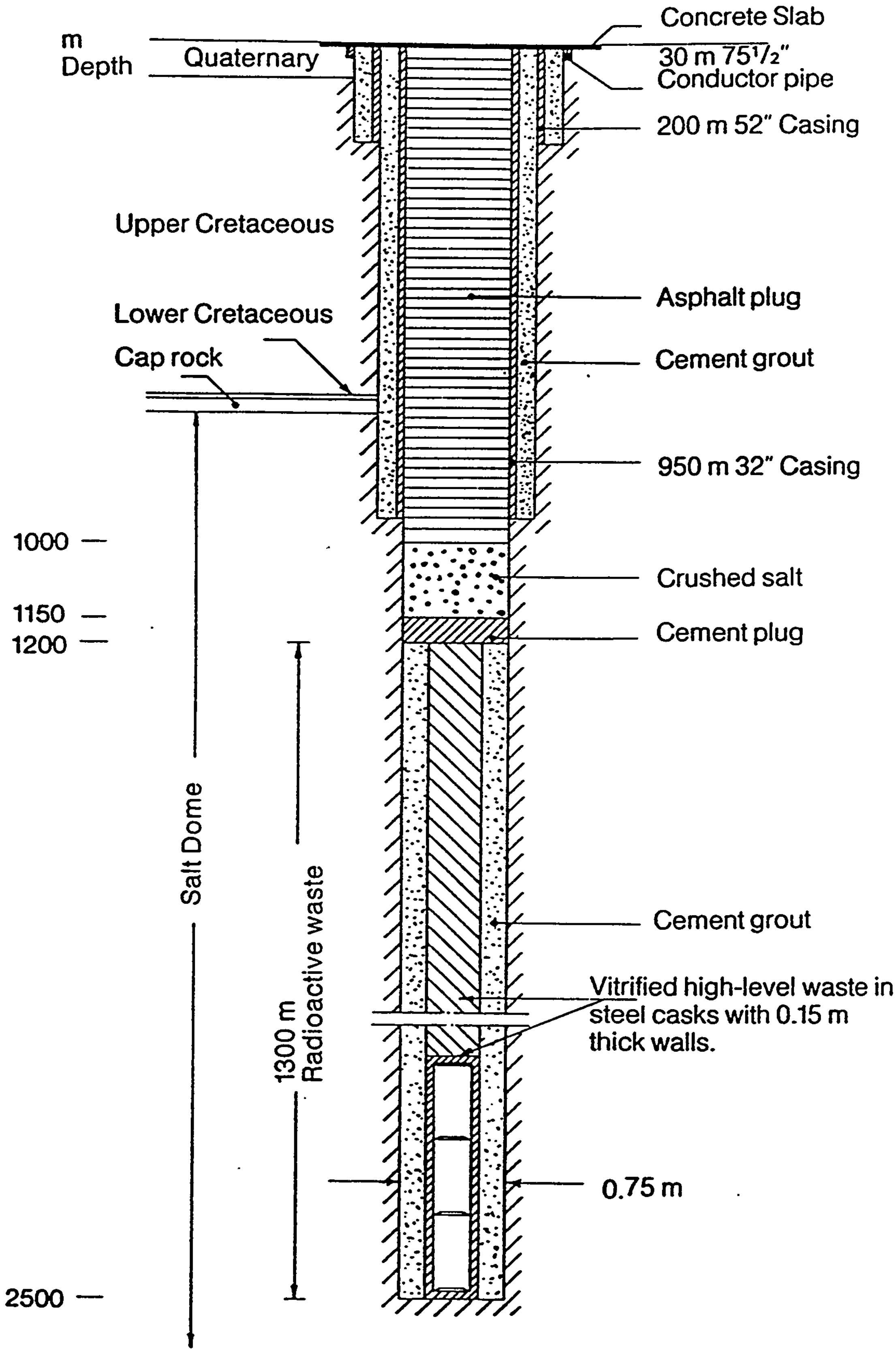

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Figure 8.2.10. Deep Hole Disposal Facility

8.2.6 Buffer and Sealing Materials

8.2.6.1 Mined Underground Facility

The buffer material around the containers in the drifts will consist of the pipe and crushed salt backfill. After all containers are deposited in the disposal drifts, the remaining openings will be backfilled with crushed salt. The effects of temperature and pressure will eventually reconsolidate the crushed salt backfill and reseal the openings in the salt dome. After backfilling of the repository, the shafts will also be sealed to prevent water migration into the salt dome. Figure 8.2.11 shows a proposed shaft sealing concept. A plug of coarse basalt rocks bonded with cement will act as foundation for the upper sealing materials. A chemical seal ring topped by crushed salt will be placed on top of the basalt. Above the salt, a layer of concrete will be topped by a plug of bentonite concrete, which will expand if it contacts water. The final seal from near the base of the shaft lining to the surface will be the asphalt. The asphalt will act as a barrier against the inflow of water both in the short-and long term. An asphalt mixture with 63% bitumen and 37% limestone powder will have a specific gravity of about 1.3. This will be greater than that of saturated brine so that the pressure of the asphalt column will exceed the hydrostatic pressure of the surrounding formation thereby preventing water inflows. Finally the top of the shaft will be covered with steel plates and concrete and the surface areas reclaimed to their original condition.

8.2.6.2 Deep Borehole Facility

On completion of the waste disposal activities, the remaining brine will be pumped from the borehole. Figure 8.2.12 shows the sealing arrangement proposed for the rest of the borehole to the surface. First a 50 m (164 ft) plug of cement will be placed to be followed by a crushed salt backfill to a depth of about 970 m (3,160 ft). The salt will be reconsolidated by temperature and pressure effects and thus the waste containers will eventually be enclosed completely by natural salt materials. The remaining borehole from the top of the salt back to the surface will be filled with the same type of asphalt backfill as was described previously for the mined repository. The asphalt plug will extend about 20 m (66 ft) below the bottom of the last casing into the salt and thus provide a tight seal against water migration into the repository since the pressure of the asphalt column will exceed the hydrostatic pressure of the water that might migrate down behind the cemented casing.

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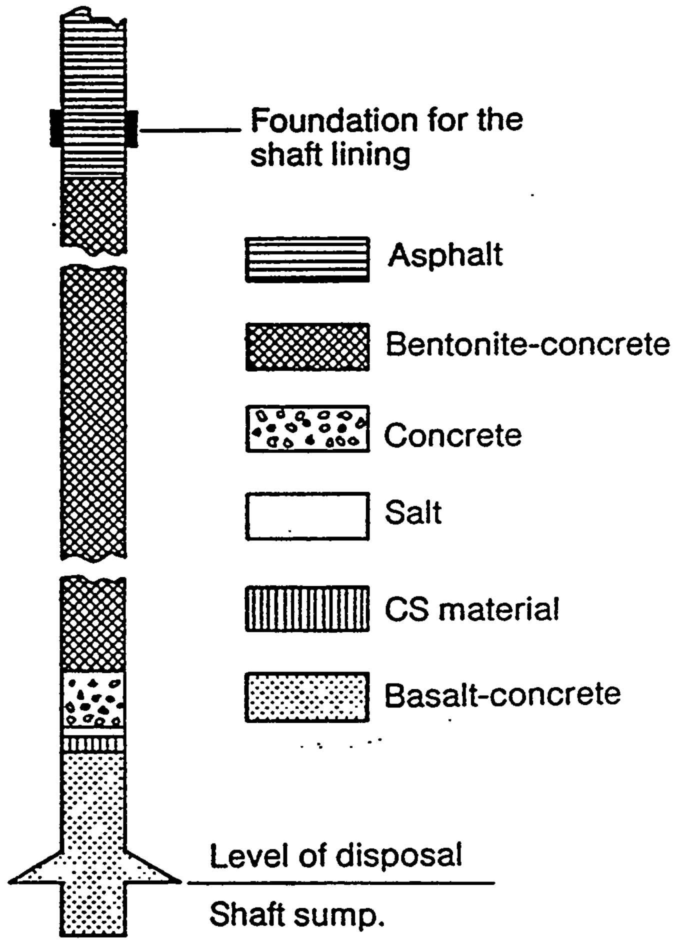

Figure 8.2.11. Sealing of Shaft

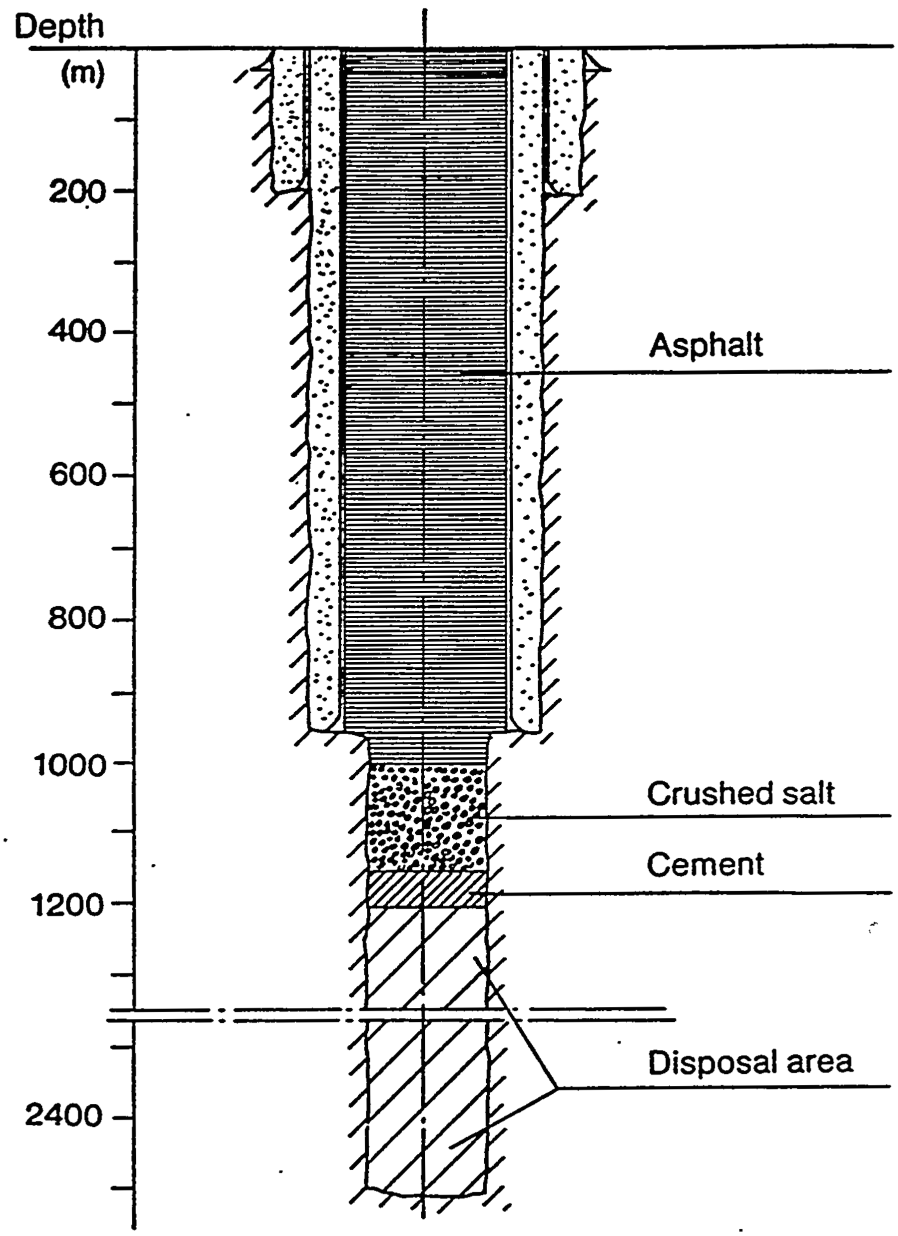

Figure 8.2.12. Deep Hole Sealing

8.2.7 Summary

Two salt dome repository concepts were presented above - a mined underground facility and a deep borehole disposal facility. Both have advantages and potential problems and these are outlined briefly in the following sections.

8.2.7.1 Advantages

1. Salt formations are considered suitable to host repositories by several countries (Unites States, Germany, Holland) because of the favorable properties of the salt. These include the practical absence of water, high thermal conductivity, self-sealing of openings due to its creep properties, relative abundance in large deposits, and ease of construction and maintenance of excavations in salt.

2. A large amount of mining experience in salt as well as understanding of salt behavior through extensive investigation programs exist in the world today.

3. The massive salt domes can provide a substantial natural barrier around a repository. Barriers of at least 200 m (660 ft) are envisioned for the salt dome repositories in Europe.

4. The deep borehole disposal concept has several advantages over the mined repository:

   a. Disposal at great depth gives a larger barrier around the waste containers and any water inflow or radionuclide release has to travel a longer path. There is also less risk of releasing the waste containers to the biosphere through diapirism.

   b. The waste containers will be enclosed by the salt sooner at the greater depths but stronger containers will be required to offset the greater pressures.

   c. The excavation volume per m^3 of waste is less for the deep borehole (2.5 m^3 against 75 m^3 for the mined facility).

   d. The borehole concept represents the least disturbance to the salt dome.

   e. Sealing and backfill material requirements are substantially less.

f. All operations are conducted from the surface, an important safety consideration.

g. Construction and operation durations are less than for the mined repository.

5. The mined repository, however, also presents some advantages over the deep borehole concept:

a. Installation of the lining and seals to isolate the formations above the salt will be done in situ and inspection of the geology at the seal locations is possible.

b. Disposal and backfilling/sealing of the waste containers at their final location will be carried out under visual supervision.

c. The repository area can be characterized better from an underground facility.

d. The greater depths of the deep borehole concept may cause problems during operation since it may be difficult to keep the hole open for the period required.

8.2.7.2 Potential Problems

1. Despite the many advantages of salt, several potential problems are associated with waste disposal in salt:

a. Salt is highly soluble, has poor sorptive characteristics and brine solutions are high corrosive.

b. Construction of the access shafts or boreholes through the water-bearing formations above the salt using specialized techniques such as freezing and lining with steel and concrete, or using large hole drilling techniques is always associated with the risk of water inflow into the repository.

c. The long-term effect of the temperature on the salt dome stability and properties is still under investigation as is the stability of the waste container.

d. Sealing and backfilling of the repository will have to ensure that no water can penetrate for a long time period.

e. Salt deposits are often associated with valuable resources such as potash and hydrocarbons.

8.2.7.3 Summary

Disposal of nuclear waste in salt formations is considered a viable option by many countries and several are actively pursuing the option. The Danish program prefers the deep disposal option whereas the U.S. program is pursuing the mined geologic repository in bedded salt, although disposal in a salt dome was one of the options considered during the initial stages of the program. However, any work done by the Danish program in the salt domes will be of relevance to the U.S. salt repository program.

8.2.8 The Low- and Intermediate-Level Concepts

Research on the low-level waste repositories has been conducted in connection with the high-level waste program. A proposed concept is to use a mined underground facility, also located in a salt dome.

8.3 SALT: THE WEST GERMAN DISPOSAL CONCEPT

This chapter outlines the high-level nuclear waste disposal program of the Federal Republic of Germany. The waste disposal policy is based on the condition that all types of radioactive wastes are disposed of in a geologic repository, especially in rock salt formations. The descriptions in this chapter describe the reference concept in which the low-and intermediate-level wastes are disposed of in caverns or bunkers, and the high-level wastes from reprocessing are placed in vertical boreholes drilled into the floor of the access drifts. The repository is to be located in the Gorleben salt dome in the eastern part of West Germany. An alternative study was also conducted in which the spent fuel is disposed of directly without reprocessing. This concept is summarized in Section 8.3.8.

8.3.1 Geographic Location

Figure 8.3.1 shows the location of the repository sites. The salt domes are located in the northern part of West Germany where extensive deposits of the Zechstein salt are found.

8.3.2 Summary of Geologic Setting

The Gorleben salt dome is about 14 km (8.7 mi) across and its base is about 3,100-3,300 m (10,170-10,830 ft) below mean sea-level. Its width decreases with depth and at the repository depth of 800-1,300 m (2,630-4,270 ft) it is about 3 km (1.9 mi) wide. The top of the salt is approximately 300 m (980 ft) below the surface. The overlying formations

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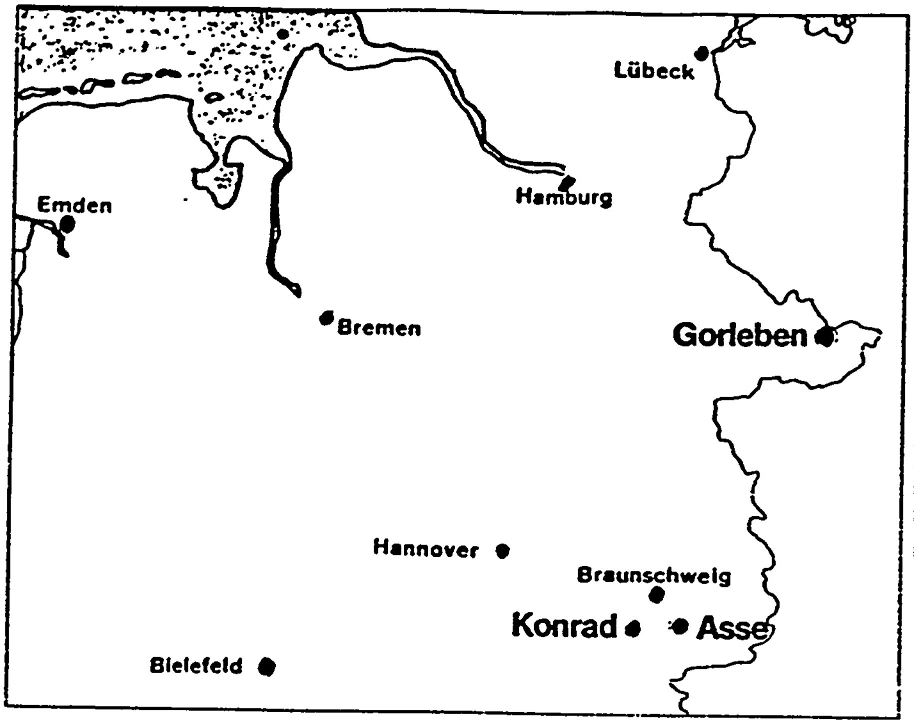


Figure 8.3.1. Repository Locations

consist mainly of sands, gravels, clays, silts and tills. These formations are partly unconsolidated and water bearing and therefore shaft sinking requires the use of ground freezing. Figure 8.3.2 shows the salt dome configuration.

8.3.3 Repository Concept Description

This section outlines the proposed German reference radioactive waste disposal concept which includes the reprocessing of spent fuel elements. An alternative case of direct disposal of spent fuel elements, which was also subject of a detailed investigation, is discussed in Section 8.3.8.

8.3.3.1 Nuclear Waste Disposal Strategy

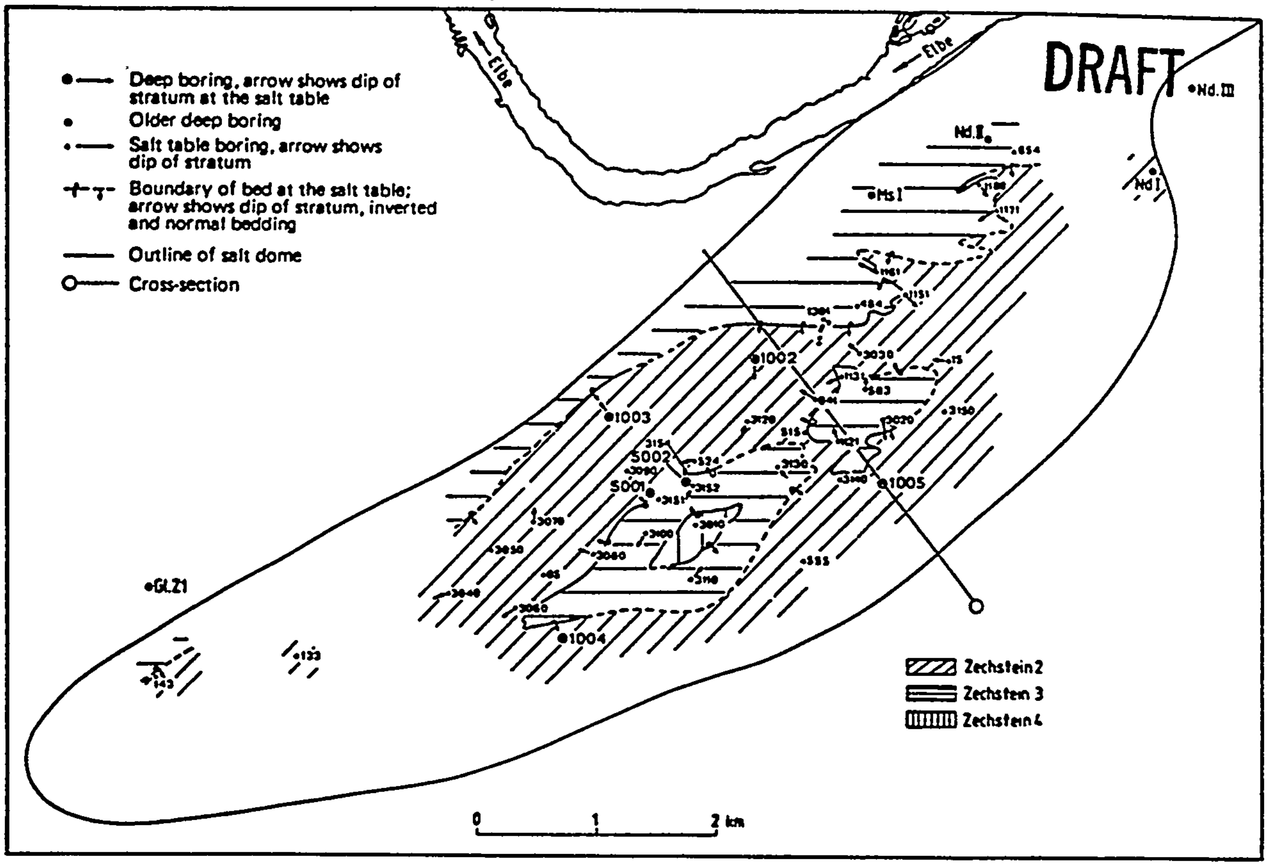
The initial high-level nuclear waste disposal strategy is to store the spent fuel elements for a period of about 7 years to allow the heat and radioactivity to decay, followed by reprocessing. Reprocessing contracts exist with COGEMA in France and BNFL in Britain. A reprocessing plant is also planned to be opened in West Germany. The first reprocessing high-level wastes are expected to start arriving back in Germany in the early 1990s; and therefore, storage of these wastes will have to be provided for about 10 years since the repository is not expected to be operational until about the year 2000. The nuclear power capacity is expected to be about 30-35 GWe by the year 2000 which will give rise to about 11,000 MT (12,100 tons) spent fuel and produce radioactive wastes amounting to approximately

- Between 250,000 and 400,000 drums of low-level wastes
- Between 70,000 and 100,000 drums of intermediate-level wastes
- About 4,600 vitrified block (canisters) of high-level wastes.

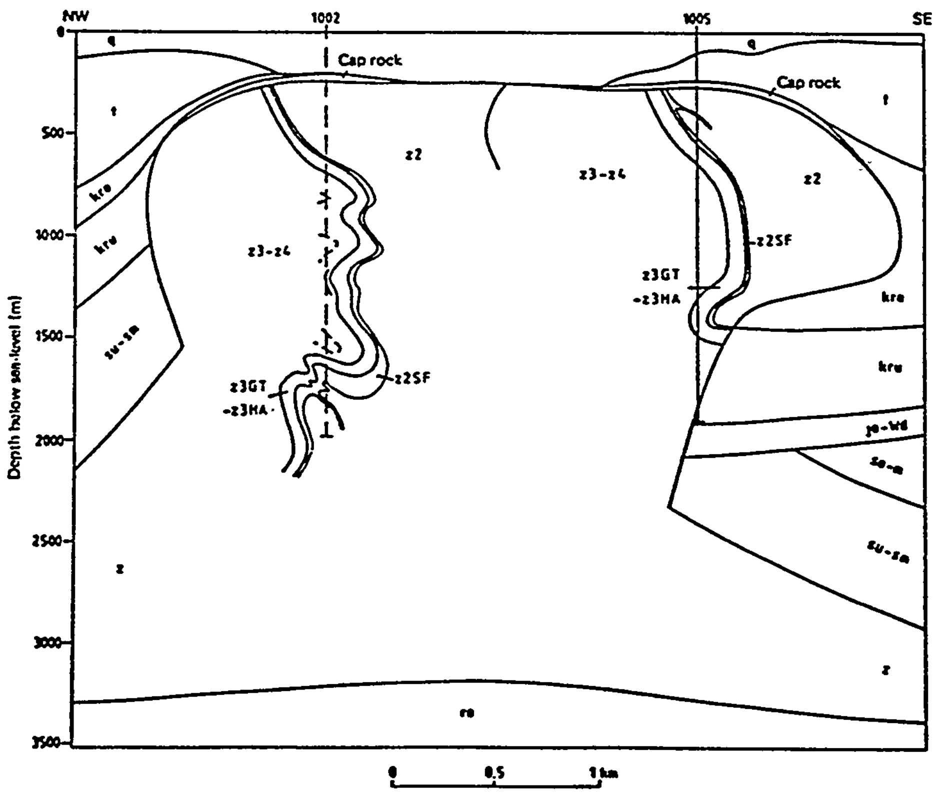
8.3.3.2 Surface Facilities

Surface installations other than those at the repository are expected to include away-from-reactor storage facilities and spent fuel reprocessing plants. A 1,500 MT (1,650 tons) capacity dry storage facility is currently operating at Gorleben. A second dry storage facility is to be constructed at Ahaus, and a reprocessing plant is proposed for construction at Wackersdorf. The exact disposition of these facilities will depend on future policy and political decisions.

The exploratory shaft site investigation program was started at the Gorleben salt dome in early 1984 and will continue until well into the



Gorleben salt dome: geological map of salt table.



Cross-section of Gorleben salt dome. q: Quaternary; t: Tertiary; kro: Upper Cretaceous; kru: Lower Cretaceous; jo-Wd: Upper Jurassic-Wealden; so-m: Upper Bunter-Muschelkalk; su-sm: Lower-Middle Bunter; z: Zechstein; ro: Rotliegende.

Figure 8.3.2. Gorleben Salt Dome Configuration

1990s. Two shafts are currently under construction from which the salt body will be characterized using horizontal drifts in the salt and exploratory drilling from those drifts. If the salt dome is found suitable to host the repository, the surface and underground facilities will be expanded for radioactive waste disposal. Figure 8.3.3 shows the surface facilities during the site characterization studies. The expanded repository surface facilities will also include the following areas:

- Reception building for receiving the waste packages arriving by rail and truck
- Reloading facility for inspection of the waste packages, combining and preparation of the packages for transport underground, and temporary storage of waste packages
- No. 1 shaft service building including personnel and equipment hoist and installations for salt and material handling
- No. 2 shaft service building including the waste package handling facilities
- Ventilation equipment such as fan houses, fans, ducting, etc.
- Administration and office buildings
- Shops and warehouses
- Salt storage areas
- Surface and underground services such as power, water, compressed air, etc.
- Security and emergency services.

All waste handling will take place through shaft No. 2. Waste packages will arrive by rail or truck. After inspection, each package is transferred to a transport dolly and it will remain on this dolly during lowering in the shaft and transport to the underground disposal location.

8.3.3.3 Underground Configuration

The underground facility was designed using principles generally in use in salt mines in Germany. The aim was to design a facility with a high degree of stability and optimal emplacement conditions, at the same time conforming to the safety requirements for waste disposal. These

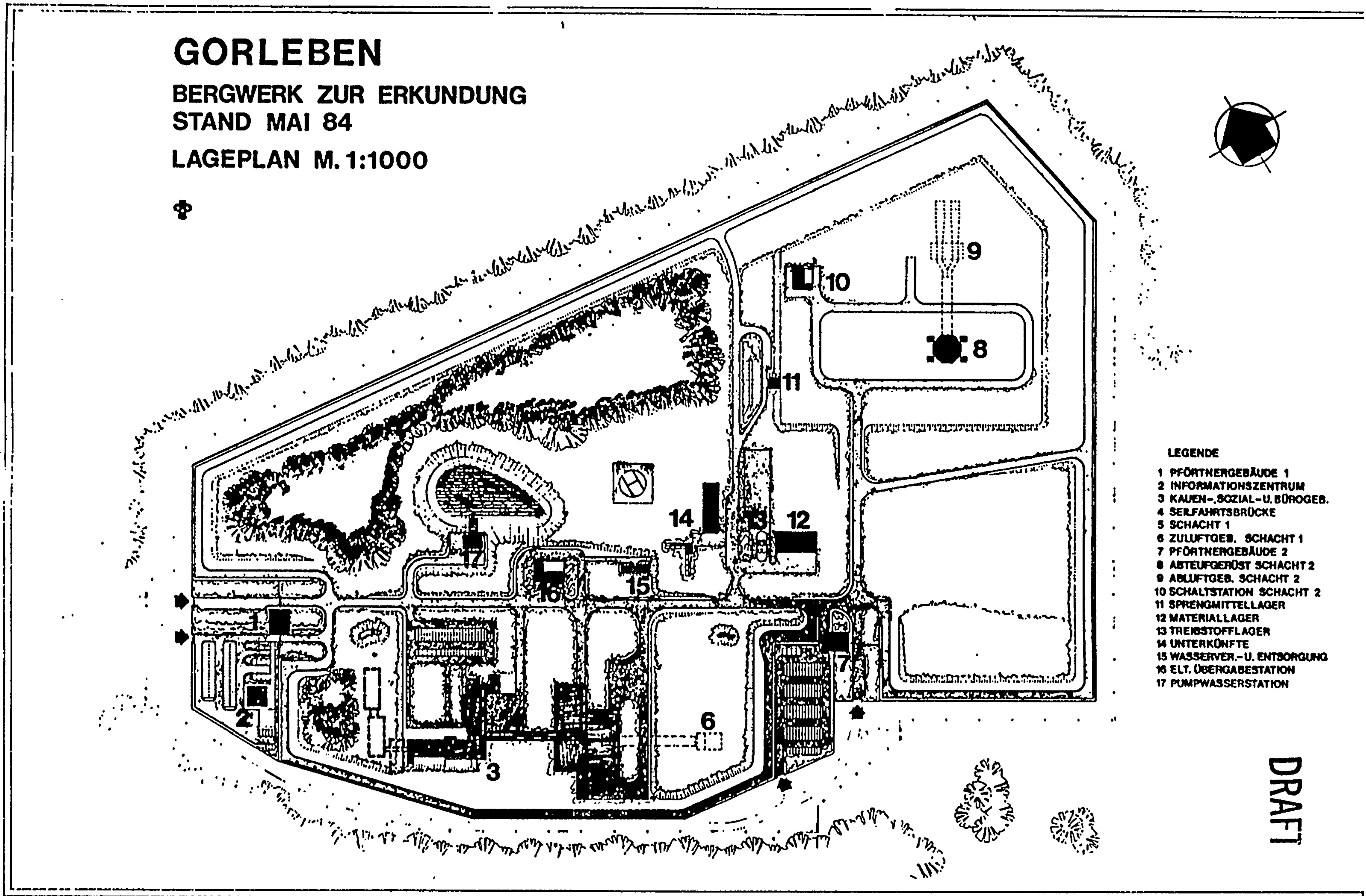
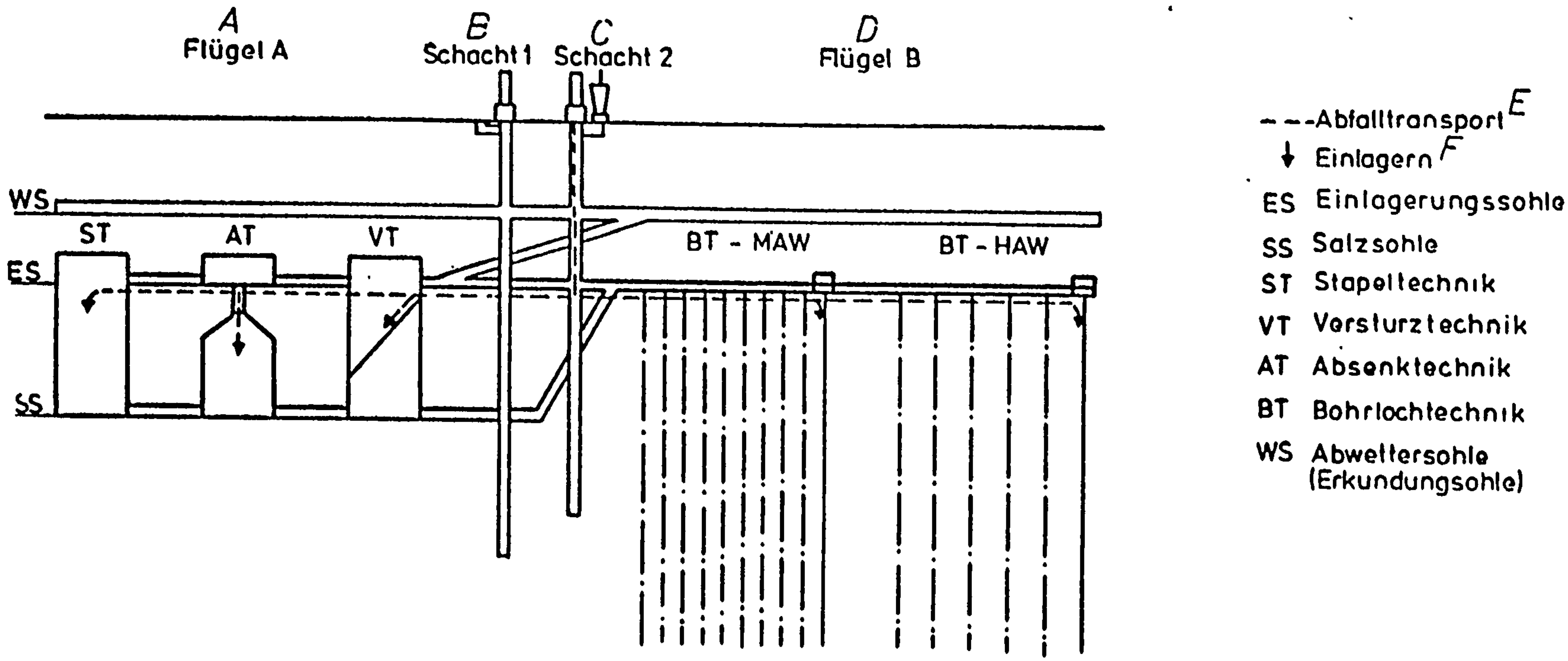


Figure 8.3.3. Gorleben Exploratory Site Surface Facilities

requirements include separate transport routes for the excavated salt and the waste packages, short waste transport routes, and short waste package handling times. Figure 8.3.4 shows a general arrangement of the repository. The two exploratory shafts will be converted to the repository shafts with shaft No. 1 being the ventilation intake and main access shaft and shaft No. 2 being the ventilation exhaust and waste transport shaft. The shaft depth will be about 940 m (3,080 ft). Repository development will take place from the shafts and existing exploratory drifts. The underground openings will be arranged on three basic levels. The upper level will be the original exploration level and act as the return air level. About 30 m (100 ft) lower is the main emplacement level, and 16.5 m (55 ft) below that, the salt handling level. The emplacement and return air levels will extend over the full horizontal area of the repository, while the salt handling level will extend only below the cavern disposal section. The repository will be divided into two sections, depending on the disposal method plus a central facilities section (Figure 8.3.5). The central (shaft pillar) section will contain the shafts and ancillary underground facilities such as equipment repair shop, warehouses, etc. Section A of the repository will contain the caverns for the low-and intermediate-level wastes and will be divided into different disposal areas, depending on the emplacement technique used (Figure 8.3.6). The three basic emplacement techniques were tested at the Asse salt mine repository, and include either dumping or stacking of the low-level waste containers, and lowering of intermediate-level waste packages into a sealed cavern. These techniques will also be incorporated into the repository design as shown in Figures 8.3.4 and 8.3.6. Section B of the repository will consist of a series of parallel drifts and the 300 m (980 ft) deep disposal boreholes will be drilled into the floors of these drifts. Two disposal areas are proposed within Section B (Figure 8.3.7). The disposal area for the heat-generating waste packaged in 400-liter (approximately 110 gal) drums will be nearest to the central shaft area. The area will consist of 8 parallel, 600 m (1,960 ft) long drifts with about 29 disposal holes per drift, spaced at roughly 15 m (50 ft) intervals. The high-level vitrified wastes will be stored in the second area which will consist of about 45 drifts, each with 10 disposal boreholes spaced at 50 m (165 ft). In order to keep the maximum temperature at the salt/waste canister interface below 200 °C (390 °F), it was proposed that "dummy" containers (ie. containing no waste) be inserted as spacers between the high-level waste packages in the borehole. Figure 8.3.7 shows the layout of Section B.



Section through the IWM model repository (source: DBE)

A Wing A
B Shaft 1
C Shaft 2
D Wing B
E waste transport
F emplacement
ES emplacement level
SS salt level
ST stacking technique
VT dumping technique
AT lowering technique
BT borehole technique
WS exhaust air level (exploration level)

Figure 8.3.4. Gorleben Waste Repository Layout

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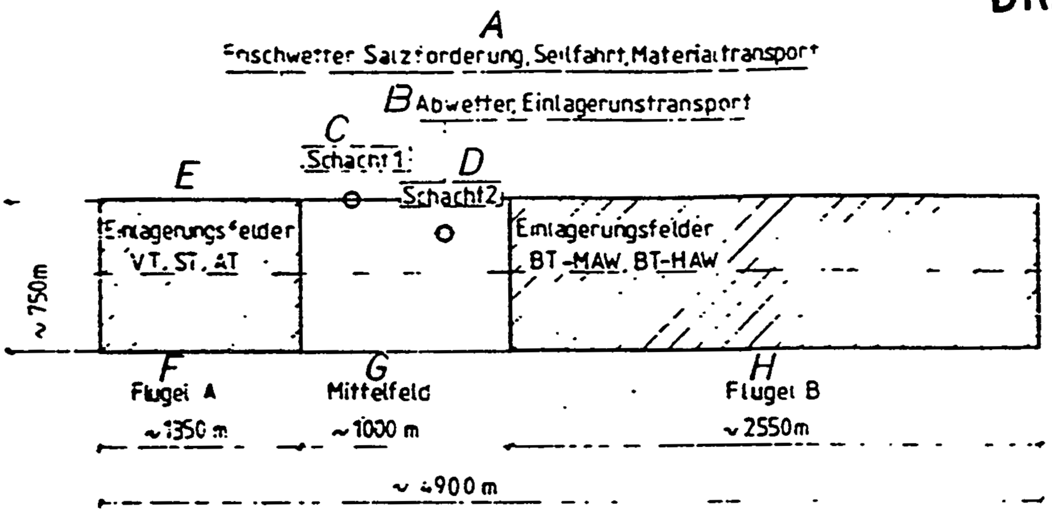


Figure 3-5: Layout of the underground openings of the IWM model repository

A fresh air, salt transport, personnel & equipment hoist, materials transport
B exhaust air, emplacement transport
C shaft 1
D shaft 2
E emplacement fields
F wing A
G central field
H wing B

Figure 8.3.5. Waste Disposal Sections

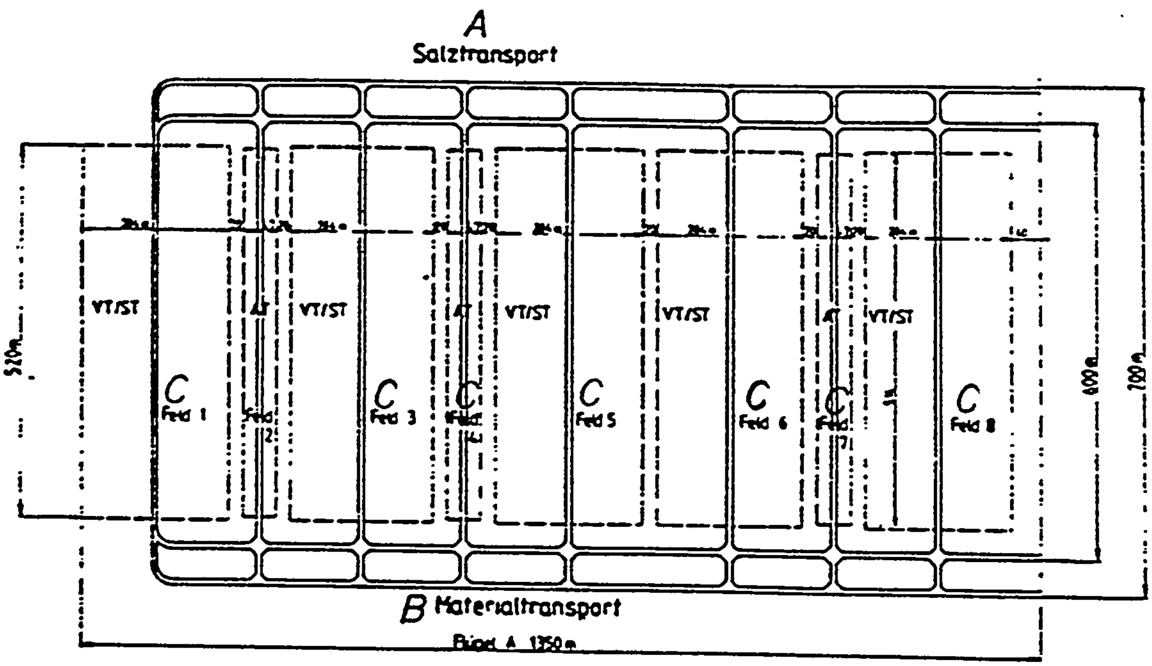
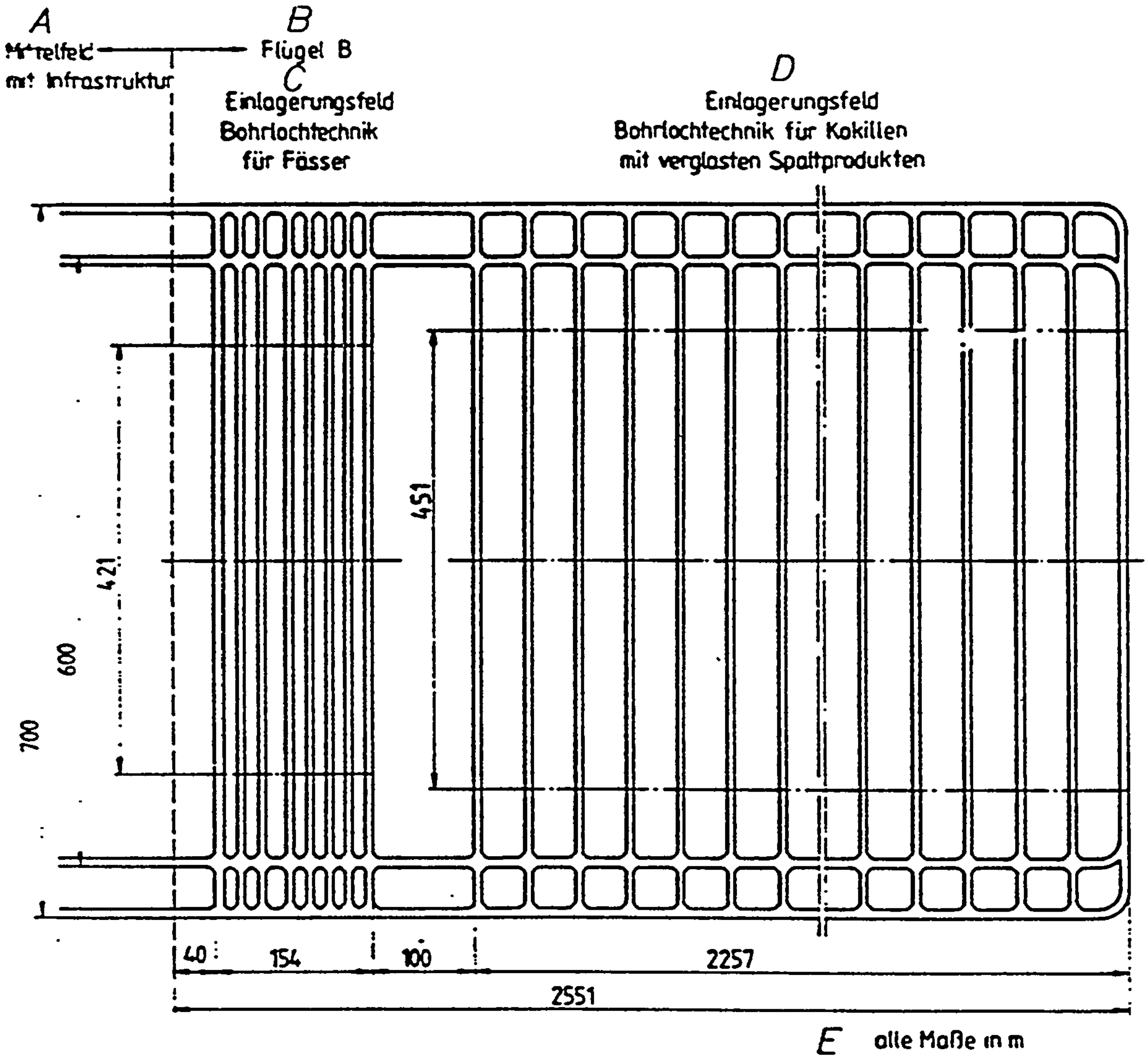


Figure 3-6: Emplacement fields in Wing A of the IWM model repository (source: DBE, TA 15)

A salt transport B materials transport C field

Figure 8.3.6. Repository Section A



Emplacement fields in Wing B of the IWM model repository
(source: DBE, TA 15)

A central field with infrastructure
B Wing B
C emplacement field, borehole technique for drums
D emplacement field, borehole technique for canisters of vitrified fission products
E all dimensions in m

Figure 8.3.7. Repository Section B

8.3.4 Repository Construction

8.3.4.1 Schedule

The Gorleben waste disposal program is planned to evolve in two basic steps:

- Extensive in situ characterization of the whole repository area, and
- Expansion of the exploratory site to full repository dimensions if the site characterization yields positive results.

The sinking of the two exploratory shafts started in early 1984 and is scheduled to be completed in 1989. Site characterization will then commence at a depth of about 840 m (2,760 ft). It is planned to characterize an area of about 19 km^2 (7.4 mi^2) covering the proposed repository area by means of exploration drifts and exploration drilling from these drifts. Construction of the repository is scheduled to start in 1995 if the results of the site investigation program are favorable. In that case, high-level waste disposal will start around the year 2000 and is scheduled to last for about 50 years.

8.3.4.2 Methods and Equipment

8.3.4.2.1 Shafts. Two shafts are currently being constructed at the proposed Gorleben salt dome repository site as part of the site investigation program. The shafts will have a depth of about 940 m (3,080 ft) and a finished diameter of about 7.5 m (24.6 ft). Since the roughly 300 m (1,000 ft) overburden above the salt dome consists of unconsolidated, water-bearing formations, the freezing technique is being used to construct the shafts. After the ground has been frozen, the shafts will be sunk thorough the frozen formations into the top of the salt dome. Excavation techniques may include hand-excavation, mechanical excavation and controlled blasting to protect the freeze wall. A preliminary lining of concrete blocks with chip board joint filler will be installed to support the shaft wall during sinking. In a competent section in the salt at the top of the dome, a foundation will be constructed and keyed into the shaft wall. The final shaft lining of welded steel and concrete will then be constructed inside the preliminary concrete block lining on top of this foundation back to the surface. A backfill of asphalt will be poured into the gap between the concrete block lining and welded steel liner of the final lining. The specific weight of the asphalt will be adjusted so that its pressure will be greater than the hydrostatic head of the groundwater, and thus prevent water inflow and migration behind the liner. Figure 8.3.8 shows the configuration of the shaft lining in the upper section. (Similar linings have been proposed for salt repository shafts in other programs such as the USA and Holland). After completion of the upper shaft lining, sinking will continue in the lower shaft section in the salt. Excavation will likely be by conventional drill-and-blast methods in the center portion of the shaft with spading near the periphery to protect the shaft wall integrity. Little (rock bolts and wire mesh) or no lining is expected to be required due to the competent nature of the salt. Shaft stations will be established at the exploration level to allow breakout for the exploratory drift development.

8.3.4.2.2 Exploratory Drifts. Figures 8.3.9 and 8.3.10 show a proposed layout for the exploratory facility in the salt dome. The main peripheral drifts will be connected by crosscuts and the whole facility will be divided into sections for site characterization. Excavation methods are likely to include conventional drill-and-blast and mechanical methods to demonstrate repository excavation methods. About 25 km (82,000 ft) of exploratory drifts are expected to be excavated. Exploratory drilling will take place in the various sections to characterize the proposed repository area periphery as well as the core of the salt dome. About 120 km (394,000 ft) of exploratory drilling is planned, including several deep core holes.

8.3.4.2.3 Repository. If site exploration confirms the Gorleben dome as a suitable repository site, the exploratory facility will be expanded to the facility shown in Figures 8.3.4 to 8.3.7. As with the exploratory facility, development will likely include continuous mining techniques such as Marietta type miners as well as conventional drill-and-blast. For the excavation of the low-and intermediate-level storage caverns, the pilot-and-slash method may be applicable.

8.3.4.3 Cost

A cost estimate was published by Röthemeyer et al. in 1983, and breaks down the costs as follows (1982 $US):

- Planning up to 1982 — $ 40×10^6
- Future planning — $ 210×10^6
- Site investigations up to 1982 — $ 45×10^6

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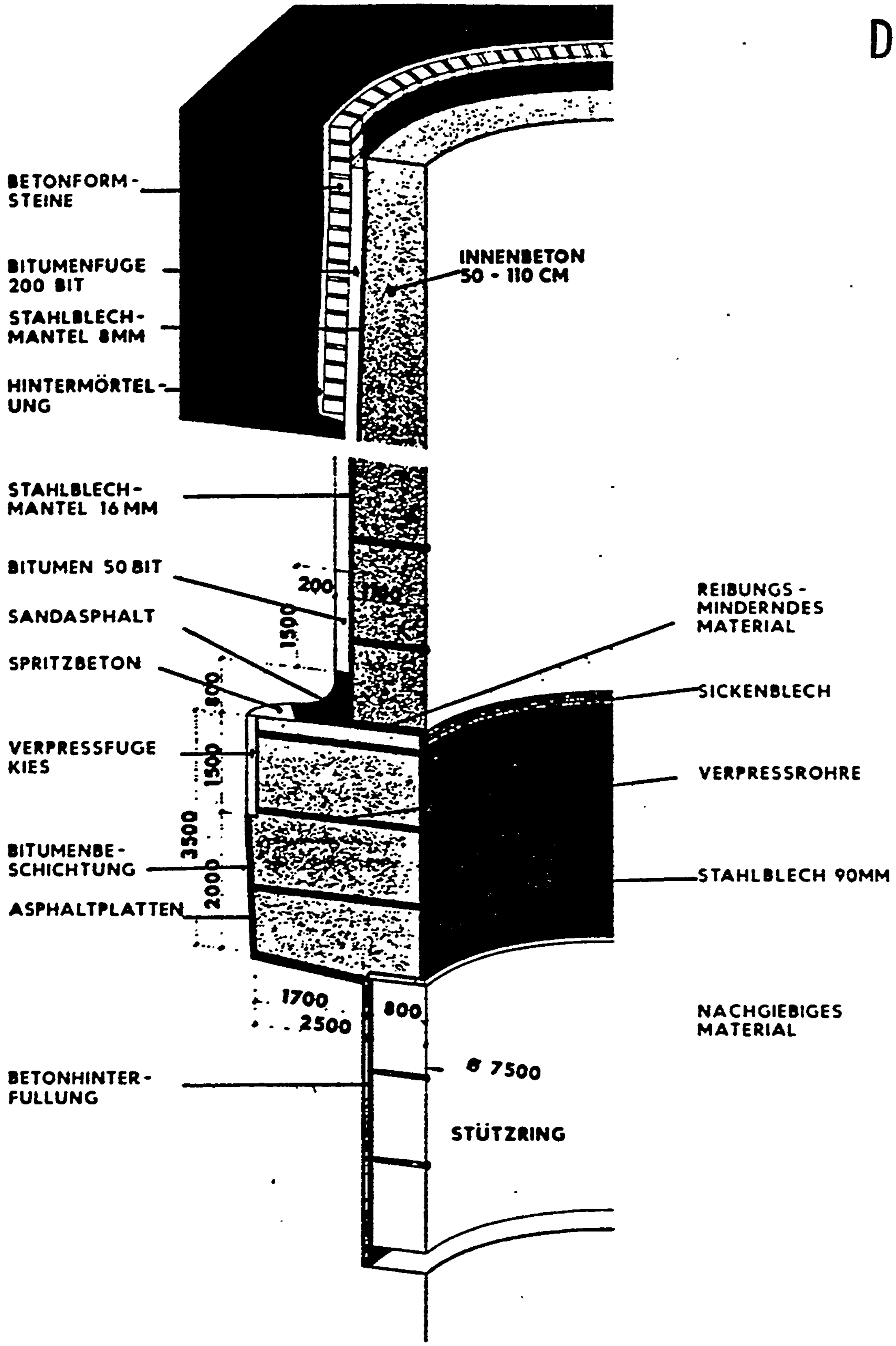

Deutsche Gesellschaft zum Bau und Betrieb von Endlagern für Abfallstoffe mbH (DBE)	SCHÄCHTE GORLEBEN AUSBAU GEFRIERSCHACHTTEIL	1983

Figure 8.3.8. Frozen Shaft Lining

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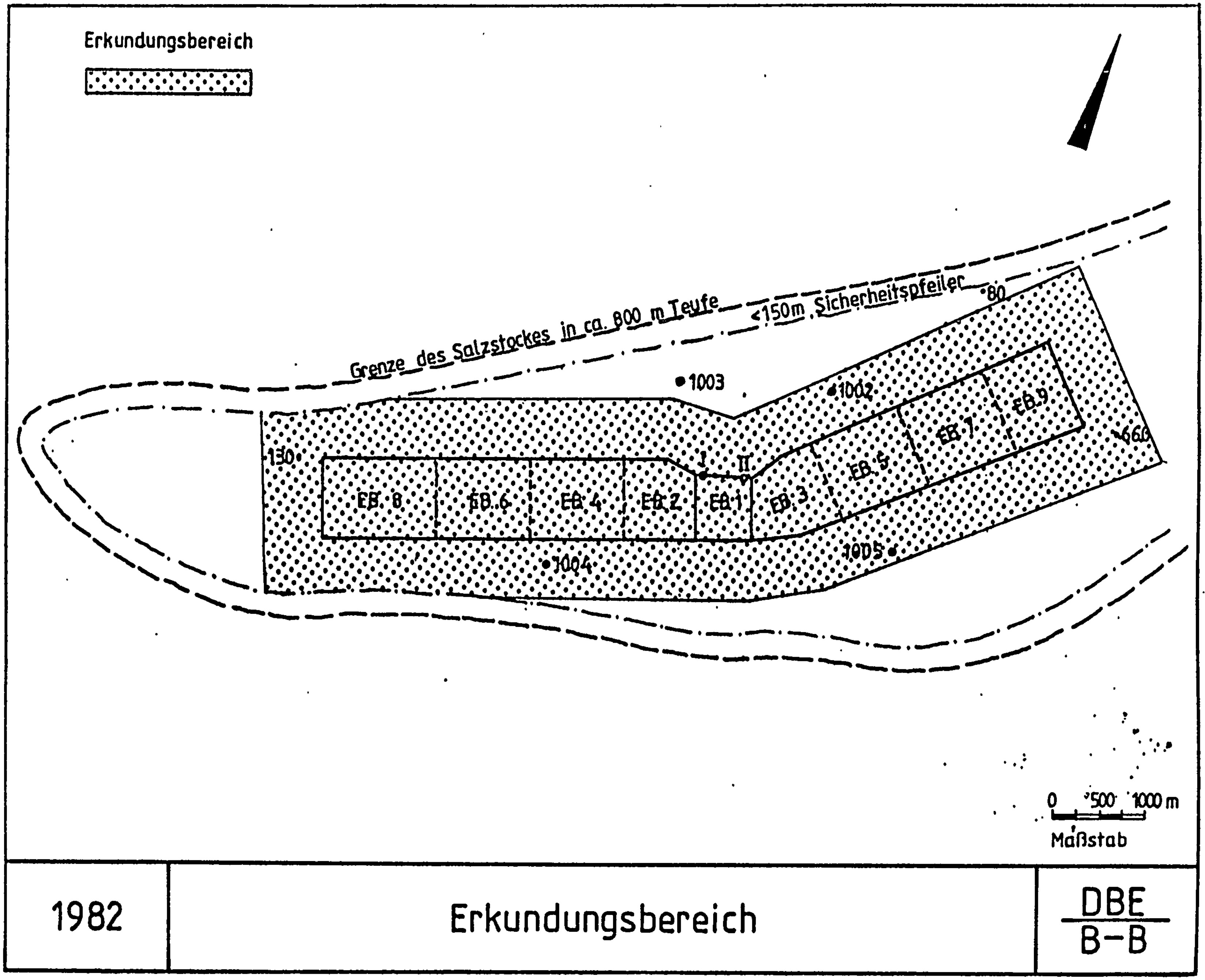

Figure 8.3.9. Gorleben Salt Dome Exploration Area

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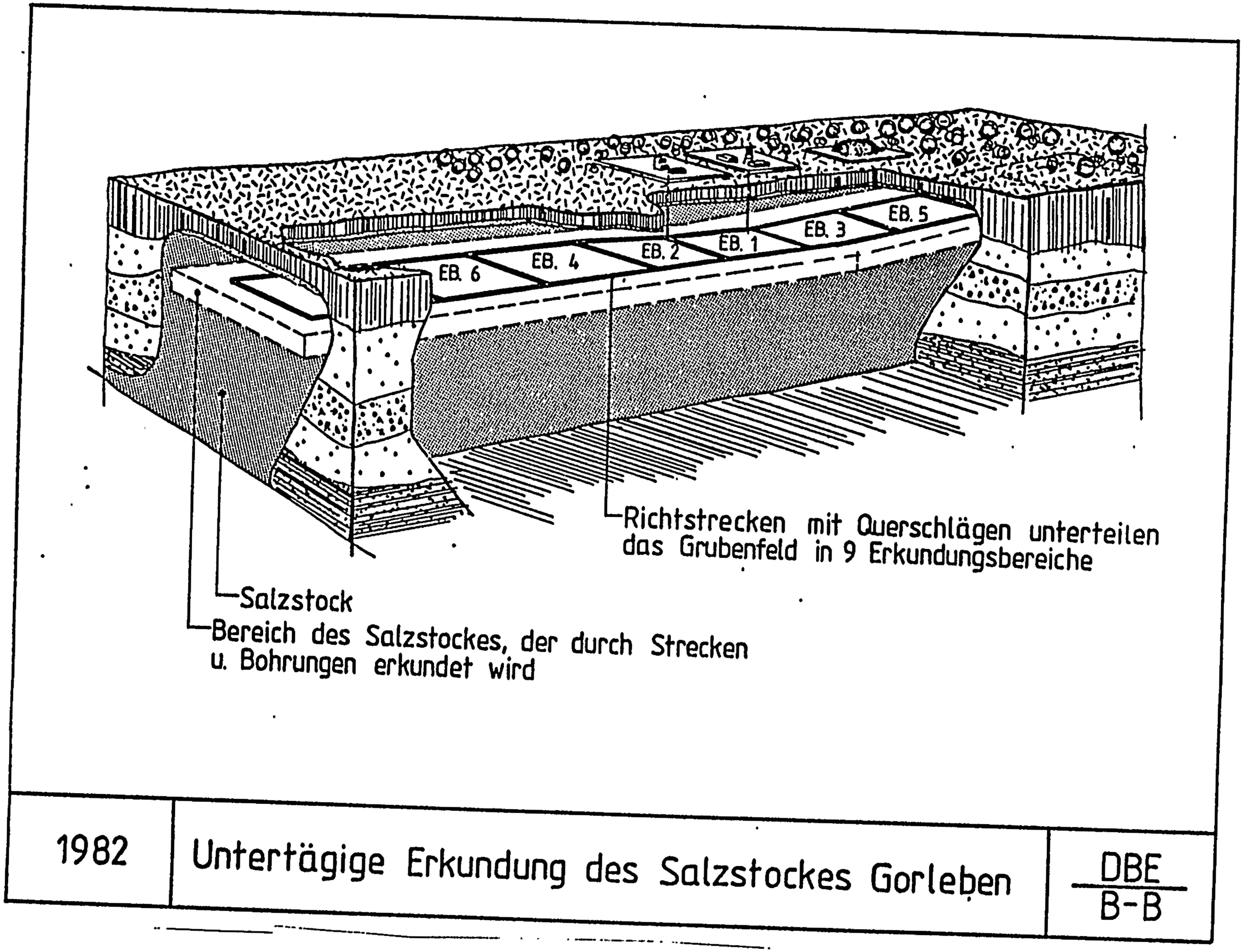

Figure 8.3.1Ø. Underground Exploratory Areas

• Future site investigation including underground investigations	$ 340 x 10^6
• Repository construction	$ 500 x 10^6
TOTAL	$1,135 x 10^6
• Operation per year	$ 50 x 10^6

8.3.5 Waste Emplacement Cycle

8.3.5.1 Waste Treatment

Radioactive wastes from the following sources are being considered for disposal in the Konrad mine (Section 8.3.9) or the Gorleben salt dome:

- Planned reprocessing plants in Germany as well as foreign plants (BNFL and COGEMA)
- Nuclear power plants
- Collection depots of the Federal States
- Nuclear research establishments
- Nuclear fuel cycle industry
- Decommissioning and dismantling of nuclear facilities
- Armed forces and medical industry.

The low-and intermediate-level wastes with negligible heat output will be immobilized with cement, bitumen or polystyrene in 200 or 400 L (55 or 110 gal) drums and several drums packaged in standardized cylindrical or rectangular containers or boxes for disposal underground. The heat generating high-level reprocessing wastes to be disposed of in the Gorleben salt dome include the following:

- Vitrified fission product solutions
- Cemented fuel element hulls and ends
- Cemented clarification fines.

High-level wastes will be stored for an interim period of about 10 to 30 years before final disposal.

In addition, the disposal of non-reprocessed spent fuel elements is also under consideration (Section 8.3.8) and a final decision on the detailed incorporation of non-reprocessed spent fuel into the Gorleben disposal program is still pending.

8.3.5.2 Waste Transport

Transport of the radioactive wastes is currently being carried out using rail and truck transport in specially designed shielded containers.

8.3.5.3 Waste Package

The low-and intermediate-level wastes with negligible heat output will be immobilized in cement, bitumen or polystyrene in 200 or 400 L (55 or 110 gal) drums and the drums will be packaged in concrete containers or boxes for underground disposal. High-level reprocessing wastes are to be immobilized in glass which will be cast into alloy or cast iron canisters. Ceramic materials are also under investigation as waste canisters.

8.3.5.4 Underground Waste Emplacement Sequence

The low- and intermediate-level wastes with negligible heat production will be stored in the vertical disposal caverns using several techniques tested at the Asse salt mine. These techniques include dumping, stacking and controlled lowering of the waste packages into the caverns. The heat producing wastes will be lowered into the boreholes drilled into the floors of the disposal drifts. Waste packages will be transported down shaft No. 2 to the various disposal areas in the repository.

8.3.5.5 Emplacement Schedule

Construction of the repository is expected to be complete around the year 2000 at which time the disposal operations are to commence. Repository operations are expected to last for about 50 years.

8.3.6 <u>Buffer and Sealing Materials</u>

Backfilling and sealing measures will be used to separate filled emplacement areas from the rest of the operating repository areas. Salt will be used as the basic backfill material for the emplacement areas. The emplacement rooms and drifts will be sealed with bulkheads, and the boreholes with plugs. Bulkheads will also be installed to seal entire disposal areas. At the completion of the repository operations, the entire underground facility will be backfilled and sealed. The shafts will then be backfilled and sealed in accordance with mine regulations to form a barrier against water inflow and radionuclide migration. Sealing materials for the shafts may include salt, concrete, and asphalt as water barriers.

8.3.7 Summary

8.3.7.1 Advantages

The advantages discussed for the other salt repository programs, such as Denmark and Holland, are also applicable to the German programs. These include:

- Favorable salt properties such as low water content and hydraulic conductivity, high thermal conductivity, great abundance in relatively homogeneous deposits, self-healing properties and low mining costs
- Massive salt domes can provide large barriers around the potential repository
- Large amount of salt mining expertise exists in Germany
- Many countries have or are conducting investigation programs in salt and thus a great deal of data exists on the behavior of the salt mass.

8.3.7.2 Potential Problems

Some major potential problem areas associated with disposal in salt domes were also noted for the other salt programs and these include:

- Salt is highly soluble, has poor sorptive characteristics and brine solutions are very corrosive
- The Gorleben salt dome is overlain by water-bearing formations and thus special shaft sinking and lining techniques are required
- Postclosure seals will have to ensure long-term isolation of the salt mass from the water

- Heat build-up may cause movement of the salt mass and this will require careful analysis and integration into the repository design.

8.3.7.3 Summary

Many countries are proposing radioactive waste disposal in salt formations, and due to the generally massive deposits, are considered one of the better repository host rocks. The Gorleben salt dome site investigation facility is currently under construction and, if events go according to schedule, the high-level repository should be one of the first to be operational in the world.

8.3.8 Alternative Spent Fuel Waste Disposal Concept

Current West German strategy also includes a thorough investigation of the final storage of spent fuel as an alternative to reprocessing, and indefinite storage of spent fuel at one or more away-from-reactor sites until the choice can be made between domestic reprocessing and spent fuel disposal. Several options are under consideration for the alternative of direct disposal of the spent fuel assemblies:

- Disposal in vertical boreholes drilled into the floor of the repository drifts
- Disposal in horizontal boreholes drilled from a repository drift
- Disposal by placing shielded containers in horizontal disposal drifts
- Disposal in deep boreholes drilled from the surface.

Of these, the disposal in boreholes in the floors of the repository drifts, and emplacement of shielded containers in horizontal drifts, appear to be the most promising. The vertical borehole technique would be similar to that described above for the high-level wastes. The horizontal emplacement would entail the placement of the spent fuel in shielded containers in repository drifts and backfilling around the containers as they are placed. Figure 8.3.11 shows a schematic of the technique.

8.3.9 The Low-and Intermediate-Level Concepts

The low-and intermediate-level waste disposal concept, which will be part of the proposed Gorleben repository, was described above. Until 1978,

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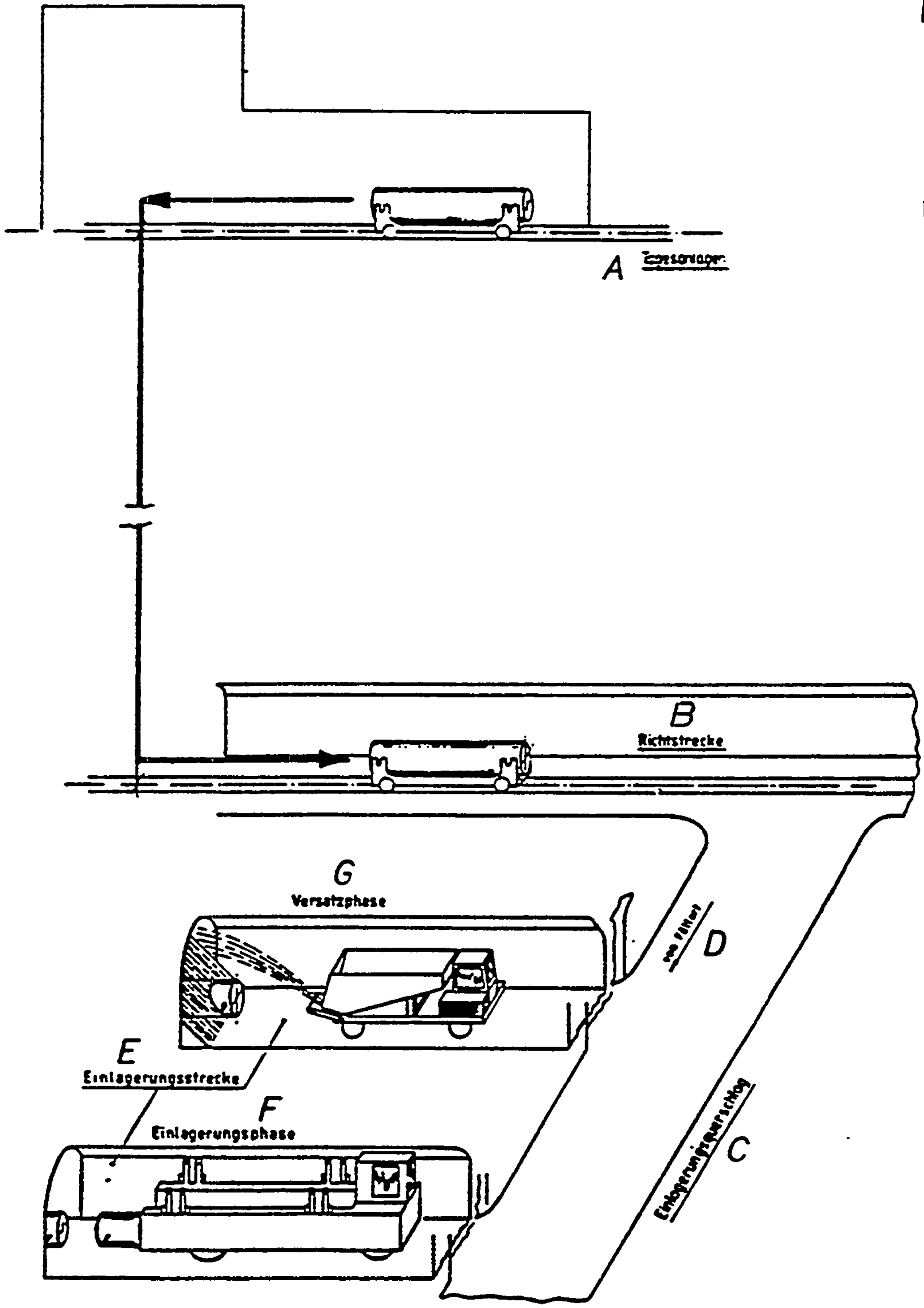

Emplacement diagram (source: DBE, TA 6)

A above-ground facilities
B development drift
C emplacement crosscut
D from filling station
E emplacement tunnel
F emplacement phase
G backfill phase

Figure 8.3.11. Spent Fuel Disposal Alternative

when the nuclear waste storage license expired, the Asse salt mine was used as the repository for low-and intermediate-level wastes as well as a research facility. The license has not been renewed to date and the mine is being used solely for research into various aspects associated with the disposal of all type of radioactive wastes. Another potential location for a low-and intermediate-level waste repository is currently under investigation at the former Konrad iron mine. Investigations have been carried out in the mine since 1975 and in 1982, based on the favorable results from these investigations, application was made for the approval of final disposal of non-heatproducing wastes in a repository constructed at the Konrad mine. If approval is granted, disposal activities could start in 1989. Figures 8.3.12 and 8.3.13 show a schematic of the proposed repository. Shaft 1 would be used as the ventilation intake shaft, as well as the hoisting shaft for the excavated iron ore, men and materials. Shaft 2 would be the ventilation exhaust shaft and provide access for the waste packages. Disposal operations and excavation operations will be separated from one another. The waste containers will be stacked in disposal caverns with a cross section of about 40 m^2 (430 ft^2). The caverns will be backfilled with material from the excavation operation. The Konrad mine, as currently envisioned, will allow excavation of storage caverns for about 650,000 m^3 (850,000 yd^3) of waste. This would mean that during its operational period of about 20 years, the mine could hold about 95% of all the radioactive wastes produced in West Germany. It has also been suggested that in case the Konrad repository cannot hold any excess wastes, the Asse mine may be reopened again for waste disposal.

8.4 SALT: THE UNITED STATES CONCEPT

This Chapter and Chapters 9 and 10 which follow describe the three repository concepts being pursued in the United States. Since 1957, when the National Academy of Sciences concluded that geologic disposal would be the most desirable method for permanent disposal of nuclear wastes, many alternatives have been considered. Studies have been conducted in different geologic media, and large amounts of data collected, analyzed and compared. The passage of the Nuclear Waste Policy Act of 1982 provided a framework for achieving the disposal of the nation's spent nuclear fuel and high-level wastes. The Act authorized one repository with a capacity of 70,000 MT (77,000 tons) of wastes, and proposes a second repository should the need arise. The repositories will be used for disposal of commercial high-level wastes or spent fuel produced by nuclear power generation facilities, as well as the small amount of defense high-level wastes. It also requires the preparation of proposals for a Monitored Retrievable Storage facility and established a funding mechanism for the waste disposal program. Several regions in the United States were under investigation and in February 1983, the Department of Energy fulfilled the first requirement of the Act by

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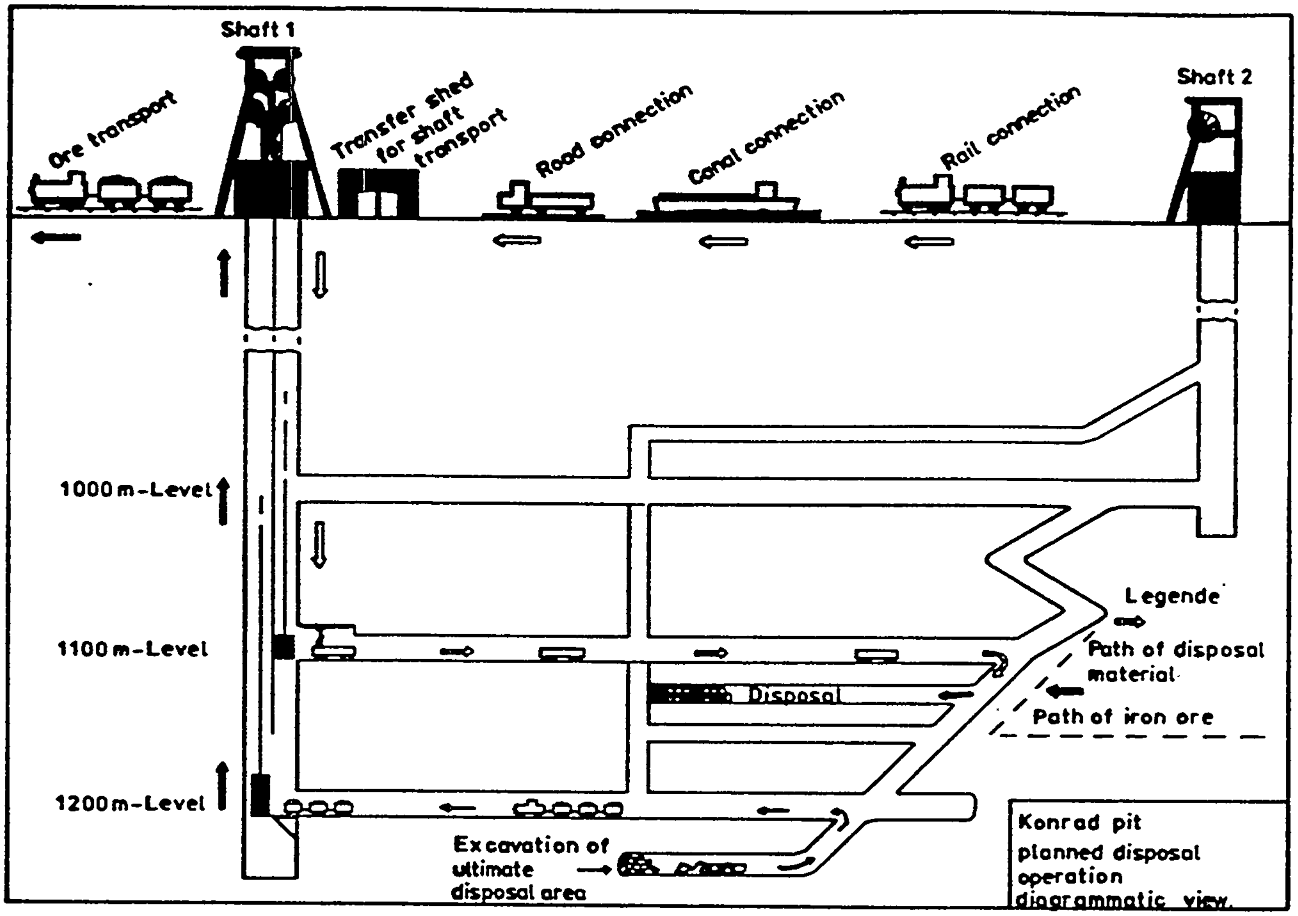

Figure 8.3.12. Konrad Repository Layout

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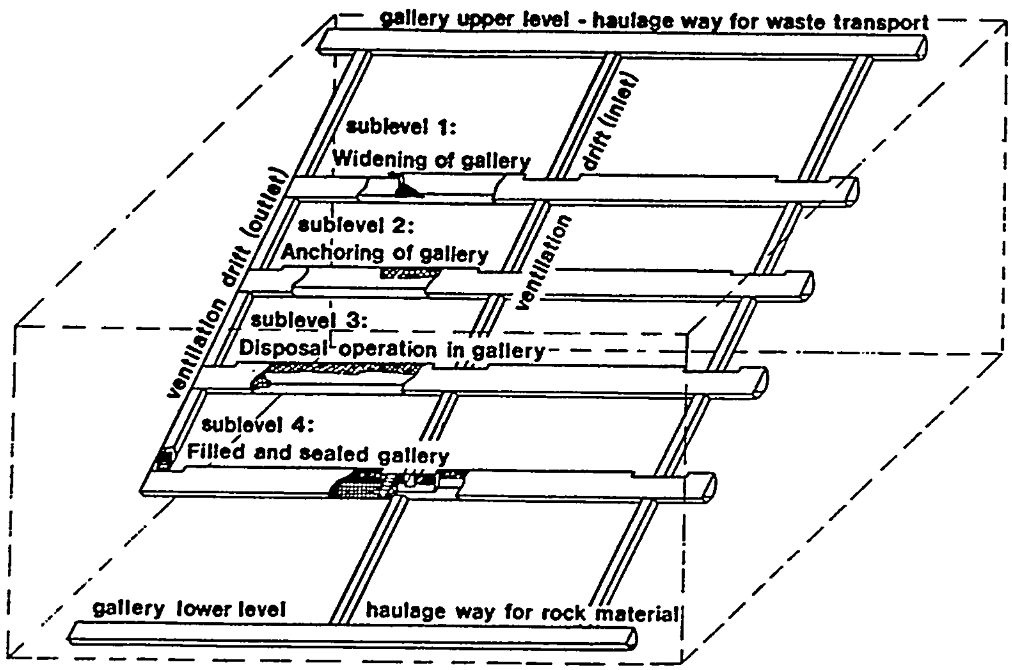

Figure 8.3.13. Waste Disposal Concept at Konrad

formally identifying nine sites as potentially suitable repository locations in three different geologic media as shown in Figure 8.4.1:

1. Vacherie Dome, Louisiana (Domal Salt)

2. Cypress Creek Dome, Mississippi (Domal Salt)

3. Richton Dome, Mississippi (Domal Salt)

4. Yucca Mountain, Nevada (Welded Tuff)

5. Deaf Smith County, Texas (Bedded Salt)

6. Swisher County, Texas (Bedded Salt)

7. Davis Canyon, Utah (Bedded Salt)

8. Lavender Canyon, Utah (Bedded Salt)

9. Reference Repository Location, Hanford Site, Washington (Basalt Flows).

On May 28, 1986, the Department of Energy announced presidential approval of the nomination and recommendation for site characterization of five of the nine sites in Washington, Texas and Nevada. Environmental assessments were published at the same time for the five recommended sites. These sites are Richton, Deaf Smith County and Davis Canyon in salt, Yucca Mountain in tuff, and Hanford in basalt. Approval was granted for the construction of exploratory shaft facilities at three of these sites:

1. Deaf Smith County, Texas

2. Yucca Mountain, Nevada

3. Hanford, Washington.

Designs for these exploratory shaft facilities are currently in preparation, and construction is scheduled to commence in the 1988-89 time frame. During the site characterization activities at each site, the Department of Energy will consult closely with the Nuclear Regulatory Commission, responsible for licencing the repository, and affected states and Indian tribes. After several years of site characterization, the Department of Energy will be able to recommend one of the sites to the president as the location for the first repository. After approval by Congress, and state and local affected groups, a Construction Authorization Application will be submitted to the Nuclear Regulatory Agency. The agency will have three years to review the application and resolve any issues with

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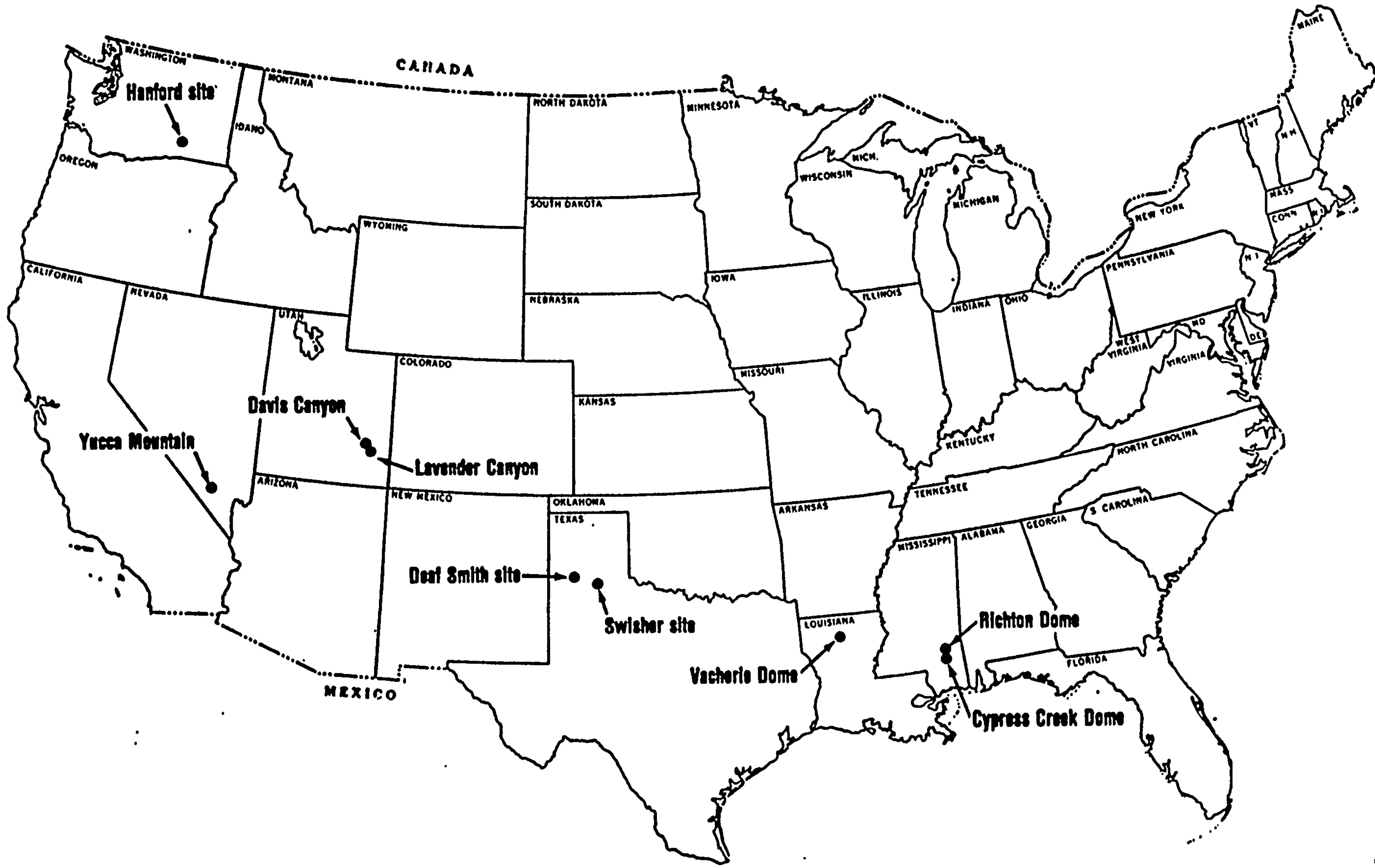

Figure 8.4.1. Potentially Acceptable First Repository Sites

the Department of Energy. Subject to approval by the Nuclear Regulatory Agency, repository construction is planned to start in the mid-1990s to meet the 1998 date of start of actual waste disposal as scheduled by the Nuclear Waste Policy Act.

Work is also being carried out on the second repository program. Crystalline rocks were investigated as a possible location for this repository. An Area Recommendation Report was published in 1986 and current activities include resolution of comments received on the report and technology development activities.

The 70,000 MT (77,000 tons) of waste represents the maximum storage capacity of the first repository and is made up of several types of nuclear waste. Commercial reprocessing is currently not practiced, and thus the reference repository described in this chapter is based on the disposal of about 62,000 MT (68,000 tons) of commercial spent fuel, and about 8,000 MT (9,000 tons) equivalent of reprocessed wastes. These reprocessing wastes are mainly defense high-level wastes (90%) and some reprocessing wastes (10%) remaining at the shut-down West Valley commercial reprocessing plant.

The rest of this chapter describes the reference repository for the Deaf Smith County, Texas site, published in the Environmental Assessment in 1986.

8.4.1 Geographic Location

The Deaf Smith County site is located in the north-central part of Deaf Smith County, in the Southern High Plains of the Texas Panhandle (Figure 8.4.2). The High Plains is nearly flat, relatively undissected tableland. The site is square with 4.8 km (3 mi) on a side and located in a rural setting with a population density of about 4 persons per square mile. Amarillo is the regional population center. Highway, rail and airplane access exists near the site. Most of the land in Deaf Smith County is cropland, although some of the land within the site is rangeland.

8.4.2 Summary of Geologic Setting

Figure 8.4.3 shows a generalized stratigraphic column at the Deaf Smith site. The column was extrapolated from Department of Energy wells drilled in Deaf Smith County. The Palo Duro Basin in the Texas Panhandle is a greatly sloping depression between the Amarillo uplift and the Matador Arch that is composed of relatively uniform sedimentary formations. The formations to be encountered in shaft construction consist of variably cemented sediments overlying generally flat-lying (less than 5° dip)

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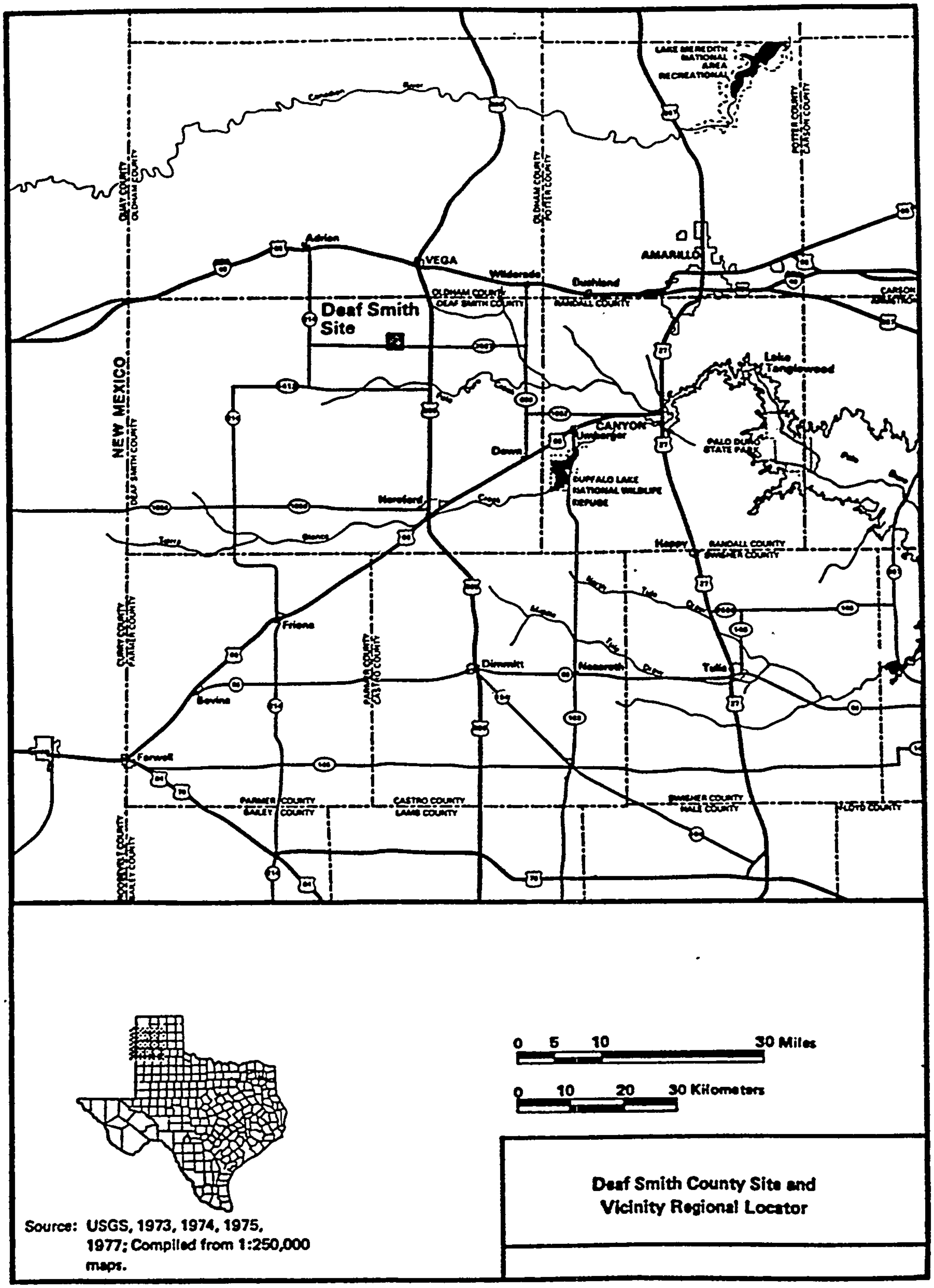

Figure 8.4.2. Salt Exploratory Shaft Site

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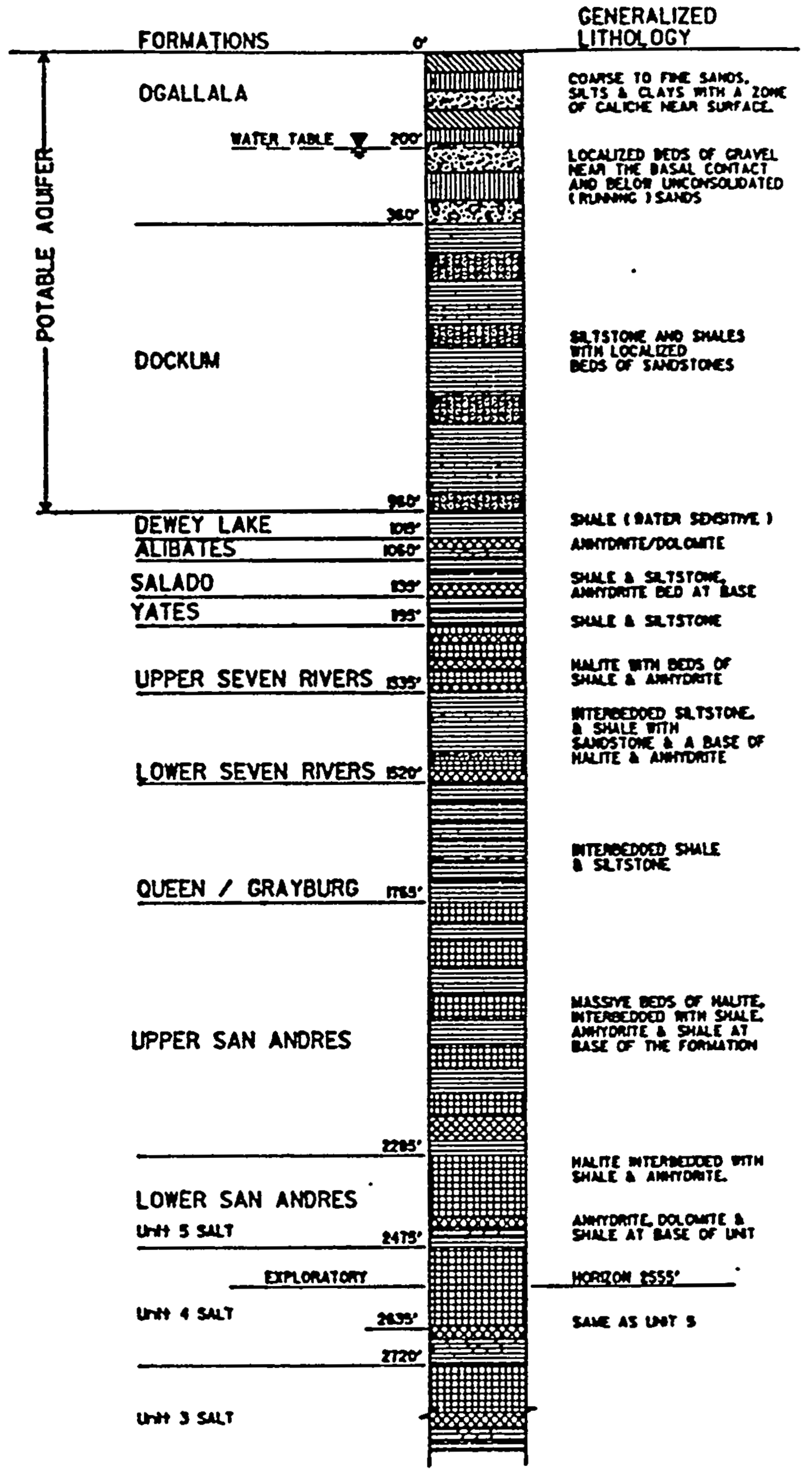

Figure 8.4.3. Generalized Stratigraphical Column, Deaf Smith County

sandstones, siltstones, shales, limestones, dolomites, and evaporite sequences ranging in geologic age from Upper Permian to Pliocene. The following is a brief summary of the lithology anticipated:

- The Ogallala formation is a major freshwater aquifer and extends from the surface to a depth of about 110 m (360 ft). It consists of variably cemented fine sands, silts, clays, and thin caliche beds grading to coarse sands and gravels near the base of the formation. The sediments are weakly cemented in some locations but are generally unconsolidated.

- The Dockum formation consists of about 290 m (600 ft) of variably indurated siltstones with localized beds of sandstones and shales in the mid-to-lower region of the formation. The sandstone is generally cemented weakly. The Santa Rosa sandstone yields large fresh water quantities in localized basin areas.

- The Dewey Lake formation underlies the Dockum group and consists primarily of siltstones and can include some shale beds.

- The Alibates formation consists mainly of anhydrite beds with some interbedding of shales and siltstones.

- The Salado and Yates formation consists of siltstones interbedded with shales.

- The Upper Seven Rivers formation is about 40 m (140 ft) thick and contains the first significant salt beds. It consists of interbedded siltstones, shales and some anhydrite. The Lower Seven Rivers formation is about 55 m (185 ft) thick and consists of beds of siltstones and shales with the possibility of some halite and anhydrite.

- The 75 m (245 ft) thick Queen/Grayburg formation generally consist of interbedded shales and siltstones and may include some halite bands.

- The 230 m (520 ft) thick Upper San Andres formation is made up of the cyclic evaporite sequences of limestone, dolomites, anhydrites and halite. Some siltstones and shales may also be present.

- The Lower San Andres Unit 5 salt is expected to start at a depth of about 700 m (2,285 ft) and consists of about 30 m (100 ft) of halite interbedded with mudstone to shale seams. The Unit 5 salt is separated from the Unit 4 salt below by anhydrites, dolomites and shales.

- The top of the Lower San Andres Unit 4 salt is expected to be at a depth of about 755 m (2,475 ft). The salt sequence is anticipated to be about 50 m (150 ft) thick and interbedded with mudstone and shale seams of less than 2-3 cm (1 in) thickness at a spacing of about 1.5 to 3 m (5 to 10 ft). The exploratory facility is planned to be located at a depth of about 780 m (2,555 ft).

8.4.3 Repository Concept Description

8.4.3.1 Nuclear Waste Disposal Strategy

Current strategy in the Unites States is to dispose the spent fuel directly in one or two geologic repositories. There are approximately 100 operating nuclear power reactors in the Unites States at this time. Reprocessing is not practiced at this time and the reference disposal concept is based on the disposal of about 62,000 MT (68,000 tons) of unreprocessed spent fuel. Disposal of defense high-level wastes and reprocessing wastes remaining from the operation of the closed-down West Valley plant, a total of about 8,000 MT (9,000 tons) equivalent, will, however, also require disposal in the repositories. Disposal is to begin in 1998 after an extensive site investigation program. This is currently in process at three potential sites. The first repository is to be constructed at one of these sites. Predictions of spent fuel accumulations will determine the start date of the second repository although technology development programs and area studies are in process. Other options that may also be included in the repository system may be the phase start-up of the disposal operations and a Monitored Retrievable Storage facility. Phase start-up would mean an initial waste disposal rate of about 400 MT (450 tons) per year which would be increased to the required rate of about 3,000 MT (3,400 tons) per year in 2001.

The National Waste Policy Act mandates that the repository shall be designed and constructed to permit retrieval of the waste packages emplaced in the repository on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated. The capability to retrieve the waste packages would have to be demonstrated prior to a decision to backfill the waste storage rooms, and would be maintained regardless of whether the storage rooms have been backfilled. For design purposes, it was assumed that the actual retrieval, if retrieval proves to be necessary, would take approximately as long as the period used for waste emplacement and repository construction.

Waste age for the reference repository design has been assumed to be 10 years. This is a conservatively low estimate of the expected average age of the fuel, and future design activity will refine this estimate. The range of the age of the fuel that will arrive at the repository will vary, depending on the growth of nuclear power demand, and is from 5 to 40 years.

In the United States, radiation protection standards which the geologic disposal system must meet are given in the draft Environmental Protection Agency (EPA) standards of 40 CFR Part 191 (EPA, 1985). Limits are established for the cumulative releases of specific radionuclides over a period of 10,000 years that are intended to limit cumulative doses to the general public. Based on these standards, the Nuclear Regulatory Commission has determined the technical requirements that must be met in order to satisfy the EPA standards (NRC, 1986). These include a requirement for substantially complete containment for a period of at least 300 and up to 1000 years after permanent closure of the repository; and a requirement that the release rate of any radionuclide from all the waste packages, following the containment period, shall not exceed 1 part in 10^5 per year of the curie inventory of that radionuclide calculated to be present at 1,000 years following closure, excepting radionuclides released at a rate less than 10^{-8}/year of the original waste inventory that still remains 1,000 years after emplacement. While not specified, in practice the containment requirement is generally applied to the container; and the controlled release rate requirement to the waste package.

8.4.3.2 Surface Facilities

Surface facilities will include the repository facilities and, if found appropriate, a Monitored Retrievable Storage facility located at some strategic location in the United States.

A Monitored Retrievable Storage facility (Figure 8.4.4) would serve as a centralized spent fuel and nuclear waste consolidation and packaging facility with storage capacity. Spent fuel from the nation's commercial nuclear power reactors would be shipped to the Monitored Retrievable Storage facility. In the facility, the spent fuel would be prepared for final disposal and then shipped by dedicated transport to a geologic repository. If necessary, interim storage of some of the spent fuel could also be provided. Conceptual designs for a Monitored Retrievable Storage facility using two storage concepts are being completed. Storage in sealed storage casks is the preferred option and is shown in Figure 8.4.5. A sealed storage cask is a large, steel-lined reinforced concrete cylinder, holding steel canisters of spent fuel, sealed with a thick concrete shield plug and welded steel lid. The wall thickness is about 1 m (3 ft), height and diameter about 7 m (22 ft) and 3.5 m (12 ft), respectively, and the cask has weight of about 220 tons. The alternate storage method of field drywell storage uses in-ground, dry, sealed, metal canisters for storing the spent fuel canisters. Each drywell would hold one spent fuel canister and would

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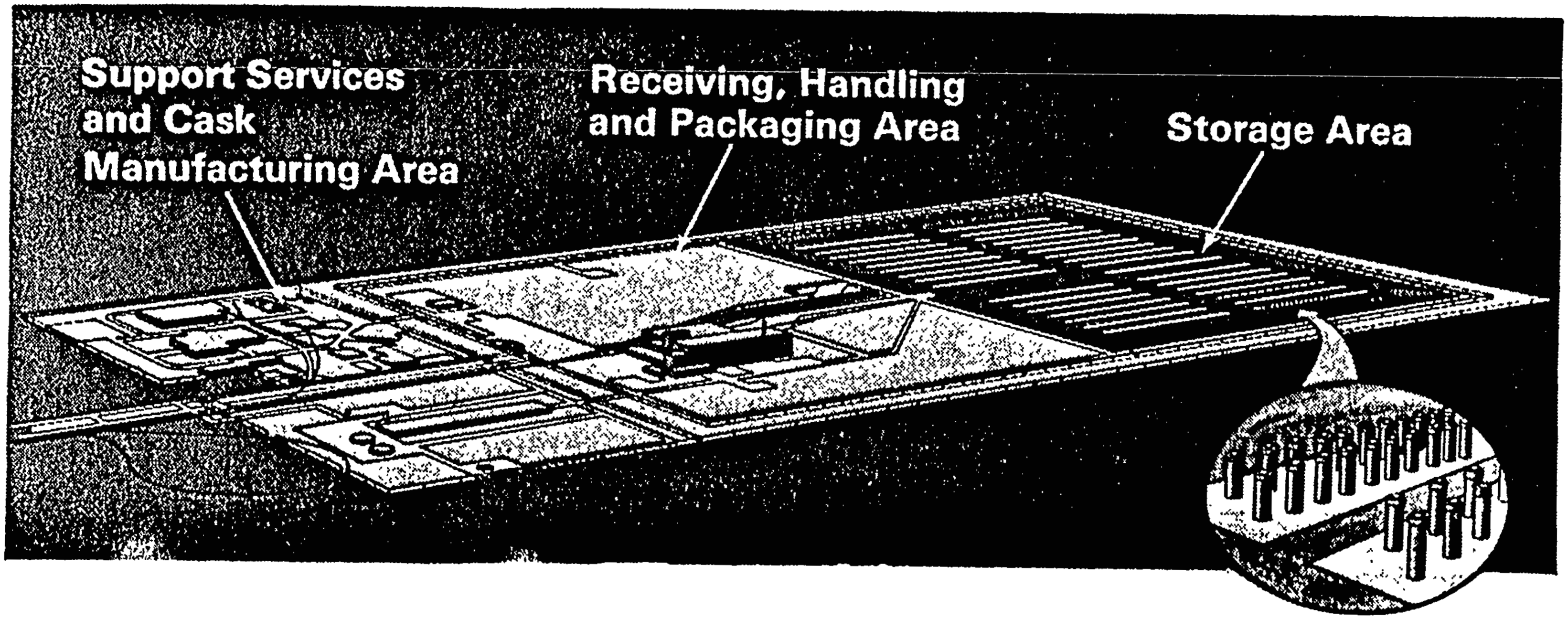


Figure 8.4.4. Artist's Concept of a Monitored Retrievable Storage

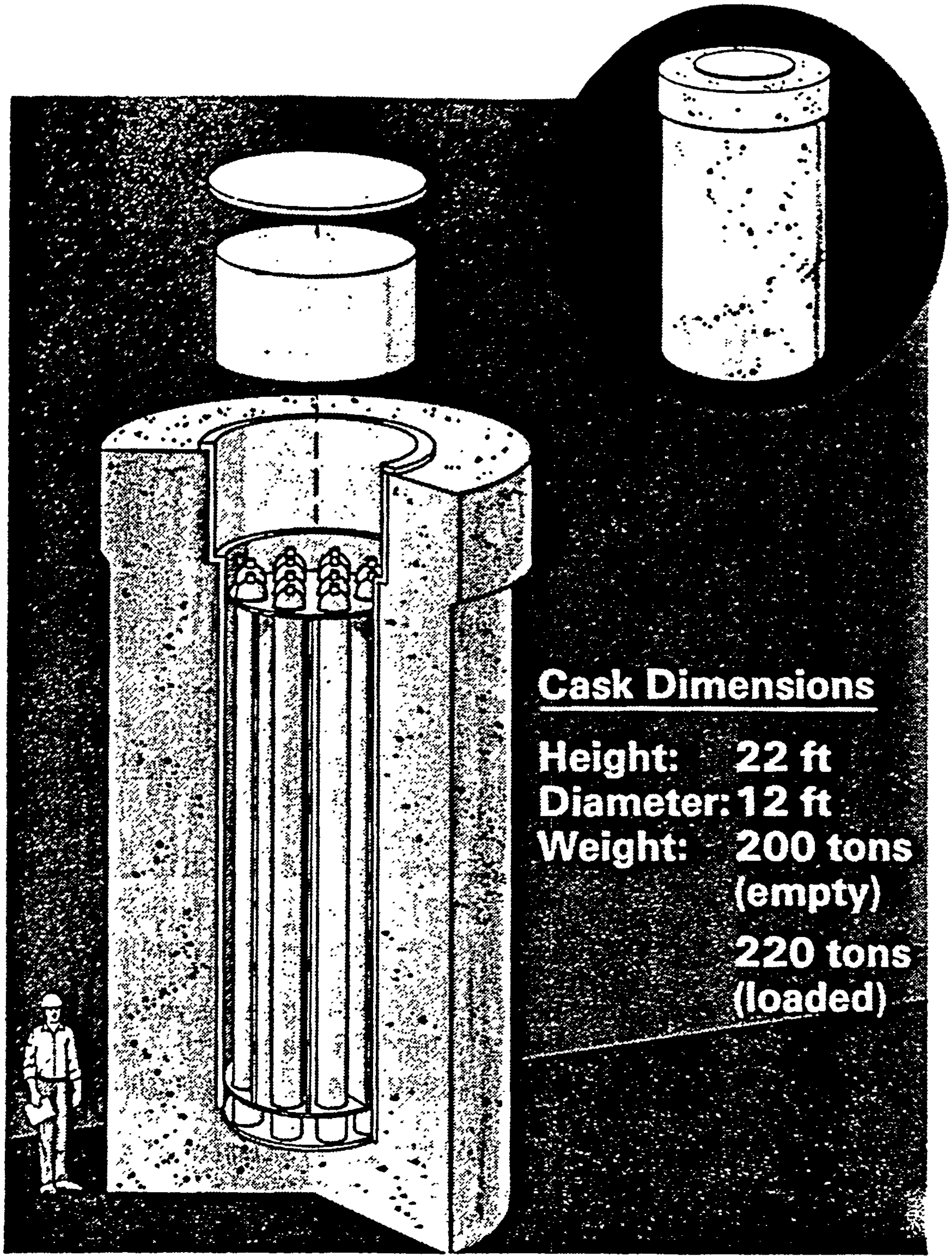


Figure 8.4.5. Artist's Concept of a Sealed Storage Cask

be about 6 m (20 ft) deep (Figure 8.4.6). Other installations at the Monitored Retrievable Storage facility will include the waste receiving and handling building and support services. The option of including a Monitored Retrievable Storage facility in the waste disposal system is still under review.

Repository surface facilities (Figure 8.4.7) will include the following four major components:

- Waste handling building
- Mine support facilities (headframe, salt storage pile, ventilation exhaust, and service buildings)
- Administration and support buildings
- Utilities and utility support facilities (power supply, work shops, waste water treatment).

These surface components will be located on approximately 165 hectares (408 acres) of land, exclusive of the access corridor (i.e., road, rail, and utilities). The underground areas will extend beyond the surface facility, and therefore a controlled area of about 23.3 square kilometers (9 square mi) will be established. Road and rail access to the site will be designed and constructed to support the expected shipments of personnel, material, and waste. Offsite acquired utilities, such as electrical power and natural gas, will run to tie-in points to service the repository. During construction, topsoil will be removed, used for onsite landscaping, or stockpiled for future restoration. During construction and operation, excavated salt will be stockpiled in a storage area designed to minimize runoff. A perimeter fence will enclose the entire surface facility with the exception of the visitor center and parking lot. An inner security fence will surround the radiological operations areas including the shafts, the receiving and packaging facility, and other waste-related support facilities. Figure 8.4.8 shows the repository site layout plan.

The primary functions of the surface waste receiving, handling, and packaging facilities are to receive, handle, and package the nuclear wastes arriving at the repository and to transfer the packaged waste to the repository subsurface. The waste handling and packaging facility (Figures 8.4.8 and 8.4.9) will be the largest building on the site and will be adjacent to the waste shaft headframe. Access to the facility will be by both road and rail. The waste handling and packaging facility will provide for the following operations:

- Washing down contaminated shipping casks
- Unloading the cask from the transport vehicle

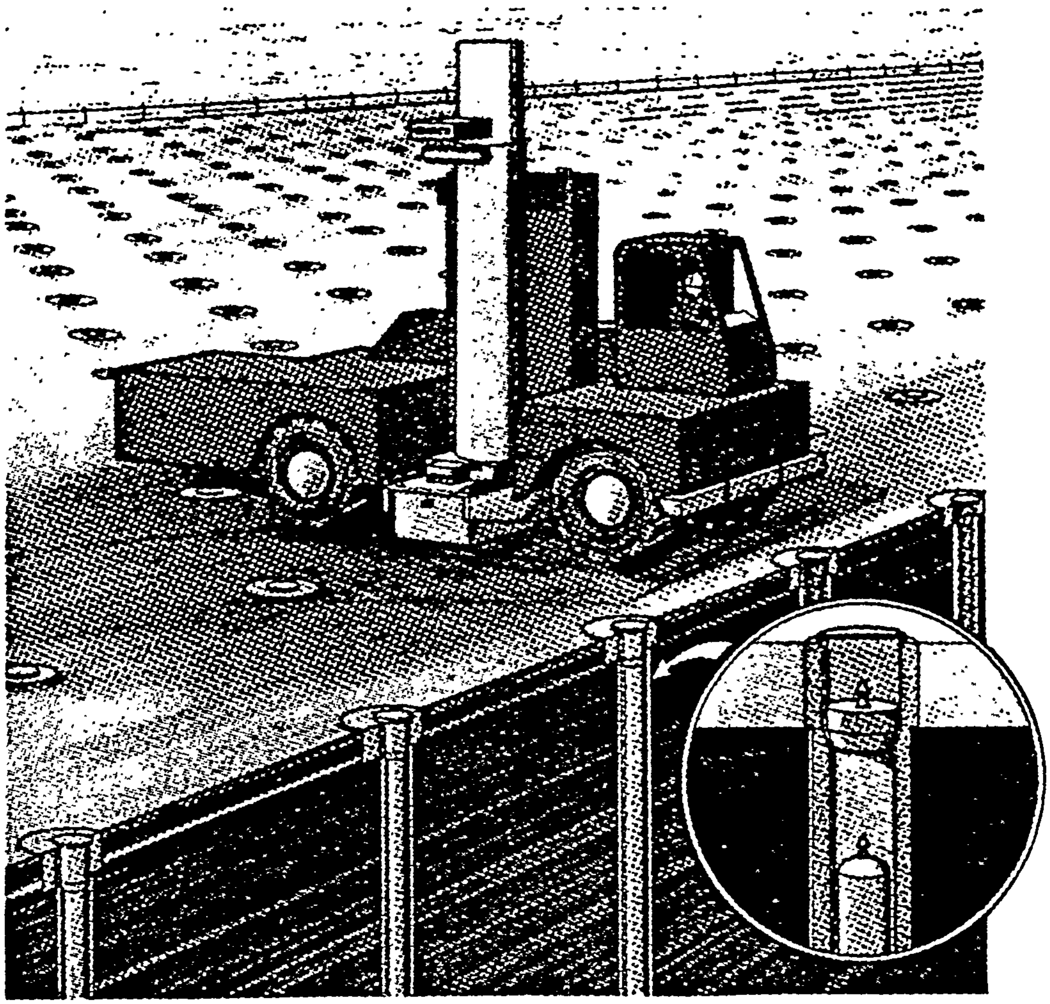
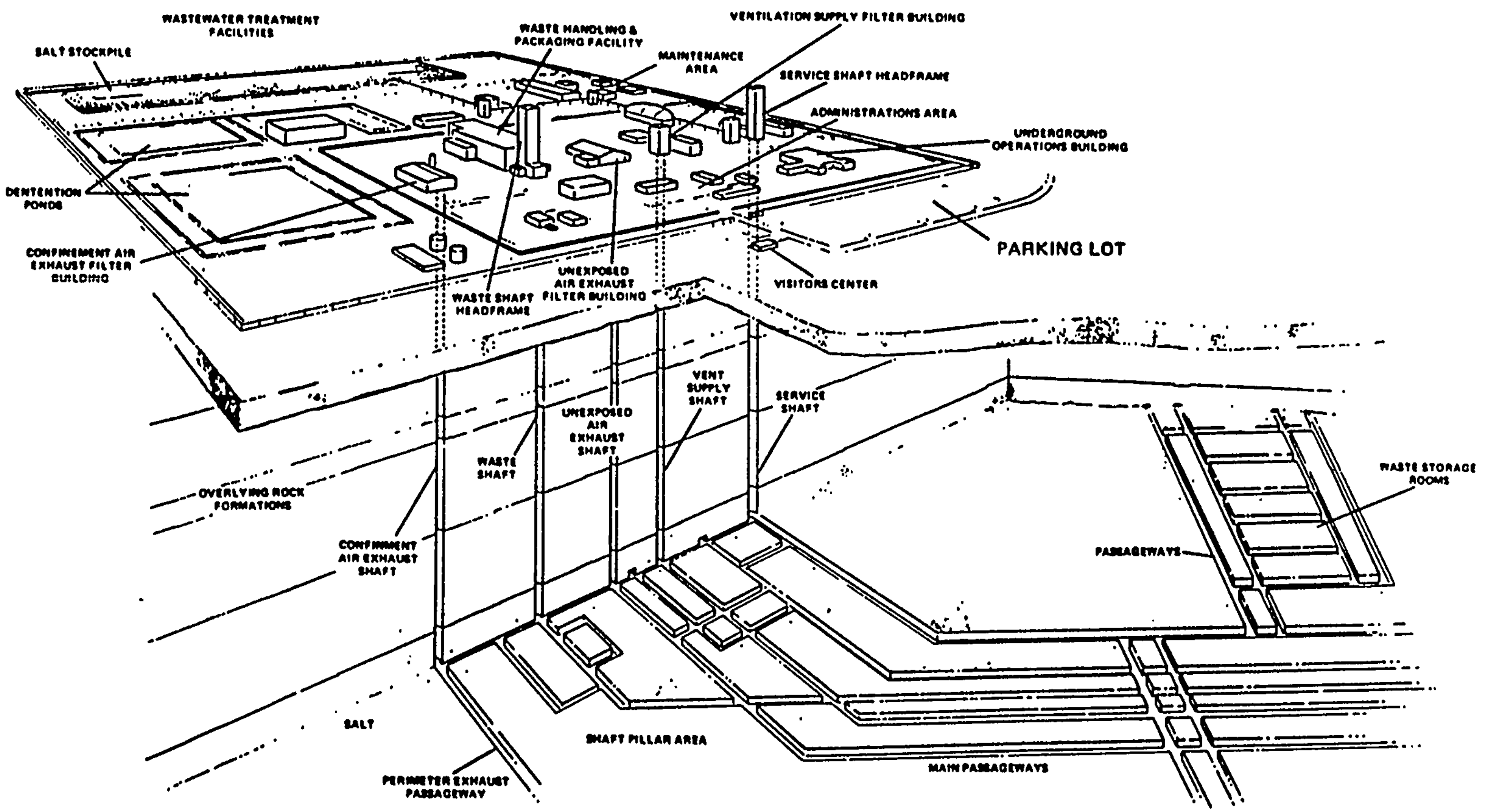


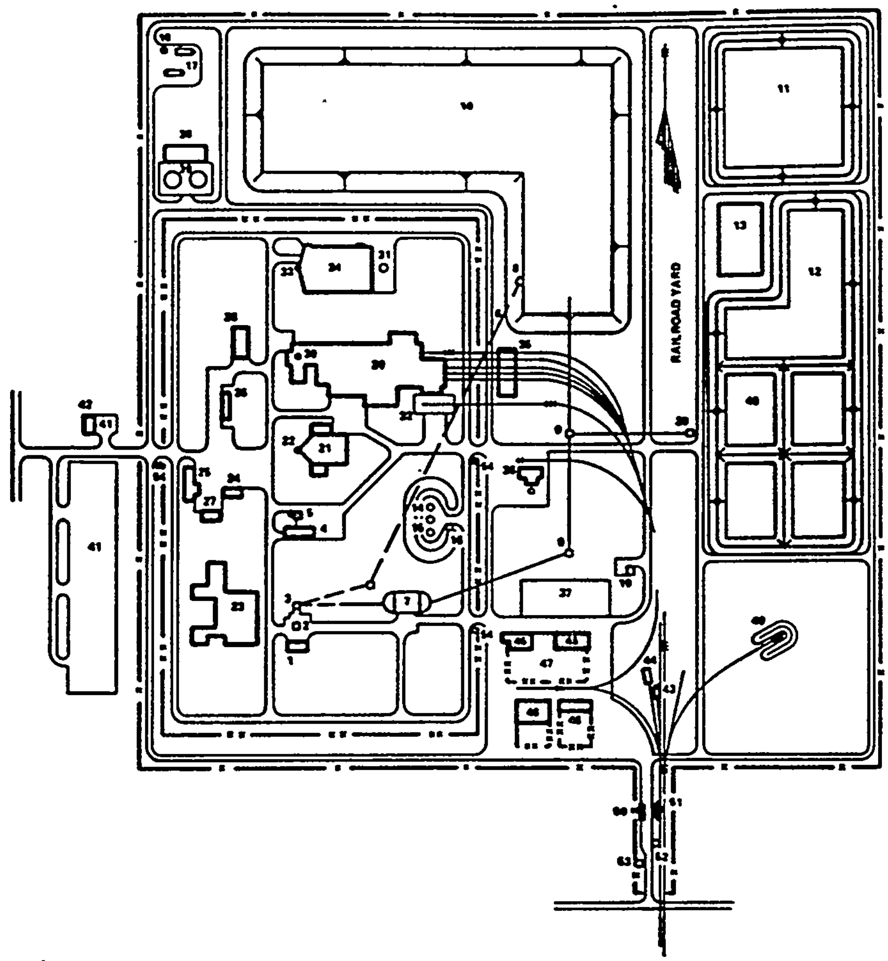
Figure 8.4.6. Artist's Concept of Drywell Storage



Source: Adapted from SCC, 1984a.

Nuclear Waste Repository Facility Concept

Figure 8.4.7. Salt Repository Concept



Explanation

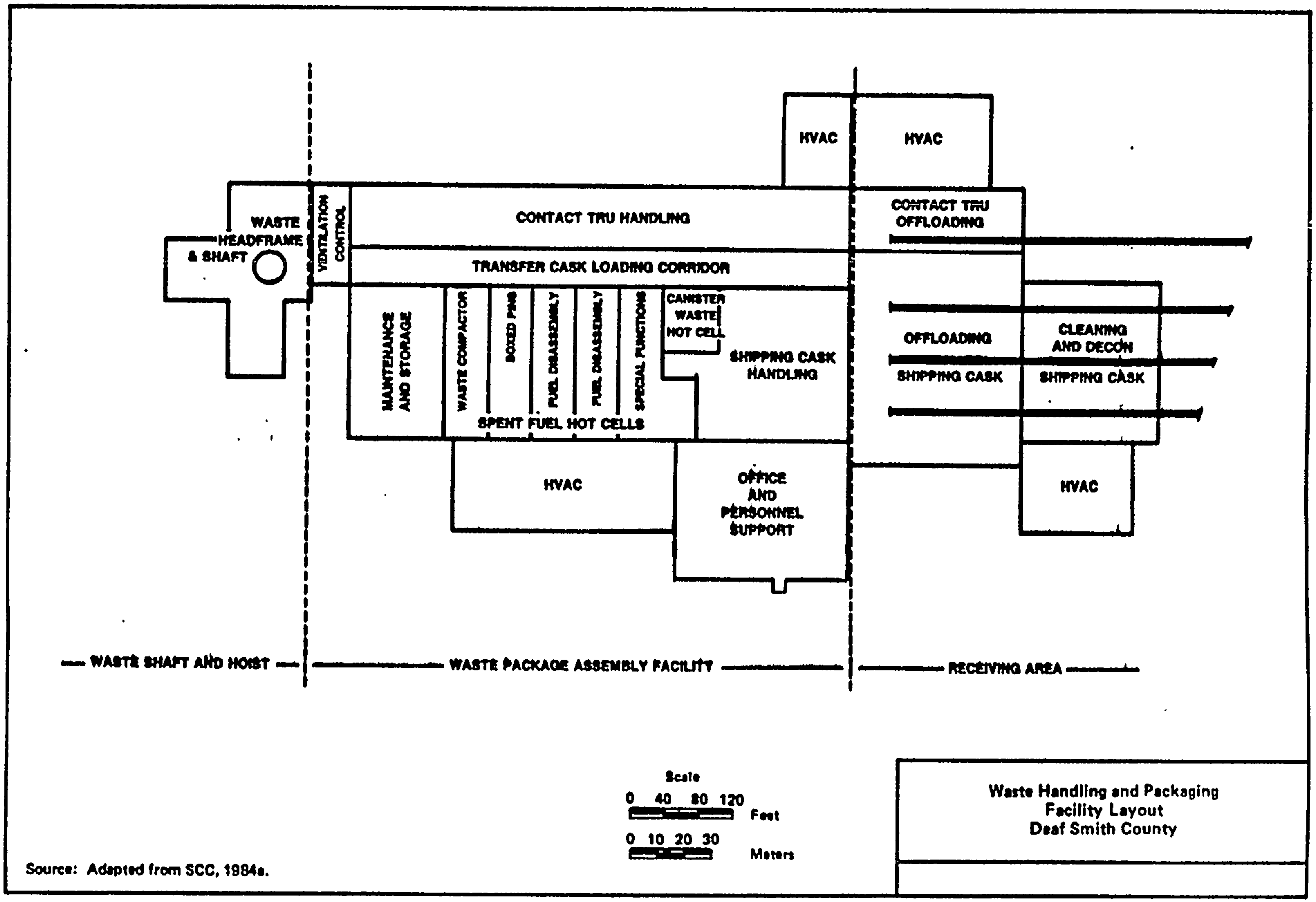
1 Service shaft filter building
2 Service shaft
3 Transfer tower
4 Ventilation supply filter building
5 Ventilation supply shaft
6 Reclaim tunnel
7 Live storage & surge building
8 Reclaim hopper
9 Salt transfer tower
10 Salt stockpile
11 Salt run-off pond
12 Treated wastewater holding pond
13 Wastewater treatment facilities
14 Potable water tank
15 Fire water storage tanks
16 Fire/potable water pumphouse
17 Cooling towers
18 Chemical feed building
19 Railroad control tower
20 Salt loadout structure
21 Unexposed air exhaust filter building
22 Unexposed air exhaust shaft
23 Underground operations building
24 Firehouse, security & medical treatment center
25 Administration & operation control center
26 Electric power building
27 Environmental & instrument laboratory
28 Compressor chiller building
29 Waste handling & packaging facility
30 Waste shaft
31 Confinement exhaust stack
32 Storage yard
33 Confinement air exhaust shaft
34 Confinement air exhaust filter building
35 Non-contaminated railcar/truck washdown area
36 Steam plant
37 Abandoned material & equipment storage yard
38 Raw water treatment
39 Raw water storage tanks
40 Storm water detention ponds
41 Parking lot
42 Visitors center
43 Railroad inspection pit
44 Railroad maintenance building
45 Maintenance building
46 Storage shed
47 Storage yard
48 Warehouse
49 Suspect Railcar/truck storage area
50 Truck inspection pit
51 Railcar inspection pit
52 Monitoring station
53 Guardhouse

Source: Adapted from SCC, 1984a.

Scale
0 500 1000 Feet
0 150 300 Meters

Repository Site Layout Plan
Deaf Smith County

Figure 8.4.8. Salt Repository Surface Facilities



Source: Adapted from SCC, 1984a.

Figure 8.4.9. Waste Handling and Packaging Facility

- Removing the waste from the shipping cask
- Inspecting and recording identification of waste forms
- Temporarily storing waste packages
- Packaging waste forms
- Welding waste packages closed
- Inspecting completed waste packages
- Transferring waste packages in a transfer cask to the waste shaft.

The functions required to receive the shipping casks from the rail and truck transport vehicles and to prepare the waste form for underground emplacement, up to and including the transfer to the waste shaft hoist, will be performed within the confines of the waste handling and packaging facility. The overall functional-flow schematic diagram for the waste handling and packaging facility is shown in Figure 8.4.10. Waste handling activities begin with the receipt of the transport vehicle from the cask contamination survey station. As necessary, the casks will be reinspected and contaminated casks will be washed down in a separate decontamination wash bay. Casks will then be unloaded from the transport vehicle and positioned for removal of the waste forms. With the exception of contact-handled transuranic wastes, all the waste forms will be removed from the shipping casks into the hot cell portion of the facility where the waste forms will be inspected, sorted, and temporarily stored. Contact-handled transuranic waste is not as radioactive as spent fuel or high-level waste and will be handled outside the hot cells. It will be unloaded, inspected, and placed on pallets before being transferred to the waste hoist. The functions performed within the hot cells will depend on the waste type. Spent fuel assemblies will be dismantled and spent fuel rods will be repackaged. These packages and canisters of commercial and defense high-level wastes will be placed in containers. The canisters will pass out of the hot cell area via transfer casks used to move the waste packages to and down the waste hoist, and through the underground workings to the emplacement area.

The waste handling and packaging facility will have several support areas not directly involved in handling radioactive waste. The most significant of these in size and importance will the heating, ventilating, and air conditioning system, which will constitute the major engineered safety system for the containment of the radioactive materials.

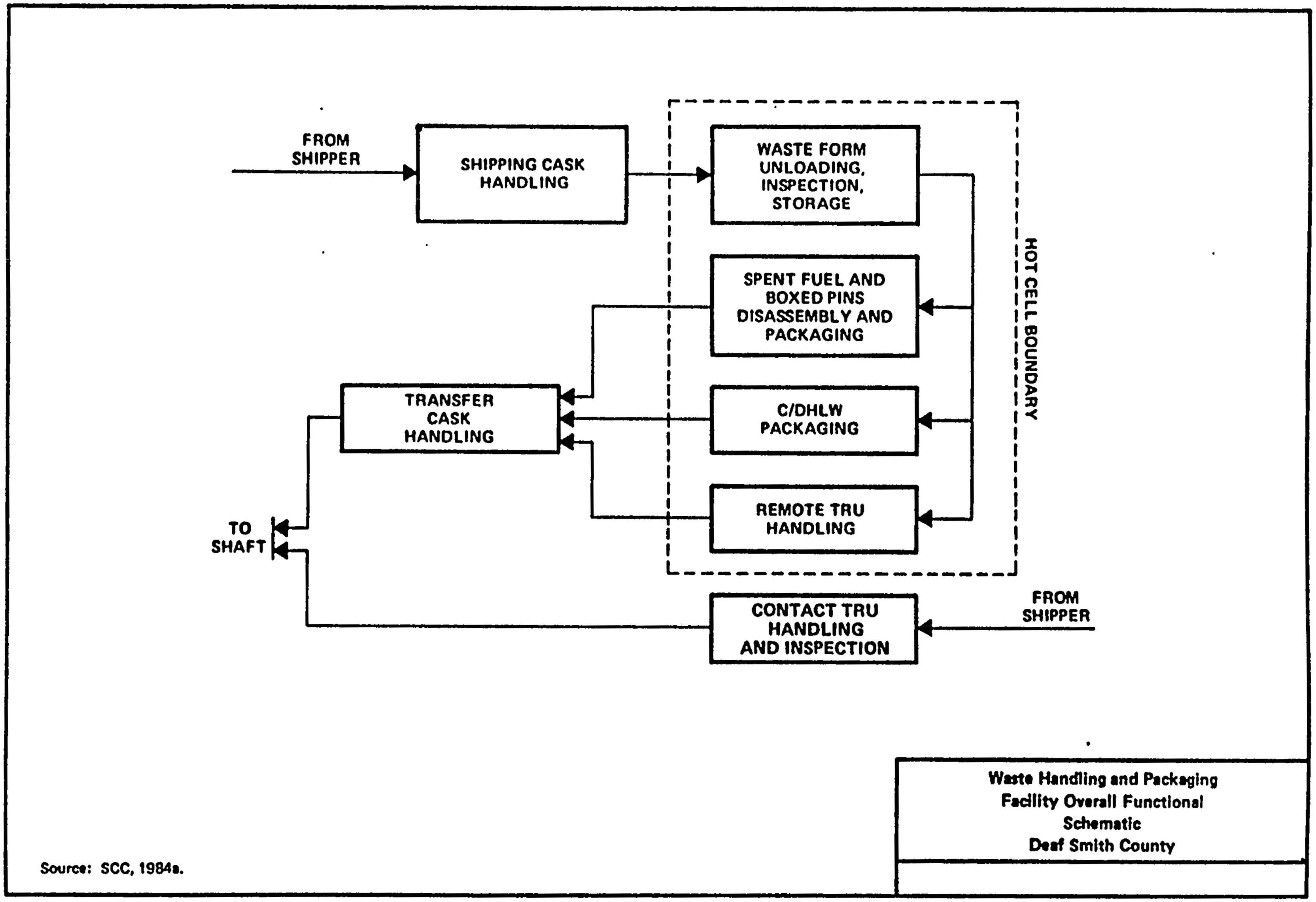


Figure 8.4.10. Waste Handling Functional Schematic

8.4.3.3 Underground Configuration

8.4.3.3.1 Repository Shafts. The repository shafts will provide the means for ingress and egress of personnel, transport of materials and nuclear waste packages, and ventilation of the underground workings of the repository. The reference design (Figure 8.4.11) proposes five shafts to serve the requirements of the repository:

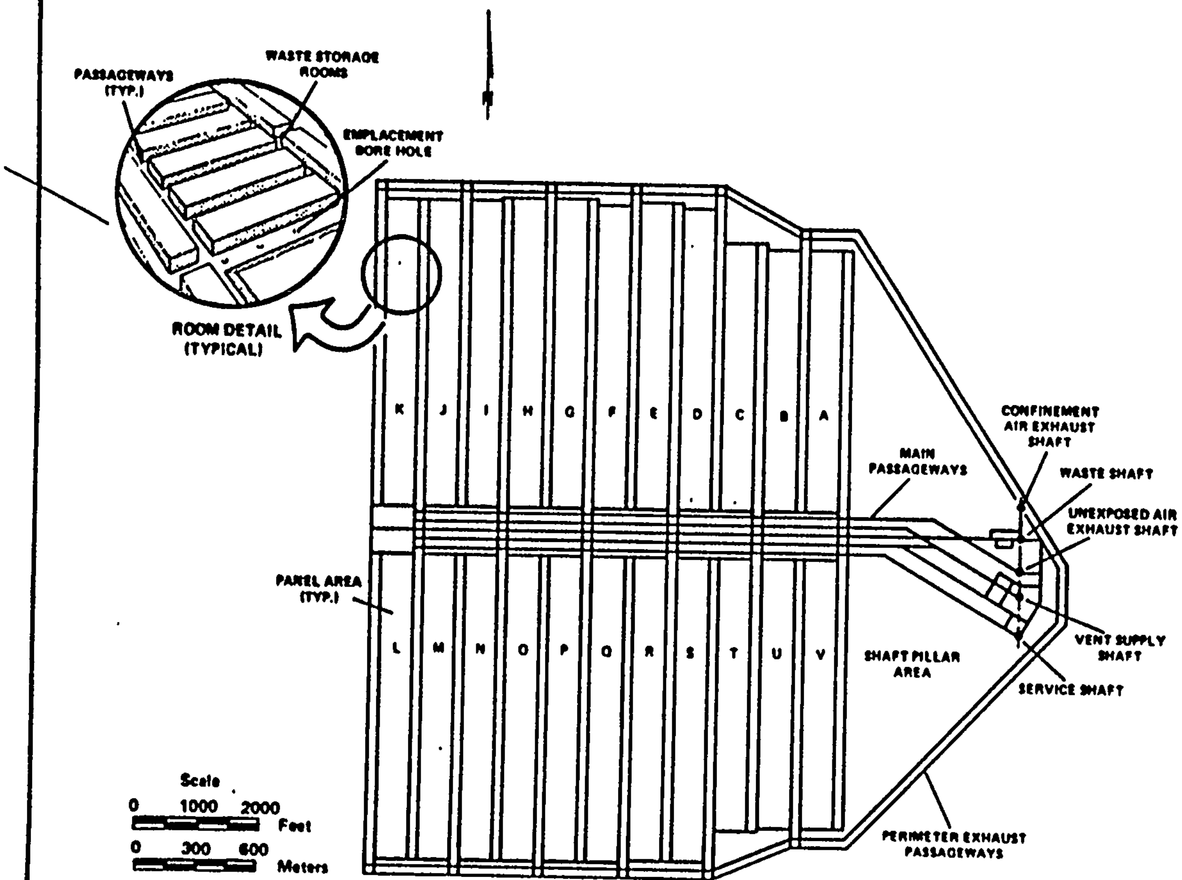
- Waste shaft - transport of waste containers from surface to underground
- Service shaft - transport of personnel, equipment, materials, salt, supply air, and utility lines
- Ventilation supply shaft - supply air, utility lines, and emergency egress
- Unexposed air exhaust shaft - exhaust air from the underground development area, utility access
- Confinement air exhaust shaft - exhaust air from the waste emplacement areas.

Consideration is being given to utilizing the two shafts that will be constructed during the site characterization phase as repository shafts. These shaft will have a smaller diameter (3.6 m or 12 ft) and therefore could perhaps replace one of the ventilation shafts.

All shafts will be circular in cross section and range in size from 6.4 to 9.4 m (21 to 31 ft) finished inside diameter. The depth of the shafts in the repository horizon will vary from the repository depth to 107 m (350 ft) below the repository depth to accommodate the excavated salt hoist skips in a loading position, the salt spillage storage area, and hoist equipment. The repository depth will most likely be the same as the proposed exploratory shaft facility depth which is estimated to be about 780 m (2,555 ft) at this time.

The shaft pillar, which encompasses all five shafts, is an area containing little excavation and no emplacement of nuclear wastes. It will be offset or placed to one side of the underground excavations. This shaft pillar offset will isolate shaft shafts and main surface facilities from any potential effects of subsidence or uplift over the storage room panels.

The shaft liner designs and installation will take into account the site-specific stratigraphy and major aquifers. Seals will be incorporated to isolate the aquifers from the waste storage level and from each other.



PANEL TABLE

PANEL	WASTE	NO ROOMS
A	TRU	46
B	TRU	46
C	DHLW/CHLW	25/29
D	CHLW	56
E	CHLW	57
F	SF	73
G	SF	74
H	SF	74
I	SF	73
J	SF	73
K	SF	73
L	SF	73
M	SF	73
N	SF	73
O	SF	73
P	SF	73
Q	SF	73
R	CHLW	56
S	CHLW	56
T	DHLW/CHLW	26/29
U	TRU	47
V	TRU	47

NOTES:
1. ALL ROOMS ARE NOT SHOWN, SEE ROOM DETAIL.

Repository Underground Layout
Deaf Smith County

Source: Adapted from SCC, 1984a.

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Figure 8.4.11. Salt Repository Underground Layout

The design and construction of the shafts and liners will take into account the eventual decommissioning of the repository. Figures 8.4.12 and 8.4.13 show typical shaft lining and seal systems proposed for the exploratory shafts, which are likely to be similar to the repository shaft linings and seals. A watertight composite liner of steel and concrete is proposed in both shafts to the top of the Unit 4 salt formation to prevent any migration and inflow of ground water into the facility.

The waste shaft, service shaft, and ventilation intake shaft will be equipped with hoisting systems. These include guided conveyances for personnel and material transport and emergency hoisting systems. The waste shaft will be furnished with a large capacity hoisting and waste cage system to transport the heavy waste canisters. The ventilation exhaust shafts are not envisioned to have any hoisting facilities.

8.4.3.3.2 <u>Repository Underground Facilities</u>. The repository underground excavations will consist of rooms for waste emplacement, branch and main passageways for transporting the mined salt and waste packages, and drifts for ventilation. Additional areas will be excavated for equipment maintenance, parts and materials storage, operations control and monitoring, first aid and emergency systems, and other miscellaneous facilities normally associated with underground operations.

Figure 8.4.11 shows a simplified layout of the repository underground area showing a completed repository development at the end of operations. There will be five main passages driven down the approximate center of the repository storage room area:

- Fresh air intake for both development and emplacement operations
- Return airway from the development areas
- Waste transport heading
- Access heading to both development and emplacement areas for personnel and equipment
- Mined salt transport drift.

The headings will have a rectangular section approximately 9 m wide and 5 m high (30 ft by 15 ft). There will be about 10,200 m (33,500 ft) of drifts in the main access area and 6,100 m (20,000 ft) of drifts in the shaft pillar zone. The waste storage areas of the repository will consist of branch headings, exhaust branches, and 22 panels of storage rooms. The branch headings will provide access to the individual storage rooms and will be connected by cross cuts. During the excavation of storage rooms, the

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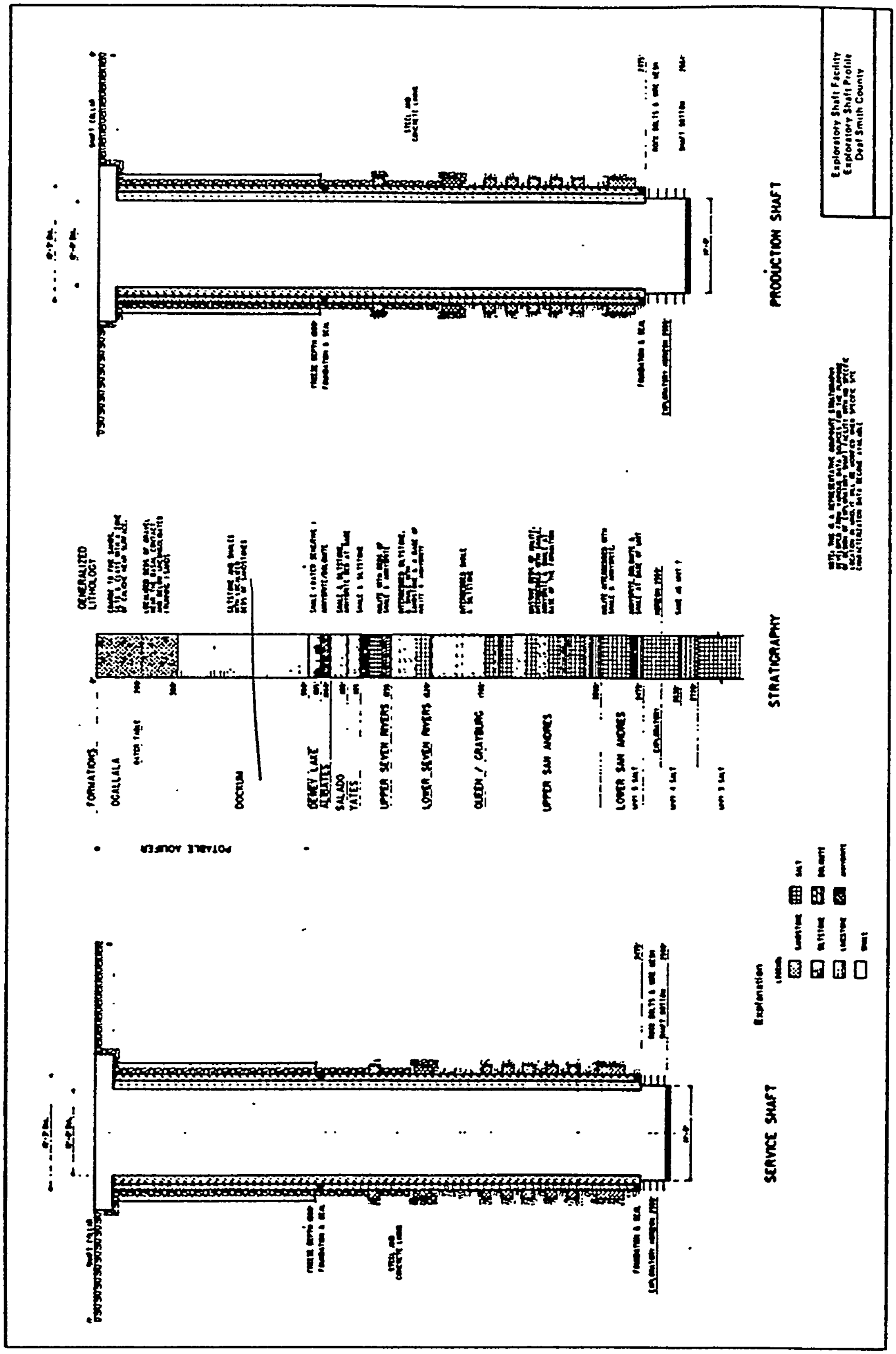

Figure 8.4.12. Salt Exploratory Shaft Profile

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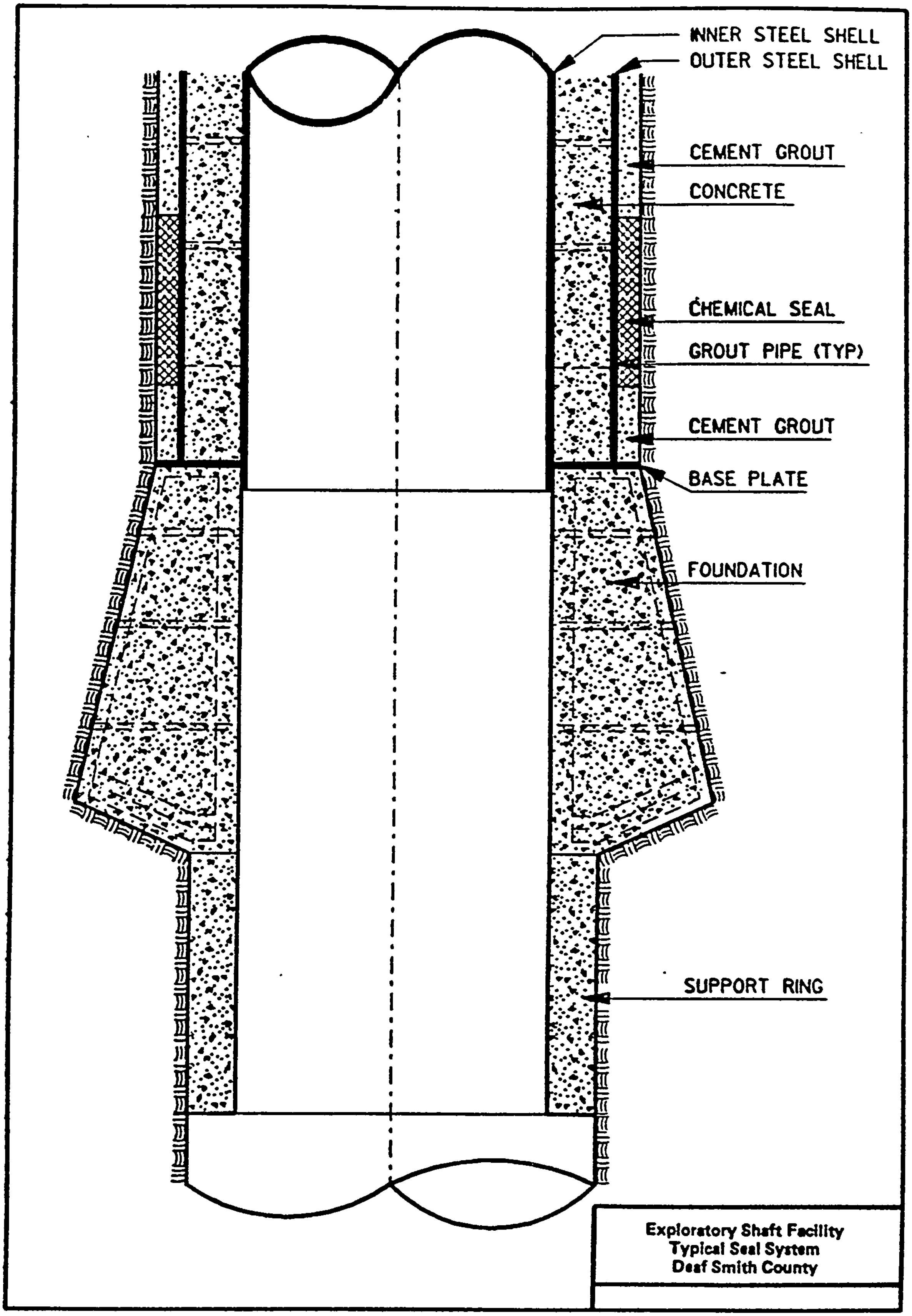

Figure 8.4.13. Salt Exploratory Shaft Typical Seal System

headings will provide a path for intake and exhaust air. During emplacement operations, the headings will be used to transport waste to the individual rooms and provide fresh air. The exhaust passages will be connected to the exhaust end of the storage rooms and to the perimeter exhaust drifts. These drifts in conjunction with the perimeter exhaust headings, will provide a completely isolated path for confining exhaust emplacement air.

Each waste type will be emplaced in separate panels in the repository. A typical waste emplacement panel is shown in Figure 8.4.14. The three access drifts, spaced at 30 m (100 ft), will connect the main access headings to the perimeter exhaust drifts. The emplacement rooms will be excavated perpendicularly to these access headings. The spacing of the roughly 360 m (1,180 ft) long rooms will depend on the type of waste. Low-heat waste rooms will be spaced at about 36 m (120 ft) and high-level, high heat waste at about 52 m (170 ft). The emplacement rooms will have an approximate height of 4.5 m (15 ft) and a width of around 7.5 m (24 ft). Emplacement of the waste canisters will be in boreholes drilled either in the floor or in the rib of the emplacement rooms. Both concepts are being evaluated at this time.

8.4.4 Repository Construction

8.4.4.1 Schedule

The sequence and duration of repository construction activities is shown in Figure 8.4.15. The total time for construction is 7.25 years, including three months for equipment tests.

Construction of the repository will begin with development of offsite utilities and transportation access corridors and establishment of the permanent water supply. Site preparation and grading would begin immediately and start in the vicinity of the service shaft. Three months of this twelve month activity are on the critical path. The next phase will involve sinking, lining, and equipping of the service shaft and stations, which will be complete before at-depth excavation begins. The shaft and shaft station construction activities are estimated to require approximately 48 months of which the first 39 are on the critical path.

Before waste emplacement begins, the shaft pillar and mine development will be completed to a point at which one year's waste receipt could be emplaced. The activity is estimated to require about 42 months and is in the critical path. Preoperation testing of emplacement equipment and ventilation systems will be completed before actual waste emplacement activities begin. This testing is estimated to require approximately three months.

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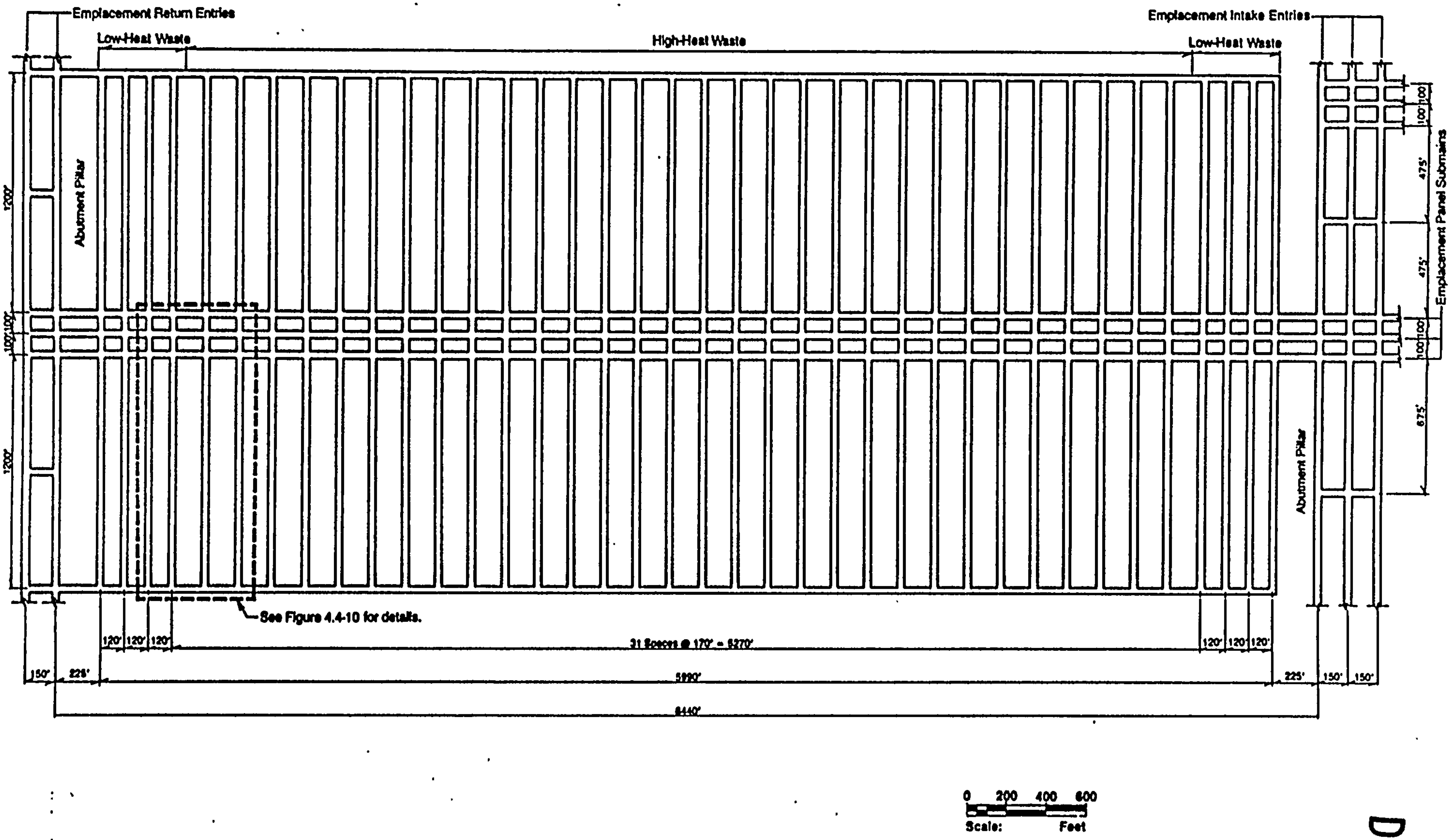
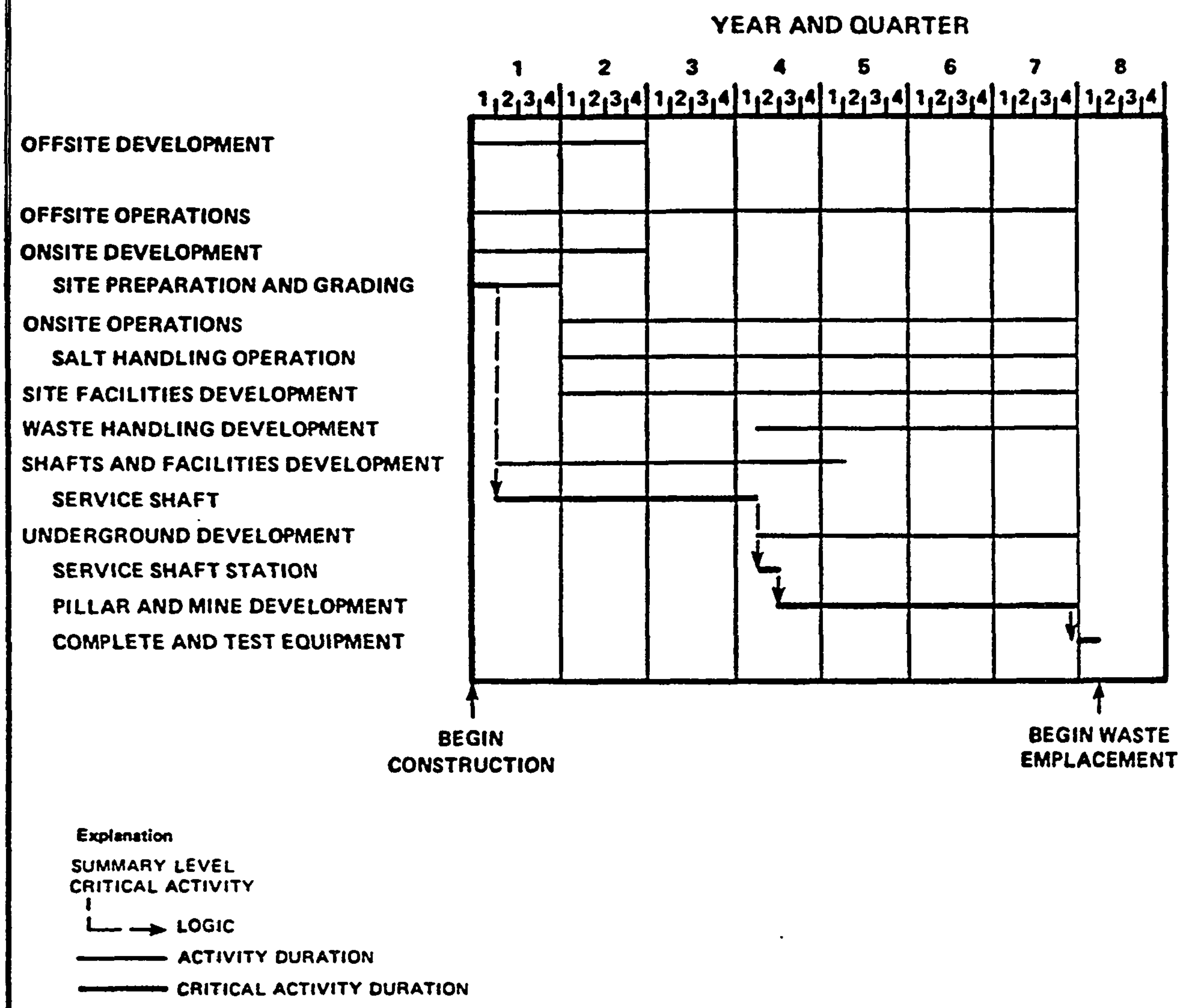


Figure 8.4.14. Salt Repository Typical Waste Emplacement Panel



Source: SCC, 1984a.

THIS INFORMATION IS BASED ON PRELIMINARY AND INCOMPLETE DATA AND IS SUBJECT TO CHANGE.

Schedule Summary Bar Chart
Deaf Smith County

Figure 8.4.15. Salt Repository Construction Schedule

8.4.4.2 Methods and Equipment

8.4.4.2.1 Shafts. All of the shafts may be excavated using conventional drill-and-blast methods, although in some sections, such as the frozen zones, the application of non-blasting methods may be preferred. Other excavation options are also under investigation. Potentially high ground-water inflows and unstable ground in the sediments above the salt will require special pretreatment by freezing to allow shaft sinking through these sediments. The ground freezing principle is shown in Figure 8.4.16 and involves the drilling of a number of freeze hole around the prospective shaft excavation from the surface to competent, dry formations below the aquifers. In the case of the exploratory shaft facility, this depth is estimated to be about 310 m (1,000 ft). A refrigeration plant continuously circulates cold brine through these pipes and forms a frozen column of ground inside which shaft excavation can be carried out. Figure 8.4.17 shows the sequence of shaft construction proposed for the exploratory shaft facility. This method will likely be similar to that which will be used for the repository shafts. Conventional controlled blasting methods or, preferably, mechanical excavation techniques will be used to excavate the shafts to below the frozen zone. A foundation and seal area will be established in a competent formation and a water-tight shaft lining installed back to the surface to completely isolate the aquifer regions from the lower formations and prevent water migration around the lining. The lining will consist of a welded steel cylinder supported by an inner concrete lining. A backfill such as bitumen will be placed between the original preliminary construction lining and the final steel liner to act as seal and prevent water migration. After completion of the lining, the freezing process can be terminated and the ground allowed to thaw. A similar process will be used to also place a watertight lining from below the frozen zone to the top of the Unit 4 salt, but temporary ground freezing will probably not be needed. Grouting techniques will most likely suffice to control any water inflows. The completed lining will thus provide a watertight and sealed connection between the surface and the salt formation as shown in Figure 8.4.12. A similar shaft construction and lining system is currently being employed during the sinking of the Gorleben exploratory shafts in Germany.

The scheduled sequence of shaft development will begin with the service shaft and ventilation intake shaft, allowing initial underground development to begin. The remaining shafts will be completed concurrent with the initial underground development. If the exploratory shafts are used as part of the final repository, initial development may also take place from these until the main shafts are completed.

The shafts will be furnished after excavation and lining. This will encompass installation of hoisting systems, utility lines, instrumentation, etc.

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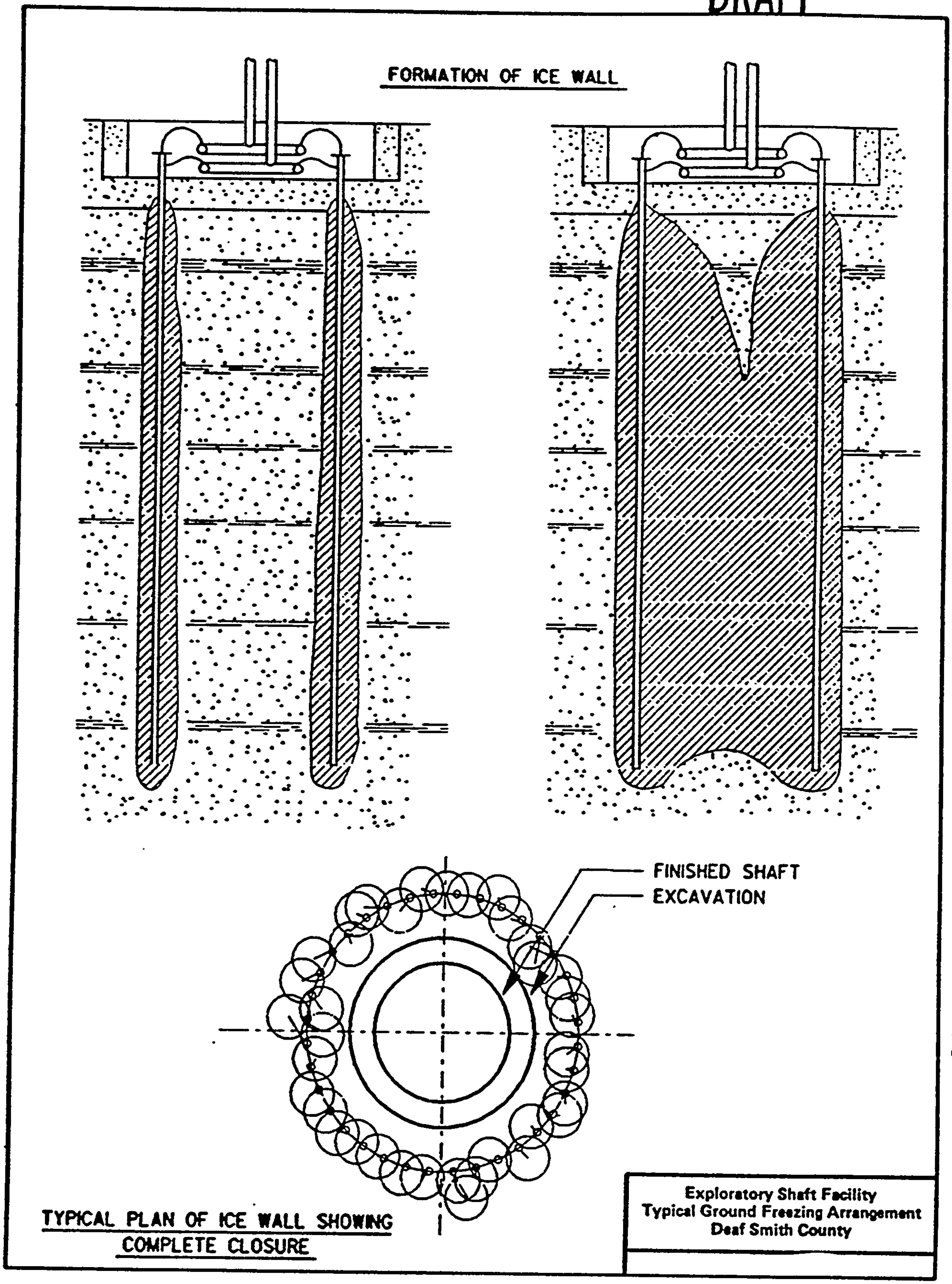

Figure 8.4.16. Typical Ground Freezing Arrangement

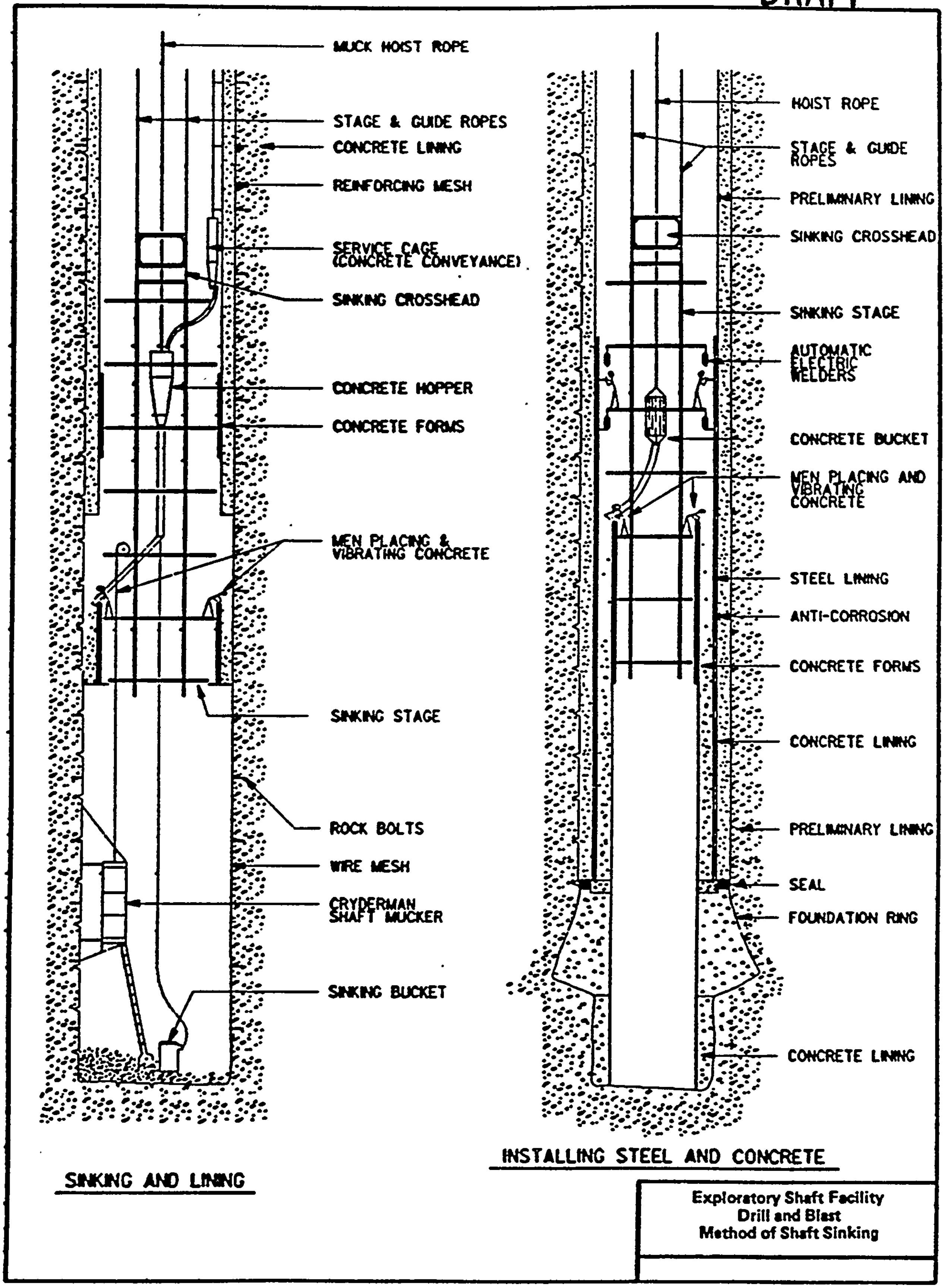

Figure 8.4.17. Typical Shaft Sinking and Lining Arrangement

8.4.4.2.1 Underground Development. It is proposed that underground development be performed mechanically. Drill-and-blast methods will be required for the initial development of the shaft station areas and staging rooms for the mechanical excavation machines. The machines have the advantage of higher excavation rates, safer operation and minimum disturbance of the surrounding rock mass.

The initial development phase of the repository will begin when the service shaft and ventilation intake shaft have been completed. The underground excavations to be performed in this initial development phase will include:

- Excavation of the entries in the shaft pillar
- Excavation of the five main access headings to access all panels necessary for the various waste types in the first year of emplacement
- Excavation of the panel drifts, exhaust drifts and perimeter exhaust drifts necessary for the first two years
- Excavation of storage rooms required for the first two years of waste emplacement
- Completion of the remaining shafts.

8.4.5 Waste Emplacement Cycle

8.4.5.1 Waste Treatment

Reprocessing of commercial spent fuel is currently not being practiced in the United States so that the majority of the nuclear waste will be in the form of spent fuel assemblies. About 62,000 MT (68,000 tons) of spent fuel were estimated to require disposal in the first repository. Another 8,000 MT (9,000 tons) equivalent of defense high-level reprocessing waste and commercial reprocessing waste from West Valley will also be disposed of in the first repository. The spent fuel assemblies will be repackaged in special containers in preparation for underground disposal. The high-level reprocessing wastes will be vitrified in borosilicate glass and encased in disposal containers.

8.4.5.2 Waste Transport

Waste transport systems are still under analysis. Rail and truck transport are the options being investigated. Transport cask configurations are also the subject of scientific and engineering studies.

8.4.5.3 Waste Package

Reference waste package designs have been produced for consolidated spent fuel, spent fuel assemblies and defense high-level wastes emplaced either horizontally or vertically in boreholes in salt. The waste package consists of a canistered waste form and a low-carbon steel container. The container, identical in concept for all waste forms, is shown schematically in Figure 8.4.18. Dimensional differences are shown in Table 8.4.1 and are not affected by emplacement mode.

Table 8.4.1. Waste Package Dimensions for Salt (Dimensions are in cm).

	Waste Form		
Dimension	HLW[1]	SFA[2]	CSF[3]
Waste form canister O.D.	61.0	67.3	62.0
Container O.D.	84.8	94.8	88.4
Container wall thickness	10.7	12.8	11.9
Container head thickness	16.8	19.8	18.3
Corrosion allowance	1.1	2.3	2.3

Source: ONWI, 1987

1 High-level waste
2 Spent fuel assembly
3 Dismantled and consolidated spent fuel assemblies

No packing is intentionally included with any waste package. However, crushed salt will be used to fill the emplacement hole above a vertically emplaced container for shielding purposes; and it is assumed that crushed salt will fill the annulus around the container. A solid salt plug is provided for shielding in the horizontal borehole.

The waste for each package is provided in canistered form. In the case of defense high-level waste, the vitrified waste form is supplied in a stainless steel canister. For the spent fuel assemblies, each canister

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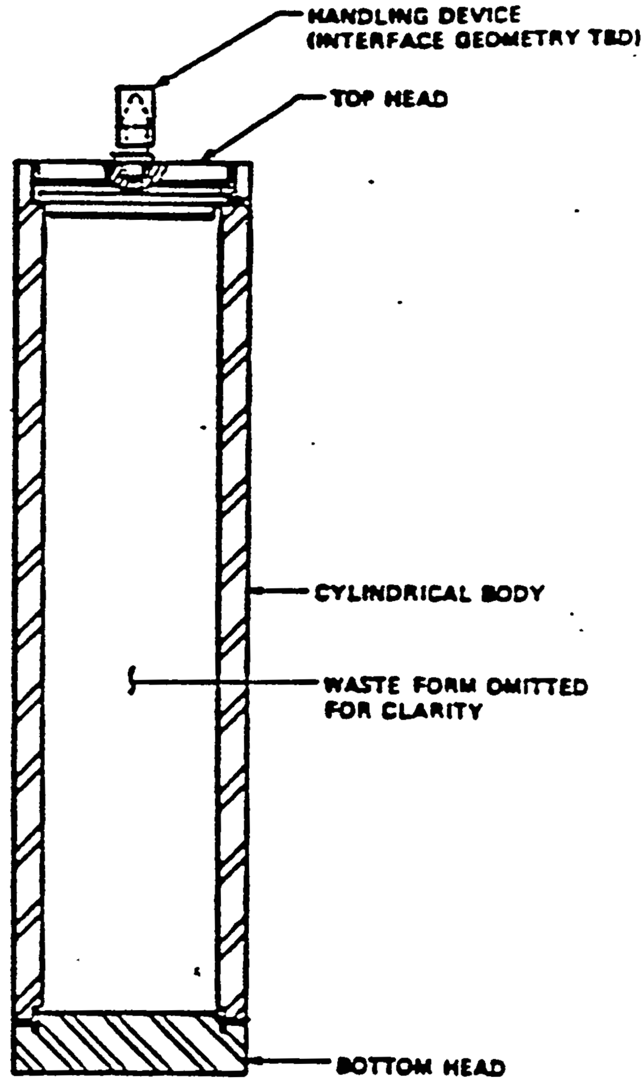

Figure 8.4.18. Schematic of Reference Conceptual Waste Package in Salt

consists of four components: a baseplate, a cylindrical shell, a fuel cage, and a cover plate with lifting pintle. Fuel cages are designed for either 4 pressurized water reactor assemblies or 9 boiling waster reactor assemblies. Each cage is made of 3 circular plates, each having a square hole to position either a 2 by 2 or 3 by 3 array of assemblies. The plates are spaced longitudinally in the canister by long steel rods. For the consolidated spent fuel assemblies, the cylindrical canister is segmented into 6 congruent compartments (Figure 8.4.19). Each compartment will hold the rods from either 2 pressurized water or 5 boiling water reactor assemblies. A vacant hexagonal section at the center of the canister cross section accommodates a stem which extends from the canister baseplate for handling purposes.

The material chosen for the waste container is a cast, low-carbon steel, ASTM 216-77, Grade WCA (Westinghouse, 1986). This steel is considered to have adequate corrosion resistance and strength while being plentiful, easily formed, and weldable. The container is designed to resist a lithostatic pressure of 18.0 MPa (2,600 psi), the pressure at repository depth. Early imposition of this external pressure on the container is expected due to the rapid creep of the salt. To the thickness required for structural support is added a 1000 year corrosion allowance. The corrosion allowance is calculated assuming an unlimited supply of brine. While limited availability of brine, because of the limited amount of water in salt, is the expected condition, unlimited brine provides a conservative basis for design. This is considered prudent because the corrosion data base is limited and because of the heterogeneous nature of the salt at the selected salt site and the possibility of brine flow through interconnected porosity around the waste package.

8.4.5.4 Waste Emplacement

The anticipated sequence of repository operations will begin when the waste form has arrived at the repository by rail or truck in the shipping container. At the waste handling and packaging facility, the waste form will be removed from the shipping cask and enter the hot cell. Depending on the form of the waste, additional operations may be performed in the hot cell. For consolidation of the spent fuel, the waste form will move into the dismantling station where the fuel rods will be removed from the fuel assembly or box, consolidated, and placed into a waste package canister. This spent fuel waste package canister or the commercial high-level waste and defense high-level waste canisters will then be moved to the packaging station where they will be placed inside the waste containers, and the container lid installed. After inspection of the welds, etc., the waste package will then move to the transfer cask station where the package will be loaded into the transfer cask. The transfer cask, a heavy, shielded container similar to the shipping cask, will be transferred to the waste shaft collar, loaded onto the waste cage, lowered to the repository level, and unloaded into an underground transport vehicle.

Underground emplacement operations will begin upon completion of the initial development phase. During this first year of operation, the room

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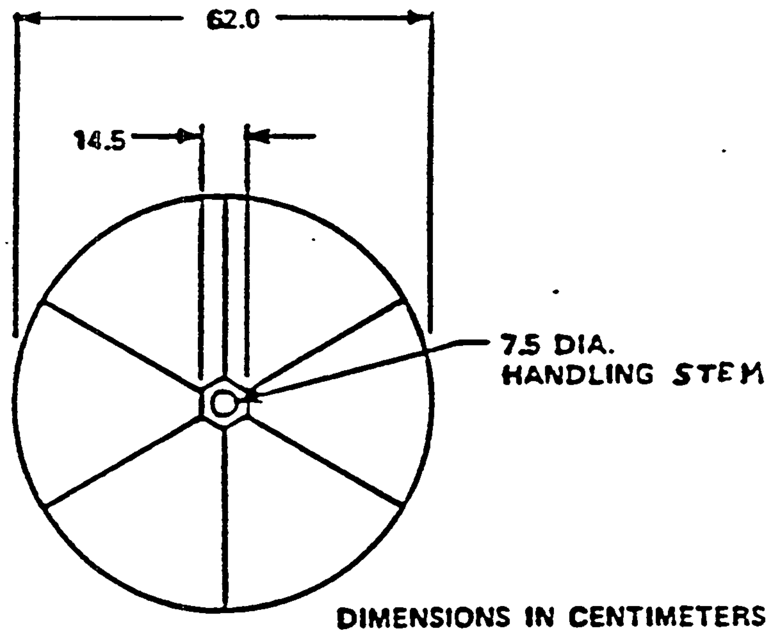


Figure 8.4.19. Consolidated Spent Fuel Canister Cross Section in Salt

excavation operations will be moved to the side of the main access headings opposite the emplacement operations. Once the waste package transfer cask is unloaded from the cage onto the underground transporter, the transporter will move the waste in the cask to the designated storage room and position the cask at the vertical or horizontal disposal borehole. The waste package will be transferred into the borehole through an open port in the transfer cask. The borehole will then be backfilled with crushed salt and a plug installed. The transporter will return to the shaft with the empty cask which will be hoisted to the surface for another waste package.

After completion of disposal operations, the repository will be decommissioned and decontaminated. During the decommissioning period, the contaminated portions of the waste handling facilities could be cut up, packaged, and moved to the underground repository. The remaining underground openings and shafts will be backfilled and sealed using the originally excavated salt and other selected sealing materials.

8.4.6 Backfilling and Sealing

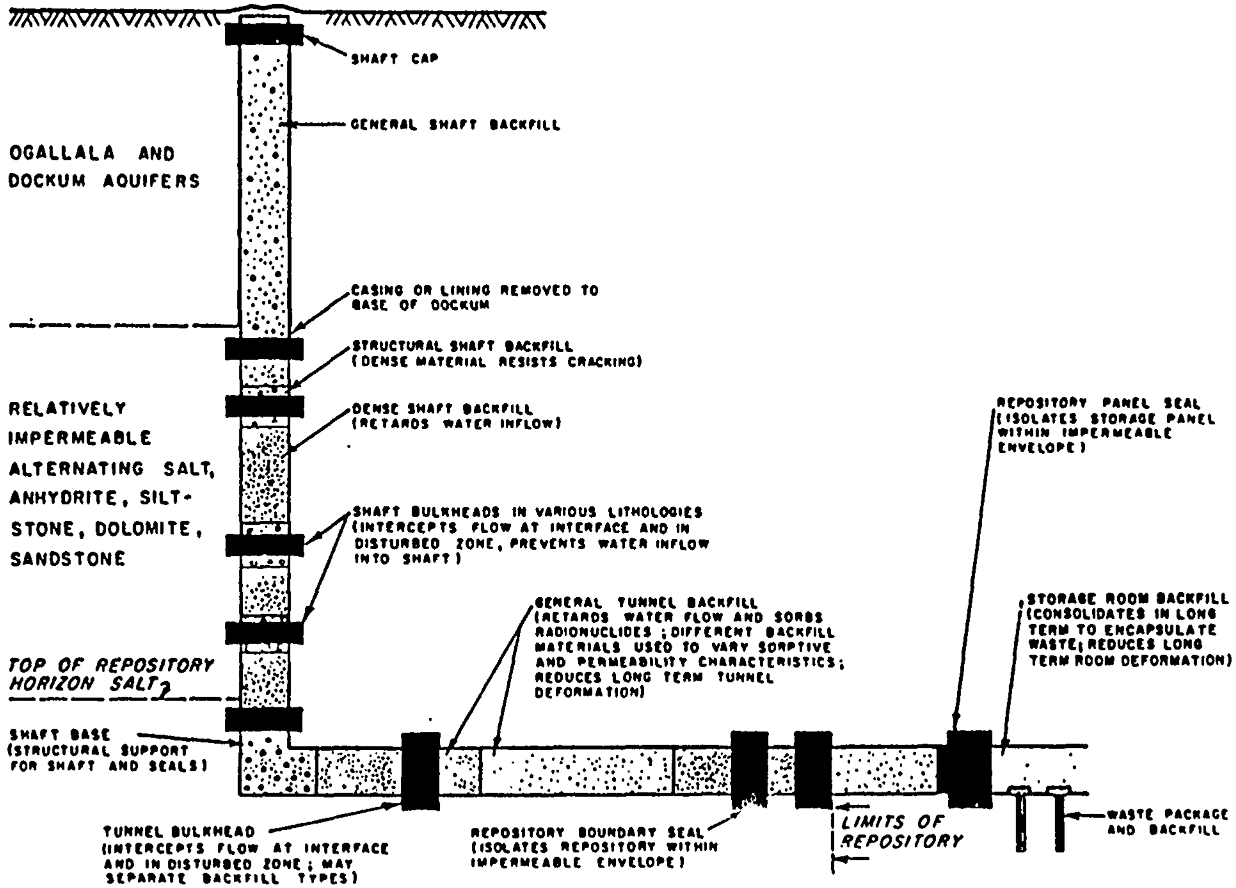
After all contaminated materials for surface decommissioning and decontamination are emplaced underground, the repository underground facilities will be closed by installing a repository sealing system. The repository seals will consist of subsystems to seal the rooms, drifts, and passageways; the shafts; and the boreholes. A schematic diagram of the sealing system is shown in Figure 8.4.20.

8.4.6.1 Room, Drift, and Passageway Seals

The bulk of the horizontal openings will be filled with a crushed salt backfill, which under stress and heat conditions, will ultimately (tens to hundreds of years) reform into a solid mass of salt similar to the pre-emplacement conditions. Groups of rooms will be isolated with precompressed salt blocks installed as masonry walls to provide isolation while the crushed salt is compacting. In the passageways connecting the shafts to the main disposal area, multiple concrete bulkheads will be installed. Portions of the drift between concrete bulkheads will be backfilled with an earthen material to provide redundancy.

8.4.6.2 Shaft Seals

After the drifts connecting the shafts have been sealed, the base of the shafts will be filled with structural concrete to support the loads imposed by the shaft sealing materials. Most of the shaft volume will be backfilled with crushed salt, in the salt horizons only; dense earthen material, a mixture of clay and locally available materials; or general backfill, locally available material such as that removed during shaft sinking. At selected locations along the shaft, bulkheads will be constructed as shown in Figure 8.4.21.



Repository Sealing
System Schematic
Permian Basin Repository

Source: Adapted from Kelsall et al., 1985, BMI/ONWI-564.

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Figure 8.4.20. Salt Repository Sealing Schematic

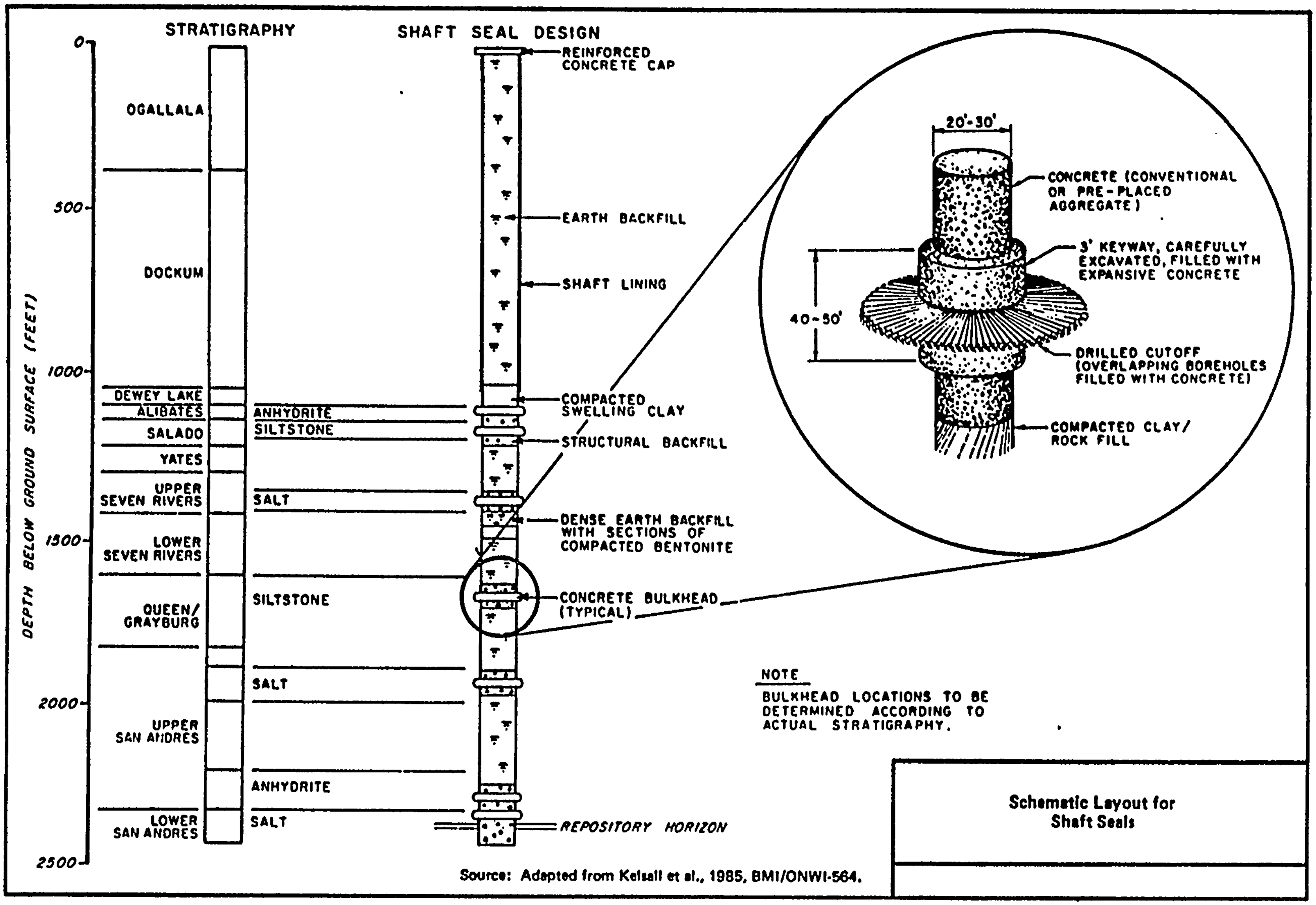

Figure 8.4.21. Salt Repository Shaft Sealing Schematic

The shaft lining will be removed at the bulkhead locations. These bulkheads will extend into the rock around the shaft to intercept and seal any fracture areas created by the sinking of the shaft and prevent the transport of water outside of the sealed shaft. At the top of the shaft, a shaft cap of concrete will be placed.

8.6.4.3 Borehole Seals

Boreholes from the surface will be plugged as shown in Figure 8.4.22. Cement grout will be the primary constituent of the seal with a clay layer for material redundancy. Where necessary to seal behind the borehole casing, the casing can be perforated using oil field technology and grout injected through the holes into the formation.

8.4.7 Summary

8.4.7.1 Advantages

The main advantages listed for the other salt programs are also applicable here. These and other advantages include:

- Favorable salt properties including low water content and hydraulic conductivity, high thermal conductivity, large deposits of relatively homogeneous formations, self-healing properties, and low mining costs
- A large amount of research has been ongoing in salt for many years and thus a great deal of data exists on the behavior of the salt mass
- Large mining projects have been conducted in salt formations and the mining methods and technology have evolved to a high degree of sophistication
- A large amount of knowledge about the area conditions exists in the oil industry.

8.4.7.2 Potential Problems

Although salt has many advantages as a repository formation, several potential problems or difficulties are associated with the disposal concept as outlined in this chapter, and further investigations are ongoing to resolve these:

- Salt is highly soluble, has poor sorptive properties and brine solutions are very corrosive

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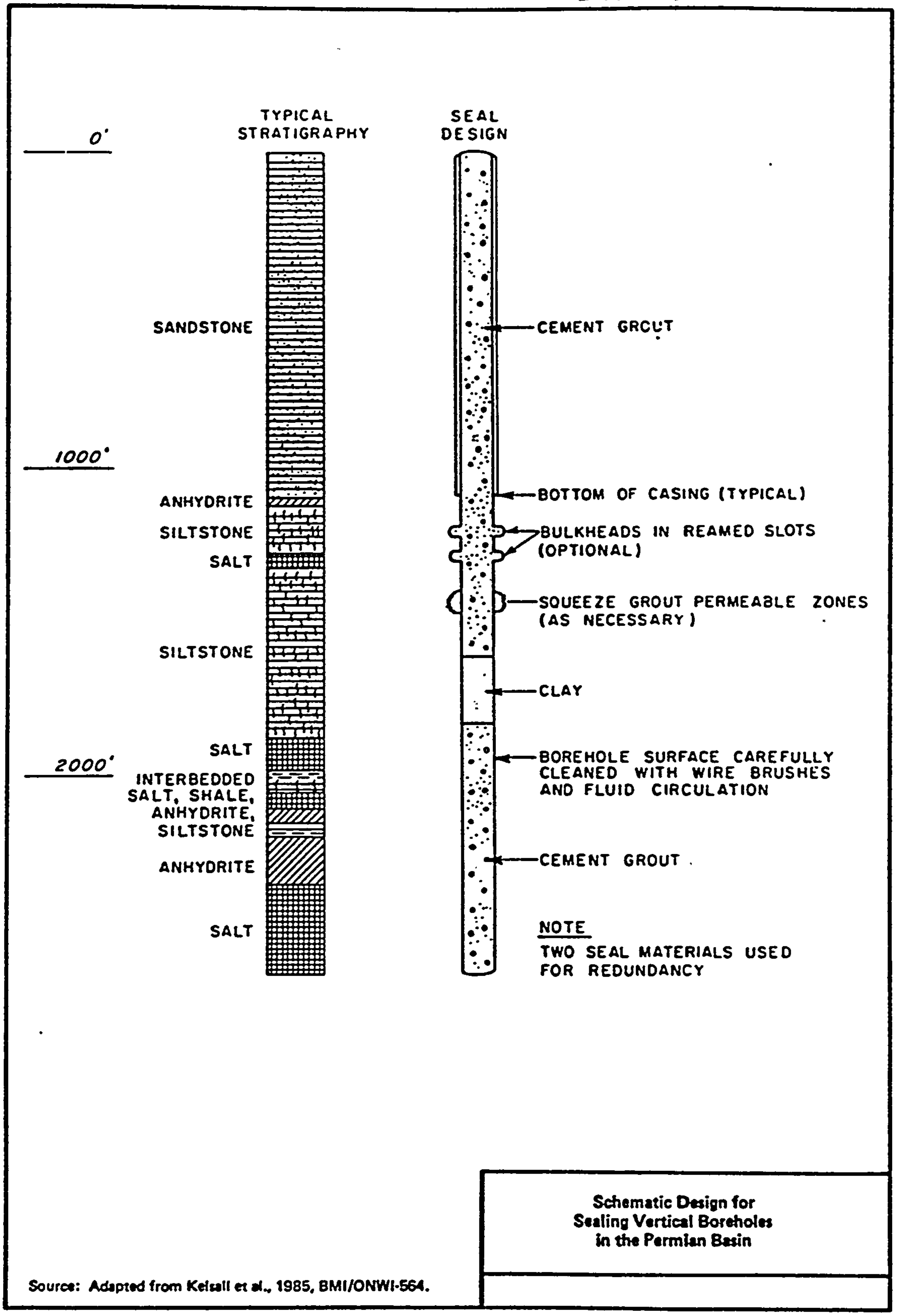

Figure 8.4.22. Salt Repository Borehole Sealing Schematic

- The overlying aquifers will require the application of freezing technology which, although it has been used for over 100 years, will always have an inherent risk factor

- High degrees of quality control will be required during the installation of the shaft linings to ensure the long operational periods and watertightness.

9.0 BASALT FORMATIONS

9.1 BASALT: THE UNITED STATES CONCEPT

This chapter describes the Basalt reference repository concept of the nuclear waste disposal program in the United States. Since 1957, when the National Academy of Sciences concluded that geologic disposal would be the most desirable method for permanent disposal of nuclear wastes, many alternatives have been considered. Studies have been conducted in different geologic media and large amounts of data collected, analyzed and compared. The passage of the Nuclear Waste Policy Act of 1982 provided a framework for the achieving the disposal of the nation's spent nuclear fuel and high-level wastes. The Act authorized one repository with a capacity of 70,000 MT (77,000 tons) of wastes and proposed a second repository, should the need arise. The repositories will be used for disposal of commercial high-level wastes or spent fuel produced by nuclear power generation facilities as well as the small amount of defense high-level wastes. It also requires the preparation of proposals for a Monitored Retrievable Storage facility and established a funding mechanism for the waste disposal program. Several regions in the United States were under investigation and in February 1983, the Department of Energy fulfilled the first requirement of the Act by formally identifying nine sites as potentially suitable repository locations in three different geologic media as shown in Figure 8.4.1:

1. Vacherie Dome, Louisiana (Domal Salt)
2. Cypress Creek Dome, Mississippi (Domal Salt)
3. Richton Dome, Mississippi (Domal Salt)
4. Yucca Mountain, Nevada (Welded Tuff)
5. Deaf Smith County, Texas (Bedded Salt)
6. Swisher County, Texas (Bedded Salt)
7. Davis Canyon, Utah (Bedded Salt)
8. Lavender Canyon, Utah (Bedded Salt)
9. Reference Repository Location, Hanford Site, Washington (Basalt Flows).

On May 28, 1986, the Department of Energy announced presidential approval of the nomination and recommendation for site characterization of five of the nine sites in Washington, Texas and Nevada. Environmental assessments were published at the same time for the five recommended sites. These sites are Richton, Deaf Smith County and Davis Canyon in salt, Yucca Mountain in tuff, and Hanford in basalt. Approval was granted for the construction of exploratory shaft facilities at three of these sites:

1. Deaf Smith County, Texas

2. Yucca Mountain, Nevada

3. Hanford, Washington.

Designs for these exploratory shaft facilities are currently in preparation, and construction is scheduled to commence in the 1988-89 time frame. During the site characterization activities at each site, the Department of Energy will consult closely with the Nuclear Regulatory Commission, responsible for licencing the repository, and affected states and Indian tribes. After several years of site characterization, the Department of Energy will be able to recommend one of the sites to the president as the location for the first repository. After approval by Congress, and state and local affected groups, a Construction Authorization Application will be submitted to the Nuclear Regulatory Agency. The agency will have three years to review the application and resolve any issues with the Department of Energy. Subject to approval by the Nuclear Regulatory Agency, repository construction is planned to start in the mid-1990s to meet the 1998 date of start of actual waste disposal as scheduled by the Nuclear Waste Policy Act.

Work is also being carried out on the second repository program. Crystalline rocks were investigated as a possible location for this repository. An Area Recommendation Report was published in 1986 and current activities include resolution of comments received on the report and technology development activities.

The 70,000 MT (77,000 tons) of waste represents the maximum storage capacity of the first repository and is made up of several types of nuclear waste. Commercial reprocessing is currently not practiced, and thus the reference repository described in this chapter is based on the disposal of about 62,000 MT (68,000 tons) of commercial spent fuel and about 8,000 MT (9,000 tons) equivalent of reprocessed wastes. These reprocessing wastes are mainly defense high-level wastes (90%) and some reprocessing wastes (10%) remaining at the shut-down West Valley commercial reprocessing plant.

This chapter describes the reference repository for the Hanford site in Washington, published in the Environmental Assessment in 1986.

9.1.1 Geographic Location

The reference repository location is situated near the center of the 1,500 square km (570 square mi) Hanford site in the south-central portion of the State of Washington (Figure 9.1.1). The area lies within the Pasco Basin, the structural and topographic low part of the Columbia Plateau, and approximately 55 km (34 mi) northwest of the junctions of the Snake and Yakima Rivers with the Columbia River. The topography at the reference

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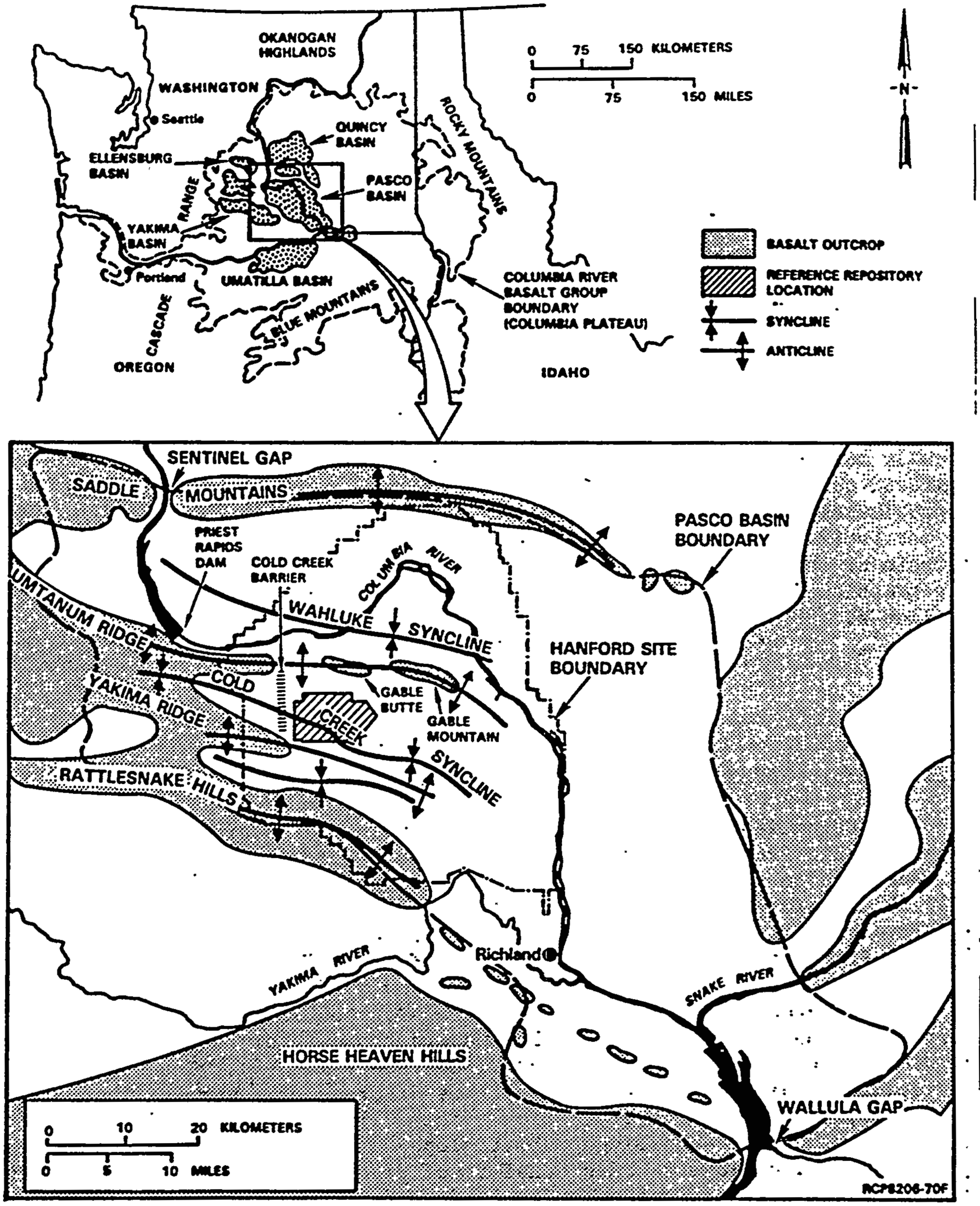

Figure 9.1.1. Reference Repository Location within the Hanford Site

repository site is relatively flat. The site is approximately 195 m (640 ft) above mean sea level and approximately 73 m (240 ft) above the current level of the Columbia River.

9.1.2 Summary of Geologic Setting

The near-horizontal layers of basalt reach depths of more than 3.3 km (10,000 ft). The highest basalt layer is covered by poorly consolidated sediments, about 183 m (600 ft) thick, and some basalt layers are separated by sedimentary beds. Figure 9.1.2 shows a generalized stratigraphical column derived from one of the boreholes at the exploratory shaft location. The Cohassett basalt flow at a depth of about 911 m (2,993 ft) is the reference repository horizon. At this location, the Cohassett is 80 m (262 ft) thick.

The overburden sediments contain an unconfined aquifer with a ground water table at about 52 m (170 ft) depth. The tops and bottoms of the basalt flows are typically brecciated and vesicular. The basalt interiors are tight, but the flow tops have greater hydraulic conductivity than the interiors. This is also true for some of the interflow sediments.

The intact basalt is very dense, strong and hard (unconfined compressive strength is around 303 MPa or 44,000 psi). Hydraulic fracturing tests have indicated that an in situ horizontal stress of about 2.5 times the vertical stress in a roughly north-south direction, and about 1.3 times the vertical stress in the east-west direction. This stress condition has been qualitatively confirmed by observations of core disking, by borehole wall spalling in many boreholes, and by analysis of microseisms in the Pasco Basin.

The thermal gradient is fairly large, and the in situ temperature at the depth of the reference repository location is about 52 °C (125 °F).

9.1.3 Reference Repository Concept Description

9.1.3.1 Nuclear Waste Disposal Strategy

The United States high-level waste disposal strategy was outlined in Section 8.4.3.1.

The repository concepts described in this chapter is based on the concept published in the Environmental Assessment paper in 1986. These descriptions in turn were based on 1982 conceptual design (Raymond Kaiser Engineer, Inc./Parsons-Brinkerhoff Quade and Douglas, Inc. 1982) and a series of later engineering studies. The information presented here is therefore not a final repository concept but represents a refinement step in the normal design process. A discussion of the 1982 reference design is given in Section 9.1.3.2, and a summary of subsequent engineering studies is given in Section 9.1.3.3.

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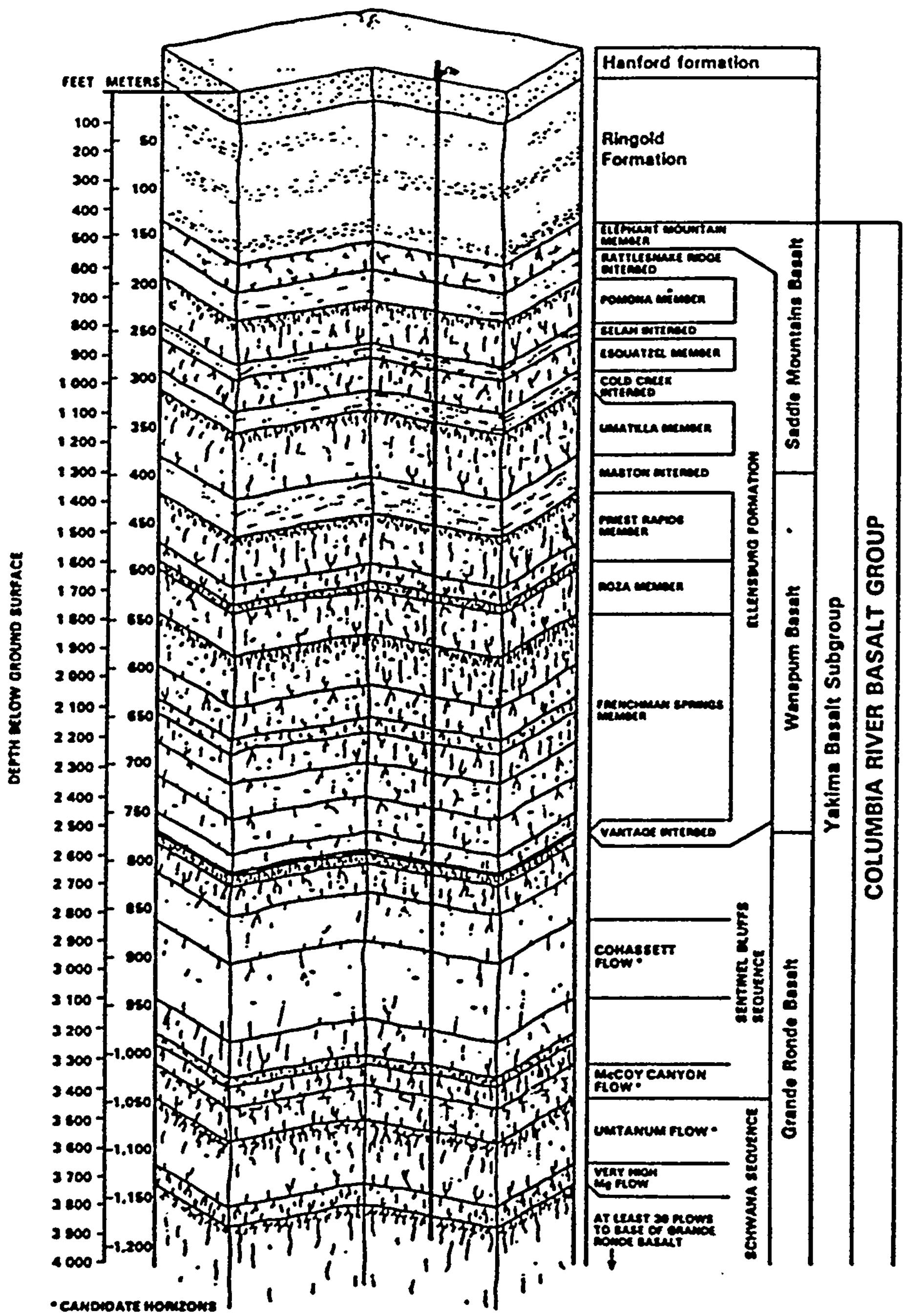

Figure 9.1.2. Generalized Stratigraphy

9.1.3.2 1982 Reference Repository Concept

The 1982 conceptual design predates the adoption of the 70,000 MT (77,000 tons) design capacity. This design was based on an ultimate capacity to dispose of spent fuel and commercial high-level waste equivalent to 47,400 MT (52,000 tons). Design of the underground facilities was based on an equal split between spent fuel and commercial high-level waste on the basis of weight. It was assumed that the spent fuel would be removed from the reactor a minimum of 10 years prior to shipping to the repository or would be converted to commercial high-level wastes. The waste-receipt rate was 1,072 containers of spent fuel per year or the heavy metal equivalent of commercial high-level waste or any combination thereof. The design provided the option of waste retrieval, starting up to 50 years after the initial waste emplacement.

An illustration of the repository facilities base on the 1982 concept is shown in Figure 9.1.3. The components of the repository are the surface facilities, the access shafts, and the underground facilities, including the shaft pillar area and waste storage areas.

Figure 9.1.4 shows the sequential emplacement process for spent fuel or commercial high-level waste as described in the 1982 conceptual design. The shipping cask would arrive at the repository by rail or truck and be moved into the waste handling facility (Figure 9.1.5). The waste container would be removed from the shipping cask and then moved to the primary hot cell to be inspected and repaired if required. The containers would be loaded into the waste-transport shaft cage, and transported down the shaft to the shaft pillar area (Figure 9.1.6). Containers would be transported from the shaft pillar area via the main entries (Figure 9.1.6) to the emplacement room by the waste transporter. Once the waste transporter is in the emplacement room, the container would be loaded in a long emplacement hole (Figure 9.1.7). Containers would be left in the emplacement borehole and could be retrieved, if required. Following the retrieval period, the emplacement holes would be packed and the underground openings backfilled with an engineered material of low permeability (this design concept is based on a mixture of bentonite and crushed basalt).

9.1.3.2.1 <u>Surface Facilities</u>. Repository surface facilities will include the repository facilities and, if found appropriate, a Monitored Retrievable Storage facility located at some strategic location in the United States. A description of the Monitored Retrievable Storage concepts was given in Section 8.4.3.2 and will not be repeated here.

The surface facilities at the repository site are shown in Figure 9.1.8 and would be located in a central process area. The central process area would include all the accesses to the surface and underground facilities. The underground facilities for the 47,400 MT (52,000 ton) repository will cover an area of about 540 hectares (1,334 acres). The central process area of about 80 ha (200 acres) is located directly above the repository and is surrounded by a security fence. A major facility on the surface will be the ventilation system. The waste handling facility,

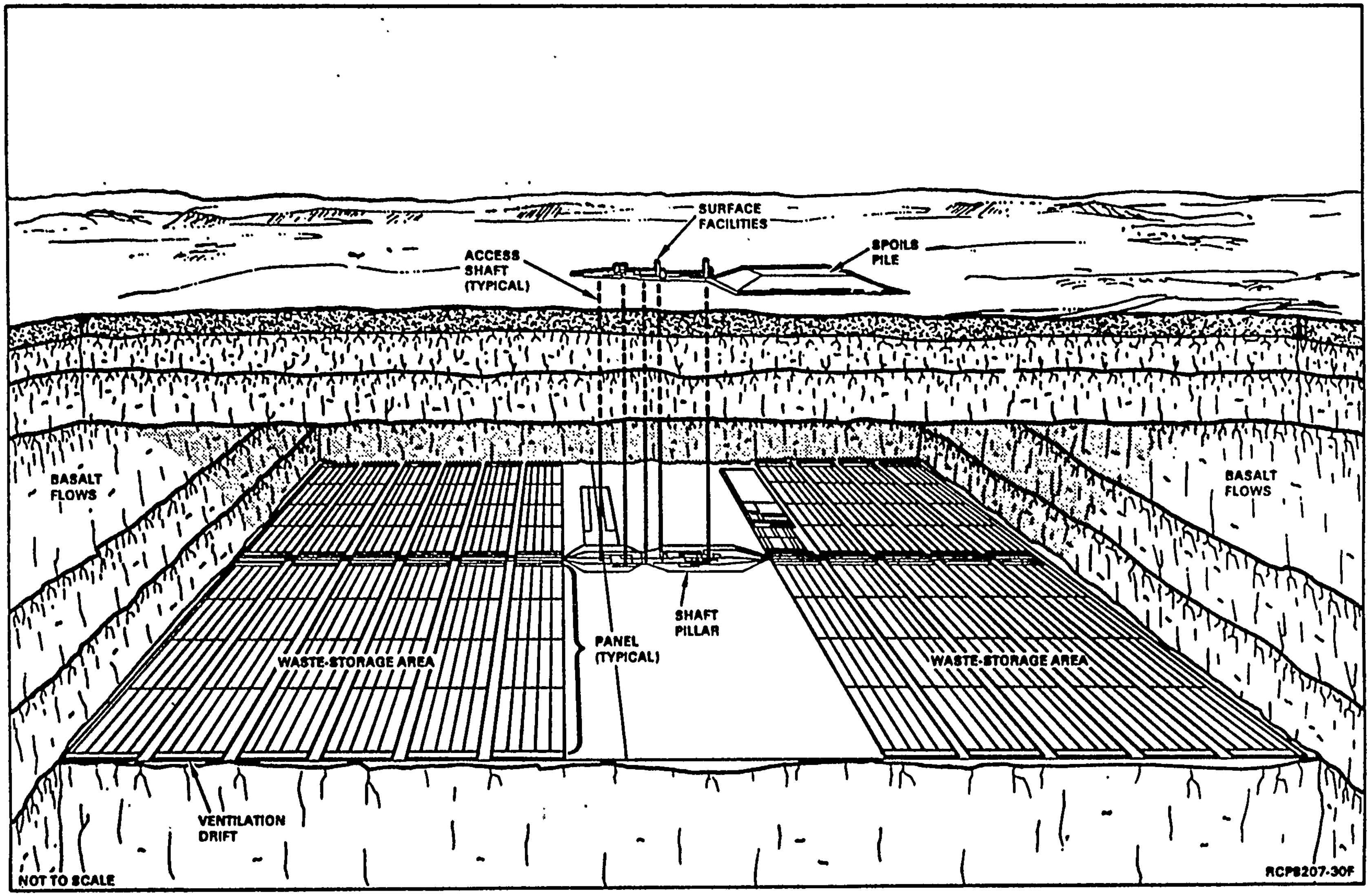

Figure 9.1.3. The 1982 Conceptual Repository Design

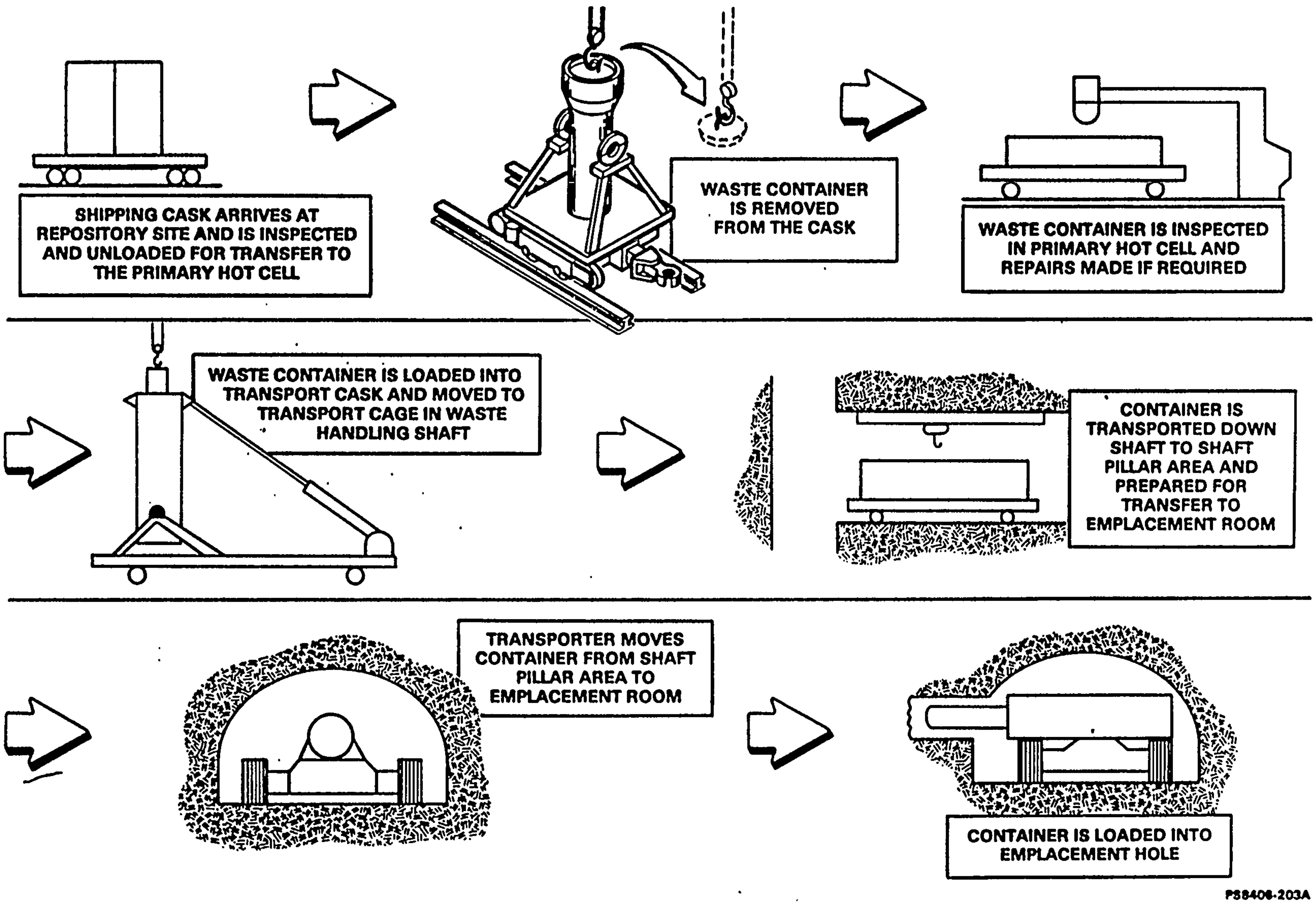

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Figure 9.1.4. Generalized Emplacement Process (1982 Reference Design)

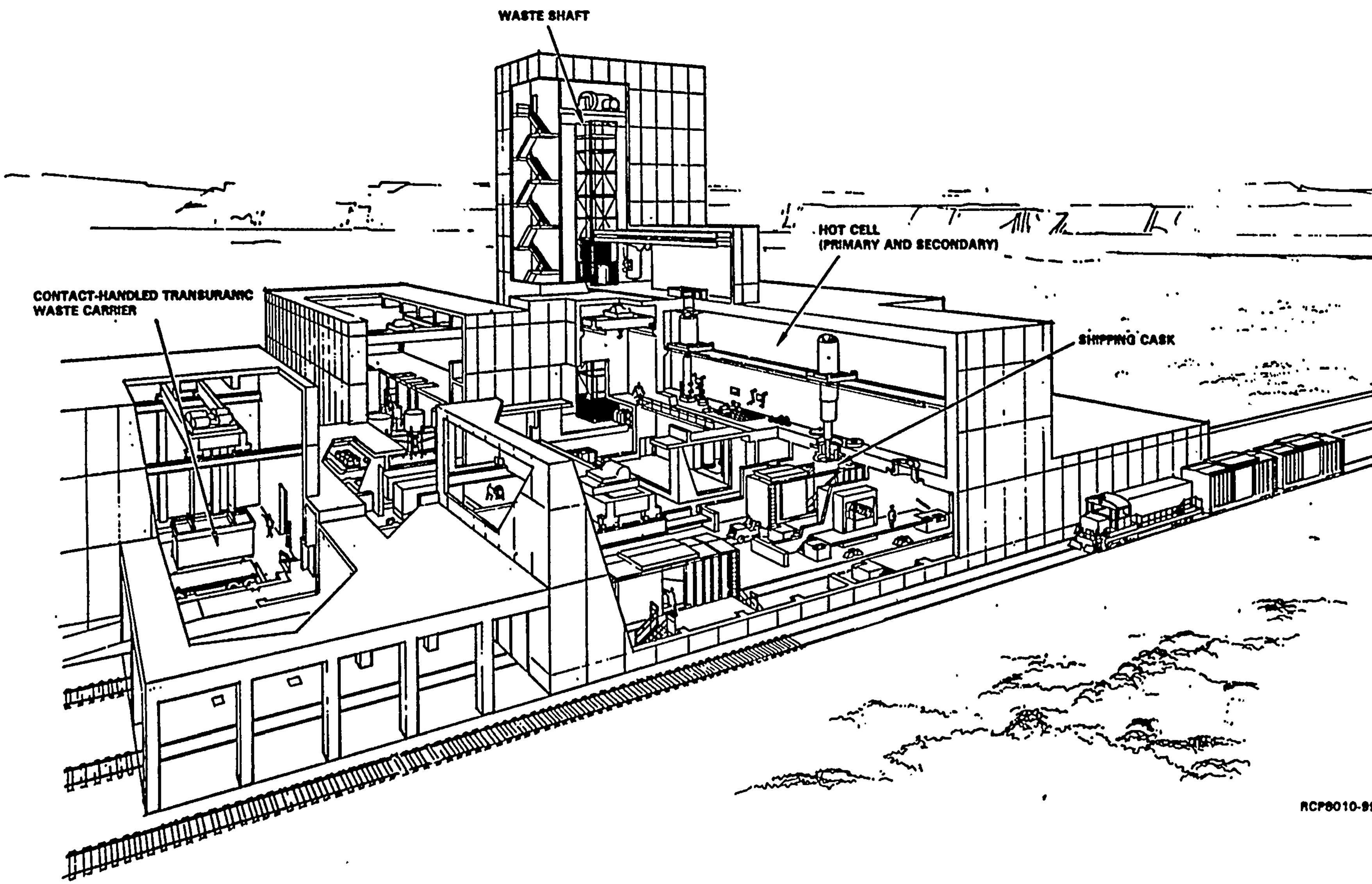

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Figure 9.1.5. Waste Handling Facility (1982 Reference Design)

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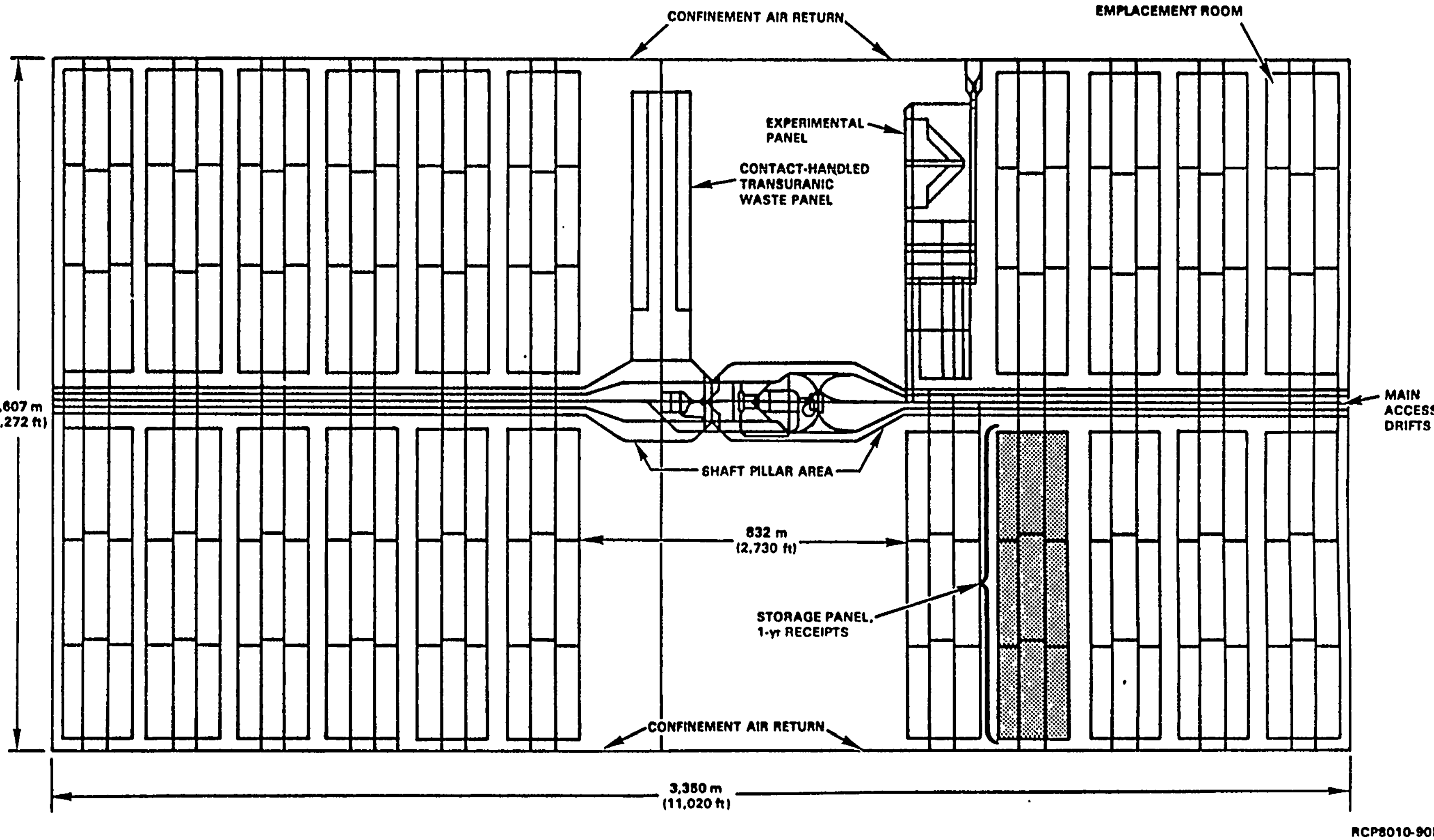

Figure 9.1.6. Underground Repository Layout (1982 Reference Design)

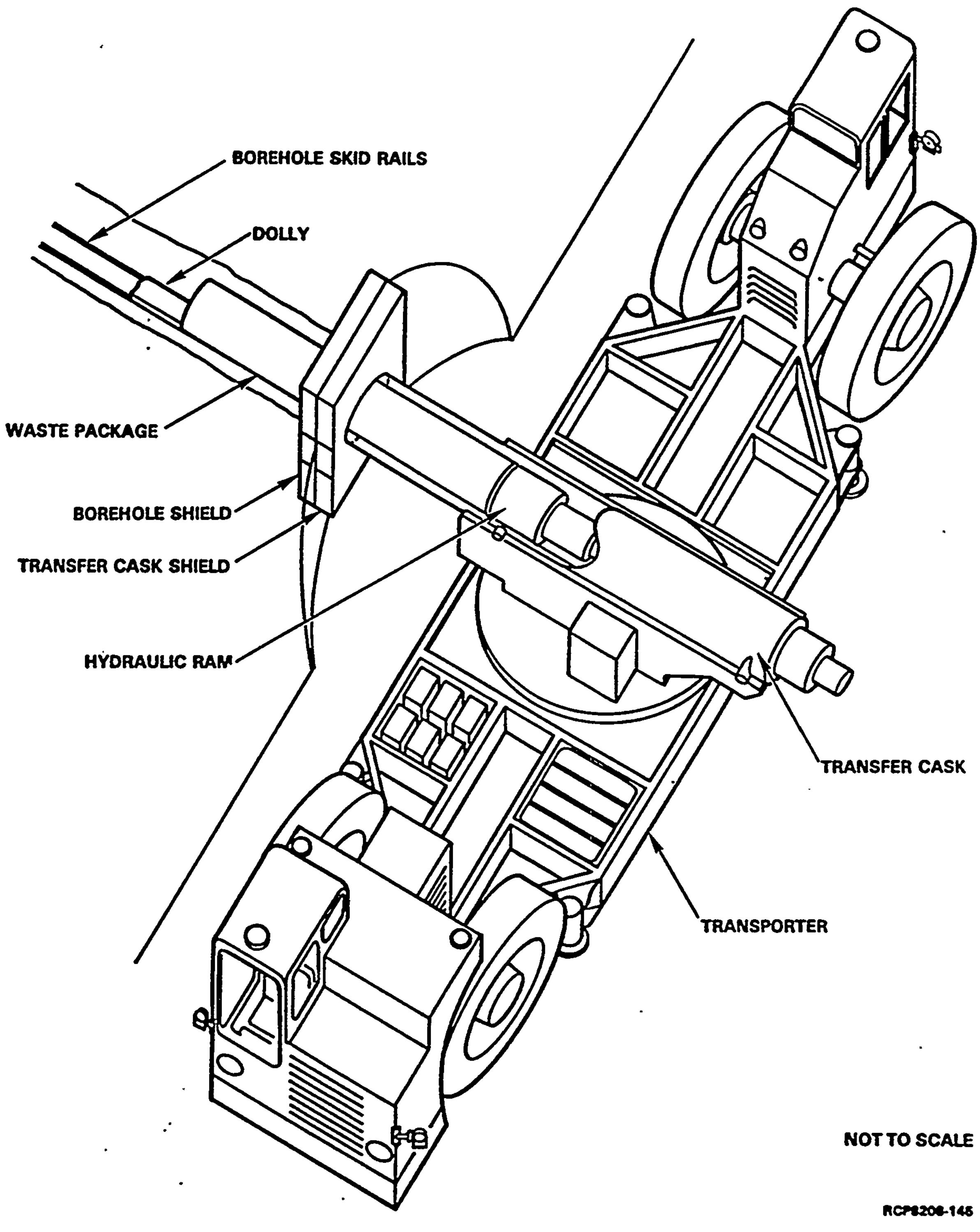

Figure 9.1.7. Emplacement Transporter (1982 Reference Design)

MAJOR FACILITIES

1. PARKING SPACE
2. HELIPAD
3. ADMINISTRATION BUILDING
4. LUNCHROOM
5. PATROL HEADQUARTERS
6. FIRE STATION
7. CONFINEMENT EXHAUST BUILDING
8. SHAFT 1 - CONFINEMENT EXHAUST SHAFT
9. COOLING TOWER
10. SHAFT 2 - WASTE-TRANSPORT SHAFT
11. WASTE-HANDLING BUILDING
12. SHAFT 3 - CONFINEMENT INTAKE SHAFT
13. CONFINEMENT AIR INTAKE BUILDING
14. PERSONNEL- AND MATERIAL-ACCESS FACILITY
15. SHAFT 4 - SERVICE SHAFT
16. MAINTENANCE
17. SHAFT 5 - BASALT-TRANSPORT SHAFT
18. MINE EXHAUST BUILDING
19. CORE STORAGE AND LABORATORY BUILDING
20. RAILCAR AND TRUCK-HOLDING AREA
21. STANDBY GENERATOR BUILDING
22. MINE-WATER-RETENTION PONDS
23. WAREHOUSE
24. PROCESS WATER EVAPORATION PONDS
25. MINE-WATER-PERCOLATION POND
26. SPOILS PILE (PARTIALLY SHOWN)
27. TRAINING CENTER
28. INDUSTRIAL SAFETY BUILDING
29. GATE HOUSE NO. 1
30. SECURITY FENCE

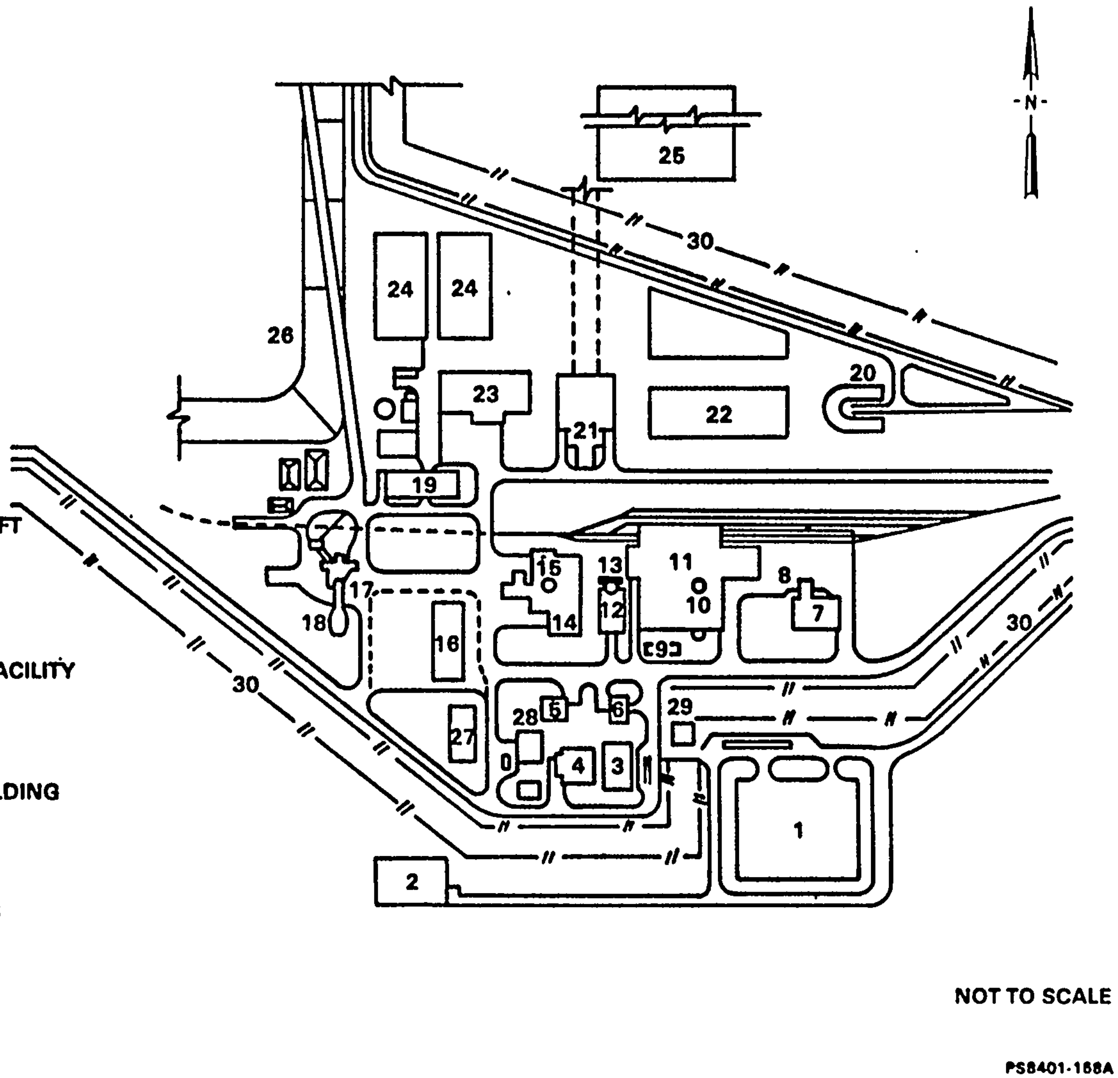

Figure 9.1.8. Surface Facilities (1982 Reference Design)

located on the surface (Figure 9.1.8), will be ventilated by two separately controlled systems: a confinement system for the waste handling and a system for the support and administrative areas. The underground facilities will also be ventilated by two separately controlled systems: a confinement system for waste emplacement and a system for the mining development areas. The confinement ventilation systems will be equipped with redundant fans and air-filtering systems. A radiation monitoring system will extend throughout the confinement airways and in the exhaust ventilation shaft. The intake ventilation air will be cooled and a cooling tower may be required on the surface to dissipate heat.

9.1.3.2.2 Underground Facilities. The principal components of the conceptual underground facility are the shaft pillar area, shafts, and underground drifts (Figure 9.1.3).

The shaft pillar area will provide a minimum number of openings for access, ventilation, and travel. It is best described in terms of its functional areas and its major accessways. The accessways consist of five shafts: basalt transfer, service, confinement air intake, waste transport, and confinement exhaust. The shafts would be constructed using large-hole drilling methods.

The main accesses to emplacement panels will be the airways and travel routes between the shafts and the waste panel areas (Figure 9.1.6). The main accesses will provide the following:

- Accessways connecting the waste shaft to the waste emplacement locations
- Haulageways and accessways for bulk materials, mining equipment, supplies, and personnel
- Accessways for utilities, dewatering lines, and maintenance facilities
- Airways for mining and confinement air systems.

The waste panels in the 1982 conceptual design will be divided into emplacement pillars designed to contain the spent fuel and processed high-level waste containers. A waste panel will be a discrete area encompassing emplacement rooms or corridors. Waste containers will be placed into long horizontal boreholes from the emplacement (vertical emplacement is also currently under investigation). The waste panels (during the operations phase of the repository) will be serviced by two independent ventilation circuits from the confinement ventilation system and the radiation monitoring system.

9.1.3.3 Additional Engineering Design Studies.

This section summarizes some of the engineering studies that have been conducted since the completion of the 1982 reference repository design to determine repository development needs and performance requirements, and to provide additional information to supplement the reference design. These studies were conducted by Raymond Kaiser Engineer, Inc./Parsons-Brinkerhoff Quade and Douglas, Inc., Westinghouse Electric Company, and Rockwell, Hanford Operations. Investigations are still ongoing and the final repository design will be derived from additional studies and results of the exploratory shaft facility site investigation activities.

The repository capacity increase study identified changes in the 1982 design for an increase in the total storage capacity to 72,000 MT (79,200 tons) of heavy metal. Using the 1982 design as a basis for comparison and comparing it to the two-phase repository concept (see below), there would be no significant increase in capital construction costs for the first 5 years resulting from an increase in capacity form 47,400 to 72,000 MT (52,000 to 79,200 tons). The operating costs, however, would increase significantly due to the extended 6-year operational period. The increase in excavation volume of the emplacement panels would be 21%. The additional ventilation capacity would require a 57% increase for fan horsepower and a 44% increase in pre-cooling air.

The shaft optimization study determined the required shaft diameter based on operational requirements, technology requirements for drilling large-diameter shafts, and shaft lining requirements. The shaft optimization study indicated that the largest shaft diameter required for the repository would have a 3.7 m (12 ft) inside diameter. This does not preclude the use of larger shafts. Separating ventilation functions into several shafts allows the use of 3.7 m (12 ft) diameter shafts. This is also feasible and nearly as economical. Based on technical evaluation and judgement, single-pass drilling is preferred over multi-pass. Both drilling methods are feasible and the choice may depend on rig availability and other nontechnical issues. A solid steel, as opposed to a concrete and steel composite lined, was recommended for all shafts.

The tunnel optimization study determined requirements for repository layout, excavation methods, rock support, and drift size and shape. The results of this study were:

- Layout - The findings indicated that a central shaft pillar area with an overall rectangular shape would provide the best layout.

- Excavation methods - The drill-and-blast method was recommended for waste panel development.

- Rock support - Use of cement-grouted dowels (grouted rock bolts) with microsilica shotcrete was the preferred alternative.

- Drift size and shape - The preferred shape in the waste panel (heated) areas was nearly elliptical, approximately 6.7 m (22 ft) wide by 3.3 m (10.7 ft) wide. The height and width would be primarily dictated by the size of the container transporter and the requirements for loading of containers in emplacement boreholes perpendicular to the emplacement room. The main entries, which would have much lower heat loads from the stored waste, might use a circular configuration because of the lower thermal stress superimposed on the in situ stress.

A waste emplacement study considered the emplacement, retrieval, and effect of the thermal loading of the containers on the repository as a whole. The main finding of the study indicated that storage of containers in horizontal boreholes perpendicular to the emplacement room offers the most advantages. Short emplacement boreholes (9 m or 30 ft or less) appeared to offer an advantage over long boreholes (62 m or 200 ft or more), as there was a much higher expectation that emplacement and then retrieval, if necessary, of waste containers could be accomplished successfully. The short emplacement holes used in the study were 76 cm (30 in) in diameter. The diameter of the boreholes may be increased should performance analysis indicate a need for a thicker container and/or packing material (backfill).

An alternate waste package study considered waste container designs in conjunction with the emplacement study.

The shaft, tunnel, and waste emplacement studies described above used a maximum storage capacity of 47,400 MT (52,000 tons) of heavy metal as in the 1982 design.

The repository underground layout study is based on a single phase of waste receipt to accommodate 70,000 MT (77,000 tons) with a minimum age of 10 years. Expected waste receipt rates are up to 1,800 MT (1,980 tons) per year during the first 5 years of operation and up to 3,000 MT (3,300 tons) of heavy metal per year thereafter until capacity is reached. Tables 9.1.1 to 9.1.5 show the differences between the repository underground layout study and the 1982 reference concept. The changes were caused by the following factors, among others:

- Increase in total repository capacity
- Change in emplacement borehole length
- Single canister emplacement in boreholes
- Change in reference basalt flow
- Change in maximum stress directions.

The physical dimensions for the repository underground layout study, on which the current design concept is based, are shown in Table 9.1.4. The repository length to width ratio changed slightly, but the areal extent increased by about 50% due to the increase of almost 50% in capacity and the change from emplacement of several canisters in one long borehole to single canisters in shorter boreholes. The number of shafts has increased to nine (Table 9.1.5). In summary, the repository underground layout is basically similar to the 1982 design, but has been sized to accommodate 70,000 MT (77,000 tons) of waste. Some of the ventilation shafts are located on the outer periphery of the repository in the shaft pillar to enhance constructibility and safety.

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Description	1982 conceptual design (RKE/PB, 1983)	Repository underground layout study (RKE/PB, 1984e)
Container heat generation (pressurized water reactor), W	1,650	2,200
Container outside diameter cm (in.)	41.7 (16.4)	50.3 (19.8)
Container length, cm (in.)	411 (162)	411 (162)
Packing method*	Pneumatic	Prepackaged sections
Packing thickness, cm (in.)	15.2 (6)	15.2 (6)
Emplacement borehole diameter, cm (in.)	76.2 (30)	89.0 (35)
Emplacement borehole pitch (center to center spacing), m (ft)	18.3 (60)	6.7 (22)
Emplacement borehole length, m (ft)	61 (200)	6.1 (20)
Containers of pressurized water reactor spent fuel per emplacement borehole	13	1
Emplacement room container length, m (ft)	0.7 (2.3)	3.3 (10.8)

*Packing is a mixture of bentonite and crushed basalt.

Table 9.1.1. Design Comparison of Waste Emplacement Features

Description	1982 conceptual design (RKE/PB, 1983)	Repository underground layout study (RKE/PB, 1984e)
Reference basalt flow	Umtanum	Cohassett
Depth to bottom of flow interior, m (ft)	1,170 (3,838)	983 (3,224)
Estimated total thickness of flow interior, m (ft)	65.5 (215)	68.1 (223)
Intact compressive strength, MPa (lbf/in^2)	207 (30,000)	290 (42,050)
Allowable rock stress in emplacement borehole, MPa (lbf/in^2)	186 (27,000)	200 (29,100)
Allowable rock stress in emplacement room, MPa (lbf/in^2)	166 (24,000)	157 (22,800)
In situ stress ratio (maximum horizontal to vertical)	2:1	2.5:1
Maximum in situ stress magnitude, MPa (lbf/in^2)	60 (8,700)	58 (8,410)
Maximum in situ stress direction	East-west	N. 3° W.

Table 9.1.2. Design Comparison of Geomechanical Data

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Description	1982 conceptual design (RKE/PB, 1983)	Repository underground layout study (RKE/PB, 1984e)
Maximum number of shafts	5	9
Maximum allowable shaft inside diameter, m (ft)	No limit defined	3.7 (12)
Main airway (drift) maximum velocity, m/min (ft/min)	457 (1,500)	457 (1,500)
Service shaft maximum velocity, m/min (ft/min)	610 (2,000)	610 (2,000)
Ventilation shafts maximum velocity, m/min (ft/min)	1,067 (3,500)	1,220 (4,000)
Ambient rock temperature, °C (°F)	57 (134)	52 (125)

Table 9.1.3. Design Comparison of Ventilation Requirements

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Description	1982 conceptual design (RKE/PB, 1983)	Repository underground layout study (RKE/PB, 1984e)
Overall underground dimensions, m (ft)	1,610 x 3,360 (5,270 x 11,020)	1,930 x 4,150 (6,330 x 13,600)
Areal extent, ha (acres)	540 (1,334)	800 (1,978)
Total length of emplacement rooms, m (ft)	27,600 (90,600)	156,100 (512,000)
Size of emplacement rooms, m (ft)	3.1 x 6.1 (10 x 20)	3.1 x 7.0 (10 x 23)
Pillar width, m (ft)	65 (212)	31 (100)

Table 9.1.4. Design Comparison of Physical Dimensions

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Description	1982 conceptual design (RKE/PB, 1983)	Repository underground layout study (RKE/PB, 1984e)
Maximum air volume and phase, m^3/min (ft^3/min)	Backfilling, 11,980 (4.2×10^5)	Operations, 30,300 (1.07×10^6)
Room cool-down time for backfilling, d	90	53
	Number and diameter of shafts	
Waste handling	One 3.7-m (12-ft) ID	One 3.7-m (12-ft) ID
Service and air intake	One 4.9-m (16-ft) ID	One 3.7-m (12-ft) ID
Basalt hoisting and air exhaust	One 4.3-m (14-ft) ID	One 3.7-m (12-ft) ID
Mine air intake	Not applicable	One 3.1-m (10-ft) ID
Mine air exhaust	Not applicable	One 3.1-m (10-ft) ID
Confinement air intake	One 3.7-m (12-ft) ID	Two 3.7-m (12-ft) ID
Confinement air exhaust	One 3.4-m (11-ft) ID	Two 3.7-m (12-ft) ID

ID = inside diameter.

Table 9.1.5. Design Comparison of Ventilation Features

In 1984, a feasibility study of the two-phase receipt approach to a repository was conducted. The features of this concepts are:

- Ninety-six year schedule from start of construction to completion of decommissioning
- A spent fuel storage capacity of 70,000 MT (77,000 tons) based on a phase 1 annual waste acceptance rate of 400 MT (440 tons) for the first 3 years, and a phase 2 acceptance rate of 500 and 1,400 MT (550 and 1,540 tons) per year for the fourth and fifth years, and 3,000 MT (3,300 tons) thereafter
- Utilizing the exploratory shafts to expedite repository construction and using one or both for repository operations
- Utilizing a Washington Public Power Supply System building near the reference repository location as a limited waste packaging and handling facility.

The two-phase concept has not been examined as closely as the single-phase concepts and will be verified or evaluated in future design activities.

Table 9.1.6 presents a comparison of the various design concepts.

9.1.4 Repository Construction

9.1.4.1 Schedule

The repository operations area will be designed so that the emplaced waste could be retrieved on a reasonable schedule, starting any time up to 50 years after waste emplacement operations are initiated, unless otherwise approved by the Nuclear Regulatory Commission. For design purposes, it was assumed that retrieval operations would take as long as repository construction and waste emplacement operations. Figure 9.1.9 shows the repository schedule, representing the beginning of the retrieval option 56 years after the start of construction. Start of waste emplacement is scheduled for 1998 as mandated in the Nuclear Waste Policy Act.

9.1.4.2 Methods and Equipment

Repository construction methods and technology will be demonstrated by the construction of the Exploratory Shaft Facility. The blind-hole drilling method is proposed for the construction of the exploratory shafts as well as the repository shafts. The exploratory shafts are to be blind-drilled and lined with a watertight steel casing, the annulus between the steel casing and rock being filled with grout. Figure 9.1.10 demonstrates the principle. The shaft will have a 183 cm (72 in) inside diameter, and will be deep

Repository characteristics	Reference design[a]	Two-phase feasibility study[b]	Other design studies[c]	Current design concept	Changes in projected environmental impacts	Changes in projected transportation impacts	Changes in projected socioeconomic impacts
1. Incorporation of exploratory shafts	No	Yes: One 1.8 m (6 ft) one 3.7 m (12 ft)	No	Yes: Two 1.8 m (6 ft)	Incorporation of exploratory shaft(s) would result in a minimal decrease in the amount of excavated material. The impact from excavated material is directly related to the size of the spoils pile, which is based on total excavated metric tons (tons). Whether these spoils are stockpiled for future use as backfill or left on the surface and reclaimed, the resulting impact is a temporary loss of surface habitat. The excavated material from the shaft(s) is expected to be an insignificant fraction of the overall spoils, resulting in an insignificant reduction of impact.	There would be no significant differences in impacts among concepts.	There would be no significant differences in impacts among concepts.
2. Number and size of shafts	Two 3.7 m (12 ft) one 4.3 m (14 ft) one 4.9 m (16 ft)	One 1.8 m (6 ft) four 2.7 m (8.75 ft) four 3.7 m (12 ft)	Seven 3.7 m (12 ft) two 3.0 m (10 ft)	Seven 3.7 m (12 ft) two 3.0 m (10 ft) two 1.8 m (6 ft) exploratory shafts	The impact related to this repository characteristic would be a temporary loss of surface habitat from placement of excavated material. The overall impact is expected to be insignificant for all four concepts compared to the total amount of material excavated from the repository.	There would be no significant differences in impacts among concepts.	There would be no significant differences in impacts among concepts.
3. Total excavated rock, metric tons (tons)	6,572,800 (7,240,400)	11,487,300 (12,654,000)	11,902,500 (13,111,400)	(d)	A notable impact related to this repository characteristic would be a temporary loss of surface habitat from placement of excavated material. The estimated surface area temporarily lost is 81 hectares (200 acres) for the reference design and 190 hectares (470 acres) for the three other concepts. Such habitat loss, in that it is temporary and a diminutive fraction of the regional shrub-steppe habitat, would be insignificant in itself and among the concepts. The most notable potential impact would be the occasional generation of fugitive dust. Recent air-quality impact analyses indicate that levels of fugitive dust would be minimal at locations of public access (Subsection 5.2.1.3.2). Standard mitigation measures would not only limit these emissions to insignificant levels, but afford little difference between concepts.	There would be no significant differences in impacts among concepts.	There would be no significant differences in impacts among concepts.
4. Total area, ha (acres): Surface Subsurface	80 (200) 540 (1,334)	190 (470) 824 (2,036)	(d) 800 (2,000)	(d) (d)	A notable impact related to this repository characteristic would be a temporary loss of surface habitat from placement of excavated material. The estimated surface area temporarily lost is 81 hectares (200 acres) for the reference design and 190 hectares (470 acres) for the three other concepts. Such habitat loss, in that it is	There would be no significant differences in impacts among concepts.	There would be no significant differences in impacts among concepts.

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Table 9.1.6. Comparison of Alternate Repository Design Concepts (Page 1 of 4)

Repository characteristics	Reference design[a]	Two-phase feasibility study[b]	Other design studies[c]	Current design concept	Changes in projected environmental impacts	Changes in projected transportation impacts	Changes in projected socioeconomic impacts
4. Total area, ha (acres): Surface Subsurface (cont.)					temporary and a diminutive fraction of the regional shrub-steppe habitat, would be insignificant in itself and among the concepts. The most notable potential impact would be the occasional generation of fugitive dust. Recent air-quality impact analyses indicate that levels of fugitive dust would be minimal at locations of public access (Subsection 5.2.1.3.2). Standard mitigation measures would not only limit these emissions to insignificant levels, but afford little difference between concepts.		
5. Preclosure period (yr):[f] Construction Operation Decommission	 8 50 (d)	 4 50 42	 4 50 (d)	 (e) 50 (d)	The most notable impact related to the repository preclosure period would be the temporary loss of surface habitat from placement of excavated material and surface facilities. Even though the construction period for each concept varies by as much as a factor of 2.6, this difference is insignificant compared to the total time estimated for construction, operation, and decommissioning. Such habitat loss, in that it is temporary and a diminutive fraction of the regional shrub-steppe habitat, would be insignificant in itself and among the concepts. The most notable potential impact would be the occasional generation of fugitive dust. Recent air-quality impact analyses indicate that levels of fugitive dust would be minimal at locations of public access (Subsection 5.2.1.3.2). Standard mitigation measures would not only limit these emissions to insignificant levels, but afford little difference between concepts.	Radiological transportation impacts are reduced due to radioactive decay of the waste forms to an insignificant degree if waste shipments are delayed beyond the planned time. For a given waste quantity, variations in waste receiving period do not change overall transportation impacts, but do change annual transportation impacts in inverse fashion (i.e., lengthening the receiving period reduces annual transportation impacts). Impacts resulting from those differences are considered to be less significant than those resulting from the reference design.	Small differences among concepts in duration of periods will not result in significant differences in potential impacts.
6. Total capacity, metric tons (tons) of heavy metal	47,400 (52,200)	70,000 (77,000)	70,000 (77,000)	70,000 (77,000)	A notable impact related to this repository characteristic would be a temporary loss of surface habitat from placement of excavated material. The estimated surface area temporarily lost is 81 hectares (200 acres) for the reference design and 190 hectares (470 acres) for the three other concepts. Such habitat loss, in that it is temporary and a diminutive fraction of the regional shrub-steppe habitat, would be insignificant in itself and among the	For a given waste type (e.g., spent fuel), transportation impacts are roughly proportional to the amount of waste shipped. Transportation impact changes associated with other waste types are identified in Item 8. Impacts resulting from those differences are considered to be less significant than those resulting from the reference design.	Greater capacity may increase the size of peak work force and preclosure period; however, differences in impacts are not expected to be significant. Increased population and demands for community services and facilities are small compared to the existing populations and infrastructure.

Table 9.1.6. Comparison of Alternate Repository Design Concepts
(Page 2 of 4)

Repository characteristics	Reference design[a]	Two-phase feasibility study[b]	Other design studies[c]	Current design concept	Changes in projected environmental impacts	Changes in projected transportation impacts	Changes in projected socioeconomic impacts
6. Total capacity, metric tons (tons) of heavy metal (cont.)					concepts. The most notable potential impact would be the occasional generation of fugitive dust. Recent air-quality impact analyses indicate that levels of fugitive dust would be minimal at locations of public access (Subsection 5.2.1.3.2). Standard mitigation measures would not only limit these emissions to insignificant levels, but afford little difference between concepts.		
7. Peak receipt rate, metric tons (tons) of heavy metal per year	2,370 (2,650)	3,000 (3,300)	3,000 (3,300)	Spent fuel: 3,000 (3,300), West Valley high-level waste, and defense high-level waste: 400 (440)	Potential radiological impacts would be directly proportional to maximum waste receipts per year. The radiological impacts for the reference design are summarized in Subsection 6.2.2.1. Radiological releases from any repository concept will be regulated by 10 CFR 20, 10 CFR 60, and 40 CFR 191 and can be considered insignificant.	For a given annual waste type (e.g., spent fuel), transportation impacts are roughly proportional to the annual receipt rate. Changes in transportation impacts for alternative waste types are identified in item 8. Impacts resulting from those differences are considered to be less significant than those resulting from the reference design.	Differences in receipt rates may result in differences in peak work force; however, changes in impacts are not expected to be significant.
8. Waste inventory	50% spent fuel 50% commercial high-level waste	100% spent fuel	100% spent fuel	89% spent fuel; 11% defense high-level waste and West Valley high-level waste	The reference design would have the highest potential impacts on a per metric ton (ton) of waste basis due to the greater spacing requirement; therefore, a greater volume of rock would be removed per metric ton (ton) of waste emplaced. The lower heat load per equivalent metric ton (ton) of heavy metal of defense high-level waste and West Valley high-level waste would permit closer spacing of waste containers. However, the equivalent metric tonnage (tonnage) per container would be less than for spent fuel. Consequently, the volume of rock excavated and total underground layout area to accommodate defense high-level waste and West Valley high-level waste may increase with the current design concept. Standard mitigation measures would limit this to insignificant levels, affording little difference between concepts.	Transportation impacts of replacing spent fuel by an equivalent amount (spent fuel weight basis) of high-level waste are increased for a mixture of defense high-level waste and West Valley high-level waste, and reduced for possible future commercial high-level waste. These changes result from shipping cask payload variations. Refer to Section 5.1.7 for a discussion of the relationships between shipments, payloads, and waste types. The differential impacts are less significant than those impacts resulting from the reference design.	There would be no significant differences in impacts among concepts.
9. Emplacement configuration	Long horizontal boreholes	Short horizontal boreholes	Short horizontal boreholes	Short horizontal boreholes	Short borehole emplacement would increase the size of the repository and the size of the spoils pile. The resulting potential impacts are identified in item 3 and are considered insignificant. However, short borehole emplacement would provide increased reliability for emplacement and retrieval; therefore, potentially decreasing worker exposure and decreasing potential long-term radiological impacts.	There would be no significant differences in impacts among concepts.	There would be no significant differences in impacts among concepts.

Table 9.1.6. Comparison of Alternate Repository Design Concepts (Page 3 of 4)

Repository characteristics	Reference design[a]	Two-phase feasibility study[b]	Other design studies[c]	Current design concept	Changes in projected environmental impacts	Changes in projected transportation impacts	Changes in projected socioeconomic impacts
10. Waste handling buildings	1	1 offsite 1 onsite	1	2	There would be no significant differences in potential impacts among concepts.	There would be no significant differences in impacts among concepts. For the two-phase repository concept, an existing receiving facility on the Hanford Site would be inserted into the transportation route; the transportation impacts (excluding the waste receiving and handling impacts) of this additional receiving facility would be insignificant.	There would be no significant differences in impacts among concepts.
11. Peak utilities demand: Water, m^3/min (gal/min) Electricity, kW/day Fuel m^3/day (gal/day)	 13 (3,400) 29,000 5 (1,330)	 (g) (g) (g)	 (g) (g) (g)	 (g) (g) (g)	There would be no significant differences in potential impacts among concepts.	There would be no significant differences in impacts among concepts.	There would be no significant differences in impacts among concepts.
12. Peak work force: Construction Operations	 1,100 900	 (g) 950	 (d) (d)	 (d) (d)	There would be no significant differences in potential impacts among concepts.	There would be no significant differences in impacts among concepts.	There would be no significant differences in impacts among concepts. The accompanying changes in population and increased demands for community services and facilities would be small compared to the existing populations and infrastructure.
13. Access improvements: Road, km (mi) Rail, km (mi)	 4.8 (3.0) 4.8 (3.0)	 2.0 (1.2) 6.6 (4.1)	 (d) (d)	 (d) (d)	The most notable impact related to this repository characteristic would be a temporary loss of surface habitat from access improvement. This impact is expected to be insignificant for all four concepts compared to the total amount of habitat lost due to the repository.	The access route distances shown are preliminary design-basis estimates that have not yet been verified by access route designs. The differences in access improvements among concepts are judged to be small; therefore, there would be no significant differences in transportation impacts among concepts.	There would be no significant differences in impacts among concepts.

[a]1982 conceptual design (RKE/PB, 1983).
[b]Two-phase repository feasibility study (Rockwell, 1984).
[c]Repository underground layout study (RKE/PB, 1984e).
[d]Estimated to be nearly the same (within plus or minus 10 percent) as for the two-phase repository concept.
[e]Complies with mission plan (DOE, 1985).
[f]Preclosure periods defined as follows:
Construction--Start of construction to start of waste emplacement.
Operation--Start of waste emplacement to end of caretaker period (may include backfilling of portions of the underground facility if such work is started prior to the end of the caretaker period).
Decommission--End of caretaker period to completion of shaft and borehole sealing.
[g]Estimated to be nearly the same (within plus or minus 10 percent) as for the reference design case.

Table 9.1.6. Comparison of Alternate Repository Design Concepts (Page 4 of 4)

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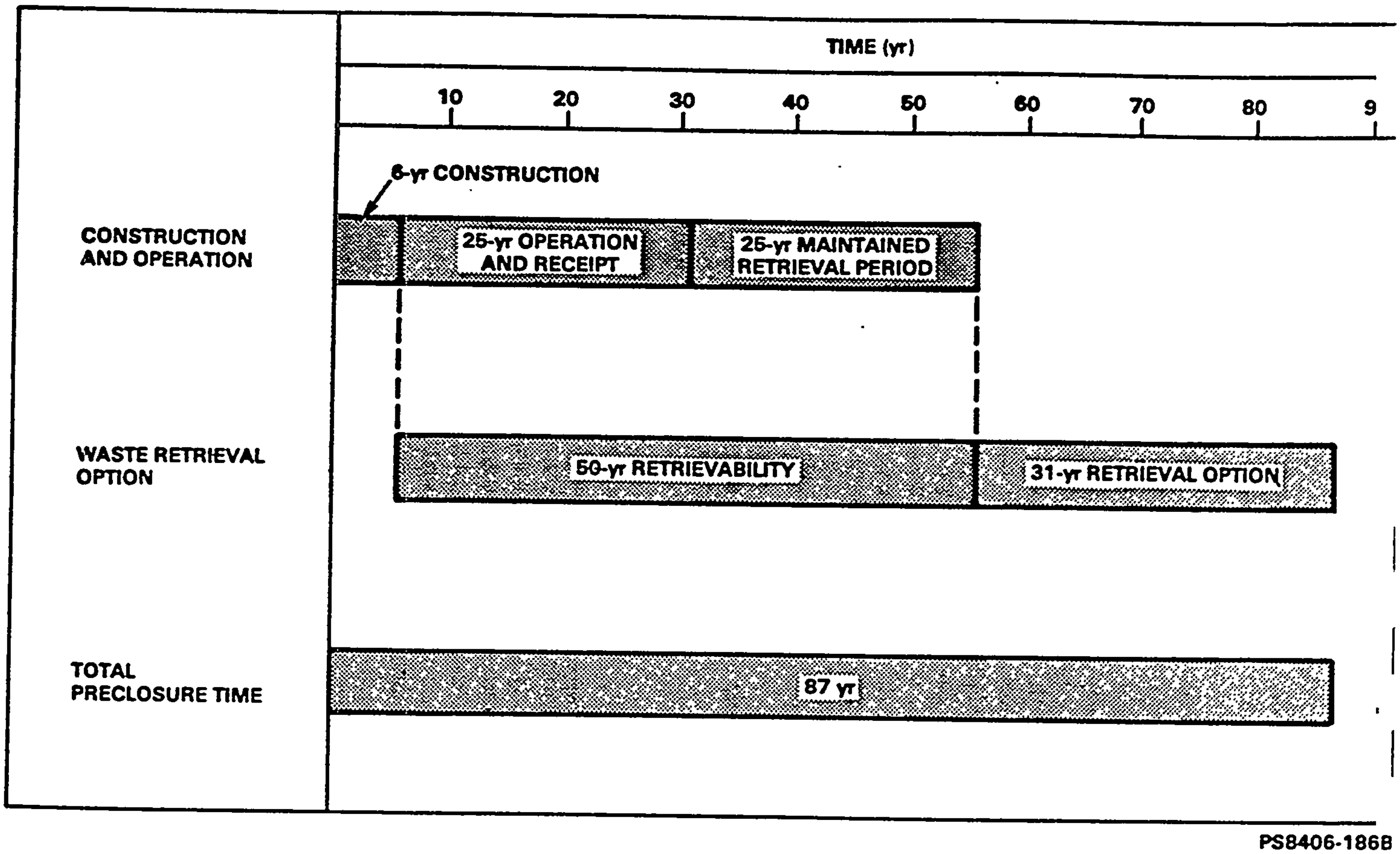

Figure 9.1.9. Repository Schedule

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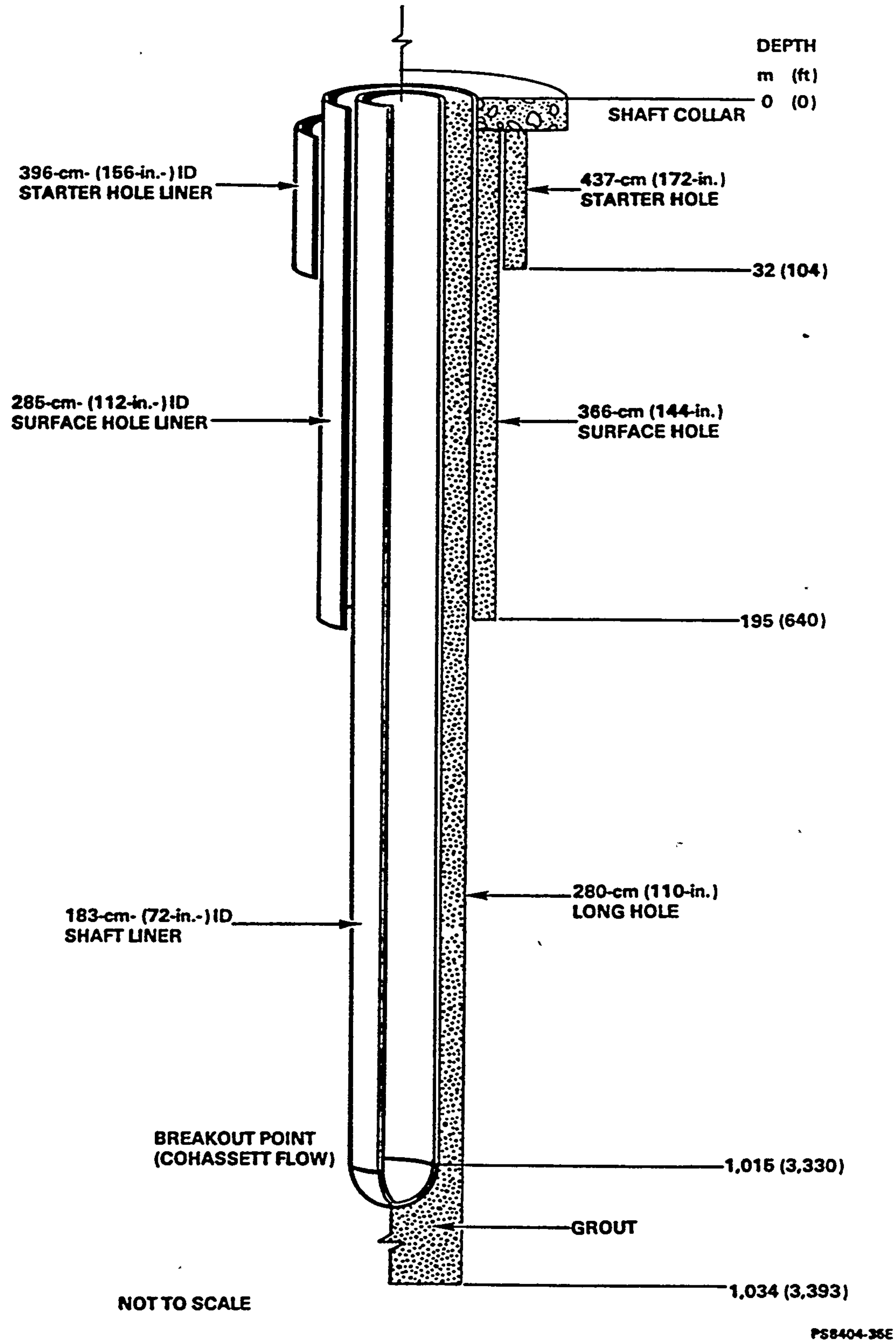

Figure 9.1.10. Drilled Shaft Concept

enough to permit shaft station breakout in the Cohassett flow interior. The shafts will be drilled to a depth of about 1,034 m (3,393 ft). During drilling, the shaft will be filled with drilling fluid to provide static support to the shaft wall, cool and lubricate the drill bit and reamers, and carry cuttings to the surface by fluid circulation. The shaft will penetrate several aquifers and the hydrostatic head of the drilling fluid would minimize ground-water inflows. After the completion of drilling, a steel liner will be floated into place and cement grout injected between the liner and rock to cement the liner in place. Several casings will be installed, one inside the other (Figure 9.1.10), to provide a starter hole at the surface and a surface casing to isolate the sediments above the top of the basalt flows. After completion of shaft drilling, the drill rig will be removed from the completed shaft and the headframe erected. Using a shaft sinking work deck, or similar arrangement, the shaft can then be furnished. At the appropriate horizon, the liner will be penetrated to allow station excavation, followed by horizontal level development.

Due to the strength of the basalt, the most effective repository development method will most likely be drill-and-blast methods. The method and equipment (e.g., jumbos, haulage vehicle, etc.,) are still under investigation to find an optimum combination. Application of controlled blasting methods is also under investigation to minimize the damaged rock zone around the excavations.

9.1.5 Waste Emplacement Cycle

9.1.5.1 Waste Treatment

Reprocessing of commercial spent fuel is currently not being practiced in the United States so that the majority of the nuclear waste will be in the form of spent fuel assemblies. About 62,000 MT (68,000 tons) of spent fuel were estimated to require disposal in the first repository. Another 8,000 MT (9,000 tons) equivalent of defence high-level reprocessing waste and commercial reprocessing waste from West Valley will also be disposed of in the first repository. The spent fuel assemblies will be repackaged in special containers in preparation for underground disposal. The high-level reprocessing wastes will be vitrified in borosilicate glass and encased in disposal containers.

9.1.5.2 Waste Transport

Waste transport systems are still under analyses. Rail and truck transport are the options being investigated. Transport cask configurations are also the subject of scientific and engineering studies.

9.1.5.3 Waste Package

The reference consolidated spent fuel waste package concept is shown in Figure 9.1.11. It consists of the consolidates spent fuel waste

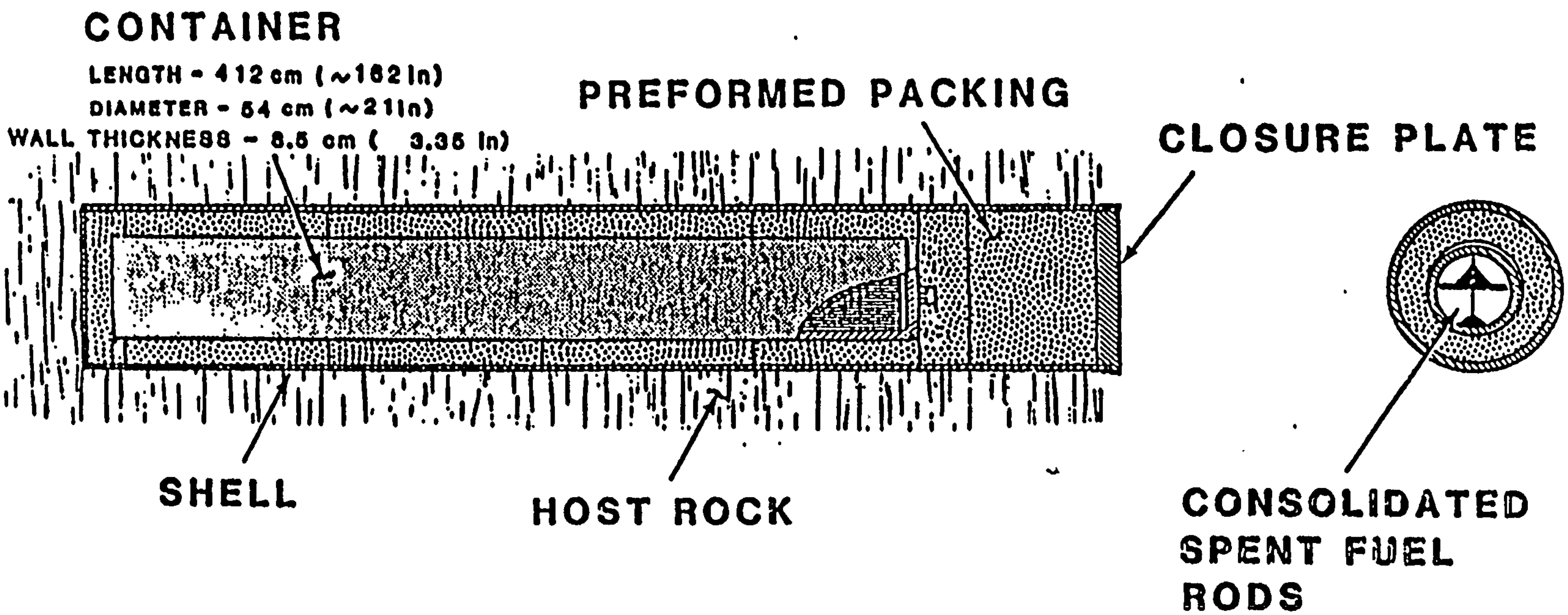

Figure 9.1.11. Waste Package Reference Concept for Consolidated Spent Fuel

form, a carbon-steel container, preformed packing material sections, and a steel shell. A single waste package is inserted in a horizontal borehole. Previous plans to emplace multiple packages in long horizontal boreholes were abandoned because of the complexity of emplacement and retrieval. The container is designed to hold the rods from four pressurized water reactor assemblies generating about 2,200 watts in total. A cruciform insert divides the container interior into four equal quadrants. For traceability, all the rods from one spent fuel assembly are placed in one quadrant. The container is filled with a protective gas prior to sealing. The option to use this container design with fuel of higher heat output (younger or higher burnup spent fuel) could be exercised by placing a smaller amount of fuel in the container. This could be accomplished by altering the internal structure of the container or by the addition of a nonfissile filler, with basalt a preferred candidate. The packing material is a mixture of 25% sodium bentonite and 75% crushed basalt that is fabricated into preformed units. These are loaded into a carbon-steel shell above ground. The assembled packing and shell are transported below ground where they are inserted into a horizontal borehole. The container of consolidated spent fuel is then inserted into the center of the packing, end packing sections are inserted, and the closure plate installed.

The design concepts for spent fuel assemblies and commercial high-level waste are identical with that for consolidated spent fuel as illustrated by Figures 9.1.12 and 9.1.13. Dimensions change to accommodate the various waste forms and these are shown in Table 9.1.7 for the three designs. In the case of spent fuel assemblies, four assemblies are placed into a basket which is inserted into the container cavity. Void space around the basket is filled with crushed basalt.

Table 9.1.7. Waste Package Dimensions for Emplacement in Basalt. Dimensions are in mm (in).

	CSF(1)	SFA(2)	HLW(3)
Container O.D.	541 (21.3)	805 (31.7)	813 (32.0)
Wall Thickness	85 (3.35)	85 (3.35)	89 (3.5)
Head Thickness	127 (5.0)	127 (5.0)	103 (4.1)
Packing O.D.	871 (34.3)	1135 (44.7)	1146 (45.1)
Packing Thickness	152 (6.0)	152 (6.0)	152 (6.0)
Shell O.D.	894 (35.2)	1158 (45.6)	1168 (46.0)
Shell Thickness	8 (0.3)	8 (0.3)	8 (0.3)

Source: Gilbert/Commonwealth, 1986.

(1) Dismantled and consolidated spent fuel assemblies
(2) Spent fuel assembly
(3) High-level waste

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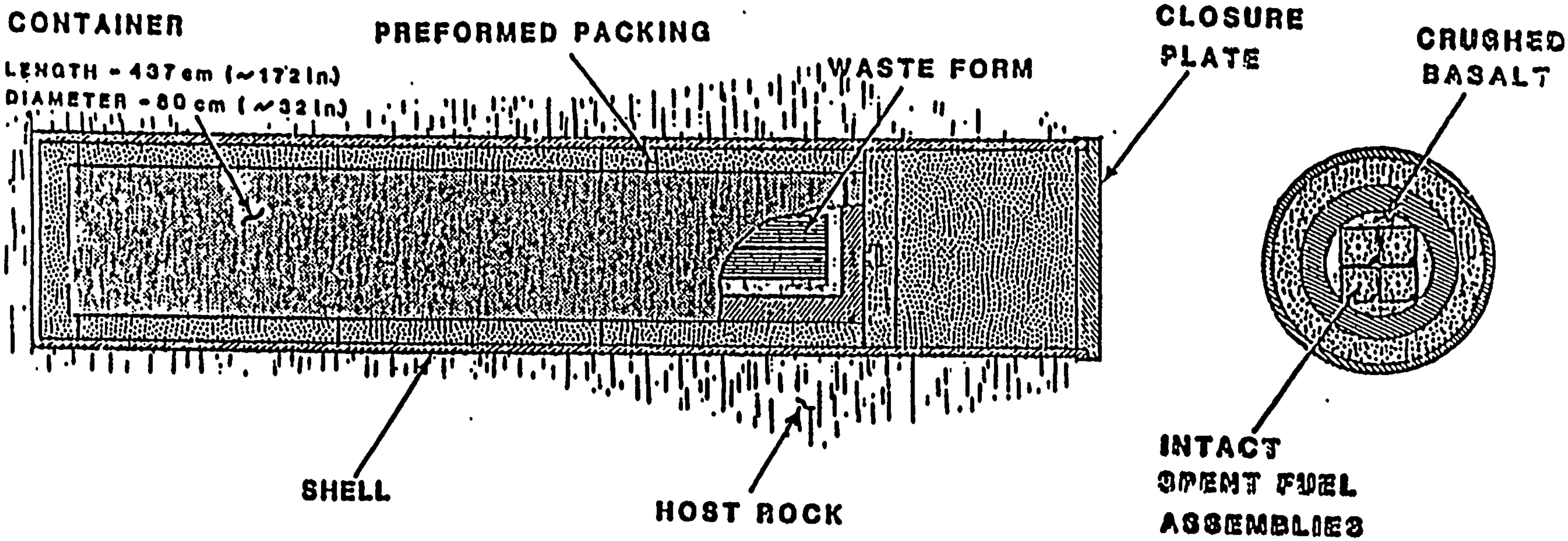

Figure 9.1.12. Waste Package Reference Concept for Intact Spent Fuel

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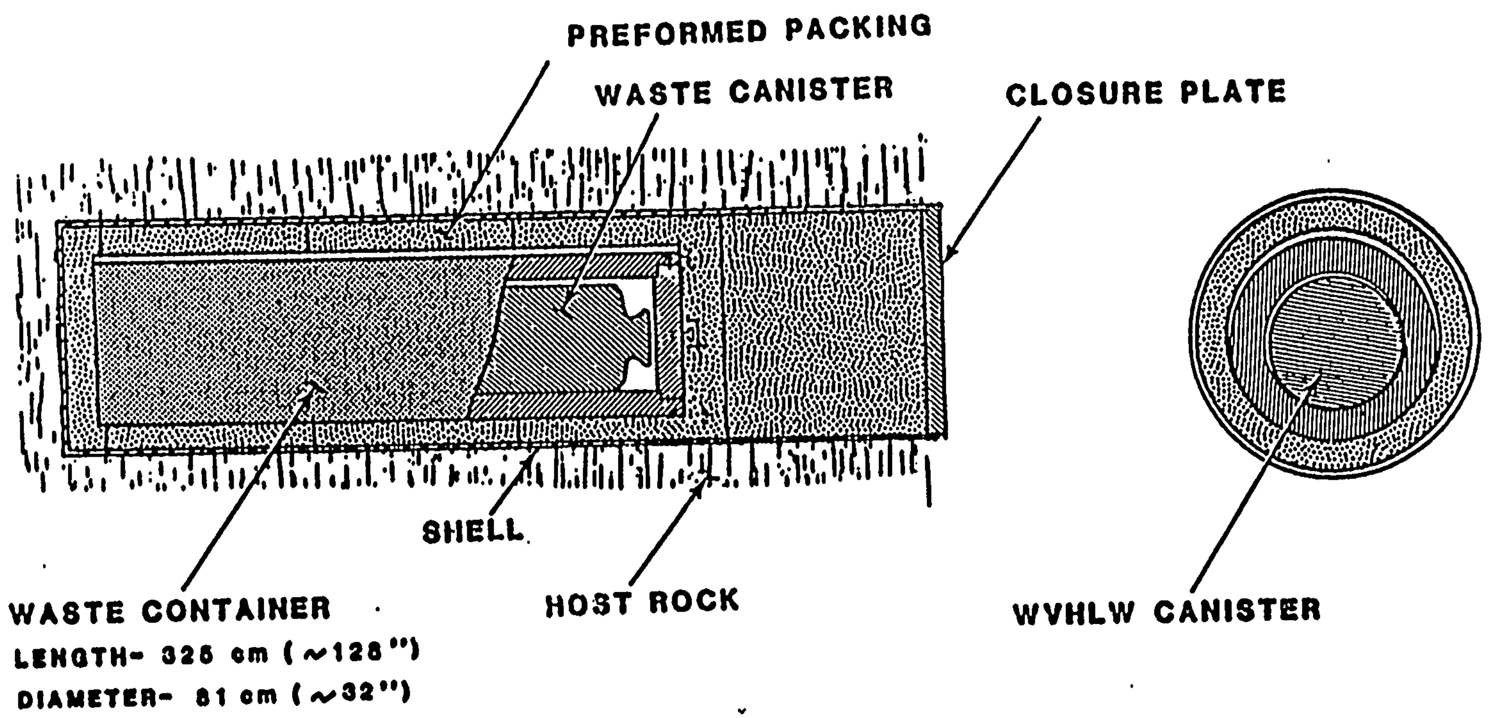

Figure 9.1.13. Waste Package Reference Concept for High-Level Wastes

Similarly, for high-level waste, after the waste canister is placed into the container, the remaining void space is filled with crushed basalt. The reference heat output for a canister of high-level waste, as produced by West Valley, is a maximum of 255 W, significantly less than the spent fuel waste forms. The canister is made of stainless steel.

An alternate design concept for use with consolidated spent fuel also has been developed as illustrated in Figures 9.1.14. In this alternate design, the total packing thickness is increased to 1.0 m (3.28 ft). Because of the thicker packing and its low thermal conductivity, a reduction from 4 to 3 consolidated spent fuel assemblies is necessary to prevent calculated waste form temperatures from exceeding specified temperature limits.

9.1.5.4 Underground Waste Emplacement

The concepts being studies for waste emplacement were summarized in Section 9.1.3.3 and 9.1.5.3. Tables 9.1.1 and 9.1.4 to 9.1.6 outline the proposed emplacement concept parameters. Horizontal emplacement is the preferred concept at this time to reduce the vertical extent of the repository, but vertical emplacement is also being investigated due to the advantages that are offered by easier canister handling.

Figures 9.1.15 to 9.1.17 show the single canister horizontal borehole emplacement method under consideration. The emplacement holes would be drilled on either side of the disposal room and filled with the packing and shell liner. The waste package is inserted inside the packing material and the borehole is plugged.

9.1.6 Backfilling and Sealing

The primary components of the repository seal system are the shaft seals since the shafts provide a direct connection between the underground facility and the accessible environment. Backfill is placed in the underground facility to restrict ground-water movement between the waste emplacement areas and the shafts, and, in some cases, provide structural support to underground openings in some areas of the repository, or may be provided to act as a sink for sorption of radionuclides in the waste emplacement areas. Backfill will also be placed in the shafts, as well as boreholes drilled from the surface to within the basalt flows a repository depth. Crushed basalt mixed with bentonite is currently being proposed as the major component of backfill provided to inhibit groundwater flow. The composition of backfill materials to be used in the repository seals will be further analyzed during advanced conceptual studies.

Crushed backfill and bentonite mixtures are also proposed for the emplacement rooms and access drifts. Concrete bulkheads will be placed also to isolate drifts and emplacement rooms. Injection grouting may be used to seal fractures that exist in the exposed rock.

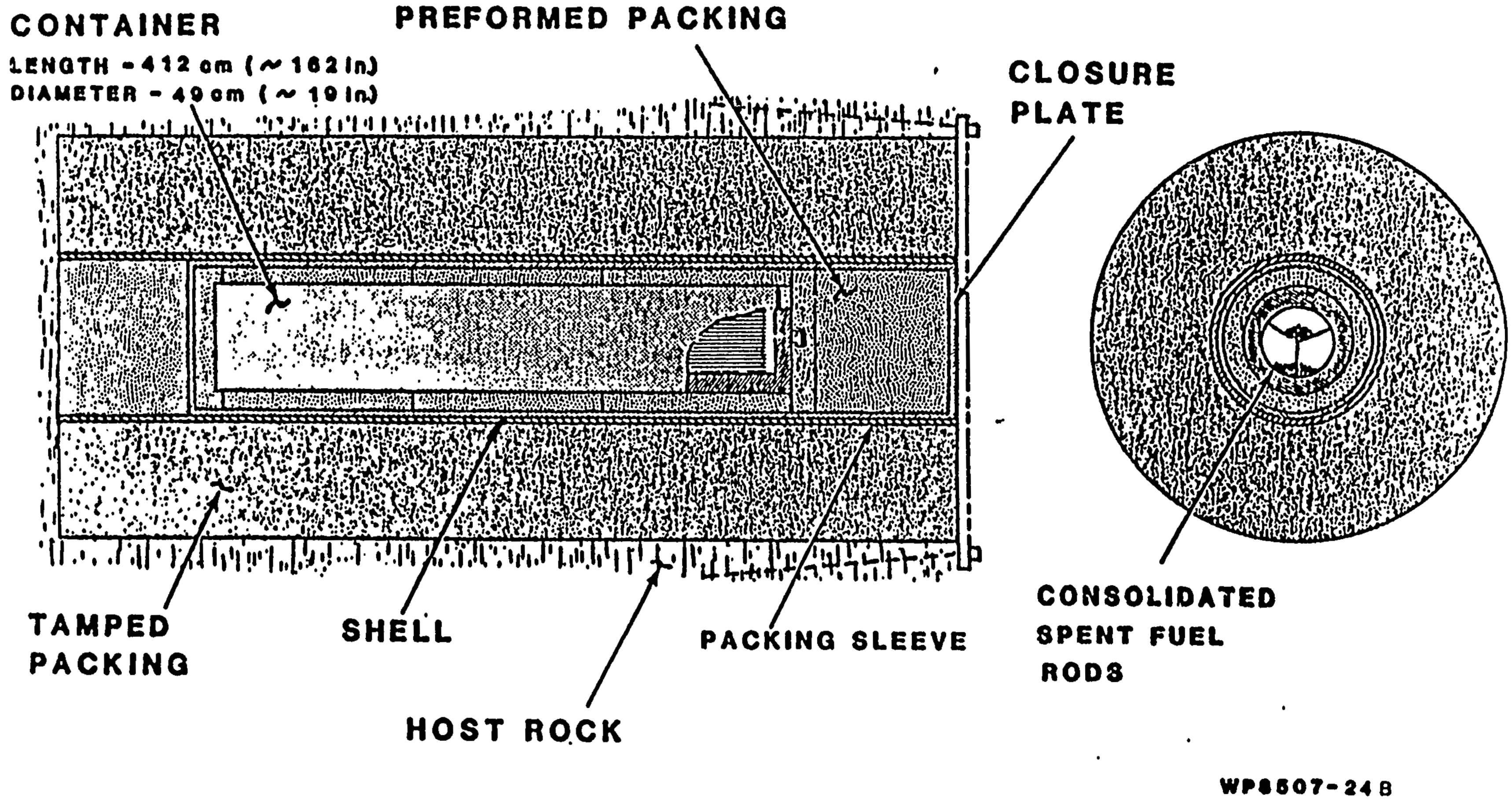

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Figure 9.1.14. Alternate Waste Package for Consolidated Spent Fuel

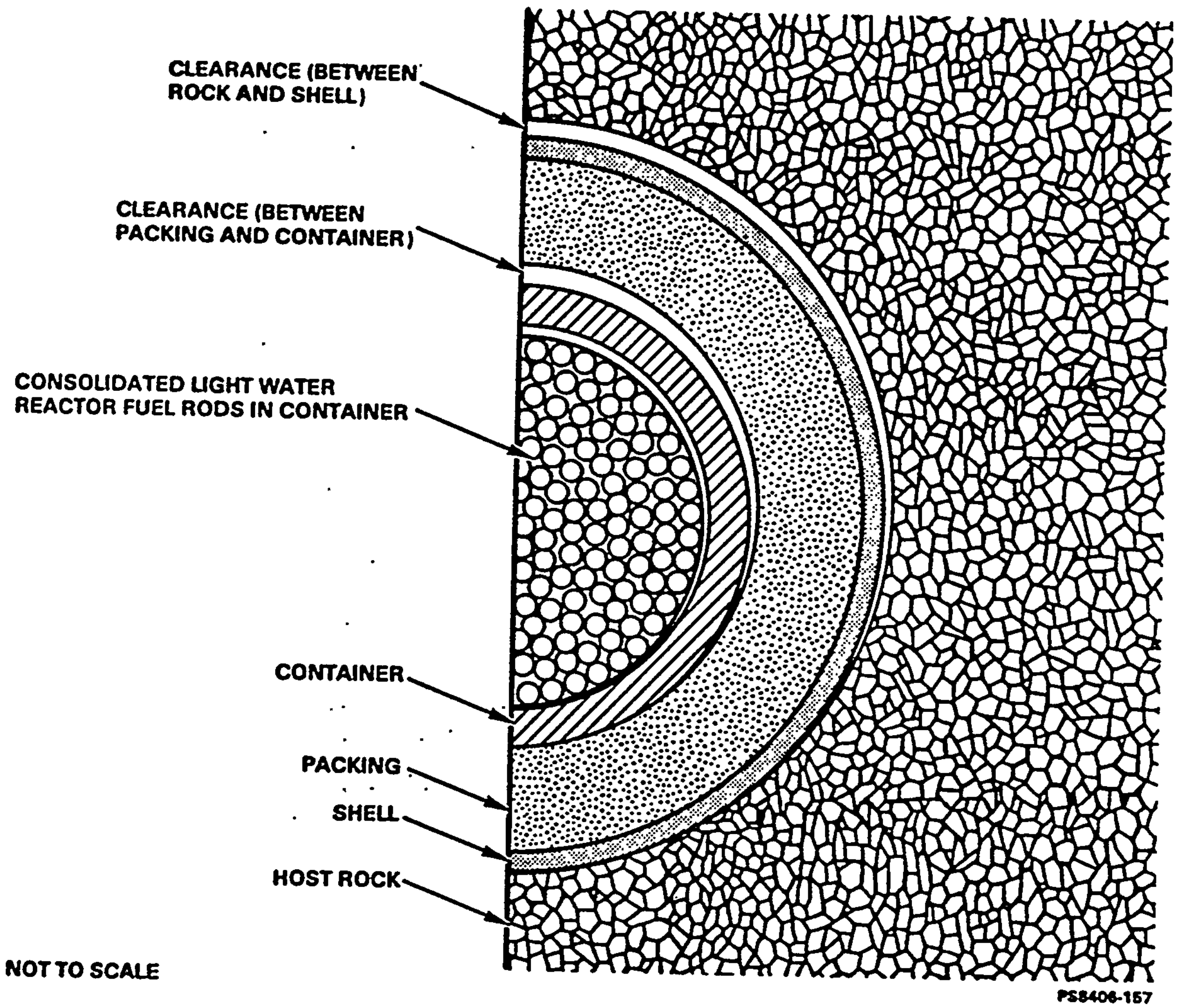

Figure 9.1.15. Typical Cross Section of Emplacement Borehole

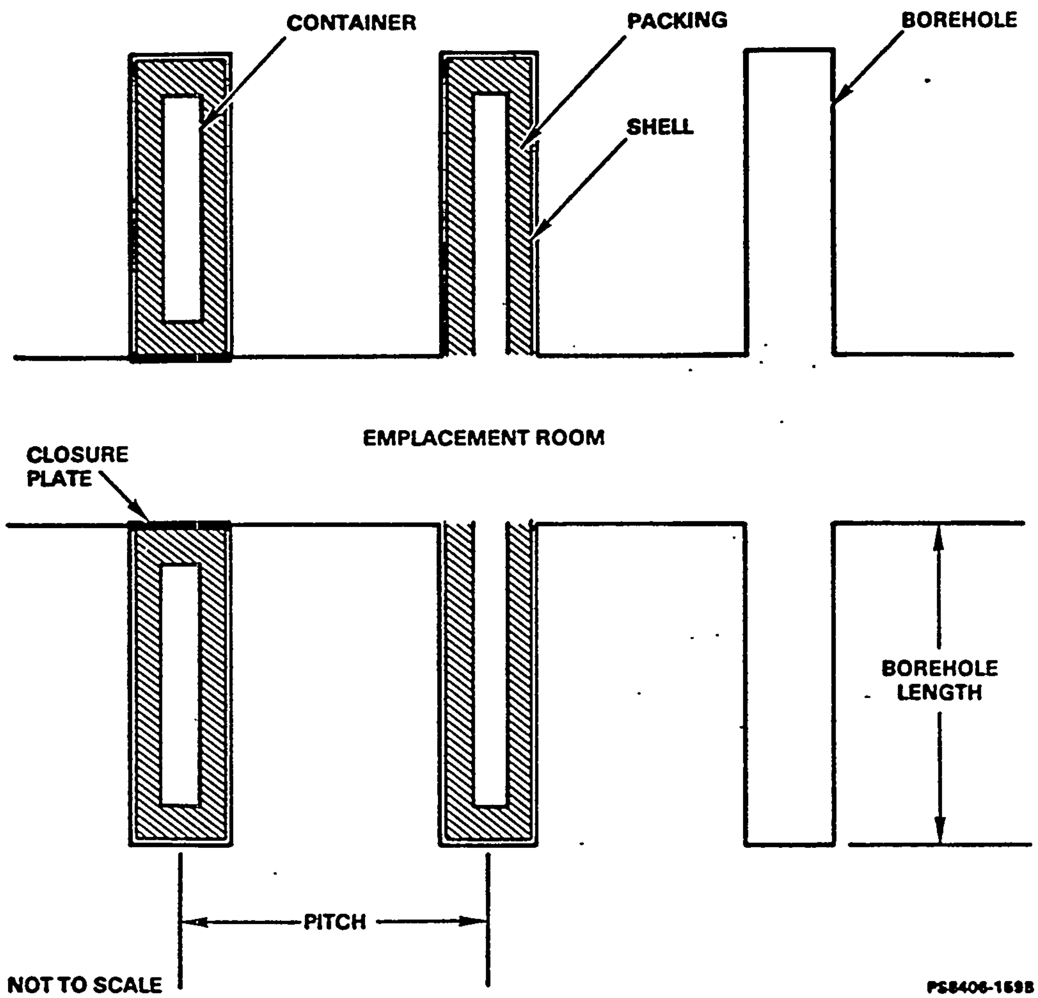

Figure 9.1.16. Typical Plan View for Waste Emplacement

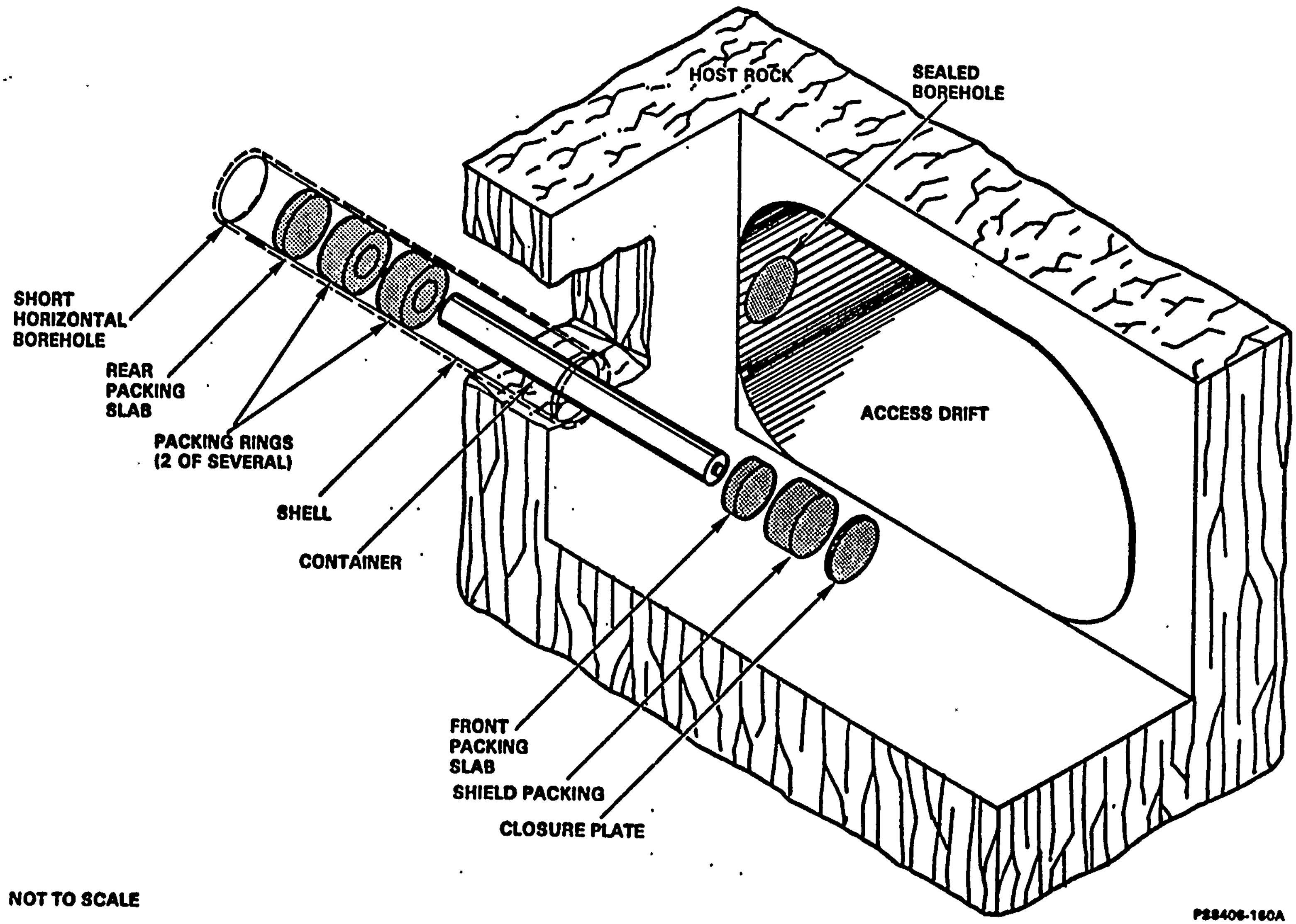

Figure 9.1.17. Waste Emplacement Sequence Concept

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In shafts and boreholes drilled from the surface, low-permeability materials other than backfill may be provided as redundant barriers to ground-water flow. Concrete materials, metals, and vitrified basalt are being considered for such barriers. Where shaft seals are placed in shafts, shaft liners and grout will be removed from areas of intact rock prior to seal emplacement.

9.1.7 Summary

9.1.7.1 Advantages

1. Shaft drilling is a safer shaft construction method, though limited in depth and/or diameter in hard rock.

2. Horizontal emplacement limits the vertical extent of the repository.

9.1.7.2 Potential Problems

1. Shaft drilling does not afford the capability to inspect the geologic conditions as conventional shaft sinking would.

2. The presence of overlying aquifers can be a potential problem.

3. Horizontal emplacement requires development of special emplacement techniques.

10.0 TUFF FORMATIONS

10.1 TUFF: THE UNITED STATES CONCEPT

This chapter describes the tuff reference repository concept of the nuclear waste disposal program in the United States. Since 1957, when the National Academy of Sciences concluded that geologic disposal would be the most desirable method for permanent disposal of nuclear wastes, many alternatives have been considered. Studies have been conducted in different geologic media and large amounts of data collected, analyzed and compared. The passage of the Nuclear Waste Policy Act of 1982 provided a framework for achieving the disposal of the nation's spent nuclear fuel and high-level wastes. The Act authorized one repository with a capacity of 70,000 MT (77,000 tons) of wastes, and proposed a second repository, should the need arise. The repositories will be used for disposal of commercial high-level wastes or spent fuel produced by nuclear power generation facilities as well as the small amount of defense high-level wastes. It also requires the preparation of proposals for a Monitored Retrievable Storage facility and established a funding mechanism for the waste disposal program. Several regions in the United States were under investigation and in February 1983, the Department of Energy fulfilled the first requirement of the Act by formally identifying nine sites as potentially suitable repository locations in three different geologic media as shown in Figure 8.4.1:

1. Vacherie Dome, Louisiana (Domal Salt)
2. Cypress Creek Dome, Mississippi (Domal Salt)
3. Richton Dome, Mississippi (Domal Salt)
4. Yucca Mountain, Nevada (Welded Tuff)
5. Deaf Smith County, Texas (Bedded Salt)
6. Swisher County, Texas (Bedded Salt)
7. Davis Canyon, Utah (Bedded Salt)
8. Lavender Canyon, Utah (Bedded Salt)
9. Reference Repository Location, Hanford Site, Washington (Basalt Flows).

On May 28, 1986, the Department of Energy announced presidential approval of the nomination and recommendation for site characterization of five of the nine sites in Washington, Texas and Nevada. Environmental assessments were published at the same time for the five recommended sites. These sites are Richton, Deaf Smith County and Davis Canyon in salt, Yucca Mountain in tuff, and Hanford in basalt. Approval was granted for the construction of exploratory shaft facilities at three of these sites:

1. Deaf Smith County, Texas

2. Yucca Mountain, Nevada

3. Hanford, Washington.

Designs for these exploratory shaft facilities are currently in preparation and construction is scheduled to commence in the 1988-89 time frame. During the site characterization activities at each site, the Department of Energy will consult closely with the Nuclear Regulatory Commission, responsible for licencing the repository, and affected states and Indian tribes. After several years of site characterization, the Department of Energy will be able to recommend one of the sites to the president as the location for the first repository. After approval by Congress, and state and local affected groups, a Construction Authorization Application will be submitted to the Nuclear Regulatory Agency. The agency will have three years to review the application and resolve any issues with the Department of Energy. Subject to approval by the Nuclear Regulatory Agency, repository construction is planned to start in the mid-1990s to meet the 1998 date of start of actual waste disposal as scheduled by the Nuclear Waste Policy Act.

Work is also being carried out on the second repository program. Crystalline rocks were investigated as a possible location for this repository. An Area Recommendation Report was published in 1986 and current activities include resolution of comments received on the report and technology development activities.

The 70,000 MT (77,000 tons) of waste represents the maximum storage capacity of the first repository and is made up of several types of nuclear waste. Commercial reprocessing is currently not practiced, and thus the reference repository described in this chapter is based on the disposal of about 62,000 MT (68,000 tons) of commercial spent fuel and about 8,000 MT (9,000 tons) equivalent of reprocessed wastes. These reprocessing wastes are mainly defense high level wastes (90%) and some reprocessing wastes (10%) remaining at the shut-down West Valley commercial reprocessing plant.

This chapter describes the reference repository for the Yucca Mountain site at the Nevada Research and Development Area in Nevada, published in the Environmental Assessment in 1986.

10.1.1 Geographic Location

The Yucca Mountain site (Figure 10.1.1) is located on and immediately adjacent to the southwestern portion of the Nevada Test Site, which is in Nye County, Nevada, about 105 km (65 mi) north west of Las Vegas. The site lies within the Basin and Range physiographic province, a broad region

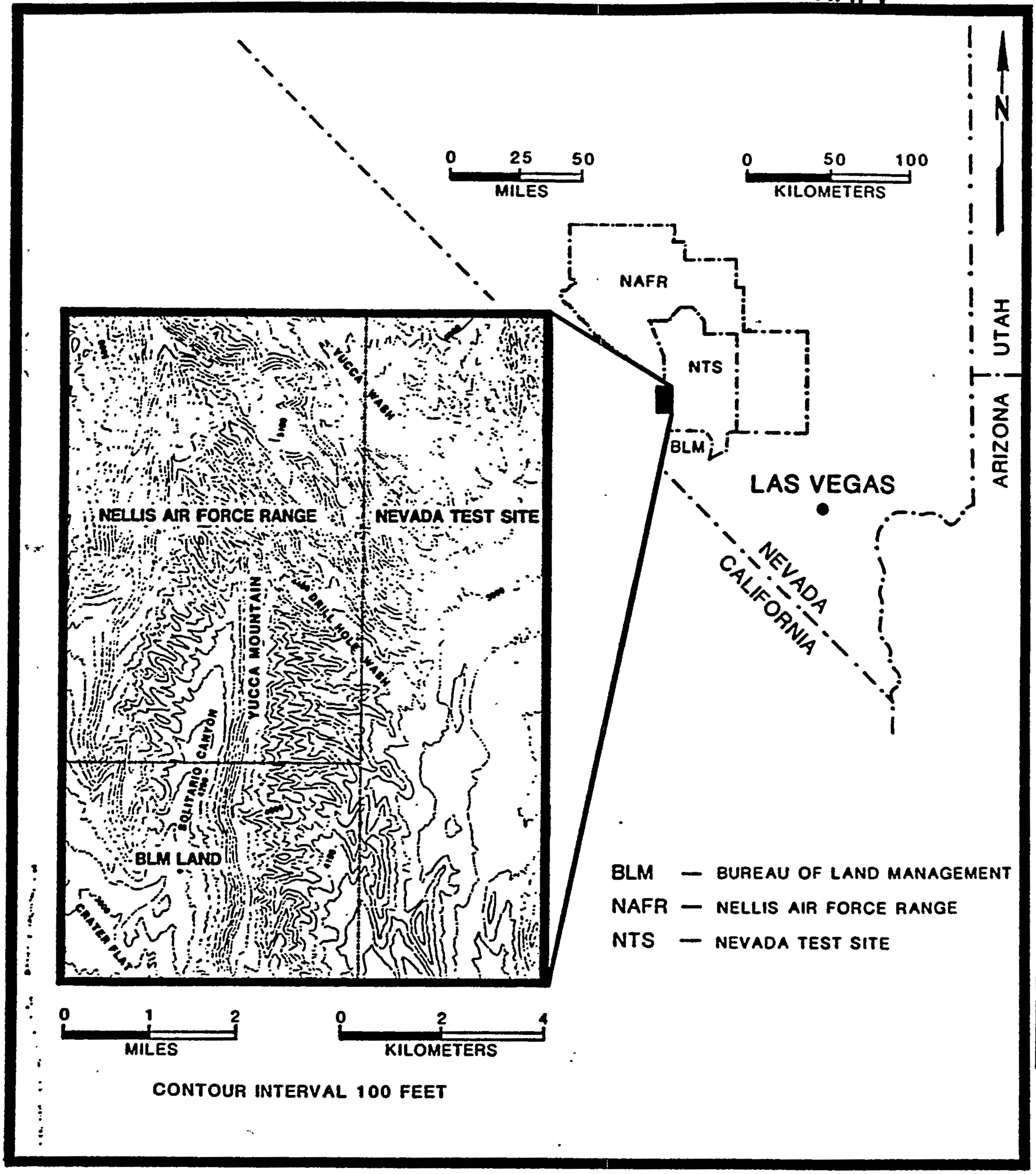

Figure 10.1.1. Location of Yucca Mountain Site in Nevada

of generally linear mountain ranges and intervening valleys. Figure 10.1.2 shows the physiographic features in the region. The Yucca Mountain site is located exclusively with lands controlled by the Federal Government.

10.1.2 Summary of Geologic Setting

Yucca Mountain is in the southern part of the Great Basin, a part of the Basin and Range Physiographic Province in which all surface waters drain into closed basins rather than flowing into the ocean. As shown in Figure 10.1.3, the rocks in this province can be divided into four groups in order of decreasing geologic age:

- Precambrian crystalline basement rocks
- Upper Precambrian and Paleozoic sedimentary rocks that have been folded, faulted, and uplifted to form large mountain ranges that eventually eroded to a gentle plain
- Tertiary tuffaceous volcanic material such as that which forms Yucca Mountain
- Alluvium derived from the erosion of the surrounding mountains.

The tuffaceous rocks occur in layers at least 2,000 m (6,500 ft) thick. These rocks are composed chiefly of rhyolitic ash-flow tuffs, with smaller amounts of dactitic lava flows and flow breccias and minor amounts of tuffaceous sedimentary rocks and air-fall tuffs.

Faulting and volcanism that produced the early features of the Basin and Range Province took place concurrently approximately 10 to 40 million years ago. In the vicinity of Yucca Mountain, tectonic activity has steadily decreased over the last 10 million years. Minor volcanic activity has continued during basin filling and, most recently, produced thin, areally restricted flows and cones of basaltic material on Crater Flat, west of Yucca Mountain. Yucca Mountain and areas to the west and south have had a relatively low level of seismic activity throughout the historical record.

At Yucca Mountain, a repository would be constructed in the unsaturated zone 200 to 400 m (650 to 1,300 ft) above the water table. The movement of ground water in the saturated zone is typified by a very lowflux of water moving downward primarily through the intergranular pores of the tuff layers. In the saturated zone below, water moves laterally through fractures and pores in both the tuffs and in the underlying carbonate-rock aquifers.

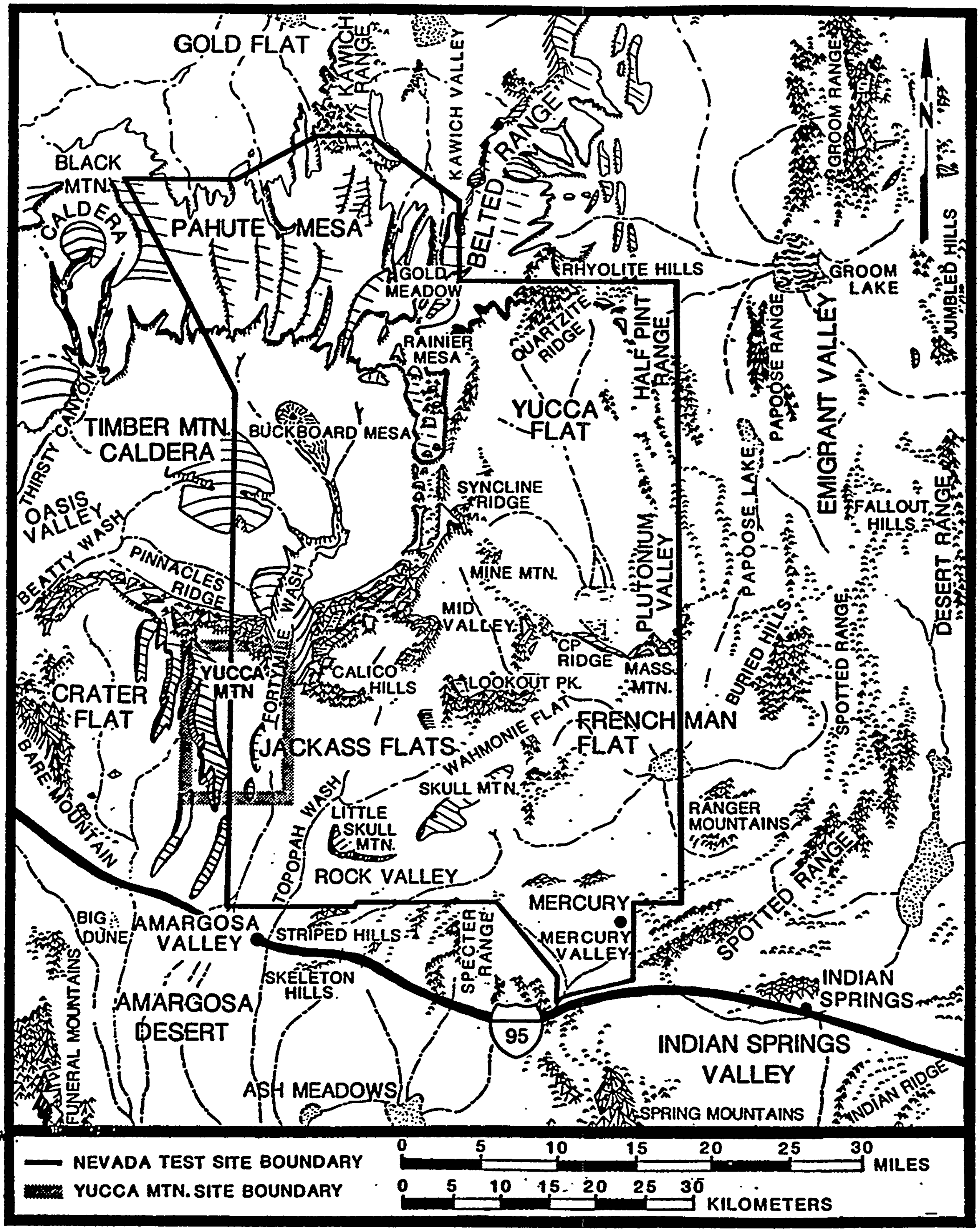

Figure 10.1.2. Physiographic Features of Yucca Mountain

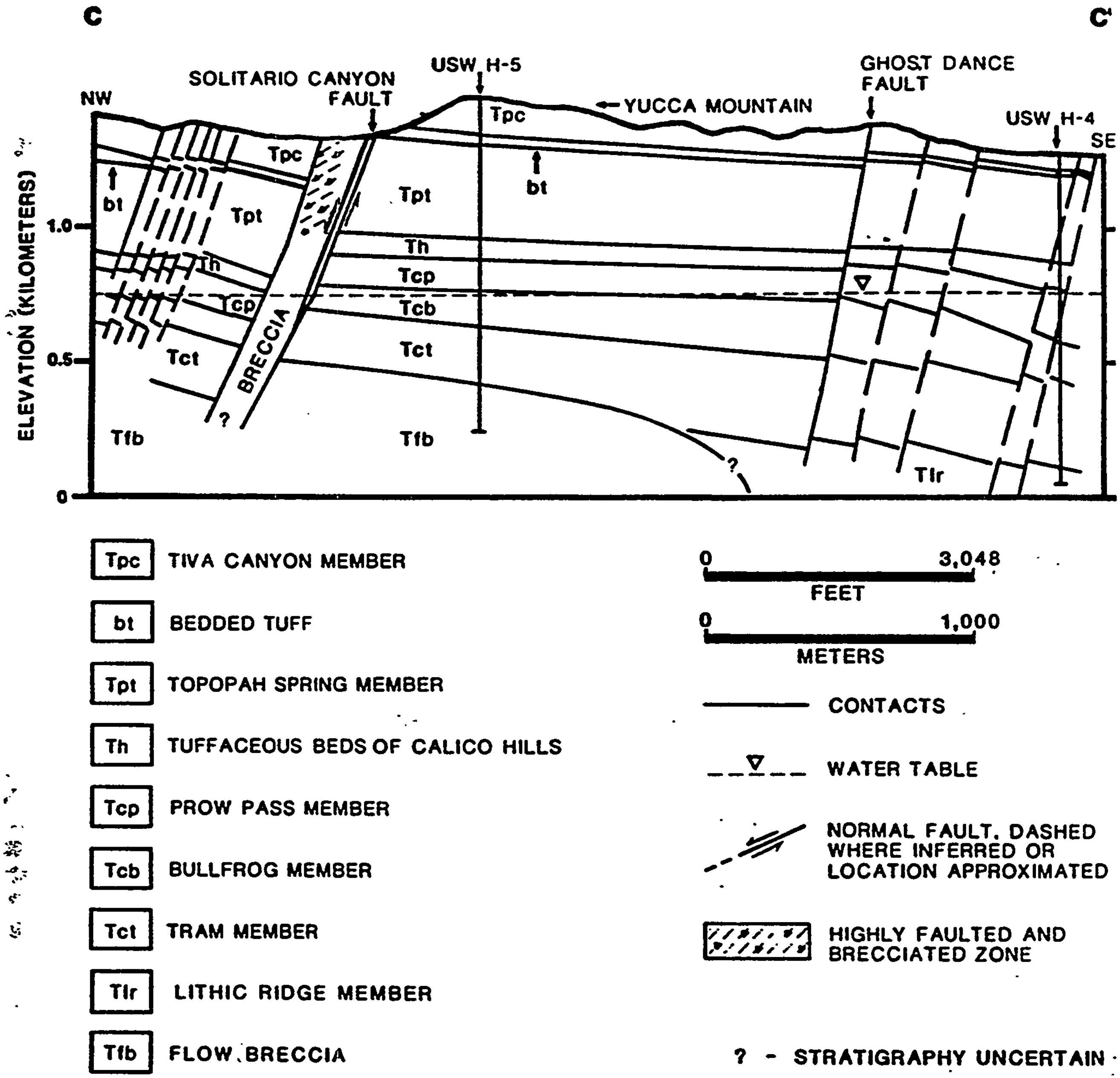

Figure 10.1.3. Geologic Section of Yucca Mountain Site

10.1.3 Repository Concept Description

10.1.3.1 Nuclear Waste Disposal Strategy

The United States high-level waste disposal strategy was outlined in Section 8.4.3.1.

10.1.3.2 Repository Concept Derivation

Some of the more important features of the repository in tuff are illustrated in Figure 10.1.4. Although it is an artist's rendition of the two-stage repository design concept, it serves as a guide to the following discussion of the evolution of the Yucca Mountain repository design. The conceptual design of the prospective repository consists of a surface facility, an underground facility, and a means of access from one to the other. Figure 10.1.4 shows ramps as the means of access from the surface to the underground repository where mined access drifts connect with other mined drifts in which the waste is emplaced. The waste would be emplaced in holes drilled either horizontally into the walls of the emplacement drifts or vertically into the floors.

Three different design concepts can be identified in the continuing evolution of the repository design. The first was the reference repository design described in the preliminary design report by Jackson (1984). This concept was summarized in Section 5.1 of the December 1984, draft Environmental Assessment for Yucca Mountain. The second is the two-stage repository design concept (MacDougall, 1985). This design has evolved through minor changes to a concept called the current design concept that is described in the Mission Plan. The characteristics of and expected differences in the three design concepts are summarized in Table 10.1.1. The most important differences among these concepts are the proposed waste inventory and the staging of construction and waste receipt activities. The reference design concept was a single-stage facility designed to accept a waste inventory of 35,000 MT (38,500 tons) of spent fuel and 35,000 MT (38,500 tons) equivalent of commercial high-level waste and reprocessed West Valley waste. In the two-stage repository concept, the repository would accept only spent fuel (70,000 MT or 77,000 tons) and would be constructed in two phases and operated in two stages. In the current design concept, the repository would receive 62,000 MT (68,000 tons) of spent fuel and 8,000 MT (9,000 tons) equivalent of defense high-level waste (including commercial high-level waste from the West Valley Demonstration Project); it would be constructed in two stages; and it would be able to receive spent fuel as early as five years out of the reactor.

The two-stage repository design is the design for which the most complete data are available. This design integrates preliminary repository concepts embodied in the reference repository design concept with recent changes and additions as described in the "Generic Requirements for a Mined

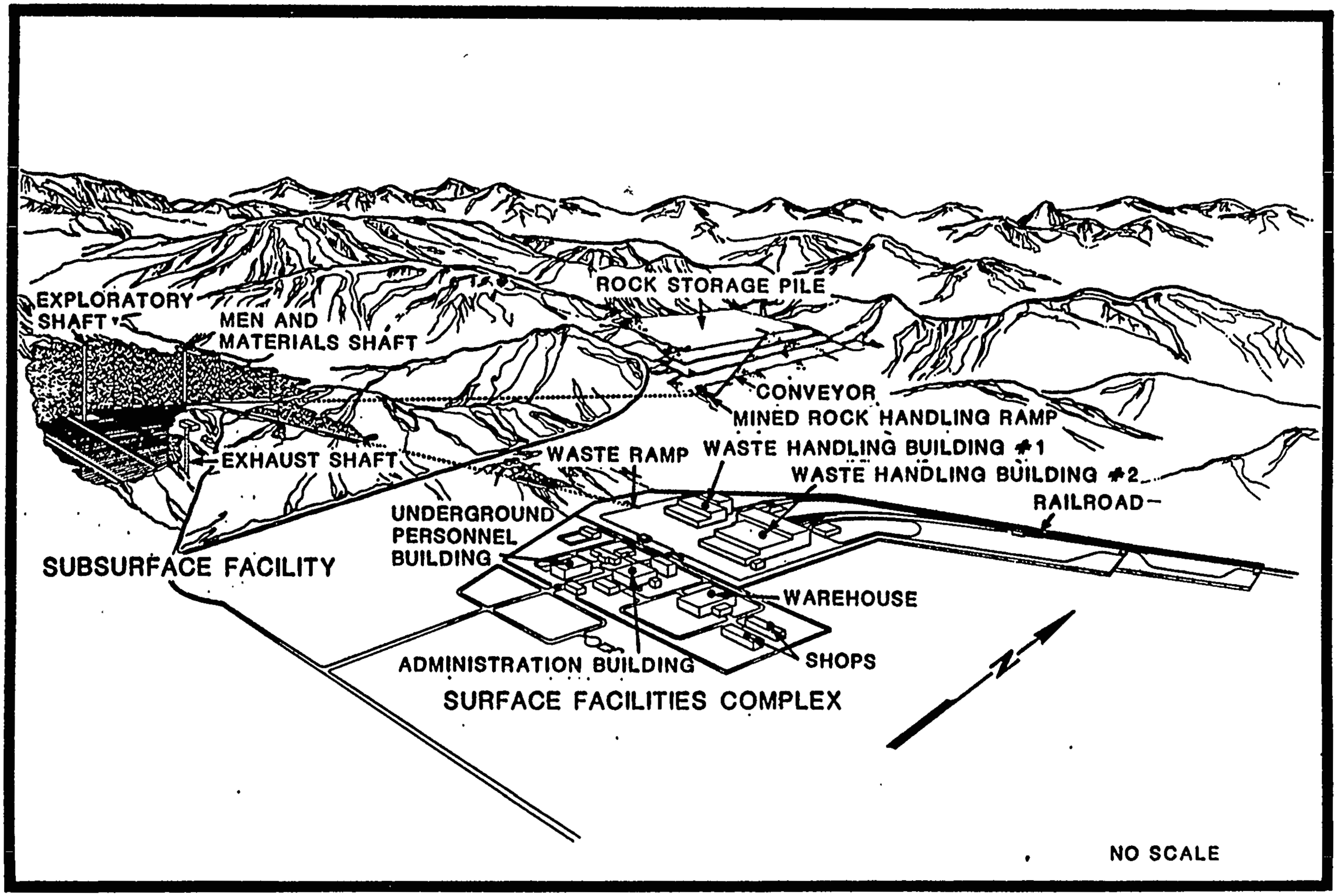

Figure 10.1.4. Artist's Rendition of Yucca Mountain Site

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REPOSITORY CHARACTERISTIC[b]	REFERENCE DESIGN (DRAFT EA) Vertical	REFERENCE DESIGN (DRAFT EA) Horizontal	TWO-STACK DESIGN (FINAL EA) Vertical	TWO-STACK DESIGN (FINAL EA) Horizontal	CURRENT DESIGN CONCEPTS (MISSION PLAN) Vertical	CURRENT DESIGN CONCEPTS (MISSION PLAN) Horizontal	CHANGES IN IMPACTS[a] Socio-economic	CHANGES IN IMPACTS[a] Environ-mental	CHANGES IN IMPACTS[a] Trans-portation
INCORPORATES EXPLORATORY SHAFTS?	YES	YES	YES	YES	YES	YES	NSD	NSD	NSD
SUBSURFACE ACCESS									
Ramps - Waste and air intake	15-ft x 20-ft	15-ft x 20-ft	24-ft dia.	24-ft dia.	21-ft dia.	19-ft dia.			
- Muck and mine exhaust	15-ft x 20-ft	15-ft x 20-ft	19-ft dia.	19-ft dia.	24-ft dia.	20-ft dia.			
Shafts - Men and material	20-ft dia.	14-ft dia.	25-ft dia.	25-ft dia.	20-ft dia.	20-ft dia.	NSD	NSD	NSD
- Repository exhaust	16-ft dia.	14-ft dia.	20-ft dia.	20-ft dia.	20-ft dia.	20-ft dia.			
- Supply	16-ft dia.	10-ft dia.	12-ft dia. ES	12-ft dia. ES	12-ft dia. ES	12-ft dia. ES			
- Supply	12-ft dia. ES	12-ft dia. ES	6-ft dia. ES	6-ft dia. ES	6-ft dia. ES	6-ft dia. ES			
Excavated rock - tons	20,000,000	2,200,000	21,600,000	6,580,000	20,700,000	4,630,000		(c)	
Total area - Main surface complex	75 acres	75 acres	150 acres	150 acres	150 acres	150 acres	NSD	NSD	NSD
- Subsurface	1520 acres	1520 acres	1520 acres	1520 acres	1520 acres	1520 acres			
Preclosure period[d] - Construction	1993-1998	1993-1998	1993-2000	1993-2000	1993-2000	1993-2000			
- Operation[e]	1998-2047	1998-2047	1998-2047	1998-2047	1998-2047	1998-2047	NSD	NSD	NSD
- Decommission	2048-2052	2048-2052	2048-2055	2048-2050	2048-2055	2048-2050			
Total capacity	70,000 MTU	70,000 MTU	70,000 MTU	70,000 MTU	70,000 MTU	70,000 MTU	NSD	NSD	NSD
Annual receipt rate[f] - MTU	Yr 1-23 3,000	Yr 1-23 3,000	Yr 1-3 400	Yr 1-3 400	Yr 1-3 400	Yr 1-3 400			
	Year 24 1,000	Year 24 1,000	Year 4 900	Year 4 900	Year 4 900	Year 4 900			
			Year 5 1,800	Year 5 1,800	Year 5 1,800[h]	Year 5 1,800[h]	(g)	(g)	(g)
			Yr 6-27 3,000	Yr 6-27 3,000	Yr 6-24 3,400[h]	Yr 6-24 3,400[h]			
			Year 28 100	Year 28 100	Year 25 1,500	Year 25 1,500			
Waste inventory - spent fuel	35,000 MTU	35,000 MTU	70,000 MTU	70,000 MTU	62,000 MTU	62,000 MTU			
- CHLW	35,000 MTU	35,000 MTU					NSD	NSD	NSD
- TRU	20,000 Pkgs	20,000 Pkgs							
- DHLW					8,000 MTU[i]	8,000 MTU[i]			

Table 10.1.1. Comparison of Alternate Repository Design Concepts
(Page 1 of 2)

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REPOSITORY CHARACTERISTIC	REFERENCE DESIGN (DRAFT EA) Vertical	Horizontal	TWO-STAGE DESIGN (FINAL EA) Vertical	Horizontal	CURRENT DESIGN CONCEPTS (MISSION PLAN) Vertical	Horizontal	CHANGES IN IMPACTS[a] Socio-economic	Environ-mental	Trans-portation
Waste handling buildings	One	One	Two	Two	Two	Two	NSD	NSD	NSD
Peak annual usage									
Water - gallons per year	58,600,000	58,600,000	120,000,000	120,000,000	120,000,000	120,000,000			
Electrical - kWh per year	137,000,000	82,000,000	115,000,000	83,000,000	115,000,000	83,000,000	NSD	NSD	NSD
Diesel - gallons per year	1,660,000	946,000	5,500,000	5,500,000	5,500,000	5,500,000			
Peak annual number of direct workers									
Construction period	3,348	2,800	1,905[j]	1,651[j]	1,905[j]	1,651[i]			
Operation - Emplacement phase	2,313	1,442	1,905[j]	1,651[j]	1,905[j]	1,651[i]	NSD	NSD	NSD
- Caretaker phase	594	453	162	146	162	146			
Decommissioning period	1,548	653	412	441	412	441			
Access improvements - Highway	16 miles	16 miles	16 miles	16 miles	16 miles	16 miles			
- Railroad	85 miles	85 miles	100 miles	100 miles	100 miles	100 miles	NSD	NSD	NSD
Total construction materials									
Concrete - cubic yards	554,400	264,700	547,300	266,700	547,300	266,700			
Structural steel - tons	26,100	19,700	201,930	80,940	201,930	80,940	NSD	NSD	NSD
Number of stages	One	One	Two	Two	Two	Two	NSD	NSD	NSD
Fuel consolidation?	Yes	Yes	Stage Two	Stage Two	Stage Two	Stage Two			

[a] Change noted for the difference between the two-stage design and the current design. NSD = No substantial difference.
[b] MTU = metric tons uranium; CHLW = commercial high-level waste; TRU = transuranic waste; DHLW = defense high-level waste.
[c] Less excavation and surface area disturbed will result in less habitat destroyed and more fugitive emissions.
[d] Except for September 1993 (start of construction), the dates indicated above are from January thru December of the year listed.
[e] "Operation" is defined to include the emplacement and the caretaker or retrievability phases.
[f] Year 1 = 1998, i.e. 1st year of waste receipt.
[g] Increasing the number of shipments increases the traffic impacts.
[h] 3,400 MTU includes 3,000 MTU spent fuel plus 400 MTU high-level waste (including DHLW and West Valley high-level waste).
[i] Includes DHLW and West Valley high-level waste.
[j] Construction and operation periods overlap in the year of maximum direct employment. See tables 5-5a and 5-5b.

Table 10.1.1. Comparison of Alternate Repository Design Concepts (Page 2 of 2)

Geologic Disposal System." This document stipulates the following design requirements:

- The quantity of waste emplaced in the repository may not exceed 70,000 MT (77,000 tons) as spent fuel, or its equivalent in high-level waste, until a second repository is in operation. Although the waste form most likely to be received for disposal is spent fuel, the design will not preclude the capability to receive, handle, and dispose of reprocessed commercial high-level waste and defense high-level waste.

- The repository will be designed to permit the initiation of waste retrieval operations at any time during the waste emplacement phase and up to 50 years after emplacement operations have begun, for recovery of any or all of the waste.

- The receipt rate during the first five years will increase from an initial rate of 400 MT (440 tons) per year to 1,800 MT (1,980 tons) per year. For the remainder of the emplacement phase it will be 3,000 MT (3,300 tons) per year.

- A surface facility with a surge storage capacity for accommodating the equivalent of a three-month accumulation of waste receipts will be provided (i.e., 100 MT (110 tons) equivalent for Stage 1 operation and up to 750 MT (825 tons) equivalent for Stage 2 operation). This capability will help to minimize the impact of scheduled or unscheduled interruptions in repository operations or the off site transportation system and waste shippers. The storage facility will be capable of accommodating both the waste receipts from off site sources and the waste packages prepared on the site.

Under the current design concept the repository would receive defense high-level waste at a rate of 400 MT (440 tons) equivalent per year beginning in 2003, the sixth year of operation. The waste would be in the form of borosilicate glass contained in waste disposal containers approximately 0.6 m (2 feet) in diameter, 3 m (10 feet) high, and weighing about 1.8 metric tons (4,000 pounds). Shipment may be by either truck or rail. If shipment were by truck, this design would result in approximately three shipments per day for defense waste or 800 waste disposal containers per year. In either the two-stage repository concept or the current design concept, the Stage 1 waste handling building, designed to receive up to 400 MT (440 tons) per year, would no longer be used to receive spent fuel after the year 2002 when the Stage 2 facility becomes fully operational. In the current design concept, the Stage 1 facility could then be used for the receipt and handling of defense waste beginning in the year 2003. Since the defense waste has lower thermal radiation levels than spent fuel, the Stage 1 facility would be totally suitable to perform this function.

10.1.3.3 Surface Facilities

The actual location of the surface facilities has not yet been determined. However, a candidate location has been identified for the purpose of preparing the Environmental Assessments. The candidate location for these facilities is along the gently sloping east side of Yucca Mountain, as shown on Figure 10.1.5. The surface facilities complex proposed at Yucca Mountain would encompass approximately 60 hectares (150 acres) of land, all of which would be enclosed by a security fence.

A preliminary site plan of the proposed surface facilities at Yucca Mountain is shown in Figure 10.1.6. The surface facilities complex will be used for waste handling and packaging operations in support of the underground activities and to provide general repository support services. The restricted-access area for waste handling and packaging facilities will include buildings and equipment for receiving and packaging all incoming wastes. A facility would also be constructed for processing all the radioactive waste generated by onsite operations, such as protective clothing, decontamination fluids, and ventilation filters.

Support facilities for the repository will include offices for administrative, management, and engineering staff; a firehouse; medical, training and computer centers; a vehicle maintenance and repair shop; security buildings; a machine and sheet metal shop; and an electrical shop. Warehouses will be constructed to store bulk materials, equipment, spare parts, and supplies.

Facilities for environmental and instrument laboratories will also be constructed. Surface facilities in support of the underground repository operations include personnel change rooms and showers, as well as space to store mining equipment and vehicles.

Power lines will be extended to the site from existing local utilities on the Nevada Test Site and a new substation constructed at the site. Other utilities that may be installed to support the repository included steam generating equipment, emergency electrical generators, compressor and chiller systems, cooling towers with water treatment (if required), water treatment system, and fuel storage.

Construction away from the main surface facilities complex will consist primarily of an access route connecting with the main highway, a rail line, a bridge, the mined rock handling and storage facilities, and ventilation facilities above each exhaust shaft.

A Monitored Retrievable Storage facility, located at some strategic location in the United States, may also be integrated into the nuclear waste disposal cycle. A description of the Monitored Retrievable Storage functions and concepts was given in Section 8.4.3.2 and will not be repeated here.

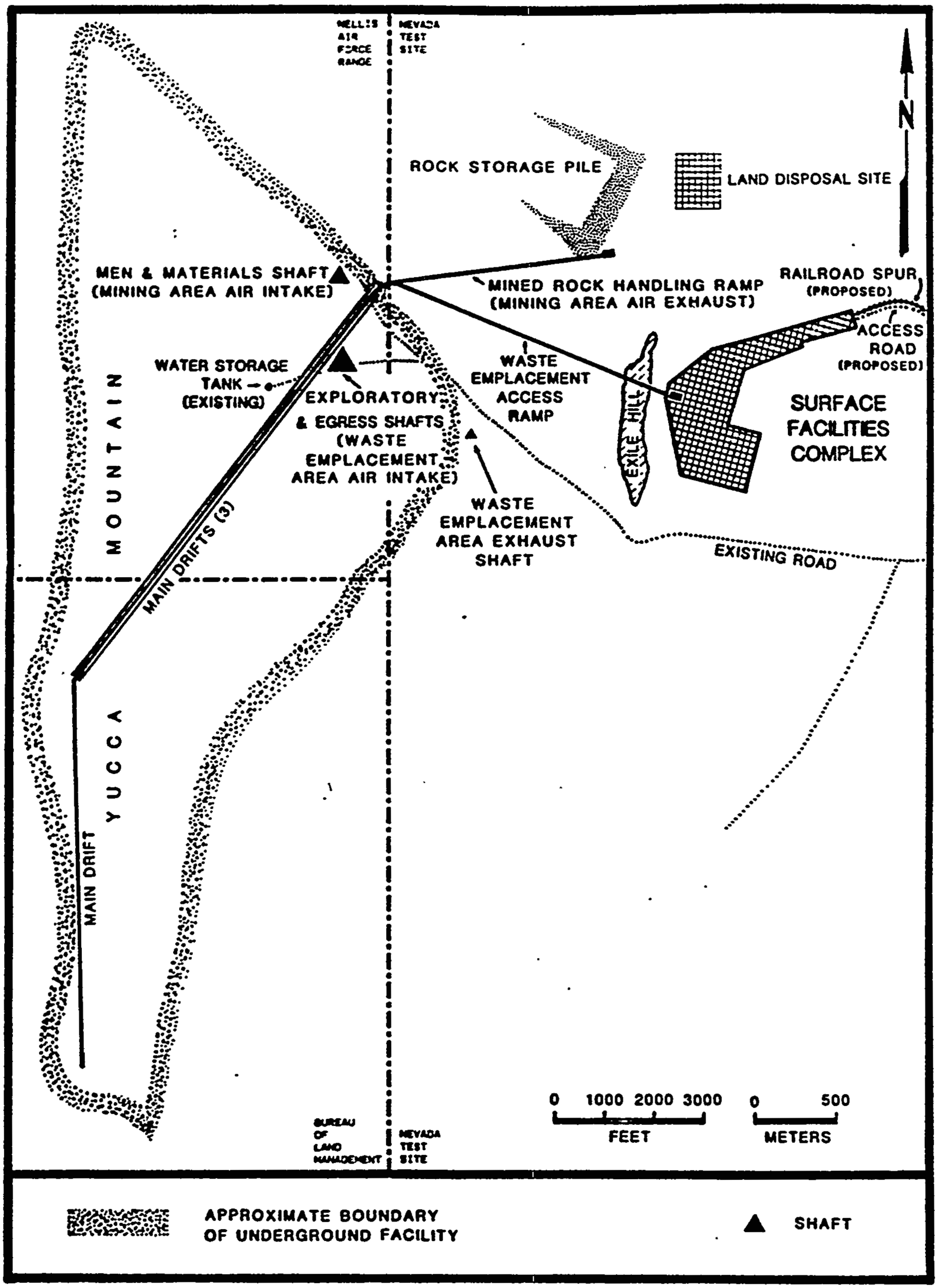

Figure 10.1.5. Two-Stage Repository Site Plan

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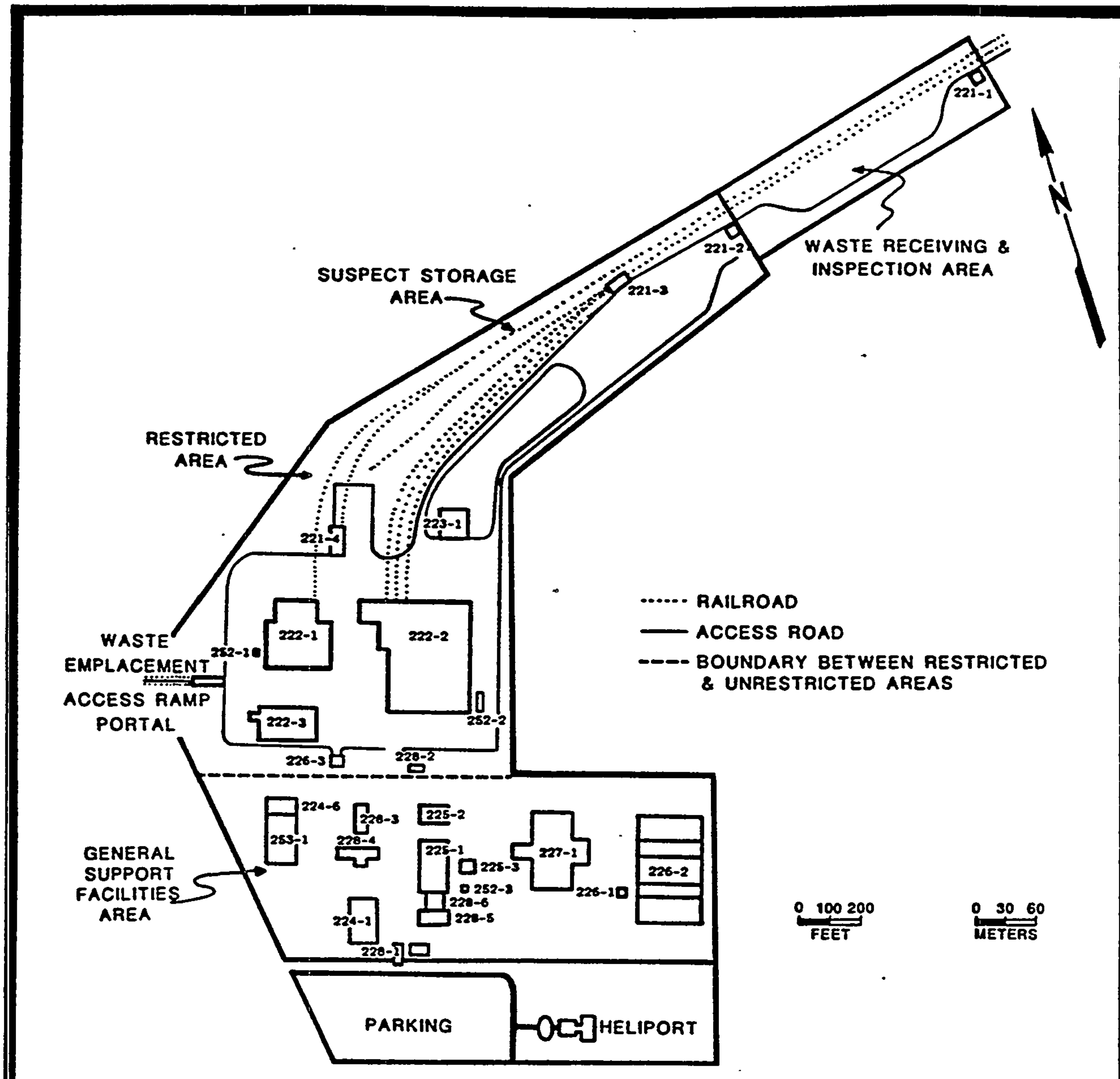

221-1 SECURITY
221-2 HEALTH PHYSICS
221-3 WASHDOWN FACILITY
221-4 DECONTAMINATION BUILDING
222-1 WASTE HANDLING BUILDING NO. 1
222-2 WASTE HANDLING BUILDING NO. 2
222-3 PERFORMANCE CONFIRMATION FACILITY
223-1 WASTE TREATMENT BUILDING
224-1 UNDERGROUND PERSONNEL FACILITY
224-6 MAIN SUBSTATION & STANDBY GENERATOR BUILDING
225-1 ADMINISTRATION BUILDING
225-2 MOCKUP BUILDING
225-3 COMPUTER BUILDING
226-1 MOTOR POOL & FACILITY SERVICE STATION
226-2 SHOPS
226-3 RESTRICTED AREA GARAGE
227-1 CENTRAL WAREHOUSE
228-1 SECURITY STATION NO. 1
228-2 SECURITY STATION NO. 2/ HEALTH PHYSICS STATION NO. 3
228-3 FIRE STATION
228-4 MEDICAL CENTER & DORMITORY
228-5 FOOD SERVICE FACILITY
228-6 TRAINING AUDITORIUM
252-1 WATER CHILLERS NO. 1
252-2 WATER CHILLERS NO. 2
252-3 WATER CHILLERS NO. 3
253-1 UTILITY ELECTRIC SUBSTATION

Figure 1Ø.1.6. Preliminary Surface Site Plan

10.1.3.4 Underground Configuration

Six access excavations will connect the surface to the underground areas. These excavations, used for ventilation air intake and exhaust, the transport of materials, are as follows:

- The men-and-material shaft will be used to transport personnel and materials to and from the underground facilities. This shaft will be 7.6 m (25 ft) in diameter and approximately 335 m (1,110 ft) deep.

- The waste handling ramp or decline will be used to transport waste underground. This ramp will be 7.4 m (24 ft) in diameter and approximately 2,042 m (6,700 ft) long.

- The mined-material handling ramp will be used for the mined-material conveyor system and as an exhaust outlet for construction area ventilation. The ramp will be 5.8 m (19 ft) in diameter and approximately 1,417 m (4,650 ft) long.

- The waste emplacement area exhaust shaft will serve as the exhaust outlet for ventilation during waste emplacement. The 6.1 m (20 ft) diameter shaft will be about 304 m (1,000 ft) deep.

- The 3.7 m (12 ft) diameter exploratory shaft, constructed during site characterization, will be used to supply air for repository waste emplacement operations. It will be approximately 450 m (1,480 ft) deep.

- The 1.8 m (6 ft) diameter emergency egress shaft of the exploratory shaft test facility will be used to supply air to the repository waste emplacement support facilities. This shaft will be approximately 365 m (1,200 ft) deep.

The underground facilities will be located within Yucca Mountain approximately 1.7 km (1 mi) west of the proposed location of the surface facilities complex (Figure 10.1.5). This facility will encompass about 615 hectares (1,520 acres) of underground area. The repository horizon will be more than 230 m (750 ft) below the surface within the Topopah Spring Member of the Paintbrush Tuff. (The exploratory shaft facility is planned to be excavated at a depth of about 370 m or 1,210 ft). The water table in the vicinity of Yucca Mountain is approximately 200 to 400 m (650 to 1,300 ft) below the potential repository horizon. Except for possible scattered pockets of perched water, the underground openings are expected to be dry. An artist's rendition of the proposed surface facilities is shown in Figure 10.1.7.

The subsurface facilities consist of main access drifts to the emplacement areas, the emplacement drifts, and service areas near the shafts and ramps. The layout of the facilities depends upon whether the waste is emplaced vertically or horizontally. For vertical emplacement, waste

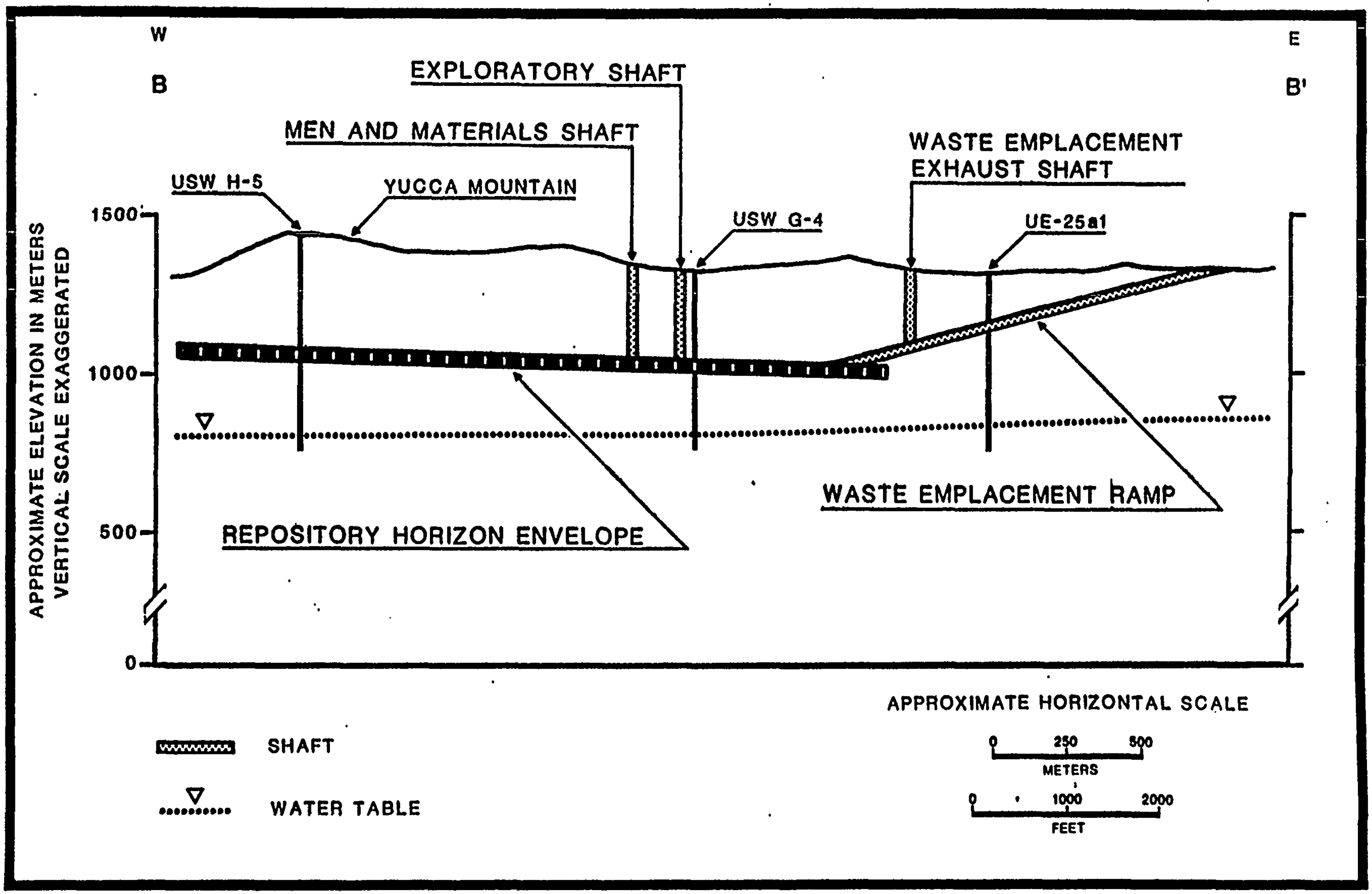

Figure 10.1.7. Artist's Rendition of Underground Layout

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disposal containers would be emplaced in vertical boreholes in the floors of the emplacement drifts. An extraction ratio of 24 percent has been adopted for the vertical emplacement alternative. Cross-sectional dimensions of these openings are listed in Table 10.1.2. The total amount of rock excavated for the facility would be about 21.6 million tons.

For horizontal emplacement, waste disposal containers would be emplaced in horizontal boreholes in the drift pillars (walls). The underground layout for horizontal waste-emplacement requires considerably less excavation. The total amount of rock excavated for the facility would be about 6.6 million tons. Table 10.1.2 lists the dimensions of the openings for horizontal waste emplacement.

Table 10.1.2. Dimensions of Underground Excavations for Vertical and Horizontal Waste Emplacement.(1)

Excavation	Vertical Emplacement		Horizontal Emplacement	
	Height m (ft)	Width m (ft)	Height m (ft)	Width m (ft)
Access corridors	4.6 (15)	6.4 (21)	4.6 (15)	6.4 (21)
Emplacement drifts	6.4 (21)	4.6 (15)	4.6 (15)	6.4 (21)

(1) Data from MacDougall (1985).

Conventional mining equipment, as well as machinery designed specifically to transport wastes to the emplacement locations, would be required underground. The service areas required underground include medical facilities, warehouses, personnel change rooms, and maintenance areas.

The excavated rock would be placed near the site in a hypalon-lined rock storage pile (Figure 10.1.5). The rock-storage pile would be constructed on the surface using conventional mined-rock handling equipment and would be sprayed with water to suppress dust. Runoff from precipitation would be intercepted by dikes, ditches, and liquid-collection sumps. The present drifts to maintain the structural integrity of the underground openings. If backfilling of a portion of the repository is required before closure and decommissioning, some of the excavated rock would be used for that purpose.

10.1.4 Repository Construction

10.1.4.1 Schedule

The lifetime of the repository at Yucca Mountain, before it is permanently closed, may be divided into several periods: construction, operations, and decommissioning. These periods are illustrated in Figure 10.1.8 and 10.1.9 for the two emplacement options under consideration. All of Stage 1 and part of Stage 2 facilities will be constructed and some of the underground facilities will be excavated during the first 4.3 years (1993 to 1998) of the 7.3 year construction period. The Stage 2 facilities would be completed in the last 3 years of the construction period (by 2000), which overlap with the first 3 years of the operations period. The operations period, which will last for 50 years, will consist of two phases. Radioactive waste will be received and emplaced during the 28 year emplacement phase. The underground facilities and surrounding environment will be monitored during this phase. The 22 year caretaker phase will follow completion of waste emplacement operations; the facilities, as well as the surrounding environment, will continue to be monitored, and the retrievability option will be maintained in compliance with Nuclear Regulatory Commission requirements for ensuring retrievability at any time up to 50 years after waste emplacement begins. Retrieval, if required, would take about 30 years. The decommissioning and closing of the repository will take about 8 years for the vertical emplacement repository, and about 3 years for the repository with horizontal waste emplacement.

10.1.4.2 Methods and Equipment

Design work completed to date indicates that area and geometric requirements, mine ventilation requirements, the requirements for stability of the underground workings, and retrievability considerations will be satisfied by a conventional room and pillar design. Excavation may be conducted using either a drill-blast-mucking technique or a continuous mechanical mining machine.

Shaft sinking of the shaft may be by conventional drill-and-blast technique or shaft drilling technique. The 3.7 m (12 ft) exploratory shaft is planned to be excavated by drill-and-blast techniques. The 1.8 m (6 ft) emergency egress shaft for the exploratory shaft facility is expected to be raisebored once access has been established to the prospective shaft bottom from the main exploratory shaft. The 6.1 m (20 ft) waste emplacement exhaust shaft may be constructed using the pilot-and-slash method by enlarging a raisebored pilot hole to the final shaft diameter.

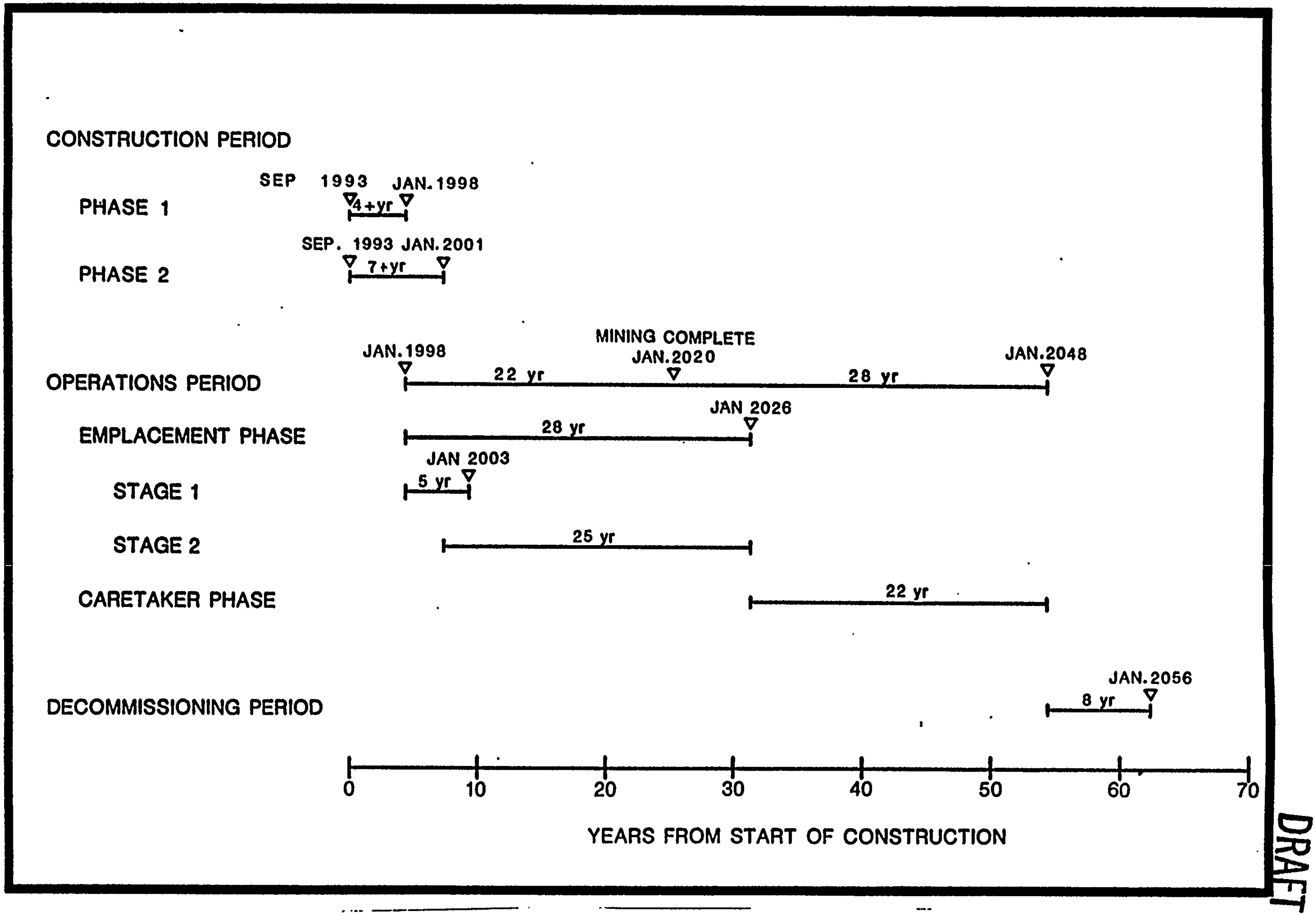

Figure 10.1.8. Repository Schedule for Vertical Emplacement

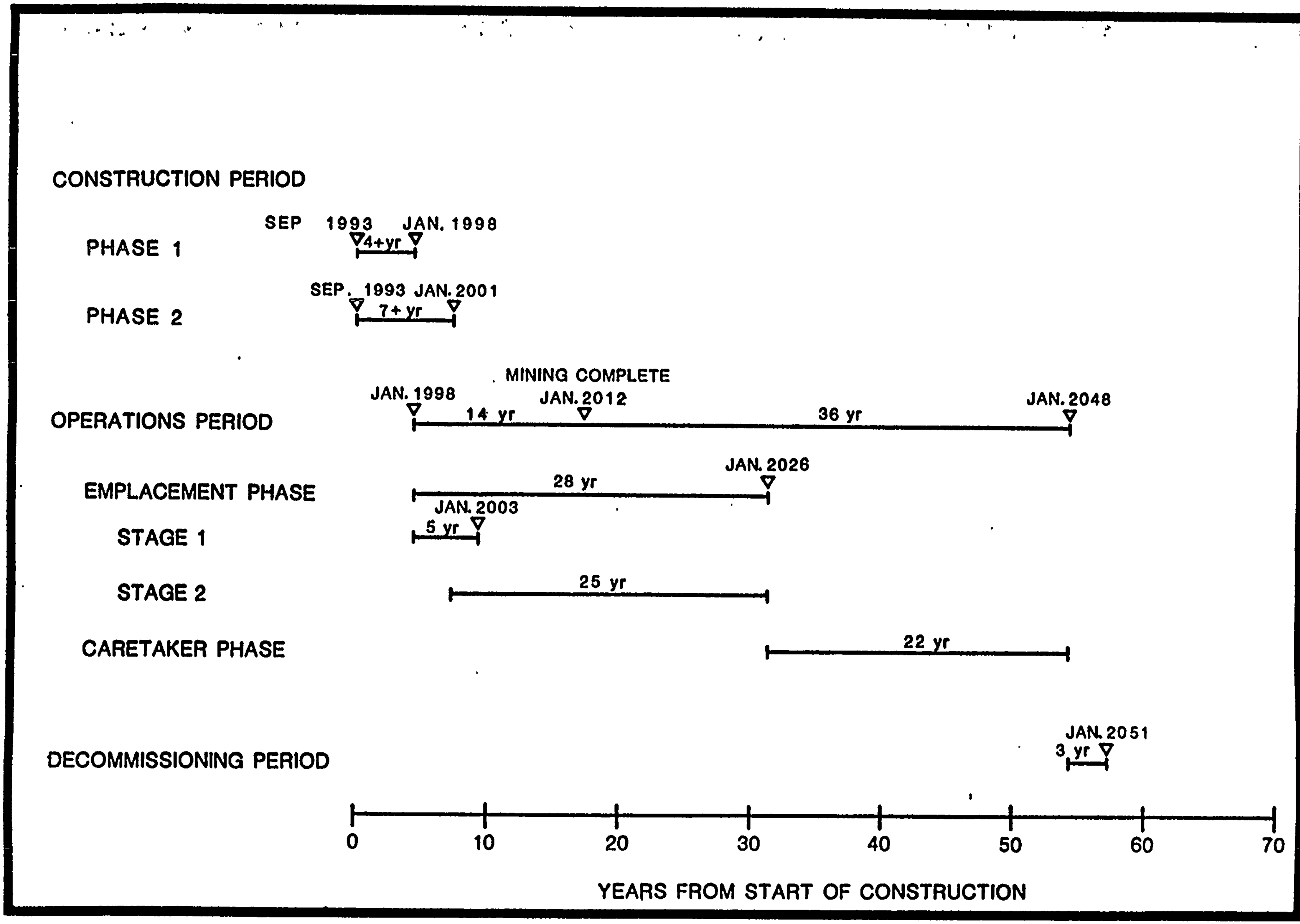

Figure 10.1.9. Repository Schedule for Horizontal Emplacement

10.1.5 Waste Emplacement Cycle

10.1.5.1 Waste Treatment

Reprocessing of commercial spent fuel is currently not being practiced in the United States so that the majority of the nuclear waste will be in the form of spent fuel assemblies. About 62,000 MT (68,000 tons) of spent fuel were estimated to require disposal in the first repository. Another 8,000 MT (9,000 tons) equivalent of defense high-level reprocessing waste and commercial reprocessing waste from West Valley will also be disposed of in the first repository. The spent fuel assemblies will be repackaged in special containers in preparation for underground disposal. The high-level reprocessing wastes will be vitrified in borosilicate glass and encased in disposal containers.

10.1.5.2 Waste Transport

Waste transport systems are still under analysis. Rail and truck transport are the options being investigated. Transport cask configurations are also the subject of scientific and engineering studies.

10.1.5.3 Waste Package

Reference waste package designs have been developed for consolidated spent fuel, spent fuel assemblies, and for defense and commercial West Valley high-level waste. The packages are intended for horizontal emplacement in boreholes and consist of either canistered spent fuel assemblies, consolidated spent fuel rods supported by an internal spider structure or canistered high-level waste within a thin-walled stainless-steel container. The container diameter and wall thickness is identical for all waste forms. Package designs are shown schematically in Figures 10.1.10 and 10.1.11.

No packing is included with any waste package.

The internal support structure for consolidated spent fuel is the same for either pressurized water reactor or boiling water reactor rods and can accommodate the rods from either six pressurized water reactor or 18 boiling water reactor assemblies. These are arranged in six segments around a central hexagonal space reserved for the placement of compacted non-fuel hardware from consolidation operations.

This same internal structure will accommodate 6 square canisters of consolidated spent fuel, each containing the fuel rods from 2 pressurized water reactor assemblies, or 6 intact boiling water reactor spent fuel assemblies. A variation of the internal structure will accept 3 canisters of consolidated spent fuel, each containing the rods from a single pressurized water reactor assembly, or 3 intact pressurized water reactor spent fuel assemblies.

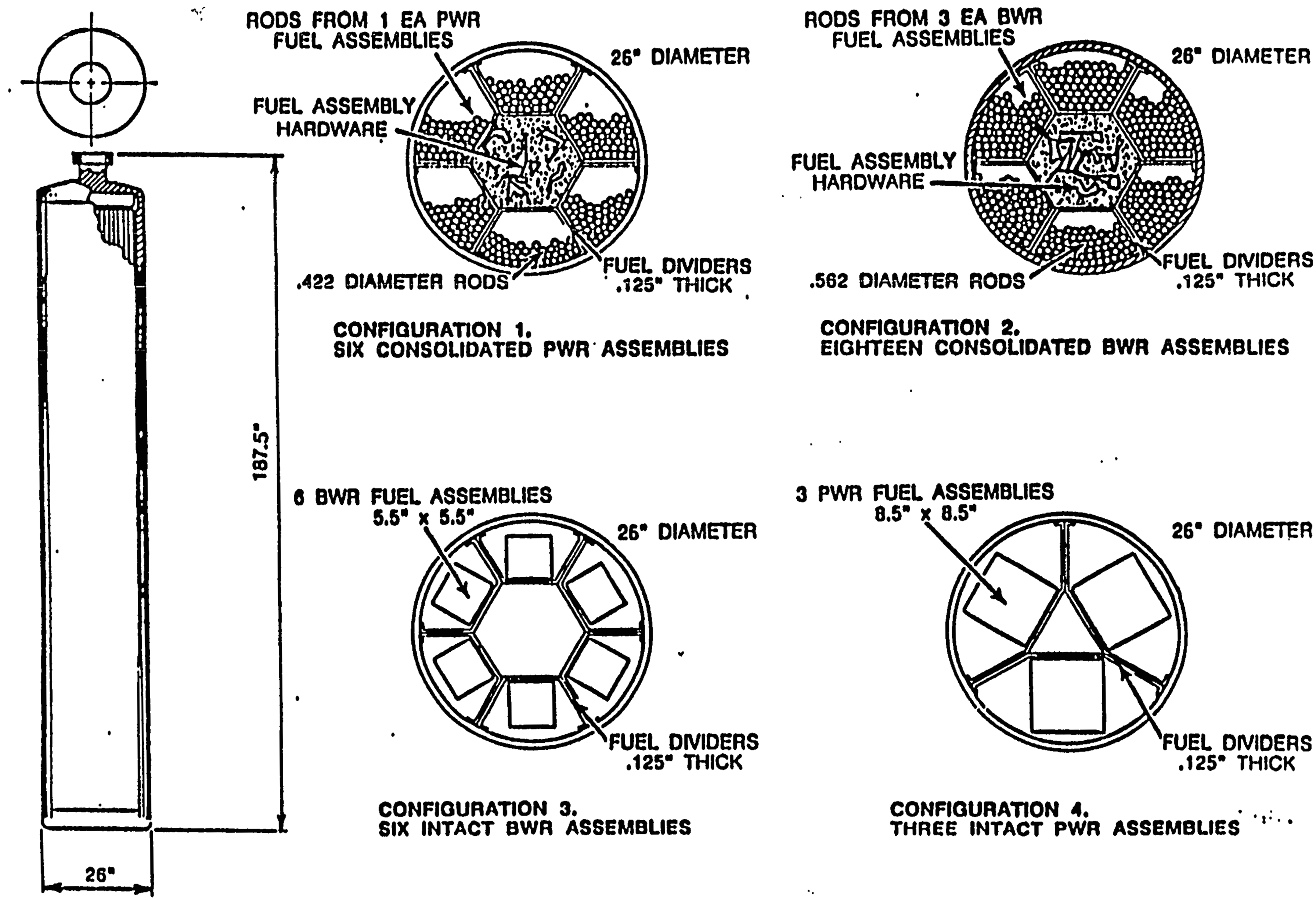

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Figure 10.1.10. Waste Package Reference Concept for Spent Fuel

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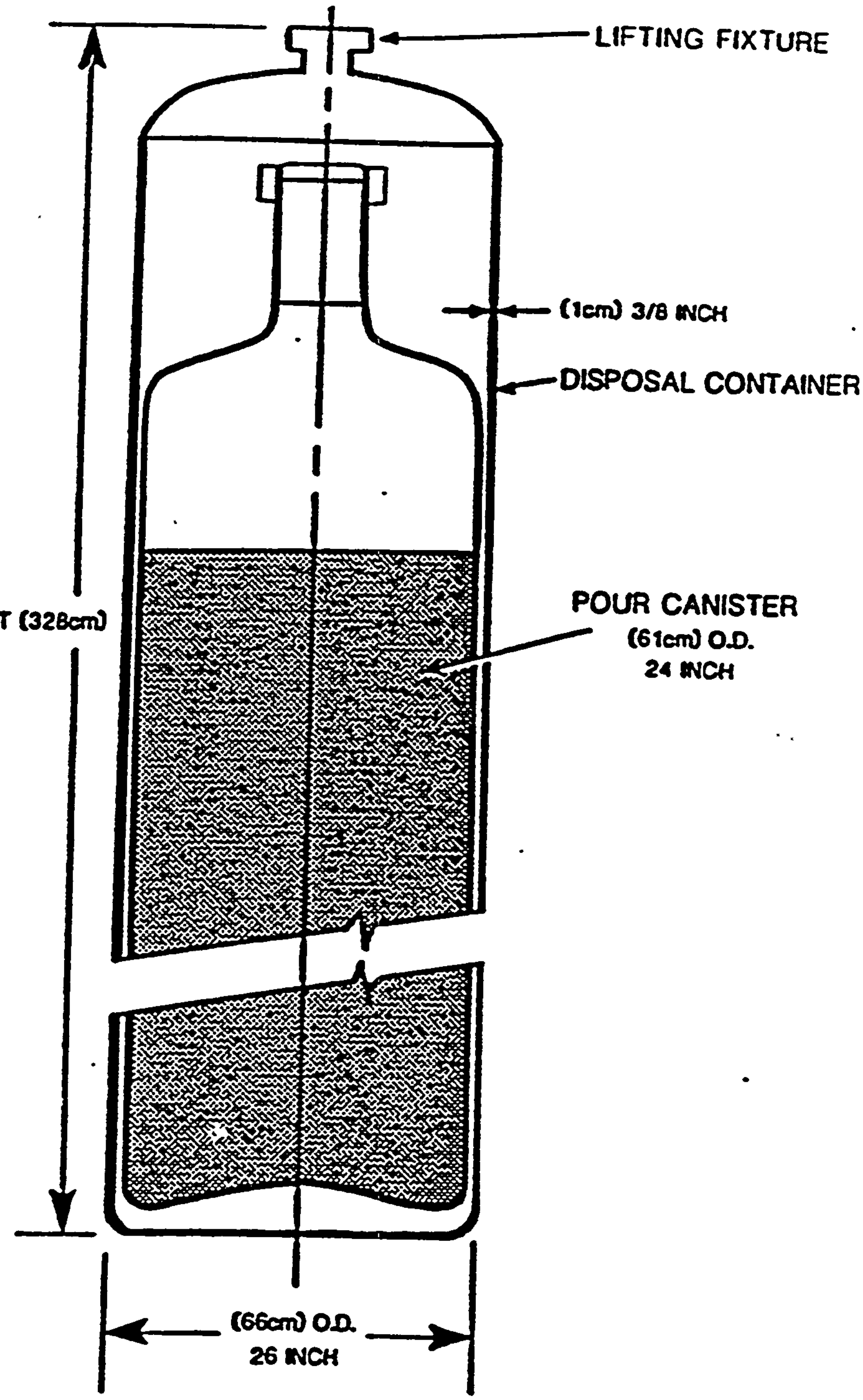

Figure 10.1.11. Waste Package Reference Concept for High-Level Waste

The various options permit an array of heat loadings that will accommodate spent fuel with a wide range of age (5 yrs and older) and burnups.

All waste packages containing spent fuel are to be filled with an inert cover gas prior to final closure.

Defense and commercial West Valley high-level waste is in vitrified form in a stainless-steel canister. The canister is inserted in the container for disposal.

Primary reliance for containment in a repository in tuff will be placed on the container which is designed to perform this function for a period of up to 1,000 years. No packing is included in the waste package design.

The tuff repository will be located at a horizon that is about 350 m (1,150 ft) below the surface and above the static water table. Under these conditions, the waste package will be exposed to atmospheric pressure. Since tuff is a structurally stable rock, no structural postclosure requirements are placed on the container. The only concern is corrosion.

Emplacement of the thermally hot waste packages will vaporize any tuff pore water and any incoming vadose water, so that the dominant environment of the container should be a steam-air mixture. The boiling point at the repository elevation is 96 °C (205 °F). Since the steam-air environment is less corrosively aggressive than an aqueous environment, waste-package loading and waste-package spacing in the repository is designed to maintain temperatures above the boiling point for a minimum of 300 years, the minimum containment lifetime requirement. Once temperatures drop below boiling, the amount of water available for corrosion and waste form dissolution will be limited to extremely small amounts. The maximum downward flux of water through the repository horizon is thought to be 0.5 mm/year (0.02 in/year).

Type 304L stainless steel was chosen as the reference material for construction of containers. It is a material that forms a passive oxide layer in an oxidizing environment that subsequently provides corrosion protection. It was chosen on the basis of its excellent general corrosion resistance in air and steam at temperatures in the range of 95° to 300°C (203 to 572°F) and in non-saline, near-neutral pH, oxic waters at temperatures below 95°C (203°F).

The major concern with the use of Type 304L stainless steel is its potential for localized corrosion as a result either of sensitization or the development of chloride concentrations. Sensitization can occur during welding of the container or during long-term storage at elevated temperatures (>150°C or 302°F). The sensitized structure is susceptible to intergranular corrosion and intergranular stress corrosion cracking. Concentration of chloride ions could result by periodic surges of water through fractures in the host rock above the repository that evaporate when it contacts the hot surface of the container. Pitting, crevice corrosion, and stress-corrosion cracking are corrosion processes associated with high chloride concentrations.

In the event Type 304L stainless steel is found to be inadequate, other back up materials are being evaluated. These are other 300 series austenitic stainless steels, Alloy 825, a high-nickel iron-chromium austenitic alloy, and copper and copper-base alloys.

10.1.5.4 Underground Waste Emplacement

The activities planned to occur during the emplacement phase include waste receipt, processing, and placement; continued underground construction to waste emplacement rooms and supporting services; the initial retrieval option period; and storage and management of mined rock for potential use as backfill.

During Stage 1 operations (Figures 10.1.8 and 10.1.9), surface and underground facilities will be constructed to receive and emplace a limited amount (400 MT or 440 tons per year) of spent fuel, unconsolidated fuel. This would be packaged at the site (assuming that it was not done elsewhere at, for example, a Monitored Retrievable Storage facility) for disposal in the repository. The Stage 2 facilities to be completed 3 years later than the Stage 1 facilities, will have a capacity of 3,000 MT (3,300 tons) per year and they would be capable of receiving other types of waste and consolidating spent fuel. Receipt rates will gradually increase during the early years of repository operation.

The receiving building (Figure 10.1.6) will provide for:

- Rail and truck inspection stations for inspecting incoming and outgoing traffic
- A suspect storage area where incoming shipments that do not meet repository acceptance standards would be held until corrective action can be taken
- A loading area for incoming and outgoing shipments
- A vehicle washdown facility
- A loading and unloading bay where shipping packages will be removed and loaded onto their carriers
- A decontamination station in the waste handling building where waste packages will be checked and decontaminated
- A station in the waste handling building where cask closures will be prepared for connecting the casks to the hot cell port for unloading.

Before disposal, waste packages will be sealed in waste disposal containers designed to meet the minimum lifetime requirements set by the Nuclear Regulatory Commission. To meet these requirements, the minimum life time of the waste packages would be between 300 and 1,000 years under the expected underground environmental conditions in the repository. These waste disposal containers are one component of a system of engineered barriers, including waste forms, overpacks, and packing materials that may be used as part of the repository system.

The waste disposal containers will be placed either in vertical holes in the floor of the disposal drifts (vertical emplacement) or in long horizontal holes in the walls (horizontal emplacement). If the waste is placed horizontally, each borehole would contain multiple waste disposal containers, if vertically, each borehole would contain one waste disposal container.

The surface and underground facilities at the repository that handle radioactive waste will be operated at less than atmospheric pressure. Exhaust air from the surface facilities will be processed through a prefilter and a series of high efficiency particulate filters before going being discharged to the atmosphere. Exhaust from the underground waste storage rooms will be directed to a surface building where the exhaust will be monitored and filtered if necessary prior to discharge into the atmosphere. The ventilation system for the underground construction areas will be physically separated from the waste emplacement ventilation circuit.

The following sections summarize briefly the vertical and horizontal emplacement procedures described in recent reports published by Sandia National Laboratories.

10.1.5.4.1 Horizontal Emplacement. The basic operations required for the horizontal emplacement of radioactive waste packages include:

- Preparing a horizontal emplacement borehole
- Loading a waste canister onto a dolly and into a waste transporter cask at the surface storage facility
- Moving the loaded waste transporter underground through the access ramp and access drift to the emplacement borehole
- Moving the waste package onto a dolly through a temporary shielding mechanism into the borehole
- Repeating the waste package and transporter operations until a full complement of waste packaged is emplaced
- Installing a shield plug in the borehole and removing the shielding mechanism
- Installing a cover on the borehole.

Figure 10.1.12 shows a proposed layout of the emplacement panels and boreholes.

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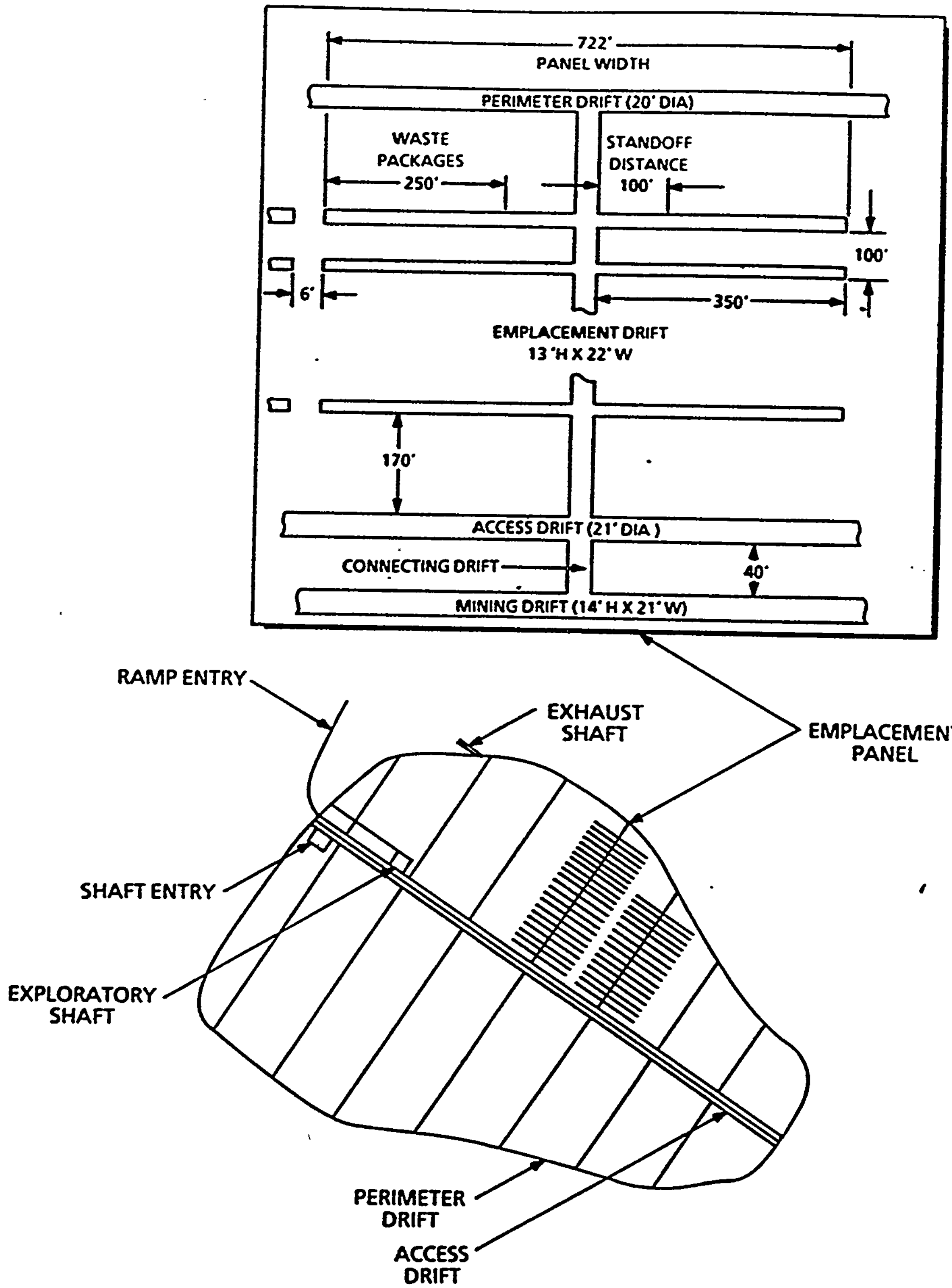

Figure 10.1.12. Repository Layout for Horizontal Boreholes

The boreholes will be spaced at about 52 m (170 ft) and drilled 107 m (350 ft) horizontally into the wall of the emplacement drift. The configuration of the borehole is shown in Figure 10.1.13. The first 4.4 m (14.5 ft) of the 0.94 m (37 in) diameter borehole will be enlarged to accommodate the entry liner. The steel borehole liner and entry liner configurations are shown in Figure 10.1.14. A 107 m (350 ft) long liner will accommodate about 15 waste packages with a stand-off distance of 30 m (100 ft). The stand-off distance is the distance from the last waste package emplaced to the borehole collar. The borehole preparation procedure is shown in Figure 10.1.15.

The waste package emplacement procedure from the collection of the waste package to the completion of emplacement is shown in Figure 10.1.16. The transporter will be equipped with a shielded transport cask mounted on a rotary table and hydraulic levelling arrangement to allow collection of the waste canister at the surface and insertion into the emplacement borehole (Figures 10.1.17 and 10.1.18). Alignment guides bolted to a concrete pad at the borehole location will assist in the location of the transporter at the borehole before insertion of the waste canister (Figure 10.1.19). The waste canisters will be mounted on a dolly which is pulled into/pushed out of the transport cask during collection and emplacement operations. Figure 10.1.20 shows the arrangement. The dolly consists of a curved steel plate with rollers so the waste canisters can be inserted into the steel lined emplacement borehole. Figure 10.1.20 shows the emplacement borehole during emplacement operations.

After waste canister emplacement is complete, a modified forklift (Figure 10.1.21) will be used to insert a shield plug in the borehole and install cover over the borehole (Figures 10.1.22 and 10.1.23).

Retrieval of the waste canisters, if necessary, will be carried out using the same equipment and by reversing the operations described above (Figure 10.1.24).

10.1.5.4.2 Vertical Emplacement. The basic operations required to emplace a waste package in a vertical borehole include:

- Prepare the waste emplacement borehole
- Loading the waste package into a transporter cask at the surface storage facility
- Moving the loaded transporter underground through the access ramp to a vertical borehole in an emplacement drift
- Lowering the waste package through a temporary shielding mechanism (shielding closure or collar) into the borehole
- Installing a shielding device (plug) or shielding material (backfill) above the emplaced waste package

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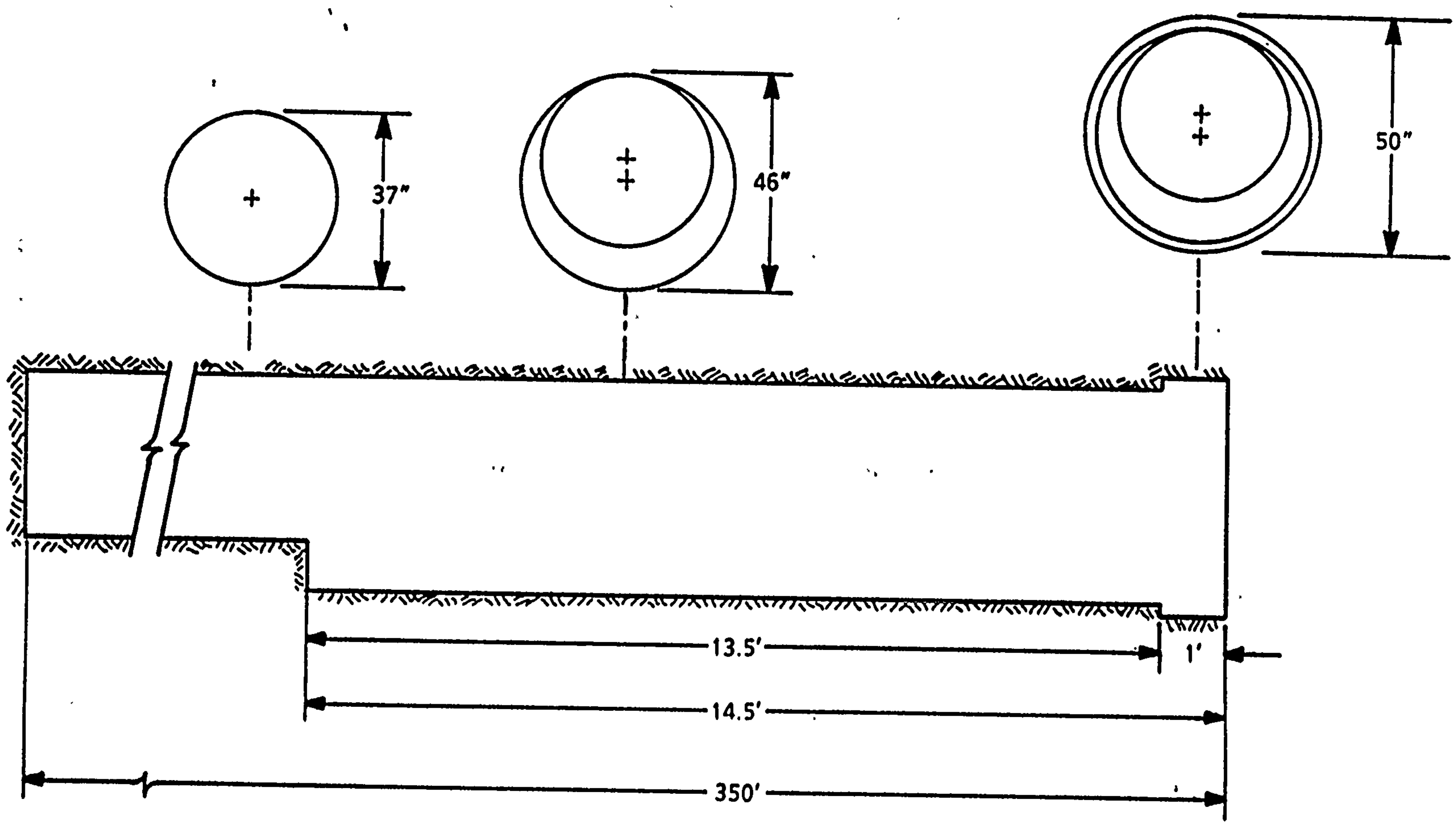

Figure 10.1.13. Horizontal Borehole Configuration

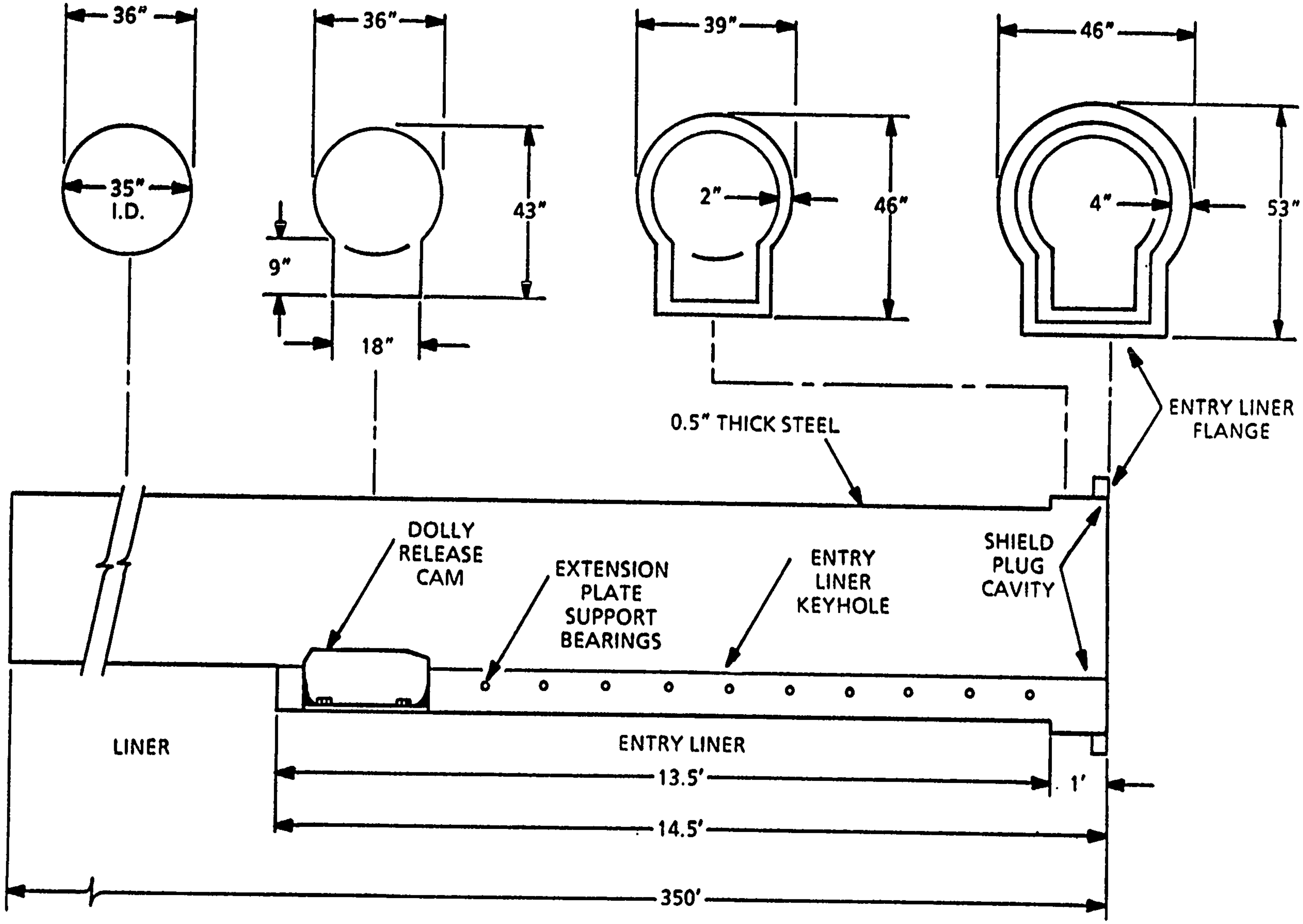

Figure 10.1.14. Liner and Entry Liner

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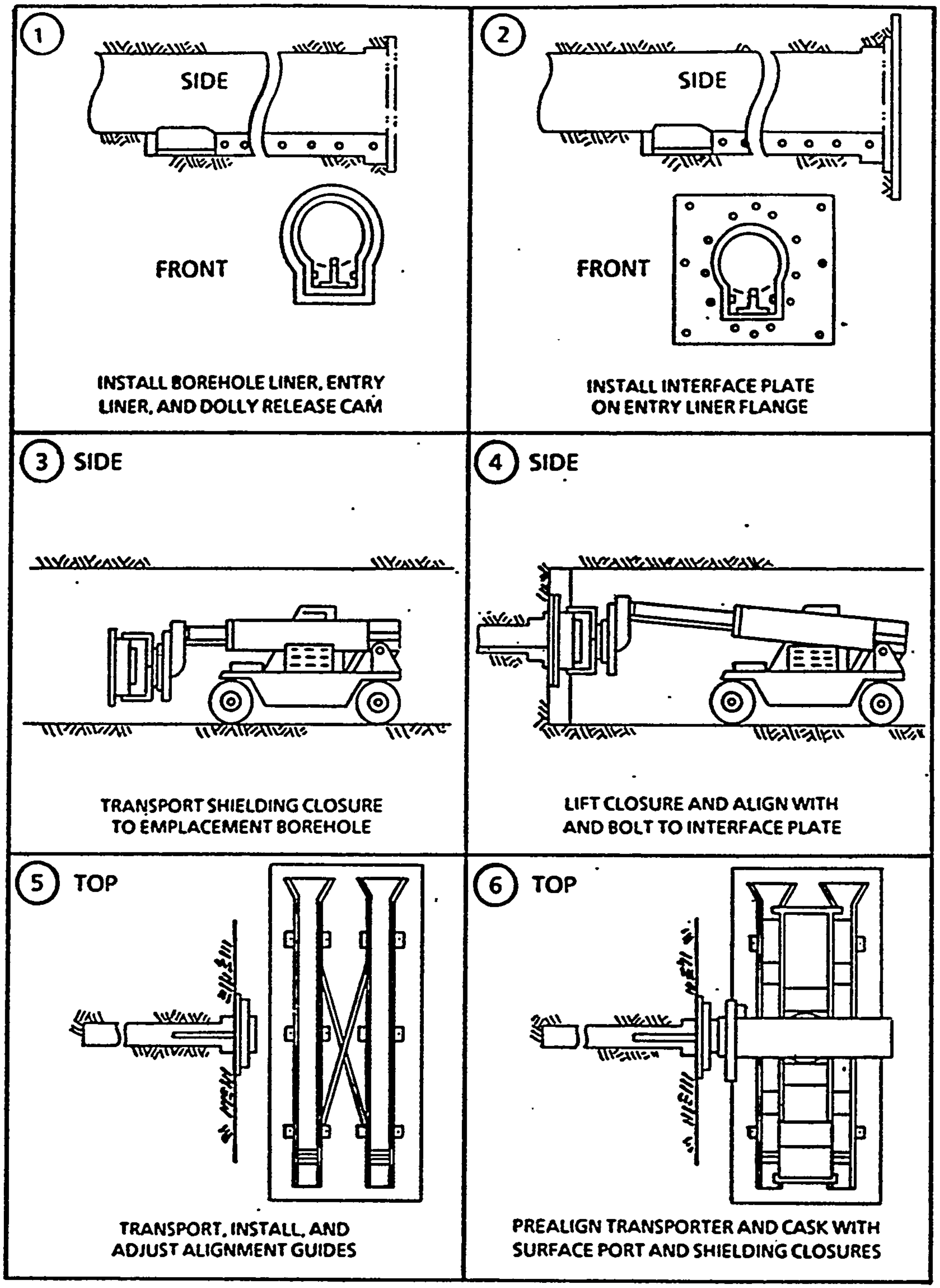

Figure 10.1.15. Horizontal Borehole Preparation

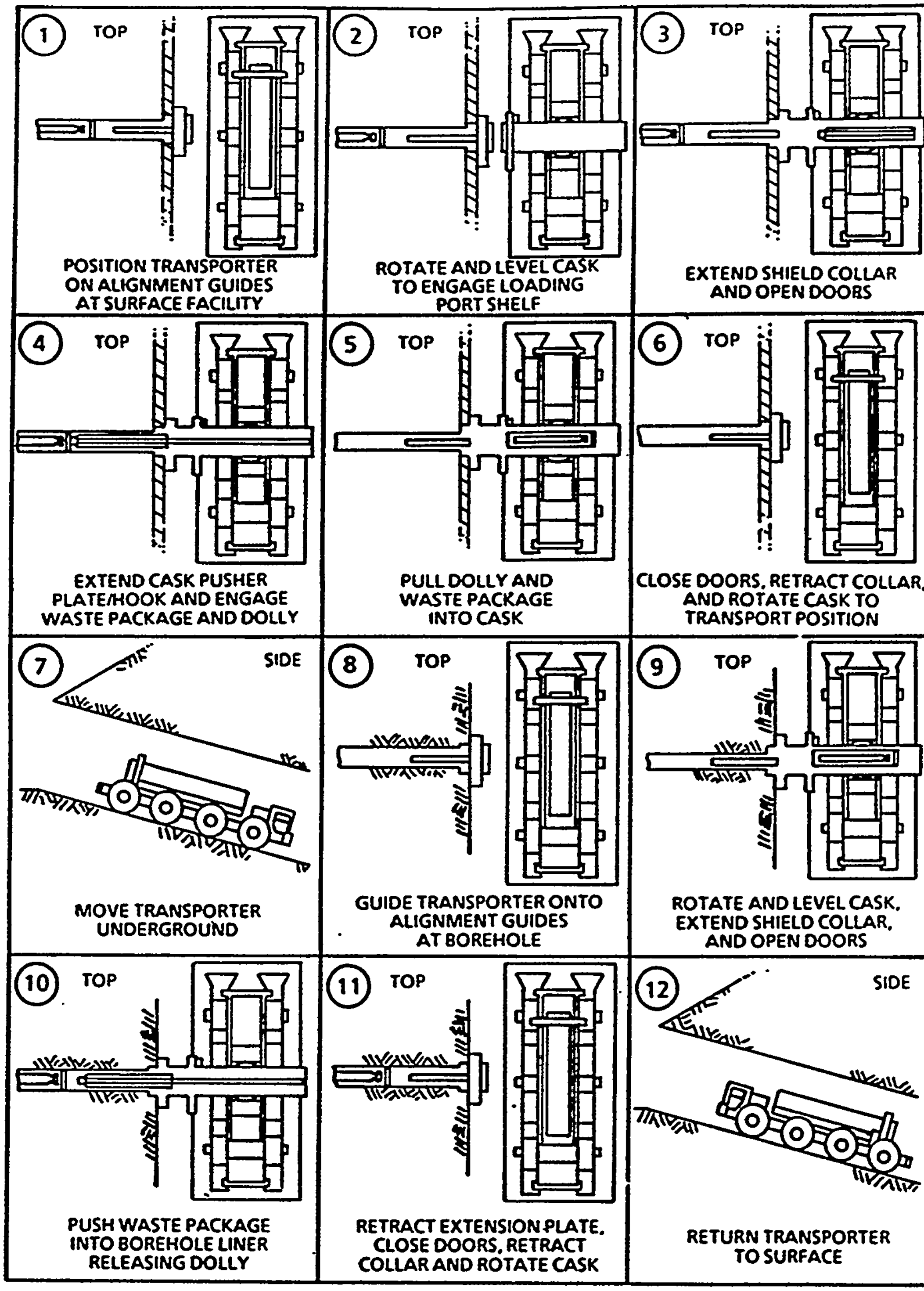

Figure 10.1.16. Waste Package Emplacement

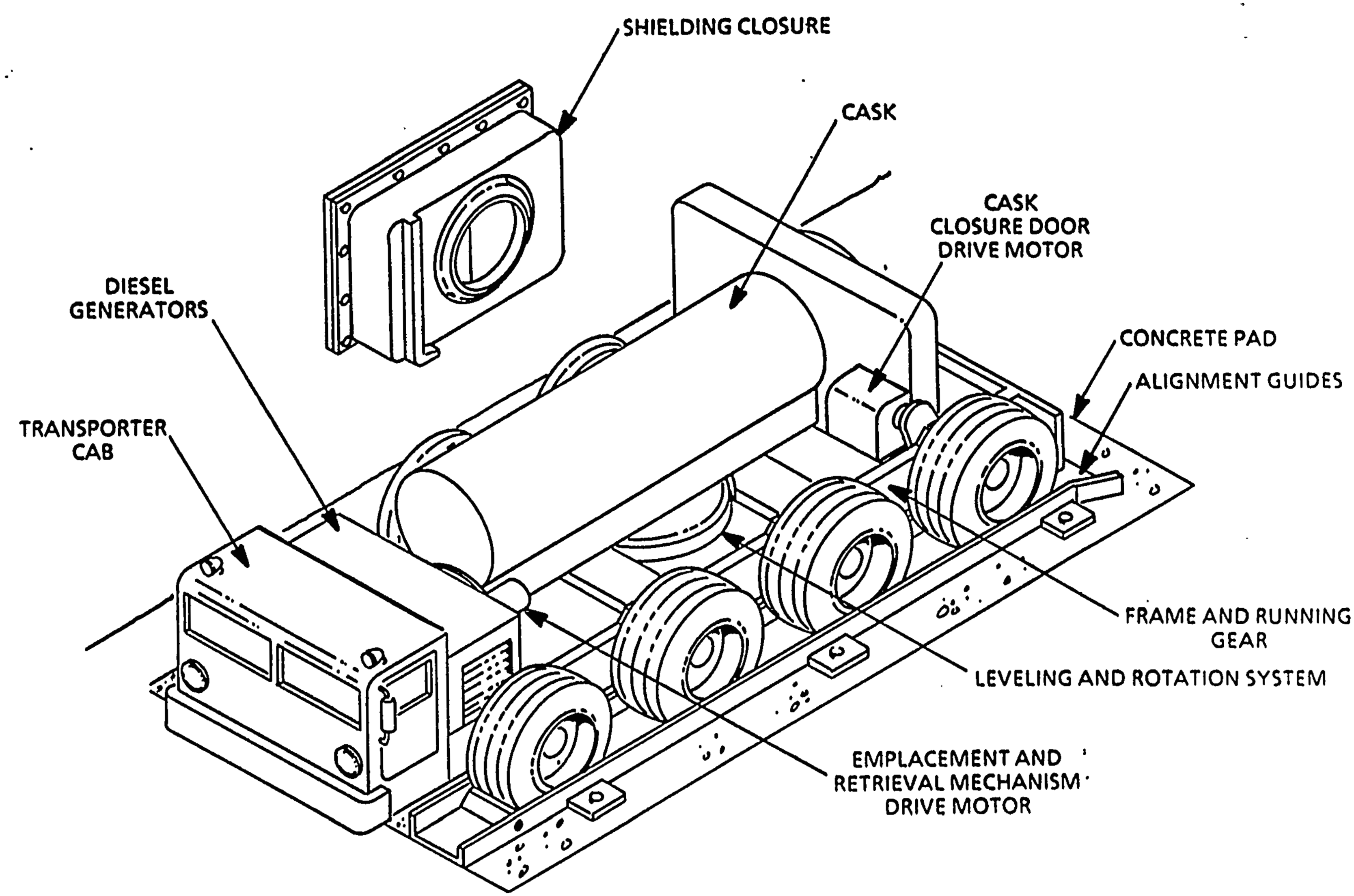

Figure 10.1.17. Waste Package Transporter in Transport Mode

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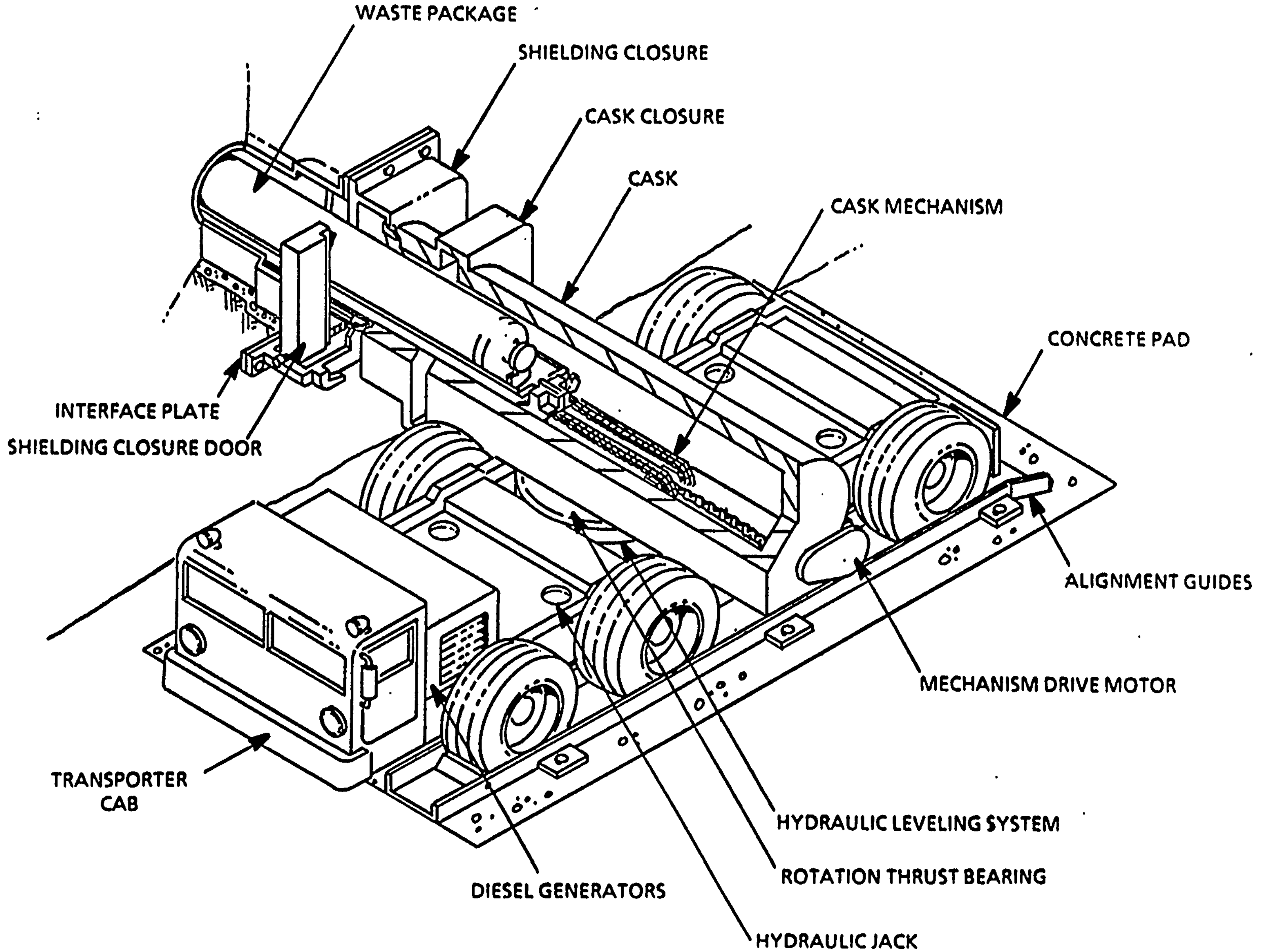

Figure 10.1.18. Waste Package Transporter in Emplacement/Retrieval Mode

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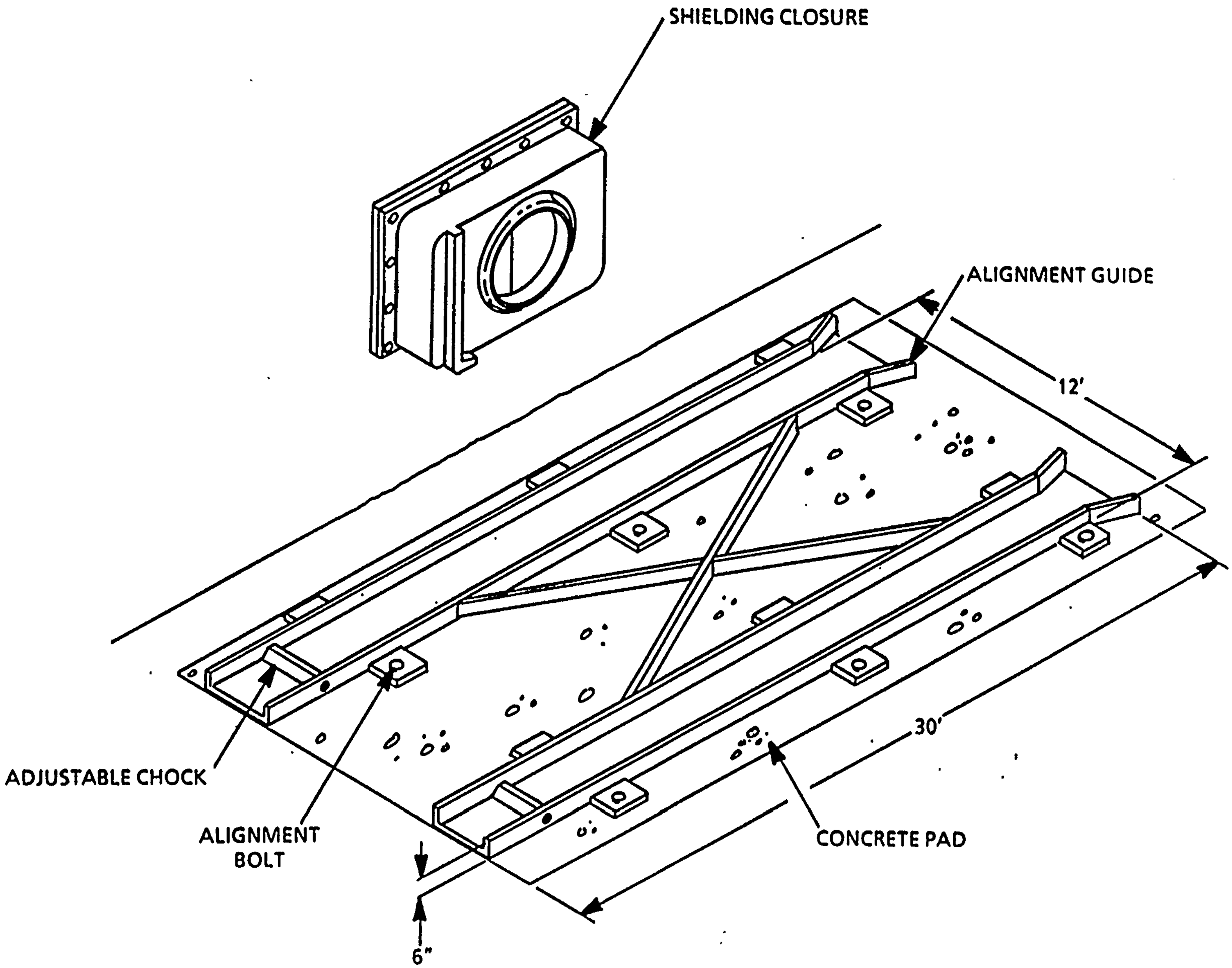

Figure 10.1.19. Transporter Alignment Guides

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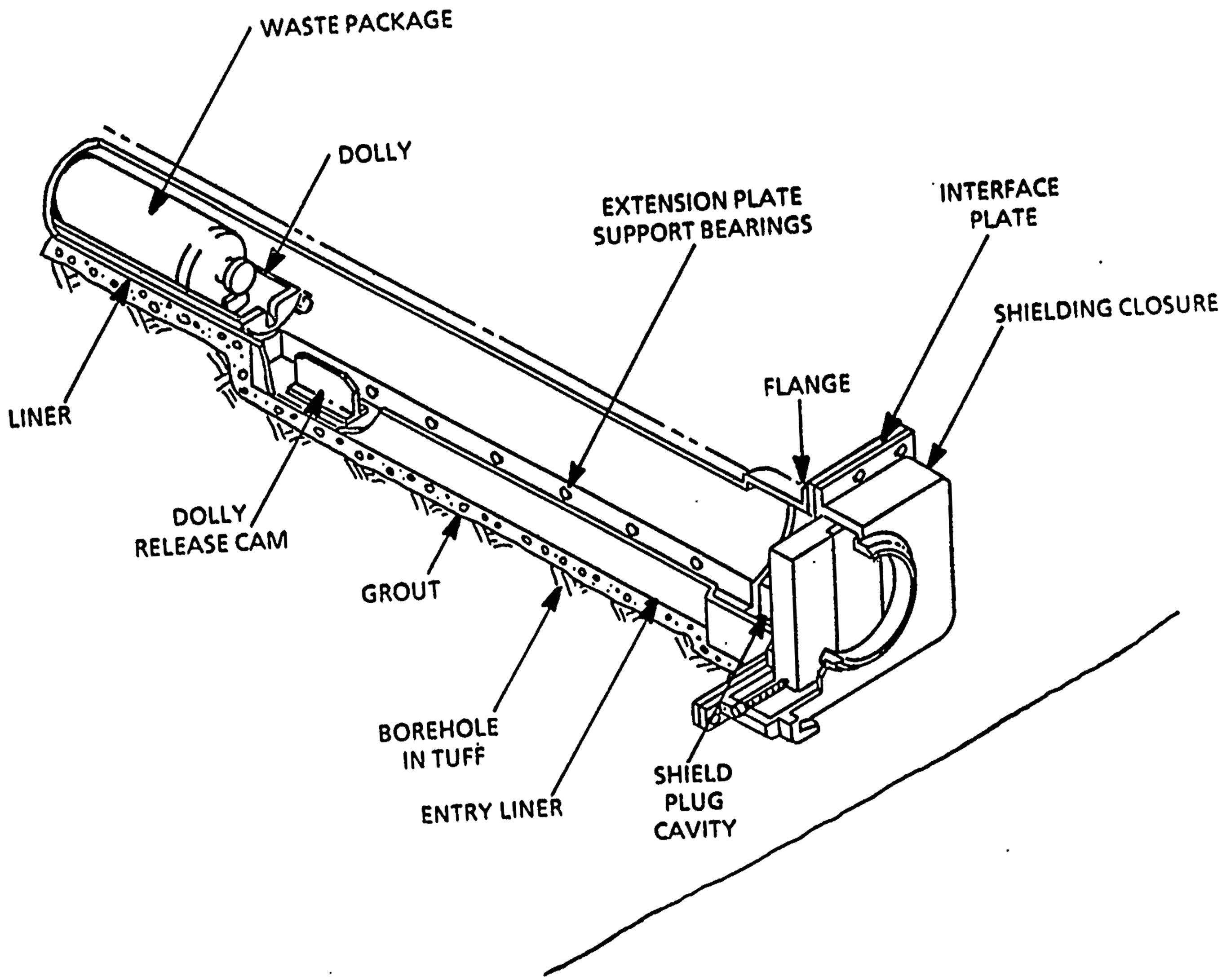

Figure 10.1.20. Horizontal Borehole during Emplacement or Retrieval

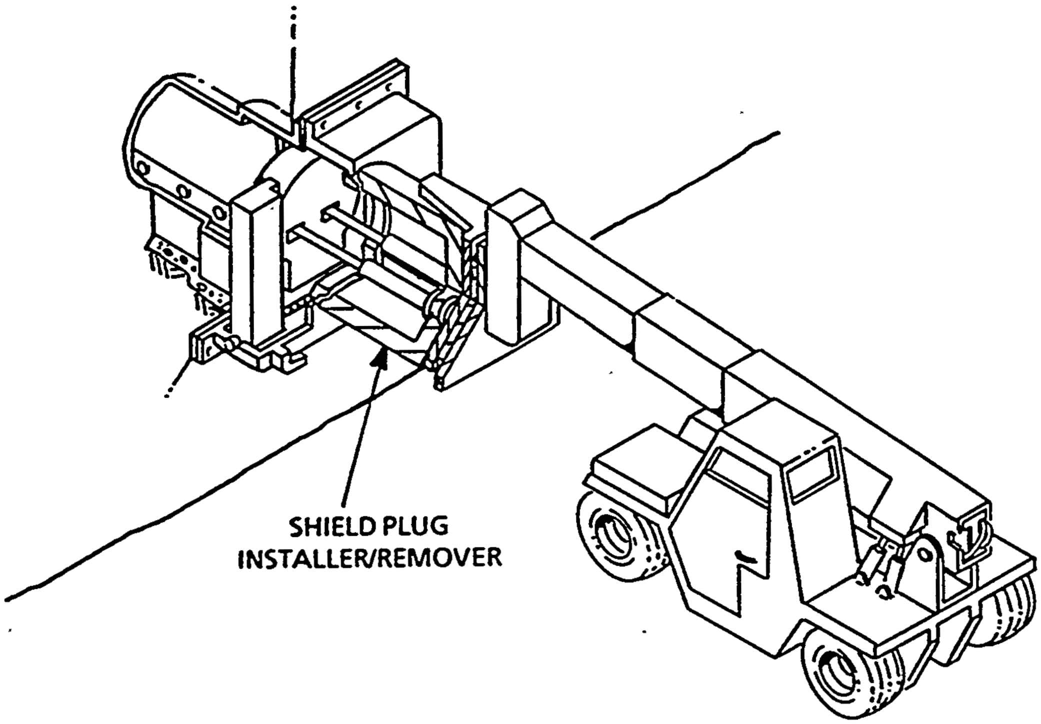

Figure 10.1.21. Shield Plug Installer/Remover

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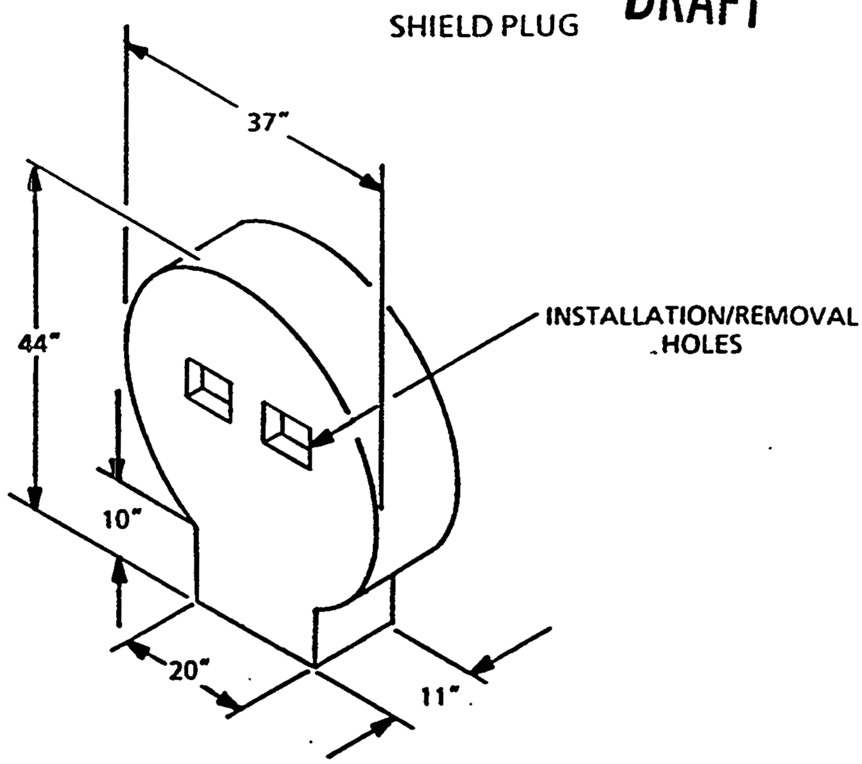

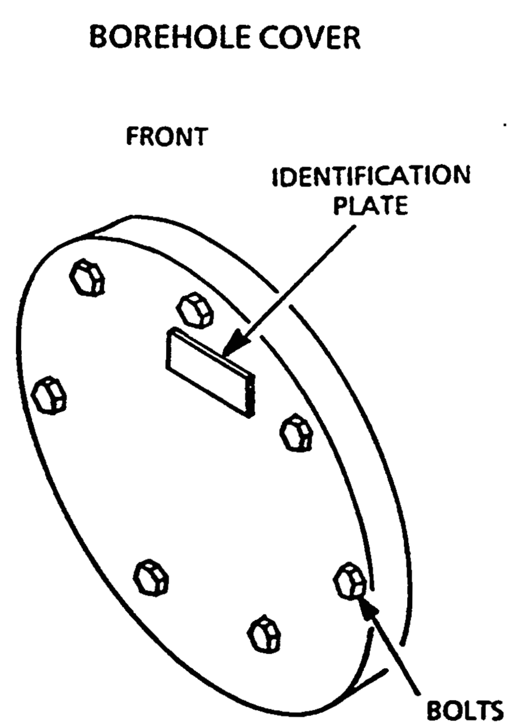

Figure 10.1.22. Borehole Plug and Cover

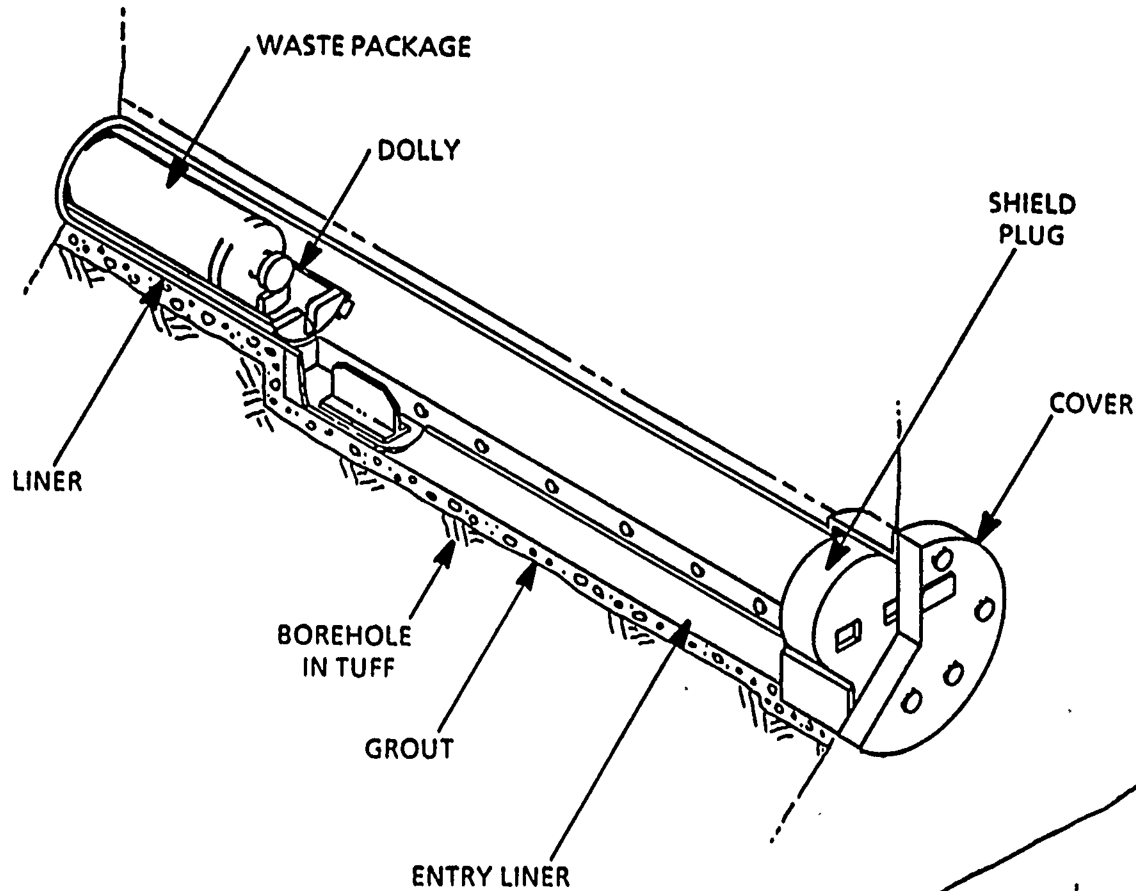

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Figure 10.1.23. Horizontal Emplacement Borehole after Closure

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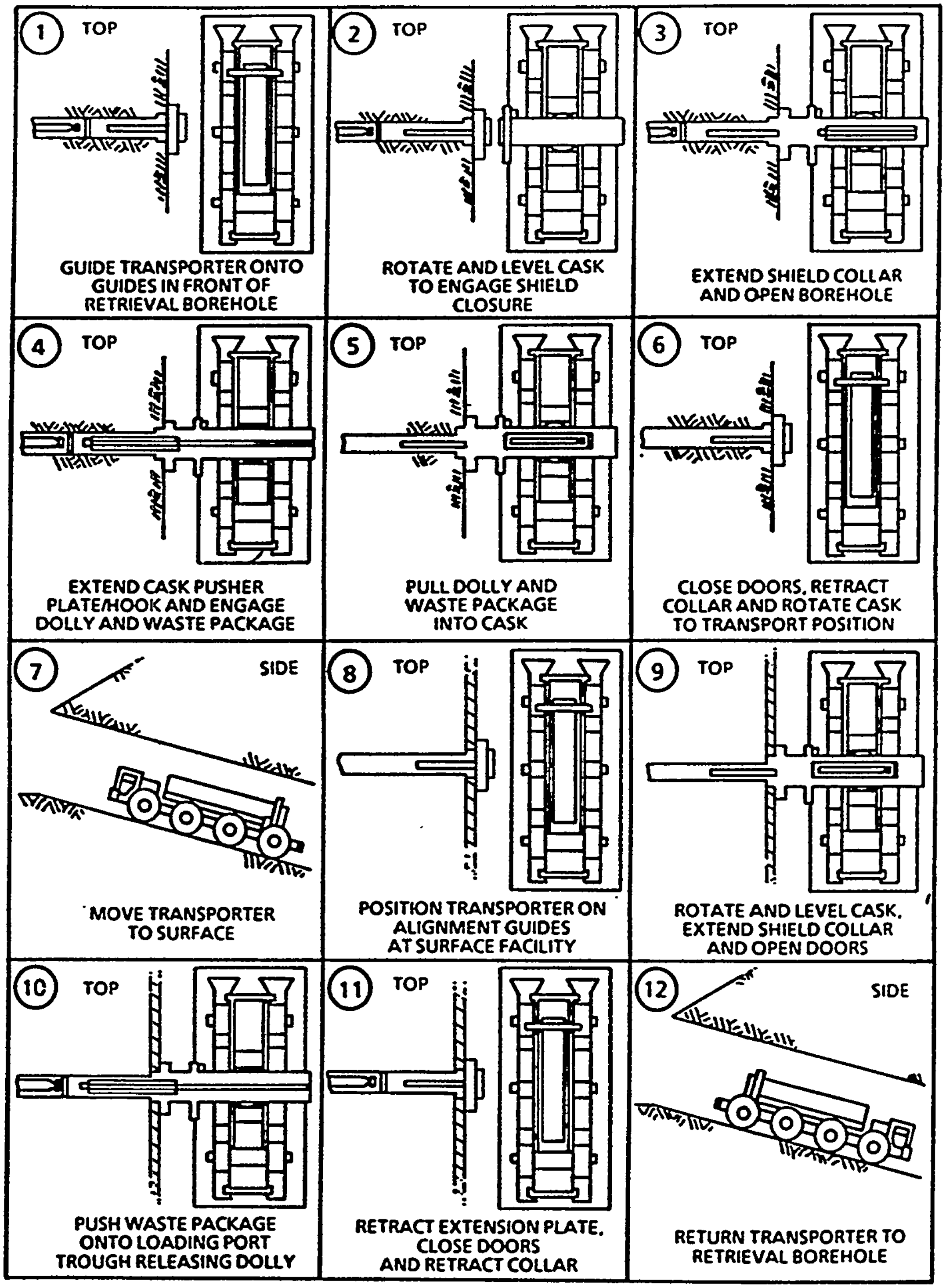

Figure 10.1.24. Waste Package Retrieval

- Removing the temporary shielding mechanism
- Installing a cover over the borehole.

Figure 10.1.25 shows the two methods being considered for vertical borehole emplacement and Figure 10.1.26 shows the configuration of the boreholes for the two methods. The plug method uses a cast iron or steel plug to seal the borehole above the waste package. With the backfill method, the emplacement borehole is filled with a backfill material to seal the waste package in the borehole. The emplacement drifts will be spaced at about 45 m (150 ft) and the borehole spacing will be about 4 m (13 ft).

Figure 10.1.27 shows the emplacement operation for the plug method of emplacement from collection of the waste canister at the surface, emplacement in the borehole and removal of the transporter. The emplacement borehole will be drilled into the floor of the emplacement drift and will be about 7.6 m (25 ft) deep (Figure 10.1.26). The diameter of the hole will be about 0.73 m (29 in) with the upper 1.8 m (6 ft) enlarged to 0.86 m (34 in) to accommodate the liner for the plug (Figure 10.1.28), grouted into the borehole. The liner extends far enough into the borehole support the upper part of the waste canister and is equipped with bevelled shoulder on the inside and outside to support the plug (Figure 10.1.28). A support plate is installed in the bottom of the borehole (Figure 10.1.29) to center the waste container and isolate it from the bottom of the borehole. A shielding enclosure will be installed at the borehole collar (Figure 10.1.25). The transporter is shown in the transport mode in Figure 10.1.30 and in the emplacement mode in Figure 10.1.31. A section through the emplacement operation is shown in Figure 10.1.32. When the waste canister has been lowered into the borehole, the transporter is removed and returned to the surface. A modified forklift (Figure 10.1.33) will place the plug installer (Figure 10.1.34) over the borehole and lower the plug into place (Figure 10.1.35). A cover will then be installed over the borehole as a final seal (Figure 10.1.29). Figure 10.1.36 shows the completed disposal borehole. Retrieval of the waste package can be achieved by reversing the emplacement procedure.

The emplacement borehole for the backfill method is shown in Figure 10.1.26. The depth of the borehole will be about 7.6 m (25 ft), the diameter about 0.72 m (28 in) and the upper 1 m (3 ft) enlarged to 0.86 m (34 in) to accommodate the thimble (Figures 10.1.25 and 10.1.28). A shielding enclosure is placed at the borehole collar (Figure 10.1.25). Figure 10.1.37 shows the emplacement operation which is basically similar to the plug method except that the transporter remains over the borehole after lowering the waste canister to aid in placing the backfill. The backfill will be installed using a mobile hopper and feeder unit (Figure 10.1.35) which can place the backfill into the borehole through the transport cask over the borehole. When the borehole is full, the equipment is removed and a borehole cover installed (Figure 10.1.29). The completed disposal borehole is shown in Figure 10.1.36. For the retrieval operations, should they become necessary, a backfill drill would be needed to drill out the backfill to expose the top of the waste canister so it can be removed again with the transporter (Figures 10.1.38 and 10.1.39).

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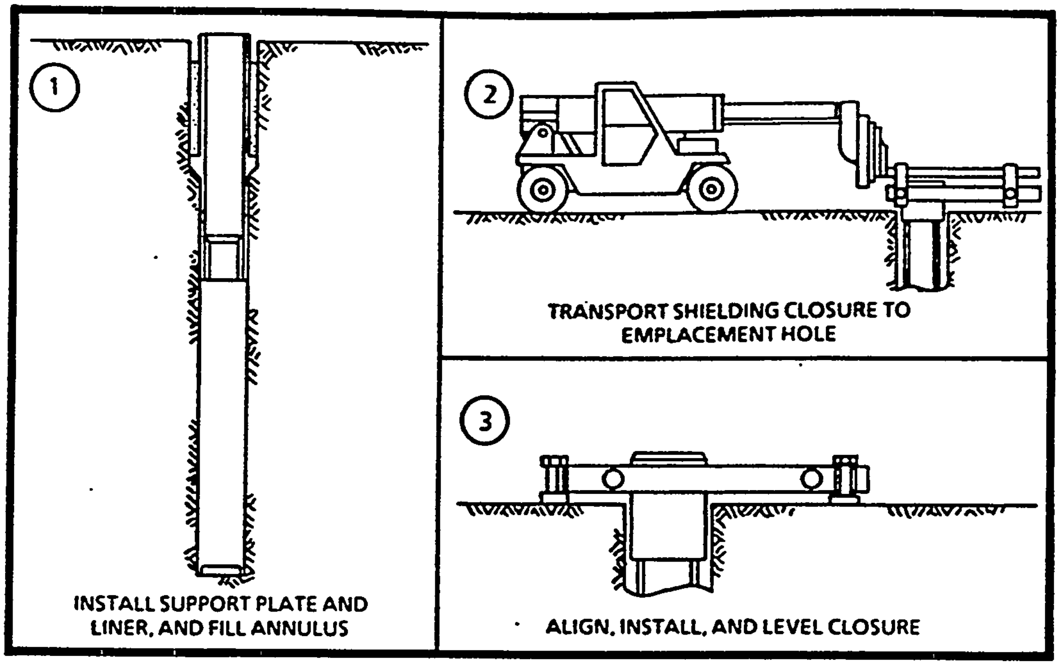
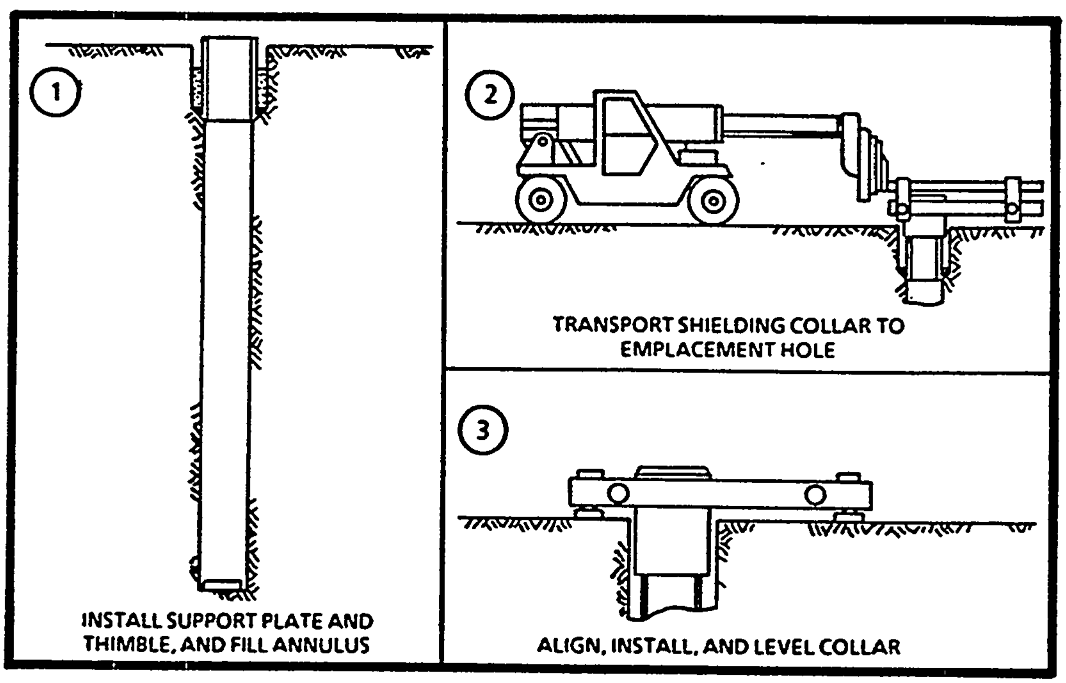


PLUG METHOD



BACKFILL METHOD

Figure 10.1.25. Vertical Borehole Preparation

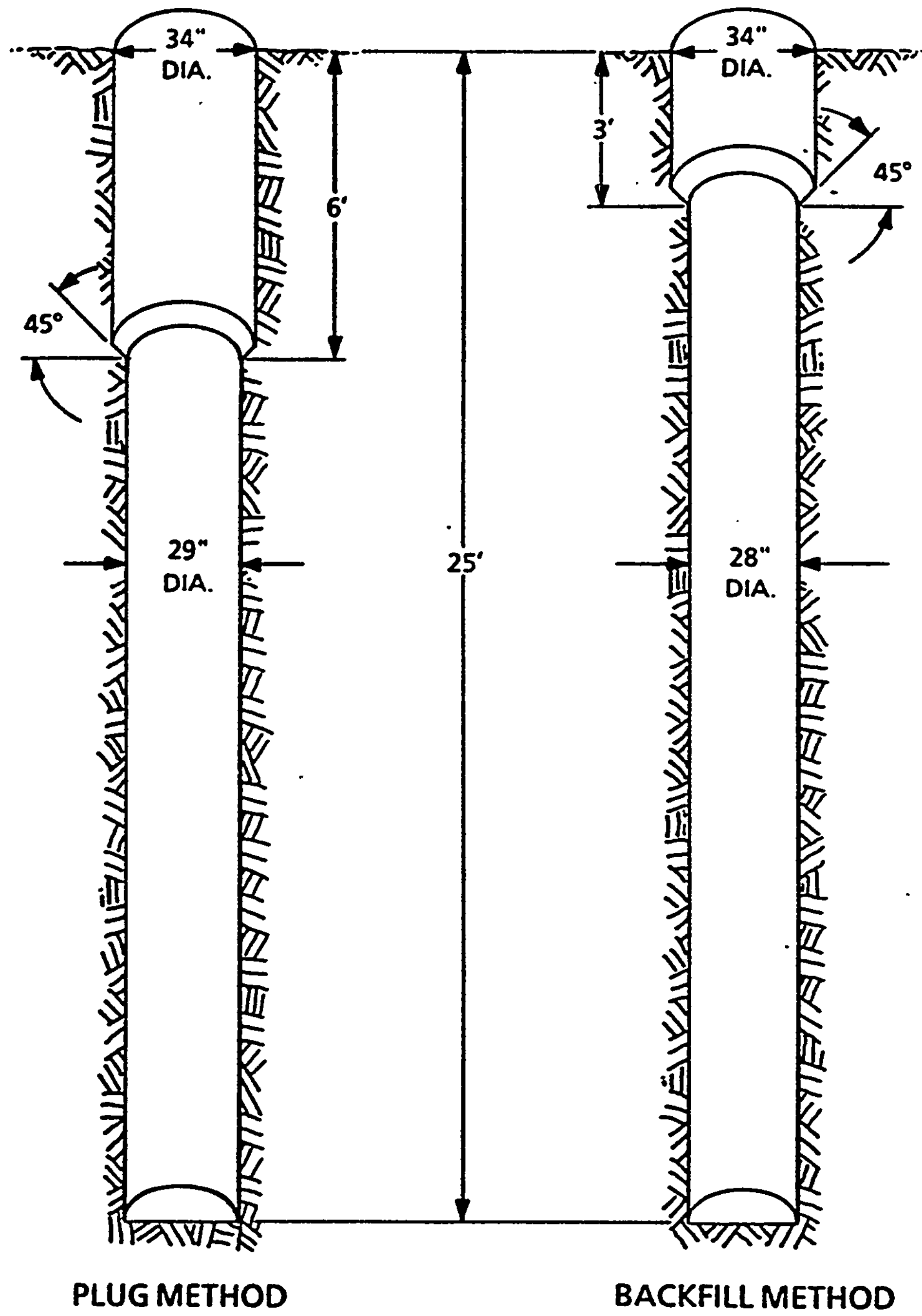


Figure 10.1.26. Boreholes for Plug and Backfill Methods

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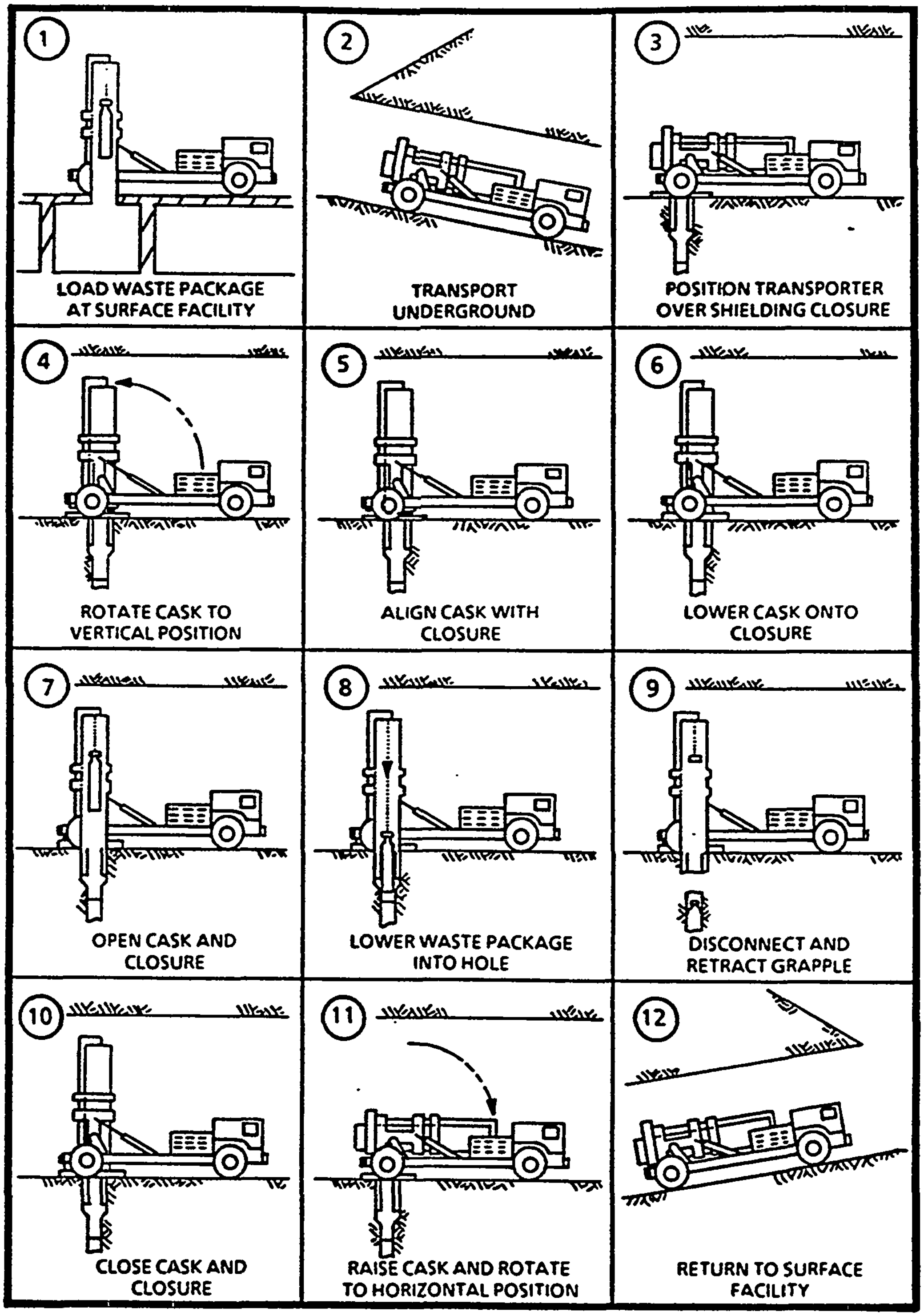

Figure 10.1.27. Vertical Emplacement for Plug Method

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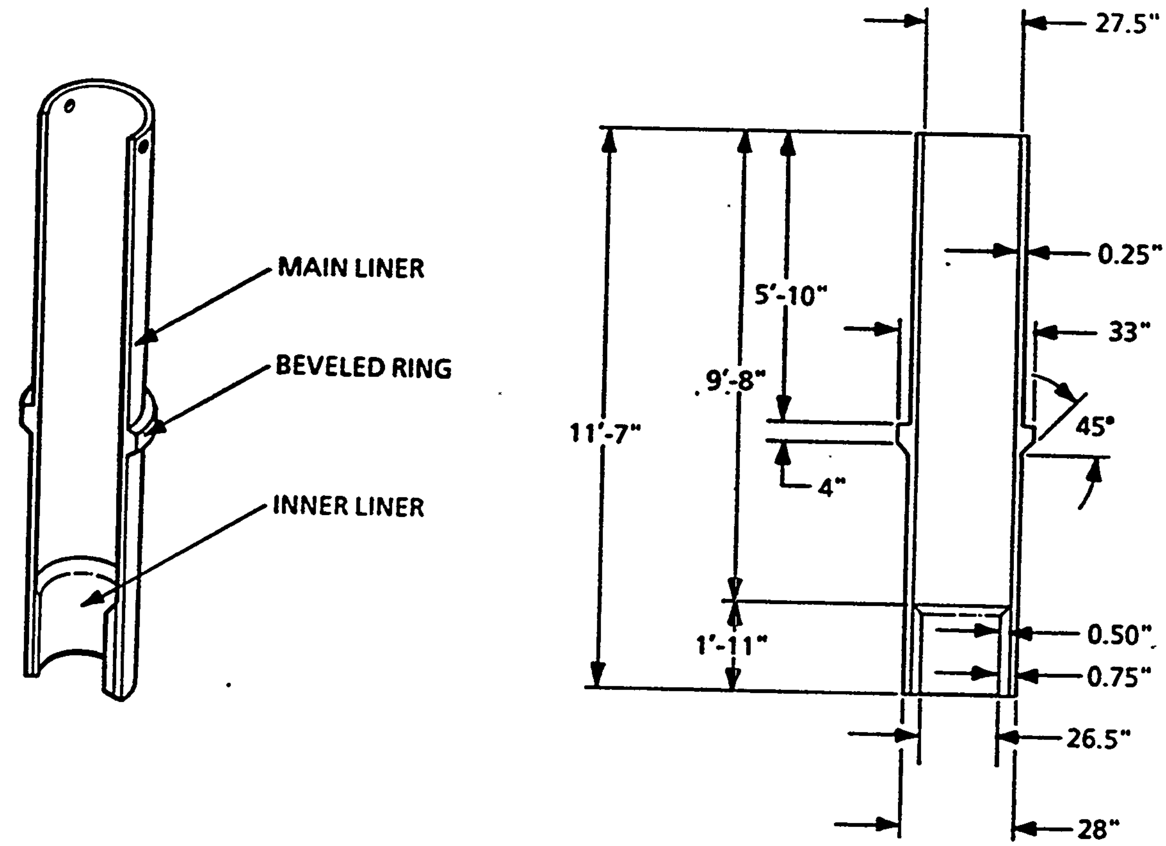
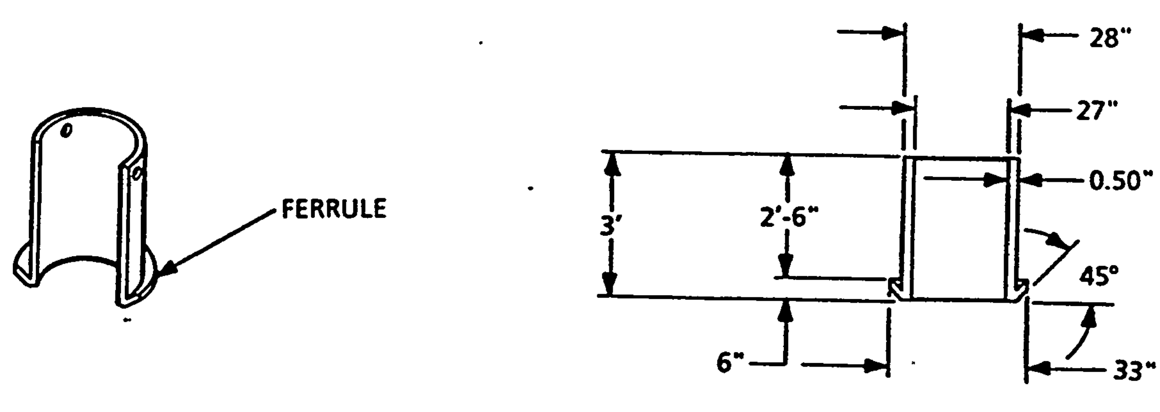


LINER FOR PLUG METHOD



THIMBLE FOR BACKFILL METHOD

Figure 10.1.28. Liner and Thimble

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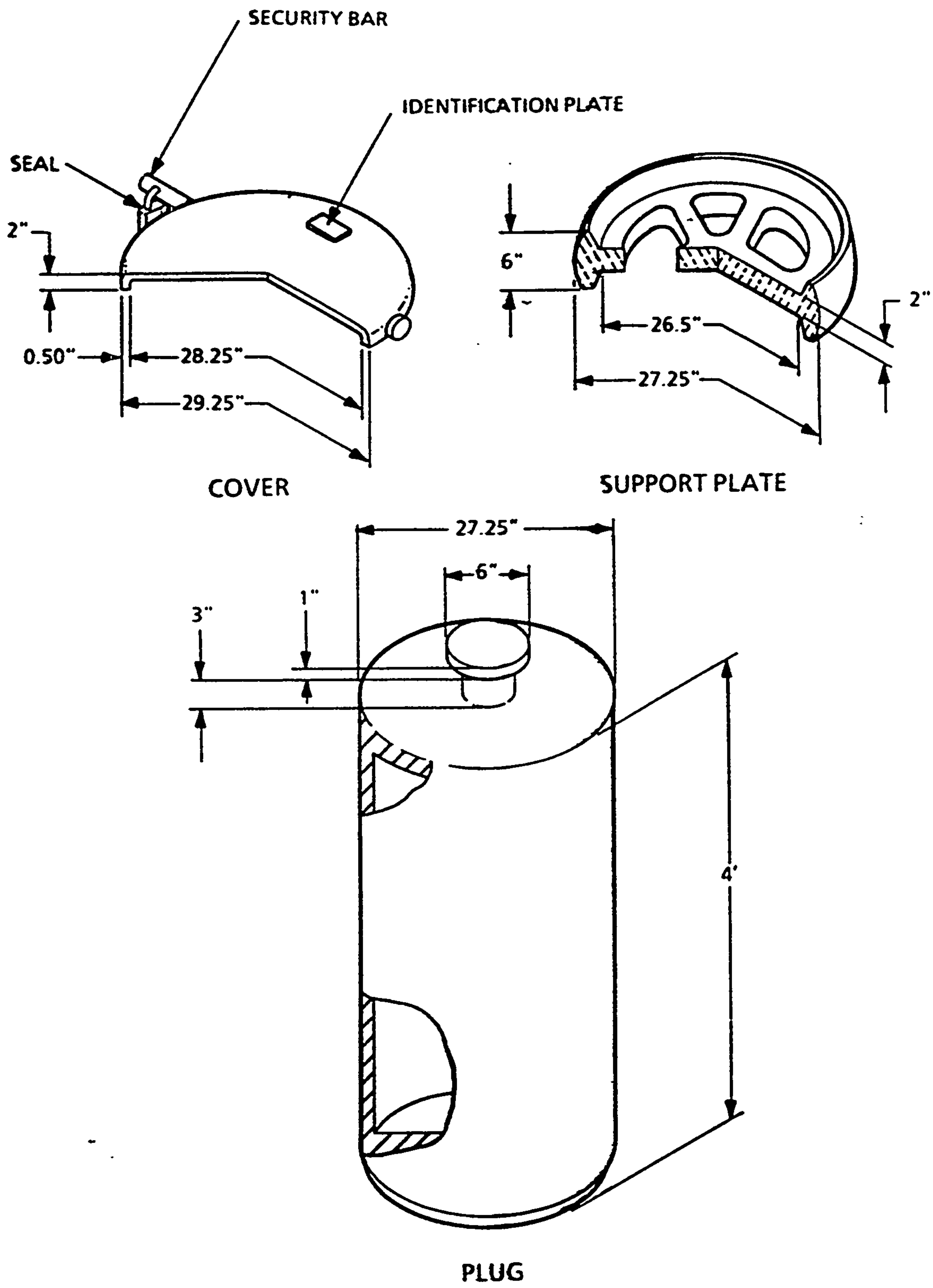

Figure 10.1.29. Support Plate, Plug, and Cover

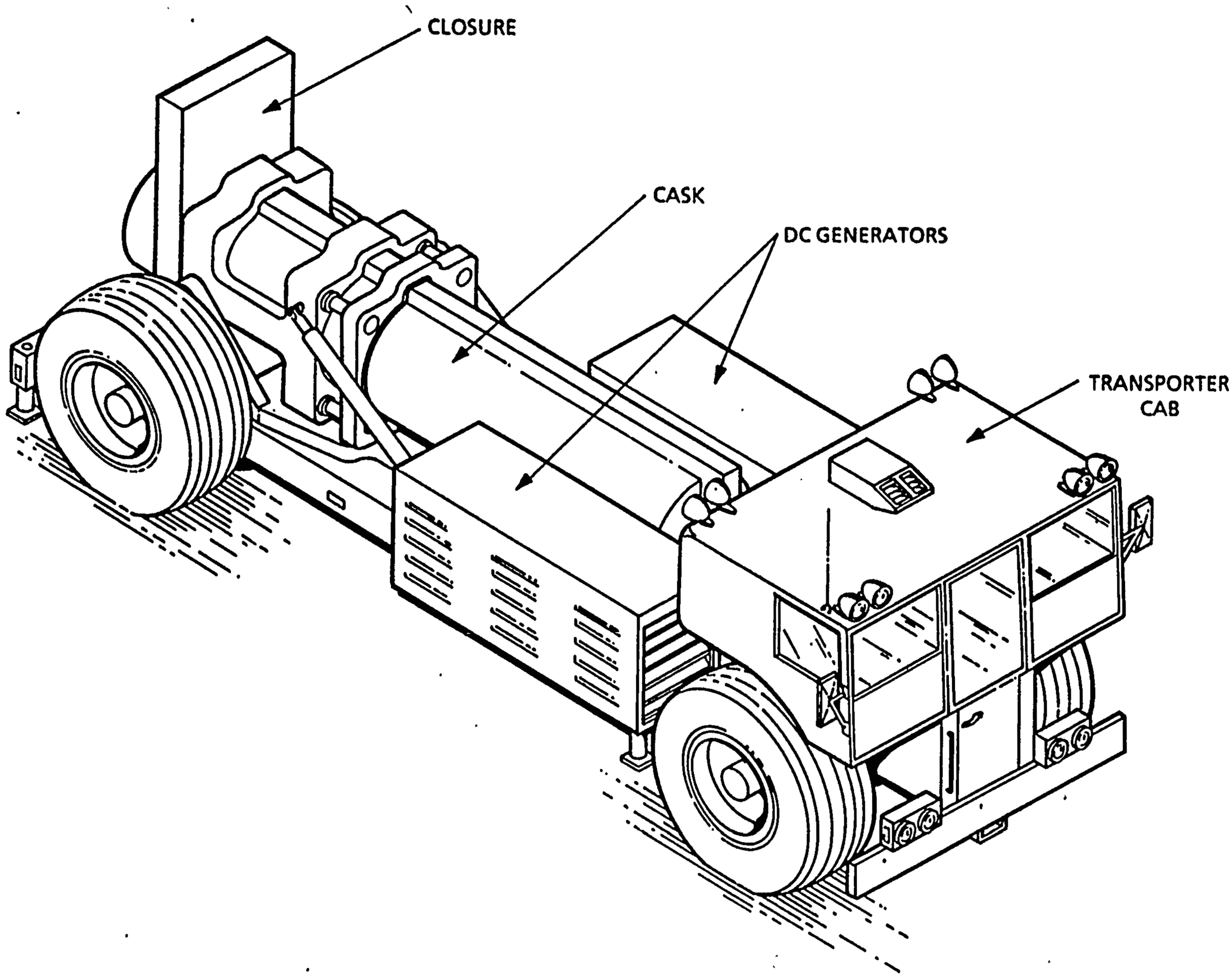

Figure 10.1.30. Waste Package Transporter in Transport Mode

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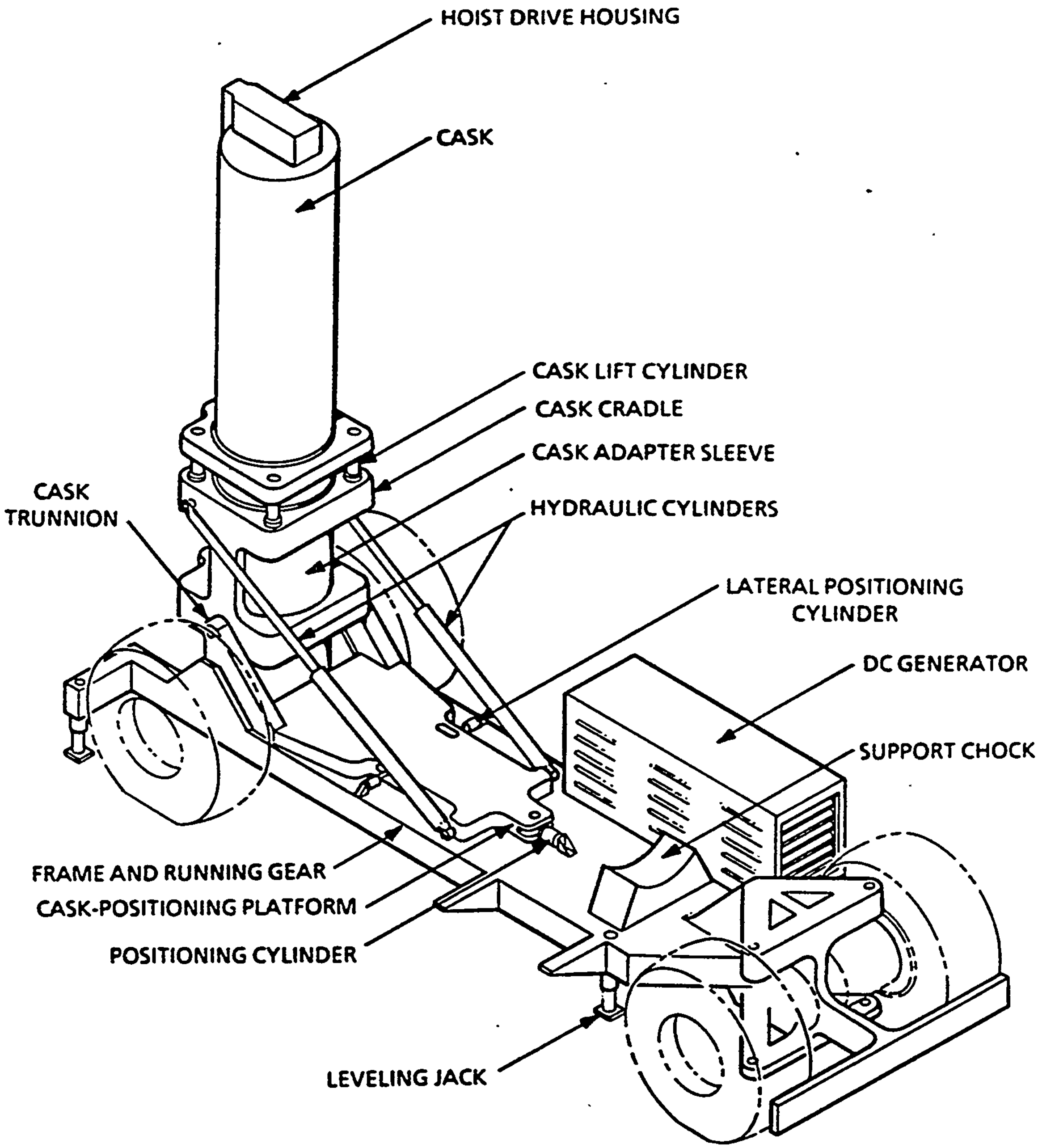

Figure 10.1.31. Waste Package Transporter in Emplacement Mode

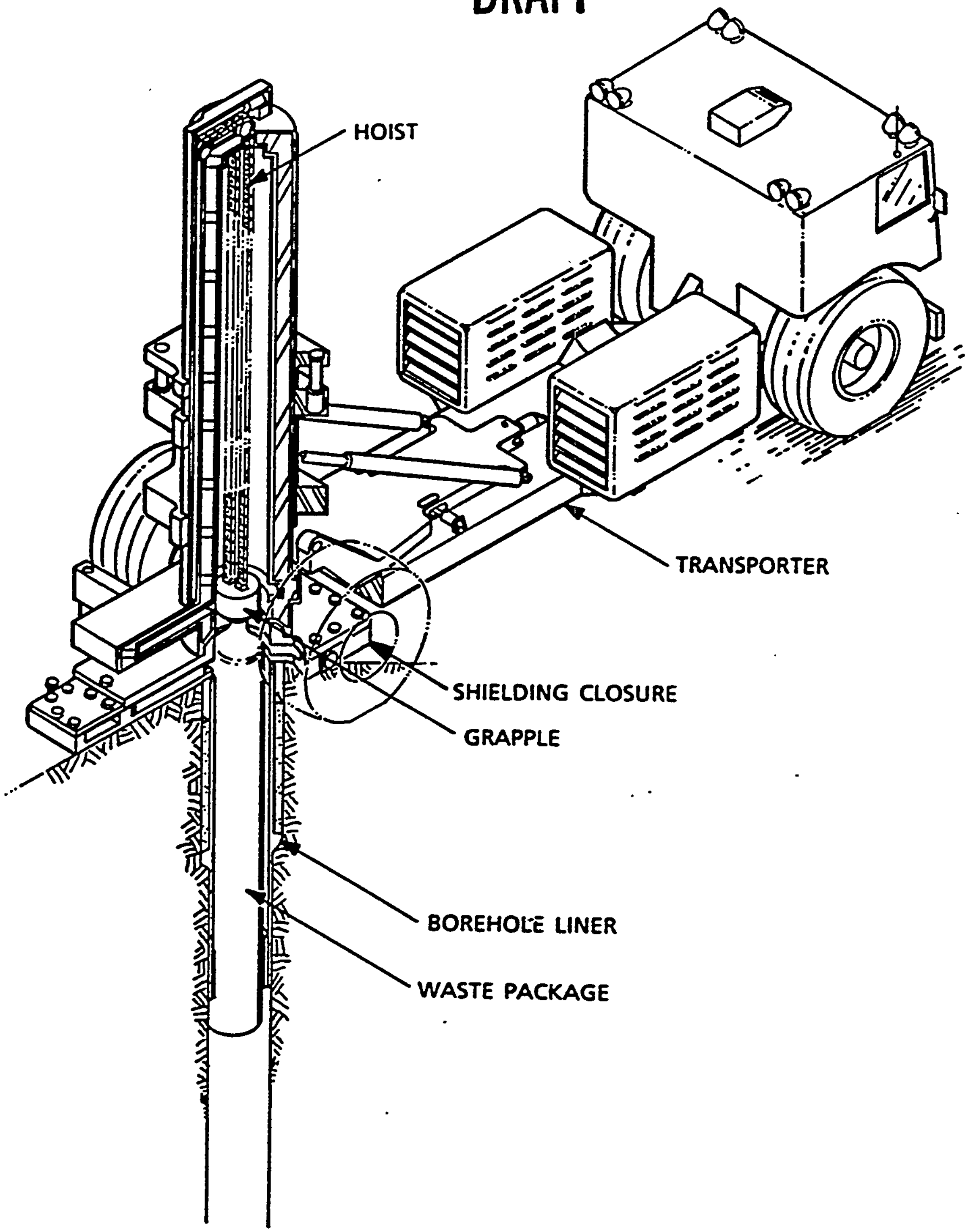

Figure 10.1.32. Vertical Emplacement Operations

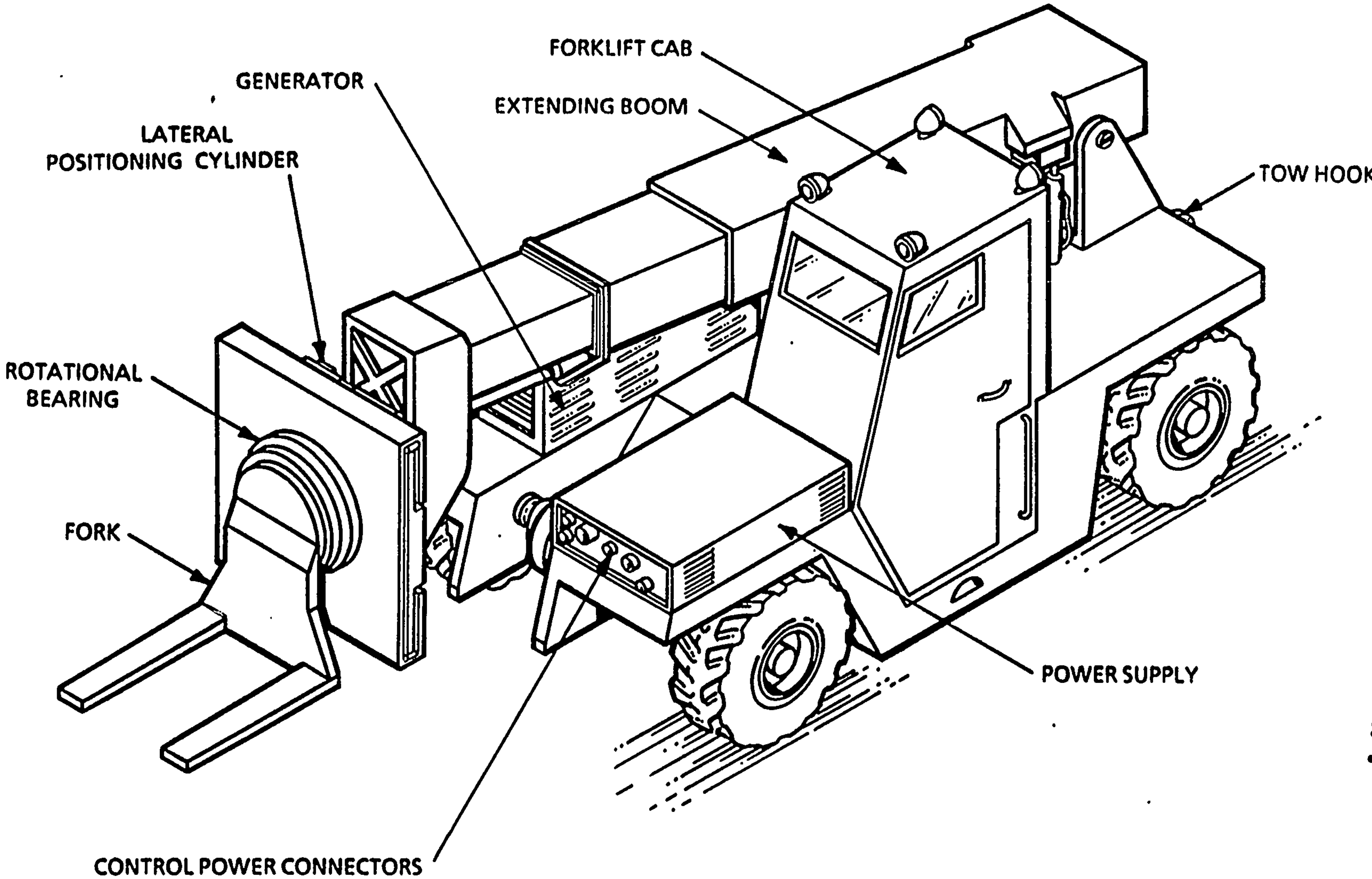

Figure 10.1.33. Modified Forklift

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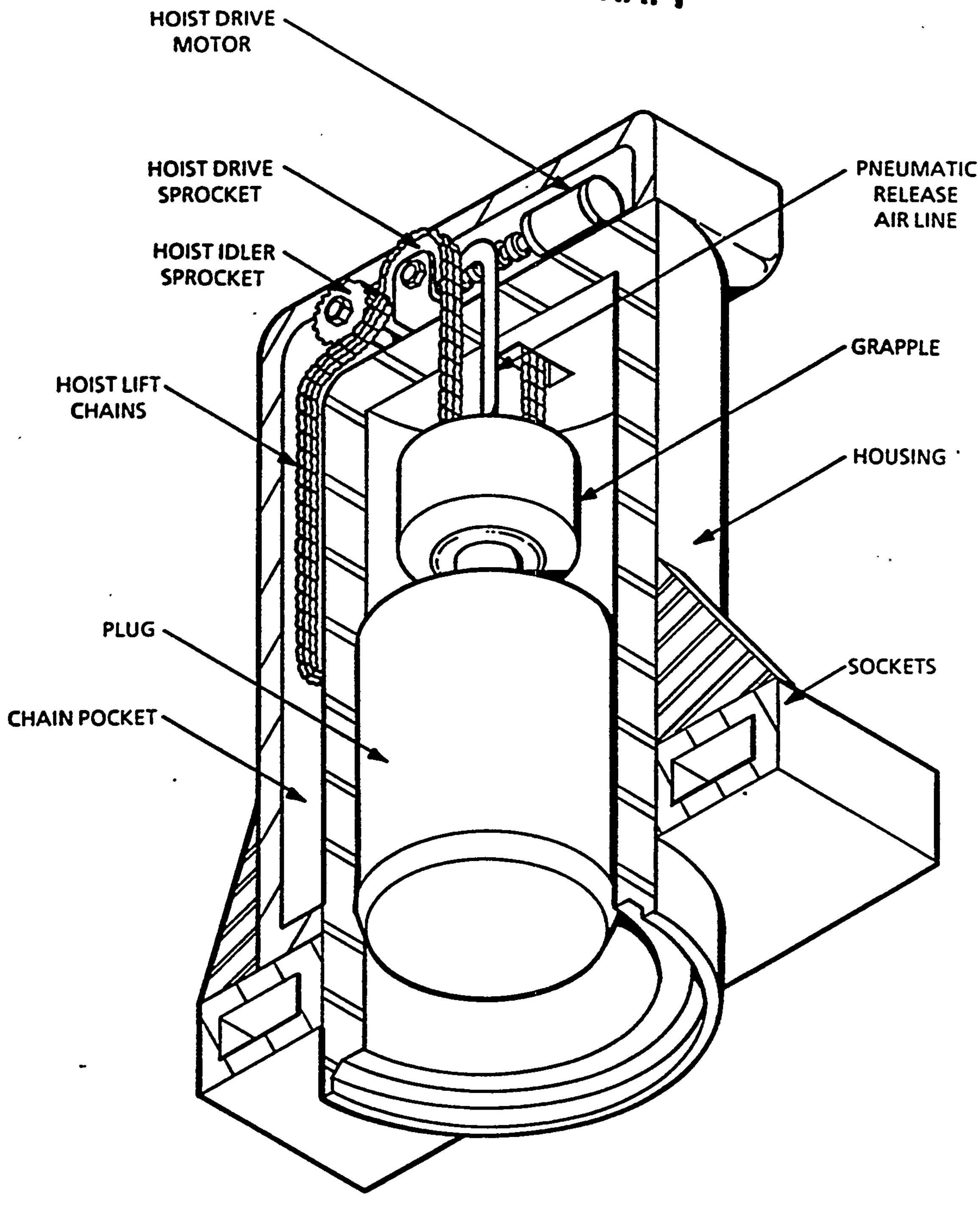
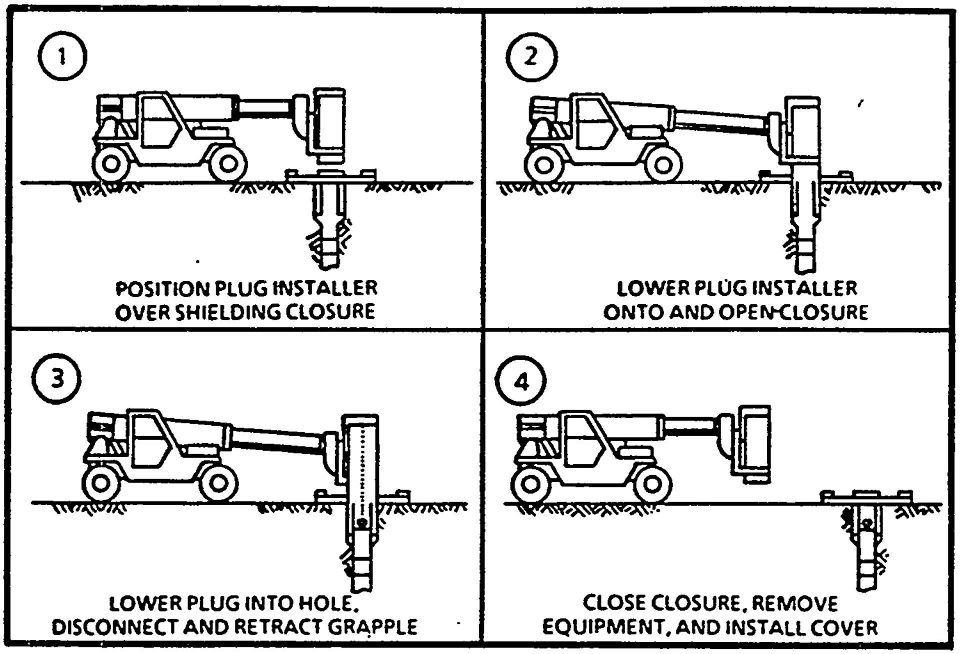
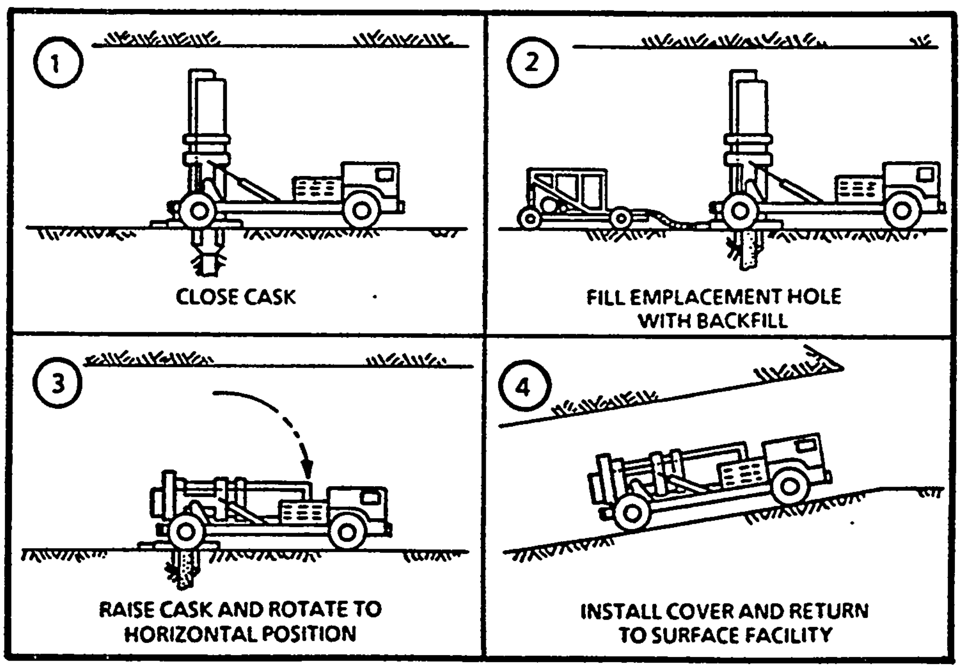


Figure 1Ø.1.34. Plug Installer/Remover



PLUG METHOD



BACKFILL METHOD

Figure 10.1.35. Plug and Backfill Installation

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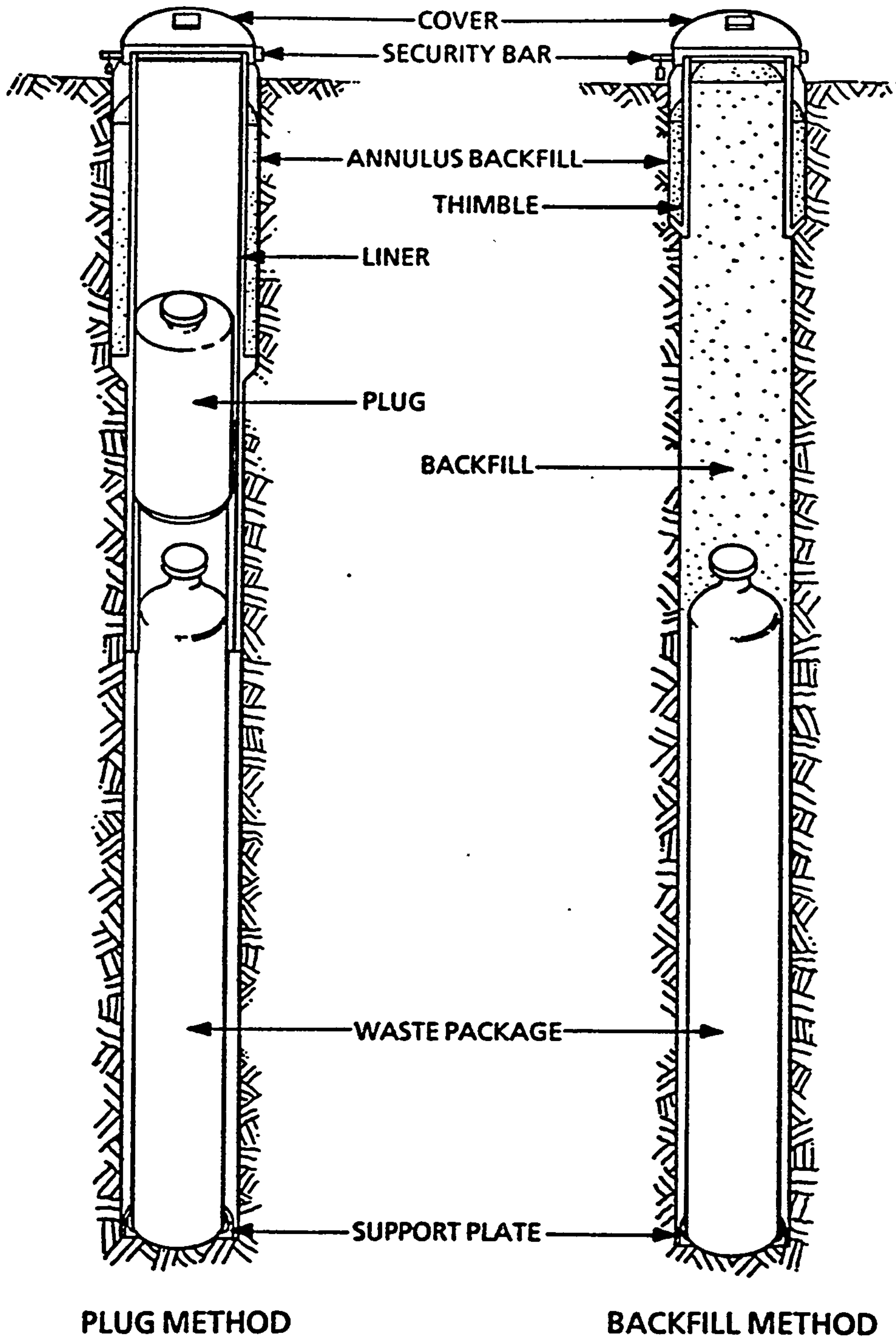

Figure 10.1.36. Plug and Backfill Methods with Emplaced Waste Package

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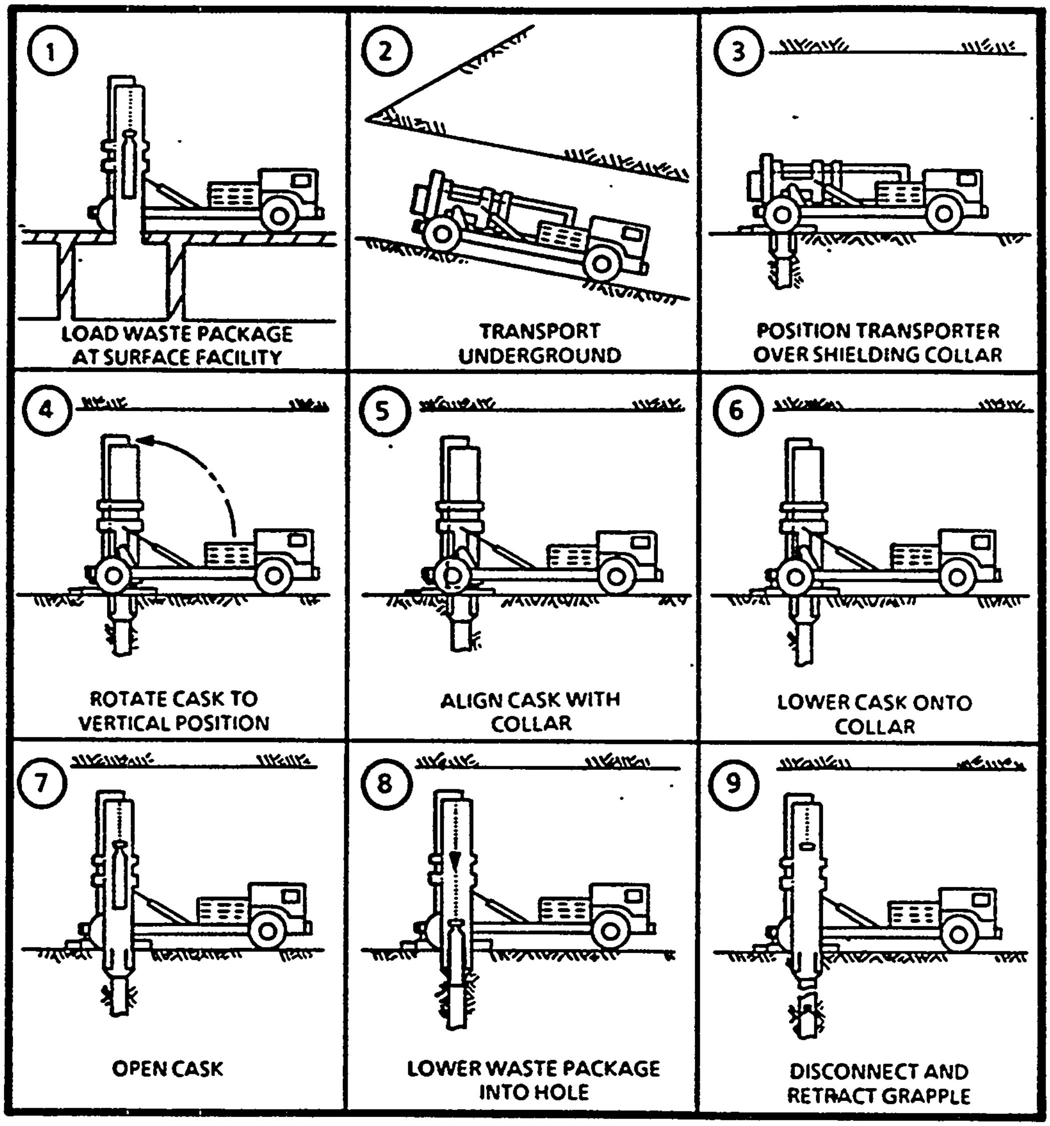

Figure 10.1.37. Vertical Emplacement for Backfill Method

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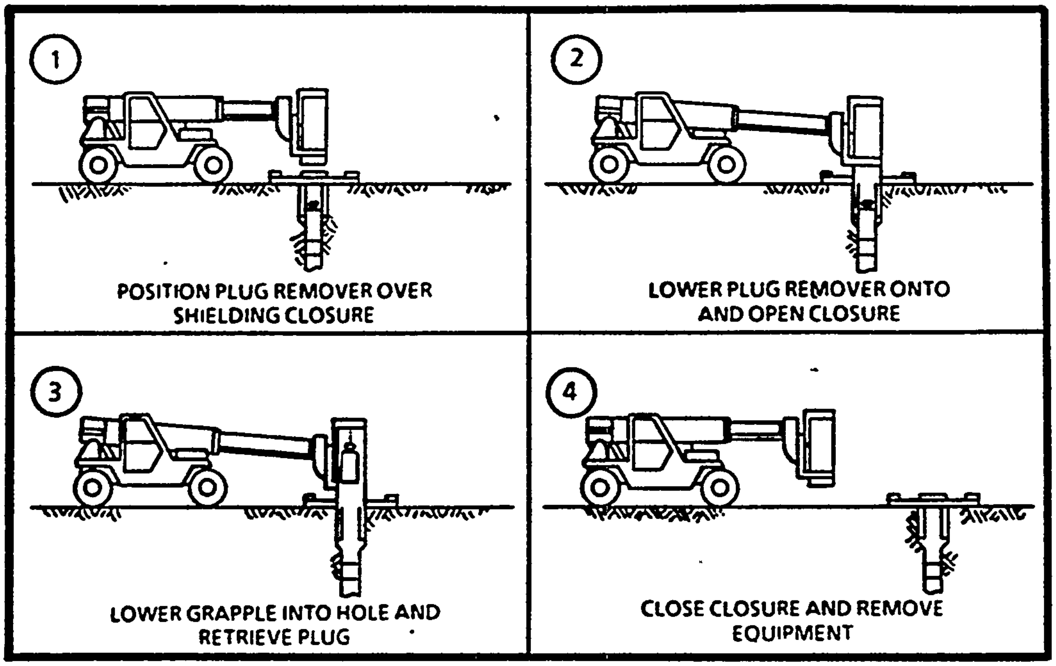

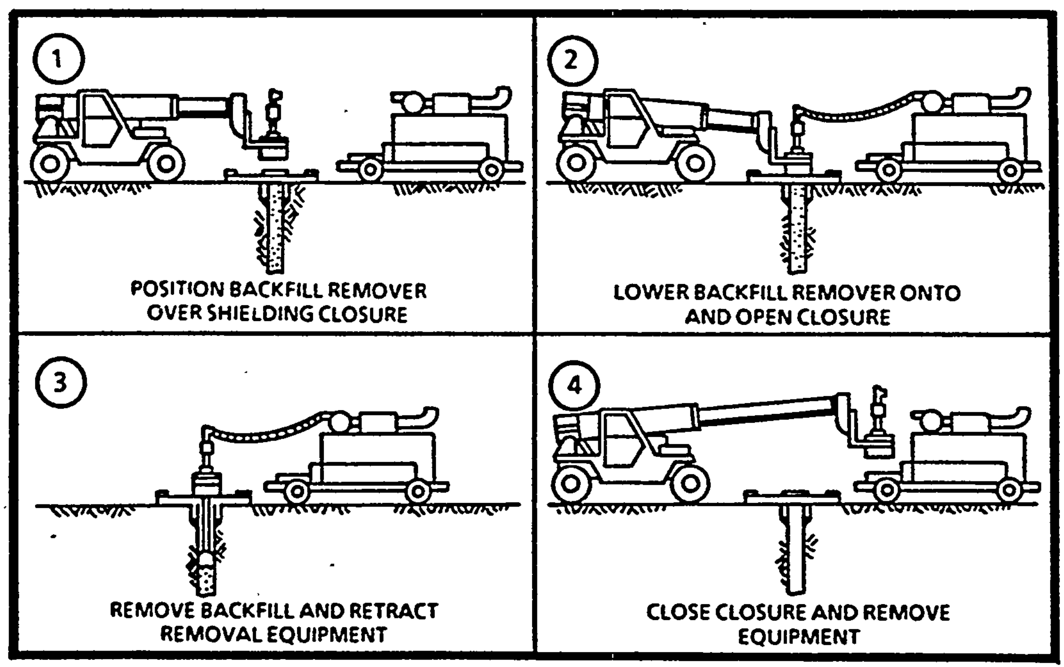

Figure 10.1.38. Plug and Backfill Removal

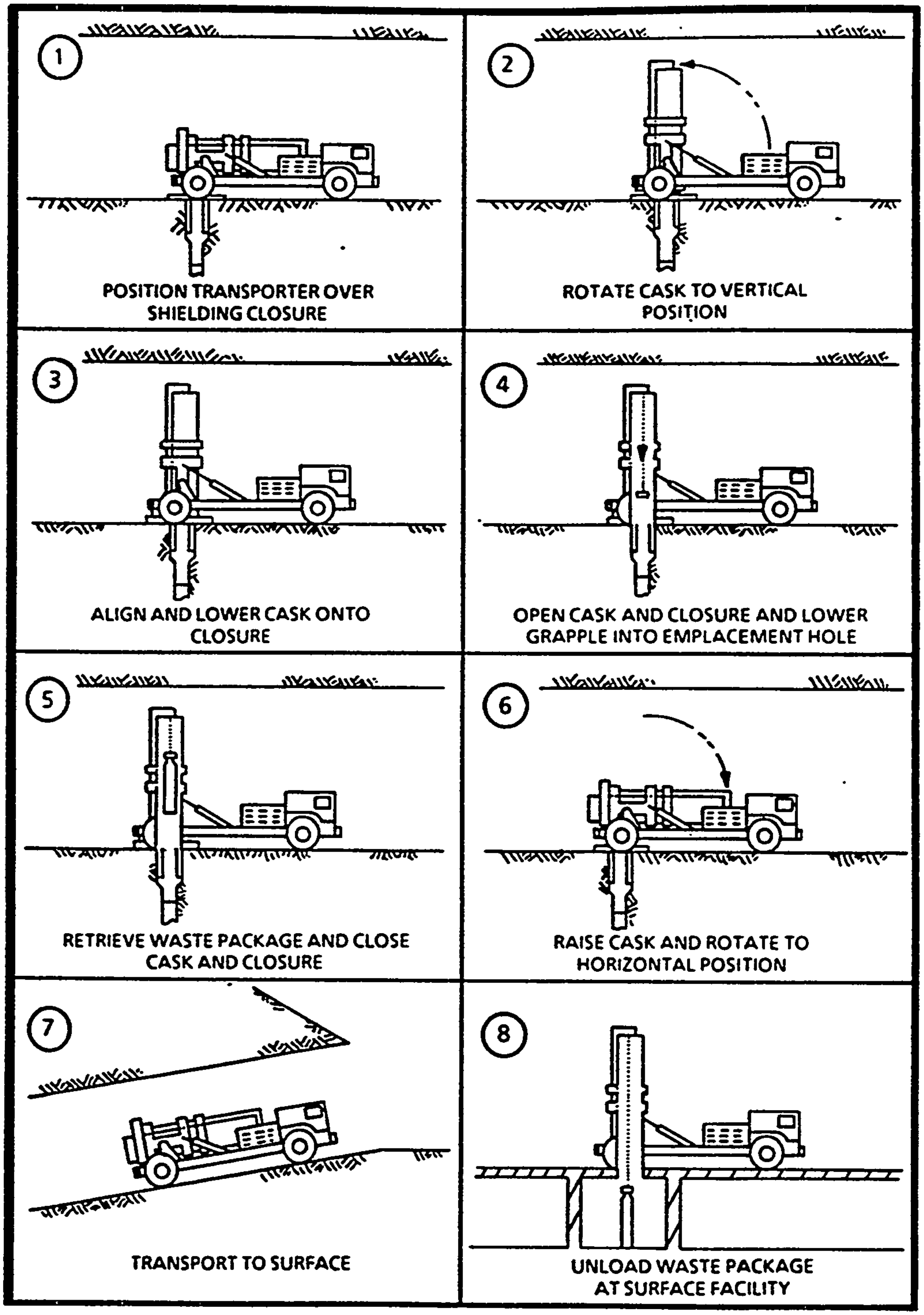

Figure 1Ø.1.39. Vertical Retrieval

10.1.6 Backfilling and Sealing

After the planned 22-year caretaker period during which retrievability must be ensured and after the performance confirmation program has been completed, the Department of Energy will apply to the Nuclear Regulatory Commission for permission to close the repository. After approval has been granted, decommissioning of the repository will begin. To decommission the underground facilities, all salvageable materials will be brought to surface. During closure, all underground access areas will be sealed and backfilled using multiple materials and techniques to ensure that the seals and backfill offer isolation properties equivalent to or better than the host rock. One of the materials under investigation as a backfill material is crushed tuff, which could be produced using the excavated material from the repository construction operations.

10.1.7 Summary

10.1.7.1 Advantages

1. The use of ramps or declines for access to the repository will have great advantages in the construction phase and during the emplacement operations.

2. Ramps provide a cost effective alternative since expensive hoisting installations would not be required.

3. Transportation of personnel, materials, and waste canisters will be easier and safer.

4. The ramps will provide a safer and quicker means of egress in case of emergencies.

10.1.7.2 Potential Problems

Potential problems exist with all repository concepts due to the limited understanding of the long-term behavior of the isolation system and host rock under repository conditions. However, extensive investigations are being conducted in most rocks, including tuff, to increase the understanding of the behavior of the waste disposal system components and ensure the suitability of the host rock formations.

11.0 OTHER WASTE DISPOSAL CONCEPTS

All national nuclear waste disposal programs currently favor disposal in mined geologic repositories. Other concepts have been studied by several countries and most attention has been directed towards subseabed disposal. Alternative concepts include very deep boreholes, disposal in outer space, deep well injection, rock melting, waste partitioning and transmutation, and ice sheet or island disposal. This chapter briefly describes these alternatives and although each has advantages and disadvantages, none have been shown to be superior to the mined repository from an economic, scientific or political point of view.

11.1 SUBSEABED DISPOSAL

Disposal of high-level wastes in the sediment in deep ocean floors is being investigated by several countries. Advantages of the method are considered to be the stable, uniform and isolated disposal environment and the tremendous capability of the ocean to disperse and dilute any radioactive releases. The disadvantages include the lack of monitoring capability of the disposal area, lack of knowledge about the marine transport properties, irretrievability of the waste packages, and complications of disposal in international waters. Two basic disposal methods are being investigated. One is the placement of waste canisters into boreholes drilled into the ocean floor. The other is the dynamic penetration of the seabed by a ballistically shaped waste package which is released several hundred meters above the ocean floor. The waste canister penetrates the ocean floor by freefall or artificial propellant to the required depth of around 20 to 30 m (65 to 100 ft). The borehole alternative would allow simpler canister design and deeper burial but is relatively expensive. Several canisters may also be placed into each hole. The penetration alternative is a simpler and more economic method but has disadvantages due to the limited penetration depth, lack of direct control over emplacement location, the need for complex canister shapes and the need to relocate the ship before each release.

11.2 VERY DEEP BOREHOLE DISPOSAL

The Very Deep Borehole concept requires a number of very deep holes drilled into a stable rock formation. Depths between 3,000 and 6,000 m (10,000 and 20,000 ft) are envisioned. Canisters would be lowered into these holes either in a controlled manner or by freefall. The concept is similar to the option described for the Danish disposal system. Canisters are stacked to within 3,000 m (10,000 ft) of the surface and the remainder of the borehole would be sealed with a combination of plugs and backfill of clay, cement and soil materials. Advantages of the method include the distant removal of the wastes from the biosphere and surface climatic changes, occupational safety since no personnel would be required to work in an underground environment, reduction of thermally induced gradients to the biosphere and minor effects on the existing thermal conditions by the heat production. Limitation on drilling technology such as small diameters and depth restrictions, high costs and lack of knowledge of the earth's interior are the main disadvantages.

11.3 SPACE DISPOSAL

This concept involves placing reprocessed wastes into solar orbit roughly half-way between the earth and the sun. The nuclear wastes would be launched into earth orbit aboard the space shuttle. A recoverable rocket would project the waste package towards the sun and a second rocket would establish the wastes in solar orbit where they would remain for at least a million years. This option has the great advantage of isolating the wastes not only from the biosphere but also from the entire planet and appear technically feasible. However, high costs and the technical risks associated with the launches and accidental reentry are major concerns with the method.

11.4 DEEP WELL INJECTION

This option is similar to the method developed for the disposal of oil field brines in the oil and gas industry. In the case of nuclear waste, one of two methods may be used. Liquid wastes may be injected into porous or fractured formations at depths of 1,000 to 5,000 m (3,280 to 16,400 ft). Alternatively, shale formations at depths of 300 to 500 m (890 to 1,640 ft) may be hydro-fractured and radionuclides suspended in clays and cements injected into the openings. Low cost and prior experience in the petroleum industry are the major advantages. Absence of multiple barriers, irretrievability of the wastes, lack of definition of the repository zone and potential reconcentration of the radionuclides represent major drawbacks to the method.

11.5 ROCK MELTING

In this concept a molten mass of nuclear waste is injected into a large cavity at the bottom of a shaft 2,000 to 3,000 m (6,500 to 10,000) ft deep. The heat generated by the wastes melts the surrounding rock and during cooling the wastes are embedded in a natural rock matrix which strongly resists leaching. This projected stability of the waste-rock matrix and the low mining costs are major advantages of the method but inherent uncertainties in the waste-rock interactions during loading, melting and cooling make this a doubtful alternative.

11.6 WASTE PARTITIONING AND TRANSMUTATION

These options represent waste treatment alternatives rather than disposal concepts and may be used in connection with other disposal schemes. Waste partitioning is the chemical separation of the waste constituents so that each may be disposed of individually. Transmutation is the radiation treatment through which some of the more dangerous radionuclides are rendered less toxic. Drawbacks of these alternatives are the extremely high costs and difficulties and risks associated with actinide separation.

11.7 ICE SHEET DISPOSAL

The large thicknesses and remoteness of polar ice regions have led to considerations of disposal of nuclear waste in ice sheets but the concept is not regarded favorably at the present time. Several emplacement options have been proposed. One involves the simple, free melting descent of a waste canister through the ice to the bedrock. Another option is the anchoring of the canister to limit the descent and make retrieval possible. A third option is surface storage but melting and additional accumulation of precipitation would also eventually bury the wastes. Transportation problems, uncertainties in ice behavior and international complications are some of the reasons this concept is not feasible at this time.

11.8 ISLAND DISPOSAL

In this concept the continental mined repository would be placed on a remote oceanic island. Remoteness, dilution and dispersion in the ocean, benefits of continental structures, presence of few economic resources and apparent simpler ground water hydrology are some of the benefits proposed. However, high costs and risks associated with waste transport, geohydrological uncertainties and international political considerations outweigh the benefits and hence this option is not under serious consideration.

12.0 SUMMARY

In summary, several general conclusions were noted during the investigations for this report. These are listed in the following.

1. There are three basic geologic high-level waste disposal concepts under consideration by the various countries researched. These concepts and their variations are as follows:

   a. Disposal in long boreholes drilled from the surface (Denmark, Italy).

   b. Disposal in mined tunnels, drifts or rooms:

      (1) Variation 1: Prepare tunnel with backfill before disposal of containers so that waste containers are completely surrounded by a backfill material such as bentonite (Switzerland).

      (2) Variation 2: Stacking waste canisters horizontally or vertically in tunnels and backfilling as disposal proceeds (France).

      (3) Variation 3: Disposal of the waste canisters with or without extended ventilation of the waste packages before backfilling and sealing the repository (France).

   c. Disposal in boreholes drilled from mined tunnels, drifts or rooms:

      (1) Variation 1: Horizontal boreholes for single or multiple waste packages (United States).

      (2) Variation 2: Inclined boreholes for single or multiple waste packages (Belgium).

      (3) Variation 3: Vertical boreholes for single or multiple waste packages (West Germany, Holland, Sweden, United States) arranged in single or multiple rows (Canada) in the floor of the disposal room.

2. Crystalline rocks, salt and clay are the most common rock types being considered as potential high-level waste repository host rocks. This is primarily because these are the most common rock types in most countries. Less commonly considered rocks are tuff, basalt, shales and metamorphic rocks. Most countries are, however, investigating all potential rock types available within its borders. Since the second repository program in the United States has been considering crystalline rocks as a potential host

rock for the second repository, the interest and results of investigations in crystalline rocks abroad will be of substantial benefit to the second repository program in addition to any site work carried out in the United States. In fact, the Department of Energy has been active in several international cooperative programs with Sweden, Switzerland, Canada, and others investigating the characteristics of crystalline rocks.

3. All countries are proposing to construct an in situ underground research laboratory for site characterization, to demonstrate the suitability of the host rock, and to demonstrate repository construction, emplacement and sealing technology.

4. Disposal of high-level waste will not take place in any for at least another decade or more and none have therefore completed a final design for a repository.

5. Machine excavation of shafts and tunnels is preferred, where rock properties permit, to reduce the damaged zone around the repository excavations. Most countries have conducted preliminary conceptual investigations into repository construction and operation and all have concluded that the technology and experience to safely construct and operate a repository is currently available.

6. The presence of large aquifers above the potential repository formations (e.g., West Germany, United States) is not considered a technical drawback since the shaft sinking and lining technology (such as freezing) to penetrate and seal shafts in these types of ground conditions have been demonstrated at numerous locations throughout the world.

7. Most countries consider long term storage (up to 50 years or more) of the spent fuel or high-level waste as a favorable option and are planning to integrate a long term storage facility into their waste disposal program. However, West Germany and the United States are considering shorter times (minimum of 10 and 5 years, respectively).

8. Reprocessing of the spent fuel is the preferred option in most countries. However, they are also investigating direct disposal of spent fuel as an option due to political and environmental considerations.

9. The disposal concepts show a wide range in the degree to which reliance is placed on either the waste package or the geologic setting to contain and isolate the waste. In the Swedish concept, prime reliance is placed on the waste package because of uncertainty, caused by lack of data, as to the performance of the geologic setting. On the other hand, in both the German and Belgian concepts for the disposal of high-level waste, prime reliance is based on the fact that the waste will be encapsulated

by the rapid creep of an extremely impermeable host medium. Disposal concepts by other nations fall in between these extremes.

10. In general it appears that horizontal or inclined boreholes are preferred for waste emplacement in bedded media while vertical boreholes are preferred in massive rock formations. While in-room or in-drift emplacement concepts have been adopted only for massive domal salt and crystalline rock formations, the basis for selection of the concept does not appear to be related to the thickness of the rock formation but to other factors such as the simplicity of emplacement or the potential to minimize drift diameters. In the absence of shielding by the host rock for a waste package emplaced in a borehole, in-drift disposal packages are provided with shielding to facilitate their safe handling.

11. Bentonite packing and backfill is a design feature for all waste packages emplaced in saturated, non-plastic media. Sodium bentonite, with or without sand or crushed rock, is the preferred type because of its swelling properties.

Table 12.1.1 summarizes the high-level waste disposal concepts for most countries with a nuclear power program.

Table 12.1.1. Summary of High-Level Nuclear Waste Programs.
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COUNTRY	NO. OPERAT'G REACTORS	SPENT FUEL @ YEAR 2000 (Metric tU)	DISPOSAL METHOD (WASTE TREATMENT)	ROCK TYPE	INTERIM STORAGE
Argentina	2	5,800	Reprocess, Dispose of vitrified wastes in mined repository Disposal Options: - not defined	Cryst.	AR, AFR 20 years
Australia	0	0	R & D, HLW immobilization, Analog studies	-	-
Austria	0	0	R & D, HLW immobilization,	Cryst.	-
Belgium	7	3,000	Reprocess, Dispose of vitrified wastes in mined repository, Disposal Options: - Inclined boreholes	Clay	AR, AFR 50 years
Brazil	1	1,000	Not defined, Reprocess	-	-
Canada	17	33,900	Dispose spent fuel or HLW in mined, geol. repository, Disposal Options: - Boreholes in floor of disposal rooms	Cryst.	AR 50 years
Finland	4	900	Dispose spent fuel or HLW in mined repository Disposal Options: - Boreholes in floor of disposal rooms - Tunnel disposal	Cryst.	AR, AFT 40 years

(AR = At-Reactor; AFR = Away-From-Reactor)

Table 12.1.1. Summary of High-Level Nuclear Waste Programs
(Page 2 of 4)

COUNTRY	NO. OPERAT'G REACTORS	SPENT FUEL @ YEAR 2000 (Metric tU)	DISPOSAL METHOD (WASTE TREATMENT)	ROCK TYPE	INTERIM STORAGE
France	48	20,000	Reprocess, Store vitrified HLW indefinitely, Dispose in mined repository, Disposal Options: - Boreholes in floor of disposal rooms - Tunnel disposal - Disposal with or	Granite, Salt, Clay, Shist	AR, AFR Indef.
W. Germany	21	11,000	Reprocess, Dispose of vitrified HLW in mined repository, Also, study spent fuel disposal, Disposal Options: - Deep boreholes in floor of disposal rooms	Salt Dome	AR, AFR min. 10 years
India	6	5,000	Reprocess, Dispose of HLW in mined repository Disposal Options: - not defined	Granite Shist	AR, AFR min. 30 years
Italy	3	2,000	Reprocess, Dispose of vitrified HLW in mined repository, Disposal Options: - Deep boreholes from surface - Boreholes in floor of disposal rooms	Clay	AR, AFR 50 years
Japan	33	20,500	Reprocess, Dispose of vitrified HLW in mined repository, Disposal Options: - Boreholes in floor of disposal rooms	Granite Tuff Diabase	AR, AFR min. 30 years

(AR = At-Reactor; AFR = Away-From-Reactor)

Table 12.1.1. Summary of High-Level Nuclear Waste Programs (Page 3 of 4)

COUNTRY	NO. OPERAT'G REACTORS	SPENT FUEL @ YEAR 2000 (Metric tU)	DISPOSAL METHOD (WASTE TREATMENT)	ROCK TYPE	INTERIM STORAGE
S. Korea	6	4,440	Undefined		
Holland	2	420	Interim storage until site is found, Also reprocessing abroad with HLW disposal in Holland, Disposal at foreign sites also under investigation, Disposal Options: - Deep boreholes in floor of disposal rooms	Salt Dome	AFR decades
Pakistan	1	440	Undefined		
S. Africa	2	714	Reprocess, Dispose of vitrified HLW in mined repository, Disposal Options: - Undecided	Granite	AR Decades
Spain	8	3,400	Store spent fuel for 10 years, move to AFR at geologic repository site until repository is ready, Disposal Options: - Undecided	Granite, Salt Dome	AR for 10 years, AFR for decades
Sweden	12	7,800	Store spent fuel at AFR facility, Enclose spent fuel in copper canister, Disposal in mined repository, Disposal Options: - Boreholes in floor of disposal tunnels CLAB AFR storage facility ready since 1985	Cryst.	AR for min. 1/2 year, AFR for min. of 40 years.

(AR = At-Reactor; AFR = Away-From-Reactor)

Table 12.1.1. Summary of High-Level Nuclear Waste Programs
(Page 4 of 4)

COUNTRY	NO. OPERAT'G REACTORS	SPENT FUEL @ YEAR 2000 (Metric tU)	DISPOSAL METHOD (WASTE TREATMENT)	ROCK TYPE	INTERIM STORAGE
Swiss	5	2,000	Reprocess, Dispose of vitrified HLW in mined repository, Disposal Options: - Tunnel disposal - Tunnel disposal of spent fuel	Cryst.	AR, AFR for 40 years
U.K.	38	3,000	Reprocess, Store vitrified HLW until repository solution is found, Siting decisions deferred after some investigations, Disposal Options: - Deferred	Cryst., Salt Clay	AR, AFR for at least 50 years
U.S.	97	40,100	Direct disposal of spent fuel in mined repository, Disposal Options: - Two phased repositories - Boreholes in floor or wall of disposal tunnel - Tunnel disposal	Tuff, Salt, Basalt, Cryst., Sedimen. Metamor.	AR, AFR min. 10 years or until repository ready
U.S.S.R.	47	NA	Reprocess, Dispose of vitrified HLW in mined repository, Disposal Options: - Unknown	NA	AR, AFR

(AR = At-Reactor; AFR = Away-From-Reactor)

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