Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste

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Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste

III Facilities

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1 INTRODUCTION

The KBS project was initiated in response to the requirements of the "Conditions Act", concerning the safety accountability of nuclear power facilities and was commenced in early 1977. The objective has been to describe a safe way of handling and storing high-level radioactive waste. In order that current schedules for the fueling of nuclear power plants under construction could be met, it was necessary for KBS to submit an initial report in late 1977.

This means that KBS had approximately 9 months to prepare the present report. In this short period of time, it has not been possible to study all alternative solutions in detail. Instead, it has been necessary to make an early selection of those approaches which were deemed to have the best potential for achieving satisfactory results from the point of view of safety. Nor has there been enough time for cost calculations with satisfactory accurancy. The design presented here does therefore not make any claims on technical-economical optimality; it is presented merely as one possible alternative.

The facilities which are described here are designed for the handling and storage of vitrified high-level waste. Corresponding facilities for spent nuclear fuel will be described in a later report. The status of the work for this later report is described in Appendix 1 to Volume I.

The construction and operation of the described facilities is based on technology for which experience is available from previous nuclear installations or from other fields.

Certain parts of the report are based on work done outside of KBS. Thus, the preliminary study on transportation systems and a central fuel storage facility conducted by PRAV (Swedish National Council for Radioactive Waste Management) comprises the basis for chapters III:2 and III:3. The continued work on facilities design and the preparation of a siting application for the central fuel storage facility is being pursued within SKBF (Swedish Nuclear Fuel Supplies Company). The design of the central fuel storage facility as presented in the preliminary study and this report may therefore be subject to revision. The intermediate fuel storage facility for waste cylinders described in chapter III:5 is modelled after a similar plant at Marcoule in France, whose designers - St Gobain Technique Nouvelle - have collaborated in the KBS project.

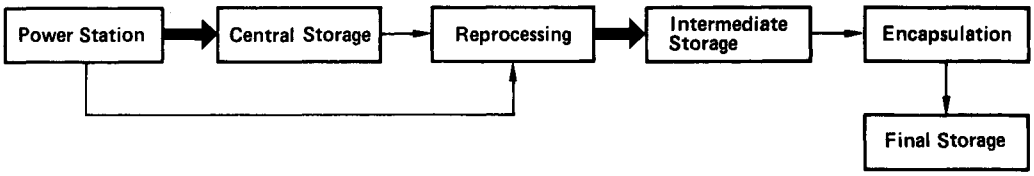
As in the AKA Committee (Government Committee on Radioactive Waste) study it is assumed that the final disposal of the high-level waste will be accomplished by deposition in crystalline bedrock possessing the appropriate properties. The calculations and evaluations in this report are based on a repository depth of 500 m.

The facilities described here have been designed for a total waste capacity corresponding to approximately 9 000 metric tons of uranium - which is the quantity which would be produced by 13 reactors operating over a period of 30 years. It should be emphasized that the technical solutions which are presented here would not be significantly altered by an increase or decrease of this figure.

The schedule for continued study work and construction of the facilities is presented in chapter I:14. This schedule includes the following main dates.

1987	Choice of site for intermediate storage facility
1990	Intermediate storage facility completed and ready for reception of vitrified waste
2000	Choice of site for final repository (incl. encapsulation station)
2020	Encapsulation station and final repository completed and ready for reception of vitrified waste.

2 TRANSPORTATION SYSTEMS



2.1 STANDARDS AND GOVERNMENT REGULATIONS

Relevant portions of the IAEA's "Regulation for the Safe Transport of Radioactive Materials" must be observed in connection with the transportation of spent nuclear fuel and other radioactive material /2-1/.

Both the spent fuel and the vitrified high-level waste contain so much radioactivity that they must be transported in containers which meet international requirements. The requirements which are applicable here are the IAEA regulations for type B packaging, which are described in greater detail in section 2.4.3.

Every planned transport must be preregistered with the Nuclear Power Inspectorate, including specification of identification data for the selected fuel elements and a preliminary time-table for the transport. Administrative routines for this work will be established by the Nuclear Power Inspectorate before the transportation system is put into operation. Physical protection of the transports will also be arranged in accordance with the directives of the Nuclear Power Inspectorate.

2.2 GENERAL PRINCIPLES, FLOW CHART

Figure 2-1 shows in the form of a flow chart the various transports involved in the back end of the fuel cycle. Solid lines indicate the transports which require type B packaging while broken lines indicate internal transports where type B packaging is not required.

The Swedish nuclear power plants are situated on the coast and have their own harbours. It is assumed that the central fuel storage facility and the intermediate storage facility will also be located near harbours. The transports will therefore be accomplished primarily by sea, with only short road transports to and from the harbours.

Transport stages A, D, F and G (Fig. 2-1) are thus road transports where the transport cask is carried by a trailer. This type of transport has already been practised in Sweden at, for example, the Oskarshamn plant in connection with the shipping of spent fuel to the English reprocessing plant at Windscale (Fig. 2-2).

When the transport cask is loaded onto the ship for shipment to a

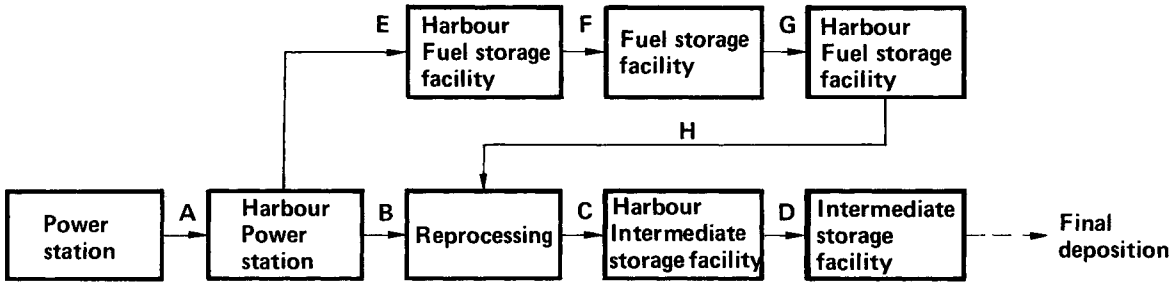


Figure 2-1. Flow chart of transportation within the back end of the fuel cycle. B packaging is required for transport stages A–H. Transports A, D, F and G go by road.

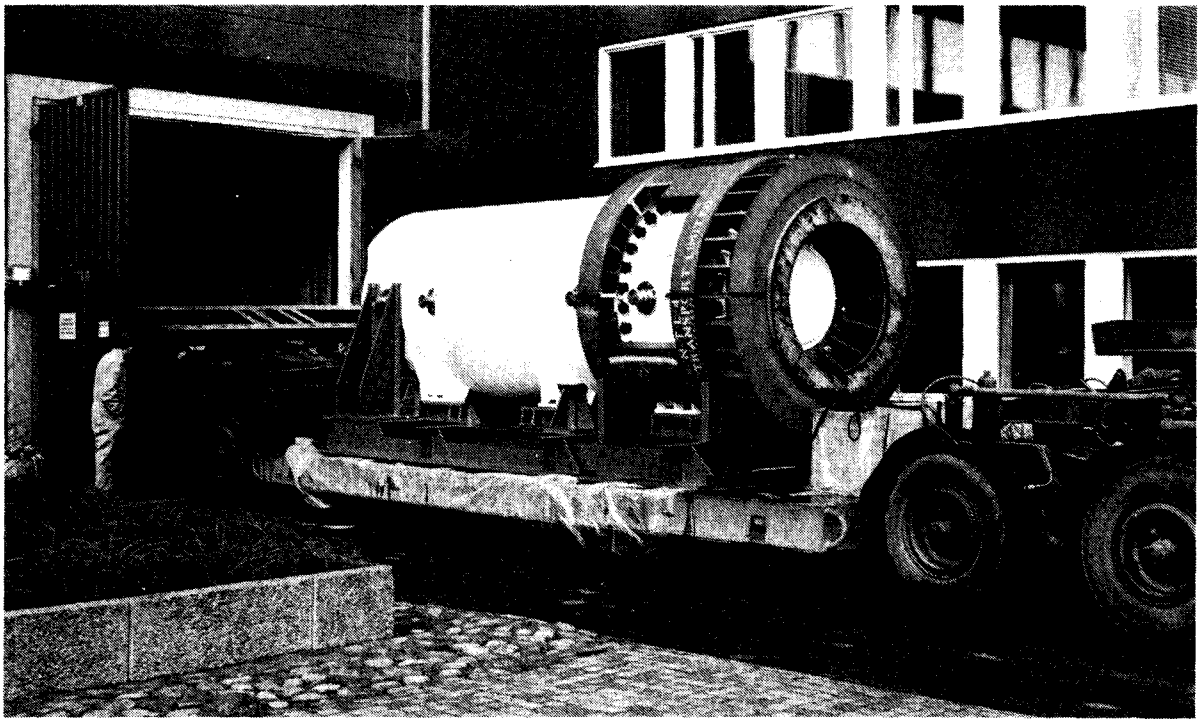


Figure 2-2. Trailer and transport cask (B packaging) outside of Oskarshamn plant. This equipment has been used to transport spent fuel to the harbour for further shipment to the reprocessing plant at Windscale.

reprocessing plant, responsibility for the transport cask is assumed by the reprocessing company's transportation organization. The Swedish transportation organization assumes responsibility only when the casks containing the vitrified high-level waste are unloaded in a Swedish harbour (transport stages B, C and H).

Shipments of spent fuel from the various power plants to the central fuel storage facility by sea are handled by the Swedish transportation organization (transport stage E).

2.3 EXISTING EQUIPMENT AND SYSTEMS

The transportation of spent nuclear fuel imposes special requirements on the transport equipment and on the supervision of the transports. The fuel contains fission products which make it highly radioactive. It also generates heat. Since the nuclear fuel contains fissionable material as well, the risk of criticality must be taken into consideration. This imposes special demands on the design of a transport cask. A large cooling surface area is required for effective heat dissipation. At the same time, the surfaces of the cask should be smooth in order to facilitate cleaning. The radiation shield must incorporate a material of high density in order to shield gamma radiation, but also a material of low density in order to shield neutron radiation.

European transport casks currently in use weigh between 30 and 70 metric tons and can transport between 1 and 2.5 tons of nuclear fuel. They are of French, German or English design. These three countries currently operate a joint company called Nuclear Transport Limited (NTL) which dominates the European market.

During the period 1966-1977, some 700 metric tons of spent nuclear fuel have been transported from light-water reactors to various European reprocessing plants. In the beginning, only relatively low burnup fuel was transported, while in recent years high burnup fuel (30 000 MWd/t) has been transported after only 6-9 months of cooling time at the reactor. Fuel has been transported to the following plants through 1976:

- WAK, Karlsruhe, West Germany 85 tons
- La Hague, France 255 tons
- Windscale, England 270 tons
- Eurochemic, Mol, Belgium 90 tons

Transport casks with a maximum weight of 40 metric tons are normally carried on the public road network, while transport casks of higher weight are shipped by rail. The transportation of spent nuclear fuel from Italy, Spain, West Germany, Holland and Sweden to the English reprocessing plant at Windscale has been done by boat.

Table 2-1. Western European transport casks for spent nuclear fuel.

Type	NTL 2	NTL 3	NTL 4	NTL 5	NTL 8	NTL 9	NTL 10	NTL 11	NTL 12	NTL 14	Exl 13/3A	Exl 14
Government approval No.	F 59	1106	1132	1124		F 136A		1146		1154	1126/1141	1147
Government approval (Present status)	Lic.	Lic.	Lic.	Lic.	Lic.	Lic.	Pending	Lic.	Pending	Lic.	Lic.	Pending
Owner	NTL	NTL	NTL	NTL	NTL	NTL	NTL	NTL	NTL	NTL	BNFL	BNFL
Capacity Fuel elements/mm PWR BWR	 4/200 9/114	 7/200 -	 7/200 19/114	 7/200 12/140	 3/125 -	 - 7/140	 12/230 -	 7/215 17/140	 12/215 30/140	 5/230 -	 5/215 14/140	 5/215 14/140
Thermal capacity kW	15	30	35	35	35	25	100	42	100	50	30	40
Weight (metric tons)	32	52	65	69	36	34	104	75	95	82	72/72.5	100
Payload (metric tons of uranium)	1.1	2.0	2.3	2.3	1.4	1.4	6.2	3.3	5.7	2.7	2.7	2.7
Cavity length (mm) diam. (mm)	3875 440	3380 864	4370 864	4675 864	4280 3x230	4520 474	5050 1220	4630 914	4580 1220	5160 914	4674/4776 864	4887 914
Coolant	Air	Water	Water	Water	Air	Air	Air (water)	Water	Air (water)	Water	Water	Water
Means of transport	Road	Rail	Rail	Rail	Road	Road	Rail	Rail/ sea	Rail/ sea	Rail/ sea	Rail/ sea	Rail/ sea
Number of casks in operation	2	3	1	1	2	2	-	-	-	-	3/4	-
Number of casks in production or on order	-	1	-	-	-	-	1	5	1	2	-	-

Table 2-2. American transport casks for spent nuclear fuel.

Type	NFS-4	NAC 1/2	TN-8	TN-9	IF-300	NLI 10/24
Government approval No.	6698	9010	9015	9016	9001	9023
Government approval (present status)	Licensed	Licensed	Licensed	Licensed	Licensed	Licensed
Owner	NFS NAC	NLI	TNY	TNY	GE	NLI
Capacity PWR (fuel elements/mm) BWR (" " ")	 1/215 2/140	 1/215 2/140	 3/215 -	 - 7/140	 7/215 18/140	 10/215 24/140
Thermal capacity, kW	11	11	35	25	61	70
Weight (metric tons)	22	22	36	34	64	91
Payload (metric tons of uranium)	0.5	0.5	1.4	1.4	3.5	4.7
Cavity length (mm) Cavity diameter (mm)	4521 343	4521 321	4280 3x230	4520 474	4578 953	4559 1143
Coolant	Water	Air	Air	Air	Water	Air/He
Means of transport	Road	Road	Road	Road	Rail	Rail
Number of casks in operation	6	3	-	-	4	-
Number of casks in production or on order	-	-	2	3	-	4

The trend is towards increasingly large transport casks. Transport casks are now being planned with a weight of 100 tons and a capacity of 6 tons nuclear fuel. Such a transport cask is expected to be in operation some time in 1978.

The current situation with regard to available and planned transport casks in western Europe is presented in table 2-1. The situation on the American market is presented in Table 2-2. As is shown in the tables, western Europe (NTL) currently occupies a leading position as a supplier of transport casks for spent nuclear fuel.

2.4 DESIGN OF A SWEDISH TRANSPORTATION SYSTEM

2.4.1 General

In parallel with the conceptual study on the central fuel storage facility, SKBF is examining various alternatives for securing a reliable supply of transport resources within Sweden.

Swedish transportation needs have been studied for the period 1976-1991. Annual discharges of fuel elements expressed in tons of uranium are reported in chapter I:2. These quantities are based on the six units now in operation and on continued expansion to thirteen units.

In 1976, discussions were initiated with European and American organizations which work with the transportation of spent nuclear fuel for the purpose of exploring the possibilities of procuring transport casks.

Nuclear Transport Limited (NTL-Europe) currently seems to be the leading company in this field. In recent years, NTL has carried out hundreds of transports in Europe to such destinations as Windscale and La Hague. The types of transport casks used by NTL are well-suited to Swedish transportation requirements.

The American consultancy company Nuclear Assurance Corporation (NAC) has designed four transport casks, designated NAC-1, which are currently in routine operation in the USA. NAC is currently designing a transport cask with a capacity of max. 3 tons of nuclear fuel. This cask is also well-suited to Swedish requirements.

SKBF is currently awaiting further developments on the transportation side. One of the reasons for this is that COGEMA announced in July of 1977 that they plan to enter the nuclear fuel transportation field. It its important that any transportation system which is adopted be compatible with any existing standard European system.

2.4.2 Capacity considerations

Capacity considerations are based only on the need for transports within Sweden. Transports of spent nuclear fuel to foreign reprocessing plants and return transports of vitrified high-level

waste to Sweden are anticipated to be included in the commitments of the reprocessing company.

In calculating the annual transport volume to the central storage facility, various alternatives have been studied. The required number of transport casks and the annual number of sea transports will depend on the number of reactors in operation and on the following factors:

- Location of the central storage facility for spent nuclear fuel.

  Of the three studied sites for the central storage facility, two - Forsmark and Oskarshamn - are located in connection with nuclear power plants, which means that nuclear fuel from the reactors at these sites will be transported directly by trailer to the central facility. In the case of Studsvik, all transports will arrive by sea, which means more shipments, requiring more transport casks, since the average cycle time per cask will be longer.

- Reception capacity of the central storage facility for spent nuclear fuel.

  The reception capacity of the central fuel storage facility will depend on how many casks can be handled simultaneously in the receiving section, the average receiving time per cask, the amount of space available to accomodate casks and equipment and the size of the personnel force in the receiving section. On the basis of the proposed design of the facility, it is estimated that an average reception capacity of one cask per day could be achieved.

In determining the capacity of the transportation system, the accumulated quantity of fuel which is stored at the nuclear power plants when the central storage facility is commissioned must be taken into account. In addition to an annual fuel quantity corresponding to the volume of fuel discharged from each reactor, the amount of fuel to be transported will also include this accumulated quantity, which must be transferred to the central fuel storage facility as soon as possible.

The annual discharge volume at equilibrium after expansion to 13 reactors will be approximately 1 400 fuel elements per year, corresponding to approximately 300 metric tons of uranium per year. A transport cask such as NTL 11,TN 17 or the equivalent (see 2.4.3) can transport max. 3 tons of nuclear fuel. When equilibrium has been achieved, i.e. after the fuel accumulated at the nuclear power plants has been transferred to the central storage facility, the annual number of cask transports will be approximately 100. 6-8 casks will be required for this volume. The transport distance for the sea transports from the nuclear power plants to the three studied sites for the central fuel storage facility (Forsmark, Oskarshamn and Studsvik) vary from 200 to 1 100 km.

A single shipload may consist of 4-8 transport casks. On the average, it is assumed that 24 hours will be required for all handling of each cask at the nuclear power plant. There will be a certain amount of overlap so that one cask will be transported

into the station and prepared for handling before the immediately preceding cask is returned to the ship filled with fuel. Approximately one extra day will therefore be required for a single shipload, for example a total of 7 days will be required for 6 casks.

24 hours will also be required for the reception of each filled cask at the central storage facility plus cleaning and preparation. An extra day is also expected to be required here before the ship is ready for departure, loaded with empty casks.

The various transport stages are described briefly below. The assumptions are 6 casks per shipment, a total travelling time one-way of 48 hours and no unforeseen delays. However, allowance must be made for both foreseen and unforeseen delays - such as poor weather, unplanned production stoppages at the power plants and at the central storage facility etc. - in planning the total annual transportation capacity of the system.

Typical times required for the transportation of 6 transport casks from a nuclear power plant to a central storage facility:

Transport stage	Time required
1. Journey to nuclear power plant with 6 empty transport casks onboard.	48 hours is assumed in this example.
2. Transfer from ship to trailer of one cask at a time. Lift from trailer via reactor hall level to pool, where cask is filled with fuel. Despatch.	24 hours/cask plus 24 hours for entire shipment equal 7 days for 6 casks.
3. Return journey to central storage facility.	Same as trip to plant. In this example, 48 hours.
4. Reception in the central storage facility. Cooling, cleaning, unloading of fuel. Preparation of casks for new shipment. Loading of transport casks on ship.	24 hours/cask plus 24 hours for entire shipment, i.e. total 7 days.

The time required for an entire transportation cycle will thus be 18 days, and the number of sea transports per year will be 16-17.

2.4.3 Transport casks

A transport cask consists of the following main components:

- An inner cask fitted with a neutron-absorbing substance and usually made of a heat-conducting material.
- A thick gamma ray shield made of heavy material such as lead or steel.
- A neutron shield to reduce neutron emission, mainly from curium-242 and -244.
- Heat-dissipating flanges on the outside of the transport cask or an air-cooling system.

- A shock absorber to protect the transport cask's cover and its connections.

A transport cask must also meet the safety requirements of the IAEA transport regulations for type B packagings. This means that it must be able to withstand:

- A 9-metre free fall onto a hard surface.
- Free fall from a height of 1 metre against a solid steel cylinder with a diameter of 15 cm.
- Heating for 30 minutes to 800°C.
- Submersion in water to a depth of 15 metres.

The transport cask must also meet the requirements imposed on type A packaging by the IAEA regulations.

The transport casks which will be used for the transport of spent fuel in Sweden will most probably be of European design. The casks which are the most likely candidates are NTL 11 and NTL 12:

NTL 11 (Fig. 2-3) is a refined version of the English Exellox transport cask, which has been used for transport between Oskarshamn and Windscale. NTL 11 was put into commercial operation in the autumn of 1977 for transport from the Wurgassen reactor in Germany to the French reprocessing plant at La Hague. The cask consists of an outer steel container, which serves as the pressure vessel, plus an inner lead container, which serves as the gamma ray shield. The inner lead container is covered with a stainless steel lining in order to facilitate internal cleaning of the transport cask. NTL 11 will be used exclusively for "wet" transports, in which the transport cask is filled with water during fuel transportation.

NTL 12 (Fig. 2-4) is of French design and has the largest capacity of any transport cask on the market. The cask consists of a 300 mm forged steel container which constitutes both the pressure vessel and the gamma ray shield. The transport cask is lined with stainless steel. In order to provide adequate neutron shielding, the steel cask is covered with a 100 mm thick organic material. The large quantities of heat (max. 100 kW) are dissipated by a large number of copper fins, 30 cm in length, on the outside of the transport cask.

NTL 12 can be used for the transport of spent fuel with either water or air as a coolant.

At present, one NTL 12 cask is being manufactured in Germany and will be ready for use some time in 1978. The French reprocessing company COGEMA recently ordered 5 NTL 12 casks. The NTL 12 cask will be used primarily for transports from nuclear power plants and the central fuel storage facility to reprocessing plants.

A number of Swedish and western European nuclear power plants (BWR) of somewhat older vintage are not equipped for the handling of NTL 12 casks. For this reason, the manufacture of a slightly smaller version of the NTL 12, designated TN 17, is planned. TN 17 will be ready for operation in 1979-1980.

As was mentioned under 2.4.1, the French company COGEMA has announced its plans to enter the spent nuclear fuel transporta-

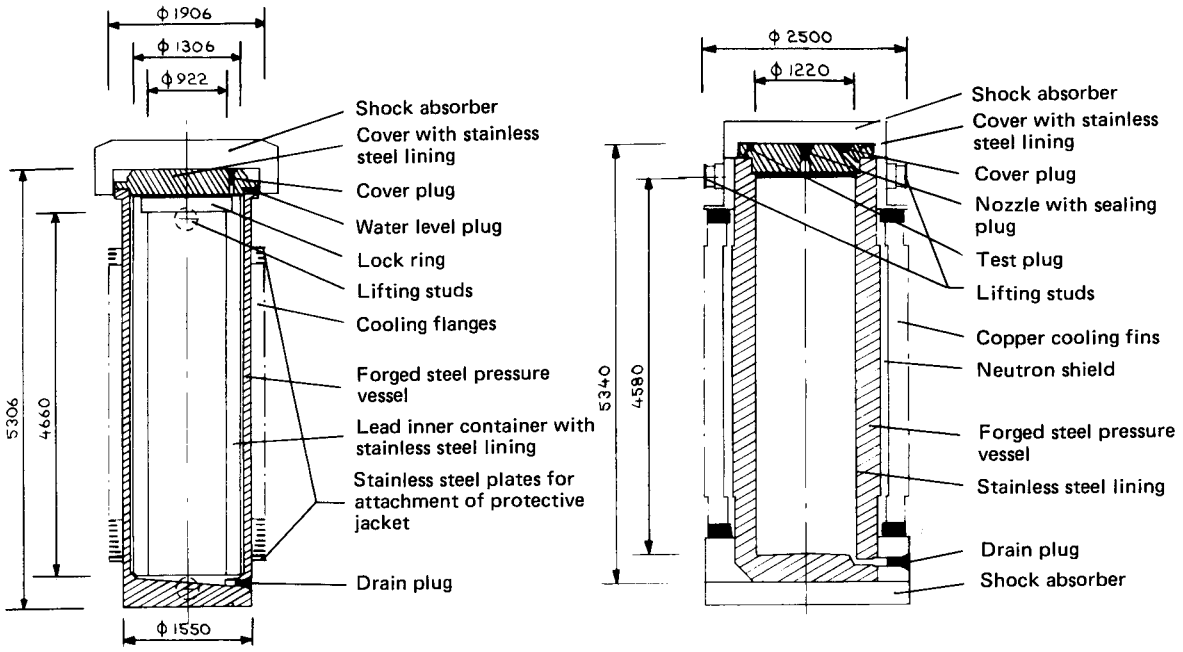


Figure 2-3. NTL 11 transport cask for spent fuel (left).

Figure 2-4. NTL 12 transport cask. This cask can be used for either spent fuel or vitrified waste (right).

tion field and plans to manufacture transport casks designated LK 80 B and LK 100 B. Data aside from capacity and dimensions have not yet been obtained for these casks.

2.4.4 Sea transports

It is assumed that spent nuclear fuel will be transported to the central storage facility by sea. The construction of a ship designed especially for that purpose is considered warranted.

A suitable size for such a ship is approximately 1 000 tons dwt. Such a ship can take up to 8 transport casks of the foreseen size, e.g. NTL 11 or TN 17, at a time. Available Swedish tonnage in this size class is very limited. Moreover, it is difficult to adapt existing ships to the requirements which must be met by a ship which is used regularly for the transportation of spent nuclear fuel. Existing ships could be chartered for occasional transports, but since fuel will be transported throughout most of the year, this alternative would be uneconomical in the long run.

The transport vessel should be equipped with effective steering and mooring equipment. Its draught will be limited to 3-4 m, which means that existing channels and harbours can be used. The ship will be designed for conventional cargo handling or for roll-on roll-off. With conventional handling, the cargo is lifted directly down onto the holds by means of dock-based cranes. This method is used today at the Swedish nuclear power plants. With roll-on roll-off handling, the transport vehicle - the trailer -

can drive on and off the ship without requiring any lifts by harbour cranes. The harbours at all of the nuclear power plants can be adapted for such rational handling.

The cargo must be secured in the transport vessel in such a manner that it will not come loose in the event of a collision or if the ship runs aground. The hull is divided by watertight bulkheads for added security against sinking. Should the ship nevertheless go to the bottom, it must be easy to locate. It will therefore be equipped with some such device as an underwater transmitter which is automatically activated if the ship should sink. The shipping lanes and channels are shallow enough to permit salvage of both ship and cargo.

The hull of the ship must be designed for running through ice. But a vessel of the size in question cannot function as an icebreaker, which means that the assistance of icebreakers will be required under difficult ice conditions.

A preliminary study for a ship project has been conducted /2-2/. According to this study, the time for delivery of a vessel of the type described here from a Swedish shipyard is currently 1 1/2 to 2 years.

2.4.5 Operation of the system

Below is an outline of a possible Swedish transportation organization:

- The transportation organization (personnel, material etc.) is a unit within the organization of the central fuel storage facility.
- Transport casks are procured by cooperation with an existing transportation organization. Alternatively, manufacturing under license may be considered.
- Auxiliary equipment (trailers, tractors etc.) is procured by the transportation organization.
- The transport vessel is built and owned by a Swedish shipping company. The transportation organization charters the ship as needed (probably year-round).
- The transportation organization is contracted by the Swedish nuclear power industry to undertake the necessary transports.

2.5 TRANSPORTATION OF VITRIFIED HIGH-LEVEL WASTE

2.5.1 General

The transportation of vitrified high-level waste in the form of waste cylinders from European reprocessing plants will be undertaken by the reprocessing company or by a transportation organization contracted by the reprocessing company.

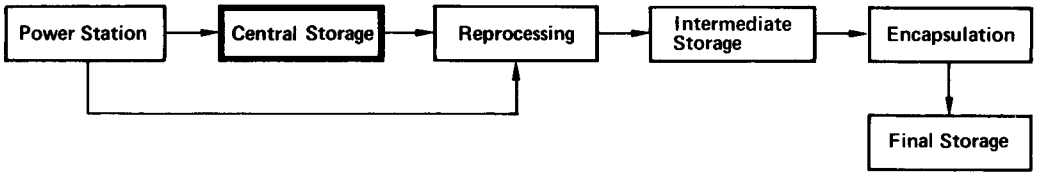
Waste cylinders will be transported from the reprocessing plant to Sweden in transport casks which are in principle identical to those which are used for spent nuclear fuel. NTL 12 is one of the types which may be used. This cask can transport up to 6 tons of

nuclear fuel with a maximum permitted heat generation of 100 kW. Calculations carried out for this transport cask show that 15 waste cylinders can be transported. 17 kW of heat is thereby generated, which is well below the permitted limits for the cask. The gamma ray and neutron shield is fully adequate to meet IAEA standards.

2.5.2 Scope of the transports

After expansion of the Swedish nuclear power industry to 13 reactors, the annual volume of discharged nuclear fuel will be about 300 metric tons of uranium, corresponding to 300 waste cylinders. The ships which are used today for the shipment of transport casks have a cargo capacity corresponding to 6 type NTL 12 casks. One transport vessel would thus be able to transport a maximum of 90 waste cylinders, corresponding to 3-4 shipments, per year, if all nuclear fuel is reprocessed.

3 CENTRAL STORAGE FACILITY FOR SPENT FUEL



3.1 GENERAL

The following chapter is based on the conceptual study carried out by the National Council for Radioactive Waste Management (PRAV) concerning a central storage facility for spent nuclear fuel /3-1/.

The size and design of the central storage facility has been based on a total storage capacity of 3 000 metric tons of spent fuel. The fuel will be stored in the central storage facility for an estimated maximum period of 10 years, after which it will be transported either to reprocessing or other storage.

The central storage facility will also be used to store discarded components from the reactor core. In some cases, these components will undergo mechanical treatment prior to storage in the fuel pools. It is assumed that the facility will be situated in rock.

The facility has three main sections: Receiving section, storage section and auxiliary systems section (see Fig. 3-1 and 3-2).

The fuel arrives at the central storage facility in transport casks which are unloaded, cleaned and cooled in the receiving section, after which the fuel is unloaded.

The fuel is stored in the storage section in a number of fuel pools of the same basic type as those in a nuclear power station.

The auxiliary systems section contains equipment for cooling, cleaning of the coolant water, waste treatment, process monitoring and power supply.

Storing spent fuel in water-filled pools is basically a relatively simple operation, from which many years of experience are available from nuclear power plants. However, due to the high handling frequency and large volume of fuel stored in the central storage facility, careful thought must be given to layout and systems design before the design of the facility can be finalized. The design goals shall be optimum operational availability, safety and economy. The design described below is tentative in certain respects, but nevertheless provides an idea of the basic principles for the handling and storage of spent fuel in a central storage facility.

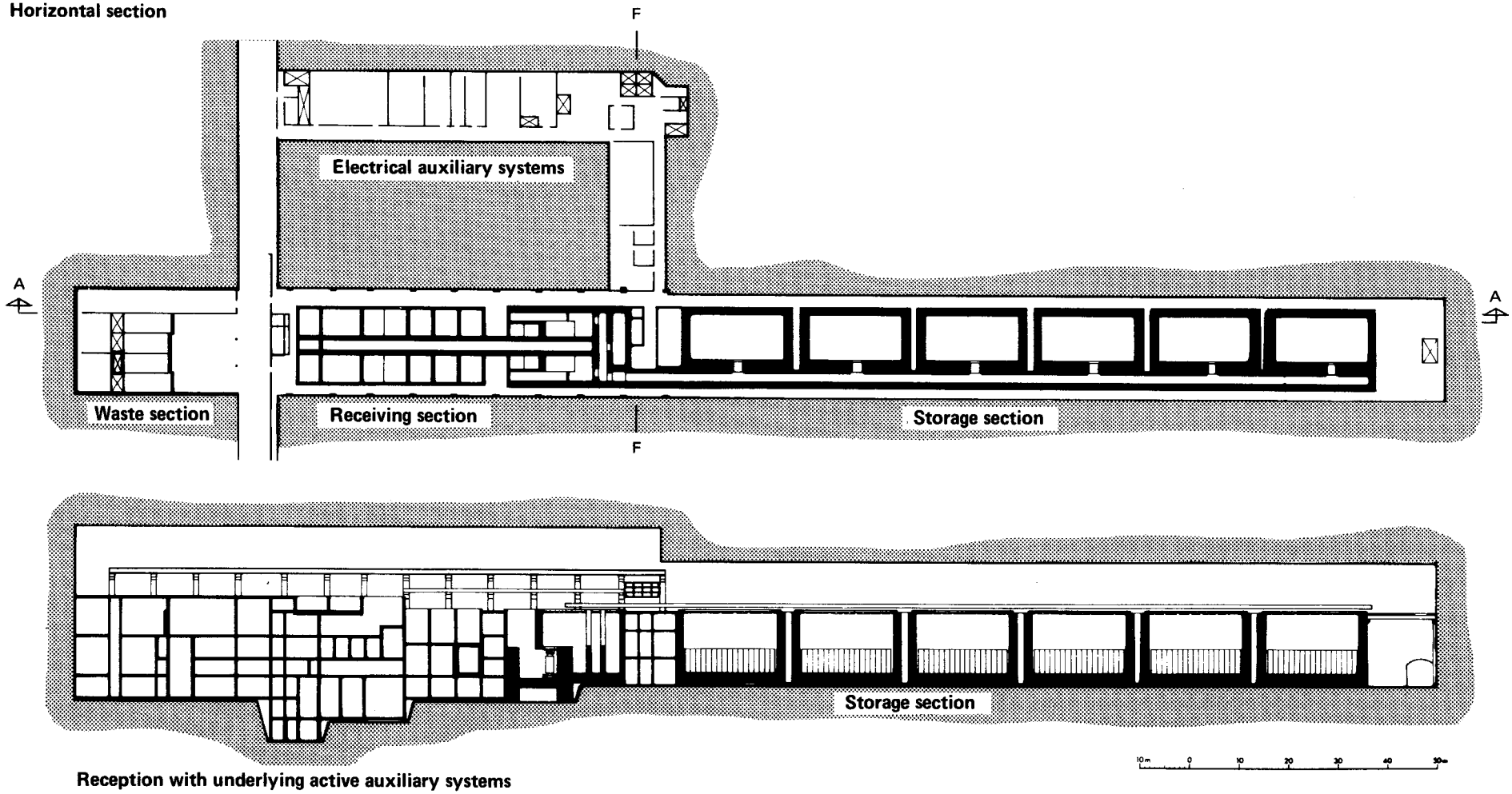


Figure 3-1. Horizontal and longitudinal sections of central storage facility for spent fuel. (From conceptual study by National Council for Radioactive Waste Management).

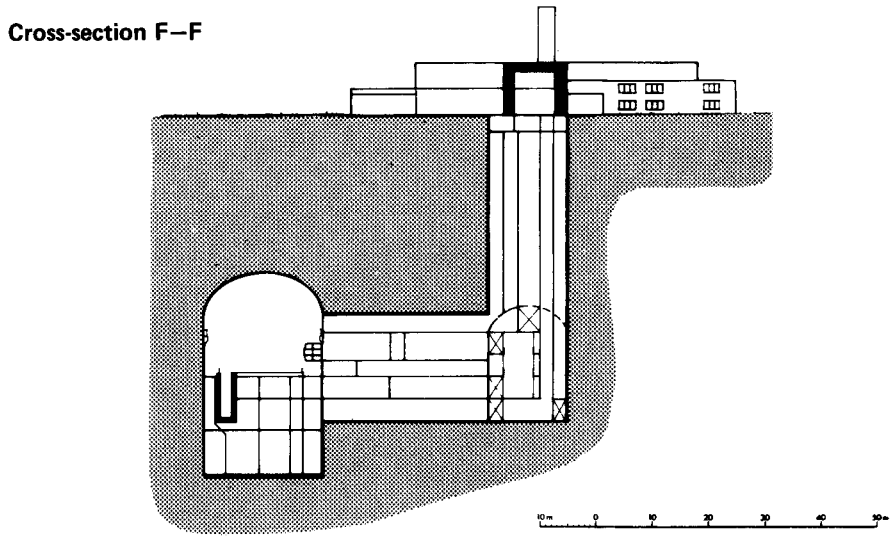


Figure 3-2. Cross-section of central storage facility for spent fuel. (From conceptual study by National Council for Radioactive Waste Management).

3.2 DESIGN REQUIREMENTS AND PRINCIPAL DATA

3.2.1 Principal data

The capacity of the facility is based on the following design data.

Storage capacity, fuel	3 000 tons uranium
Storage capacity, BWR elements	12 000
" , PWR elements	1 800
" , core components	700 tons
Storage cassettes, 25 BWR elements	480
Storage cassettes, 9 PWR elements	200
Number of storage pools	6
Normal amount of uranium per pool	500 tons
Water volume per pool	2 000 m^3
Total max. cooling requirement	6.5 MW
Seawater flow for cooling	400 kg/s
Temperature increase of coolant seawater	5°C
Pool temperature, normal	20-30°C
" , max. in normal operation	60°C
" , max. at reduced cooling capacity	100°C
Receiving capacity	1 transport cask per day
Total excavated rock	250 000 m^3

3.2.2 Design principles

The plant will be designed and constructed to modern technical standards in compliance with government laws and regulations.

The design of the facility shall be based on a service life of at least 60 years. Exceptions can be made for replaceable components.

Buildings and systems shall be designed to provide some protec-

tion against sabotage and acts of war. The fuel storage pools and a system which supplies them with water will be earthquake-proofed.

The facility will be equipped with diesel generators for stand-by power supply in the event of a failure of the external mains power system.

3.2.3 Separation and redundancy

Systems which provide cooling of the fuel or prevent or restrict the release of radioactivity shall be designed with redundant configuration. This redundancy shall be designed to ensure high operational availability and so that the malfunction of one component will not jeopardize the function of the system.

The temperature of the water in the fuel pools will be allowed to rise to max. 60°C if only one heat exchanger or one pump is out of service. Air temperature and humidity will also be allowed to rise in the event of the failure of one component. If the regular cooling system fails completely, the temperature of the water in the pools will rise to 100°C after about one week. In order to guarantee that the fuel is kept covered with water, the facility will be equipped with a make-up water system which can supply the required quantity of make-up water to the pools from a storage tank. This system will be physically separated from the pool's normal cooling system and will not require an electrical supply.

The pools will be designed to withstand the stresses to which they can be subjected in connection with boiling.

3.2.4 Fire

The facility will be equipped with fire detection and extinguishing systems. Fire zones will be designated for evacuation and fire fighting.

A fire must not be able to disable the electrical power supply to both pool systems. A fire in the operation control centre may be permitted to temporarily disable both process lines. Manual start of at least one of the process lines shall be possible from a place other than the operation control centre.

3.2.5 Operation control centre

The facility will have an underground operation control centre and a number of local control rooms. It will also be possible to monitor certain vital process parameters from a monitoring station on the surface.

3.2.6 Type of fuel transport

The systems and equipment in the receiving section will be designed to receive water-filled fuel transport casks. Casks for dry transport of the fuel will not be used for fuel transports to the facility, but may be used for transports from the facility.

The saturation pressure in the cask is normally 2-3 bar when it arrives filled with fuel.

It is assumed that the facility will be located near a harbour.

3.3 DESIGN OF FACILITY

3.3.1 General

The facility will be located in rock in order to satisfy requirements on protection, especially of the storage section, from sabotage and acts of war. Since it has been judged expedient to locate the receiving section directly adjacent to the storage section, the storage and receiving sections will be located in a line in a rock tunnel approximately 21 m wide, 25-35 m high and 280 m long (see Fig. 3-1 and Fig. 3-2).

The facility's waste system and the cooling and cleaning systems for the receiving and storage pools will be located in the lower part of the receiving section.

Electrical power supply and monitoring equipment for the equipment in the rock caverns will be located in a gallery which runs parallel to the gallery for reception and storage and in a transept between these galleries.

Other auxiliary system components will be installed in a building on ground level. The entry, administration and service sections of the facility will be located in connection with the auxiliary system section on the surface.

The surface units will be connected to the rock caverns through a vertical shaft for communications, pipes, cables and ventilation.

Nuclear fuel, core components and other heavy materials will be transported into the facility through descent tunnels from ground level down to the receiving section and auxiliary systems section at a depth of 50 metres below the surface. The gradient in these tunnels will be about 1:10. One loop of the transport tunnel will also pass through the far end of the storage section. Additional tunnels may be required to expedite quick and economical blasting of the rock caverns.

The receiving and storage sections will be designated as controlled areas in accordance with radiation protection regulations.

Personnel will be admitted to these areas via the shaft from the surface installation down to the changing quarters in the transept between the two galleries.

The only controlled area on the surface is the exhaust fan area.

3.3.2 Receiving section

Most handling of arriving and departing radioactive material takes place in the receiving section. The gallery measures 20 m in width and 35 m in height.

The receiving section contains equipment for reception, cleaning, cooling and unloading of the fuel from the transport casks. Next to the receiving section is a workshop for maintenance of the transport casks.

After the transport cask has been unloaded from the ship, it is transported down to the off-loading station in the receiving section on a trailer. The off-loading station is designed as a lock and is situated in a transverse passage underneath the floor of the receiving hall.

The transport cask with its transport cradle are lifted up through the transport opening in the lock by an overhead crane and placed in one of the radiation-shielded holding pens at floor level.

In the holding pen, the shock absorber and the fastenings which anchored the cask during transport are detached. A special lifting yoke is connected to the cask, after which it is raised by the crane and carried to special cells for testing, cooling and cleaning of the water in the casks.

The cask is first provided with a jacket which protects the cooling flanges from contamination during the following operations. The condition of the fuel is then checked by sampling the water in the cask. The cask is then connected to a special circulation system for cooling and cleaning which reduces the temperature and thereby also the pressure in the cask. The radioactivity level in the outgoing water is checked during the process, providing further indication of any defects in the fuel cladding and of the progress of the cask-rinsing operation.

Following this phase of the reception process, the cask is removed to a receiving pool. With the proposed design of the receiving section, this can be done in two different ways, depending on the type and design of the cask which is used.

Casks of standard type intended for Swedish transportation requirements will be lowered down in a shaft and placed on a transport wagon which can be moved via a horizontal transport passage to a holding pen underneath the receiving pool.

The top part of the cask is connected to a transport opening in the bottom of the pool by means of a mobile mechanical sealing device.

The above-described means of transport to the receiving pool requires a certain type of cask and can therefore not be applied generally. Other types of casks may also be used, e.g. for transports to foreign reprocessing plants. These casks are lowered directly down into the receiving pool in the conventional manner.

The advantage of the former method is that contamination of the outside of the cask during unloading can be completely eliminated.

The unloading process takes place under water in the receiving pool. The fuel is unloaded by a series of tools which are stored in the pool or are mounted on a gantry crane which covers the entire work area. The gantry crane is equipped with a telescopic device for handling the fuel elements and with lifting equipment for handling of tools, transport cask covers and linings etc.

The unloading operations are basically the same for a transport cask in the pool as for a transport cask in the holding pen underneath the pool. The following operations are carried out:

- The cover is removed and placed next to the cask.
- The fuel elements are lifted out of the cask by means of the telescopic device and taken to a storage cassette in the receiving pool. Each element can be inspected for cladding damage and, if necessary, the positions in the cassette can be covered with a lid and connected to a vacuum extraction system.

If the cask insert has to be replaced for the transport of another type of element, the following operations are performed:

- The insert if lifted up by the auxiliary hoist on the gantry crane and moved over to a side part of the pool which contains equipment for decontamination of the entire insert.
- A new insert is installed in the cask.

When the cask has been emptied, the cover is fitted. If the cask has been immersed in the pool, the cask and its protective jacket are externally washed during the lifting operations to prevent contamination of the transport path across the floor. If the cask has been in the holding pen underneath the pool, washing in connection with transport up to floor level should not be required.

The cask is then conveyed to a station for decontamination and inspection before it leaves the facility. This station is located adjacent to the cooling station and, like the cooling station, consists of a sub-floor cell. The protective jacket is removed and external parts are washed so that the surface activity of the cask is reduced to acceptable values.

After transport to the holding pen and placement on the transport cradle, the cask is ready for despatch and loading onto the ship.

The capacity of the receiving section has been estimated to be about one cask per day.

3.3.3 Storage section

The storage section consists of six water-filled pools connected with each other and with the receiving section through a transport channel There is a door in each pool leading into the transport channel. The pools are located separately and in a row, one after the other, in the rock gallery, which is about 20 m wide and 25 m high at this point.

Each pool normally contains about 500 tons of fuel and has a water volume of about 2 000 m^3 and a depth of about 12 m.

The pools are lined with stainless steel so as to permit inspection for leakage. Furthermore, the pools are equipped with a special leakage monitoring system and are covered.

The fuel is stored vertically in special cassettes in the pools. The cassettes are portable and are also used for transporting the fuel from receiving pool to storage pool. A special cassette-handling crane, which runs on beams on either side of the pools, is used to transport the cassettes.

The cassettes are of standard dimensions. One cassette for BWR fuel can hold 25 elements, while one cassette for PWR fuel can hold 9 elements. A total of 680 cassettes will be required for 3 000 tons of uranium, with the expected distribution between BWR and PWR elements.

The spent core components consist primarily of the fuel channels (boxes) which enclose the fuel elements in a BWR reactor, spent control rods, neutron emitters and detector equipment from the reactor cores. It is assumed that this material will be stored in stainless steel cases with the same external dimensions as the fuel storage cassettes after reduction of their volume by means of chopping and compacting of bulky components. An estimated 20-30 or so storage cases will be required up until 1990.

Adjacent to the receiving pool section is a special pool for handling (chopping and compacting) of core components.

Catwalks run alongside the receiving section outside of the paths of the overhead crane. Supply and discharge pipes for pool cooling run underneath these levels. The supply pipes are connected to the pools on the long sides, while the discharge pipes carry water away from the overflow weirs along the short sides of the pools.

3.3.4 Auxiliary systems section

The auxiliary systems section is divided into above-ground and underground installations. The radioactive systems are located underground and close to the receiving section in order to minimize the number of radioactive pipes and avoid long radioactive pipe ducts. There is also an inactive "uncontrolled" unit underground, which mainly contains electrical power supply and monitoring equipment for the underground systems, and an operation control centre from which the facility is controlled and monitored.

The surface systems include a saltwater cooling system, parts of an intermediate cooling system, a compressed air system, ventilation system, power supply, switchgear, diesel generators etc.

Radioactive cooling and cleaning systems

The cooling systems for pool water are located in connection with the storage pools and between the storage pools and the receiving

pools. The discharge pipes from the pools lead to a level control well underneath the receiving hall. The pumps and heat exchangers for the pool cooling systems are located on the level below the level control well. These heat exchangers are cooled via an intermediate cooling system. The heat exchangers for the intermediate cooling system are located in the surface building and are in turn cooled by seawater. The intermediate cooling system is connected to the pool cooling system via pipes in the communications shaft.

The cooling and cleaning systems for cask and pool water are located next to the delivery lock and close to the cask handling positions in the receiving section. A radioactive pipe duct runs between the two handling lines in the receiving section.

The filters for the cleaning systems are located on either side of this pipe duct in radiation-shielded cells. A service room for these filters and a charging room from which the filters are covered with filter material are located on the level above the filter cells.

Underneath the filters are pipe and ventilation ducts and underneath these are tanks for used filter material. The filter material is pumped out of these tanks via a radioactive pipe duct to transport casks for spent filter material. These casks are stationed in radiation-shielded positions in the waste section of the receiving section next to the transport lock, from which they can be transported out via the lock and the transport tunnel.

Besides equipment for spent filter material, the waste systems include systems for recovery, treatment and discharge of water. These systems incorporate an evaporator and a number of collection tanks for water of various grades and in various stages of the treatment process. They are located for the most part underneath the workshop section next to the receiving hall. Pipes run via the radioactive pipe duct.

A control station for controlled areas is located underneath the transport lock and in connection with the waste systems. Certain parts of the process can be monitored and controlled from this control station.

Electrical systems

The entire electrical section is an uncontrolled, i.e. inactive, area with the exception of certain personnel and ventilation areas. Personnel enter the electrical section at one end of the rock chamber via the communications shaft.

The facility's operation control centre is situated so that it provides a direct view over the receiving and storage halls. Personnel quarters are located adjacent to the operation control centre.

The electrical section is divided into fire cells to that electrical systems which belong to different redundant components and sub-systems are physically and atmospherically isolated from each other.

Auxiliary systems on the surface

The surface building contains an electrical section, diesel generators, cooling system, ventilation system and office and service quarters. The diesel generator section contains an extra ventilation level and the electrical section has an underground cable level.

The surface facility is an uncontrolled area, with the exception of certain areas for ventilation of the controlled parts of the rock cavern facility.

The surface building is entered from the outside via an entrance hall on ground level which is supervised from a guard room. The guard room is also situated to permit supervision of the entrance to the communications shaft. The communications shaft is surrounded by a thick concrete missile shield. Office and personnel quarters are located adjacent to the entrance hall.

The ventilation systems are divided into a controlled section, which serves the underground controlled receiving and storage sections, and an uncontrolled section, which serves the underground electrical systems and the surface building.

The controlled surface section is entered via a changing room in the surface personnel section.

The cooling system consists of pumps and heat exchangers which belong to the seawater cooling circuit and serves the intermediate cooling system for the pool water cooling and treatment systems.

The power supply systems are located at ground level above a cable level which is connected with the cable levels for the underground electrical systems via cable conduits in the communications shaft.

3.4 SERVICE LIFE AND DECOMMISSIONING

It is estimated that the central storage facility will have an economic life of approximately 60 years. This does not mean that the facility will no longer be useful for its purpose after this period of time. Naturally, machinery and equipment must be maintained and renovated as needed during the lifetime of the facility, but it serves no purpose to anticipate a longer service life at this time.

When the central storage facility has served out its life, decommissioning is facilitated by the location of the facility in rock. Decommissioning may proceed as follows:

- Fuel is removed to another storage facility, to reprocessing or to direct disposal.
- High-level components other than fuel are removed to final disposal.
- The facility is thoroughly decontaminated. Scrap and building components which constitute low- and medium-waste are taken away for disposal.

The facility can then be used once again for nuclear or other activities. If the rock caverns are not to be utilized for other purposes, but rather sealed off, the work of dismantling and decontamination may be reduced.

The decommissioning of a central fuel storage facility poses fewer problems than the decommissioning of a nuclear power plant. This is primarily due to the fact that the central storage facility does not contain heavy equipment or permanent installations which are highly radioactive.

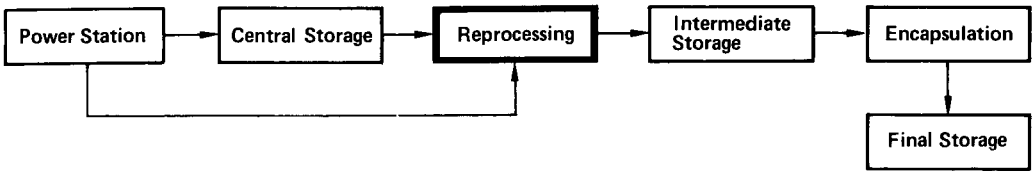
3.5 OPERATION OF FACILITY

The central storage facility for spent nuclear fuel will be under the supervision of the same authorities as a nuclear power plant, namely the National Nuclear Power Inspectorate, the National Institute of Radiation Protection etc. These authorities issue directives and regulations governing both the design and the operation of the facility.

Administrative surveillance of the fuel will be carried out under the supervision of the Swedish Nuclear Power Inspectorate (SKI) and the International Atomic Energy Agency (IAEA).

The operating personnel, an estimated 100 or so persons, will receive both theoretical and practical training in matters such as radiation protection, criticality, design and function of systems and components and operating and maintenance technology. Practical training of the personnel will include on-the-job duty at operative nuclear power plants with a special emphasis on fuel handling.

4 REPROCESSING AND VITRIFICATION



The spent nuclear fuel which is to be reprocessed will be transported either directly or via the central fuel storage facility to a reprocessing plant. No such plants are currently planned in Sweden, so reprocessing services must be purchased from abroad. For this reason, the design and operation of these plants will be dealt with in less depth than the other plants which are included in the nuclear fuel cycle. The main emphasis of this chapter is on the properties of the vitrified waste which will be returned to Sweden for final storage.

4.1 REPROCESSING

4.1.1 Processes

The plants for the reprocessing of spent nuclear fuel which have already been erected, are under construction or are in the planning stage are all based on variations of the American Purex process. In brief, this process involves chopping the fuel elements, dissolving the fuel in nitric acid, separating uranium and plutonium from the fission products in the fuel by means of extraction with an organic solvent, separating the uranium and plutonium from each other and final refinement of the uranium and plutonium.

Reprocessing of the spent fuel divides the fuel into four fractions containing uranium, plutonium, cladding waste and high-level waste in solution. Fig. 4-1 shows a flow scheme of the reprocessing of spent nuclear fuel from light-water reactors.

Light-water reactors, which dominate new reactor construction, use fuel elements with uranium in the form of an oxide. Burnup in this fuel amounts to about 30 000 MW per day per ton of uranium. This is much higher than in the English and French gas-cooled power reactors and in military reactors. Most reprocessing experience is for fuel from the latter type of reactor.

Processing this fuel presented more technical difficulties than expected. The problems were associated with the mechanical chopping of the fuel, the dissolution of uranium oxide, the separation of solid particles from the liquid and the higher radiation level, which leads to some disintegration of the organic process liquids. The reprocessing of light-water reactor fuel has now been demonstrated in 5 plants. Eurochemic in Belgium, WAK in West

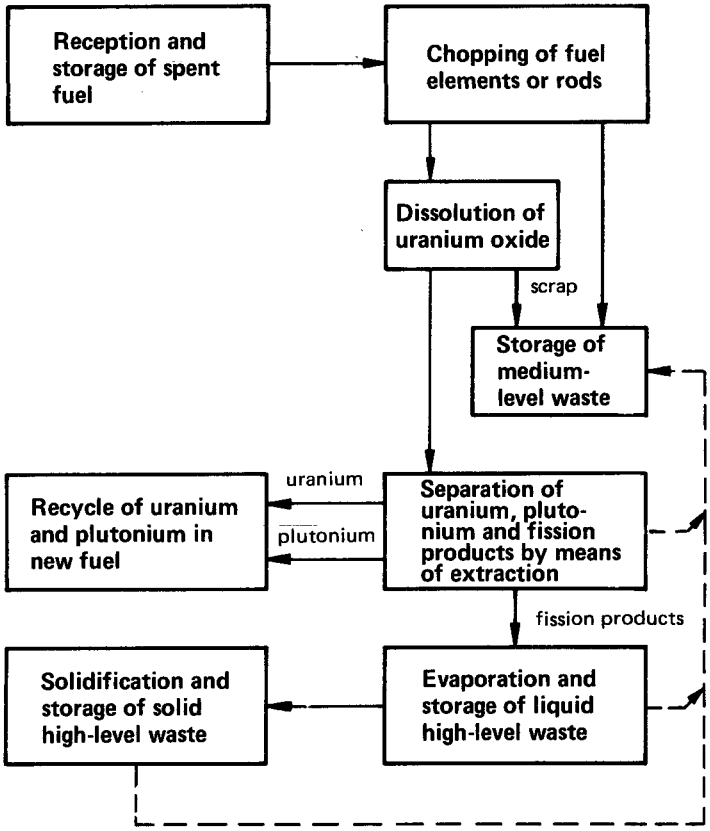


Figure 4-1. Process scheme for reprocessing of spent fuel from light-water reactors.

Germany, Windscale in England, La Hague in France and Nuclear Fuel Service in the USA.

Parallel to the main process are a number of auxiliary systems for treatment of the solvent to render it suitable for reuse, recovery of nitric acid and treatment of gas-borne and liquid waste. These systems require expensive and advanced technology for process control as well as for personnel protection.

The fuel entering the plant is stored in water pools which provide cooling and radiation shielding. The fuel is taken out of the pools after a cooling period of at least one year for pretreatment, in which the fuel elements are freed of their external structural parts and chopped into 5-8 cm long pieces.

The chopped material is transferred in a basket to a dissolver, where the uranium oxide with its fission products and transuranium elements is dissolved in boiling nitric acid. The leached hulls remain undissolved and are transferred to waste canisters for storage. This scrap contains induced radioactivity in the zircaloy cladding as well as traces of fuel. Gaseous fission products - mainly ioding, krypton and tritium - are evolved. The iodine is separated in the exhaust gas system by means of alkaline scrubbing and special filters. Methods are also available for separating krypton from the exhaust gas.

The solution containing the fuel dissolved in nitric acid is fed into a system where two immiscible liquids - the nitric acid-fuel solution and an organic solvent - flow in opposite directions

while the liquid phases are alternately remixed and separated (countercurrent extraction). Uranium and plutonium are highly soluble in the organic solvent under the prevailing conditions, while the solubility of the fission products is very low. The liquid phase flowing out of the system contains 99.9% of the fission products /4-21/. This liquid contains the high-level waste.

The high-level waste solution then goes to processing stages for concentration and storage. After separation in the first extraction stage, the product flow is delivered to another extraction stage. Here, conditions are adjusted so that plutonium is converted to a chemical form which is virtually insoluble in the organic phase, while the solubility of the uranium remains unchanged. As a result, uranium and plutonium are separated from each other. By means of additional refinement stages, both uranium and plutonium achieve the desired purity.

Uranium is finally obtained in the form of uranyl nitrate solution, which is calcined to uranium trioxide. The calcinate is transported to a plant for conversion to uranium hexafluoride and renewed enrichment. Plutonium is stored in the form of plutonium nitrate solution and converted to plutonium dioxide before it can be used for fuel manufacture.

The high-level waste solution contains 99.9% of the fission products, about 0.1% of the original quantity of uranium, about 0.5% of the original quantity of plutonium and all of the other transuranium elements. The solution is evaporated and normally stored for a period of time in the form of liquid concentrate in cooled and monitored stainless steel tanks. After storage, the high-level waste is calcined and vitrified.

4.1.2 Reprocessing plants

The high-level processes in a reprocessing plant must be carried out behind radiation shields in "hot cells" with metre-thich concrete walls. The sections in which uranium- or plutonium-bearing substances are handled must be designed for absolute security against criticality. This is achieved by geometric delimitation of the process equipment, limitation of the quantity of fissile material in the solutions or the addition of neutron-absorbing substances.

A reprocessing plant is divided into different sections on the basis of radiation levels and activity contents. A number of protective barriers are used. For example, the stainless steel vessel for dissolving the fuel is provided with special ventilation which maintains a negative pressure in the vessel. The dissolver is enclosed in a cell whose floor and walls are lined with stainless steel. The cell is ventilated so that it is at a lower pressure than surrounding spaces with lower activity levels.

The principles of plant maintenance are very important in the design of a reprocessing plant. Remote-controlled maintenance is normally required for the mechanical equipment in the cell where the fuel elements are chopped up. The cell is equipped with manipulators and other equipment which permit repairs without human contact.

A high surface finish is of vital importance to permit thorough cleaning (decontamination) of pipes and vessels prior to maintenance. Traps, pockets and sharp corners in pipelines where radioactive sludge could accumulate and be difficult to remove must also be kept to a minimum.

Whereever highly radioactive liquids are transferred between process vessels, an attempt is made to avoid the use of equipment with moving parts (mechanical pumps etc.). Instead, steam injectors, gas lift pumps or level differences between the vessels are used. This reduces the risk of leakage and minimizes maintenance.

4.1.3 Operational experience

The reprocessing industry has a 30-year history. Since the 1940s, fuel from military production reactors, research and experimental reactors and gas-cooled power reactors have been reprocessed. The technology for such reprocessing has been demonstrated in Europe on an industrial scale for many years at Windscale, Marcoule, La Hague and Eurochemic.

Experience from the first stage in reprocessing - receiving and storage of the fuel in pools - have led to the installation of more efficient systems for cleaning the pool water. Equipment for isolating and covering leaking fuel is another means to reduce the dose load on personnel who work in the receiving section.

High reliability in the unit for chopping the fuel elements is of fundamental importance in ensuring high operational availability in reprocessing. At the reprocessing plant in Karlsruhe (WAK), the chopping cell has not been entered since it was put into service 4 1/2 years ago. A La Hague, where the entire fuel bundle is chopped, there have been some initial difficulties, but these seem to have been overcome now.

A large quantity of insoluble fission products is obtained from the dissolution of high-burnup oxide fuel as compared with fuel from gas-cooled reactors. These fission products are obtained in the form of a fine powder. In addition, zircaloy chips are formed when the fuel rods are chopped. The solid particles can interfere with the extraction process and must therefore be separated. This can be accomplished by means of centrifuging or filtering.

One of the main problems in the extraction process for high-burnup fuel has been the radiolysis of organic solution accompanied by the precipitation of zirconium butyl phosphate, which can disrupt the process. The extraction apparatus must therefore be designed to provide minimum contact time between the organic solvent and the radioactive solution. Pulse columns and centrifugal contactors can give contact times which are a factor of 10 lower than mizer-sidimenters. The French atomic energy commission has developed special multi-stage centrifugal contactors which will be used at the plant for oxide fuel in La Hague.

Zirconium and ruthenium are the fission products which are most difficult to separate from uranium and plutonium in the first extraction cycle. They can therefore contribute to a higher radiation level in the following stages for separation and purification of the uranium and plutonium solutions. This has been a

problem in plants where the equipment was originally designed for fuel of lower burnup.

The exhaust gas cleaning process in a reprocessing plant must be able to separate the iodine isotopes I-131 and I-129 efficiently. I-131 is short-lived (half-life 8 days) and need only be taken into consideration when the spent fuel is reprocessed less than 6 months after it is discharged from the reactor. This is not a possibility which need be considered in the case of light-water fuel. Owing to reduced limits for the release of I-129 and problems in the handling of iodine-containing alkaline washing solutions, solid filters with silver-impregnated catalysts are now being used to an increasing degree. Tests of such filters have shown that they reduce iodine releases to less than 1/1000th.

The Eurochemic, WAK, Windscale and La Hague plants have demonstrated the reprocessing of oxide fuel on an industrial scale. At La Hague, improved apparatus has been developed and tested for the chopping of oxide fuel, the separation of solid particles and liquid extraction with short contact time. The outlook is therefore favourable for satisfactory operational reliability and plant availability in future large-scale operation.

4.1.4 Working environment and safety

The main factor which distinguishes the working environment in a reprocessing plant from the working environment in other chemical plants is, of course, the level of radioactivity. The radiation environment in a reprocessing plant can be kept under control by effective measurement and monitoring of radiation levels and registration of personnel doses. Such direct registration and monitoring are often not possible with respect to chemical environmental factors. The recommendations of the International Commission for Radiological Protection (IRCP) limit the annual dose to radiologically employed personnel to a maximum of 5 rems. The fundamental goal of radiation protection work shall be to keep radiation doses as low as is practically possible.

The internal environment at La Hague

The new receiving plant for fuel from light-water reactors has been in operation for a short period of time. Experiences from the processing of fuel from gas-cooled reactors are, however, considered sufficiently representative to provide a picture of the expected working environment situation which will be associated with operation with light-water fuel.

The reprocessing plant has a unit for company medical services (Service Médical de Travail; SMT) as well as a medical laboratory (Laboratoire d'Analyse Médical; LAM, which performs routine toxicological and radiotoxicological analyses. The SMT unit employs 12 persons while the LAM unit employs 16 /4-22/.

Radiation protection of the plant is under the supervision of Service Central Protection de la Radiation Ionizée (SCPRI) an agency under the Ministry of Public Health. ICRP standards are followed. The mean dose per employee in radiological work was

350 mrems/year in 1975. Employees in the decladding section (approx. 60 persons) and employees who work with decontamination (approx. 50 persons) had received a mean dose of 1600-1700 mrems per year. These two groups are exposed to the highest doses in the reprocessing plant.

Exposure data and other working environment matters are evaluated and discussed monthly by a committee which includes representatives of both COGEMA and the trade union.

There has been a health and safety committee at La Hague with local representatives from the employees and the company management for many years. Following a strike, a larger health and safety committee was appointed in November of 1976 with representatives from the employees' central trade union associations and the company management in Paris. This committee submitted its final report /4-23/ in June of 1977. The committee's recommendations are unanimous. They contain 47 points aimed at improving the working environment. The different points are of varying scope. 11 of the points had been acted on by June of 1977. There is a timetable for each point and all points are to be implemented by 1981. COGEMA's board has decided to implement the 47-point programme in accordance with the committee's proposal.

The 47-point programme is divided into the following sections:

- short- and medium-range reforms
- medium-range large-scale investments
- recruitment of new personnel (such as radiation protection personnel)
- safety equipment
- organization and methods
- studies
- training
- technical problems
- personnel problems
- technical organization and methods

Representatives of the employees on the health and safety committee agree that safety is good /4-24/, but that the 47-point programme must be implemented in order to provide adequate safety margins.

External environment

Releases into the air and water from the plant at La Hague are carefully monitored. The French radiation protection authorities have established the following limits for water releases:

β-radiation	40 000 curies/year
tritium	60 000 "
α-radiation	90 "

Water releases were measured at β = 32 000 curies, tritium = 11 000 curies and α = 13 curies for 1975. β releases were reduced in 1976 to 19 000 curies /4-25/.

An extensive network of monitoring stations at La Hague measures atmospheric emissions (mainly krypton) and water releases and

analyzes the levels of radioactive elements in the environment. In 1975, 2 200 samples were taken, on which 6 800 analyses were performed /4-26/. Samples are taken from the air, rainwater, streams, groundwater, plants, milk, seawater, sand and sediment, algae, crustaceans and fish. Samples are taken both near the plant and farther away.

In summary it can be said that radioactivity releases from La Hague to the most highly exposed group of people are estimated at 1 mrem/year from the consumption of fish and crustacean and 5 mrems/year from atmospheric emissions, mainly of krypton-85. Additional information is provided in /4-26/. These values can be compared to the natural external radiation level at La Hague, which is 100 mrems, and internal radiation from potassium-40 in the human body, which is 25 mrems /4-21/.

4.2 VITRIFICATION

Methods for the infusion of the high-level waste in glass are currently being developed in a number of countries. At the French PIVER pilot plant in Marcoule, 15 tons of high-level glass of the borosilicate type have been produced. A batch process using a special furnace is employed. The material is evaporated, calcined and melted to glass in the same apparatus. The furnace which is used is made of inconel and is heated by means of induction /4-27/.

The waste solution is mixed with a weak nitric acid solution which is added to the furnace and which contains the vitrifying additives in the form of tiny particles. The temperature is increased at the rate of 100°C per hour up to 1 150°C, at which point calcination takes place and vitrification begins. After 3-4 hours, the molten glass is allowed to run down into a container of chromium-nickel steel /4-20/.

An initial industrial prototype plant, AVM (L'Atelier de Vitrification de Marcoule) is based on previous experience from the PIVER plant and is currently being trial-operated with inactive material. The plant will be put into radioactive operation in early 1978 /4-28/. It will solidify the high-level waste from the reprocessing of relatively low-burnup fuel from gas-cooled reactors and fuel from research reactors. The construction of a similar plant for the solidification of waste from reprocessing of oxide fuel is also planned in La Hague.

The AVM process is continuous. First, the high-level liquid is dried to a powder (calcinate) which is then fused with borosilicate glass in a furnace at about 1 100°C. A homogeneous glass if formed, since the borosilicate glass mass dissolves all of the metallic oxides in the high-level waste. The glass is then cast in a chromium-nickel steel container. When the container is full, a lid is welded on so that it is hermetically sealed.

There is an intermediate storage facility for high-level waste at Marcoule where 154 waste cylinders with a total activity of 5 million curies have been stored for the past 6-8 years. The storage facility is located underground and is of concrete construction with vertical round holes in which the glass cylinders are stacked on top of one another. A 1 1/2 m thick concrete plug is

inserted into each hole. There is no danger to persons above the storage facility.

4.3 SOLUBILITY OF THE VITRIFIED WASTE

4.3.1 Time dependence of leaching rate

The glass is made up of a network with a coherent, three-dimensional structure containing silicon, boron and aluminium oxides. Other substances are then bound in this network.

The substances incorporated in the glass are leached out in two different ways /4-1/. Elements integrated in the network are dissolved directly from the surface. This mechanism applies for example to silicon, boron, aluminium and plutonium. In the case of cesium and strontium, the elements are first replaced by hydrogen ions in the glass lattice. This results in a diffusion-controlled leaching rate which diminishes with time. After a period of a few weeks or less, leaching of these elements as well will be determined by a direct dissolution of the surface /4-2/.

The exact shape of the leaching curves depends on the structure of the silicic-acid-rich film which is formed in contact with the leaching solution. It has been found experimentally that even elements which initially exhibit great differences in leaching rate will have very similar leaching rates after a few months. This means that the leached quantities will be proportional to the levels of the various elements in the glass dissolved from the surface, and that the rate of surface dissolution can be equated with the leaching rate.

In the case of laboratory-fabricated French glass of the light-water reactor type which contains 20% fission products, it has been found after about 3 months that the leaching rate per day is constant in tests with high water flow rates. The leaching rates are then about 2.10^{-7} grams per cm^2 and day at room temperature. This corresponds to a dissolution rate of 3.10^{-4} mm per year /4-3, 4-29/. Even lower values (down to 5.10^{-11} grams per cm^2 and day) have been obtained for strontium after 15 years in field tests with buried blocks of Canadian nepheline syenite glass /4-4/.

In order to calculate the leaching of a radioactive element from a glass body, the leaching rate is first multiplied by the surface available for leaching and then by the fraction which the radioactive element comprises of the glass.

In the final repository, the water flow around the encapsulated glass will be very low (see II:5). The leaching rate will thereby be lower than the rates determined at high rates of water flow /4-30/. This question is treated in greater depth in section IV:6.3.

4.3.2 Influence of groundwater composition

Variations in the composition of the leaching agent can affect its attack on the glass. This applies especially to its content

of substances which can break up the Si-O bonds in the network. Hydroxide and fluoride ions react in this manner /4-5, 4-6/, which means that the pH of the water can be an important factor. The concentration of fluoride ions in most groundwaters is low, and they have maximum effect at low pH values. Low pH values do not necessarily mean low water flow rate, since the leaching mechanism for glass generates a borate-silicate buffer with a pH of about 9. Bentonite also stabilizes the pH value at this level.

Leaching resistance can theoretically also be affected by the substances incorporated in the glass itself which, upon long-term contact between a small quantity of leaching liquid and glass, can build up to higher concentrations in the liquid /4-7/. When glass of this type is used - as it will be for the vitrification of radioactive waste - sodium, boric acid and silicic acid concentrations of several hundred ppm may build up. Sodium ions may, by rediffusion into the glass, reduce the corrosion rate somewhat. French experiments with salt solutions and solutions containing fission products indicate that the effect of the dissolved substances on the leaching rate is slight /4-8/.

Other substances in the leaching solution may at least temporarily reduce the leaching rate by forming a protective film of e.g. carbonates or sulphates on the surface under stationary conditions. At the pH which can be expected to prevail in the groundwater, precipitation reactions will also take place in the liquid, for example of carbonates, as well as the complexing of uranium and plutonium, which affects the further transport of these elements. An important factor in this respect is the hydrogen carbonate concentration. Chloride, which plays an important role in the corrosion of metallic materials, does not influence the corrosion of French glass. This has been shown both by the fact that it does not affect the mechanism of glass leaching and by the aforementioned French experiments.

It can be added that English experiments have found leaching rates which are 10-30 times higher for ion-free water than for natural water /4-9/. But ion-free water has never been encountered and cannot exist in the Swedish bedrock.

4.3.3 Influence of pH on resistance to leaching

Experiments have been conducted at Marcoule in France /4-8/ with glass containing radioactive fission products in order to study the influence of the pH on the leaching rates, especially for cesium-137 and strontium-90.

The experiments, which were conducted on a French glass, showed that the leaching resistance of the glass did not change within the pH interval 4-11. At pH 3, the leaching rate was 10 times as great, and at pH 14 it was 20 times as great as at pH /4-11/. These results indicate that glass with incorporated fission products is considerably less pH-sensitive with respect to the leaching rate than ordinary soda glass. This is especially true within the alkaline range, where a substantial increase in the leaching rate is noted for ordinary glass at pH 9-10. Resistance to acid, however, is poor according to information from Eurochemic and Marcoule. However, pH values below 4 are extremely im-

probable under the conditions which prevail around the glass body in final storage.

In order to confirm the resistance of the borosilicate glass to moderately alkaline solutions, measurements are being carried out at Studsvik of the leaching at 70°C of an inactive borosilicate glass at pH 10.5 and 8.5. This glass has a chemical composition which is the same as the glass which is to be used for final storage, i.e. with about 9% fission products. Results obtained to date indicate a doubling of the leaching rate at pH 10.5 in comparison with pH 8.5. The leaching rate at pH 8.5 is comparable to that for radioactive French glass.

Among the prerequisites which must be fulfilled by potential sites for a final repository for high-level waste in Sweden is the condition that the storage site must have a very low groundwater flow (on the order of a few decilitres per m^2 and year). This means that only a relatively small quantity of water will come into contact with a large area of glass. Since the mechanism for the reaction between glass and water leads to the release of alkali, one would expect the pH to rise.

However, measurements at Studsvik performed on a borosilicate glass containing inactive simulated fission products show that the pH remains below 9.5 at 70°C. The original pH of the leaching agent was thereby 8.5. In another experiment, in which groundwater was in contact with pulverized glass containing simulated fission products at room temperature for about 9 months, a pH of about 8 was measured. In this experiment, the ratio between the glass surface area and the leaching volume was fairly similar to the ratio for the glass bodies at low water flow. One reason why a greater pH increase is not obtained is that the glass contains boron oxide, which neutralizes dissolved alkali when it goes into solution and stabilizes the pH via a buffer system of borates.

The large quantity of filler material which surrounds the waste also has a similar stabilizing effect on the pH value, keeping it between 8 and 9.

4.3.4 Influence of temperature

All experience from leaching tests with glass of varying composition shows that the leaching rate and thereby the rate of attack on the glass increases sharply with the temperature. This is an important parameter, since the waste will heat the surrounding bedrock. Measurements have been made at Marcoule of the increase of the leaching rate for cesium and strontium, whereby it was found that the leaching rate is about 4 times greater at 50°C, 10 times greater at 70°C /4-11/ and 35 times greater at 100°C than at 25°C. Interpolation gives 3 times greater leaching rate at 40°C than at room temperature.

Trials are being conducted at Studsvik at room temperature and at 60°C.

The temperature conditions in the final repository are illustrated in Fig. 4-2. Naturally, leaching cannot begin until the canister and the chromium-nickel cylinder have been penetrated.

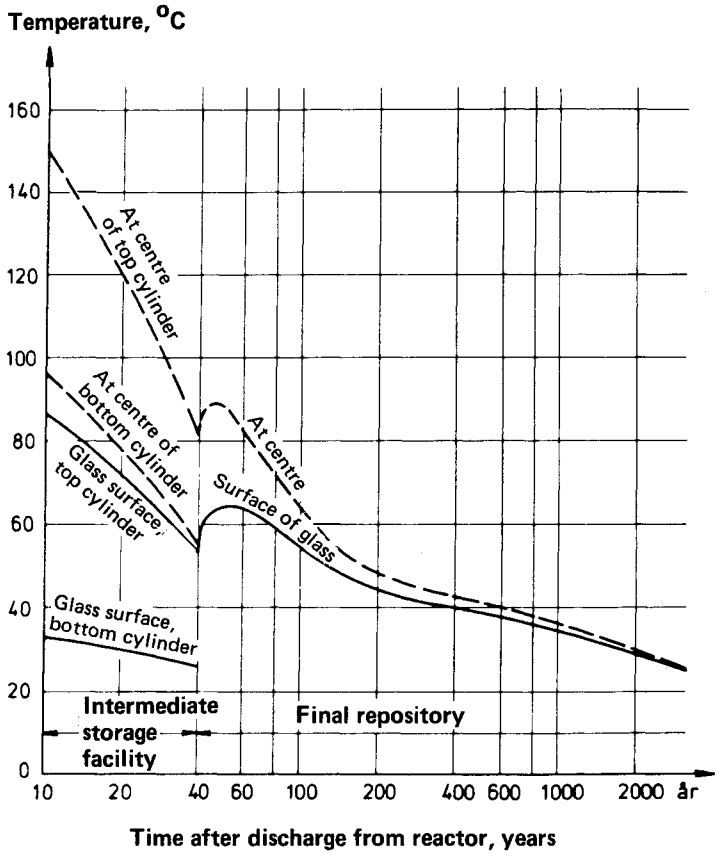


Figure 4-2. Temperature of the vitrified waste in the intermediate storage facility and the final repository.

4.3.5 Leaching tests with French glass at Studsvik

Tests were begun in July of 1977 with the leaching of French glass containing high-level fission products. In addition to gaining experience in the testing of radioactive glass, these tests are aimed at exploring the effects of the conditions which can be expected to prevail in a final repository in the Swedish bedrock. The variables which are of greatest interest in this respect are temperature, pH and groundwater composition. The radioactive glasses containing waste from light-water fuel which are now being leached contain 20% fission product oxides in comparison with the approximately 9% which are planned for the final storage of reprocessed Swedish waste.

The active glass is leached in an apparatus which is similar to the one used in France /4-8/. An initial stage of dynamic leaching is followed by a stage of static leaching (at higher temperature).

The composition of the leaching agent was as follows (pH 8.5):

HCO_3^- - 300 mg/l	F^- - 1,5 mg/l	K^+ - 2 mg/l
Cl^- - 27 mg/l	Na^+ - 125 mg/l	Mg^{2+} - 3 mg/l
SO_4^{2-} - 9 mg/l	Ca^{2+} - 6 mg/l	SiO_2 - 8 mg/l

Of the glasses which were leached, two were of types which are being considered for the solidification of light-water reactor waste, but in this case with approximately 20% fission products. Radioactivity was about 1 000 Ci, which means that most of the fission product content consisted of simulated fission products. Another cylinder contained light-water reactor glass with simulated fission products and approximately 3% plutonium dioxide.

The leaching solutions from the glasses containing radioactive fission products were measured with respect to strontium-90 and cesium-137. The leaching solutions from the plutonium-bearing glass were measured with respect to plutonium and, in some cases, americium. It should be noted in this connection that normal glass containing fission products only contains about 0.02% plutonium. It was necessary to boost the content by a factor of more than 100 so that the plutonium could be analyzed in the leaching solution.

The French glasses give leaching rates of 2.10^{-7} g per cm^2 and day after about 100 days at room temperature. The value for the plutonium glasses is roughly the same.

Results obtained to date at Studsvik for strontium and cesium show leaching rates at 25°C of 6.10^{-7} and 2.10^{-6} g per cm^2 and day after 40 days.

The leaching rates for strontium and cesium declined during the test period, which lasted for 40 days. If the experiment had been extended to 100 days, it is estimated that the leaching rates would have approached the same value which was obtained at Marcoule, namely 2.10^{-7} g per cm^2 and day. In order to permit the measurement of leaching rates at higher temperatures, the temperature was increased after 40 days to 60°C. The first values at 60°C show that the leaching rate increased by a factor of 10 /4-12/.

The leaching rate for plutonium at 25°C has been measured at Studsvik to be about 3.10^{-7} g per cm^2 and day after 30 days. The first values from 60°C show that the change of the leaching rate with temperature is less than in the case of strontium and cesium /4-12/.

When evaluating the results of the tests at Studsvik, it is important to note that they were conducted with approximately 20% fission products in glasses containing 42.9 and 46.0% SiO_2, respectively. In their reprocessing contract with COGEMA, SKBF has chosen a glass with only about 9% fission products, which means that the SiO_2 content will be more than 50%. The glass will therefore have a lower leaching rate for fission products and actinides /4-18/.

4.4 DURABILITY OF THE WASTE GLASS

4.4.1 General on glass - an amorphous material

Glass possesses a certain amount of fluidity. This property is valuable for withstanding the stresses which arise over a long period of storage. Certain observations confirm this:

a) Energy accumulation due to ionizing radiation is low and release does not take place instantaneously. Dislocations can be absorbed by the "plastic" glass structure better than by crystalline materials.

b) Helium formation in glass caused by alpha radiation over a long period of time does not lead to embrittlement. The unordered glass structure contains voids which can absorb large quantities of gases in the dissolved state. These gases can diffuse more readily through the glassy structure.

c) Electron irradiation over long periods of time does not produce any demonstrable effects, but the same energy dose in a short period of time sometimes leads to the formation of small bubbles. The glass structure has a certain "selfhealing" capacity.

Glass seems to have a good capacity to dissolve the highly variable mixture of radionuclides contained in the waste. There are a total of about 40 different elements in the waste, and as time passes, many of these elements are transformed into new elements by radioactive decay, which means different-sized atoms and new chemical properties. An unordered structure can more easily absorb these changes than an ordered crystal structure /4-18/.

4.4.2 Resistance of French borosilicate glass to radiation

Radiation damage may be caused by gamma, beta, alpha and neutron radiation. Alpha radiation, which consists of the heaviest particles and is quickly retarded in the glass, is considered to present the greatest risk for radiation damage.

Gamma radiation could only damage the glass structure indirectly by the formation of secondary electrons. Possible damage by gamma radiation is therefore covered by experiments with beta radiation /4-33/.

Beta radiation

In order to study the effect of internal beta radiation in high-level glass, specimens have been irradiated in French laboratories at Saclay /4-11, 4-13, 4-14/.

The beta dose for waste glass containing 9% fission products at various periods of time after vitrification is shown below (vitrification 10 years after discharge of spent fuel from reactor).

Years after vitrification	Rads
1	$5.8 \cdot 10^{9}$
10	$5.2 \cdot 10^{10}$
30	$1.2 \cdot 10^{11}$
100	$2.2 \cdot 10^{11}$
300	$2.4 \cdot 10^{11}$
500	$2.4 \cdot 10^{11}$

The additional increment of beta radiation after 500 years is very slight.

In the French tests, irradiation lasted 12 days with a total dose of $1.2 \cdot 10^{11}$ rads. Such accelerated radiation tests are thought to place a heavier load on the glass specimen than tests at normal dose rates, since the glass structure has less time to adjust to any structural changes.

When the irradiated glass specimens were compared with unirradiated specimens, it was found:

- that no accumulation of energy (Wigner effect) could be demonstrated;
- that the leaching rate for cesium-137 and strontium-90 had not changed due to irradiation;
- that structural ecamination by means of X-ray diffraction did not reveal any crystallization due to irradiation. Examinations by means of infrared spectrometry and scanning electron microscopy did not reveal any structural changes either.

British experiments /4-33/ with beta radiation in high-voltage electron microscopes showed that borosilicate glass was not affected by a beta radiation dose which was 100 times higher than the expected total dose to the glass. The tests were conducted at room temperature.

Alpha radiation

The greatest risk for radiation damage can be expected from alpha radiation /4-11, 4-13/. In order to examine the effects of such radiation, glass specimens were prepared which contained alpha emitters which emit a dose corresponding to storage for 1 000 years in a period of only 1-2 years. Actinide levels are then about 100 times higher than in normal waste glass. Possible effects include formation of helium, accumulation of energy, structural changes, changes of leaching values and changes of mechanical properties.

German experiments /4-34/ involving doping the glass with curium isotopes show that no significant change can be expected in the properties of borosilicate glass even after storage for 10 000 years.

Table 4-1 shows the load to which waste glass containing 9% fission products is subjected by alpha radiation when the spent fuel has been reprocessed 10 years after discharge from the reactor. This late reprocessing leads to a high level of americium-241 in the glass. If the fuel is instead reprocessed 3 years after discharge from the reactor and the high-level liquid is vitrified 10 years after discharge, all of the figures in the table can be divided by a factor of approximately 2.5.

Table 4-1. Alpha radiation for waste glass contains 9% fission products. 0.5% of Pu to the waste. Reprocessing 10 years after discharge from reactor.

Number of years after discharge from reactor	Number of alpha particles per cylinder	Number of alpha particles per g glass	Number of rads	Developed energy kWh/l	Developed He mm^3/g glass	moles/ cylinder
30	$6.9 \cdot 10^{22}$	$1.76 \cdot 10^{17}$	$1.4 \cdot 10^{10}$	103	6.5	0.119
100	$2.1 \cdot 10^{23}$	$5.4 \cdot 10^{17}$	$4.4 \cdot 10^{10}$	317	20	0.343
300	$4.8 \cdot 10^{23}$	$1.23 \cdot 10^{18}$	$9.7 \cdot 10^{10}$	695	46	0.797
1 000	$9.6 \cdot 10^{23}$	$2.5 \cdot 10^{18}$	$2.0 \cdot 10^{11}$	1 417	92	1.60
3 000	$1.24 \cdot 10^{24}$	$3.2 \cdot 10^{18}$	$2.5 \cdot 10^{11}$	1 806	120	2.06
10 000	$1.42 \cdot 10^{24}$	$3.6 \cdot 10^{18}$	$2.9 \cdot 10^{11}$	2 083	140	2.36
30 000	$1.68 \cdot 10^{24}$	$4.3 \cdot 10^{18}$	$3.4 \cdot 10^{11}$	2 444	160	2.79
100 000	$2.11 \cdot 10^{24}$	$5.4 \cdot 10^{18}$	$4.3 \cdot 10^{11}$	3 111	200	3.50
300 000	$2.77 \cdot 10^{24}$	$7.1 \cdot 10^{18}$	$5.6 \cdot 10^{11}$	4 056	260	4.60
1 000 000	$5.71 \cdot 10^{24}$	$1.46 \cdot 10^{19}$	$1.18 \cdot 10^{12}$	8 528	540	9.48
3 000 000	$1.17 \cdot 10^{25}$	$3.0 \cdot 10^{19}$	$2.4 \cdot 10^{12}$	17 220	1 100	19.4
10 000 000	$1.76 \cdot 10^{25}$	$4.5 \cdot 10^{19}$	$3.6 \cdot 10^{12}$	26 110	1 700	29.2

The following specimens were fabricated in order to simulate these effects:

	Weight of actinide	Weight of glass block	Energy kWh per litre and year
Americium-241	50 g	2 000 g	70
30% plutonium-238	50 g	2 000 g	100
Curium-244	0,6 g	50 g	810

Virtually all of the developed energy is dissipated in the form of heat. A smaller quantity of energy is accumulated in the glass. This quantity has been measured to be about 40 joules per gram glass after one year of storage with a high actinide level. This energy is released gradually in connection with heating. So there is no sudden, rapid release of energy resulting in a rapid rise in temperature.

Helium formation

The tests with americium and plutonium have now been in progress for 597 and 525 days, respectively. Calculations show that 5 (5.2) mm^3 of He have been formed per gram glass. The test with curium has not yet been completed. It is estimated that 50 mm^3 of He per g glass have already been formed in this test.

The unordered structure of the glass can dissolve helium, but if

the quantity becomes too great, there is a risk that tiny bubbles will form and that the glass will become more brittle.

The results of the test with americium indicate no change of the mechanical properties of the glass after 597 days, while the specimen with plutonium was difficult to measure in the first place. This is because a high level of plutonium results in phase separation in the glass - a problem which is not encountered at normal plutonium levels in the waste glass.

In order to obtain comparative data regarding the solubility of gases in glass, natural vulcanic glass which is about 400 000 years old has been analyzed. It contains 200 mm^3 of gas per g glass (of which 70% H_2) and was found not to be brittle. The entrained gas is evolved upon heating to 1 050°C /4-11/.

Table 4-1 shows that helium formation in tests on glass with a high actinide content corresponds to relatively short storage periods for our waste glass. The curium test corresponds to a storage period of about 300 years. The natural glass containing 200 mm^3 He/g, however, corresponds to a storage period of 100 000 years /4-32/.

Helium which is formed inside the glass can either remain in place, in which case the amount of helium per g glass increases, or diffuse through the glass to the open space at the top of the cylinder. The diffusion rate of helium increases with increasing temperature in the glass. British tests /4-33/ reveal significant helium diffusion at 170°C. Even if all the helium which is formed diffuses to the top of the cylinder, however, the gas pressure in the cylinder after 10 million years will only be about 30 atg, which is lower than the water pressure on the outside.

Neutron radiation

Neutron radiation in the glass originates primarily from the (α-n) reaction. The maximum estimated neutron emission rate is two neutrons per million alpha particles. Approximately 60% of the neutrons have an energy of more than 1 MeV. The effects of neutron radiation are slight in comparison with alpha radiation /4-32/.

4.4.3 Mechanical properties

When the glass has been cast into the chromium-nickel steel cylinder, a lid is welded on, producing a hermetically sealed container. When the cylinder is cooled, the steel will shrink more than the glass, which means that the glass will be subjected to compressive stress /4-16/. The glass is very resistant to such stress. Upon rapid cooling, cracks may form in the glass, resulting in surface enlargement. The cylinder may be cooled in connection with decontamination or lowering into a water pool. COGEMA has conducted experiments at Marcoule involving the extremely rapid cooling of inactive cylinders, after which the glass cylinders were examined. When the glass is cast, it becomes bonded to the chromium-nickel steel surface, making it difficult to determine which cracks formed due to cooling and which formed when the metal was pried loose. According to these tests, surface

enlargement corresponds to a factor of about 2-10. If excessive surface enlargement is suspected, the glass can be remelted in its steel container by heating to about 1 000°C and gradual cooling without thermal shock.

4.4.4 Thermal stability - crystallization

Most data on French borosilicate glass concerns waste from gas-cooled reactors. Tests have also been conducted on borosilicate glass from light-water reactors, however. The composition of such glass is:

SiO_2	50.60%
Na_2O	8.60%
B_2O_3	14.00%
NiO	1.00%
Fission products	25.80%

This glass has a melting temperature of about 1 000°C, minimum crystallization temperature 640°C and maximum crystallization temperature 930°C. The maximum crystallization rate is 0.01 µm/minute at about 800°C.

At the transformation point (approx. 550°C), there is no risk of crystallization for borosilicate glass with approx. 50% silicon dioxide, even over a long period of storage.

Since no experience is available from the storage of borosilicate glass for thousands of years, however, it was decided that it would be of interest to study which factors influence the rate of crystal growth and what would happen if crystallization should eventually occur over a long period of time.

The tests at Marcoule /4-11/ show that crystallization tendency increases as the level of fission products increases and as the level of molybdenum increases. For glass with 9% fission products and approx. 20% boron oxide, crystallization tendency is low.

In order to study the effects of crystallization on borosilicate glass, test blocks were heated at Marcoule for one year at 500°C and 600°C as well as for 100 hours at 800°C /4-11/. After heat treatment, the specimens remained intact and were not cracked. It is known that the size of the crystals in glass which has crystallized at 800°C are relatively large. But if crystallization of the borosilicate glass were to occur over a long period of time at a temperature below 550°C, the crystals would be smaller. The risk of cracking is then even less than for crystallization at 800°C.

Leaching tests were conducted on glass blocks which were heat treated and thereby crystallized.

The leaching rate for these specimens was compared with the leaching rate for non-heat-treated specimens. It was found that heat treatment increases the leaching rate slightly for certain

elements, such as cesium, while it reduces the leaching rate for other elements, such as strontium. The same tendency was noted in inactive Swedish tests /4-17/. The difference in leaching rate are small - on the order of 50%. The probable explanation is that upon crystallization, the glass forms a glass-ceramic material in which certain elements are bound more tightly in the crystals, while other elements are concentrated at the boundaries between the crystals and are thereby more easily leached out.

The conclusion is that:

- the risk of crystallization is low below 550°C;
- if crystallization should nevertheless occur over a long period of time, the glass will not crack and changes in leaching rates will be small.

At the Glass Research Institute in Växjö /4-17/, glass bodies have been fabricated of French type borosilicate glass containing 9% inactive simulated fission products. The composition of the glass is given in table 4-2.

Table 4-2. The composition of the test glass is:

Constituent	% by weight	Constituent	% by weight	Constituent	% by weight
SiO_2	53.0	BaO	0.46	CdO	0.026
Na_2O	11.3	Y_2O_3	0.15	SnO	0.014
B_2O_3	19.4	ZrO_2	1.28	Sb_2O_3	0.0036
Al_2O_3	2.1	MoO_3	1.63	CeO_2	0.75
UO_2	3.9	MnO_2	0.77	La_2O_3	0.71
Fe_2O_3	1.3	CoO	0.21	Nd_2O_3	1.21
Cs_2O	0.88	NiO	0.37	Pr_2O_3	0.35
SrO	0.26	Ag_2O	0.011		

In order to study physical changes in connection with crystallization, some of the glass bodies were heat-treated at 800°C for 14 days, whereby a certain amount of crystallization occurred. The results show that crystallization does not have any significant effect on strength, coefficient of expansion, transformation point, softening point or density. Nor do the speciments crack. In the case of cesium, the leaching rate after crystallization increased by about 50%, while the leaching rate for strontium decreased.

4.4.5 Molybdate phase

The fission products contain inactive molybdenum which, after calcination is present in the form of molybdenum oxide /4-17/. At a fission product level of 9% in the glass, the molybdenum level

is about 1.6% MoO_3. When the glass is melted together with the fission products, it may sometimes happen that a phase consisting primarily of sodium molybdate separates from the glass. Under the most unfavourable circumstances, Bonniaud at Marcoule found that 0.5% of the glass would consist of a separate molybdate phase /4-31/. This separation probably takes place within fixed temperature interval (600-800°C, according to British findings), after which the phase redissolves in glass at higher temperatures. The molybdate phase consists mainly of inactive components, but may also incorporate some active strontium and cesium. It might also possibly dissolve small quantities of actinides. The molybdate phase is soluble in water, whereby the constituent strontium and cesium will also come out in the water.

The development work on high-level glass has thus far indicated the following methods for counteracting the formation of molybdate phase:

- A lower level of fission products in the glass (low MoO_3 content).
- A high boron oxide level reduces molybdate phase, according to French experiments.
- Avoid contamination with sulphate - reduces molybdate phase, according to English experiments /4-19/.

At Marcoule it is estimated that no more than 1% of the glass cylinders will contain molybdate phase.

4.4.6 Important parameters for French borosilicate glass

The following parameters characterize French borosilicate glass which is obtained in the vitrification of high-level waste /4-15/:

- Leaching rate at 25°C: 2.10^{-7} grams per cm^2 and day.
- Factor for increase of leaching rate at 70°C: 10 times.
- Increase of surface area in connection with handling and transport: 2-10 times.
- Transformation point (= temperature below which crystallization does not take place): 550°C.
- Increase of leaching rate if crystallization occurs: 50%.
- Density: 2.8 g per cm^3.
- Thermal conductivity: 1.2 W per metre and degree Celcius.

The chemical composition of French waste glass from light-water reactors is:

	% by weight	
SiO_2	54.9	
Na_2O	11.7	
B_2O_3	20.0	
Al_2O_3	2.2	
$NiO + Fe_2O_3$	1.1	(from structural components of the fuel bundles)

	% by weight
Actinide oxides	1.1
Fission products	9.0

A typical composition of the actinide oxides is:

	% by weight
UO_2	0.8
NpO_2	0.19
PuO_2	0.0074 *)
Am_2O_3	0.14
Cm_2O_3	0.003

*) COGEMA calculates that 0.15% of the plutonium in spent fuel is retained in the waste glass at La Hague, which corresponds to a level of 0.0074% PuO_2. The safety analysis, however, has assumed a PuO_2 level in the glass which corresponds to 0.5% of the level in the spent fuel.

Typical contents of the most important fission products are:

	% by weight		% by weight
MoO_3	1.63	Pr_2O_3	0.40
Nd_2O_3	1.21	PdO	0.40
ZrO_2	1.15	Rb_2O	0.27
CeO_2	0.66	Tc_2O_3	0.27
RuO_2	0.61	Sm_2O_3+Eu_2O_3	0.27
BaO	0.55	Y_2O_3	0.18
Cs_2O	0.54	Rh_2O_3	0.18
Gd_2O_3	0.54	SrO	0.14

The distribution of the fission products varies with the degree of burnup of the spent fuel and the time after discharge.

The steel cylinder in which the waste glass is cast is made of a heat-resistant stainless steel (type Z 15 CN 24-12: chromium 24%, nickel 12-13%, carbon 0.15%). The cylinder is illustrated in Fig. 4-3. Its dimensions and weight are given below:

Diameter	400 mm
Overall height	1 500 mm
Thickness, stainless steel	3 mm, cylinder shell
	4 mm, cylinder ends
Glass volume	150 litres
Glass weight	420 kg
Total weight, approx.	470 kg

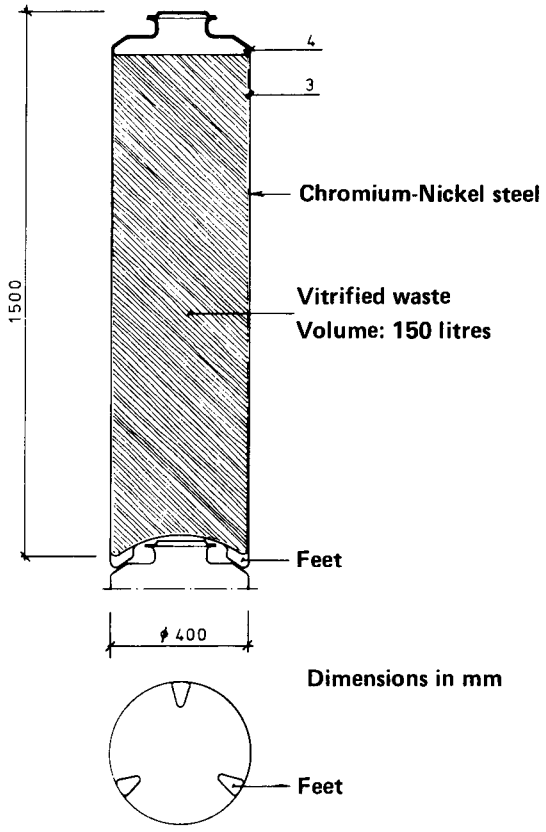
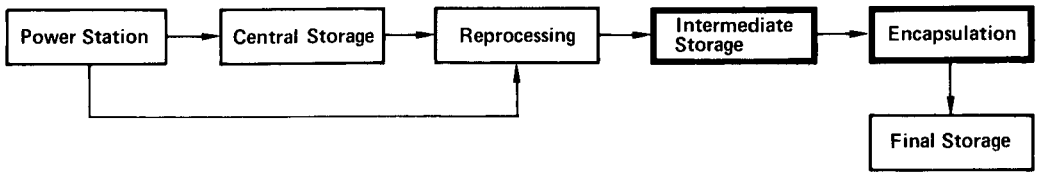


Figure 4-3. Waste cylinder. The vitrified waste is cast in a container made of chromium-nickel steel. The container is sealed with a welded-on lid. The feet enable the waste cylinders to be stacked on top of each other.

5 INTERMEDIATE STORAGE AND ENCAPSULATION



5.1 GENERAL

An intermediate storage facility and an encapsulation station for the waste cylinders from the reprocessing plant will be constructed adjacent to the final repository. (Possibilities for alternative siting of these facilities are discussed in chapter I:11).

The waste cylinders will be stored for 30 years in the intermediate storage facility (this storage period can be extended). The purpose of intermediate storage is:

- to reduce heat flux from the waste in the final repository. During a 30-year period, heat flux decreases from approx. 1 200 W to approx. 525 W per waste cylinder /5-1/
- to postpone the encapsulation of the waste and the construction of the final repository. This provides time for further development and optimization of the encapsulation procedure and the design of the final repository.

Before the waste cylinders are placed in the intermediate storage facility, they pass through a receiving section, where they are unloaded from the transport cask in which they arrive at the facility and where the inside of the transport cask is checked and, if necessary, decontaminated.

When the waste cylinders are to be transferred to the final repository after the end of the storage period, they pass through an encapsulation station, where they are encased in a lead-titanium canister. The purpose of this canister is to provide long-term resistance to corrosion and radiation shielding. The radiation shielding reduces radiolysis of the groundwater in the final repository to a negligible level and also simplifies handling.

The plant thus has three main sections: receiving, intermediate storage and encapsulation. Most of the facility is located underground to provide protection against external forces (acts of war and sabotage). The facility can store 6 000 waste cylinders and receive and encapsulate 300 per year. The design of the facility is based on existing technology. The intermediate storage facility is similar to the storage facility for vitrified waste which is currently in operation at Marcoule in France. The Marcoule facility was designed by the French company Saint Gobain Techniques Nouvelles, which has also been engaged by KBS for this project /5-2/.

For a more detailed description of the facility, see the drawings at the end of this chapter.

5.2 DESCRIPTION OF FACILITY

5.2.1 Layout

The layout of the plant is illustrated by Fig. 5-1, which also shows its location in relation to the final repository.

The surface installations consist primarily of an entrance building with administration and service premises. This building is of conventional design and is not described further in this report. The other parts of the facility are located underground with a rock cover approximately 30 metres thick.

The underground part of the facility consists of two rock galleries laid out in the form of a T with reception and encapsulation in a line and with intermediate storage perpendicular to them. The two rock galleries are separated from each other (communication is provided through two smaller tunnels) in order to avoid large spans at the point of intersection. The rock galleries have a maximum span of 20 metres and a height of 30 metres, which is not exceptional compared to existing rock cavern facilities in Sweden and in other countries. The rock galleries are stabilized by means of conventional construction methods.

The underground part of the facility can be entered via a tunnel, through which the transport cask with the waste cylinder arrives on its trailer, or via a vertical shaft for personnel etc. Connection with the final repository is provided through a horizontal tunnel from the encapsulation section to a vertical shaft which leads down to the final repository.

See I:14 for the schedule for the construction of the facility.

5.2.2 Reception

The waste cylinders from the reprocessing plant arrive at the facility in a transport cask on a trailer (see chapter 2). The cylinders have a diameter of 40 cm, a height of 150 cm and a volume of 150 litres. An NTL 12 transport cask can hold 15 waste cylinders.

After external washing in the arrival hall, the transport cask is lifted from the trailer through an air lock into the receiving room (see Fig. 5-2), where it is placed on a wagon in a vertical position. The radioactivity of the air which is blown through the cask is monitored to check whether the cask is internally contaminated. The bolts which retain the cover are removed, and the cask is moved to a position underneath the unloading cell and connected to an opening in the floor of the cell. When all glass cylinders have been lifted out of the transport cask into the unloading cell, the cask is flushed with water if the monitoring indicated that its inside was contaminated. The transport cask can then be returned to the reprocessing plant.

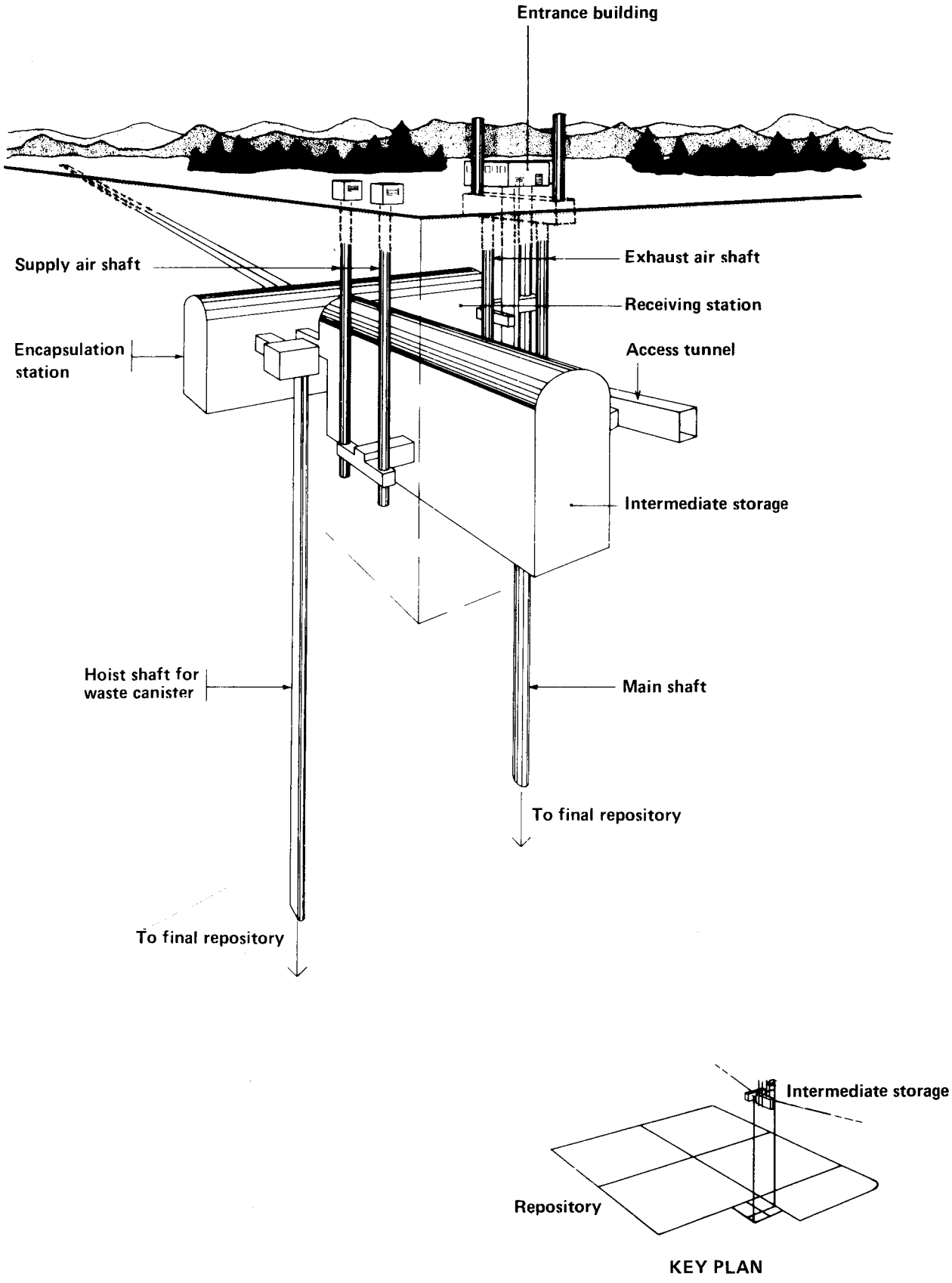
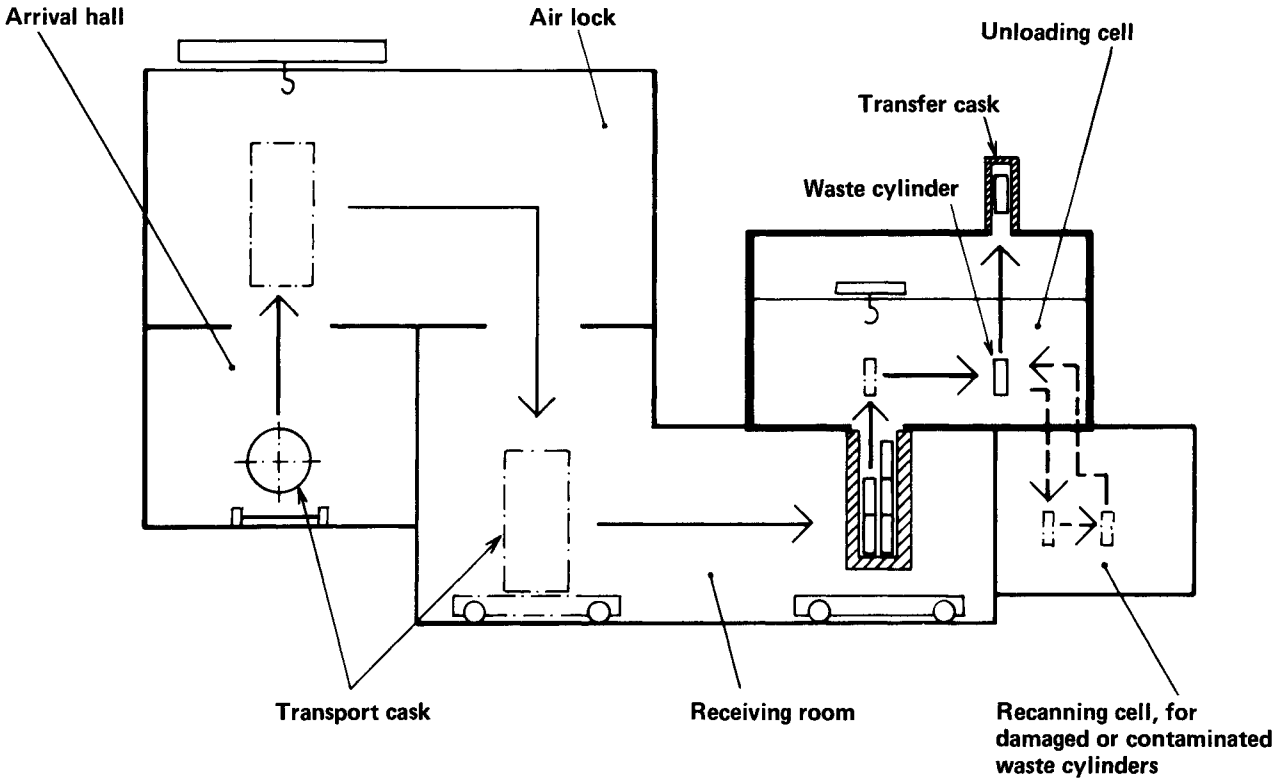
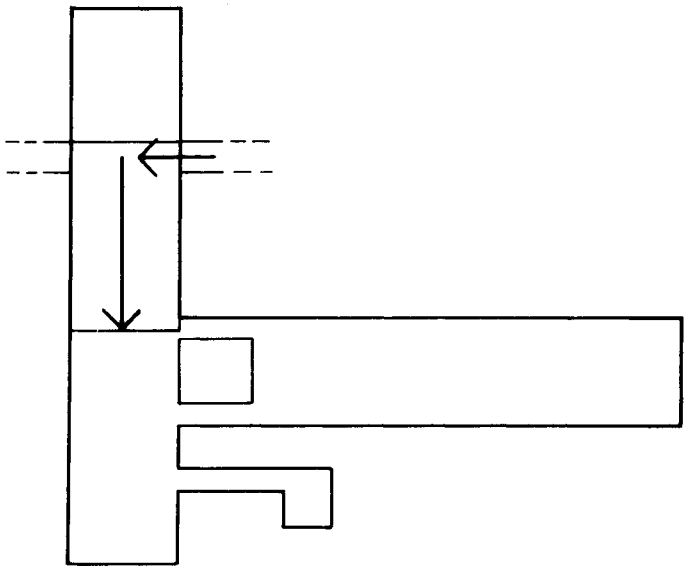


Figure 5-1. Perspective drawing of plant for intermediate storage and encapsulation. It is located underground with a rock cover approximately 30 metres thick. The plant is located above the final repository.





KEY PLAN

Figure 5-2. Schematic diagram of reception. In this section, transport casks are received and the waste cylinders are unloaded. Damaged or contaminated cylinders are encased in an outer container of chromium-nickel steel.

The cylinders are unloaded from the transport cask in the unloading cell. This cell is enclosed in concrete of sufficient thickness to provide radiation shielding for the personnel. The cell has four handling stations, each equipped with a radiation-shielded window and a pair of master-slave telemanipulators so that work in the cell can be done from the outside.

Materials are moved inside the cell by means of a remote-controlled overhead crane with a lifting capacity of 8 tons. When it is not being used or when it requires maintenance, the crane is moved to an intervention cell through an opening which can be closed by a radiation-shielded sliding door.

The waste cylinders are brought into the unloading cell through an opening in the floor which is connected to the transport cask in the receiving room. The opening is closed by means of a radiation-shielded sliding door when it is not being used.

When the transport cask has been connected, its cover is removed by the overhead crane and placed in a sealed box in order to prevent spread of any contamination.

The waste cylinders are lifted out of the transport cask and placed in a temporary storage in the cell by the overhead crane, which is equipped with a special grapple.

When a waste cylinder is to be transferred to the intermediate storage section, it is placed by the crane in a position underneath an opening in the roof of the cell. This opening is covered by a radiation-shielded sliding door when it is not being used.

If the inside of the transport cask has been found to be contaminated, all waste cylinders from such a cask are assumed to be contaminated and are taken to the recanning cell, where they are encased in an outer container similar to the one with which they were provided in the reprocessing plant in order to prevent contamination of the intermediate storage section.

The recanning cell has two handling stations, each equipped with a radiation-shielded window and a pair of master-slave telemanipulators. It is connected to the unloading cell through two openings in the roof, one to lower the waste cylinders into the cell and one to lift them out. The openings can be sealed by concrete plugs.

A radiation-shielded arrangement makes it possible to bring empty outer containers and their lids into the cell from the maintenance cell. They are placed on a carousel which brings them into a position where they can receive a waste cylinder as it is lowered from the unloading cell through the opening in the roof. In the next position, the lid is placed on the filled outer container and welded in place. It is then moved into a position underneath the second opening and lifted up into the unloading cell by the overhead crane.

In the unloading cell, the outer containers are decontaminated externally by washing with water under high pressure. They are then moved by the overhead crane into a position underneath the opening in the roof of the cell for transfer to the intermediate storage system.

5.2.3 Intermediate storage

The waste cylinders are transferred to the intermediate storage section inside a radiation-shielded transfer cask.

The transfer cask is enclosed in a lead jacket 25 mm thick which is lined on the inside with stainless steel and covered on the outside with a 20 cm thick layer of polyethylene. These layers provide adequate radiation protection for the operating personnel. The transfer cask has its own ventilation system with a fan and filters at the air intake and outlet. When a waste cylinder is inside the transfer cask, the ventilation system is used for cooling (when required) and to check whether the outside of the waste cylinder is contaminated by monitoring of the radioactivity of the filter at the air outlet.

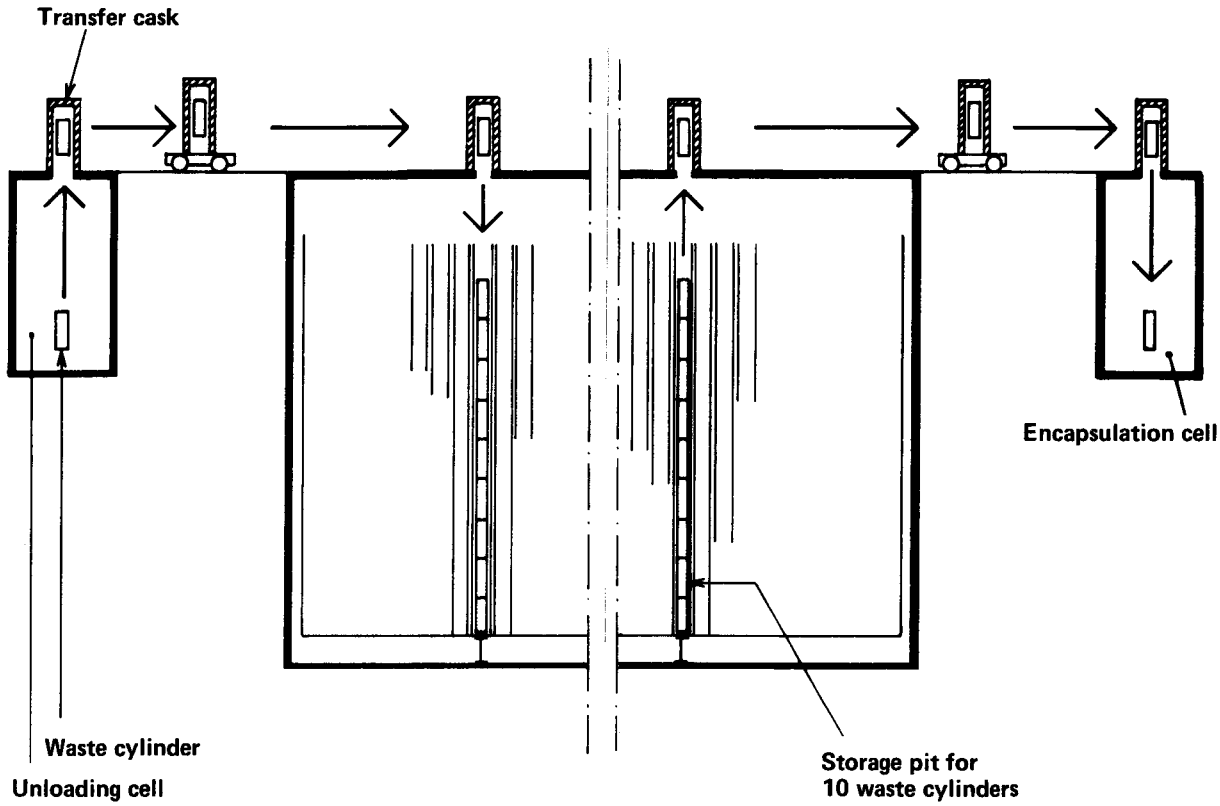
The transfer cask is positioned over the opening in the roof of the unloading cell, which is covered by a radiation-shielded sliding door. There is a similar door in the bottom of the transfer cask, and both doors are opened simultaneously. The waste cylinder, which is positioned underneath the opening in the unloading cell, is then lifted into the transfer cask by the hoist with which the cask is equipped. When the waste cylinder is inside the cask, both sliding doors are closed and the transfer cask is taken to the intermediate storage hall by a portal crane on rails to a position where it can be reached by the overhead crane in the intermediate storage hall.

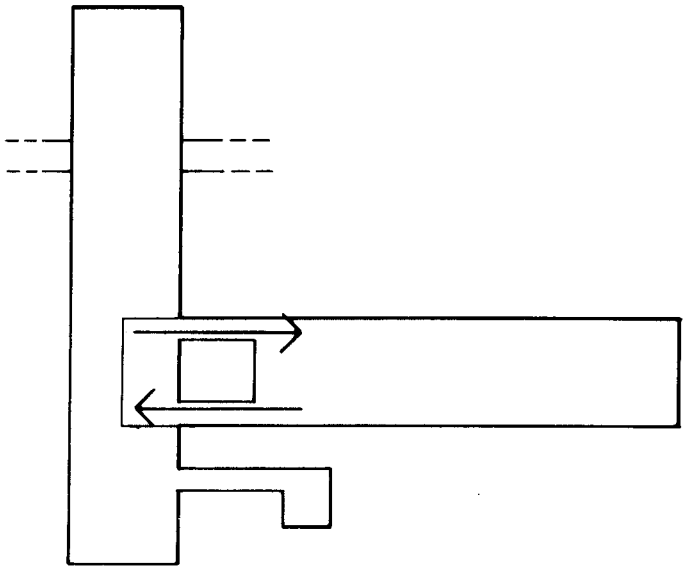
In the intermediate storage section, the waste cylinders are stored in steel pits inside a steel frame in a concrete trench (see Fig. 5-3). Each trench contains 150 steel pits spaced at centre-to-centre intervals of just under 1 metre and each with room for 10 waste cylinders stacked on top of one another. Each trench thus holds 1 500 waste cylinders. The intermediate storage section has four trenches in two groups with room for ventilation equipment between the groups. Each group has its own ventilation system. The total storage capacity of the facility is thus 6 000 waste cylinders.

The storage trenches are covered by a concrete slab which is thick enough to provide radiation protection for the intermediate storage hall above it. Furthermore, the air pressure in the hall is maintained at a higher level than that in the trenches, so air from the trenches cannot enter into the hall. Above each storage pit, the concrete slab has a hole which is sealed with a removable concrete plug.

The waste cylinders are cooled by the circulation of air through the storage pits by the ventilation systems in the intermediate storage section. The ventilation systems communicate with the atmosphere through ventilation shafts and stacks on the surface (see also under 5.2.5).

When a waste cylinder is to be deposited into a storage pit, a mobile radiation-shielded sliding door and a plug removal cask are positioned above the pit. After the plug has been lifted into the cask, the sliding door is closed and the cask with the plug is lifted away. The transfer cask containing the waste cylinder is then positioned on top of the sliding door, which is opened at the same time as the transfer cask door. The waste cylinder is





KEY PLAN

Figure 5-3. Intermediate storage. The waste cylinders are transferred to the intermediate storage section inside a transfer cask. After storage for at least 30 years, the cylinders are transferred to the encapsulation cell.

then lowered into the storage pit by means of the hoist in the cask. The doors are then closed, the transfer cask is removed and the plug is replaced by the reverse of the procedure which was used to remove it. All handling is done by the overhead crane in the intermediate storage hall, which has a lifting capacity of 35 tons.

5.2.4 Encapsulation

When the waste cylinders are to be transferred to the final repository after 30 years (or more) in intermediate storage, they are moved to the encapsulation cell by means of a procedure which is the reverse of that which was used when they were transferred from the unloading cell to the intermediate storage section.

Like other cells, the encapsulation cell is enclosed in concrete of sufficient thickness to provide radiation protection for the operating personnel.

The cell has five handling stations, each with a radiation-shielded window and a pair of master-slave telemanipulators (see Fig. 5-4).

Inside the cell, material is moved by means of a remote-controlled overhead crane with a lifting capacity of 8 tons. When it is not being used or when it requires maintenance, the crane is moved to the intervention cell through an opening which can be closed by means of a radiation-shielded door.

The waste cylinder is brought into the cell from the transport cask through an opening in the roof of the cell (which is closed by means of a radiation-shielded sliding door when it is not being used) and is placed on a wagon which serves the five handling stations.

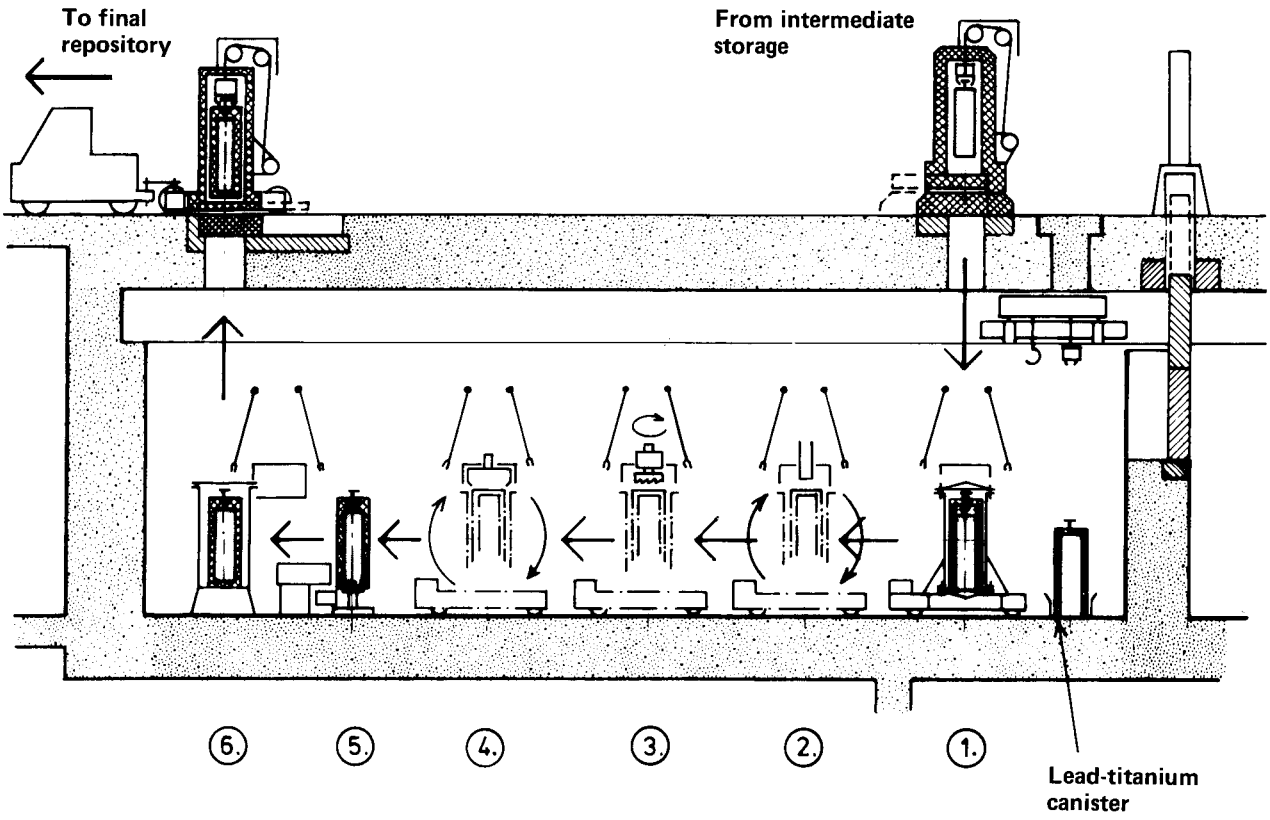
At the first station, a prefabricated part of a lead-titanium canister is placed over the waste cylinder. This part of the canister (which is fabricated outside of the facility) is brought into the cell through an opening in the roof. The opening is sealed by a concrete plug when it is not being used.

At the second station, the canister is turned over so that the waste cylinder is upside-down and molten lead is poured into the canister, filling the space between the prefabricated part and the waste cylinder as well as the space above the waste cylinder. The lead is brought into the cell from a furnace situated in a room above the cell.

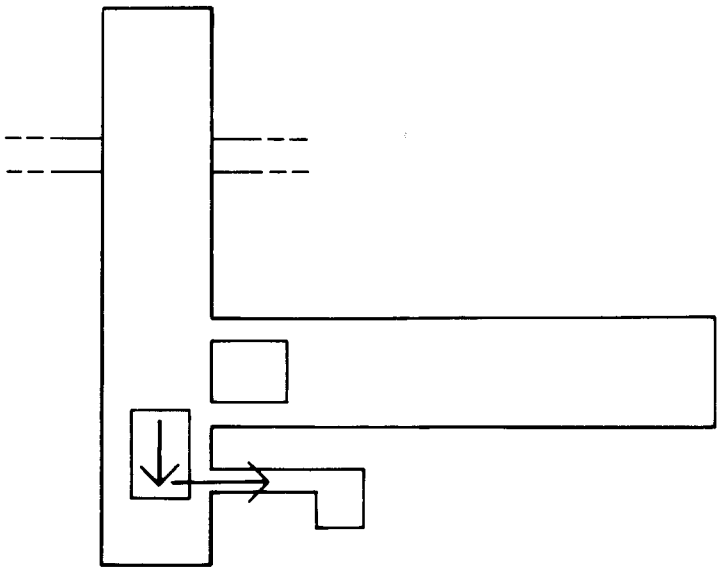
At the third station, the surface of the lead which was pured at the preceding station is machined (following cooling) in order to facilitate the attachment of a titanium lid.

At the fourth station, a titanium lid is placed on the canister by means of a remote-controlled handling device and is welded to the titanium shell of the prefabricated part by an automatic welding machine. After welding is finished, the canister is turned to the upright position again.

At the fifth station, the canister is rotated so that the lid



1 A prefabricated lead-titanium canister is placed over the waste cylinder
2 Casting of lead in the canister
3 Machining of the cast lead surface
4 Welding of titanium lid on to the canister
5 Inspection of lid weld
6 Tightness testing of canister



KEY PLAN

Figure 5-4. Schematic diagram of the encapsulation.

weld passes before an X-ray transmitter, which exposes a film in a device which permits film to be changed from the outside of the cell. The canister is then placed in a box in which its tightness is checked by means of helium under vacuum.

The finished canister (see Fig. 5-5) is then placed in position underneath an opening in the roof of the cell and is ready to be taken to the final repository.

The prefabricated part of the canister is fabricated using a lead casting technique which is also used in the fabrication of transport casks.

An alternative method of fabrication is extrusion, which is used in cable manufacture /5-3/. In this technique, a lead cover is joined to the prefabricated part by means of pressing in the cell, whereby a homogeneous lead container is obtained. In this version as well, the prefabricated part has a titanium shell and a titanium lid is welded on, after which the canister is checked

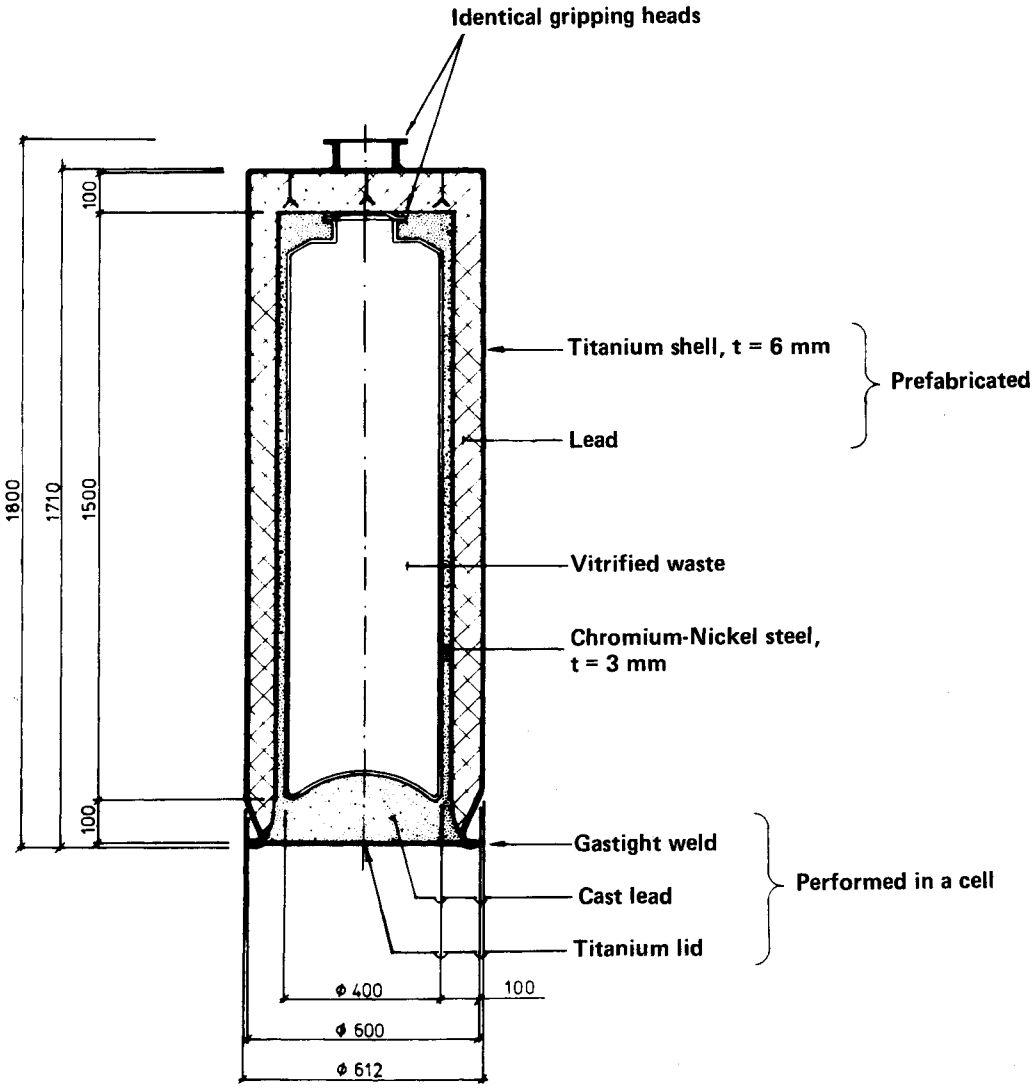


Figure 5-5. Lead-titanium canister for waste cylinders with vitrified waste. Total weight approx. 3,9 ton.

as described above. In this method, lead is not cast in the cell. The final choice of fabrication method will be made on the basis of a technical-economical evaluation.

5.2.5 Auxiliary systems

The facility will contain systems for decontamination of transport casks and waste cylinders, for floor and groundwater drainage etc. These systems are similar to the ones in a nuclear power station. "Own" low- and medium-active waste (water, filters, solid waste etc.) will be collected and sent to plants which are equipped to receive and treat such material.

Diesel-powered generators will supply auxiliary power to vital systems (ventilation, draiage etc.) in the event of an external power failure. In the event of malfunctions of pumps for the groundwater draiage system, the water will be collected in a basin with enough capacity to prevent the facility from being flooded, even in the event of an extended power failure.

A cascade of pressure differentials will be maintained by the ventilation system in the underground part of the facility in accordance with the potential risk of contamination in the various areas. The intake air will be filtered and conditioned to provide pleasant working conditions.

Air for the unloading cell and the encapsulation cell will be supplied by means of a screw device equipped with a damper and non-return valve in such a manner as to prevent positive pressure in the cell.

The ventilation system for the intermediate storage section is designed to maintain negative pressure in the storage trenches. It has a capacity of 150 000 m^3/h for each group of two trenches. The air enters through a low pressure chamber on the surface and passes through a bank of filters before entering the storage trenches.

All exhaust ventilation air from the underground part of the facility passes through absolute filters with an efficiency of 99.99% before it is released into the atmosphere. (It is possible that modifications in the design of the facility will eliminate the necessity of such filters in parts of the ventilation system.)

Air is circulated in the storage trenches by two fans, with a third fan in reserve. These three fans are located in a room next to the storage trenches. A fourth fan, with the same capacity but located on the surface for better accessibility in emergency situations, provides additional reserve capacity.

If only one fan is in operation, 65% of the air flow provided by two fans can be maintained. In the event of a failure of all fans, a bypass line with an automatic damper permits air to circulate with natural convection without passing through the filter systems.

In normal operation, the temperature of the exhaust air will be $80^{o}C$ /5-2/ when the temperature of the supply air is 20^{o} and when

the total heat produced by the waste cylinders is maximal (3 000 kW). With only one fan in operation, this temperature increases to 112°C after 40 hours. If all fans are out of operation and the storage section is being cooled by natural air convection alone, the temperature of the exhaust air will be 336°C after 40 hours. The surface temperature of the hottest cylinder will only be a few degrees higher than the temperature of the exhaust air and the centre temperature of the glass will only be 20-30° higher than the surface temperature. Since the glass does not crystallize at temperatures below 550°C (see chapter 4), the waste cylinders will not be damaged even if all fans are out of operation.

The facility has been designed in such a manner that the hot air will not cause any damage to building structures and installations. To this end, the storage trenches and ventilation ducts are lined with steel sheet with an air space between the sheet and the concrete.

5.3 CHARACTERISTICS OF ENCAPSULATION MATERIAL

5.3.1 General

In the final repository, the waste canisters are subjected to the action of the groundwater in the rock. The encapsulation material should therefore possess good resistance to such action.

The waste glass exhibits a very low leaching rate in water (see chapter 4), providing an essential barrier against the escape and dispersal of the radioactive substances. But the solubility of the glass increases with the temperature, so the glass should not come into contact with the groundwater during the period when its temperature is high due to the heat generated by the waste. An additional barrier against the escape of radioactive substances to the biosphere should also be provided during the period when the toxicity of the waste is very high (see Fig. 5-6).

However, the chromium-nickel steel container in which the vitrified waste is enclosed in the reprocessing plant is not accredited with any appreciable service life in chloridic groundwater /5-4/. Encapsulation with corrosion-resistant material is therefore necessary to prevent the glass from coming into contact with the groundwater for a long time after deposition.

A combination of lead and titanium will be used for this encapsulation. The lead also serves as a radiation shield.

5.3.2 Corrosion properties of titanium

A detailed study has been conducted concerning the suitability of titanium as a corrosion-resistant encapsulation material for vitrified high-level waste /5-5/. The study was based not only on data from the literature on the corrosion behaviour of titanium in the corrosive environment in question (which is assumed to be equivalent to Baltic seawater at a pH of 4-10 and 100°C), but also on information from prominent titanium researchers in England,

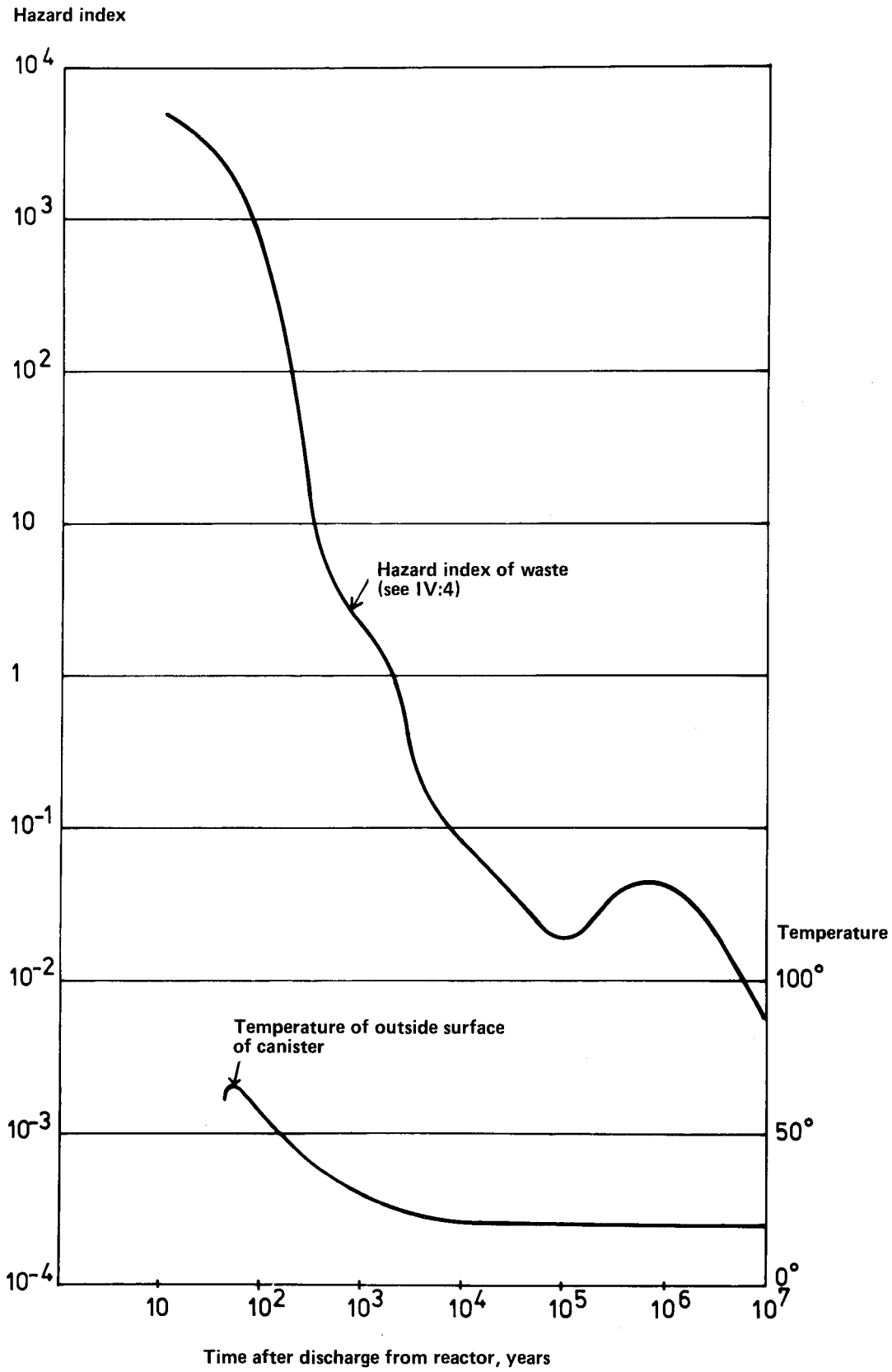


Figure 5-6. Graph showing how the toxicity of the vitrified waste (hazard index, see IV:3) and the external surface temperature of the canister vary with time. Note that the hazard index and time scales are logarithmic.

Japan, the USA and Germany. The types of corrosion which are dealt with in the study are the following:

- general corrosion, i.e. a uniformly distributed attack on the titanium surface caused either by oxidation or by a gradual dissolution of the passive film of titanium dioxide which protects the titanium
- local corrosion in the form of pitting, crevice corrosion, stress-corrosion cracking or corrosion fatigue
- hydrogen embrittlement as a result of the diffusion into the encapsulation material of hydrogen formed by corrosion or radiolysis of the water.

General corrosion

As can be seen from a pH potential diagram, the thermodynamically stable form of titanium under the storage conditions which will exist in the final repository is titanium dioxide (TiO_2). This oxide, which has the same chemical composition as the stable titanium material which occurs in nature (rutile), is formed spontaneously on the surface of the titanium (the titanium is passivated) in contact with water and protects the metal against continued corrosion. The thickness of this passive film is about 30 Å at room temperature. The passive film resists corrosion at pH 2-14, regardless of the oxygen content of the water /5-13/.

Since general corrosion in water does not normally have to be taken into consideration, the literature contains very few values for oxidation or corrosion rates at temperatures below 200°C. In one case, however, a corrosion rate of 0.25 µm/year was measured in both air-saturated and argon-saturated 3.5% NaCl solution at 60°C. This value, which was obtained under conditions which are fairly similar to those which are expected to prevail in the final repository, gives by linear extrapolation a corrosion depth of 0.25 mm over a period of 1 000 years. Another value obtained from 9 months of exposure in water from the Pacific Ocean gives a corrosion rate of 0.1 mm per 1 000 years, and the results of autoclave exposures still in progress at AB Atomenergi at 100 and 130°C give a maximum of 0.5 mm per 1 000 years /5-6 and 5-7/. The corrosion environment in the latter experiment, which has now been in progress for 100 days, is Baltic seawater adjusted to a pH of 4.5 to which 10 ppm F^- has been added.

The above corrosion data are very low and do not limit the service life of the 6 mm thick titanium casing for thousands of years. They must also be regarded as very conservative, since they have been calculated under the assumption of a constant corrosion rate over this long period of time. In actuality, the oxidation of titanium decreases with time.

Local corrosion

Of the forms of local corrosion mentioned above, corrosion fatigue can be excluded, since cyclical tensile stresses cannot occur in the canister. Stress corrosion in seawater is theoretically possible, but requires such large fracture indications and stress intensities that this type of local corrosion can also be pre-

vented if the canister is fabricated under adequate control procedures.

Pitting and crevice corrosion of titanium have been dealt with by some researchers as two separate forms of corrosion. The present study shows, however, that in actual practice, local corrosion consists mainly of crevice corrosion. The following principal criteria must be met in order for crevice corrosion to occur in titanium:

1 Very narrow crevices and a sufficiently large exposed titanium surface for the initiation of crevice corrosion via an oxygen concentration cell with the crevice as the anode and the surrounding titanium surface as the cathode.

2 A certain critical temperature which declines with rising chloride concentration and falling pH in the solution. Crevice corrosion in unalloyed titanium has not been observed below 120°C in contact with chloride solutions up to the concentration of Atlantic seawater (3.5% NaCl).

3 A mechanical roughening or cold working of the surface (for example due to scratching, grinding, etc.) appears to have an accelerating effect. This is probably due to the effect of microcrevices.

4 Contamination with base metals, especially iron, is not an absolute prerequisite, but promotes the initiation of crevice corrosion. Corresponding treatment with halogenide salts, with the exception of fluorides, is said to have a similar effect /5-8/.

The above-mentioned conditions for crevice corrosion on titanium are, of course, based on relatively short-term laboratory tests. However, nearly 25 years of experience with titanium as a design material for components which come into contact with seawater, such as heat exchangers, pipes and pumps, show that titanium's susceptibility to crevice corrosion does not increase over such a period of time. Both Swedish /5-9/ and foreign experiments /5-10/ indicate that incubation times longer than 500 hours are unlikely, probably due to an equalization of the oxygen gradient between the crevices and the surrounding titanium surface.

No local corrosion has been observed after 100 days of testing at 100-130°C in acidified (pH 4.5) Baltic seawater despite narrow crevices and scratching of the surface with iron /5-8/.

For the above-cited reasons, unalloyed titanium can be expected to have a very long service life (at least thousands of years), even when local corrosion is taken into consideration. In choosing the site for the final repository and in designing the manner of storage, appropriate measures will be taken to make sure that extremely high chloride levels in the groundwater are avoided. The system for watering the storage holes which is described in greater detail in section 6.2.3 is designated to eliminate the risk of salt enrichment (due to evaporation) during the period immediately following deposition and to keep the temperature of the canister at an adequately low level (approx. 65°C).

Hydrogen embrittlement

The risk of hydrogen embrittlement caused by hydrogen - created by radiolysis of water or by corrosion - diffusing into the titanium and causing hydride formation has been thoroughly investigated.

The total quantity of hydrogen generated by radiolysis of water over a period of 10 000 years has been estimated to be on the order of 10^{-5} g/cm^2, which corresponds to an increase of only about 4 ppm of the original hydrogen content of the titanium of 10-20 ppm. Hydrogen due to corrosion can only be formed after crevice corrosion has been initiated and has started to grow. If this extremely improbable situation arises, a local hydration will be of subordinate importance compared to the damage which has already occurred.

Thus, the presence of hydrogen is judged to be negligible, and even if a sufficient quantity of hydrogen should come into contact with the titanium surface, diffusion data show that it would take hundreds of years before the titanium would be hydrated to brittleness.

In order to guarantee that the original hydrogen content of the titanium cannot be enriched to stress concentrations, and thereby lead to delayed fracture over the long run, a hydrogen content of max. 20 ppm has been specified for the titanium used in the canister. This value corresponds to the solubility of hydrogen in titanium at room temperature and thereby renders hydride precipitation impossible.

5.3.3 Corrosion properties of lead

Like titanium, lead depends for its corrosion resistance on the formation of a protective film on the surface which impedes or prevents further corrosion. The composition and properties of the protective film depend on the nature of the surrounding medium. In a suitable environment, the film can exhibit considerable resistance to corrosion. Obviously, lead corrosion is not even a possibility until the titanium casing has been penetrated.

Besides resisting corrosion, the lead in the canister functions as a radiation shield which reduces the radiation level outside the canister to such a low level that it is of no practical importance to the corrosion behaviour of the titanium. Calculations show that concentrations of oxidizing agents induced by radiation are very low /5-11/.

Since the lead is protected by the titanium casing, general corrosion can be disregarded.

If the titanium casing is penetrated, however, some local corrosion may be expected on the lead surface which is thereby exposed.

This corrosion will thereby be highly localized and will develop in the form of pitting. If it is assumed that the reaction will be limited by the available supply of oxidants - oxygen in the surrounding water, radiolysis products etc. - the amount of lead

which can go into solution is 1.24 kg per 1 000 years and metre of canister length, i.e. slightly more than 2 kg per canister (total weight 3 900 kg). The corrosion attack will penetrate into the lead at a diminishing rate. It is tentatively estimated that pitting will penetrate the lead lining after about 500 years, but this figure is probably grossly underestimated /5-11/.

5.3.4 Summary

The Swedish Corrosion Research Institute was commissioned by KBS to investigate the corrosion resistance of the proposed encapsulation materials. The institute in turn appointed a reference group composed of specialists within the field of corrosion and materials to conduct the study.

In a status report dated 27 September 1977 and reproduced in KBS Technical Report No. 31 /5-12/, the institute and its reference group submitted the following assessment of the service life of the lead-titanium canister:

> "The corrosion resistance of the titanium casing is based entirely on the existence of a protective passivating film. Under the conditions prevailing in a final repository, this film has a self-healing capacity in the event of damage of limited extent. Under the assumed circumstances and on the basis of current knowledge, the titanium casing should have a service life of more than 1 000 years. However, this estimate is subject to a certain degree of uncertainty in that previous experience of pitting and crevice corrosion in titanium comes from experiments and applications of relatively (in this context) short duration. In order to reduce the risk of local corrosion, the storage site and storage method should be selected to avoid the possibility of extremely high levels of Cl^- in the groundwater.
>
> If the titanium casing is penetrated as a result of mechanical damage or local corrosion, the lead lining thus exposed may be attacked by galvanic corrosion. The rate of this corrosion is determined by the supply of oxygen and other oxidants which are present in the groundwater or are created by radiolysis as well as by corrosion-inhibiting constituents in the water, for example hydrogen carbonade ions. In contact with the postulated storage environment, it has been concluded that the lead lining will greatly prolong the service life of the canister.
>
> The service life of the lead-lined titanium canister is currently estimated by some members to be at least 1 000 years, while other members estimate the service life to be at least 500 years. Before a final assessment is made, further study should be conducted in this area."

The conclusions of the Corrosion Research Institute are supported unanimously by the specialists in the reference group. Supplementary statements by members of the reference group have also been appended to the status report.

In one of the supplementary statements, it is claimed that the estimates given in the status report are conservative and repre-

sent a lower limit for the durability of the encapsulation material.

It is furthermore submitted that on the basis of existing knowledge, it is highly probable that further study will reveal a considerably longer life for the encapsulation material. KBS shares this opinion.

5.4 OPERATION OF FACILITY

Only some 30-40 persons will be required for the operation of the facility.

The entire underground part of the facility is classified as a controlled area and is divided into zones according to the potential risk of contamination, in the same manner as in a nuclear power station.

All handling of waste cylinders is done by remote control when the cylinders are in radiation-shielded cells or with the aid of a radiation-shielded transfer cask. If a power failure should render motor-driven equipment inoperable, the work can be done manually.

All cells are connected to an intervention cell to which all equipment in the cells can be transferred via remote control and in which minor repairs can be effected. If major repairs are required, the equipment can be decontaminated and taken out of the intervention cell into a metal-lined room situated above the cell. From here, the equipment can be sent away for repair.

The facility's operating systems are based on existing technology and on experiences from similar systems in existing facilities.

The facility will be under the supervision of auhorities such as the Swedish Nuclear Power Inspectorate and the National Institute of Radiation Protection in the same manner as a nuclear power station. The facility will be designed in compliance with the regulations issued by these authorities and by the occupational safety authorities and in consultation with concerned personnel organizations.

With regard to working environment and safety, see III:7.

5.5 QUALITY CONTROL

In order to satisfy the stringent requirements on safety and operational availability which are imposed on the activities described here and in order to ensure absolute safe final storage, the quality of plant and material must conform to a sufficiently high standard. This requires effective quality assurance, which entails that all measures aimed at achieving and maintaining the necessary level of quality shall be planned, systematic and documented.

The owner of the facility shall also be responsible for ensuring that quality control and quality assurance activities are organized and executed in a satisfactory manner. The execution and

documentation of various quality-guaranteeing measures should be divided between the owner and an official institution, such as the Swedish Plants Inspectorate, in a manner similar to that which is followed in the case of nuclear power plants. This division shall be based on competence and on safety considerations and shall be approved by the Swedish Nuclear Power Inspectorate (SKI). Responsibility for coordinating such activities shall rest with the owner, who shall also submit periodic reports to SKI.

The owner shall also submit a report to SKI, in good time before the start of construction, specifying a programme for the organization and functions of quality control and quality assurance. Supplementary instructions shall subsequently be issued as required and the programme shall be subjected to continuous follow-up by SKI. The programme shall include the following points:

- Definition of the application of the programme to various building sections and installations based on safety classes.
- Description of the owner's organization and cooperating organizations, with specification of areas of responsibility and channels of contact.
- Directives for design examination. Designs should be examined by an independent body.
- Purchasing directives with respect to quality requirements.
- Inspection and identification of purchased material.
- Production and installation control appropriate to the importance of the product for plant safety and operational availability.
- Programme for recurrent periodic testing and inspection of certain plant components.
- Directives for operation and maintenance of the facility, including comprehensive instructions for abnormal operational situations and events.
- Routines for the submitting of reports to the supervisory authority.

A quality control plan for the vitrified waste and the canister should include the following points:

Glass body:

- Compositional analysis
- Hardness testing
- Leaching test

Chromium-nickel cylinder:

- Compositional analysis of material
- Tensile testing of material
- Material identification
- Dimensional check of material
- Welding procedure check
- Supervision of welding work
- Visual and dimensional inspection of welds
- Penetrant testing of welds before and after filling
- Identification of cylinder material before filling
- Visual and dimensional inspection of cylinder before filling
- Marking and issuing of test certificate

Titanium-lead canister:

- Compositional analysis of titanium and lead
- Tensile testing of titanium
- Dimensional check after first lead-casting
- Visual inspection of final surface
- Pressure testing of cylinder after first lead-casting
- Welding procedure check
- Purity control prior to welding
- Supervision of welding work
- Penetrant testing of welds
- Tightness testing by means of He after sealing
- Marking and issuing of test certificate

Some of these quality control procedures may take the form of random sample tests, the frequency of which shall be determined on the basis of the probability of defects.

Quality control which is related to the glass body and the chromium-nickel cylinder will be performed at the foreign reprocessing plant. The extent to which quality control and quality assurance shall be carried out by the manufacturer, an independent quality control institution or the owner shall be determined in consultation between the parties involved and the supervisory authority. The manufacturer is expected to provide sample material so that the owner can perform his own tests in Sweden.

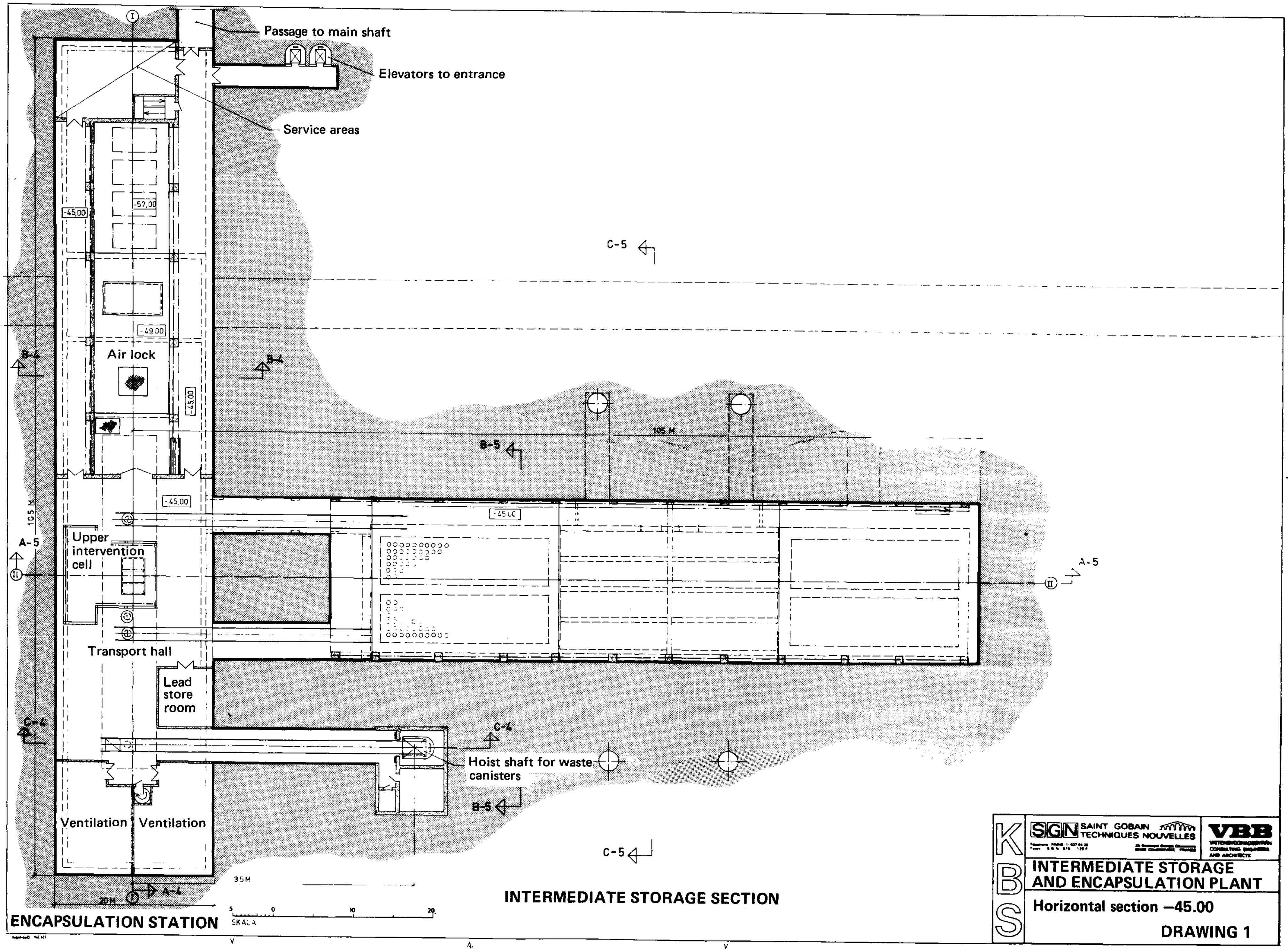
5.6 DECOMMISSIONING

When the facility is no longer required and there are no waste cylinders or canisters left in it, the facility shall be decontaminated and all "own" radioactive waste, contaminated scrap and building materials shall be taken away to facilities which are equipped to receive and treat such materials. The facility can then be modified for other use or sealed by filling with crushed rock, concrete etc.

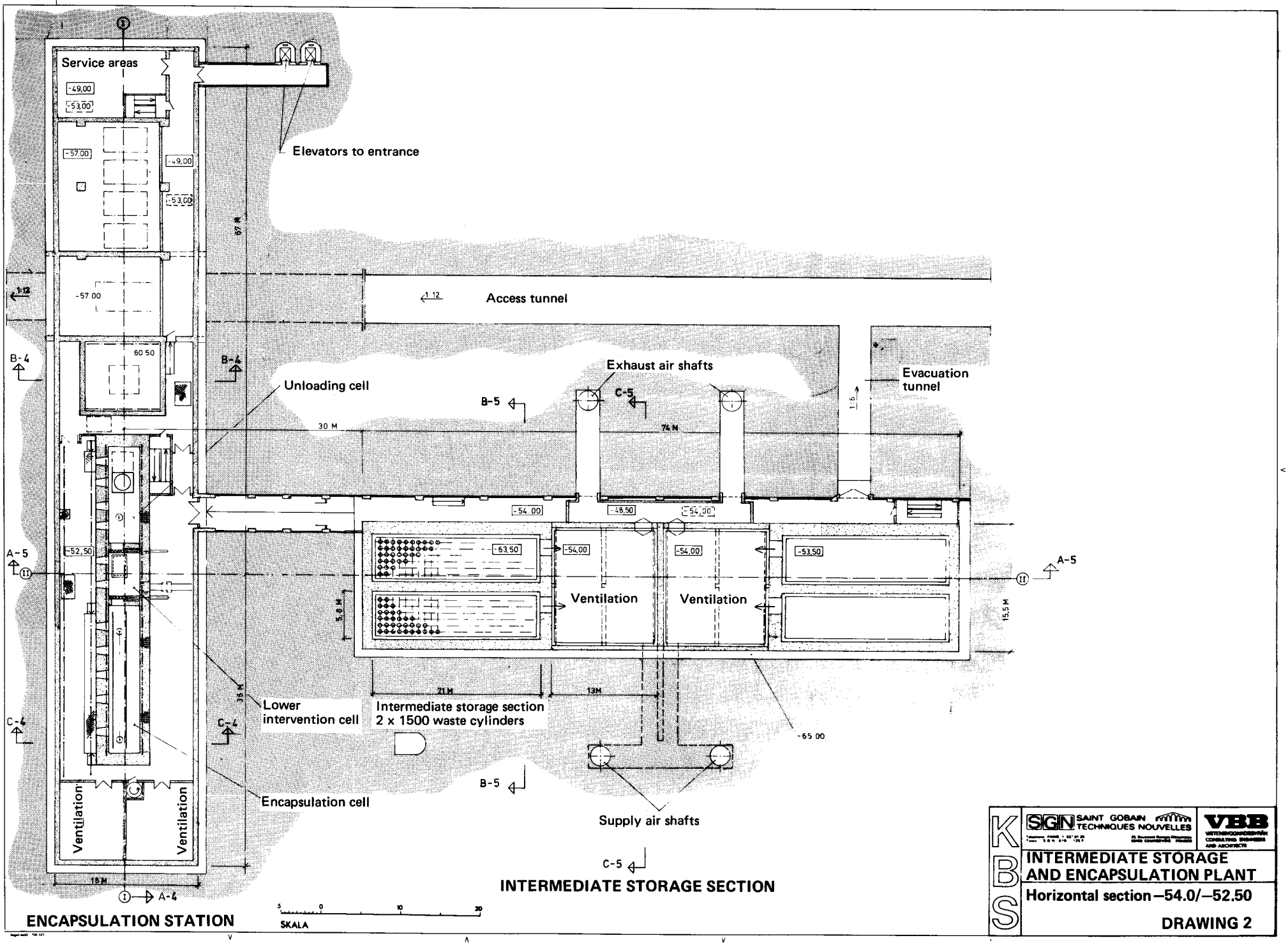
In general, very little contamination can be expected, so decommissioning should not present any difficult problems.

5.7 DRAWINGS

DRAWING	1	HORIZONTAL SECTION -45.0
"	2	HORIZONTAL SECTION -54.0/-52.5
"	3	HORIZONTAL SECTION -60/-59.5
"	4	LONGITUDINAL SECTION I-I AND CROSS-SECTIONS
"	5	LONGITUDINAL SECTION II-II AND CROSS-SECTIONS
"	6	PROCESS FLOW CHART



Passage to main shaft
Elevators to entrance
Service areas
-45,00
-57,00
-49,00
Air lock
-45,00
C-5
B-4
B-5
105 M
-45,00
Upper intervention cell
A-5
Transport hall
Lead store room
C-4
Hoist shaft for waste canisters
Ventilation
Ventilation
A-4
20M
35M
SKALA
ENCAPSULATION STATION
INTERMEDIATE STORAGE SECTION
KBS
SGN SAINT GOBAIN TECHNIQUES NOUVELLES
VBB
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT
Horizontal section −45.00
DRAWING 1



KBS

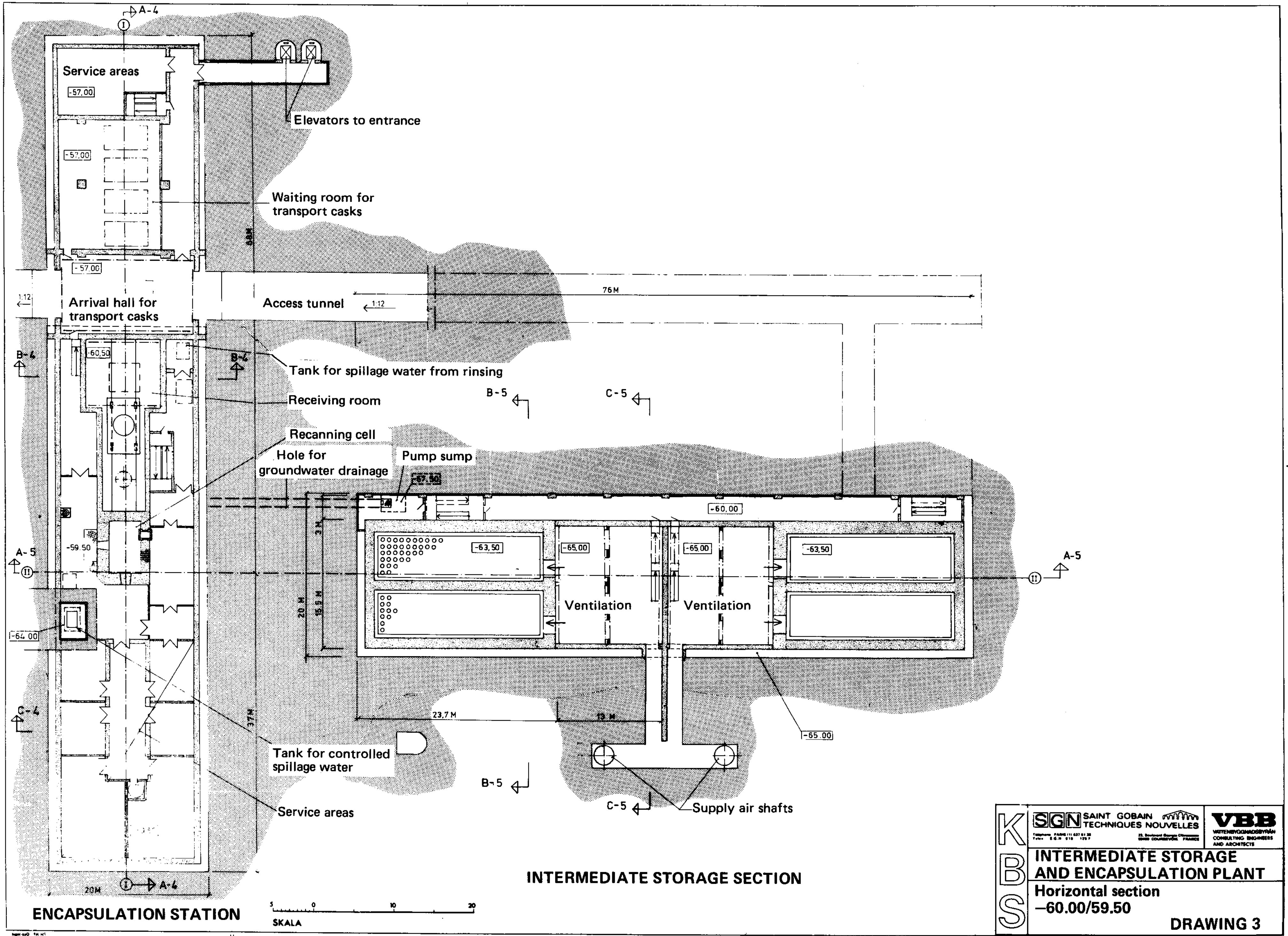
SGN SAINT GOBAIN TECHNIQUES NOUVELLES

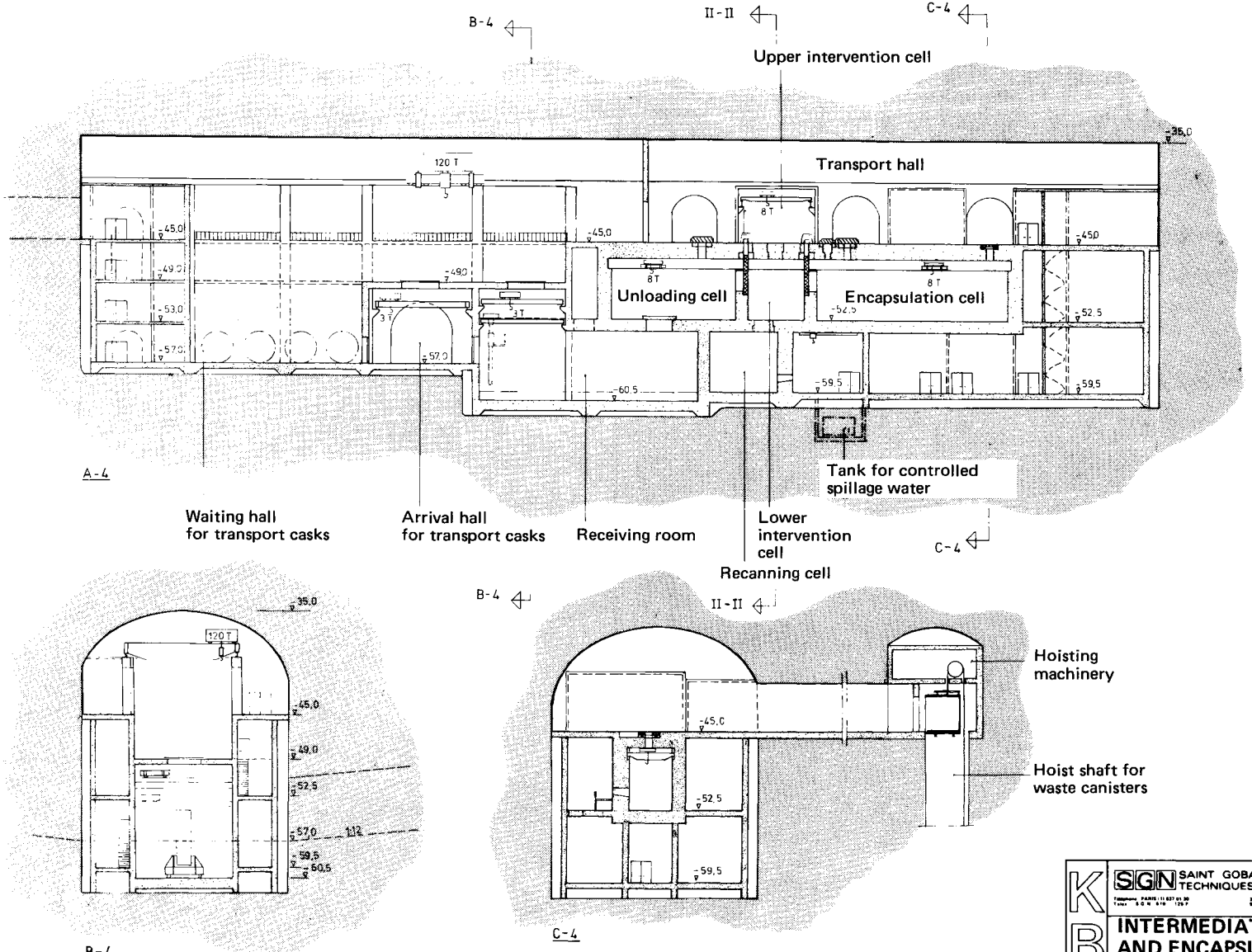
VBB

INTERMEDIATE STORAGE AND ENCAPSULATION PLANT

Horizontal section –54.0/–52.50

DRAWING 2


Service areas
Elevators to entrance
Waiting room for transport casks
Arrival hall for transport casks
Access tunnel
76M
Tank for spillage water from rinsing
Receiving room
Recanning cell
Hole for groundwater drainage
Pump sump
-60.00
-63,50
-65,00
Ventilation
Ventilation
-65,00
-63,50
23,7 M
13 M
-65.00
Tank for controlled spillage water
Supply air shafts
Service areas
20M
ENCAPSULATION STATION
SKALA
INTERMEDIATE STORAGE SECTION
KBS
SGN SAINT GOBAIN TECHNIQUES NOUVELLES
VBB
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT
Horizontal section
−60.00/59.50
DRAWING 3



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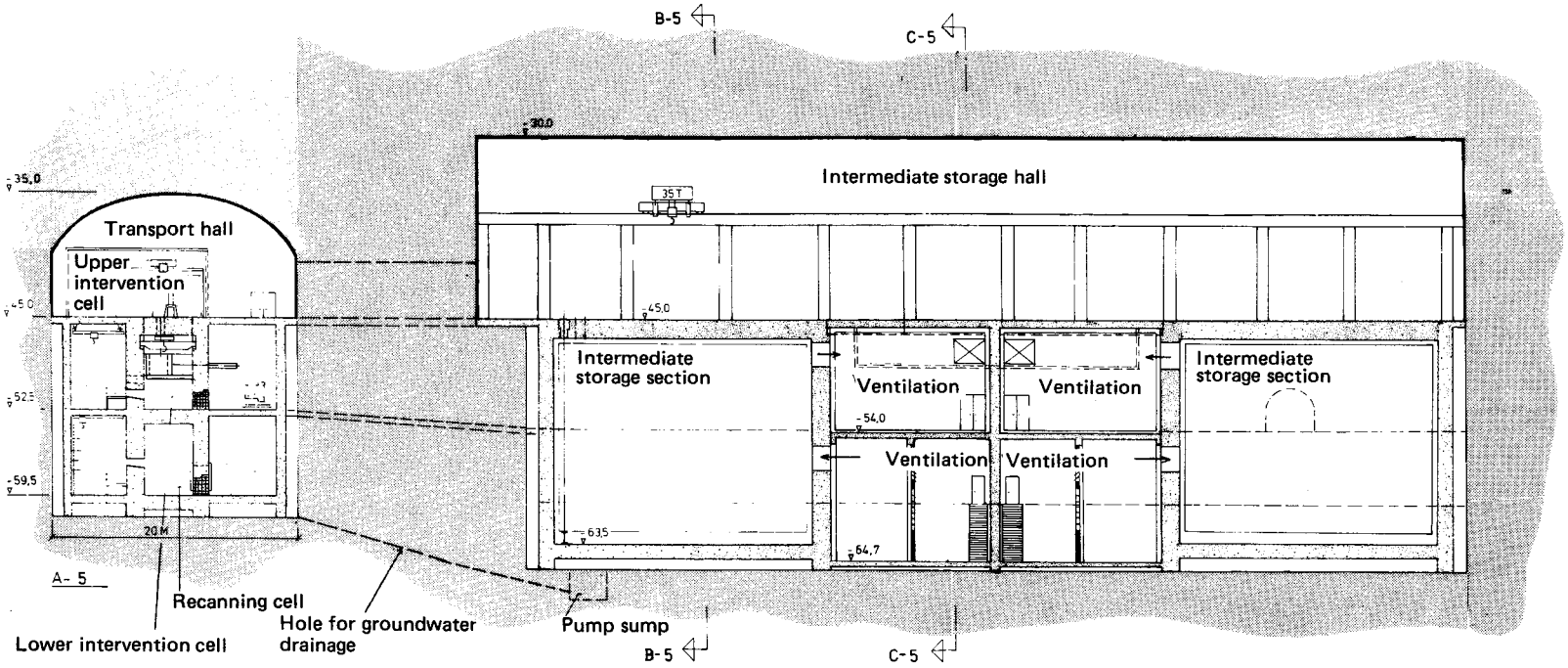
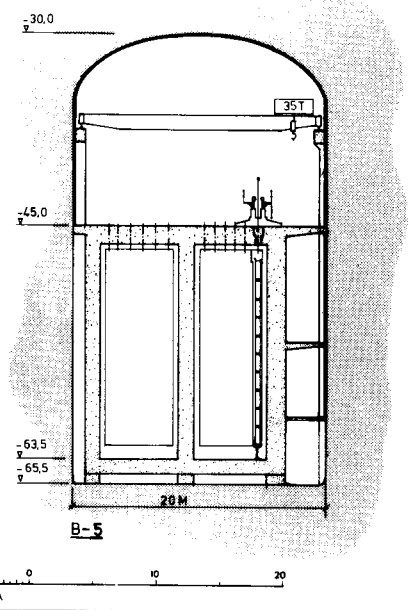
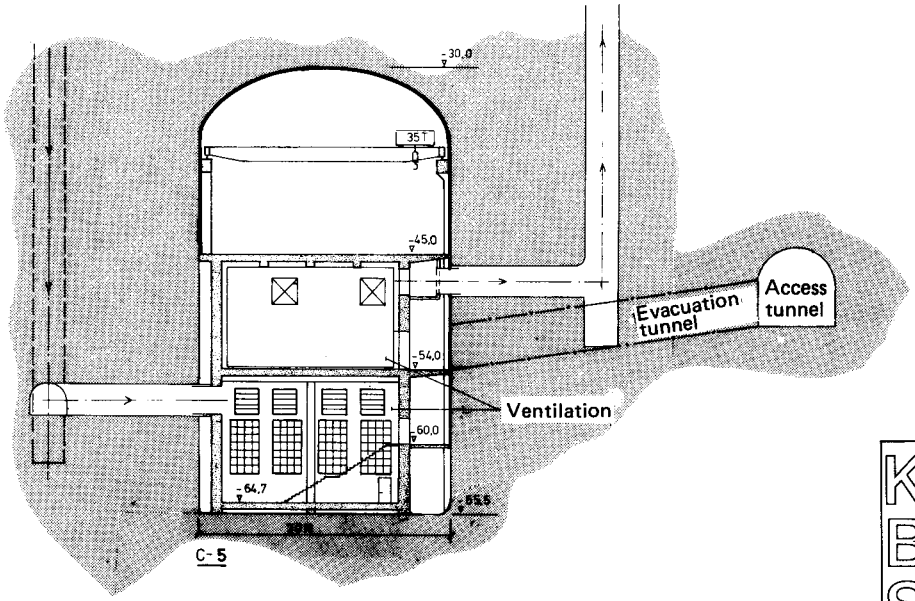
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT

Longitudinal section I—I and cross-sections

DRAWING 4







5 0 10 20
SKALA

KBS

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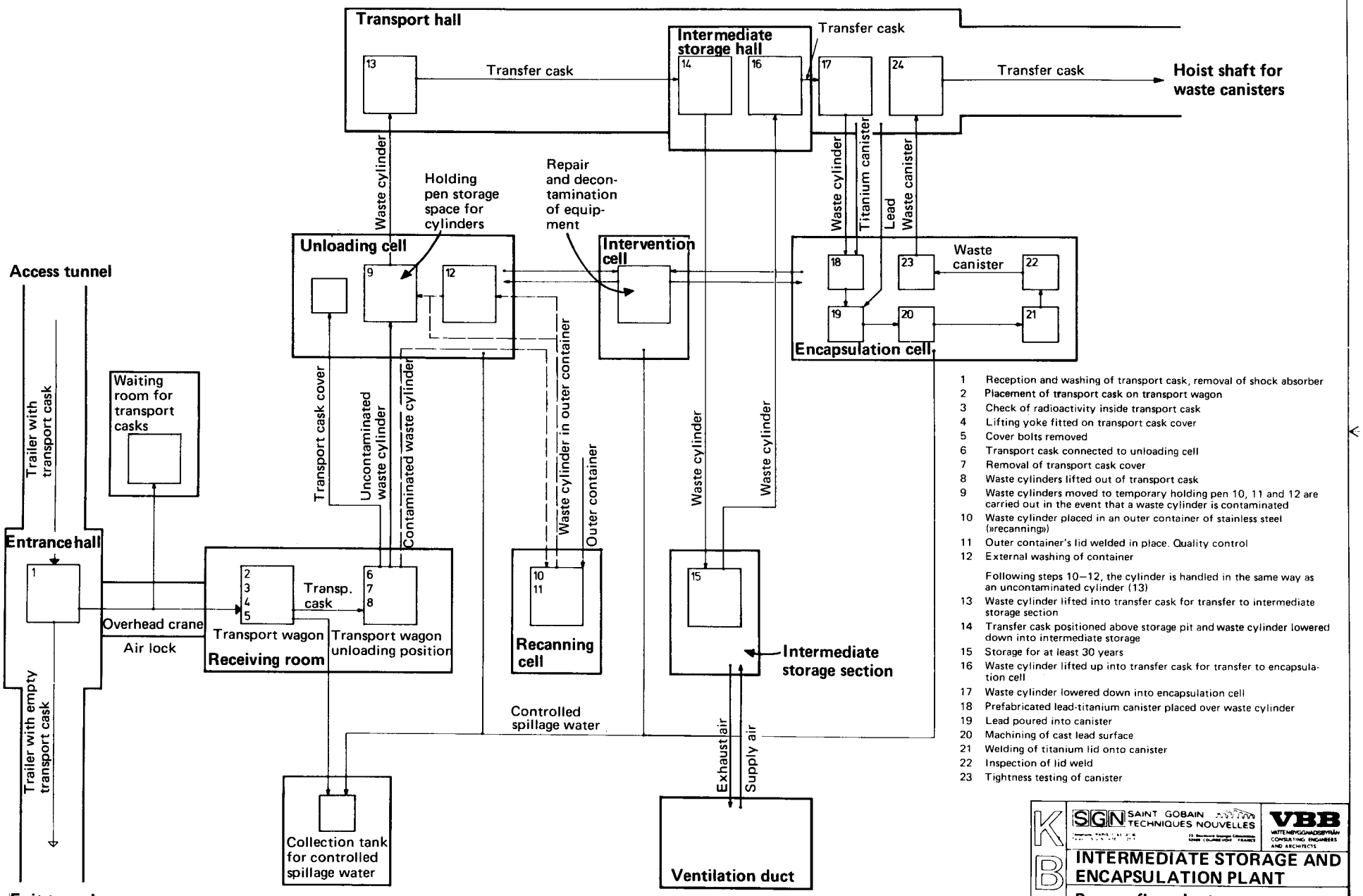
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT

Longitudinal section II–II and cross-sections

DRAWING 5



1 Reception and washing of transport cask, removal of shock absorber
2 Placement of transport cask on transport wagon
3 Check of radioactivity inside transport cask
4 Lifting yoke fitted on transport cask cover
5 Cover bolts removed
6 Transport cask connected to unloading cell
7 Removal of transport cask cover
8 Waste cylinders lifted out of transport cask
9 Waste cylinders moved to temporary holding pen 10, 11 and 12 are carried out in the event that a waste cylinder is contaminated
10 Waste cylinder placed in an outer container of stainless steel (»recanning»)
11 Outer container's lid welded in place. Quality control
12 External washing of container

Following steps 10–12, the cylinder is handled in the same way as an uncontaminated cylinder (13)

13 Waste cylinder lifted into transfer cask for transfer to intermediate storage section
14 Transfer cask positioned above storage pit and waste cylinder lowered down into intermediate storage
15 Storage for at least 30 years
16 Waste cylinder lifted up into transfer cask for transfer to encapsulation cell
17 Waste cylinder lowered down into encapsulation cell
18 Prefabricated lead-titanium canister placed over waste cylinder
19 Lead poured into canister
20 Machining of cast lead surface
21 Welding of titanium lid onto canister
22 Inspection of lid weld
23 Tightness testing of canister

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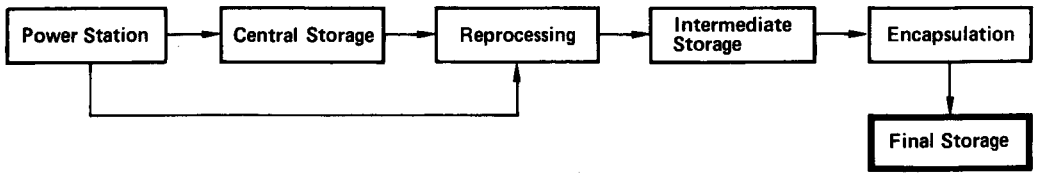
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT

Process flow chart

DRAWING 6

6 FINAL STORAGE



6.1 GENERAL

The final repository is situated in rock underneath the facility for intermediate storage and encapsulation at a depth of approximately 500 metres below the surface.

The main criterion for the design of the final repository is that it shall be possible to seal and finally abandon the facility and still retain its fundamental function: to prevent the escape of radioactive substances to the biosphere.

The studies which have been conducted of possible sites for a final repository (see volume II, Geology) have indicated that it is possible to fulfil this criterion. Bedrock and groundwater conditions at the investigated sites have proved to be such that the rock will constitute a barrier to the migration of the radioactive substances from the waste to the biosphere. The vitrified waste itself, with its low leaching rate, the canister, with its high resistance to corrosion, and the buffer material with which the storage holes, tunnels and shafts in the final repository are filled, with its special isolating properties, constitute additional barriers to such a migration. The rock also provides protection against external forces, such as acts of war, sabotage, meteorite impact etc. An evaluation of the function of the various barriers is provided in volume IV, Safety Analysis.

The final repository has been designed for the deposition of 9 000 waste canisters and on the basis of the assumption that 300 canisters will be transferred to the final repository each year from the intermediate storage facility and encapsulation station. The design of the facility is based on existing technology. For a more detailed description of the facility, see the drawings at the end of this chapter.

6.2 DESCRIPTION OF FACILITY

6.2.1 Layout

The final repository consists primarily of a system of parallel storage tunnels located approximately 500 metres below the surface, with appurtenant transport and service tunnels and shafts for communication with the surface and with the facility for intermediate storage and encapsulation. The tunnel system also in-

cludes diverse service areas (see Fig. 6-1). The encapsulated waste is deposited in vertical holes drilled in the floors of the storage tunnels.

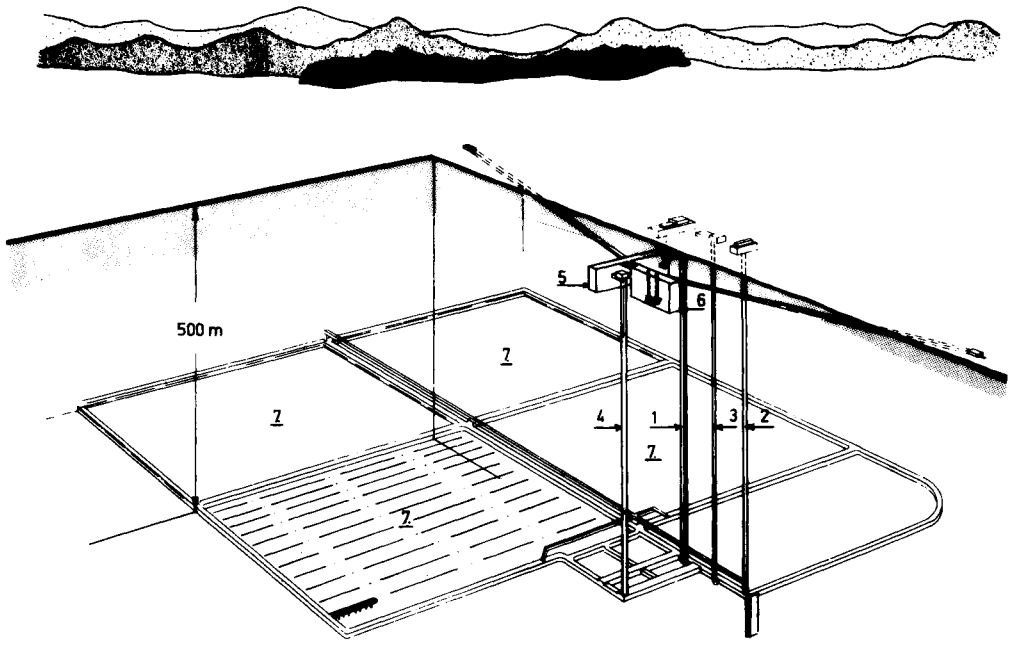
6.2.2 Design and construction of rock cavern facility

After introductory design work and preliminary studies, which may include a pilot plant, work on the rock facility will commence with the sinking of a shaft from the surface of the ground down to the level of the repository. From this shaft, drifts will be driven which will permit the excavation of other shafts by the driving of raises /6-1/.

Tunnels for service areas will be constructed next to the repository. Blast rubble will be transported to the surface via a skip. The material will be crushed first, however, since the size of the blocks which a skip can accomodate is limited.

Blasting of the repository's tunnel system will start with the hoist tunnels at the periphery and in the centre and the ventilation tunnel situated above the mid-tunnel. These tunnels will provide good general information on the characteristics of the rock at the site so that the layout of the storage tunnels can be modified if necessary in order to avoid sections of poor rock not indicated by the preliminary studies.

Blasting of the storage tunnels is then commenced with great care so as to disturb the surrounding rock as little as possible. Ver-



1 Main shaft
2 Skip shaft
3 Ventilation shaft
4 Hoist shaft for waste canisters
5 Receiving and encapsulation station
6 Intermediate storage facility
7 Final repository

Figure 6-1. Perspective drawing of final repository with plant for intermediate storage and encapsulation. The final repository consists of a system of parallel storage tunnels situated 500 m below the surface.

tical deposition holes are drilled in the floors of these tunnels for the waste canisters. Before such a hole is drilled to its full size, a smaller hole is first drilled in which the permeability of the surrounding rock is determined by means of water injection tests. If permeability is found to be sufficiently low, the deposition hole is then drilled to full size. If permeability is too high, the rock is grouted and the hole is tested again. If permeability is still too high, the hole is plugged with a mixture of sand and bentonite and the site is not used for the deposition of a waste canister. If permeability is sufficiently low, the deposition hole is drilled to full size. No hole will be drilled near faults and other joint planes in the rock.

Electrical equipment will be used for tunnel construction in order to minimize air pollution. Diesel-driven service vehicles may, however, be used. The work will be executed using conventional mining and construction methods.

The centre-to-centre spacing of the storage tunnels (25 metres) and of the deposition holes in the tunnels (4 metres) has been determined on the basis of rock mechanics considerations, including the effects of the heat generated by the canister. The deposition holes have a diameter of 1 m and a depth of 5 m. Each hole is intended for 1 canister. With the spacing selected, the Gross Thermal Loading in the initial phase will be 5.25 watts per m^2, which results in a relatively moderate increase in temperature in the surrounding rock formation (see Fig. 6-2). The effects of this heating at the surface on the climate, land elevation etc. will scarcely be noticeable /6-2/.

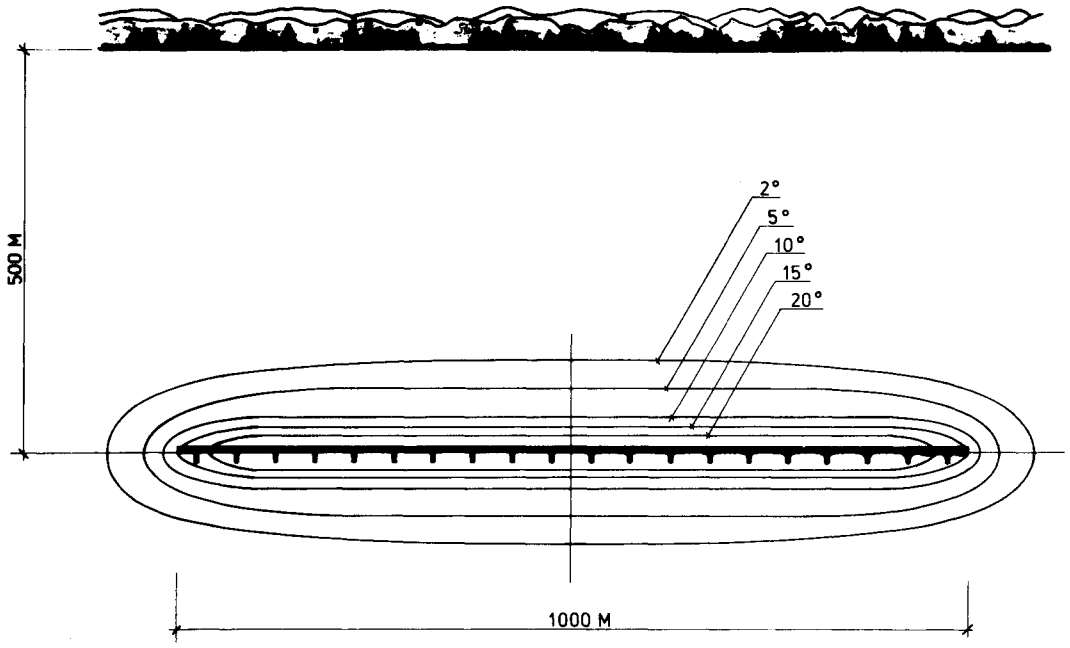
See I:14 for the schedule for the construction of the facility.

6.2.3 Deposition of waste canisters

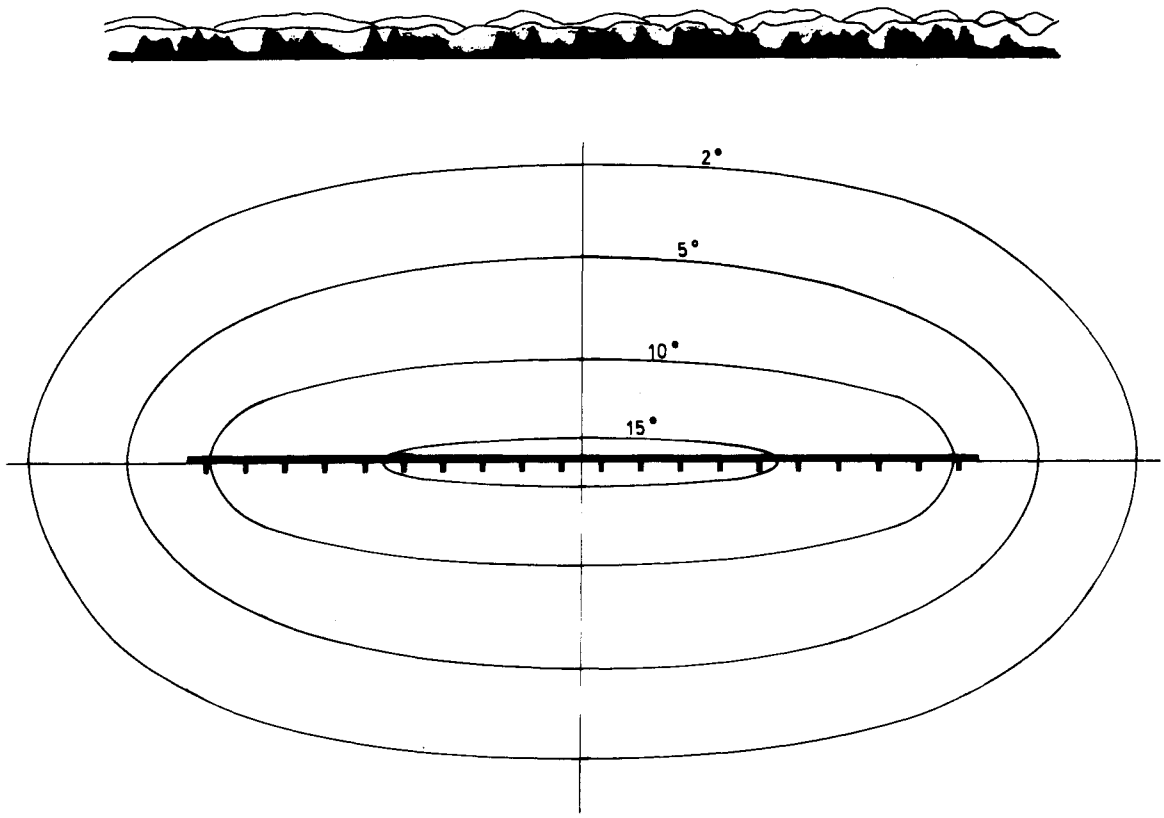
When a waste canister is to be transferred from the facility for intermediate storage and encapsulation to the final repository, it is first lifted out of the encapsulation cell into a radiation-shielded transfer cask by means of a procedure similar to that used for handling of the waste cylinders in the facility for intermediate storage and encapsulation. The cask is similar to the one described under 5.2.3, but the lead jacket is only 10 cm thick, since the radioactivity of the waste is lower following storage and since the lead in the canister provides the additional radiation protection which is required. The transfer cask is carried on a railbound wagon which is drawn by an electric tractor /6-3/.

The transfer cask is taken via a horizontal tunnel from the encapsulation cell to an elevator which runs in a vertical shaft in the rock. The elevator takes the transfer cask down to the level of the storage tunnels (see Fig. 6-3).

The elevator is of the same design as a conventional mine elevator with guides and a winding sheave and with a number of independent braking systems. The elevator cage is suspended from a number of cables which are strong enough so that a few cables alone can support the load with a good margin of safety (10-fold safety). As an additional safety precaution, there is a water pool at the bottom of the hoist shaft which dampens the impact of



50 years after deposition



600 years after deposition

Figure 6-2. Temperature increase in the rock formation around the final repository 50 and 600 years after deposition.

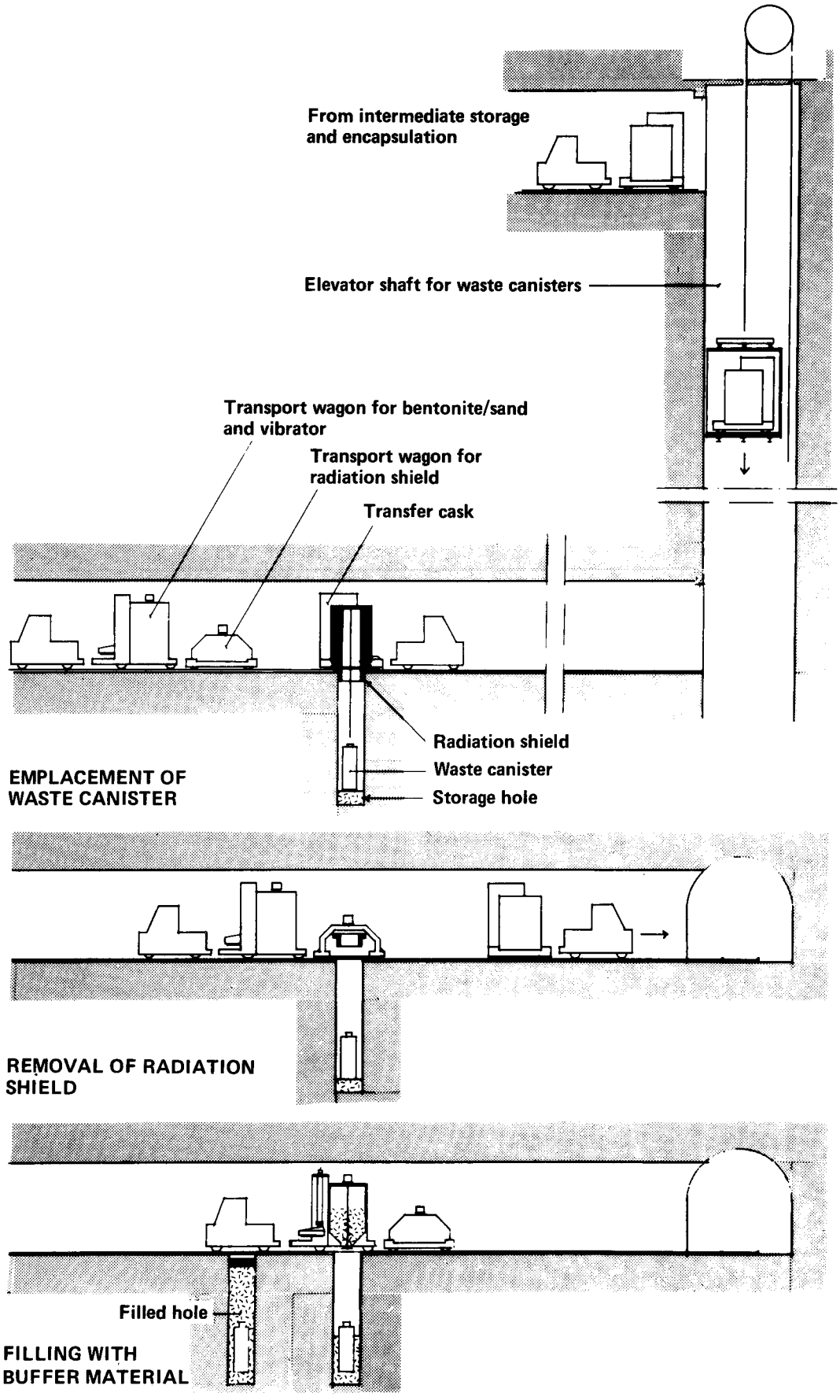


Figure 6-3. Handling of waste canisters in final repository.

a falling elevator and which provides radiation shielding in the event that a canister should be damaged.

When the elevator has reached the level of the repository, the transfer cask is taken on its wagon through the tunnel system and positioned above the hole in which the waste canister is to be deposited.

Before the transfer cask is moved into position, the deposition hole is first drained of any water and titanium irrigation pipes (see below) are installed. A bed of sand (90%) and bentonite (10%) is then deposited at the bottom of the hole. The bed is compacted by means of a hydraulically operated vibrator plate. Finally, a mobile radiation shield is positioned at the opening of the hole to protect the personnel when the canister is being lowered into the hole.

The canister is now lowered into the hole by the hoist inside the transfer cask and deposited on the sand/bentonite bed. The transfer cask is then moved away, the mobile radiation shield is removed and the hole is filled with a mixture of sand (85%) and bentonite (15%). The fill is deposited and compacted in layers 10-20 mm thick by means of hydraulically operated equipment, see Fig. 6-4 and /6-4/.

The proportions of sand and bentonite are determined by such consideration as the fact that a higher bentonite content provides lower permeability but also lower bearing strength. The bottom bed must be able to support the weight of the canister and should

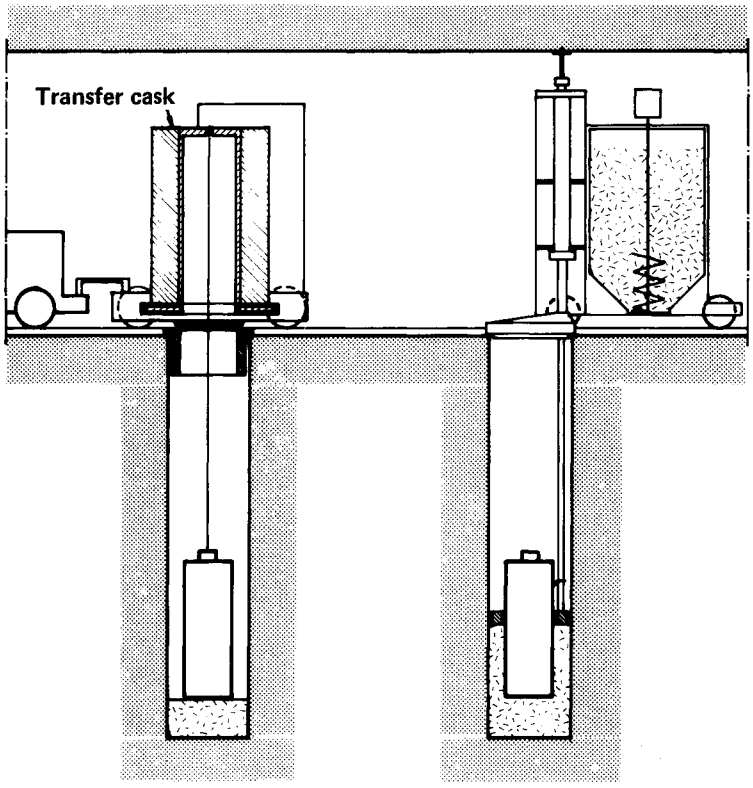


Figure 6-4. Deposition of canister (at left) and filling of deposition hole (at right). The filler material consists of a mixture of quartz sand and bentonite.

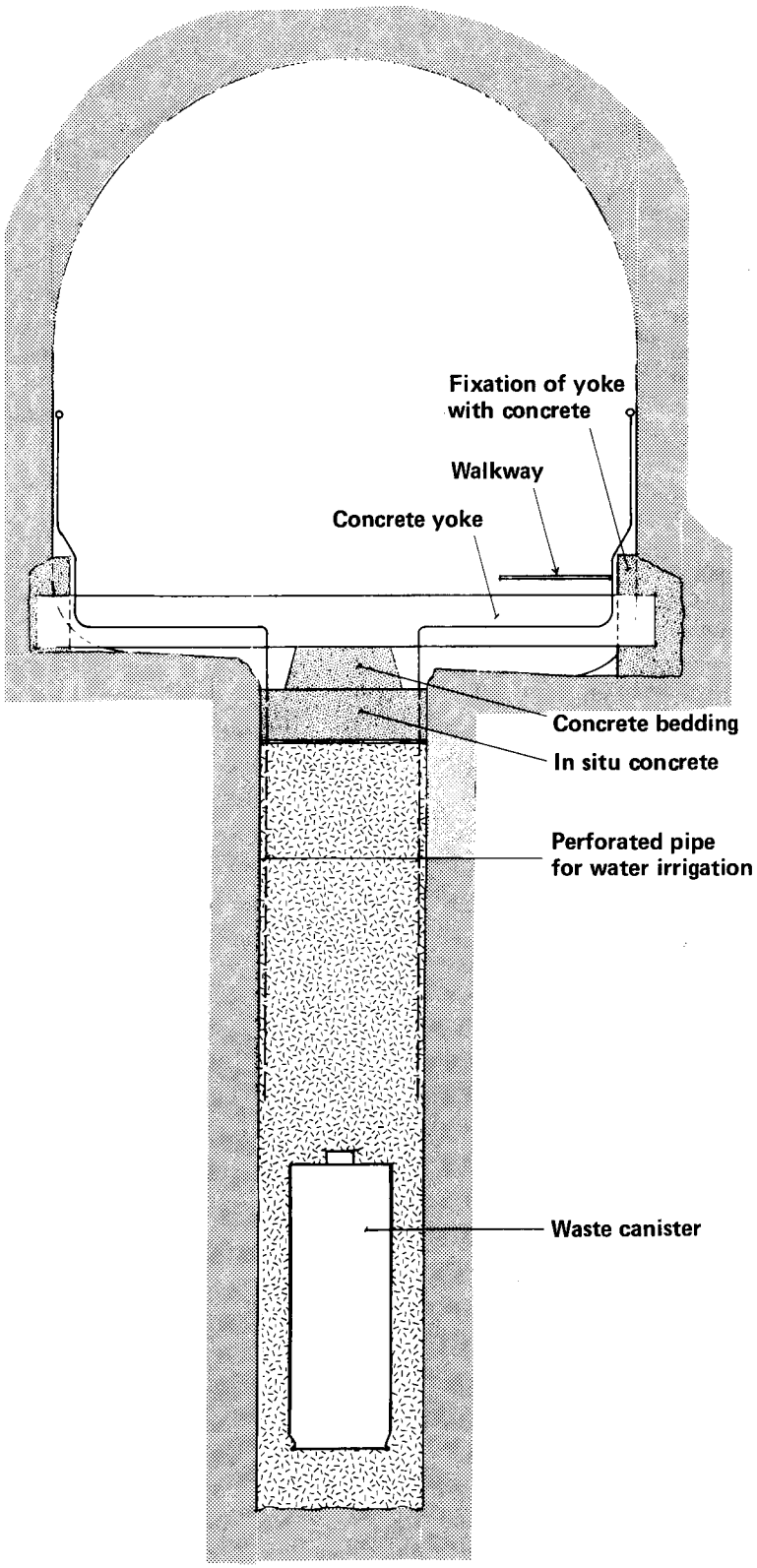


Figure 6.5. After backfilling with a mixture of quartz sand and bentonite, the deposition hole is sealed with a lid of cast-in-situ concrete. A concrete yoke is anchored above the lid.

therefore have a lower bentonite content than the rest of the fill, for which lowest possible permeability is the primary criterion. See further under 6.3 below.

After backfilling is concluded, the deposition hole is sealed with a lid of cast-in-situ concrete. A prefabricated concrete yoke is placed on top of the lid and grouted in recesses in the tunnel walls (see Fig. 6-5). The irrigation pipes are connected to a piping system through which water is injected into the hole. The purpose of the lid is to prevent water from seeping out or evaporating from the hole. It also prevents the fill from swelling when the bentonite absorbs the water. This further compacts the fill, making it denser (see 6.3). The filler material provides adequate radiation protection for the personnel who work in the storage tunnels.

The system for irrigation of the deposition holes is maintained for the entire period of time during which the repository is open. During this period, the facility is drained and ventilated and the irrigation system and the lid over the hole prevent the filler material in the holes from drying out due to the heat generated by the canisters. This would reduce the thermal conductivity of the filler material and thereby increase the temperature of the canister (which is about 65°C when the fill is water-saturated, /6-2/). Drying of the filler material could also lead to an enrichment of the salt in the groundwater. The increase in temperature and the salt enrichment would have a negative effect on the resistance of the canister to corrosion (see III:5.3). However, it is possible that further study and analysis of the effects of heat generation on the filler material and the groundwater will show that the irrigation system described here can be simplified or dispensed with altogether.

6.2.4 Auxiliary systems

The facility will contain auxiliary systems for water supply, sewerage, electric power, conpressed air, fire protection, telecommunications, transport of personnel and material etc. These systems are similar to those in conventional mining installations.

The ventilation system is designed to provide a free circulation of air in tunnels and shafts /6-1/. It shall supply the facility with fresh air and remove dust, fumes and gases from blasting and from vehicles. The air temperature shall be maintained at a pleasant level in all areas occupied by personnel.

Radioactive contamination of the air is not expected to occur. Even in the event of a handling accident, it is highly improbable that the encapsulated waste cylinder would be damaged to the extent that the vitrified waste would be shattered into such small particles that they would become airborne.

Thus, the function of the ventilation system is to create and maintain pleasant and hygienic working conditions in the final repository. It has no direct connection with the function of the waste canisters. The principles for the design of the system are illustrated in Fig. 6-6.

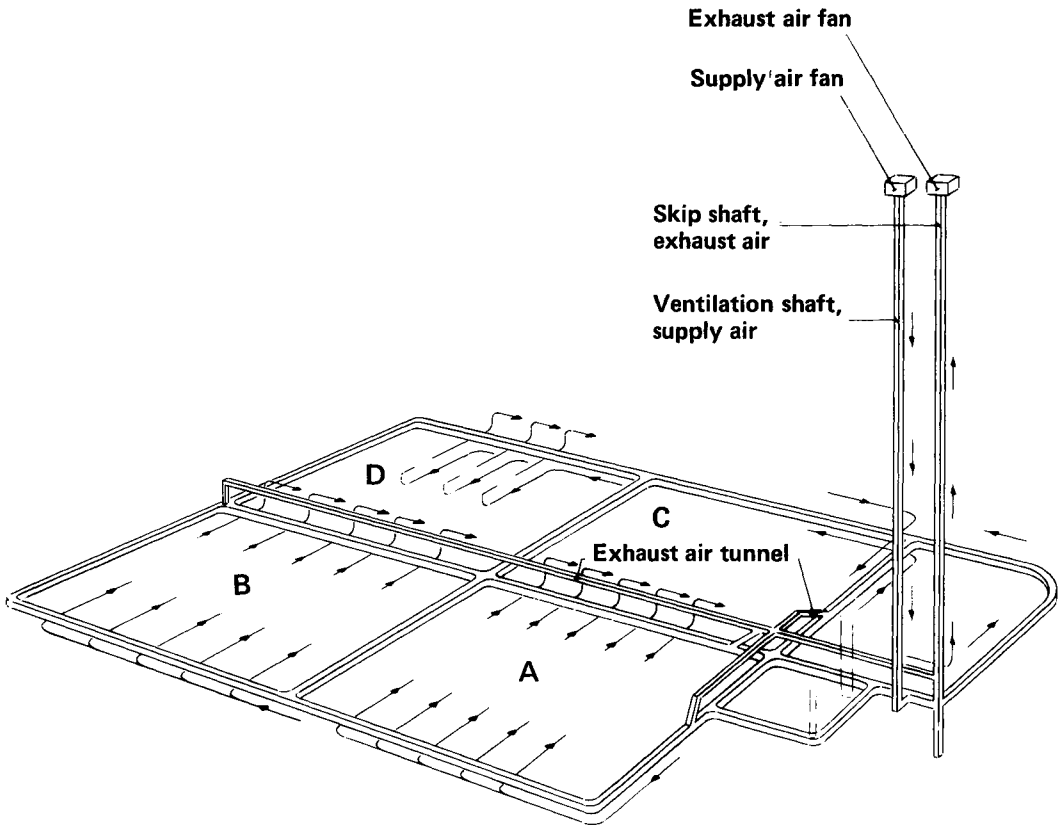


Figure 6-6. Perspective drawing of ventilation system in final repository. Deposition has been concluded within area A and is in progress in area B. Blasting has been concluded within area C and is in progress within area D.

Fans at ground level blow air down through a ventilation shaft and this air is then distributed to the tunnel system and the hoist shafts. After each storage tunnel has been blasted to its full length, it is ventilated by means of free air flow from the outer transport tunnels to the centre tunnel. From here, the air is evacuated via vertical shafts to an exhaust air tunnel situated above the centre tunnel, which also serves as an evacuation tunnel for smoke in the event of fire in any part of the tunnel system. The air is conveyed through the exhaust air tunnel to the rock hoist shaft, which also serves as an air evacuation shaft. Evacuation ducts from the service areas also empty into this shaft. Fans on the exhaust air side (at the opening of the shaft on the surface and in the tunnels which lead to the outlet shaft) also assist in the evacuation of the exhaust air.

In each storage tunnel and between the service areas and the transport passages are doors with dampers which can be used to regulate the distribution of air to the various areas according to immediate requirements, which are dependent upon the nature of the work and temperature conditions in the various parts of the facility.

Vital criteria for the design and capacity of the ventilation system are the fresh air requirement during the construction period and the necessity of keeping the temperature below approximately 25°C in tunnels where work is in progress.

Although it is assumed that both vehicles and machines used in

the facility will be electrically powered, the ventilation system has been designed so that its capacity can be made sufficient for diesel operation as well. The need for temperature reduction within parts of the tunnel system is met by the regulability of the ventilation dampers.

Even if all fans should fail, natural air convection will provide sufficient air exchange to permit work in the final repository to proceed for a limited period of time, possibly with some restrictions.

A drainage system will be provided in the final repository for the collection and removal of groundwater which leaks into the repository and spillage water from rinsing operations etc. Contaminated water from workshops, personnel areas etc. will be disposed of by a separate sewerage system.

The bottoms of the tunnels are designed so that water leaking into the tunnels is collected in a gutter and conveyed to pump sumps situated in the transport tunnels. The sumps have special chambers for sludge separation.

Sumps will also be provided in the service areas and in the shafts. Water from all sumps is pumped through pipes to the central shaft, and from there to the nearest suitable recipient on the surface. Pumping through the shaft is effected in two stages via a booster pump.

The number and size of the pump sumps will be finally determined on the basis of the leakage rate which is observed during the initial stages of construction. The sumps and the parts of the shafts below the bottom level of the repository shall have sufficient capacity to prevent flooding of the tunnels in the event of extreme flows and pump breakdowns. The pumps are operated automatically. Alarms are issued in the event of abnormal water levels.

Vital systems (ventilation, drainage, hoists, emergency systems) shall be packed up by auxiliary diesel generators.

6.3 PROPERTIES OF SAND/BENTONITE FILL

The material used to backfill deposition holes and seal tunnels and shafts (see 6.6) should possess the following properties:

- bearing capacity; to keep the canisters in place in the deposition holes and to hold back pieces of rock which may break off from the rock surface.
- plasticity; to maintain the homogeneity of the material despite minor movements in the bedrock.
- low permeability; to minimize groundwater flow in deposition holes and in backfilled tunnels and shafts.
- good thermal conductivity; to transmit the heat generated by the waste canister to the rock without the canister becoming excessively hot.
- high ion exchange capacity; to retard the migration of radioactive nuclides which may leak out from the canister.
- long-term stability against weathering, cementation or other changes; so that the material will retain the above properties throughout the service life of the repository.

Nor shall the material have a negative effect on the corrosion resistance of the canister.

Tests and studies /6-5 to 6-13/ have shown that a mixture of quartz sand and bentonite possesses most of the above-specified properties. Both materials are available in the required quantities. They can be mixed to a homogeneous material without difficulty, e.g. in an ordinary concrete mixer.

In order for the mixture to possess good bearing capacity and thermal conductivity as well as low permeability, it is desirable that the sand and bentonite fractions be mixed in such proportions as to provide a good (morain-like) particle size distribution.

Bentonite is characterized by a high swelling capacity when it absorbs water. It also has a high ion exchange capacity.

A high bentonite content increases the plasticity and ion exchange capacity of the mixture. It also improves the material's density, since its swelling will fill the pores in the material. At the same time, however, a higher bentonite content reduces the bearing capacity and thermal conductivity of the mixture and it becomes more difficult to handle.

Tests have shown that mixtures of 80-90% quartz sand and quartz filler and 10-20% bentonite provide a good balance of the desired properties.

The bentonite which is used is sodium bentonite (Volclay Mx 80 or the equivalent) in granulated form with a particle size distribution of 0.07 - 0.8 mm. It has good swelling properties, even after heating to 300°C.

The sand is pure quartz sand (98% SiO_2) with a particle size distribution of 0.063-2 mm. It has a sintering point (1 400°C) which lies well above the temperature encountered in the repository.

The results of field and laboratory tests on the mixture can be summarized as follows:

- The strength and deformation properties of the mixture are approximately the same as those of a clayed moraine. The bearing capacity of the material is composed of a cohesion component and a friction component. An increase of the bentonite content increases cohesion and reduces friction.
- Permeability varies between 10^{-8} and 10^{-11} m/s when swelling is restrained. With unrestrained swelling, permeability is higher.
- Thermal conductivity is between 0.3 and 1.7 W/m°C when the water content varies between 5 and 25%.
- Maximum dry solids density is 1.90-2.00 t/m^3 at an optimum water content of 8-12%.
- Unrestrained swelling when the material is in contact with water leads to an increase of its original volume by 5-20%. The swelling pressure of the material under restrained swelling conditions is on the order of 30-150 kPa.

Studies conducted for KBS have shown that the properties which

are of importance for bearing capacity, density, thermal conductivity and ion exchange will not undergo any essential change over long periods of time /6-8 and 6-12/.

The sand/bentonite filler material can affect the groundwater chemically by acting as a pH buffer, whereby a stabilization of the pH to a value between 8 and 9 can be expected at the temperatures in question, according to the results of studies currently in progress (cf. section 4.3.3).

6.4 OPERATION OF FACILITY

Canister deposition begins when approximately one-quarter of the total number of storage tunnels have been completed. The facility is designed in such a manner that the construction work can continue without any interference from the transport and deposition of canisters. Next to the centre tunnel, the storage tunnels are closed off by a concrete wall with a door and with dampers for regulating the ventilation flow in the storage tunnel.

The equipment for transporting and handling canisters and backfilling the deposition holes is railbound. It is pulled by an electric tractor. After deposition is concluded in a tunnel, the rails are moved to the next tunnel.

Only some 30-40 persons will be required for the operation of the facility (not including the construction work).

Up until the time the final repository is to be sealed, the storage tunnels in which canisters have been deposited can be inspected and checked and measurements can be made of rock stresses, temperatures, groundwater leakage etc.

The facility will be inspected by authorities such as the Swedish Nuclear Power Inspectorate and the National Institute of Radiation Protection in the same way as a nuclear power station. It will be designed in accordance with the regulations issued by these authorities and in consultation with concerned personnel organizations.

With regard to working environment and safety, see III:7.

6.5 QUALITY CONTROL

Besides the quality control of the rock and the groundwater which is carried out during the construction and operation period, quality control will be primarily aimed at verifying the properties of the sand/bentonite fill. This will be accomplished by sampling and analysis of delivered material, of the finished mixture at the mixing station and of the completed fill. The testing procedure is similar to that used for the core of an earth dam. The equipment for compacting the filler material in the deposition holes also has instruments which indicate and register the degree of compaction which is achieved.

For other quality control, see under 5.5.

6.6 PERMANENT CLOSURE

After the final repository has been filled with canisters to its design capacity, the facility can be kept open and inspected as long as surveillance and maintenance of the drainage and ventilations systems and other essential auxiliary systems are considered desirable. The facility can then be sealed and finally abandoned.

When it is sealed, the tunnels, shafts and boreholes are filled with a mixture of sand and bentonite similar to that used to fill the deposition holes.

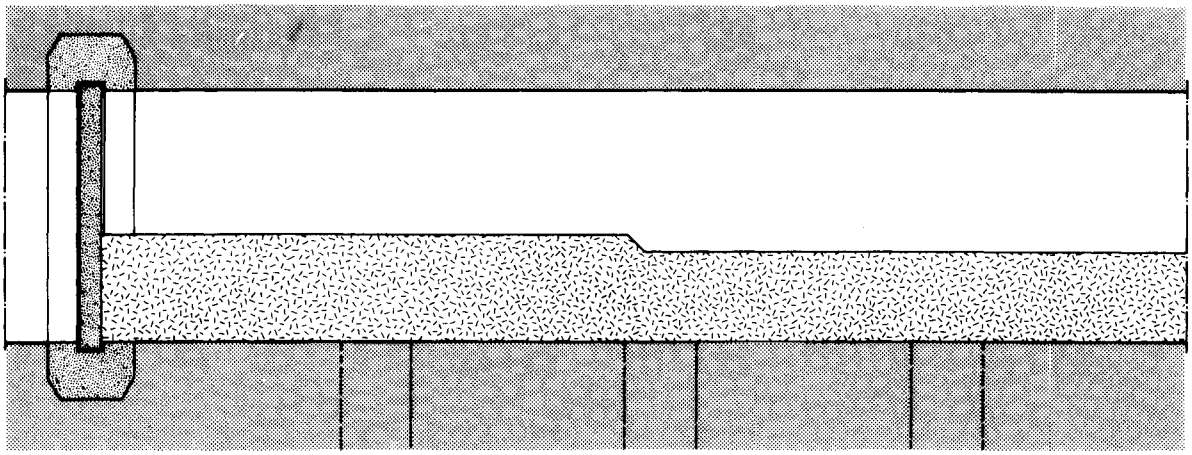
The fill is deposited in the tunnels in layers and compacted by means of vibratory rollers. The material is brought to the filling site on a conveyor belt and is spread by tractors. Before the work is begun, the tunnel floor is cleaned. If desired, the lids on the deposition holes and their yokes can also be removed.

The fill is applied in the top part of the tunnels by means of a spraying technique similar to the one which has long been used for lining the roofs of tunnels with concrete. Tests /6-4/ have shown that this technique is suitable for spraying sand/bentonite. The spraying technique and the swelling capacity of the bentonite permit complete filling of the tunnel section with a high (70-80%) degree of compaction. See Fig. 6-7 and 6-8.

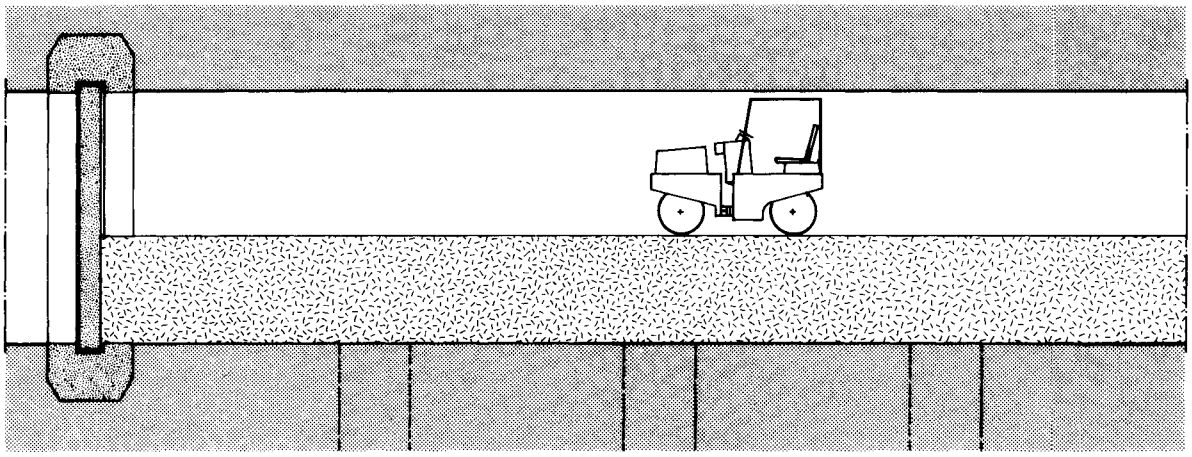
A mixture of sand and bentonite will also be used for backfilling vertical shafts. A fine-grained moraine may also be used in the upper part of the shafts. Holes drilled in connection with the preliminary study of the rock formation are filled with pure bentonite.

In this manner, all cavities and voids in the rock are filled with material which is at least as impervious as the surrounding rock. The ion-exchanging properties of the bentonite will thereby constitute an additional barrier to a migration of radioactive nuclides in the filler material.

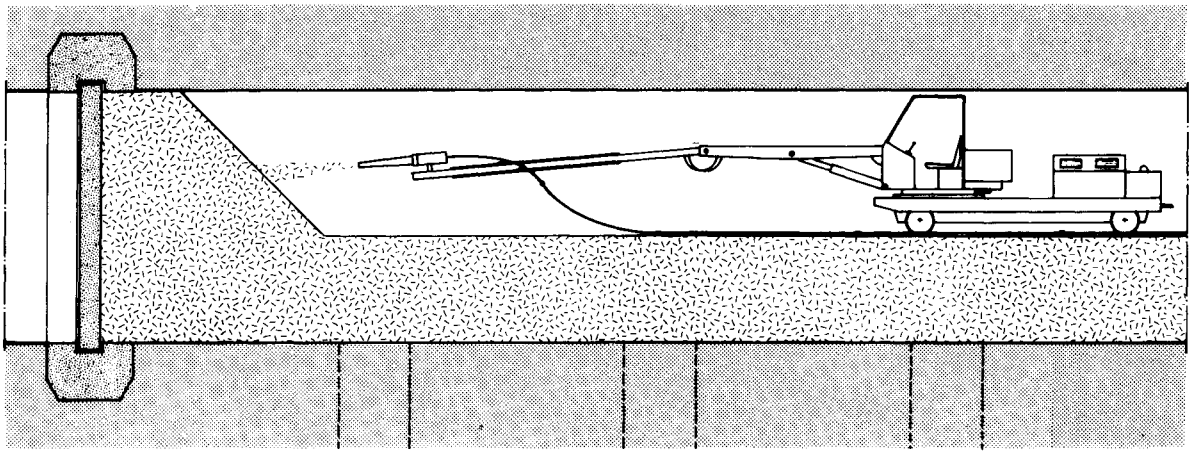
It is assumed that observations and measurements of the groundwater system, rock stresses, temperatures etc. will be performed for a certain period of time following the closure of the final repository. A programme for such activities will be drawn up in cooperation with the concerned authorities.



Deposition



Compaction



Spraying

Figure 6-7. When the final repository is sealed, the tunnels are filled with a mixture of quartz sand and bentonite. The lower part is placed by tractors and vibrorolled. The upper part of the tunnel is filled by spraying.

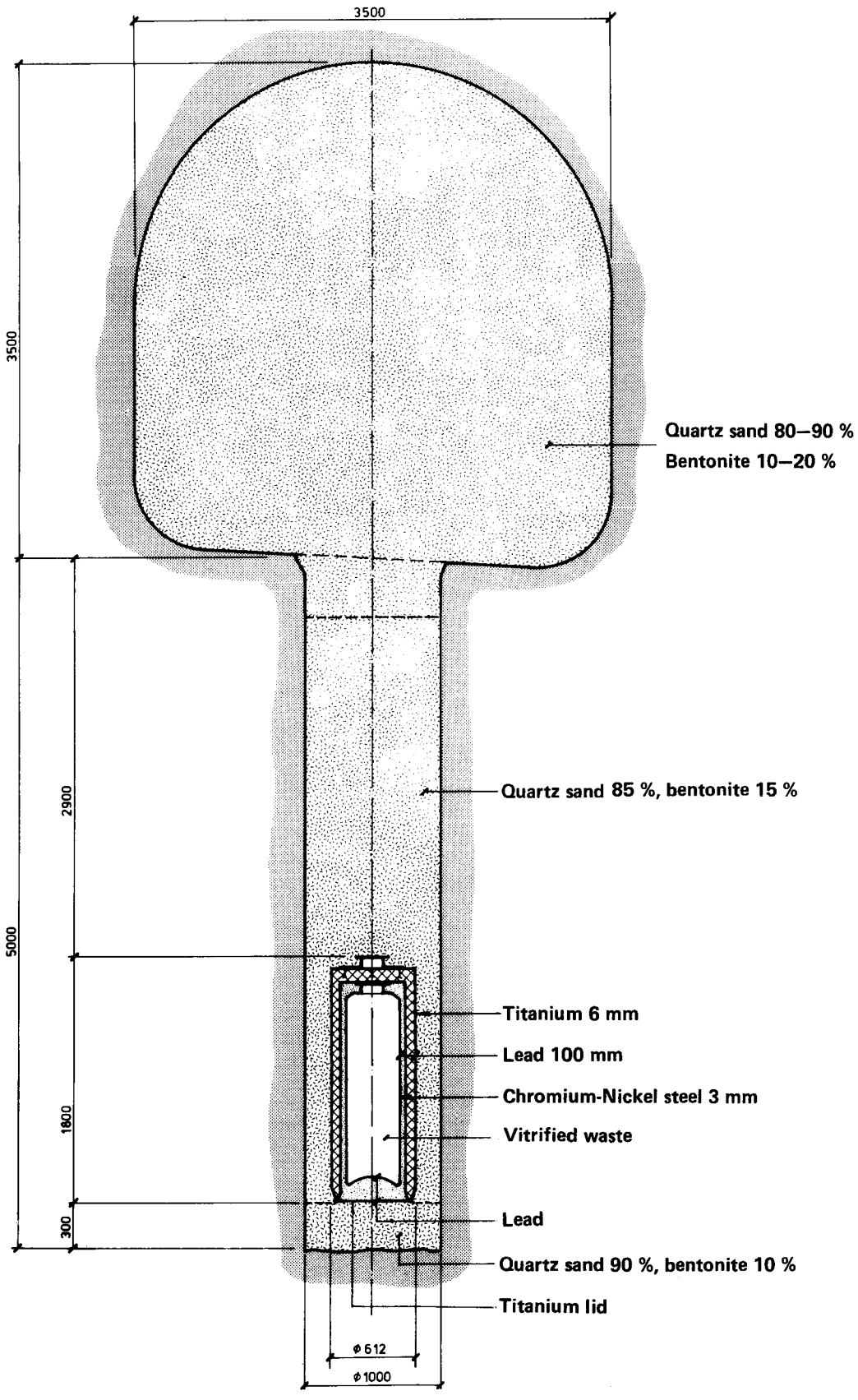
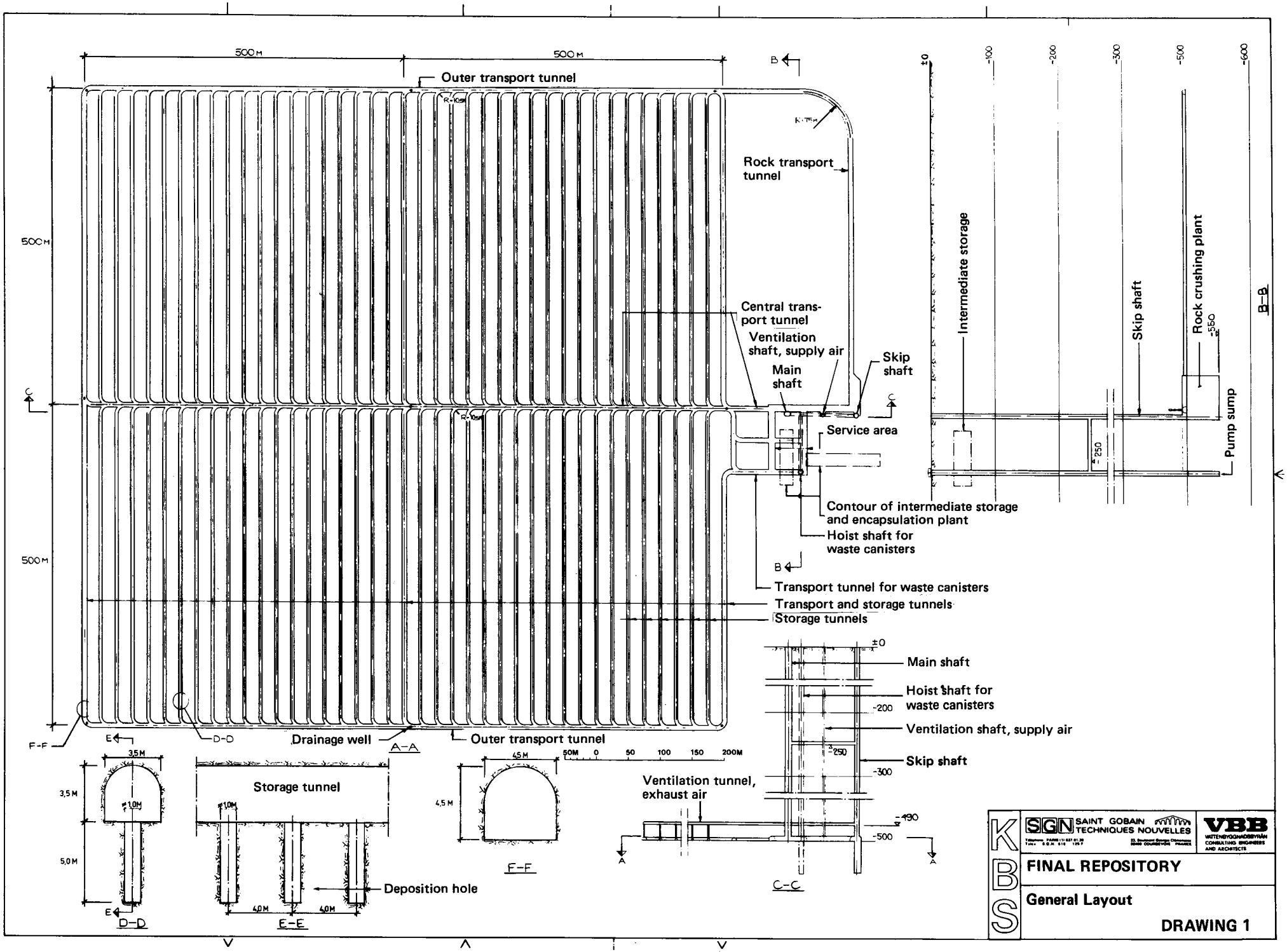


Figure 6-8. Sealed final repository.

6.7 DRAWINGS

DRAWING	1	GENERAL LAYOUT
"	2	PERSPECTIVE DRAWING
"	3	SITING EXAMPLE
"	4	CONSTRUCTION STAGES
"	5	TRANSPORT ROUTES
"	6	VENTILATION
"	7	STORAGE TUNNELS
"	8	SEALED REPOSITORY
"	9	TRANSPORT OF WASTE CANISTER FROM INTERMEDIATE STORAGE TO FINAL REPOSITORY
"	10	TRANSPORT AND DEPOSITION OF WASTE CANISTER IN FINAL REPOSITORY
"	11	SEALING OF DEPOSITION HOLES
"	12	SEALING OF TUNNELS


500 M
500 M
Outer transport tunnel
Rock transport tunnel
500 M
500 M
Central transport tunnel
Ventilation shaft, supply air
Main shaft
Skip shaft
Service area
Contour of intermediate storage and encapsulation plant
Hoist shaft for waste canisters
Transport tunnel for waste canisters
Transport and storage tunnels
Storage tunnels
Drainage well
Outer transport tunnel
A-A
B-B
C-C
D-D
E-E
F-F
Intermediate storage
Skip shaft
Rock crushing plant
Pump sump
±0
-100
-200
-300
-500
-600
-250
-550
-490
Main shaft
Hoist shaft for waste canisters
Ventilation shaft, supply air
Skip shaft
Ventilation tunnel, exhaust air
Storage tunnel
Deposition hole
3,5 M
5,0 M
4,5 M
1,0 M
4,0 M
50M 0 50 100 150 200M
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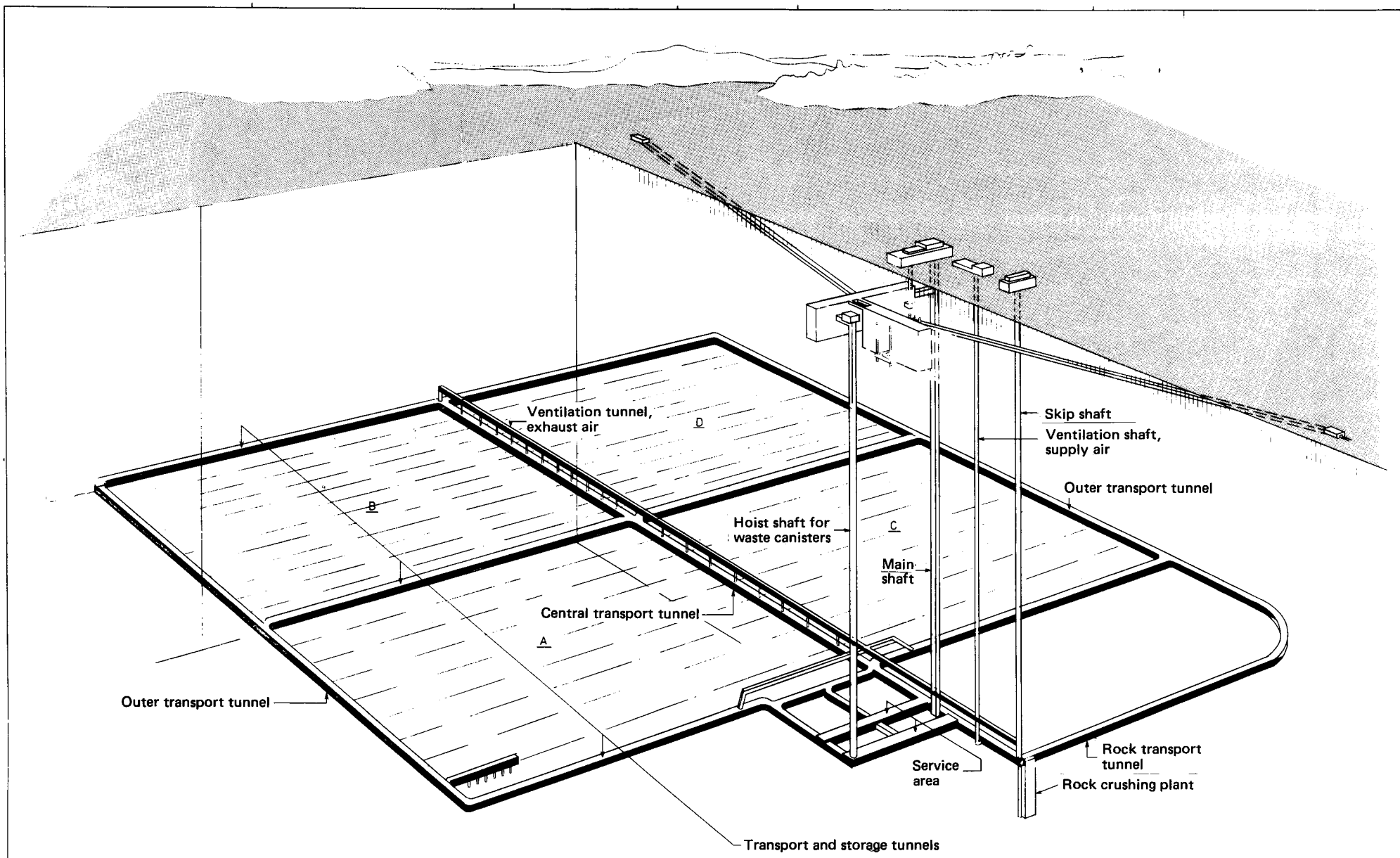
FINAL REPOSITORY
General Layout
DRAWING 1



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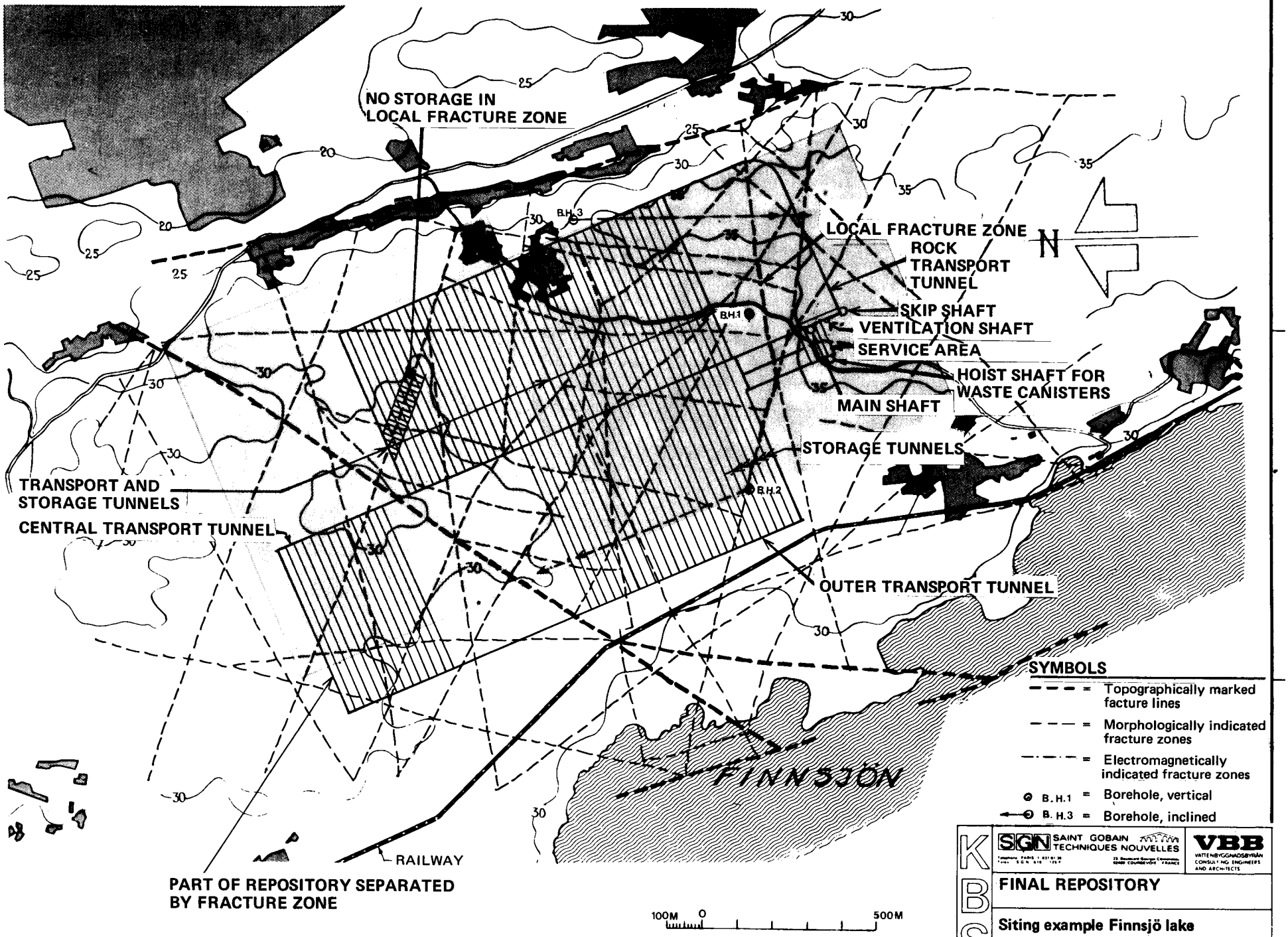
FINAL REPOSITORY

Perspective drawing

DRAWING 2


NO STORAGE IN LOCAL FRACTURE ZONE
LOCAL FRACTURE ZONE
ROCK TRANSPORT TUNNEL
SKIP SHAFT
VENTILATION SHAFT
SERVICE AREA
HOIST SHAFT FOR WASTE CANISTERS
MAIN SHAFT
STORAGE TUNNELS
TRANSPORT AND STORAGE TUNNELS
CENTRAL TRANSPORT TUNNEL
OUTER TRANSPORT TUNNEL
B.H.1
B.H.2
B.H.3
FINNSJÖN
RAILWAY
PART OF REPOSITORY SEPARATED BY FRACTURE ZONE
N
100M 0 500M
SYMBOLS
Topographically marked facture lines
Morphologically indicated fracture zones
Electromagnetically indicated fracture zones
B.H.1 = Borehole, vertical
B.H.3 = Borehole, inclined
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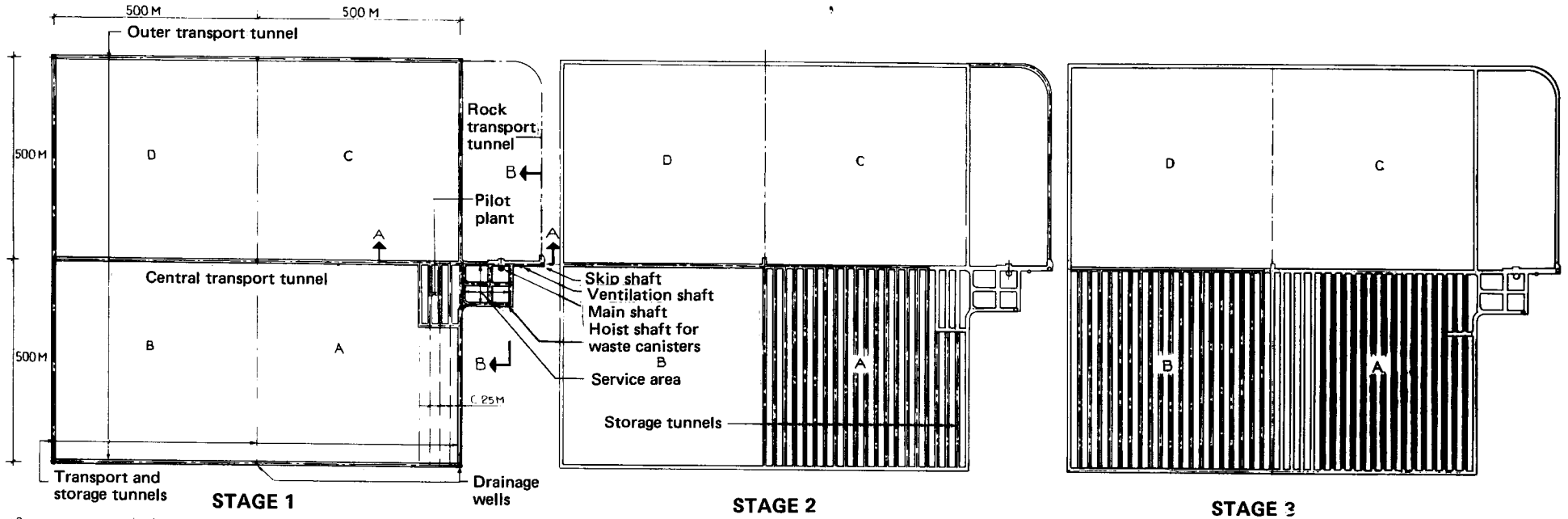
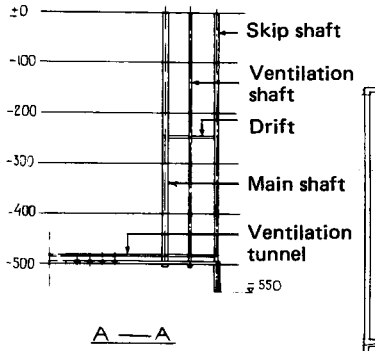
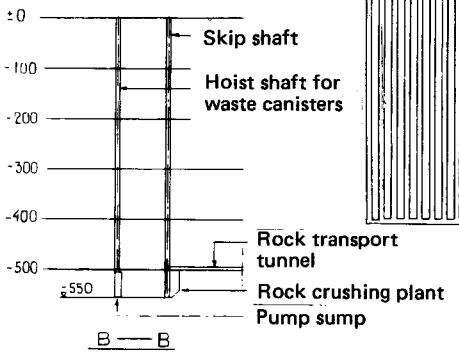
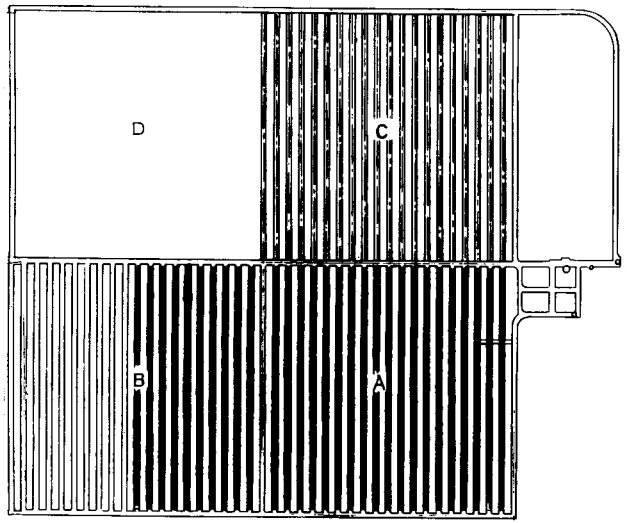
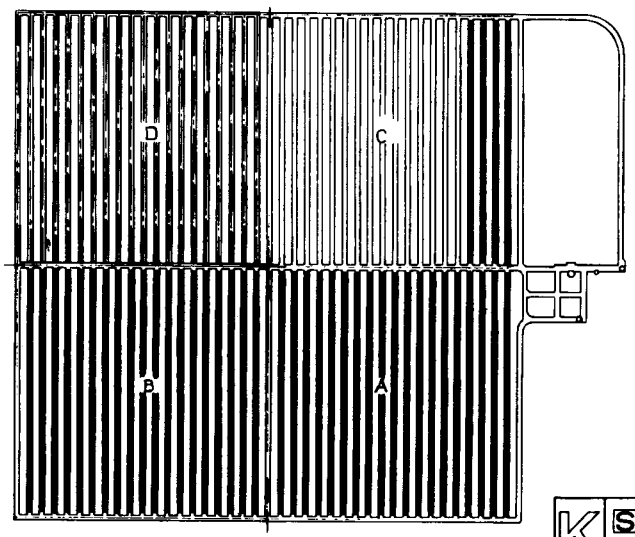
FINAL REPOSITORY
Siting example Finnsjö lake
DRAWING 3











STAGE	YEARS	BLASTING	DEPOSITION
1	1–4	Shafts, pilot plant, transport tunnels	Experimental plant
2	4–7	Area A	
3	7–13	Area B	Area A 70 %
4	13–14,5	Area C 35 %	Area A finished
5	14,5–18	Area C finished	Area B 45 %
6	18–22	Area D 75 %	Area B finished
7	22–25	Area D finished	Area C 20 %
8	25–29,5		Area C finished
9	29,5–37		Area D finished

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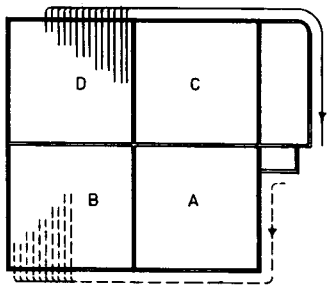
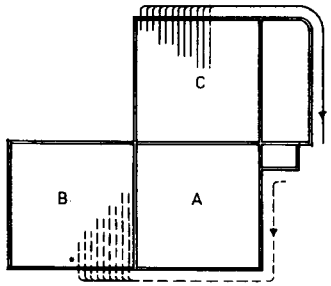
FINAL REPOSITORY

Construction stages

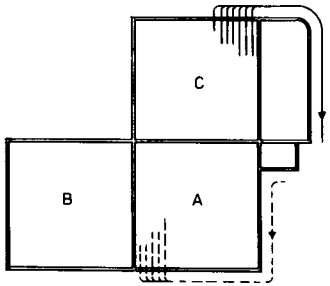
DRAWING 4



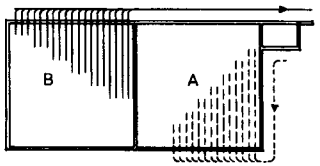
Stage 6



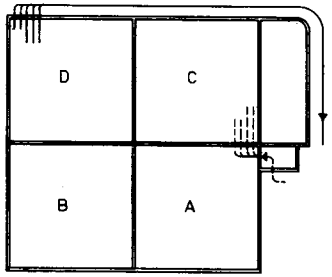
Stage 5



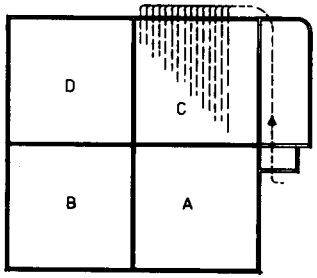
Stage 4



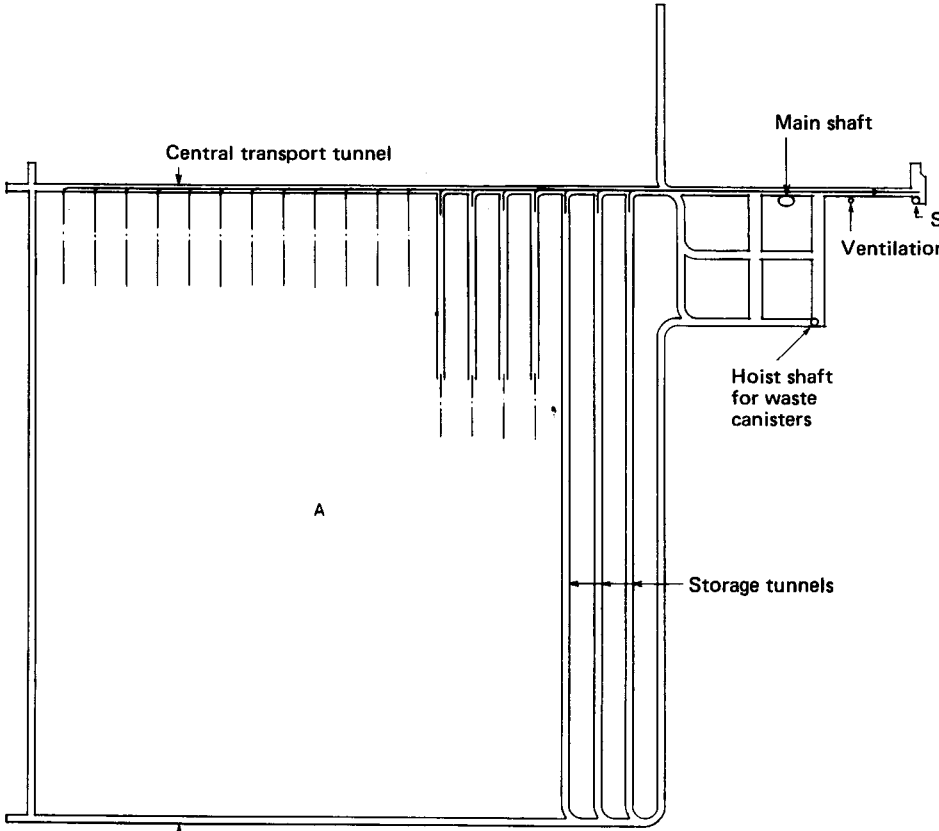
Stage 3



Stage 7



Stage 8



Stage 2

Direction of rock transport

Direction of waste transport

Note: Construction stages as per drawing 1

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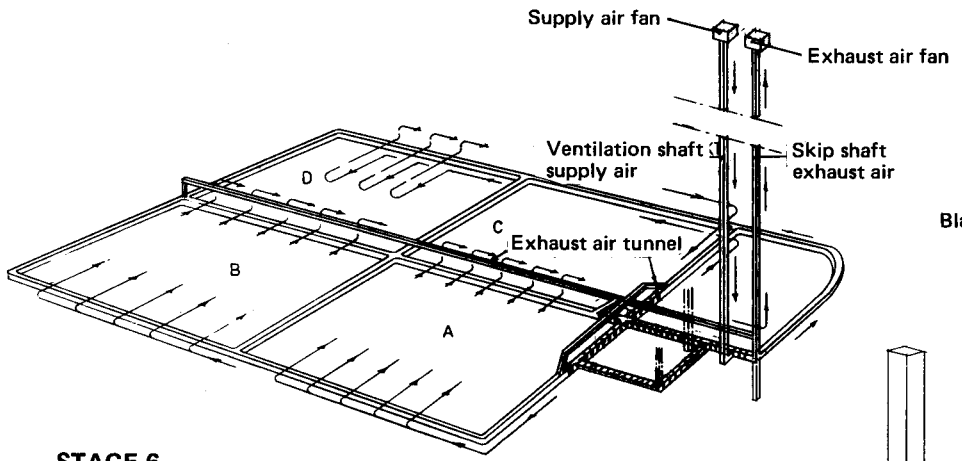
FINAL REPOSITORY

Transport routes

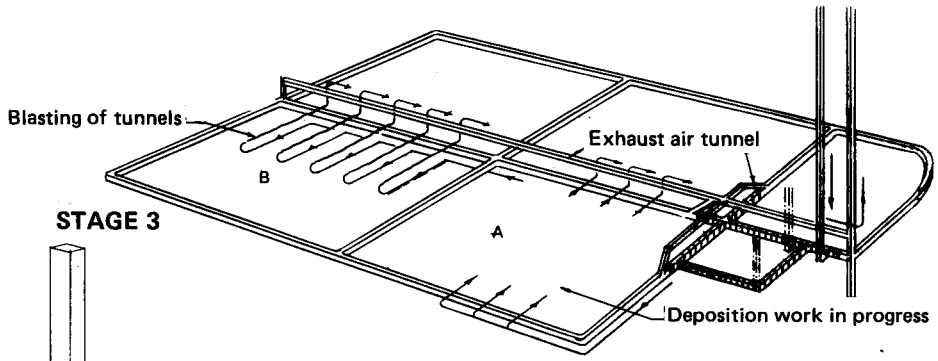
DRAWING 5



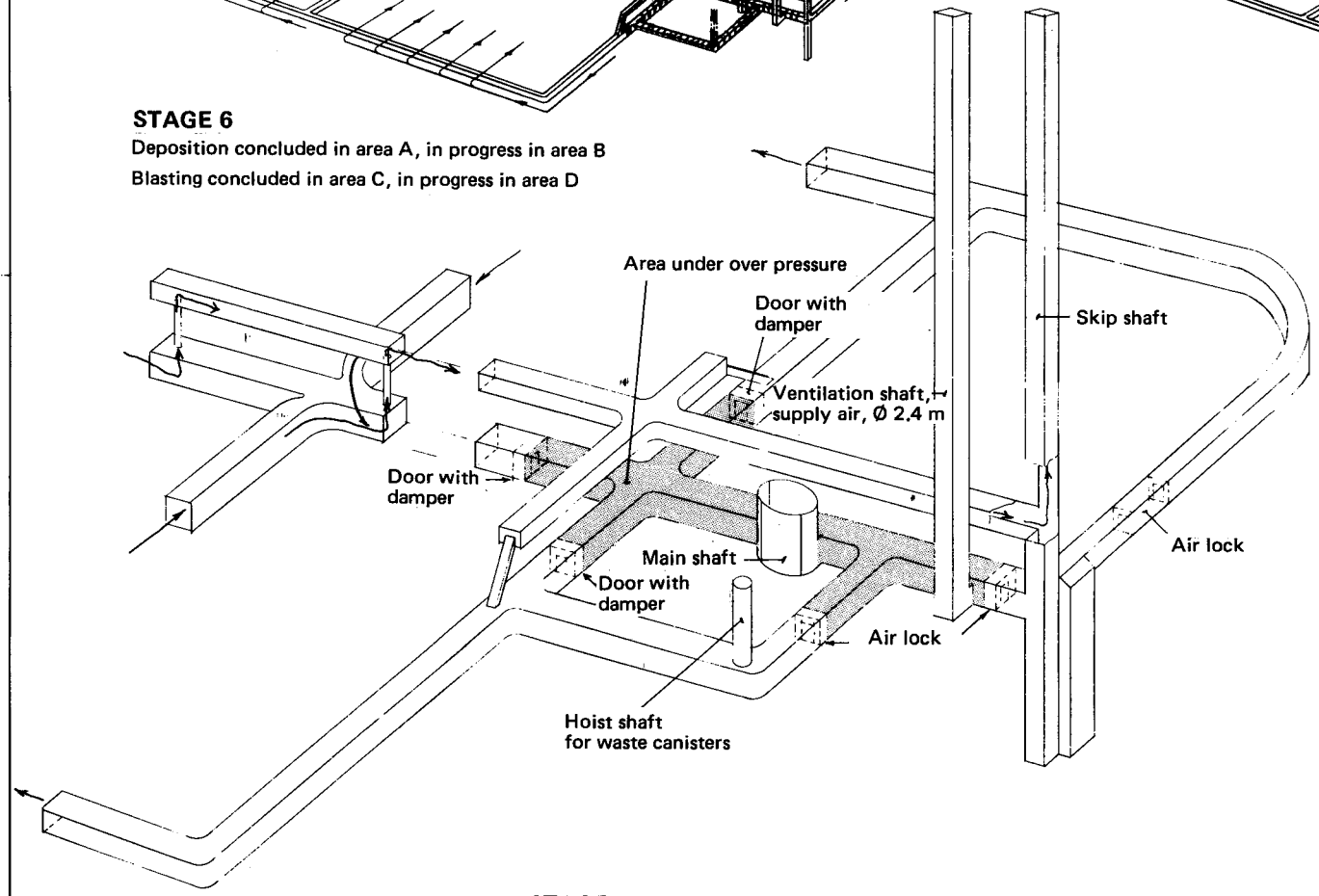
STAGE 6

Deposition concluded in area A, in progress in area B

Blasting concluded in area C, in progress in area D



STAGE 3



STAGE 8–9

Note: Construction stages as per drawing 1

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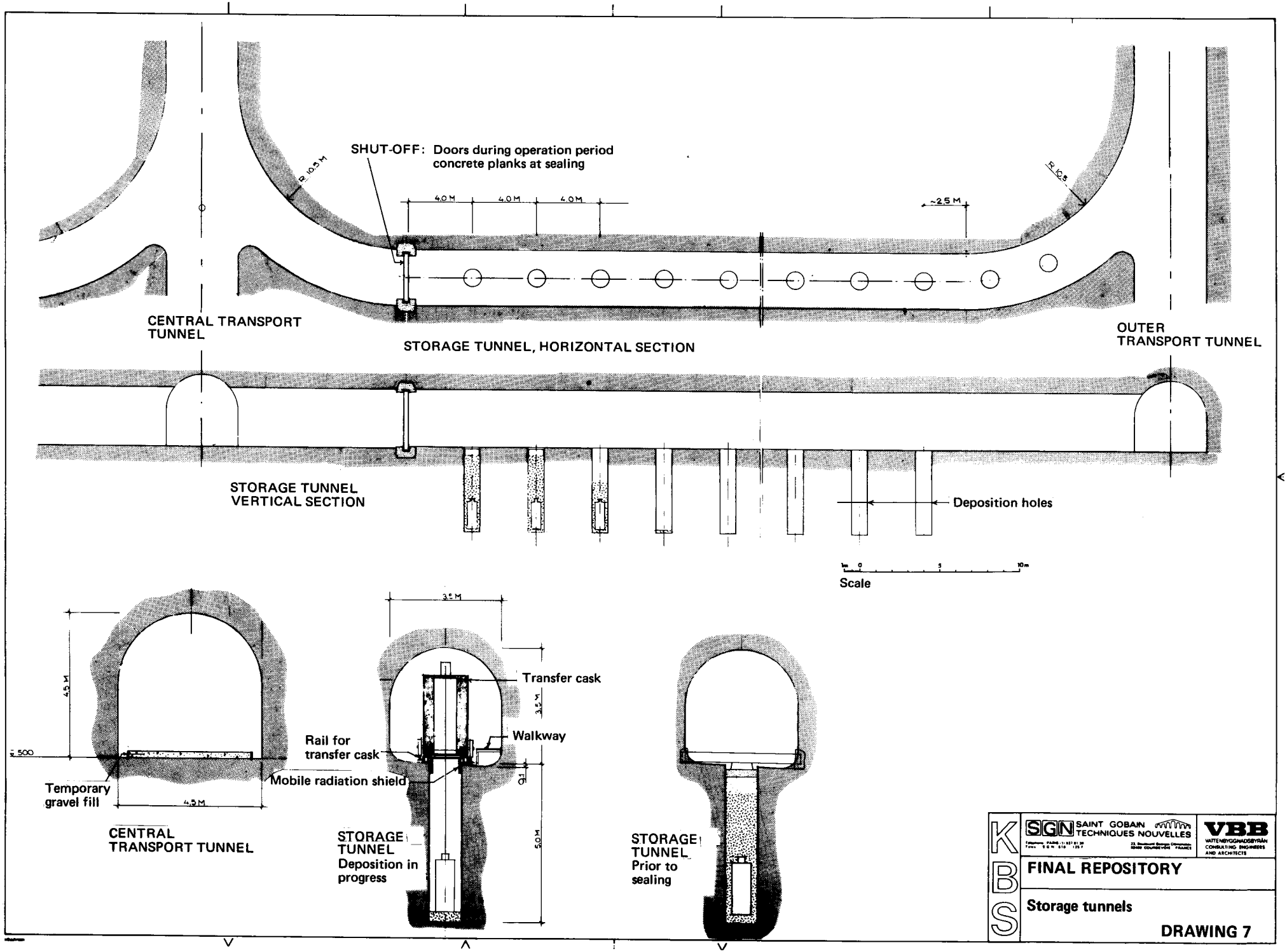
FINAL REPOSITORY

Ventilation

DRAWING 6



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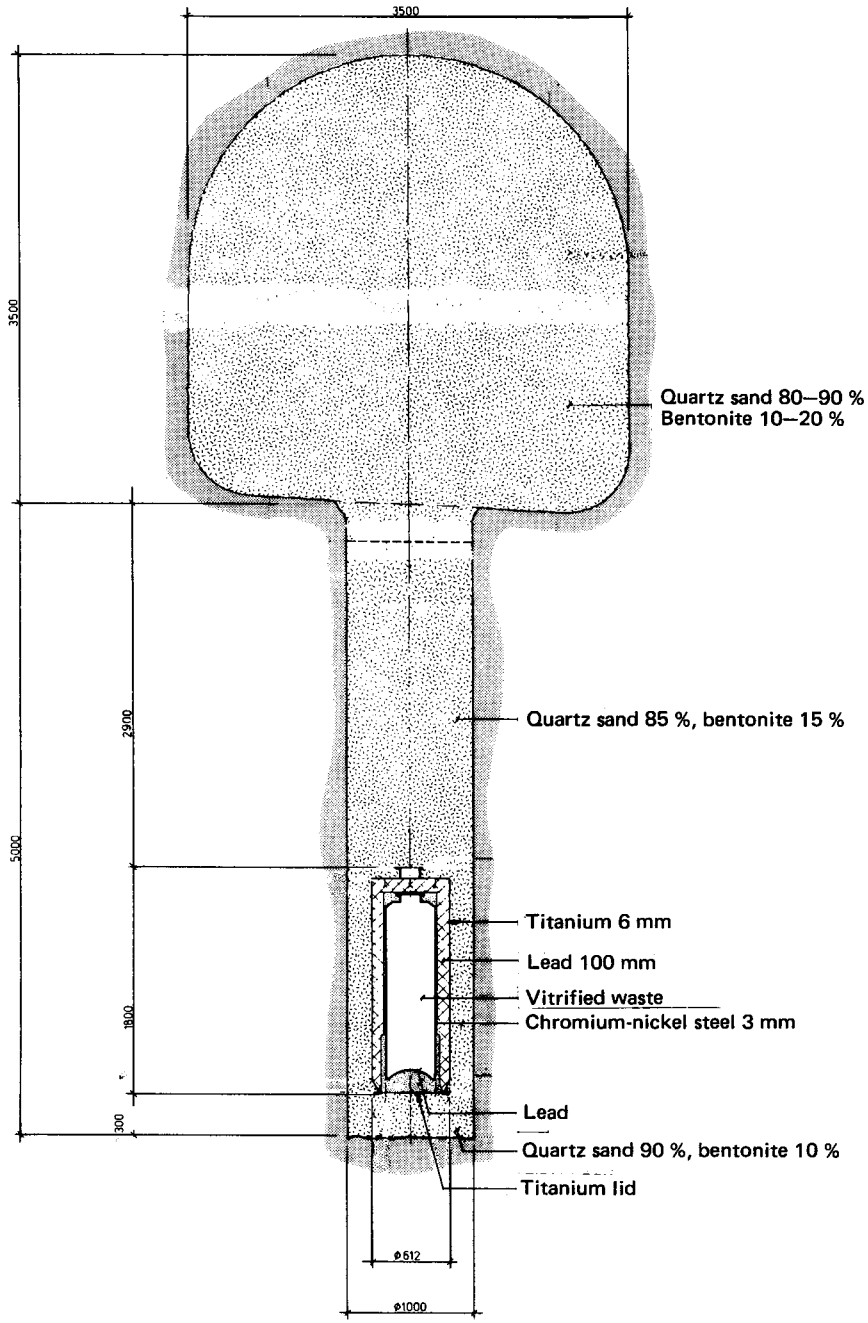
FINAL REPOSITORY

Storage tunnels

DRAWING 7



50 cm 0 2 m

Scale

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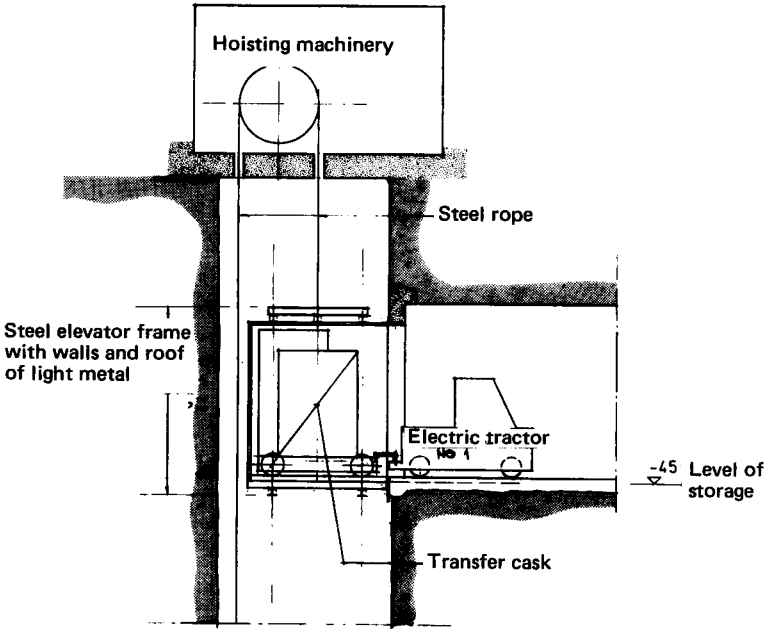
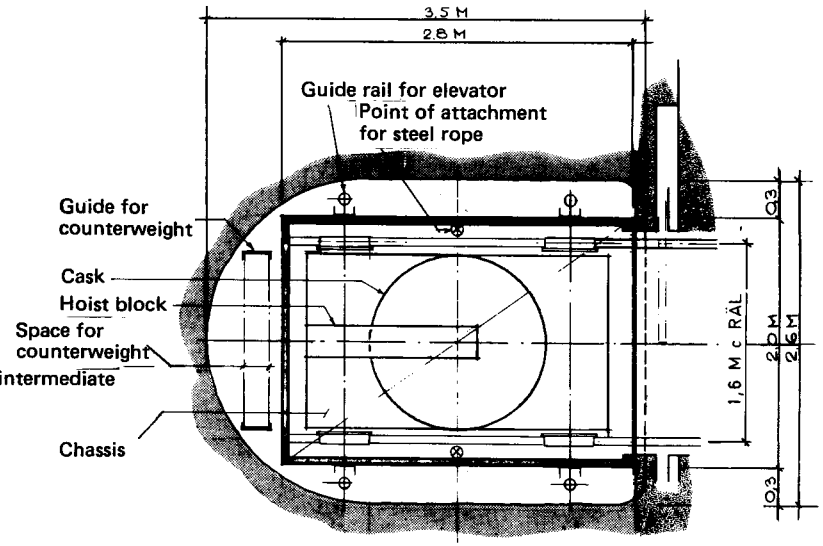
FINAL REPOSITORY

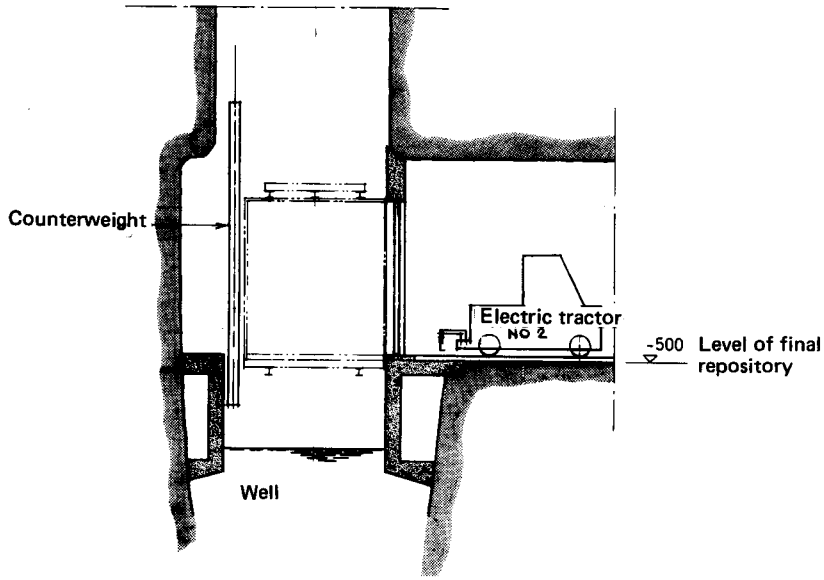
Sealed repository

DRAWING 8





MAIN SECTION THROUGH HOIST SHAFT



VERTICAL SECTION THROUGH HOIST SHAFT

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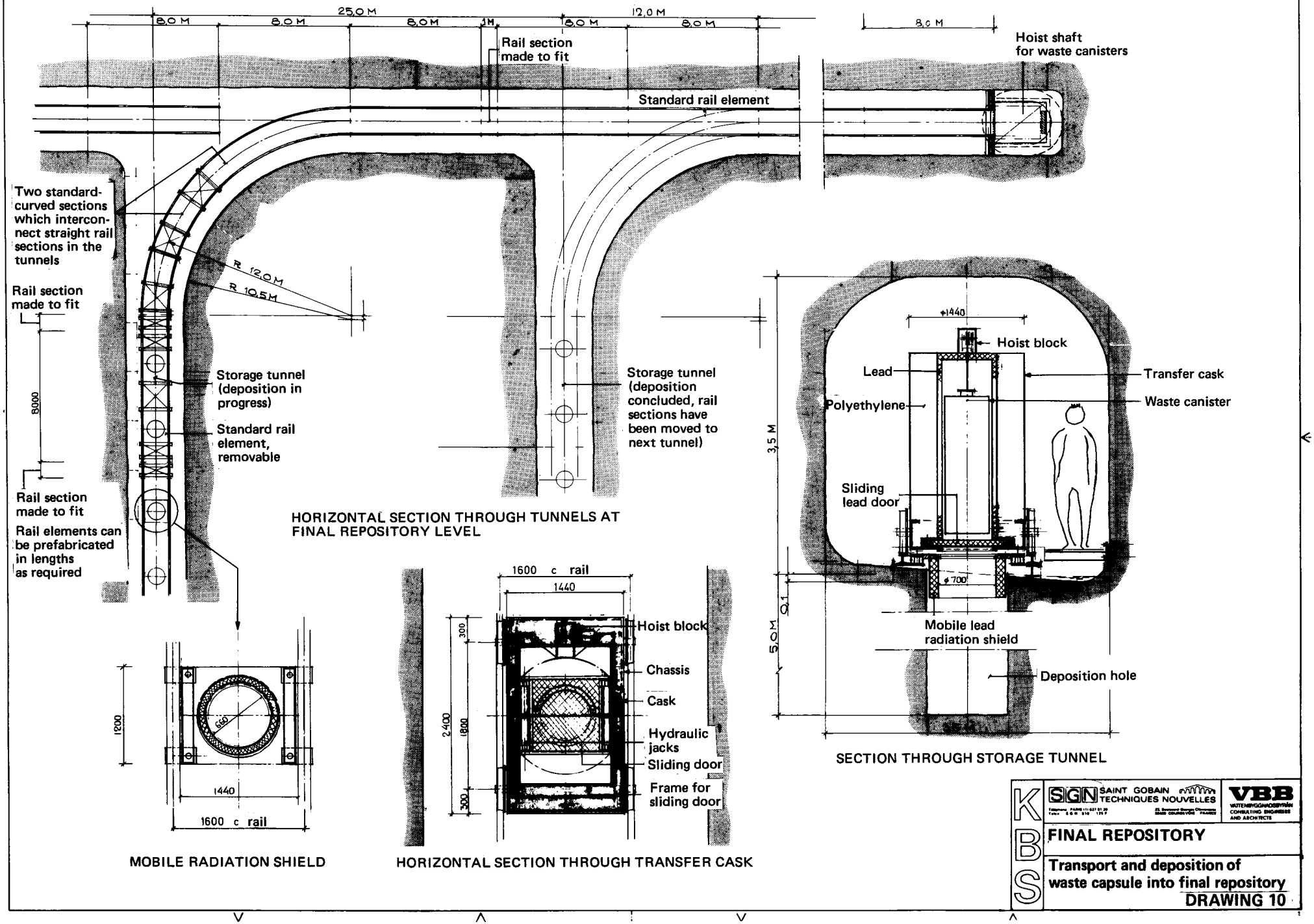
FINAL REPOSITORY

Transport of waste canister from intermediate storage to final repository **DRAWING 9**



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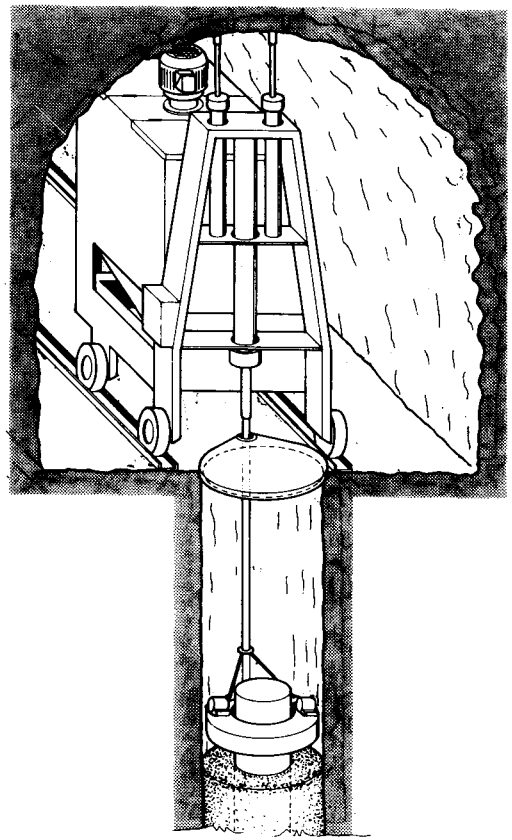
FINAL REPOSITORY

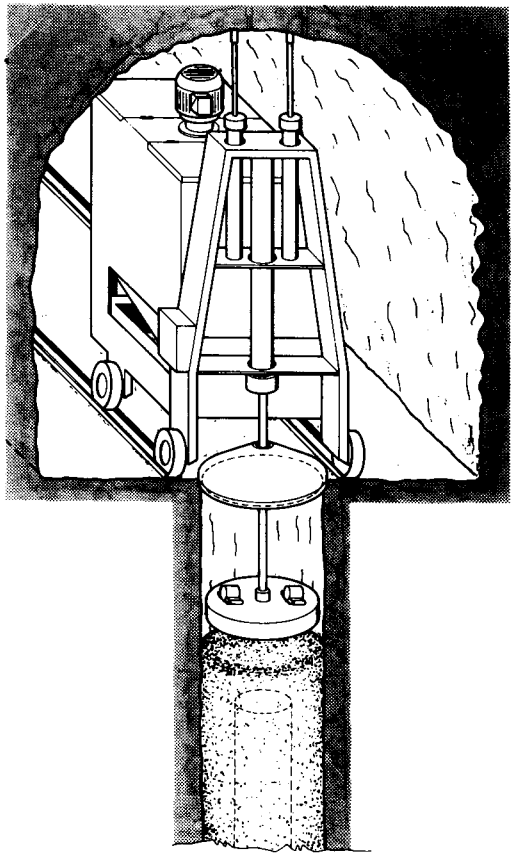
Transport and deposition of waste capsule into final repository

DRAWING 10



Compaction of side bed

Backfill with buffer material



Compaction of top bed

KBS

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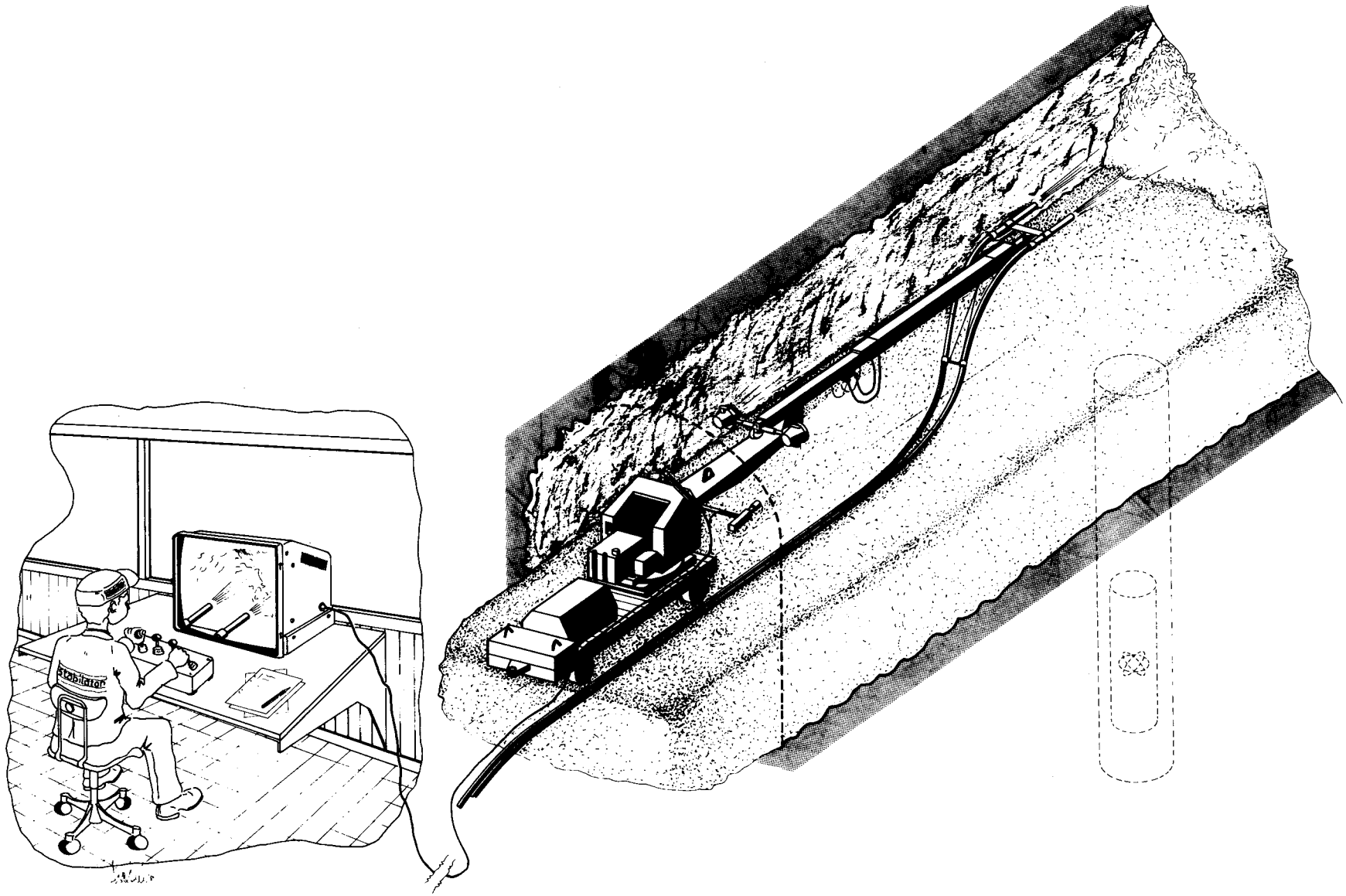
VBB VIAK-BERATUNGSBÜRO FÜR BAUWESEN AND ARCHITECTS

MARKBYGGNINGEN

FINAL REPOSITORY

Sealing of deposition holes

DRAWING 11



Spraying of buffer material

KBS

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VBB VATTENBYGGNADSBYRÅN CONSULTING ENGINEERS AND ARCHITECTS

MASKINHALLNINGEN

FINAL REPOSITORY

Sealing of tunnels

DRAWING 12

7 SAFETY

7.1 WORKING ENVIRONMENT

7.1.1 Authorities and regulations

The scope of the facilities will be outlined in the application for siting permission which is required by the provisions of Section 136 a of the Building Act. In the consideration of such applications, concerned authorities and interest organizations shall be given an opportunity to express their views. Working environment conditions can thereby be given early attention and, if necessary, be regulated in connection with the granting of permission.

The design of buildings and equipment and conditions on the worksite during the construction and operation phases shall comply with the requirements of applicable laws and regulations. In order to guarantee that full attention is given to occupational safety and health matters, both the Workers' Protection Act and various statutes and ordinances require that inspection authorities and employee organizations be given an opportunity to examine work methods and the plant of the worksite prior to the start of construction.

7.1.2 Working environment during the construction phase

Construction work on the intermediate storage facility, the encapsulation station and the final repository will include:

- Buildings on ground level for offices, personnel quarters, dining halls, workshops and storerooms, electrical installations, water works, sewage installations, ventilation systems and facilities for the reception, treatment and storage of buffer and backfill material.
- Buildings in rock galleries with approx. 30 m rock cover for reception, intermediate storage and final encapsulation of waste cylinders.
- Repository with service facilities approx. 500 m below the surface for the final storage of waste.

Certain installations on ground level as well as the upper rock cavern installation (the intermediate storage section) shall be ready for operation much earlier than the lower rock cavern installation (final repository). Blasting and construction work and

deposition will be pursued simultaneously in the final repository. There will therefore be no distinct delineation in time between the construction and the operation phases. Commissioned plant sections shall therefore be separated from ongoing construction activities.

Different types of working environment problems will be encountered in different work areas. But since existing technology will be used in the different types of construction work, the various environmental problems will be familiar. It should be possible to solve environmental problems in a satisfactory manner through cooperation between the plant owner, the inspection authorities and the concerned employee organizations.

7.1.3 Working environment during the operation phase

As during the construction phase, occupational hygiene problems will vary from one work area to another. Occupational hygiene requirements pertaining to the function of the plants are already known. The design of premises and installations shall be approved in the usual manner by inspection authorities and employee organizations.

Most of the installations for water and power supply will be located on the surface, along with certain workshops and equipment for the preparation of buffer material and backfill.

Less extensive experience is available with respect to the installations for reception, storage and treatment of buffer and backfill material containing quartz sand and bentonite. The quartz sand will contain a certain amount of fine-grained material, giving rise to some dust hazard.

In order to protect the personnel against quartz dust, the dust sources shall be enclosed and suitable ventilation shall be provided. Dust-generating processes shall be supervised from areas under pressurized ventilation. In connection with maintenance work and in the event of spillage, personnel shall be protected by the use of functionally designed work equipment and personal safety equipment.

Prepared buffer and backfill material shall be watered down prior to transport to the site of application. The dust hazard is thereby limited to preparation.

Waste cylinders are received, stored and encapsulated in the upper rock cavern facility. Encapsulation involves the casting of lead and the welding of titanium. Work with waste cylinders and canisters shall be remote-controlled from radiation-shielded areas.

Rock work will be done using largely the same techniques which are normally used in rock cavern excavation and tunnel driving. This means that working environment precautions can be based for the most part on existing technology. Work machines and transport equipment shall be electrically driven whereever possible.

7.2 RESCUE SERVICE

7.2.1 Authorities and regulations

According to the Fire Protection Act, rescue service activities are aimed at minimizing damage to human beings, property and the environment in the event of fires, oil spills, cave-ins, landslides, floods or other emergencies. According to the Act, each municipality in Sweden is responsible for providing its own rescue service and instituting preventive measures.

According to Section 14 of the Fire Protection Act, it is encumbent upon the owners of buildings, storage depots or other facilities to procure and maintain the necessary equipment for extinguishing and rescure work in connection with fires and to adopt all other measures which are necessary to prevent and combat fire, all within the bounds of reasonable cost.

The County Administration issues the fire regulations for the municipalities and may be regarded, along with the National Fire Service Board (which is responsible for issuing recommendations and instructions), as the inspection authority for the municipalities.

General rules for fire protection in connection with the erection of industrial plants are provided in the Swedish Building Code, SBN 1975, issued by the National Board of Physical Planning and Building. But these rules are not applicable to an underground facility for the final storage of nuclear waste.

The necessary examination is normally undertaken in connection with the processing of building permit applications.

7.2.2 Design of facility

It is often difficult to satisfy requirements on evacuation and fire extinguishment on large worksites during the construction phase. It may be necessary to organize a fire fighting organization on the workside in cooperation with the municipal fire brigade.

Special temporary alarm and extinguishing systems may have to be installed. But it should be possible to put the permanent systems into service as soon as possible.

Special precautions must be adopted in connection with the planning of installations which involve special fire hazards in the underground facility for the final repository. The principles for the design of such plants are summarized in "Underground fire protection", published by the Swedish Mine-Owners' Association in 1976.

Evacuation routes and fire ventilation devices will be planned and designed in consultation with fire authorities in the same manner as in mines.

7.2.3 Enforcement and routines

The person who is made responsible for the fire protection of a facility shall, in consultation with the municipal fire chief, ensure that the personnel are well-acquainted with the steps which are to be taken in the event of an alarm in different work areas. He shall also be responsible for enforcing fire protection requirements in connection with the various construction and operation phases.

7.3 RADIATION PROTECTION

7.3.1 Authorities and regulations

Matters pertaining to the handling of radioactive waste as well as occupational hygiene conditions in connection with work in a radioactive environment are dealt with by the National Institute of Radiation Protection with the support of the Radiation Protection Act.

In 1977, a proposal was submitted for certain amendments to the Atomic Energy Act. The proposal is aimed at making the Swedish Nuclear Power Inspectorate responsible for the inspection and supervision of the handling and storage of radioactive waste products. Regulations pertaining to permissible releases as well as radiation protection matters would, however, continue to be handled by the National Institute of Radiation Protection.

Design plans will be submitted to the National Institute of Radiation Protection for critical examination prior to the start of construction.

The law requires that a radiology officer approved by the National Institute of Radiation Protection be present at the commissioning of plants. The radiology officer is responsible for ensuring that the rules and regulations issued by the institute are complied with.

7.3.2 Enforcement and routines

The facilities will be divided up so that premises for the handling and storage of radioactive substances are separated in a safe manner from other activities. Such separation shall be provided in the final repository by intervening space.

Activities in the final repository mainly comprise the handling of encapsulated radiation sources with known activity contents. Required radiation shielding can therefore be calculated with good accuracy.

All work operations with waste cylinders shall be remotely controlled with a radiation shield between the operator and the cylinder. For transports outside of specially shielded compartments, the cylinders shall be enclosed in radiation-shielded transfer casks.

The rules for reporting of personal doses etc. are established by

the radiation protection authorities. The authority also issues rules governing how special operations entailing abnormal dose loads shall be reported for evaluation before the work is commenced.

7.4 PHYSICAL PROTECTION

7.4.1 Authorities and regulations

The Swedish Nuclear Power Inspectorate (SKI) is, according to the Atomic Energy Act and with the support of its provisions, the inspection authority for the physical protection of fissionable material and nuclear energy facilities. The authority notifies the owners of the facilities of regulations and directives and supervises and enforces compliance therewith. The expression "physical protection" encompasses a series of overlapping safeguards against attack, sabotage and other acts of violence.

For matters pertaining to physical protection, there is an advisory board (the board for the control of fissionable material) whose function is to supervise activities, provide advice concerning the application of existing agreements and make proposals for revisions of international agreements for the control of fissionable material.

With regard to police activities in connection with physical protection, SKI cooperates with the National Police Board, whose instructions direct them to cooperate with agencies whose activities pertain to police activities. To the extent specified in their instructions or in special regulations, the National Police Board is required to issue directives to lower police authorities and to direct police activities.

The county administrations are the highest police authorities within their respective counties and the county police chief - who is an officer in the country administration - is directly responsible for upholding order and security within the county. Owners of nuclear power facilities shall cooperate with the police district in question with regard to matters pertaining to physical protection.

Directives and regulations governing the physical protection of commissioned nuclear power installations and the transport of fissionable material within the country can be issued by SKI.

Special regulations for facilities for the handling and storage of spent nuclear fuel and high-level waste have not been issued. Such facilities are considerably less technically complicated than nuclear power plants, so the regulations governing physical protection at such facilities should be simpler. The information required for the formulation of detailed directives and regulations will not be available until the facilities have reached the detailed planning stage.

The KBS studies have assumed that physical protection shall be basically the same as at a nuclear power plant, i.e. divided into the following main components:

1 District or peripheral protection, which consists of a security fence provided with devices which detect and issue alarms in the event of unauthorized entry. It shall be possible to verify the cause of the alarm through closed-circuit television.

2 Shell protection, which comprises sufficiently robust building structures in combination with security control arrangements at points of passage into and out of the facility.

3 Special protection for safety-related equipment. This protection may consist of physically separate redundant systems, protective building structures or administrative rules for admission etc.

The structural design of the physical protection must conform to the requirements on evacuation and extinguishing in the event of fire. The detailed design of the installation will therefore be submitted to the fire authorities for approval.

7.4.2 Design of facility

The need for physical protection measures has been taken into account in the design of the facilities for the handling and storage of high-level material.

The rock chambers, with their few and easily supervised points of access, offer good opportunities for physical protection with a high level of security. Tunnel doors are designed to satisfy the requirement for protection against unauthorized entry, together with installed alarm devices. Ventilation openings, water intakes and vital surface installations will be protected against unauthorized entry and/or demolition.

Other points of access to the facilities will be designed so as to permit security control of personnel and inspection of arriving vehicles and cargoes.

For fire protection and other reasons, consequence-mitigating safeguards will be provided in the form of physical separation and redundancy of safety-related equipment and of vital systems. Such safeguards may include, for example, back-up battery supplies for vital functions and auxiliary power supply and generating equipment which considerably increase the inherent protection of the plant and thereby also protect against sabotage.

The facilities will be guarded by a permanent guard staff as well as by monitoring equipment. The guard staff can also be assigned functions within fire protection.

In the detailed design of protected areas, special attention will be devoted to the geographic situation of the facilities and the consequent feasibility of assistance from the police. Information on protective and alarm devices as well as a list of particularly vital parts of the facility will be submitted to the inspection authority as part of the safety report on the facilities.

Due to strict government regulations and control, the siting of sensitive facilities in rock and the nature of the technical

equipment, the probability of acts of sabotage is judged to be very low. Since the consequences of such acts would be limited, the facilities should not be attractive targets for potential saboteurs.

7.4.3 Transport and operation

High-level radioactive material will be transported in a containment which virtually eliminates the possibility of mechanical damage or attack. The size of the transport casks will be determined by other requirements, including mechanical strength.

All transports, both by land and by sea, will be guarded in accordance with detailed plans drawn up in advance. In the event of fire, accident or other disturbances which may jeopardize the safety of a transport, telecommunications with stand-by potential will be provided to agencies in the community which could be affected.

Before commencement of the planned operation of the facilities, special safety plans shall be drawn up. These plans shall specify both administrative and technical measures for physical protection and shall be submitted to the licensing authority for examination and approval. Plans for personnel recruitment and competency requirements for the operating personnel shall also be specified.

7.5 WARTIME PROTECTION

7.5.1 Authorities and regulations

According to Section 136 a of the Building Act, applications for siting permission shall be submitted to the government for each facility for the handling and storage of high-level material.

Opinions concerning such applications shall be obtained from the Commander-in-Chief of the Swedish Armed Forces, who shall hereby judge the proposed site in the light of defence plans. These opinions shall be considered by the government in its review of the siting application.

Facilities which may be vital to the country's power supplies in wartime shall be examined by the Board for the Wartime Protection of Power Stations for approval of the protection level of the facilities.

Facilities for the handling and storage of spent nuclear fuel are, however, not directly necessary for power production and distribution in wartime. They are therefore not among the types of facilities which the Board for the Wartime Protection of Power Stations is instructed to deal with.

The requirements of wartime protection pertains first and foremost to protection against damage which may lead to releases of radioactivity. SKI shall therefore issue the directives and guidelines which may be called for from the viewpoint of wartime protection, in consultation with the Commander-in-Chief of the

Swedish Armed Forces and the National Institute of Radiation Protection.

7.5.2 Design of facility

The emplacement of the intermediate storage facility in rock provides good protection against conventional weapons. Conventional bombs cause ground vibrations upon contact, but the 30 m rock cover should insulate the facility against such damaging factors. Possible effects shall be taken into consideration in the design and construction of the concrete enclosure of the rock chambers. The final repository, with a rock cover of 500 m, is adequately protected even from nuclear weapons.

The effects of airborne shock waves in the intermediate storage facility caused by a bomb exploding outside of the facility have been reduced by the design of the access descents to provide a blow-through path. Ventilation of the intermediate storage facility has been designed in such a way that cooling can be accomplished by natural convection in the event of fan failure. This also provides some protection against airborne shock waves. This protection is enhanced by means of a stack design which permits blow-through at the intake and outlet points.

The intake and outlet openings of the ventilation stacks are protected by concrete cover. The lower portions of the stacks may be constructed in the form of thick concrete cylinders. The vertical portion of the ventilation shafts down into the rock are provided with bomb traps.

Electric lead-in bushings into the rock cavern facilities can be bomb-protected by concrete encasement and drainage openings. Where required, redundant connections can be provided at safe distances.

REFERENCES

CHAPTER 2

2-1 IAEA Safety Series No. 6, Regulation for the Safe Transport of Radioactive Materials, 1973 Edition.

2-2 Salénrederierna, Tekniska Avdelningen. Sjötransport av använt kärnbränsle och radioaktivt avfall till svenskt centrallager, 1977.
("Salén Shipping Line, Technical Department. Sea transports of spent nuclear fuel and radioactive waste to a Swedish central storage facility, 1977.")

2-3 Saint Gobain Techniques Nouvelles, Project for the handling and storage of vitrified high-level waste. KBS Technical Report No. 35.

CHAPTER 3

3-1 Programrådet för radioaktivt avfall. Centralt lager för använt bränsle. En förstudie.
("National Council for Radioactive Waste Management. Central Storage facility for spent fuel. A preliminary study.") July 1977.

CHAPTER 4

4-1 Sanders, D M, Hench, L L, "Mechanisms of Glass Corrosion", J. Amer. Ceram. Soc., 56(1973)343.

4-2 Mendel, J E, Review of leaching Test Methods and the Leachability of Various Solid Media Containing Radioactive Waste, BNWL-1765(1973).

4-3 Blomqvist, G, Utlakning av franskt, engelskt och kanadensiskt glas med högaktivt avfall,
("Leaching of French, English and Canadian glass containing high-level waste,")
KBS Technical Report No. 8, 1977-05-20.

4-4 Merritt, W F, AECL, The Leaching of Radioactivity from Highly Radioactive Glass Blocks Buried below the Water Table: Fifteen Years of Results. IAEA SM-207/9. International Symposium on the Management of Radioactive Waste from the Nuclear Fuel Cycle, Vienna, 1976-03-22--26.

4-5 Diebold, F E, Discussion on Glass-Water Interactions, ARH-2905(1973).

4-6 Doremus, R H, Glass Science, Wiley-Interscience, New York 1973, pp. 242-8.

4-7 Ref. 5, p 23.

4-8 Imbert, J C Pacaud, F, Contribution à l'étude de la diffusion en rélation avec la lixiviation des verres. CEA-R-4550 (1974).

4-9 Lakatos, T, Diskussion om glasegenskaper hos AERE i Harwell 1977-03-10. Glasforskningsinstitutets rapport nr 3277(1977). ("Discussion of glass properties of AERE at Harwell, 1977-03-10. Report No. 3277(1977) of the Swedish Glass Research Institute.")

4-10 Present status of HLW vitrification process in France. COGEMA, 9 March 1977.

4-11 Laude, F, La verre comme première barrière pour le stockage à long terme de déchets de haute activité. Proceedings of the Workshop on Risk Analysis and Geologic Modelling. NEA/JRC Ispra 1977-05-23--27.

4-12 Blomqvist, G, Lakningsförsök med högaktivt franskt glas i Studsvik. ("Leaching tests with high-level French glass at Studsvik,")
KBS Technical Report No. 50.

4-13 F Laude, R Bonniaud, C Sombret, G Rabot. CEA, Centre de Marcoule. Confinement de la radioactivité dans les verres. IAEA-SM-207/36.

4-14 R Bonniaud, F Pacaud, C Sombret, CEA, Marcoule Quelques aspects du comportement de radioélements confinés à long terme sous forme de verre ou de produit à fort fase vitreuse. Verres et Réfractaires, Vol 29.1. Jan-Feb. 1975.

4-15 Project for the handling and storage of vitrified high-level waste, St Gobain, Techniques Nouvelles, KBS Technical Report No. 35.

4-16 A D W Corbet, G G Hall, G T S Spiller, Problems in the design and specifications of containers for vitrified high level liquid waste, IAEA-SM-207/3, March 1976.

4-17 T Lakatos, Beständighet hos borsilikatglas ("Durability of borosilicate glass"), KBS Technical Report No. 44.

4-18 R Bonniaud, La vitrification en France des solutions de produits de fission, Nuclear Technology, Vol. 34, Aug. 1977, p. 449.

4-19 J B Morris, B E Chidley, AERE, Harwell, Preliminary experiences with the new Harwell inactive vitrification pilot plant. IAEA-SM-207/22.

4-20 R Bonniaud, C Sombret, Statement of research in the field of solidification of high level radioactive waste in France, 77th Meeting of American Ceramic Society, May 3-8 1975, Washington D.C.

4-21 COGEMA: Le retraitement des combustibles irradiés à La Hague.

4-22 IVA. Special Report, France 1976:8.

4-23 COGEMA, Etablissement de LA HAGUE, Propositions et recommandations approvées par le CHS élargi de La Hague, 17 June 1977.

4-24 Uppgift från personalrepresentanter i hälso- och säkerhetskommittén vid La Hague till Sydkrafts styrelses arbetsutskott, oktober 1977.
("Information from personnel representatives on health and safety committee at La Hague received by executive committee of Sydkraft Board, October 1977.")

4-25 Uppgift från företagsledningen i La Hague till Sydkrafts styrelses arbetsutskott, oktober 1977.
("Information from company management at La Hague received by executive committee of Sydkraft Board, October 1977").

4-26 Service Prevention Radioprotection, COGEMA, Etablissement de La Hague, Surveillance de la radioactivité de l'environnement de l'établissement de La Hague.

4-27 R Bonniaud, C Sombret, A Barbe. Chemical Engineering Process, March 1976 p. 47.

4-28 R Bonniaud, C Sombret, L Rozand, A Barke, P Auchapt, J A Coste, French Industrial Plant AVM for Continuous Vitrification of High Level Radioactive Wastes. American Institute of Chemical Engineers SYMPOSIUM SERIES, Vol. 72, No. 154 pp. 145-150.

4-29 C Blomqvist, Beräkning av utlakning av vissa fissionsprodukter och aktinider från en cylinder av franskt glas, KBS Teknisk rapport 24.
("Calculations of leaching of certain fission products and actinides from a cylinder of French glass, KBS Technical Report No. 24.")

4-30 H Häggblom, Diffusion av cesium genom kiselsyra, AB Atomenergi, meddelande 771025.
("Diffusion of cesium through silicic acid, AB Atomenergi, bulletin 771025.")

4-31 G Blomqvist, Sammanställning av molybdatfas, AB Atomenergi AB-M-351.
("Review of molybdate phase, AB Atomenergi AP-M-351.")

4-32 R Bonniaud, Les actinides dans les verres, B.I.S.T. Commissariat à l'Energie Atomique, September 1976, page 47.

4-33 A R Hall, J T Dalton, B Hudson, J A C Marples, Development and radiation stability of glasses for highly radioactive waste. IAEA Symposium, Vienna, March 1976. IAEA-SM-207.

4-34 Grundsätzliche sicherheitstechnische Realiserbarkeit des Entsorgungszentrums, Beurteilung und Empfehlungen der RSK und der SSK, Geschäftstelle der Reaktor - Sicherheitskommission, October, 1977.

CHAPTER 5

5-1 K Ekberg, N Kjellberg, G Olsson, Resteffektstudier för KBS, Del 1 Litteraturgenomgång, Del 2 Beräkningar, AB Atomenergi, KBS Teknisk Rapport No. 7.
("Decay heat studies for KBS, Part 1 Review of the literature, Part 2 Calculations, AB Atomenergi, KBS Technical Report No. 7.")

5-2 Saint Gobain Techniques Nouvelles, Project for the handling and storage of vitrified high-level waste. KBS Technical Report No. 35.

5-3 H Folke Sandelin AB, Avesta Jernverk, VBB, Asea-Kabel, Institutet för metallforskning, Tillverkning av blytitan-kapsel, KBS Teknisk Rapport nr 34.
("Institute for Metals Research, Fabrication of lead-titanium canister, KBS Technical Report No. 34.")

5-4 S Henriksson, Uppskattning av livslängden hos behållare av 24 Cr 12 Ni-stål, Incoloy 800 och Hastelloy C-4, vid förvaring av förglasat upparbetat kärnbränsleavfall, AB Atomenergi AE-MS-151, 1977-06-13.
("Estimate of service life of container of 24 Cr 12 Ni steel, Incoloy 800 and Hastoloy C-4, in the storage of vitrified reprocessed nuclear fuel waste, AB Atomenergi AE-MS-151, 1977-06-13.")

5-5 S Henriksson, Utredning rörande titans lämplighet som korrosionshärdig kapsling för kärnbränsleavfall. AB Atomenergi, KBS Teknisk Rapport nr 11.
("Study concerning suitability of titanium as a corrosion-resistant encapsulation material for nuclear fuel waste. AB Atomenergi, KBS Technical Report No. 11.")

5-6 M de Pourbaix, S Henriksson, Metallisk kapsling, Resultat av korrosionsprovning av titan under 30 dygn.
("Metallic encapsulation. Results of corrosion testing of titanium for 30 days.")
AB Atomenergi, TPM-MS-156, 1977-08-04.

5-7 M de Pourbaix, S Henriksson, Metallisk kapsling. Resultat av korrosionsprovning av titan under 100 dygn.
("Metallic encapsulation, Results of corrosion testing of titanium for 100 days.")
AB Atomenergi, TPM-MS-170, 1977-09-20.

5-8 W R Fischer, Zur Lochfrasskorrosion, Diss. Technische Hochschule, Braunschweig, 1964.

5-9 G Schef, Långtidsförsök med ATi-24 i 25% NaCl vid 130°C. ("Long-term test with ATi-24 in 25% NaCl at 130°C:") Avesta Jernverk, LAK 91/68, 1968-11-04.

5-10 I Multer, Discussion med L C Covington hos Timet beträffande användningen av titan för avfallsförvaring. Vattenfall 1977-06-20.
("Discussion with L C Covington at Timet regarding the use of titanium for waste containment. Swedish State Power Board 1977-06-20.")

5-11 G Eklund, Korrosion av bly. Bilaga 4 till referens 5-12. ("Corrosion of lead. Appendix 4 to reference 5-12.")

5-12 Korrosionsinstitutet och dess referensgrupp, Bedömning av korrosionsbeständigheten hos material avsedda för kapsling av kärnbränsleavfall. Lägesrapport 1977-09-27 samt kompletterande yttranden. KBS Teknisk Rapport nr 31. ("The Swedish Corrosion Research Institute and its reference group, Evaluation of the corrosion resistance of materials intended for the encapsulation of nuclear fuel waste. Status report 1977-09-27 plus supplementary statements. KBS Technical Report No. 31.")

5-13 G Wranglén, Metallers korrosion och ytskydd, ("Corrosion and surface protection of metals"), page 259, Almqvist and Wiksell, Stockholm, 1967.

CHAPTER 6

6-1 Vattenfall, VBB, GAK, Utformning av bergrumsanläggningar. ("Design of rock cavern facilities"), KBS Technical Report No. 38.

6-2 R Blomqvist, Temperaturberäkningar för förglasat avfall. ("Temperature calculations for vitrified waste.") Report 3, AB Atomenergi, KBS Technical Report No. 45.

6-3 Saint Gobain Techniques Nouvelles, Project for the handling and storage of vitrified high-level waste. KBS Technical Report No. 35.

6-4 H Fagerström, B Lindahl, Hantering av buffertmaterial av bentonit och kvartssand.
("Handling of buffer material of bentonite and quartz sand.") VBB and Stabilator, KBS Technical Report No. 37.

6-5 S Knutsson, R Pusch, PM angående värmeledningstal hos jordmaterial.
("Memorandum concerning coefficient of thermal conductivity of soil material.")
Luleå Institute of Technology, KBS Technical Report No. 2.

6-6 A Jacobsson, R Pusch, Deponering av högaktivt avfall i borrhål med buffertsubstans.
("Deposition of high-level waste in boreholes with buffer substance.")
Luleå Institute of Technology, KBS Technical Report No. 3.

6-7 A Jakobsson, R Pusch, Deponering av högaktivt avfall i tunnlar med buffertsubstans.
("Deposition of high-level waste in tunnels with buffer substance.")
Luleå Institute of Technology, KBS Technical Report No. 4.

6-8 R Pusch, Influence of cementation on the deformation properties of bentonite/quartz buffer substance. Luleå Institute of Technology, KBS Technical Report No. 14.

6-9 R Pusch, The influence of rock movement on the stress/strain situation in tunnels or bore holes with radioactive canisters embedded in a bentonite/quartz buffer mass.
Luleå Institute of Technology, KBS Technical Report No. 22.

6-10 R Pusch, Water uptake in a bentonite buffer mass. A model study. Luleå Institute of Technology, KBS Technical Report No. 23.

6-11 S Knutsson, Värmeledningsförsök på buffertsubstans av bentonit/pitesilt.
("Thermal conductivity tests of buffer substance of bentonite/pitesilt.")
Luleå Institute of Technology, KBS Technical Report No. 28.

6-12 R Pusch, A Jacobsson, Long term mineralogical properties of bentonite/quartz buffer substance. Luleå Institute of Technology, KBS Technical Report No. 32.

6-13 R Pusch, Buffer mass composition with specific reference to required physical and mechanical properties. Luleå Institute of Technology, KBS Technical Report No. 33.