Waste Isolation Pilot Plant

Compliance Certification Application

Reference 347

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

I General
II Geology
III Facilities
IV Safety analysis
V Foreign activities

KÄRN-
BRÄNSLE-
SÄKERHET

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

I General

KÄRN-
BRÄNSLE-
SÄKERHET

HANDLING OF SPENT NUCLEAR FUEL AND FINAL STORAGE OF VITRIFIED HIGH LEVEL REPROCESSING WASTE

SUMMARY

In april 1977 the Swedish Parliament passed a Law, which stipulates that new nuclear power units can not be put into operation unless the owner is able to show that the waste problem has been solved in a completely safe way. The task of investigating how radioactive waste from a nuclear power plant should be handled and stored was previously the responsibility of the National Council for Radioactive Waste Management (PRAV). This Council was formed in November 1975 as the result of a proposal made by the Government Committee on Radioactive waste (the AKA Committee).

In response to the Government bill proposing the Law, the power industry decided in December 1976 to give top priority to the investigation of the waste problem in order to meet the requirements of the Law. Therefore, the Nuclear Fuel Safety Project (KBS) was organized. The first report from the KBS project entitled "Handling of spent nuclear fuel and final storage of vitrified high level reprocessing waste" was submitted in December 1977.

The requirements of the Law regarding completely safe storage

The Law stipulates that the owner of a reactor must show how and where a completely safe storage can be provided for either the high level reprocessing waste or the spent, unreprocessed nuclear fuel. "The storage facility must be arranged in such a way that the waste or the spent nuclear fuel is isolated as long a time as is required for the activity to diminish to a harmless level". "These requirements implies that measures should be taken which, during all phases of the handling of the spent nuclear fuel, can ensure that there will be no damage to the ecological system".

In the strictest meaning of the word, no human activity can be considered completely safe. The fact that such an interpretation of the wording of the Law was not intended is evident from the formulation of the statements made by the Government in support of the Law indicating that the storage of waste shall fulfil "the requirements imposed from a radiation protection point of view and which are intended to provide protection against radiation damage". Questions regarding protection against radiation damage

are regulated by the Radiation Protection Act. This means that the requirements imposed on the handling and storage of high-level waste are, in principle, the same as those which apply for other activities involving the handling of radioactive substances.

This interpretation is supported by the statements made by the Committee of Commerce and Industry in its review of the Law, in which the Parliament also concurred. The Committee thus finds the expression "completely safe" to be warranted in view of the very high level of safety required, but considers that a "purely Draconian interpretation of the safety requirement" is not intended. Draconian means "excessively severe, inhuman".

The requirements of the Law regarding the scope of this report

In the statements made by the Government in support of the Law it is said: "The descriptions to be submitted by the owner of the reactor shall include detailed and comprehensive information for the evaluation of the safety. Consequently, over-all plans and drawings will not suffice. Furthermore, it should be specifically stated in which form the waste or spent nuclear fuel is to be stored, how the storage is to be arranged, how the transportation of the spent nuclear fuel or of the waste will be carried out and whatever else may be required in order to ascertain whether the proposed final storage can be considered completely safe and possible to construct."

To fulfil these requirements, this report presents relatively detailed information on the design of facilities and the transportation systems which are part of the handling and storage chain. Certain parts of this information are relatively unessential for evaluating the safety of the waste storage, while others are vital. A detailed evaluation of the safety aspects of the proposed design is presented in a safety analysis. The handling and processing carried out abroad is also described, although more in general.

The alternatives given in the Law

The Law requires a description of the handling and final storage of either the high level reprocessing waste or the spent, unreprocessed nuclear fuel. This report deals with the first alternative. An application to the Government to charge nuclear fuel to a new reactor based on this alternative must, in addition to this report, include an agreement which covers in a satisfactory manner the anticipated need for reprocessing of spent nuclear fuel. This aspect is, however, not dealt with in this report.

A report on the second alternative, i.e. spent unprocessed fuel, is planned for publication during the firsthalf of 1978.

Layout of the report

This report has been divided into five volumes as follows:

I General
II Geology
III Facilities
IV Safety analysis
V Foreign activities

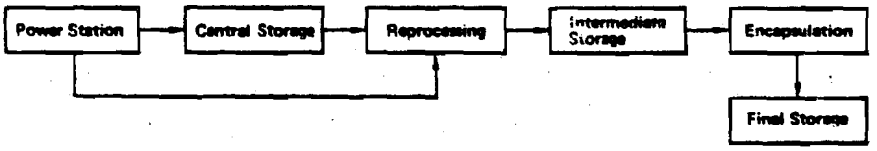
In order to provide a basis for the report, KBS has carried out a great number of technical-scientific investigations and surveys. The results of these are published in KBS Technical Reports. 56 volumes of these reports have so far been published, (see volume I, appendix 3.)

Volume I (General) can be read independently of the other volumes. It comprises mainly a summary of the more detailed reports presented in volumes II, III and IV.

Chapter 3 in volume I is a summary of the proposed method for handling and storage of nuclear fuel and high-level waste from the nuclear power plant fuel pools up to and including final storage in Swedish bedrock.

Chapter 13 in volume I summarizes the more detailed presentation of the safety analysis in volume IV. This chapter summarizes the safety evaluations of the whole handling chain from a radiological point of view. The effects of radiation have been calculated for normal conditions and for accidents. Special emphasis has been placed on the long-term aspects of the final storage of high level waste.

Final stage of nuclear fuel cycle



The handling chain for spent nuclear fuel and high-level reprocessing waste is illustrated in the above block diagram.

Nuclear power stations always have storage pools for spent nuclear fuel. They are needed so that the fuel can be discharged from the reactor and also to provide storage space for spent nuclear fuel before it is dispatched for reprocessing or for storage elsewhere.

Today, the available reprocessing capacity is limited, and it is not clear to what extent spent nuclear fuel will be reprocessed. As a result, it is necessary to extend the storage capacity for spent nuclear fuel. For economic reasons and for the planning of the back end of the nuclear fuel cycle, the extended capacity should not be provided at the nuclear power stations. Instead, a

central fuel storage facility should be constructed. This facility is needed regardless of whether the spent nuclear fuel is to be reprocessed or not before final storage. The fuel can be stored in this facility for about ten years.

As a rule, radioactive waste must be stored in the country where it is produced. The high-level reprocessing waste will be sent back to Sweden in vitrified form in 1990 at the earliest. The vitrified waste will be contained in stainless steel cylinders having a diameter of 40 cm and a height of 1.5 m. If all of the fuel is reprocessed, 9 000 cylinders will be obtained from 13 reactors that have been in operation for 30 years.

The waste cylinders will be placed initially in an intermediate storage facility where they will remain for at least 30 years before being transferred to the final storage. The cylinders will be kept in dry conditions in the intermediate storage facility, and radiactive substances cannot be released to the environment. During this storage period, the amount of heat generated by the waste will be reduced by half, thus simplifying final storage. Intermediate storage postpones the date when final storage must commence, thus providing more time to optimize the final storage method. A longer storage period than 30 years is entirely possible. Such a prolonged storage period is considered in France, for example. However, intermediate storage requires a certain amount of supervision, even though this supervision is very limited.

It is planned that the final storage, which will not have to go into operation until 2020 at the earliest, will be constructed in rock about 500 metres underground. The facility is designed in such a way that it can be sealed and ultimately abandoned. In the final storage, the waste will be exposed to the ground-water in the rock. After intermediate storage and before the waste cylinders are transferred to the final storage, they will therefore be encapsulated in a canister made of titanium and lead. These materials have good resistance to corrosion.

The siting of the facilities for the various handling stages may be arranged in different ways, in accordance with what is deemed to be practical.

Spent fuel has already been shipped abroad from Sweden for reprocessing. Similar transports will also be required between the various phases of the handling. The design and procurement of transport casks and vehicles thus form part of the waste handling.

Geological requirements for a final storage

Extensive investigations and tests have been carried out to determine the suitability of Swedish bedrock for final storage. In this connection, interest has been concentrated on precambrian chrystalline rocks. In other countries, studies have been made of storage in salt, shale and clay depending upon the natural prerequisites of each country.

Field investigations have been carried out at five sites, three of which have been selected for more detailed studies. A number

of holes have been drilled to a depth of 500 metres. It should be emphasized that the objective of this work was not to find a site now to be proposed for final storage. The purpose was to show that suitable bedrock is available within Sweden for such a facility.

The factors that will determine the suitability of a rock formation for final storage are its permeability and strength, the composition of the groundwater and its flow pattern and the delaying effects on radioactive substances when groundwater passes through cracks in the rock. Of special interest is also the risk of rock movements which could affect the pattern of groundwater flow or damage the encapsulated waste.

Assessing these factors, a depth of about 500 metres is considered to be suitable. At this depth, the bedrock contains fewer cracks and has lower water permeability than closer to the surface. This depth also gives a satisfactory protection against acts of war and such extreme events as meteorite impacts and the effects of a future ice age.

The investigations and surveys carried out have shown that the three sites selected offer satisfactory conditions for final storage. At these sites, the bedrock consists of Sweden's most common types of rock - granite, gneiss and gneissified granodiorite. Consequently, it is reasonable to expect that rock formations with equivalent conditions are also available at many other places within Sweden.

Safety of the handling chain

The extensive safety analysis carried out has shown that the release of radioactive substances which could occur in connection with normal operation or with an accident in the different stages of the handling chain within Sweden, would be insignificant in comparison with corresponding conditions at a nuclear power station. This is because the vitrified waste has a low temperature and is encapsulated without overpressure. Consequently a sudden and extensive release of radioactivity can not occur. The safety of the steps of the handling chain, which will be carried out abroad (reprocessing and vitrification), will be evaluated by Government authorities in the country concerned and are dealt with in a more superficial manner in this report.

Radioactive substances from a final storage can only be released by the groundwater. The final storage must be arranged in such a way that such a release cannot damage the ecological system. It is then important to remember that the activity of the radioactive substances in the waste diminishes very slowly. The final storage is therefore arranged so that the migration of these substances is either prevented or delayed for a long time, thus ensuring that the concentration of radioactive substances which may reach the biosphere will be harmless. For this reason, the design of the final storage provides for a number of successive barriers.

For any release of radioactive substances in the waste to the environment, the groundwater must first penetrate both the canister made of titanium and lead and the stainless steel container.

These materials have excellent resistance to corrosion. The waste cylinders will be placed in holes drilled into good-quality rock and surrounded by a buffer material consisting of quartz sand and bentonite. Since the buffer material has a low permeability, only very small amounts of water will be able to affect the encapsulated waste.

In the event of the penetration of the canister and the stainless steel container, the groundwater can affect the vitrified waste. However, the glass has a very low leaching rate under the conditions that prevail in the final storage.

The low flow rate of the groundwater, the long distance which the water must cover to reach the biosphere and the chemical processes in the crack system in the rock and in the buffer material provide effective barriers that prevent and delay the migration of the radioactive substances. Moreover, dilution in huge volumes of groundwater will take place before entry into the biosphere.

The safety of the final storage of high-level waste is dominating the safety issue. The safety analysis is based, in each phase that entails uncertainty, on assumptions and data that provide a reassuring margin of safety. Possible routes for the migration of radioactivity to the biosphere have been studied in the safety analysis, and the group of people which can be exposed to the highest level of radiation has been identified (the critical group). The critical group consists of persons taking their drinking water from a deep well drilled in the vicinity of the final storage. Under unfavourable circumstances this group can be exposed to a maximum radiation (individual dose) of 13 millirem per year in addition to natural background radiation.

This maximum additional dose of 13 millirem per year will not occur until after about 200 000 years. This long delay is caused by the retainment in the buffer material and the rock of the radioactive substances providing the highest additional dose. Radioactive substances which are not delayed relative to the flow of water in the bedrock could come into contact with the biosphere after only some hundreds of years. However, the additional dose attributable to these substances is very much lower than the value given above.

An individual dose of 13 millirem is considerably lower than the dose recommended by the International Commission on Radiological Protection (ICRP) as the upper limit for permissible additional doses for individuals namely 500 millirem per year. This limit is intended to protect individuals against delayed radiation effects such as cancer and genetic effects.

Governmental authorities impose lower limits for the operation of nuclear power plants. In Sweden, operational restrictions can be imposed and other measures taken if the additional dose tends to exceed 50 millirems per year for people living near the power plant.

In order to reduce radiation exposure as much as reasonably possible, the Swedish Radiation Protection Institute requires that nuclear power plants be designed and constructed so that the expected additional dose for the critical group living in the vicinity of the plant is less than 10 millirems per year.

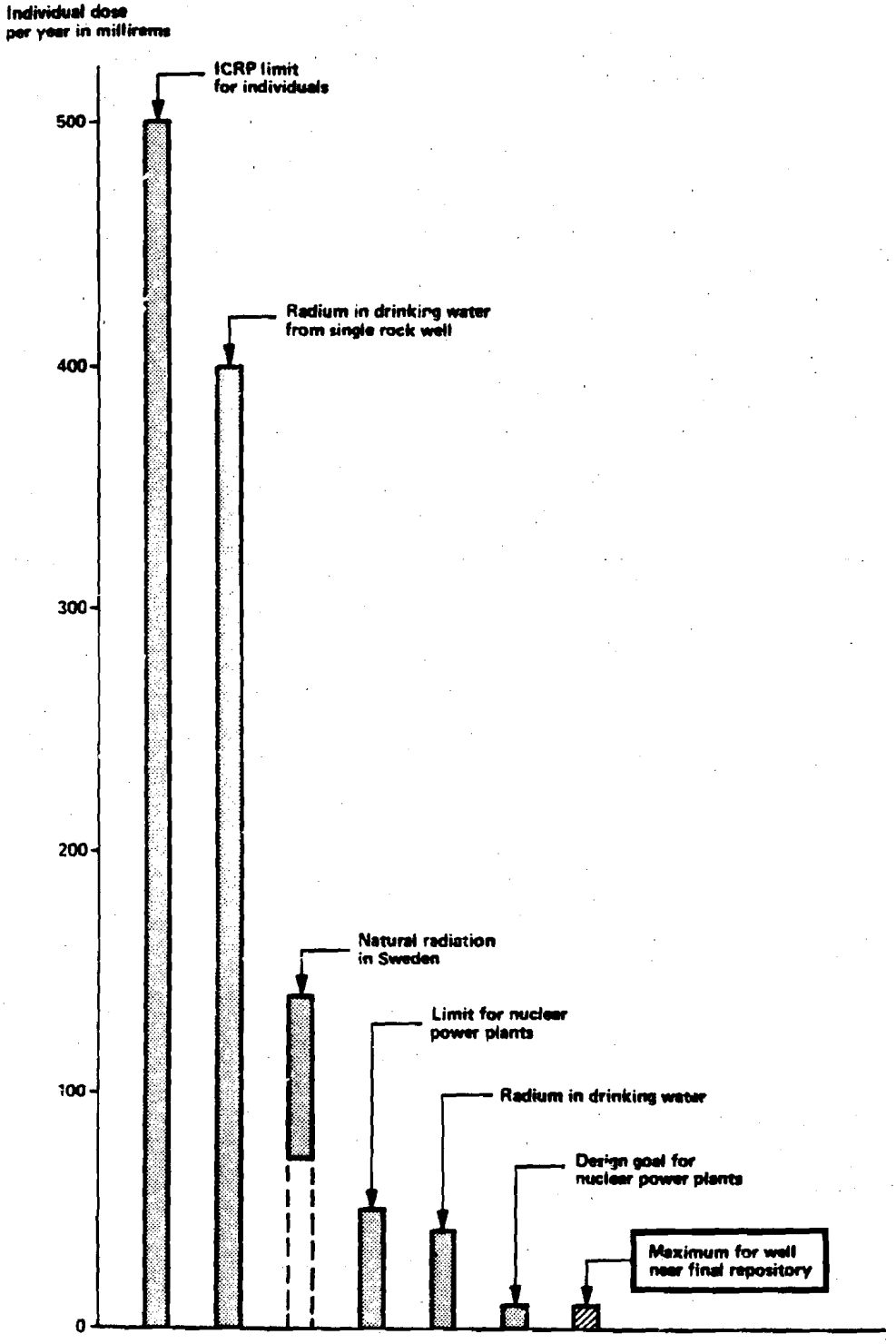
As mentioned above, the assumptions and data used in the safety analysis were selected with safety margins. It is considered probable that the dosage will be approximately 1/100th of the maximum value of 13 millirems per year given above. One reason for this is that the very low rate of water flow in the bedrock is not sufficient to break through the encapsulation or leache the vitrified waste at the rates assumed in the safety analysis presented in this report. However, verification of this lower value would require additional investigations not yet been completed.

The following bar-chart shows the dose rates mentioned above. It also indicates the dose rates from natural radiation in Sweden. As appears from the bar-chart local variations in natural radiation are considerably greater than the maximum contribution from a final storage of high-level waste obtained from 13 reactors which have been in operation during 30 years. The bar-chart also shows that the doses obtained from radium in natural drinking water in Sweden often lies considerably above the level reported for a final storage.

Moreover, the safety analysis shows that radiation doses for large population groups attributable to a final storage will be virtually insignificant and that the longterm effects on health will be negligible.

The design of the back end of the nuclear fuel cycle presented in this report thus fulfils the requirements set forth in the Law for a completely safe final storage of the high-level reprocessing waste.

Stockholm November 1977
NUCLEAR FUEL SAFETY PROJECT (KBS)



Bar graph showing the calculated maximum annual radiation doses which the final repository can give to a nearby resident and the annual dose to man from some natural radiation sources plus some established dose limits. The dose from drinking water comes from radium-226.

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1 INTRODUCTION

1.1 BACKGROUND

1.1.1 General

As of the end of 1976, 187 nuclear power reactors for civilian energy production were in operation in the world. The total installed capacity 80 GW(e), corresponds to 80 reactors of 1 000 MW(e) each.

Sweden's first commercial nuclear power plant, a lightwater reactor in Simpevarp outside of Oskarshamn, was commissioned for electrical power production in 1972. Since then, a number of reactors have been completed and there are now six nuclear units in operation in the country.

Facility	Owner	Commissioned	Capacity
Oskarshamn 1	OKG	1972	450 MW
Oskarshamn 2	OKG	1974	580
Ringhals 1	Swedish State Power Board	1976	760
Ringhals 2	Swedish State Power Board	1975	820
Barsebäck 1	Sydkraft	1975	580
Barsebäck 2	Sydkraft	1977	580

An additional six units are in different stages of construction and planning.

Facility	Owner	Ready for fueling	Capacity
Ringhals 3	Swedish State Power Board	1977	900 MW
Ringhals 4	Swedish State Power Board	1979	900
Forsmark 1	FKA	1978	900
Forsmark 2	FKA	1980	900
Forsmark 3	FKA	?	1 000
Oskarshamn 3	OKG	?	1 000

Throughout the '70s, there has been an intensive public debate in Sweden concerning problems pertaining to the safety aspects of nuclear power production and whether such production is desirable at all. From having been concentrated on problems associated with normal operation and failures during the first years, the debate has shifted emphasis in recent years to questions concerning the management of the radioactive waste arising from nuclear power

production. The problems involved with these radioactive wastes were studied by a working group appointed by the Swedish National Institute of Radiation Protection in June of 1971. This working group submitted a report in May of 1972 containing proposed guidelines for the management of radioactive waste /1-1/. This report concludes that the management of high-level waste from spent nuclear fuel "entails problems to which as yet only partial solutions have been found".

In December of 1972, the Swedish Government decided to appoint an ad hoc committee to investigate the problems related to high-level waste from nuclear power plants called the Aka Committee. The Aka Committee's findings are discussed in greater detail under 1.1.2 below.

The Government which entered office following the 1976 election set up certain conditions for the granting of permission for power utilities to charge new nuclear reactors with nuclear fuel. These conditions stipulated that the plant-owner must demonstrate where and how an absolutely safe final storage of the high-level waste can be arranged. The conditions are set forth in the "Conditions Act" ("Law concerning special permission for charging nuclear reactors with fuel") which was passed by the Swedish Parliament in April 1977 /1-3/.

In order to produce and compile material for the reports required by the Conditions Act, the power utilities formed the Nuclear Fuel Safety Project (KBS). The findings of the Aka Committee constitute the basic point of departure for the work of this Project. Research work initiated by the Aka Committee within, for example, the field of geology has been carried further by KBS. The present KBS report deals with the handling and final storage of vitrified high-level waste obtained from the reprocessing of spent nuclear fuel.

1.1.2 The Aka Committee

The Government Committee on Radioactive Waste (Aka Committee) appointed in 1972 was given further directive in May of 1974 to extend the scope of their study to include the handling and storage of low- and medium-level waste as well.

The Aka Committee submitted its findings to the Government in April of 1976 /1-2/. Its conclusions and proposals concerning the handling and final storage of high-level waste can be summarized as follows:

1 Current technology already provides satisfactory means for handling and storing spent nuclear fuel and radioactive waste.

2 It is imperative that the Swedish power utilities procure a transportation system for spent nuclear fuel as soon as possible. It is recommended that spent Swedish nuclear fuel and radioactive waste which requires heavy radiation shielding be shipped by rail or sea, whenever possible.

3 A central facility for the storage of spent nuclear fuel is needed in the country.

4 The preliminary planning of a Swedish reprocessing plant should commence as soon as possible.

5 A decision to build a Swedish reprocessing plant should also include a plant for the manufacture of plutonium-enriched fuel.

6 Solidification of the high-level waste from reprocessing in glass or ceramic material is the best method currently available for converting liquid high-level waste to solid form.

7 Studies aimed at further elucidating the requirements which must be met for the final storage of non-reprocessed spent nuclear fuel should be commenced.

8 Final storage of radioactive waste should be effected in bedrock.

9 Detailed geological studies of sites suitable for final storage should be initiated at once.

10 The power producer shall defray all costs associated with the handling and storage of spent nuclear fuel and radioactive waste.

11 It is proposed that a special government organization be formed to assume responsibility for the long-term management of radioactive waste and associated activities.

12 The proposals made by the Aka Committee regarding the management of spent nuclear fuel and radioactive waste require a comprehensive programme of research and development.

The Committee was unanimous in its proposals. Special supplementary statements were submitted by two members.

Reactions to the conclusions and proposals of the Aka Committee's report are largely positive. Criticism has been directed at those parts of the Committee's report which deal with the final storage of high-level waste, more particularly at the report's assessment of the rate of corrosion of the canister material and the extent of cracking in the bedrock. The need for further research is emphasized by many parties, especially with regard to the properties of the bedrock at greater depth. Many parties warn against a hasty commitment to a particular method for the handling of the spent fuel and the final storage. The need for a safety analysis is also emphasized.

KBS has now completed studies within the above-mentioned areas designated by the Aka report as being urgent for further study, except for the preliminary planning of a Swedish reprocessing plant. Aspects of organization and financing have not been covered by the KBS Project.

1.2 REQUIREMENTS TO BE FULFILLED BY THE NUCLEAR POWER INDUSTRY

1.2.1 Government Statement of Policy

The Government summarized its views on nuclear power in its Statement of Policy dated 8 October 1976:

"Nuclear power involves great problems and hazards. Foremost among these is the handling of the spent fuel and the high-level waste. A commitment to nuclear power cannot be made until these problems and hazards have been satisfactorily brought under control. In view of these problems, nuclear energy plants currently under construction may not be commissioned until the power company concerned can present an acceptable agreement for the reprocessing of spent nuclear fuel and demonstrate how and where an absolutely safe storage of the high-level waste can be effected. Barsebäck 2, which is completed, will be taken out of operation if a reprocessing agreement is not produced by 1 October 1977. The Government intends to enter into negotiations concerning these matters with Svensk Kärnbränsleförsörjning AB (Swedish Nuclear Fuel Supplies Inc.) and the concerned power utilities as soon as possible."

1.2.2 Conditions Act

The "Law concerning special permission for charging nuclear reactors with fuel" /1-3/ sets forth the conditions contained in the Government Statement of Policy.

§2 of the Act provides for the commissioning of nuclear reactors:

"If an application for final approval for the commissioning of the nuclear reactor has not been submitted to the Nuclear Power Inspectorate as of October 1976, the reactor may not be charged with nuclear fuel without the special permission of the Government. Permission may be granted only providing that the reactor owner

1 has produced an agreement which adequately satisfies the requirement for the reprocessing of spent nuclear fuel and has demonstrated how and where an absolutely safe final storage of the high-level waste obtained from the reprocessing can be effected, or

2 has demonstrated how and where an absolutely safe final storage of spent, un-reprocessed nuclear fuel can be effected."

1.2.3 Accountability of the nuclear power industry

The Conditions Act specifies that reports submitted by power station owners concerning the final storage of waste from reprocessed nuclear fuel shall describe the absolutely safe storage of "the high-level waste obtained from reprocessing".

Various definitions have been used for the term "high-level waste". The Aka Committee has, for example, used two definitions:

- High-level waste is waste which contains such a high level of radioactivity that it requires not only effective radiation shielding but also cooling in order to be stored in a safe manner / 1-4, page 34 /, and

- High-level waste - the waste containing fission products which is separated from the spent fuel during reprocessing /1-2, part II, page 201/.

A more precise definition based on the origin of the waste is used by the Nuclear Regulatory Commission (NRC) in the United States:

- "High-level liquid radioactive wastes" means those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels /1-5/.

Definitions based on the level of radioactivity per unit volume have also been used / 1-6/ .

Radioactivity and cooling requirements change with time during the handling and long-term storage of waste. For this reason, a definition of high-level waste based on the origin of the waste has been deemed most suitable.

In the KBS Project, the "high-level waste obtained from reprocessing" has been defined as:

- the waste with a high content of fission products which is obtained as the aqueous phase in the extraction process in the reprocessing of spent nuclear fuel.

This high-level waste will be converted to vitrified form and eventually returned to Sweden.

Other types of radioactive waste which can contain small quantities of uranium and plutonium is also obtained from reprocessing. This long-lived "alpha waste" must be specially treated prior to final storage. Methods for this treatment are not dealt with in this report. Final storage of this waste can be effected in a manner which is similar to but simpler than that which has been proposed for high-level waste. Nor does the report deal with the use of the uranium and plutonium which is obtained from reprocessing and which cannot be regarded as waste. The recovered uranium is reused in the production of nuclear fuel. The plutonium can also be used for this purpose. The use of plutonium extracted from Swedish nuclear fuel requires Government approval.

In a special explication of the Conditions Act, the accountability requirements imposed on power plant owners for describing waste storage methods have been specified in greater detail. These requirements are summarized below:

1 It is the responsibility of the reactor owner to demonstrate concrete solutions to the waste problems associated with nuclear power production.

2 In order to be granted permission to commission nuclear reactors, owners must demonstrate that the spent nuclear fuel and the high-level waste it contains will be handled in such a manner that the ecological system will not be damaged. The reactor owner must demonstrate a) how the spent nuclear fuel or waste will be handled, and b) that the handling method will provide adequate safeguards against harmful effects.

3 The basic premise must be that the high-level waste from reprocessing and the spent nuclear fuel which is not reprocessed are to be kept separated and isolated from all forms of life.

4 Detailed and comprehensive information must be provided for a safety evaluation. Thus, rough plans and sketches are not enough. In addition, it should be concretely specified:

- In what form the waste or the spent nuclear fuel will be stored.
- How the storage site will be arranged.
- How the spent nuclear fuel or waste will be transported.
- Whatever other information is required in order to determine whether the proposed final storage can be deemed to be absolutely safe and practically feasible. The primary consideration here is whether the storage scheme can meet requirements for satisfactory radiation protection.

5 The storage site shall permit the isolation of the waste or the spent nuclear fuel for as long a time as is required for the radioactivity to diminish to a harmless level.

The possible dispersion of the waste or spent nuclear fuel to the biosphere as a result of natural processes, accidents or acts of war shall also be taken into account.

6 It is not necessary that a storage facility is completed when the application for permission is submitted.

1.3 THE KBS PROJECT

1.3.1 Objective

KBS was formed by the following four nuclear power utilities: Statens Vattenfallsverk (The Swedish State Power Board), Oskarshamnverkets Kraftgrupp AB (OKG), Sydkraft AB and Forsmark Kraftgrupp AB (FKA) in order to meet the requirements of the Conditions Act which pertain to the handling and final storage of spent nuclear fuel or high-level waste.

The objective of the KBS Project is:

- to demonstrate how high-level waste or spent fuel can be handled and finally stored,

- to demonstrate where a final storage of high-level waste or spent fuel can be situated, and

- to describe the safety of the proposed arrangements for handling and storage.

1.3.2 Organization

KBS is organized as an independent project within Svensk Kärnbränsleförsörjning AB (SKBF - Swedish Nuclear Fuel Supplies Inc.). The work is being conducted in consultation and collaboration with organizations, corporations and institutions active within the field of radioactive waste handling or other technical fields of importance to the KBS Project.

The KBS Project Board has the following members:

Göran Ekberg, Sydkraft, Chairman
Bo Aler, Atomenergi
Olle Gimstedt, OKG
Lars Halle, Asea-Atom
Jonas V. Norrby, Swedish State Power Board
Erik Svenke, SKBF
Ingvar Wivstad, KBS, Project Director

Of these members, all except Lars Halle are also members of the Board of SKBF.

The Project Management Group is responsible under the Board for the implementation of the project and is made up of the following persons from the power utilities:

Ingvar Wivstad (from the Swedish State Power Board)
Per-Erik Ahlström (from the Swedish State Power Board)
Lars B. Nilsson (from OKG)

A technical committee with an advisory function is subjoined to the Project Management Group. Its members are:

Olle Gimstedt, OKG, Chairman
Tage Arnell, FKA
Lars Halle, Asea-Atom
Yngve Larsson, Sydkraft
Lars Åke Nöjd, Atomenergi
Erik Svenke, SKBF

The KBS organization is illustrated in figure 1:1.

The work has been directed by a central group of some 20 persons consisting of the Project Management Group, programme leaders (for the programme specified in the organization plan, P11 etc.) and staff functions. In addition, some 450 persons were engaged through the contracting of consultants, corporations and research institutions at technical institutes and universities. KBS has also collaborated with organizations in France, the United States and Canada which are active within the same field.

The direction of the work, various alternatives and results were discussed in reference and working groups outside of the organization itself. Through these groups, KBS was able to benefit from the experience of specialists and experts not directly engaged in the KBS project.

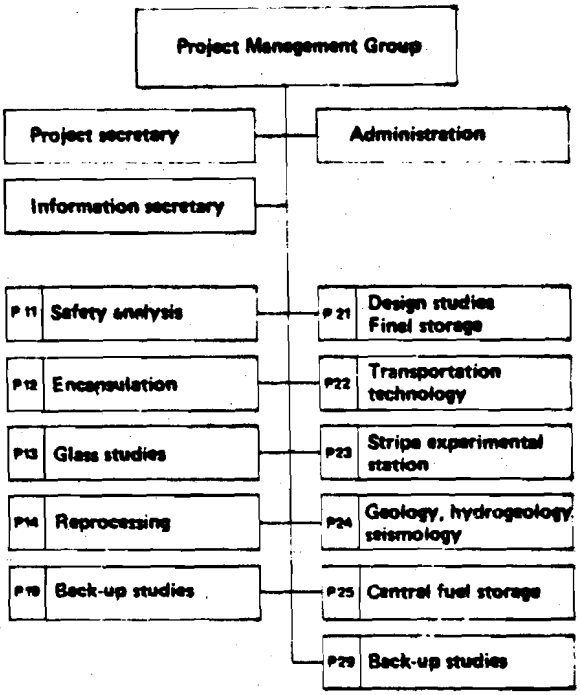


Figure 1-1. Organization plan for the Nuclear Fuel Safety Project (KBS).

The companies, institutions and experts engaged or consulted by the KBS Project are listed in Appendix 2 to this volume.

1.3.3 Premises governing the work of the project

According to agreements entered into with reprocessing companies, specific quantities of spent fuel from four reactor blocks in Sweden - Oskarshamn blocks 1 and 2, Barsebäck 2 and Ringhals 3 - will be reprocessed. Some of the waste from this reprocessing will presumably be returned to Sweden for final disposal.

Reprocessing agreements are currently lacking for other reactor blocks. It has not previously been possible to sign agreements for reprocessing of the fuel discharge from Swedish reactors after 1979. The uncertain international situation and the limited capacity of existing reprocessing facilities (chapter I:5) make it urgent to plan for a final storage of spent nuclear fuel without prior reprocessing as well.

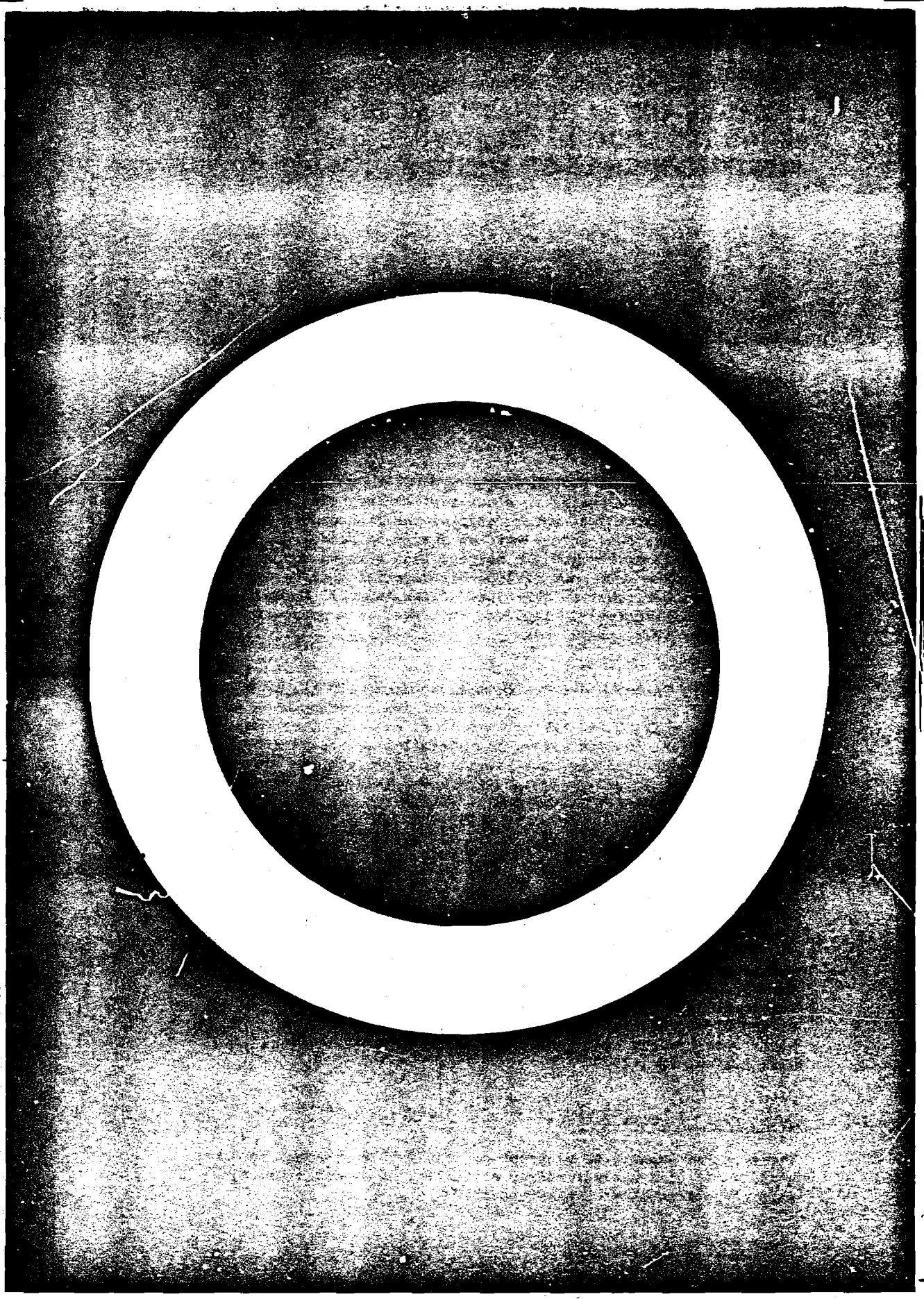
For these reasons, the KBS Project is considering both alternatives in the Conditions Act: The handling and final storage both of vitrified waste from reprocessed spent nuclear fuel and of unreprocessed nuclear fuel.

Development work on the final storage of high-level waste in other nuclear-power-producing countries has thus far been concentrated on vitrified waste from reprocessing. These problems

have been studied primarily by countries with their own reprocessing projects. As a rule, these countries plan to build storage facilities in salt formations, which are considered to be extremely stable and impervious to water penetration. The glass is enclosed in stainless steel containers and then emplaced in direct contact with the salt.

In recent years, however, attention has been turned to the final storage of high-level waste in clays and crystalline rock. The Aka Committee /1-2/ found that Sweden's primary rock formations fulfil the necessary requirements for a safe final storage of radioactive waste. KBS has arrived at the same conclusion and has therefore concentrated its work on final storage in rock.

The present report describes the handling and final storage of vitrified high-level waste from the reprocessing of spent nuclear fuel. A corresponding report on the handling and final storage of non-reprocessed spent nuclear fuel is planned for publication in the spring of 1978. A status report for this alternative is provided in Appendix 1 of this volume.



2 PREMISES AND ALTERNATIVE METHODS FOR MANAGING SPENT NUCLEAR FUEL AND VITRIFIED HIGH-LEVEL WASTE

2.1 DATA FOR SPENT NUCLEAR FUEL

2.1.1 Technical data

In order to generate energy in a nuclear reactor, the reactor is charged with uranium fuel which contains the fissionable isotope uranium 235. This fuel is consumed as energy is produced. Fission products and elements which are heavier than uranium are also formed. After some time, new fissionable material must be supplied and the spent fuel must be taken out of the reactor. Normally, roughly 1/3 of the fuel is replaced each year in a pressurized water reactor (PWR) and 1/5 per year in a boiling water reactor (BWR).

A BWR such as Forsmark 1 produces approximately 220 kWh of electrical power from each gramme of uranium. The corresponding value for a PWR is approx. 260 kWh per gramme of uranium. The composition of the fuel changes during the operation of the reactor. The spent fuel discharged from the reactor consists of:

	BWR	PWR
Uranium-235	0.7 %	0.9 %
Uranium-236	0.4 %	0.4 %
Uranium-238	95.2 %	94.1 %
Fissionable plutonium	0.5 %	0.8 %
Other plutonium	0.2 %	0.3 %
Other transuranic elements	0.05%	0.08%
Fission products	2.9 %	3.4 %

The newly-formed elements are generally unstable and decay to form stable atoms while emitting radiation. The radiation from the spent fuel comes mainly from fission products and diminishes as the elements decay. The content of radioactive elements, their half-lives and the heat generated in the spent fuel are dealt with further in I:13.3.

2.1.2 Quantities of spent fuel

The expected quantities of spent nuclear fuel discharged from the world's civilian nuclear energy production is dependent on the rate of construction of new reactors.

Table 2-1 shows the planned schedule for the construction of light-water reactors (PWRs and BWRs) in some countries up to 1990. These countries are expected to account for more than 85% of the world's total installed electric power generating capacity in the form of light-water reactors in 1980 (Eastern Europe and China not included). The figures are from the Nuclear Assurance Corporation's report of July 1977 /2-1/. The figure of 9.4 GW(e) for Sweden is based on the assumption that 12 reactors will be in operation by 1985.

Table 2-1

Expected power generating capacity of light-water reactors scheduled for construction up to 1990 in GW(e) for each country and year.

Country	1976	1980	1985	1990
Sweden	3.2	7.4	9.4	
Finland	0.4	2.2	3.2	4.2
France	0.3	14.9	39.9	58.1
West Germany	4.0	13.0	25.5	42.0
USA	40.1	76.0	158.0	225.0
Japan	6.9	14.6	30.1	59.0

Table 2-2 gives the quantities of spent nuclear fuel based on the expected construction schedule. Fuel discharge is assumed to be 28 tons of uranium per GW of installed electrical output.

The tabulated years refer to the years in which the reactor is charged with fuel. The fuel is discharged some 2 years later. Less fuel is discharged in the initial period of operation of a reactor, causing deviations from the equilibrium state assumed in the table.

Table 2-2

Quantity of spent fuel in state of equilibrium, based on the figures in table 2-1 (tons of uranium per year).

Country	1976	1980	1985	1990
Sweden	90	210	260	
Finland	11	62	90	120
France	8	420	1100	1600
West Germany	110	360	710	1200
USA	1100	2100	4400	6300
Japan	190	410	840	1700

In 1985, the quantity of spent nuclear fuel in Sweden would comprise approximately 4% of the total quantity of nuclear fuel from light-water reactors in these countries.

2.1.3 Quantities of spent fuel in Sweden

Table 2-3 gives the expected accumulated quantity of spent fuel obtained from the operation of the 13 reactor blocks specified by the 1975 Swedish Parliament as the framework for Sweden's nuclear power plant programme up to 1985. The table also shows the accumulated quantity from the 6 reactor blocks in operation in 1977. The dates assumed for the start-up of the uncommissioned blocks are:

- Ringhals 3 1978
- Forsmark 1 1978
- Ringhals 4 1979
- Forsmark 2 1980
- Forsmark 3 1984
- Oskarshamn 3 1984
- Unit 13 1986

It is assumed that Barsebäck 2 will continue to operate and that the availability factor for all blocks will be 60% during the first three years and 70% thereafter.

Table 2-3

Accumulated quantities of spent fuel in tons of uranium from the operation of 6 or 13 reactors in Sweden.

At year-end	Reactors in operation 1-6	1-13
1977	28	28
1978	120	120
1979	270	280
1980	380	420
1981	470	600
1982	570	790
1983	670	980
1984	770	1200
1985	870	1400
1990	1400	2700
1995	1900	4000

The annual quantities of spent nuclear fuel from the currently operative Swedish reactor blocks are given in table 2-4.

Table 2-4

Annual quantities of spent fuel from operative Swedish reactors in metric tons of uranium. (R = Ringhals, O = Oskarshamn, B = Barsebäck)

Year of Discharge	R1	R2	O1	O2	B1	B2
1977	-	-	13	15	-	-
1978	-	25	15	15	35	-
1979	42	29	15	17	18	32
1980	23	19	12	16	18	18
1981	21	18	12	16	17	16
1982	21	18	12	16	17	16
1983	21	18	12	16	16	16
1984	21	18	12	16	16	16
1985	etc.					

2.2 ALTERNATIVES FOR FUEL MANAGEMENT

2.2.1 General

Energy production in a reactor consumes fissionable material while forming waste products so that some of the fuel must be replaced. The spent fuel from a reactor contains:

- unconsumed uranium from which additional energy can be extracted,
- plutonium formed in the process, which can also be used for further energy production,
- elements formed by nuclear fission (fission products) or by neutron capture in uranium (transuranium elements) and which cannot be utilized for energy production in nuclear reactors. It is isotopes of these elements which are responsible for most of the radiation from the high-level waste.

Before further energy can be obtained from spent nuclear fuel, fission products and transuranium elements must first be separated from the uranium. This process is called reprocessing. After reprocessing, uranium and plutonium can be reused for fuel production while the remainder comprises waste. The high-level waste (which consists of fission products and transuranium elements separated in the extraction cycle in the reprocessing process) is converted to solid form by the addition of vitrifying substances. The vitrified waste must be stored with absolute safety for a very long period of time.

If the spent fuel is not reprocessed, all of the fuel constitutes waste which, following suitable treatment, must be stored. This form of handling of spent nuclear fuel is called direct disposal and also requires storage with absolute safety for a long period of time.

In order to avoid making a commitment to a specific method of handling which may require highly capital-intensive investments and binding agreements, reactor-owners may store the spent fuel for a long periods of time in the expectation that one of the alternatives will display clear advantages over the others.

2.2.2 The reprocessing alternative

After the fuel has been removed from the reactor, it is allowed to cool for a certain period of time in the station's spent fuel pool and may also be stored in a central storage facility for spent nuclear fuel. After this, it is transported to a reprocessing plant, where the fuel is reprocessed after some more years of storage. The fuel rods are chopped into short pieces and treated with acid, which dissolves the fuel. The fuel cladding is not dissolved and is removed. Uranium and plutonium are separated from the other elements by means of extraction with organic solvents.

The recovered uranium can be enriched in a manner similar to natural uranium and then reused as a nuclear fuel.

Plutonium in the form of a mixed oxide can also be used as nuclear fuel, in which case it replaces some of the otherwise necessary quantity of uranium-235. Through this recycling process, the uranium enrichment requirement is reduced by 15-20%. Reusing plutonium and uranium reduces the natural uranium requirement by 30-35%. Plutonium can also be stored for future use as fuel in breeder reactors. The separated high-level waste is stored for several years in liquid form in tanks, after which it is converted to solid form by the addition of vitrifying substances. The glass is then stored for a number of decades in order to allow the rate of heat generation of the waste to drop, after which it is encapsulated for final storage. The basic handling chain is illustrated schematically in figure 2-1.

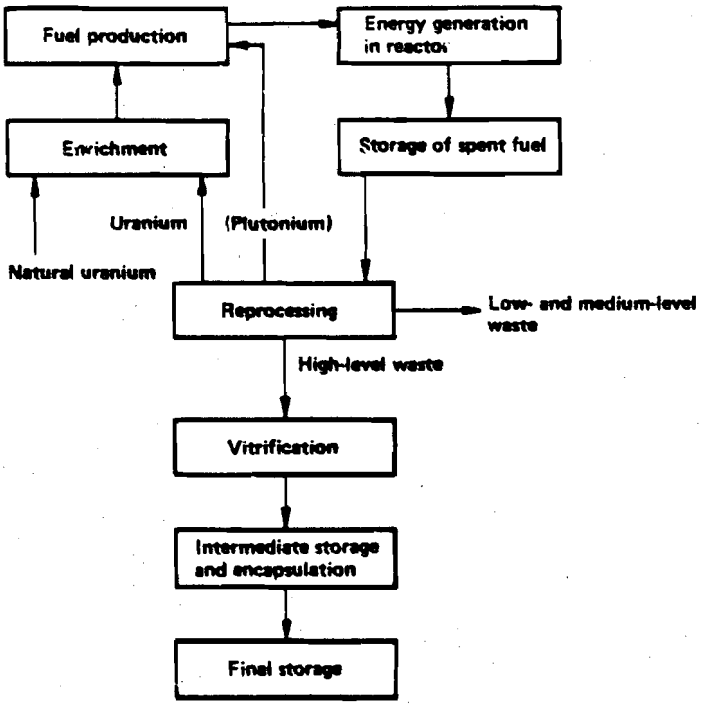


Figure 2-1. The reprocessing alternative. Flow scheme for the fuel cycle with reprocessing of spent fuel and vitrification of the high-level waste.

There are currently 4 reprocessing plants in operation in Western Europe. Two of these, La Hague in France and WAK in West Germany, can reprocess fuel from light-water reactors, while Marcoule in France and Windscale in Great Britain mainly reprocess fuel from gas-graphite reactors. Three new reprocessing plants are currently in the planning and design stage. Total available capacity for the reprocessing of fuel from light-water reactors in the 1980s will not meet the demand. Consequently, additional capacity for the storage of spent fuel is planned.

In the USA, the licensing of privately-owned reprocessing plants for civilian purposes has been postponed indefinitely, as has permission for the recycling of plutonium into the fuel cycle. Reprocessing and solidification are dealt with in greater detail in chapter I:5.

2.2.3 The direct disposal alternative

The risk that plutonium may be stolen for use in terrorist actions or for unauthorized weaponry is cited as an argument against the processing of spent fuel which produces pure plutonium at any stage. It is also feared that a proliferation of reprocessing technology will increase the risk for an accelerated proliferation of nuclear weapons.

The United States has taken the initiative for an international evaluation of the nuclear fuel cycle with regard to the risk for the proliferation of nuclear weapons (International Nuclear Fuel Cycle Evaluation, INFCE). One alternative course of action for the handling of spent nuclear fuel which should reduce this risk and which is also being considered in the United States is to regard the spent fuel as waste, i.e. not to separate and reuse uranium and plutonium (direct disposal).

In this case, the fuel is first stored to allow its radioactivity to decay. Prior to final storage, the fuel is encapsulated in a highly durable material which forms a barrier against the escape of radioactive elements to the environment. The waste is finally deposited in a final repository. The basic handling chain for the direct deposition alternative is illustrated in figure 2-2.

2.3 STORAGE TIMES AND QUANTITIES OF VITRIFIED WASTE

2.3.1 Storage times for spent nuclear fuel

The expected quantities of spent nuclear fuel from Swedish nuclear power blocks were given in 2.1.3.

An agreement for the reprocessing of spent fuel has been concluded between OKG and BNFL in Great Britain with regard to Oskarshamn 1 and 2 and between SKBF and COGEMA in France with regard to Barsebäck 2 and Ringhals 3. These agreements apply to fuel which is discharged during the 1970s.

The power stations have some storage capacity in existing spent fuel pools. This capacity can be expanded by the acquisition of new fuel racks which permit a more compact emplacement of the

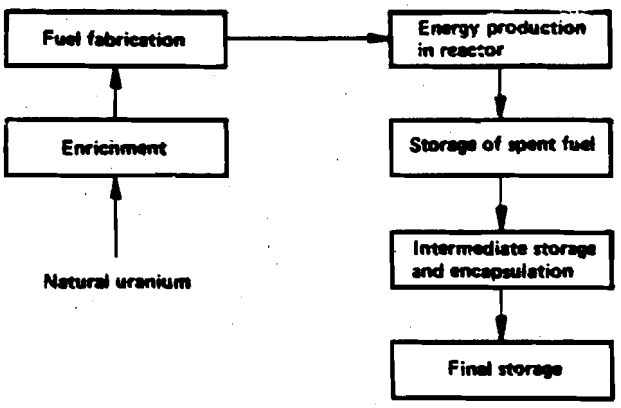


Figure 2-2. The direct deposition alternative. Flow scheme for the fuel cycle with direct storage of the spent fuel without reprocessing.

fuel elements. Table 2-5 gives the earliest dates by which the fuel must be removed from the various units, assuming an expanded storage capacity in the pools and a retained reserve capacity for unloading the complete reactor core.

Table 2-5

Dates for earliest required removal of spent fuel.

Reactor unit	First shipment Year
Oskarshamn 1	1984
Oskarshamn 2	1983
Ringhals 1	1984
Ringhals 2	1983
Ringhals 3	1989
Ringhals 4	1990
Barsebäck 1	1984
Barsebäck 2	1985
Forsmark 1	1987
Forsmark 2	1989

(From: PRAV, Central storage facility for spent fuel; a preliminary study, 1977).

In order to provide additional storage capacity for Swedish spent nuclear fuel pending shipment for either reprocessing or final storage of the un-reprocessed fuel, a central storage facility for spent fuel is required. A preliminary study of such a facility has been carried out by the National Council for Radioactive Waste Management (PRAV). The study was published in July 1977 /2-2/ and is discussed in greater detail in I:4. According to this study, the fuel storage facility should be designed for 3 000 tons of spent fuel.

2.3.2 Quantities of vitrified waste

When the fuel in a reactor is replaced, the spent fuel elements are placed in pools at the station in order to permit short-lived radioactive elements to decay. After storage for at least 6 months and (if needed) storage in a central storage facility for spent fuel, the fuel is transported to a reprocessing plant. The fuel is stored for approximately 1 year in the plant's reception pools prior to reprocessing. The reception pools serve mainly as a buffer store for the reprocessing plant, so the storage time in these pools can vary. During the early 1980s, the shortage of reprocessing capacity may result in long storage times.

Uranium and plutonium are separated and can be reused in the fabrication of new fuel or stored. The separated high-level waste is concentrated and stored in liquid form in tanks equipped with cooling systems. It is converted to solid form by the addition of vitrifying agents, after which the waste glass is cast in steel cylinders.

Under the terms of signed and planned reprocessing agreements with COGEMA, the waste cylinders will be returned to Sweden not earlier than 1990. If 13 reactors are commissioned and all of the fuel is reprocessed, no more than the following quantities of vitrified waste can have been returned to Sweden. The figures are based on 150 litres of vitrified waste per ton of uranium in the spent fuel.

Year	Number of waste cylinders	Quantity of waste in m^3	Corresponding quantity of fuel in tons of uranium
1989	0	0	0
1990	280	42	280
1995	1 200	180	1 200
2000	2 700	400	2 700

3 HANDLING AND STORAGE OF VITRIFIED WASTE FROM SWEDISH REACTORS

3.1 SWEDISH ALTERNATIVES AND COMBINATIONS

3.1.1 General

As was pointed out in the introduction, Swedish work is being concentrated on the final storage of radioactive waste in crystalline rock formations. One of the problems which must be taken into account here is the possibility that the water in the bedrock could eventually penetrate the encapsulation material around the waste and come into contact with the actual vitrified waste itself. The waste is emplaced at a depth of several hundred metres in rock of low permeability where it can safely be assumed that the water will move extremely slowly (see chapter I:7). Ion exchange reactions and other chemical processes ensure that the dispersal of most radioactive substances which are dissolved in the water takes place at a much slower rate than the movement of the water.

Knowledge regarding the movement of water and chemical conditions in rock at depths of several hundred metres was extremely limited when the KBS Project was started. KBS is therefore conducting extensive investigations into these subjects. A data base of limited scope has been assembled in the available time. In evaluating the safety of waste storage, it is therefore necessary to make conservative assumptions with regard to water movement and chemical reactions at great depths. In order to demonstrate today how the vitrified high-level waste can be finally disposed of without risking unacceptable dispersal of radioactive substances, a system involving a number of barriers against such dispersal is proposed.

3.1.2 Proposed alternative

The method for handling spent nuclear fuel which is presented in this report is based on the following principles:

1 Final storage of the waste in precambrian crystalline bedrock.

2 A series of barriers against dispersal of the radioactive substances from the final repository.

3 Flexibility in the handling chain in order to preserve freedom of action and permit the application of further technical developments.

No technical-economical optimization of the facilities, handling methods and final storage method has been carried out.

The following handling chain is proposed in order to ensure a safe final storage of the high-level waste and simultaneously retain high freedom of action and adaptibility to future technical developments:

1 After the spent nuclear fuel has been allowed to cool for at least 6 months at the power stations, it is transported to a reprocessing plant or to a central storage facility for spent fuel.

2 The fuel can be stored at the central fuel storage facility for up to about 10 years in water-filled pools. The design of the central fuel storage facility is described in chapter I:4. From the central fuel storage facility, the fuel is transported to reprocessing.

3 The fuel is reprocessed 2-10 years after it has been taken out of the reactor and the high-level waste from reprocessing is converted to solid form - vitrified. (See chapter I:5.) Vitrification is carried out using the French AVM process, which is now applied on an industrial scale.

4 The product of vitrification is high-activity cylindrical glass bodies enclosed in vessels of stainless steel - waste cylinders. Each cylinder contains the waste from approximately 1 ton of uranium. The cylinders are stored at the reprocessing plant until at least 10 years has passed from the time the fuel was discharged from the reactor. According to current agreements, Sweden has to take back waste from reprocessing in 1990 at the earliest. The properties of the glass are described in greater detail in chapter I:5.

5 From the reprocessing plant, the waste cylinders are shipped to an intermediate storage facility for high-level waste. This is designed as an air-cooled dry storage facility situated in rock with an approximately 30 m thick rock cover (see chapter I:6). The waste can be stored in this manner for a very long period of time. The capacity of the central fuel storage facility and the intermediate storage facility is sufficient to store waste from 13 reactors. A period of 30 years has been chosen for storage in the intermediate storage facility. This ensures plenty of time for optimization of the final storage method. Intermediate storage can also be extended beyond 30 years. Over a 30-year period, radiation and heat flux from the waste declines to about half.

In this study, the intermediate storage is assumed to be located adjacent to the final repository. This is not, however, intended as a necessary restriction of the location of an intermediate storage facility (see chapter I:11).

6 After 30 years of storage in the intermediate store, the waste cylinders are encapsulated in an extremely durable casing. The casing is made of titanium and is 6 mm thick. In order to reduce radiation through the titanium casing and thereby the radiolytic disintegration of the groundwater in the rock, a 10 cm thick layer of lead is inserted between the stainless steel cylinder surrounding the glass and the titanium casing. Lead also possesses excellent durability. The entire canister is shown in figure 3-1 and described in greater detail in chapter I:6. The total weight of the waste cylinder and the casing is approximately 3.9 metric tons. The external dimensions of the canister are approximately 0.6 m diameter and 1.8 m length.

7 The encapsulated waste is then taken to a final repository approximately 500 m down in the bedrock. The repository is designed as a system of tunnels approximately 3.5 m wide and high and spaced at approximately 25 m intervals. Storage holes approximately 1 m in diameter and 5 m deep are drilled

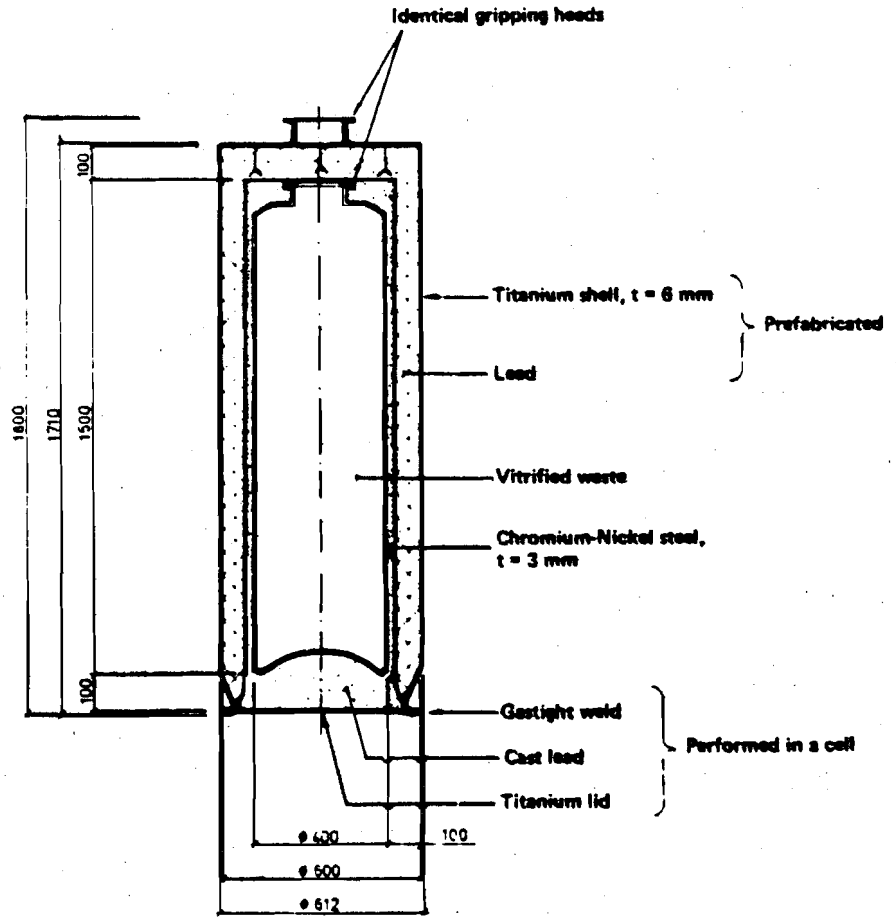


Figure 3-1. Lead-titanium canister. Waste cylinder with vitrified high-level waste enclosed in a canister of lead and titanium. Total weight approx. 3900 kg.

in the floors of the tunnels. One waste canister is stored in each hole. The centre-to-centre distance between the holes is approximately 4 m. The layout of the tunnel system with holes is illustrated in figures 3-2 and 3-3. A buffer mass consisting of a mixture of quartz sand and bentonite is packed around the waste canister. Bentonite is a clay which swells when it absorbs water. The primary purpose of the buffer mass is to fix the canister and to serve as a mechanical barrier. The material has been chosen for its mechanical stability and high durability. It also possesses low water permeability and an ion-exchanging capacity for many of the radioactive elements in the waste. The final repository is described in chapter I:8.

8 Backfill of the storage holes with buffer mass takes place immediately after deposition. Overlying tunnel systems can be kept open and ventilated as long as deposition is proceeding in the facility. During this time, retrieval of the deposited waste is in principle a simple matter. Such retrieval has not been studied more closely since it is better to extend storage in the intermediate store in case of doubt with regard to starting final storage. Such doubt may stem from current technical developments in the field of alternative uses for the waste products or a desire to await practical experiences from foreign facilities for final storage.

9 After all bore holes in the entire tunnel system have been filled with canisters, the tunnels are filled with a mixture of quartz sand and bentonite similar to that used in the

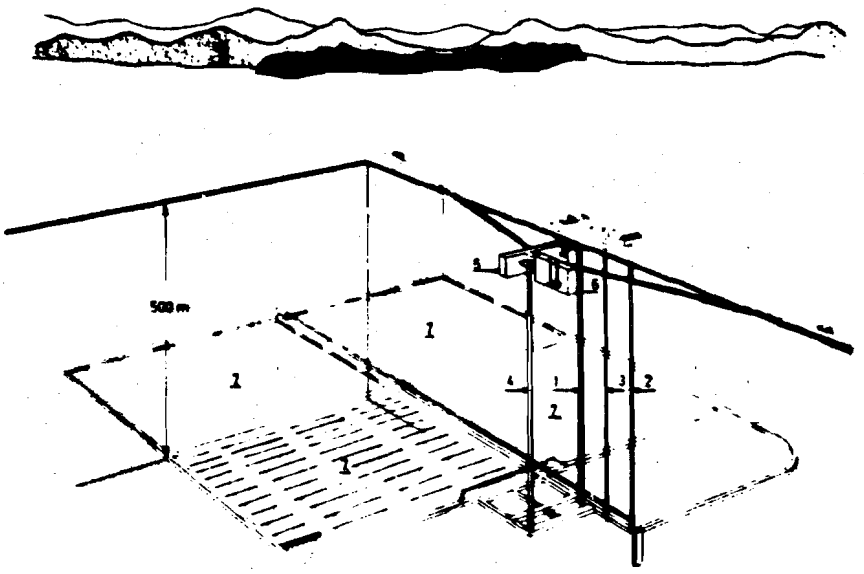


Figure 3-2. Perspective drawing of final repository with plant for intermediate storage and encapsulation. The final repository consists of a system of parallel storage tunnels situated 500 m below the surface.

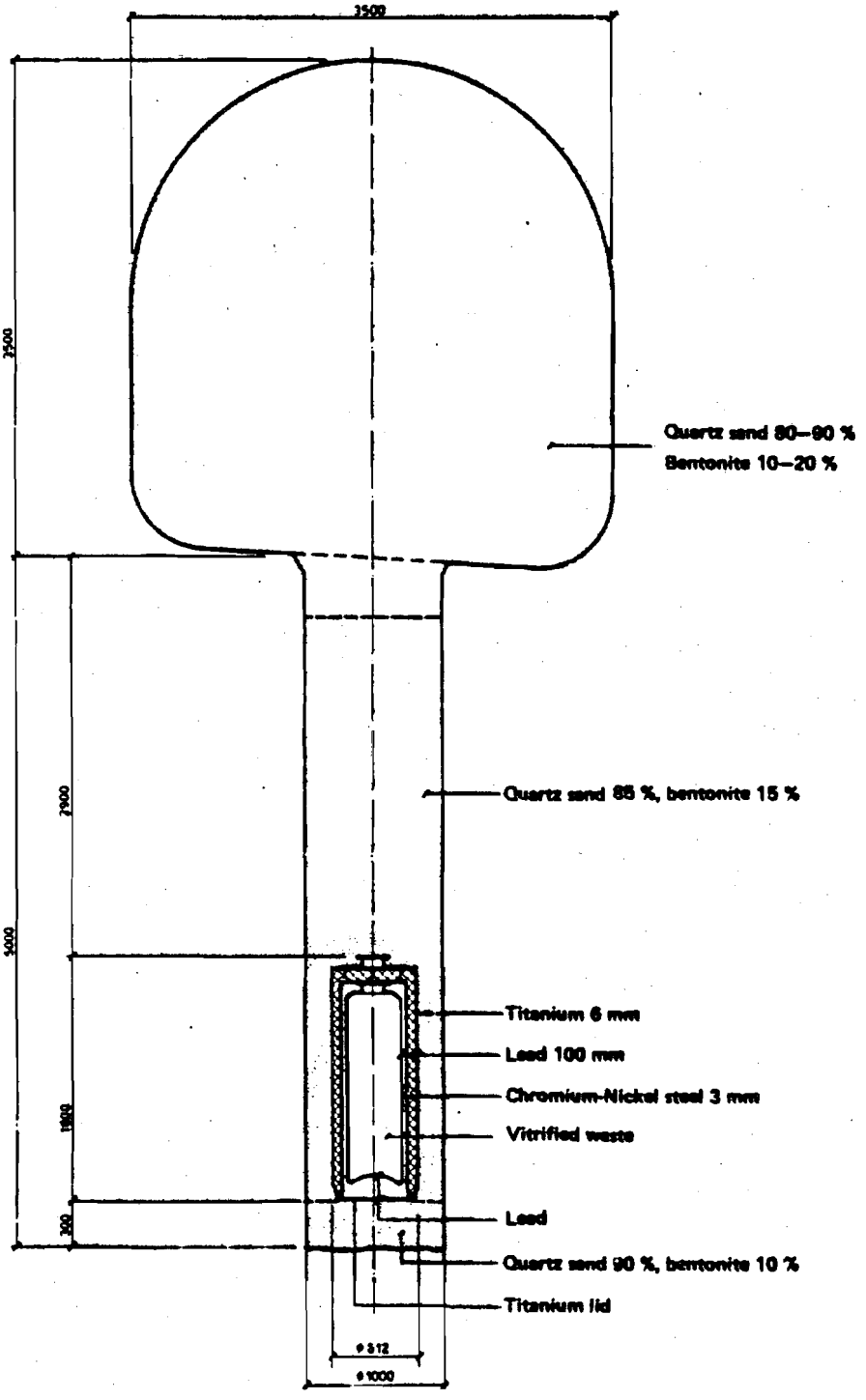


Figure 3-3. The sealed final repository. Tunnels and storage holes are completely filled with a buffer material consisting of quartz sand and bentonite.

storage holes. Access tunnels and shafts are filled in a similar manner.

10 Spent nuclear fuel can be transported using techniques which are already in use and have already proved their worth. With some slight modification, the same transport casks can be used for the transportation of vitrified high-level waste. The safety aspects of transporting high-level wastes are regulated by IAEA regulations. Transportation is dealt with in chapter I:9.

The handling chain is illustrated by figure 3-4. The dates and quantities given in the figure merely illustrate the interrelation of different types of storage facilities for a nuclear power programme of the scope outlined in the 1975 parliamentary resolution, i.e. with 13 light-water reactors in operation by 1985. If these reactors are operated for 30 years and if all the spent fuel is reprocessed, a total of approximately 9 000 waste canisters will be obtained and will have to be disposed of. A change in the assumed scope of nuclear power production in Sweden would require modification of the quantities specified in the figure. But the time schedule for the implementation of the various phases would only be altered slightly.

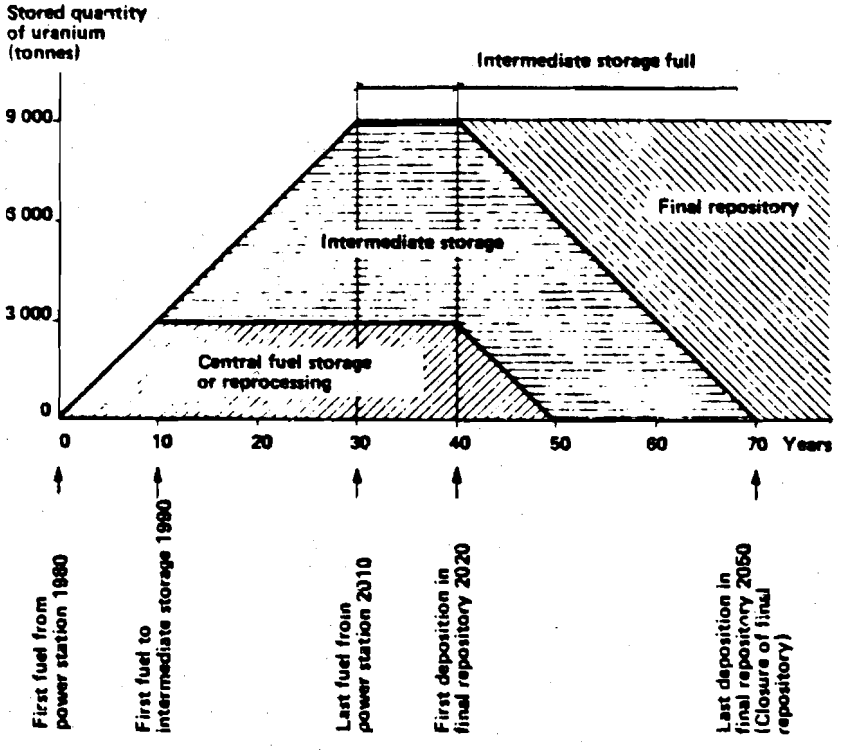


Figure 3-4. Diagram showing capacity requirements for the central fuel storage facility and final repository at different points in time.

3.1.3 Barriers against dispersal of radioactive substances

The proposed handling chain ensures the safe handling and final storage of the high-level waste. This is shown in greater detail in chapter I:13. Dispersion of the radioactive substances from the final repository is prevented or retarded by the following means:

- The radioactive elements are bound chemically in a glass which possesses high resistance to dissolution in water under the conditions prevailing in the final repository.

- The high-level waste glass is enclosed in a canister consisting of three successive metallic layers:

  - 3-4 mm stainless steel
  - 100 mm lead
  - 6 mm titanium

  Under the conditions prevailing in the repository, both titanium and lead possess outstanding resistance to penetration, so it is unlikely that water will come into contact with the actual waste glass for many millenia after deposition (see chapter I:6).

- The buffer material which surrounds the waste canister possesses good stability and very low water permeability. The circulation of water around the canister will therefore be roughly the same as in the surrounding rock.

- The buffer mass and the rock have an ion-exchanging capacity so that many radioactive substances would, if they were dissolved in the groundwater, be dispersed much more slowly than the rate of flow of the water.

- The rock formation chosen for the location of the storage tunnels must be selected with care. Groundwater movements must be small and have such a direction that it takes a long time for the water to flow from the final repository into areas in contact with the ecological system. The geological surveys carried out by the Geological Survey of Sweden (SGU) for KBS have shown that primary rock formations possessing the desired characteristics exist in Sweden (see chapter I:7).

The Finnsjö region 16 km west-southwest of the Forsmark nuclear power station has been used in this report to provide certain studies with a geographical point of reference. This does not mean that this region is actually being proposed for the site of a future final repository.

3.2 FLEXIBILITY AND DEVELOPMENT POSSIBILITIES

The handling chain proposed here entails considerable flexibility with regard to future options and technical development.

A central fuel store with the proposed capacity will permit considerable flexibility with regard to the quantities of spent nuclear fuel which may be scheduled for reprocessing over the

next 10 years. In the present situation, with a shortage of reprocessing capacity and a monopoly situation in this area, such flexibility is essential. The current debate regarding the future use of plutonium also makes a flexible strategy desirable.

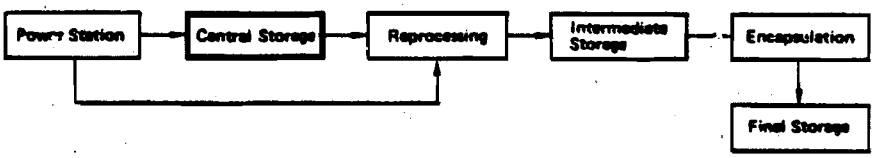
One advantage of being able to postpone a final decision on the question of large-scale reprocessing is the fact that new methods for converting the waste into solid form may be developed. These new methods may make final storage simpler and less expensive.

An intermediate storage facility for high-level waste provides extra time for a technical-economical optimization of the final storage method with unimpaired high safety. The time for final storage can be chosen when the results of the development work which is currently in process in various countries are available. The final storage site can be selected on the basis of thorough investigations of all the ecological, technical, economic and social factors which are of importance. The suitability of the site can be verified and demonstrated to the public by means of long-term tests. Studies concerning methods for utilizing the waste products can also be conducted.

The canister may be made of other materials than titanium and lead. The materials discussed in the status report on direct disposal (Appendix 1), i.e. copper or aluminium oxide (corundum, Al_2O_3), may be suitable for use in the encapsulation of vitrified waste.

It is hoped that future hydrogeological studies will reduce current uncertainty with regard to water movements etc., whereby simpler encapsulations may prove to be satisfactory. The tunnel system and the emplacement of the waste canister have been chosen on the basis of conservative considerations. Further data on the properties of the rock will probably permit a more closely packed storage of canisters as well as multi-level storage. Studies of the design of the final repository in other respects may also prove necessary in order to find simpler and cheaper solutions.

4 CENTRAL STORAGE FACILITY FOR SPENT FUEL



4.1 NEED FOR CENTRAL STORAGE FACILITY FOR SPENT FUEL

All nuclear power plants have storage pools for spent nuclear fuel. These pools are needed so that the fuel in the whole reactor core can be taken out if necessary and so that spent nuclear fuel can be stored before it is sent on to reprocessing or further storage elsewhere. The spent fuel pools must be installed close to the reactor. Space here is limited and the pools normally only have room for at most 2 or 3 years spent fuel above and beyond space for the temporary removal of the reactor core fuel. This capacity is fully adequate, provided that there is good access to reprocessing capacity or storage space. This has not been the case in the past. Sweden, along with other countries with nuclear power, must therefore expand its storage capacity for spent nuclear fuel. Additional storage capacity can be created in three ways:

- The storage capacity of existing pools can be expanded, although only to a limited extent.
- New pools can be built at every nuclear power plant.
- A central fuel storage facility serving a number of nuclear power plants can be built.

The latter alternative has been chosen as the best means of ensuring sufficient storage capacity in the long run. Compared with the total cost of storage facilities at each nuclear power plant, a central storage facility is considerably cheaper. Installations which depend not at all or only slightly on the size of the storage facility need only be built at one site. This applies, for example, to a receiving station for transport casks and to most of the auxiliary systems, both of which comprise sizable cost items. On the other hand, the transportation equipment required for a central storage facility is more expensive. However, the additional cost for transportation equipment is only approximately 10% of the total cost for a central fuel storage facility, so a saving on this item would not compensate for higher costs in the other areas.

Obviously, the amount of storage capacity which is required depends on how much fuel is to be sent for reprocessing and when this can be done. Regular shipments of fuel for reprocessing will hardly be possible before the available reprocessing capacity in Europe has been expanded so that it is in equilibrium with the annual production of spent nuclear fuel. This will probably occur no sooner than the late 1980s. Transports may also be delayed by

the fact that other countries have considerable quantities of fuel awaiting reprocessing. Thus, it is important from the viewpoint of both preparedness and negotiating strength that the central spent fuel storage facility be large enough to meet Swedish needs through the early 1990s. This corresponds to a storage capacity of 3 000 metric tons of uranium. It is assumed that the facility will be located near a harbour.

4.2 DESIGN OF FACILITY

4.2.1 Description of facility

When PRAV (the National Council for Radioactive Waste Management) was formed at the suggestion of the Aka Committee its work schedule included a preliminary study of a central storage facility for spent nuclear fuel. This preliminary study was carried out under the guidance of PRAV during 1977 by personnel from SKBF and the Swedish nuclear power utilities. The description given here is based on PRAV's preliminary study. PRAV has now handed the material over to SKBF, who are responsible for continued work on the planning and design of the fuel storage facility and application for siting approval. Certain parts of the facility may be subject to modification in this connection.

Most of the facility will be situated underground with a rock cover approximately 30 metres thick to provide protection against external forces such as acts of wars and sabotage.

The receiving and storage section is situated in a rock cavern approximately 280 m long and 20 m wide. Its height varies between 25 and 35 m. A smaller rock cavern is built parallel to this one to accommodate auxiliary systems, mainly electrical systems. A transept containing the plant control centre and communications and changing rooms will connect the two caverns. The subsurface facility will be connected to the surface building by means of a vertical shaft. Besides personnel transports, the vertical shaft will also be used to carry ventilation ducts, cables and pipes. Heavy transports will take place via a descent tunnel.

The surface building will accomodate offices, personnel quarters, auxiliary power units, ventilation fans and a seawater cooling system.

The construction and function of the facility is described in greater detail in chapter III:3.

In terms of function, the facility can be divided into a receiving section, a storage section and an auxiliary systems section (see figure 4-1).

The receiving section contains an area for transport vehicles, where the arriving transport cask is inspected externally, after which it is lifted off the vehicle and placed in a holding pen. The cask is then cleaned externally, after which it is connected to a water loop for internal cooling and cleaning. The transport cask is then transferred to a discharge pen (reception pool), where the cask is opened and the fuel elements are lifted out one

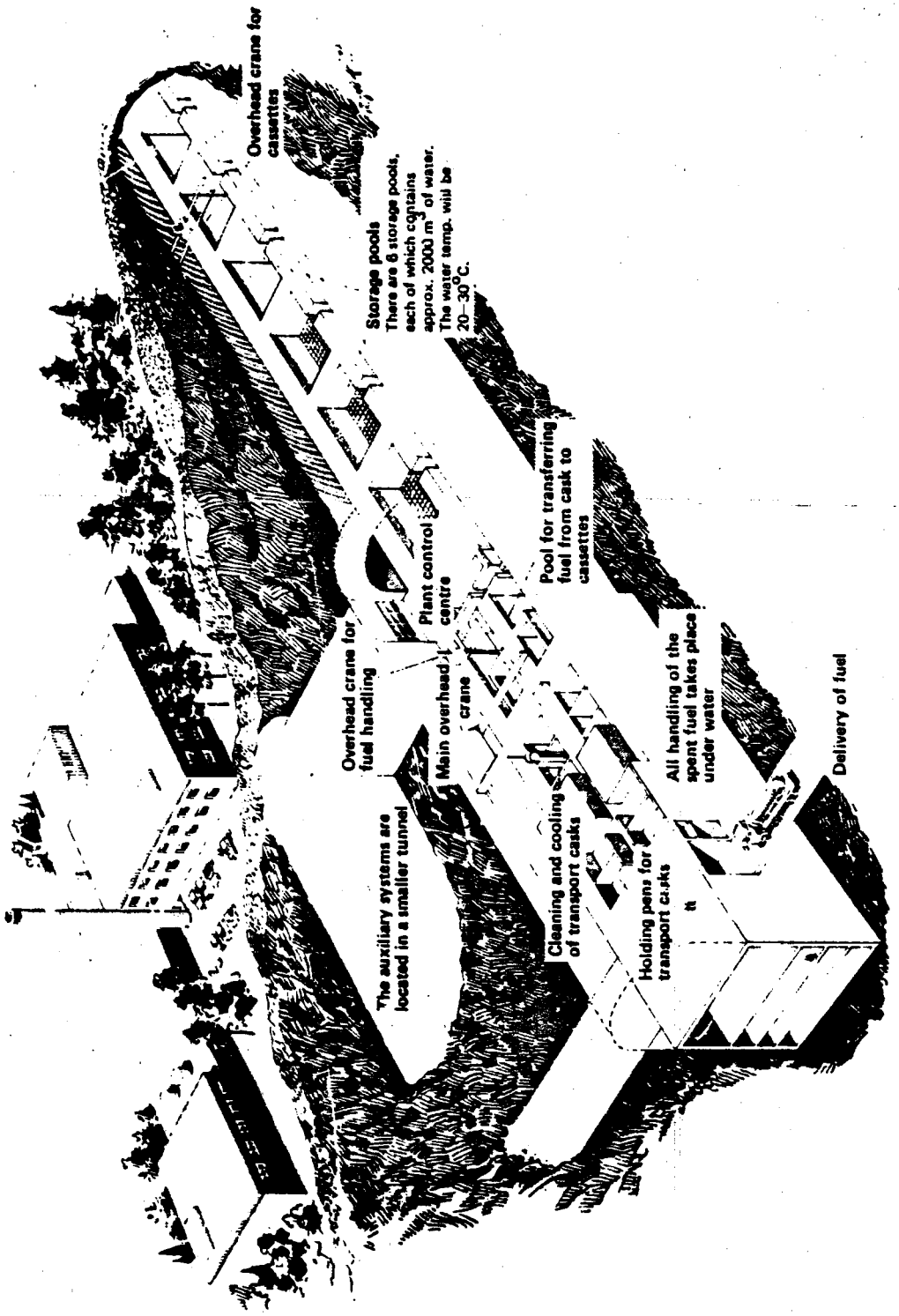


Figure 4-1. Perspective drawing of the central storage facility for spent fuel.

by one by a handling machine. The elements are then placed in special cassettes.

The receiving section also houses areas and equipment for changing and cleaning the transport cask linings.

The cassettes with the fuel elements are transferred to the storage section in a water-filled conveyor channel by means of a handling machine.

The storage section consists of 6 water-filled pools with a water depth of 12 m.

The auxiliary systems section contains cooling and cleaning systems for the receiving and storage sections and systems for handling the radioactive waste. Electrical power systems and monitoring and ventilation equipment are located separate from the radioactive systems.

4.2.2 Reasons for storage in rock

The facility is located in rock for environmental and safety reasons. A rock cavern with a 30 m rock cover provides good protection against damage due to acts of war and sabotage. Since there is good bedrock at the sites which are being considered for a central fuel store, the cost difference between a rock-enclosed facility and a surface facility with a corresponding level of protection is small. Since the buildings are large, a surface installation would have considerable impact on the landscape profile.

The above-stated reasons for locating the facility in rock apply especially to the storage section, while the receiving section could be located on the surface. Such a solution is being studied in connection with the planning of a central storage facility.

4.2.3 Expandability

The design concept of the facility with the storage pools arranged one after the other as separate units provides ample opportunity for expansion in stages. The rock caverns are blasted and the auxiliary systems are designed in such a manner as to facilitate future expansion.

Should the need arise for additional storage capacity during the 1990s, a similar facility can be built adjacent to the one which is now being planned. Potential for future expansion is being taken into consideration in the evaluation of alternative sites.

4.2.4 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 time. Continous maintenance and renovation of machinery and equipment can prolong this life.

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.
- Active components other than fuel are removed to final deposition.
- The facility is thoroughly decontaminated. Scrap and building components which constitute low- and medium-level 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 decontaminating can be reduced.

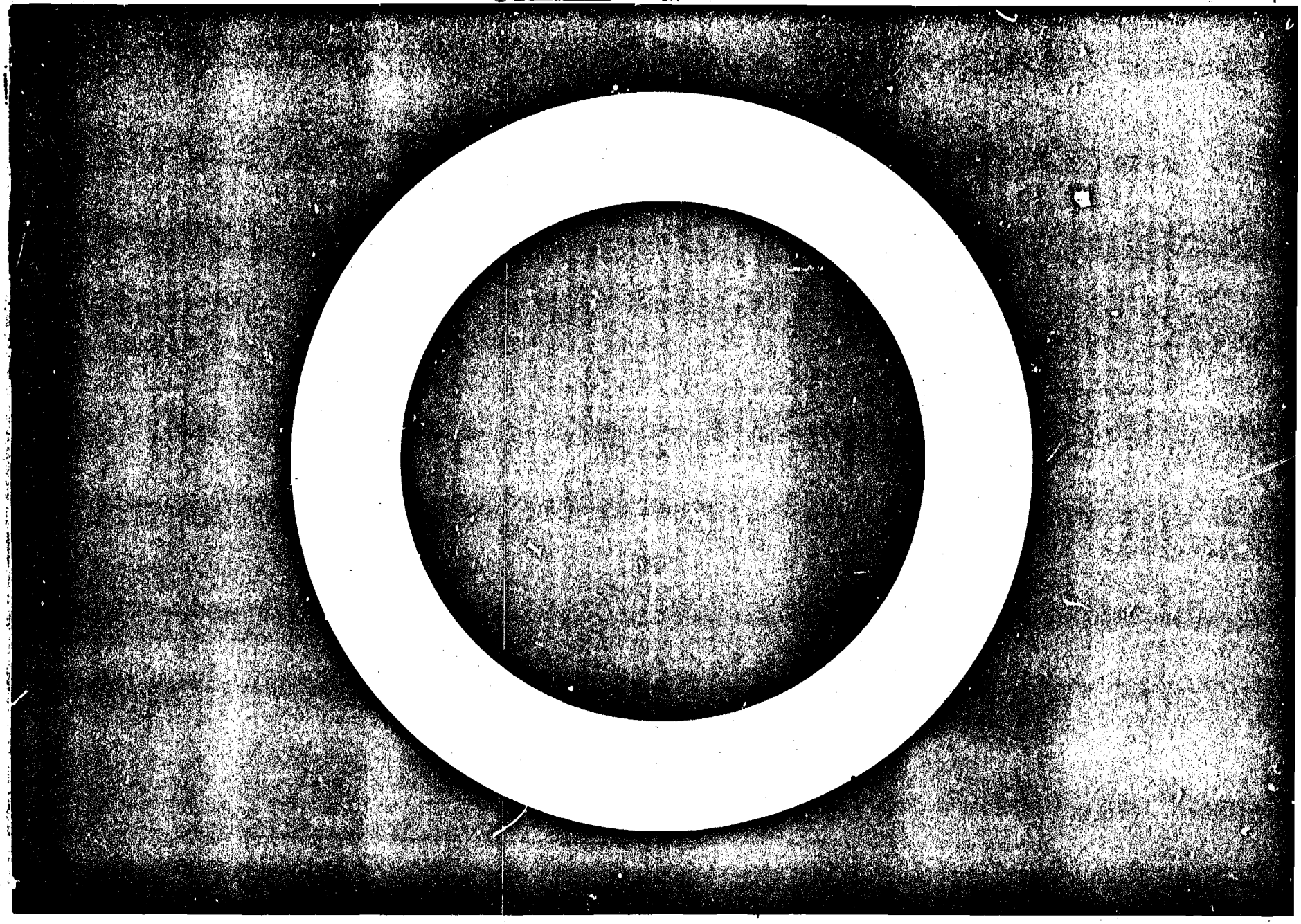
The decommissioning of a central fuel storage facility poses fewer problems than the decommissioning of a nuclear power plant. This is primarily due to the fact that the central storage facility does not contain heavy equipment or permanent installations which are highly radioactive.

4.3 OPERATION OF FACILITY

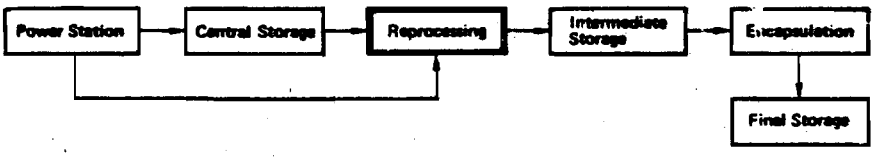
A central store for spent nuclear fuel will be under the supervision of the same authorities as a nuclear power plant, namely the National Nuclear Power Inspectorate, the National Institute of Radiation Protection etc. These authorities issue directives and regulations governing both the design and the 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 with a special emphasis on fuel handling at operative nuclear power plants.



5 REPROCESSING AND SOLIDIFICATION



5.1 INTERNATIONAL REVIEW

5.1.1 Current situation for reprocessing in Europe

There are currently four reprocessing plants in operation in Western Europe, and design work has been started on an additional three.

The first French reprocessing plant, UP1 in Marcoule, was built to reprocess fuel from the gas-cooled graphite-moderated reactors in Marcoule. It was completed in 1958 and has been in operation since that time. Marcoule will now gradually take over the reprocessing of fuel from other French reactors of this type as well, which was formerly done in La Hague. Marcoule also reprocesses fuel from a French heavy-water reactor and some fuel from the Phenix breeder reactor. The reprocessing capacity of the plant at Marcoule is approximately 1 000 metric tons per year. The PIVER plant in Marcoule has been batch-producing high-level glass since 1969. The high-level glass has been cast in containers made of chromium-nickel steel. At present, vitrified waste with an activity of around 5 million curies is stored in a subsurface air-cooled concrete storage facility in Marcoule.

The other French reprocessing plant, UP2 in La Hague, started routine operation in 1967. From the start, the plant was intended for the reprocessing of fuel for the gas-graphite reactors. In 1971, the construction of a "head end" for the reception, chopping and dissolution of light-water reactor fuel was commenced at La Hague. The plant for the separation of uranium, plutonium and high-level waste is the same for gas-graphite fuel and light-water reactor fuel. The plant is currently operated alternately with either one or the other type of spent fuel. Trial operation of this section began in 1976, when 15 tons of fuel from the Swiss boiling water reactor in Mühleberg were reprocessed. The next operating period with light-water reactor fuel will begin at the end of 1977. The strike at La Hague in the autumn of 1976 has delayed the operating schedule.

In November of 1976, the French government-owned nuclear fuel company COGEMA and its personnel organizations appointed an expanded committee for hygiene and safety with directives to propose improvements in the working environment. In June of 1977, this committee published a report with proposals for improvements in the working environment covering 47 points which were to be

implemented by 1981. COGEMA has decided to implement this programme in its entirety. Some of the proposals have already been put into effect.

The capacity of the plant at La Hague is 1 000 metric tons of gas-graphite reactor fuel per year. Starting i 1978, the plant will gradually shift emphasis to the reprocessing of light-water reactor fuel. The capacity of the plant for such fuel is 400 tons of uranium per year. This capacity will be increased to 800 tons per year as of 1981 by means of supplementary installations.

The British reprocessing plant in Windscale, which is owned by the National British Nuclear Fuels Ltd. (BNFL) was commissioned in 1964. It was built for the reprocessing of gas-graphite reactor fuel, but was then modified to permit the reprocessing of light-water reactor fuel as well. Some light-water reactor fuel was reprocessed in the early 1970s, but after an incident in 1973, the rebuilt section was closed down.

The plant in Windscale now reprocesses fuel from the British gas-graphite reactors. Its annual capacity is about 1 000 tons per year. The spent light-water reactor fuel is currently stored in water pools for later reprocessing. After the rebuilding of a "head end", the reprocessing of light-water fuel will be resumed.

The first industrial demonstration of reprocessing of ligth-water reactor fuel took place under the auspices of the joint Western European project Eurochemic, which constructed and operated a small installation at Mol in Belgium. The plant was run for 7 years and reprocessed, among other things 190 metric tons of light-water reactor fuel. After running-in and debugging, the plant operated satisfactorily. But experience showed that the process would have to be modified for a larger-scale industrial plant.

Operation of the reprocessing plant in Mol was discontinued in January of 1976. The decision to shut down the plant was made a couple of years earlier by its Western European proprietors, who foresaw a reprocessing overcapacity in Western Europe, in which case it would no longer be economical to operate a small-scale plant such as Eurochemic. The situation has now changed and the Belgian utilities are investigating the possibility of resuming operation of the plant for Belgian needs.

Another demonstration of the reprocessing of light-water reactor fuel is in progress at the West German reprocessing plant of WAK near Karlsruhe. This plant, which has an annual capacity of 40 metric tons, was commissioned in the early 1970s. After a couple of years with various operational problems, it has run smoothly over the past few years and is now reprocessing light-water reactor fuel.

Three new reprocessing plants are currently being planned in Western Europe. All will be designed especially for light-water reactor fuel. COGEMA will build a new plant in La Hague called UP3A with an annual capacity of 800 tons and a scheduled starting date in 1984/85. Plans call for the commissioning of a similar plant with the same capacity, UP3B, a couple of years later.

BNFL is planning a new plant in Windscale called THORP 1 with a

capacity of 1 000 tons per year. It is scheduled to start operation in 1985 at the earliest. The British Minister of the Environment decided in December 1976 that a public hearing should be held regarding this plant, so it is uncertain at the present time when and if the plant will be built.

Both COGEMA and BNFL plan to utilize their plants to meet both their own countries' needs as well as the needs of other countries, primarily in Western Europe and Japan. Plans call for the financing of the new plants by having domestic and foreign customers sign long-term reprocessing contracts involving advance payments.

The power utilities in West Germany have formed a joint company, DWK, which plans to build a large reprocessing plant. Discussions regarding the location of the plant are currently being held. According to present plans, such a plant would be commissioned in 1988.

5.1.2 Current situation for reprocessing in other countries

The Soviet Union is expected to start reprocessing light-water reactor fuel on a large scale during the early 1980s. It is estimated that the Soviet capacity will cover Eastern European needs.

Japan has constructed a reprocessing plant with French technology in Tokai-Mura with an annual capacity of 200 tons. In July of 1977, the plant started accepting spent nuclear fuel. An agreement has now been reached between the United States and Japan sanctioning Japanese plans to commence the reprocessing of spent fuel in the near future.

In the USA, where reprocessing technology was developed during the 1940s, federal reprocessing plants are in operation in Hanford and Savannah River. Today, these plants are reprocessing both military fuel and fuel from research reactors.

The reprocessing and recycling of plutonium from civilian nuclear power reactors in the USA will be postponed indefinitely, according to President Carter's Energy Plan.

5.2 SWEDISH REPROCESSING CONTRACTS

OKG has a contract with BNFL for the reprocessing of spent nuclear fuel from the Oskarshamn 1 and 2 reactors in England which will cover the needs of these reactors up until 1980. SKBF has signed a contract with COGEMA for the reprocessing of spent nuclear fuel from the Barsebäck 2 and Ringhals 3 reactors, which will also cover the discharged fuel needs of these reactors up to 1980.

For the present, pending resolution of the situation in the UK, SKBF is negotiating only with COGEMA regarding the reprocessing of nuclear fuel during the 1980s. Most of the fuel would be reprocessed in the planned UP3A plant.

The description of the properties of the waste glass provided in this chapter and in chapter III:4 is based on studies and infor-

mation from COGEMA. Some of the claims have been supported by Swedish verification studies of the waste glass from COGEMA.

SKBF's reprocessing agreement specifies a lower content of fission products in the vitrified waste than current French practice. This has been done to reduce the heat generation of the waste, thereby facilitating handling and storage of the material. Properties which stem from this change are easy to calculate.

According to the reprocessing agreement, SKBF and COGEMA will reach a formal agreement some time in the early 1980s on contract specification for the waste glass which are scheduled to be returned to Sweden no earlier than 1990. The product characteristics which will be specified at this time will not deviate substantially from the properties specified in this report. In any case, any deviations will be insignificant for an evaluation as to whether the high-level waste in the form of glass can be handled and stored in an absolutely safe manner.

5.3 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 short, the basic 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. Figure 5-1 shows a simplified flow scheme of the reprocessing of spent nuclear fuel from light-water reactors.

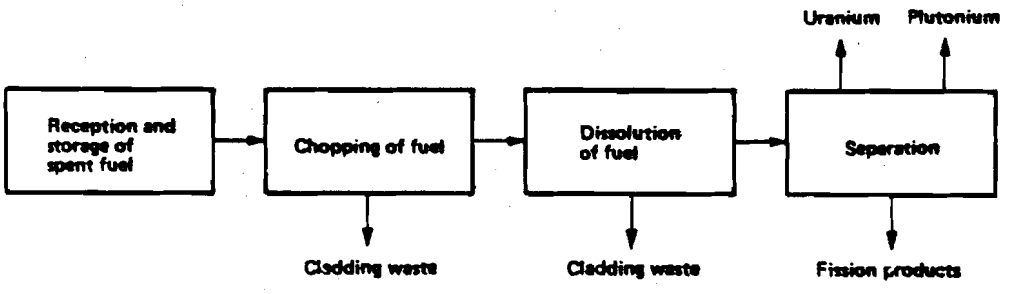


Figure 5-1. Flow scheme of the various stages of the reprocessing of spent nuclear fuel from light-water reactors.

The high-level waste solution contains an estimated 0.1% of the original quantity of uranium and 0.5% of the original quantity of plutonium plus the total quantity of fission products and other transuranium elements. The solution is evaporated in cooled and monitored stainless steel tanks. After a period of storage, the liquid high-level waste is converted to solid form. The reprocessing plant at La Hague will first convert the high-level waste to a calcinate. The calcinate will then be melted down together with borosilicate glass and cast in containers made of chromium-nickel steel. When a container is filled with waste glass, it will be hermetically sealed by the welding-on of a lid. The outside of the container (waste cylinder) will be decontaminated by rinsing with water under high pressure, thereby simplifying the subsequent handling of the waste cylinder. The cylinder will then be transported to a cooled store at the reprocessing plant.

The high-level glass will be fabricated by remote control in "hot cells" with thick concrete walls and with lead glass windows through which the process can be followed. Experience of non-continuous glass manufacture in the PIVER plant has been gained in Marcoule. A larger industrial plant for glass production by means of a continuous process, AVM (Atelier de Vitrification de Marcoule), is completed and is currently being tested with inactive glass. Production of high-level glass from Marcoule will commence in early 1978. The design of a similar vitrification facility for the reprocessing plant at La Hague has been commenced.

5.4 PROPERTIES OF VITRIFIED WASTE

5.4.1 Dimensions and radioactivity content

The vitrified waste arrives in Sweden in cylinders made of chromium-nickel steel. Each cylinder is 400 mm in diameter, 1 500 mm in length and contains 150 litres of glass (see figure 5-2). The density of the glass is 2.8 g/cm^3. Each cylinder weighs 470 kg and contains high-level waste from the reprocessing of 1 ton of uranium in spent nuclear fuel. This means that the fission products content of the glass is reduced from 20% to approx. 9% by weight, as prescribed in SKBF's reprocessing contract with COGEMA. Heat generation from each cylinder is thereby reduced to values which comply with KBS proposals for the handling and storage of vitrified waste.

According to the current contract, the waste cylinders will be returned to Sweden no earlier than 10 years after the fuel was taken out of the reactor, at which point the total radioactivity per cylinder is 4.10^5 curies. 30 years later, the amount of radioactivity will have dropped to 2.10^5 curies. Heat flux per cylinder is then a maximum of 525 W.

5.4.2 Fabrication of borosilicate glass

Researchers at the French Atomic Energy Commission initiated laboratory trials for the production of waste glass back in 1957. A pilot plant for the production of waste glass called PIVER was commissioned in 1969. This plant produces high-level glass from the reprocessing of gas-graphite fuel in Marcoule. Since 1969,

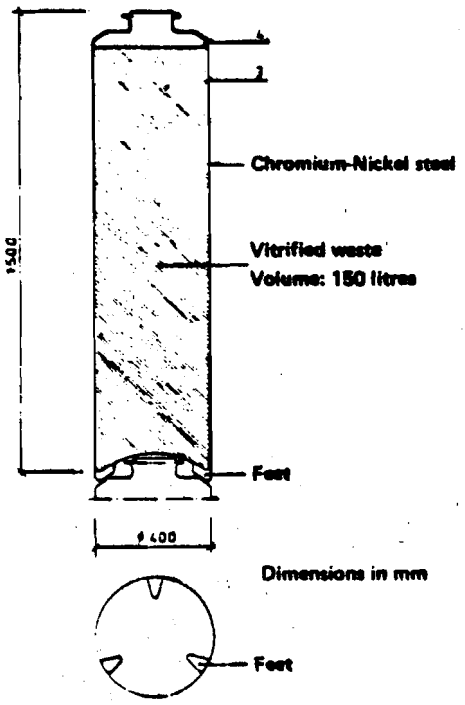


Figure 5-2. Waste cylinder. The vitrified waste is cast in a container made of chromium-nickel steel. The container is sealed with a welded-on lid. The feet enable the waste cylinders to be stacked on top of each other.

high-level waste with a radioactivity of 5 million curies has been vitrified in the PIVER plant, resulting in the production of 12 tons of glass. This glass is now being stored in a dry, air-cooled depot in Marcoule.

But the capacity of the PIVER facility is too small for the industrial production of high-level glass. A new plant has been built for continuous glass production on a larger scale in Marcoule, where operation with active material is expected to start in early 1978.

From the beginning, the French researchers concentrated their interest on borosilicate glass. The atoms in this glass are not arranged in an orderly fashion as in crystalline substances. Only atoms of a certain size can be incorporated in a crystalline structure. Glass, on the other hand, is able to dissolve the different atoms of varying size which occur in high-level waste. In addition, the structure of glass can adjust to the radioactive disintegration which takes place in the fission products and actinides and which results in their conversion to new elements.

Borosilicate glass is composed of silicon dioxide, sodium oxide and boron oxide. In the production of French high-level glass, a prefabricated inactive borosilicate glass is mixed with calcined high-level waste from reprocessing. The advantages of borosilicate glass are:

- good chemical resistance to leaching in water,
- good mechanical resistance to rapid temperature fluctuations,
- low crystallization rate,
- little increase of leaching rate if the glass crystallizes and
- good resistance to radiation damage.

5.4.3 Leaching from borosilicate glass

If groundwater comes into direct contact with glass, a very slow leaching of ions from the glass to the water takes place. At Marcoule, the leaching rate for borosilicate glass containing 20% fission products from ligth-water reactor fuel has been determined in flowing water (dynamic leaching). The leaching rate at 25°C has been calculated to be 2.10^{-7} g/cm^2 per day, which corresponds to a dissolution of 0.003 mm of glass thickness per year, or approx. 1 mm every 3 000 years. This applies for leaching with flowing water. In the case of stationary water (static leaching), which corresponds more closely to the conditions prevailing in a final repository, researchers in Marcoule have measured leaching rates which are lower than for dynamic leaching. Borosilicate glass containing only 9% fission products can be exported to exhibit a slightly lower leaching rate than glass with 20% fission products.

Experiments in Marcoule with water quality varying from tap water to seawater show virtually the same leaching rate. The leaching rate does, however, increase at low and high pH values. Measurements have found an increase of the leaching rate by a factor of 10 at pH 3 and an increase by a factor of 20 at pH 14 as compared with the leaching rate at pH 8. At less extreme pH values, the increase is less. At pH 11, for example, which exceeds the pH value which can be expected in a final repository, the leaching rate does not deviate significantly from the rate at pH 8.

In trials with French glass, a temperature increase from 25° to 70°C increased the leaching rate by a factor of 10. The temperatures which can be expected in a Swedish final repository lay between 20° and 70°C.

Samples of high-level borosilicate glass from gas-graphite reactors in Marcoule which were manufactured and tested in 1966 were subjected to test leaching again in 1976. The leaching rate proved to be virtually unchanged, 10^{-7} g/cm^2 and day, with an otherwise identical procedure. Nor is any significant change in the leaching rate expected to occur over very long periods of time.

Experiments are being conducted at Studsvik with the leaching of high-level glass from Marcoule and of high-level glass with an extra-high plutonium content. The leaching experiments are being conducted with water grades which correspond to the groundwater around a final repository. Results obtained thus far agree in essence with the French trials.

5.4.4 Thermal and mechanical properties of borosilicate glass

When the glass has been cast in the chromium-nickel steel cylinder, it will be subjected to compressive stresses due to the shrinkage of the cylinder as it cools. The ability of the glass to resist such stress is very good. However, the rapid cooling which takes place during decontamination of the cylinders may cause the glass to crack. Studies in Marcoule have shown that a surface enlargement corresponding to a factor of 2-3 takes place upon extremely rapid cooling. The maximum surface enlargement in handling and transport has been estimated to be a factor of 10.

Heat is generated inside the glass. At temperatures above 550°C, there is a risk that some of the glass will crystallize. Experiments have been conducted in Marcoule in which high-level glass blocks were held at temperatures of 800°C for 100 hours. The blocks had not cracked and the leaching rate after the heat test, exhibited only a slight change. The temperature in the centre of the glass cylinder is kept below 100°C in the final repository - well below the 550°C which can be critical. The centre temperature is below 550°C during intermediate storage as well, even in the event that all ventilation systems should fail.

The fission products in the fuel contain inactive molybdenum. In industrial glass manufacture, this molybdenum can give rise to the formation of a molybdate phase in the glass if the holding time at 800°C is long. This phase consists mainly of sodium molybdate but also contains strontium, lanthanum and perhaps some cesium and americium. The molybdate phase is soluble in water. Under the most unfavourable conditions, 0.5% of the glass could consist of molybdate phase. This phase can contain some quantities of strontium and cesium. A low fission products content counteracts the formation of molybdate phase.

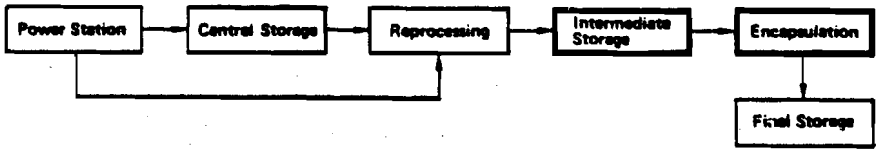
5.4.5 Resistance of borosilicate glass to radiation

The effects of radiation on borosilicate glass have been studied using high doses of beta radiation corresponding to $1.2 \cdot 10^{-11}$ rad. Storage for 1 000 years leads to a total dose of $2.4 \cdot 10^{-11}$ rad. The results of tests of irradiated samples show

- no energy accumulation (Wigner effect),
- no change of leaching rate,
- no change of structure.

The greatest risk with radiation comes from alpha radiation (helium particles). In order to study this risk, alphaemitting actinides (americium-241, plutonium-238, curium-244) were added to glass in such great quantities that a dose corresponding to that which is obtained over 1 000 years for high-level glass was obtained in 1-2 years. This type of accelerated experiment entails a more severe test than a lower dose rate over a longer period of time. This is due to the fact that the helium atoms have less opportunity to diffuse out of the glass and that the glass structure is forced to adjust more rapidly to containing a certain quantity of helium. This type of experiment has shown that there is little change in the mechanical properties of the glass. As a comparison, there are natural vulcanic glasses which contain as much as 200 mm^3 gas per gram glass without becoming brittle.

6 INTERMEDIATE STORAGE AND ENCAPSULATION



6.1 GENERAL

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

The main purpose of intermediate storage is to reduce the heat flux from the waste and thereby simplify final storage. An intermediate storage period of 30 years is foreseen during which time heat generation decreases to approximately one-half. However, this storage period can be extended, the only limitation being how long supervised storage is considered to be desirable and acceptable.

The time during which the waste is kept in intermediate storage can be used for further development and optimization of the encapsulation procedure and the design of the final repository.

The purpose of the encapsulation following intermediate storage is to enclose the vitrified waste in a corrosionresistant and tight canistor prior to deposition in the final repository. The vitrified waste cannot be leached out unless the encapsulation material has first been penetrated due to corrosion caused by the groundwater. The canister also serves as a radiation shield which reduces radiolysis of the groundwater to a level which is negligible from the viewpoint of corrosion and simplifies handling of the waste cylinders.

The plant has a storage capacity of 6 000 waste cylinders and can receive and encapsulate 300 cylinders per year.

For a more detailed description of the intermediate storage facility and the encapsulation procedure, see chapter III:5 with appurtenant drawings.

6.2 DESCRIPTION OF FACILITY

Most of the plant will be located underground with a rock cover about 30 metres thick in order to provide protection against external forces such as acts of war and sabotage. The only surface facilities will be an entrance building with administration and service quarters and ventilation inlets and outlets (see fig. 6-1).

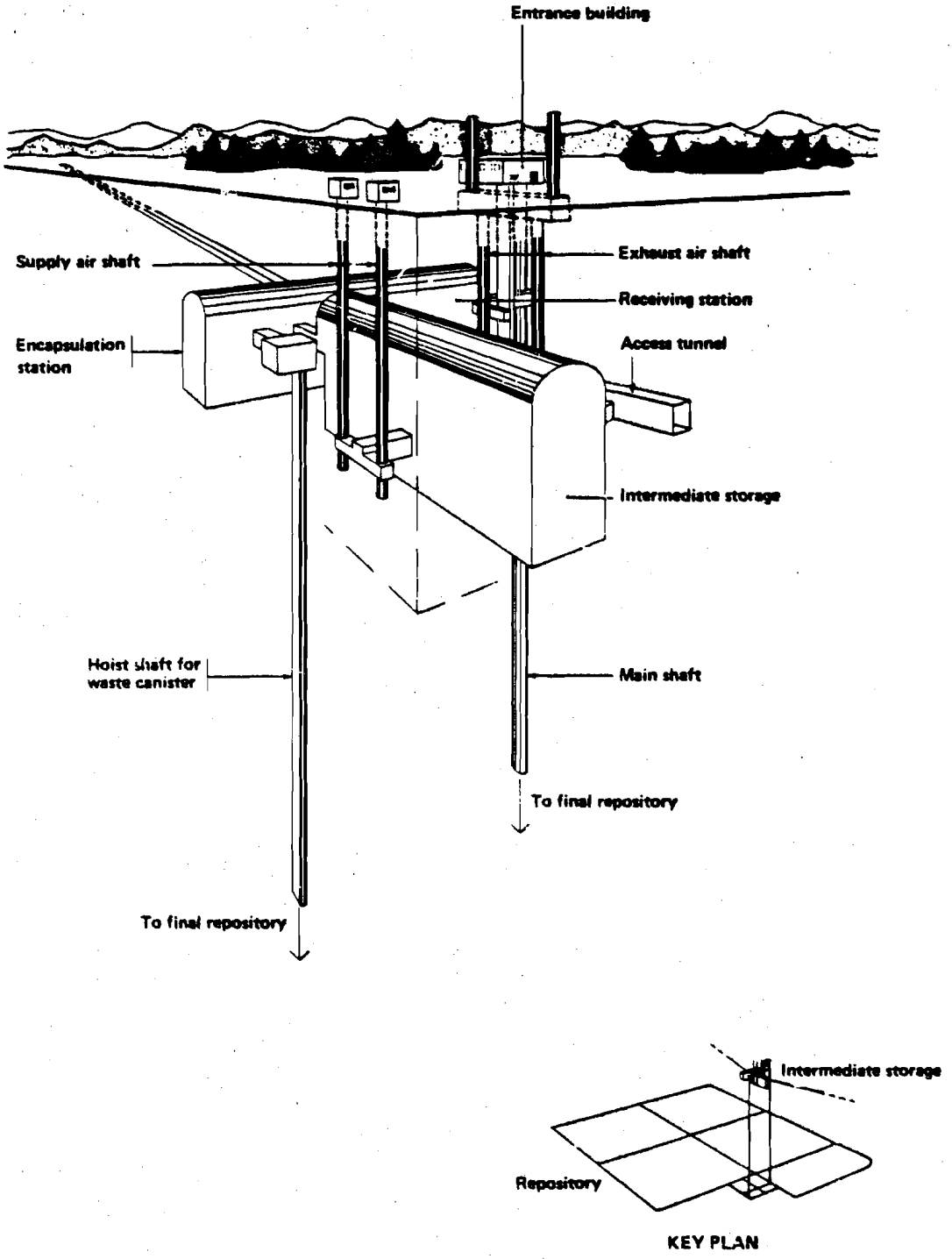


Figure 6-1. Perspective drawing of plant for intermediate storage and encapsulation. It is located underground with a rock cover approximately 30 metres thick. The plant is located above the final repository.

The various stages involved in the handling of the waste cylinders in the plant are illustrated in figure 6-2.

The waste cylinders arrive at the plant in a special transport cask on a trailer (see I:9) through an access tunnel. In the reception section of the plant, the waste cylinders are unloaded from the transport cask to the unloading cell. If the waste cylinders are found to be damaged or contaminated, they are provided with a new outer container in a recanning cell. The waste cylinders which are stored in the intermediate storage facility are therefore not radioactively contaminated on the outside.

From the unloading cell, the waste cylinders are transferred inside a radiation-shielding transfer cask, to the intermediate storage where they are placed in steel pits in concrete trenches covered by a concrete slab. Each trench contains 150 steel pits, each with room for 10 waste cylinders stacked one on top of the other - for a total of 1 500 per trench. The facility has four trenches in two groups with a total capacity of 6 000 waste cylinders.

In order to dissipate the heat emitted by the waste, air is circulated through the storage pits by means of a ventilation system with ample reserve capacity. But even in the event of o total failure of all fans, natural air convection will provide sufficient cooling to keep the temperature of the waste glass well below the critical level above which the glass may crystallize. Since the waste cylinders are clean externally, the ventilation air which is released to the atmosphere via a ventilation shaft and a chimney is not contaminated. The ventilation system can nevertheless be provided with filters and equipment for radioactivity measurement as an extra safeguard.

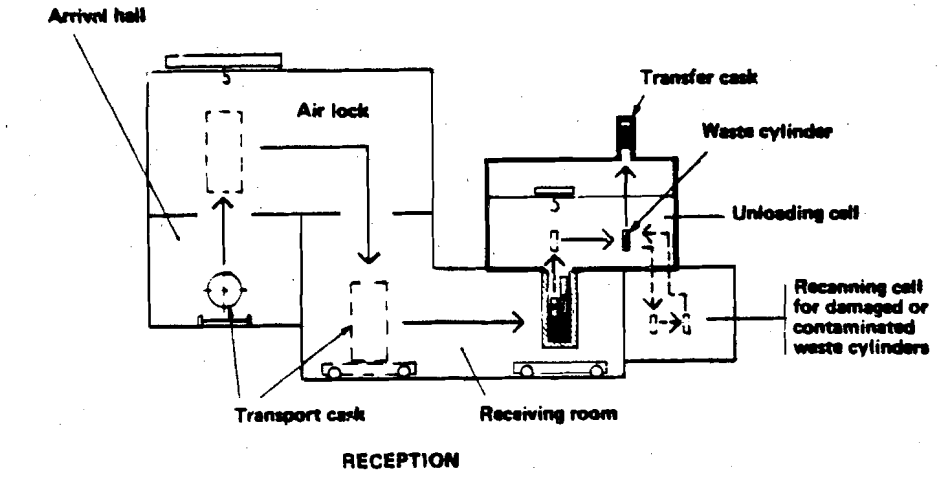
The concrete slab over the storage trenches is sufficiently thick, and the holes in the slab above the steel pits are sealed in such a manner, that sufficient radiation shielding is provided for the hall above the storage facility. Furthermore, the air pressure in the hall is maintained at a higher level than that in the trenches, so air from the trenches cannot enter the hall.

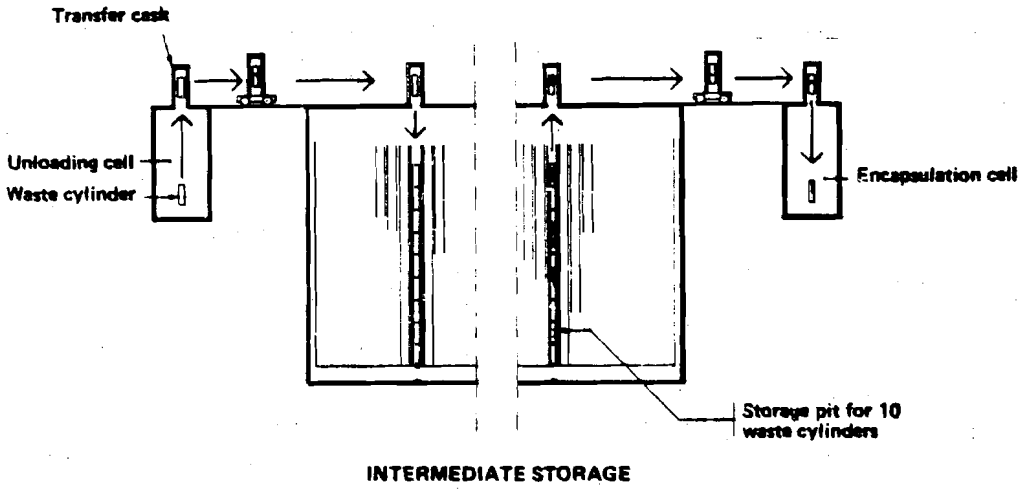
At the end of the intermediate storage period, the waste cylinders are transferred in the radiation-shielding transfer cask to the encapsulation part of the plant. There they are placed in an encapsulation cell, where they are enclosed in a lead-titanium canister (see fig. 6-3). After quality control, the encapsulated waste is transferred to the final repository.

6.3 PROPERTIES OF ENCAPSULATION MATERIAL

In the final repository, the waste canisters are subjected to the action of the groundwater in the rock. It is therefore imperative that the waste glass be protected against leaching during the period when the waste is highly hazardous (toxic, see fig. 6-4). Protection against leaching is obtained by enclosing the waste glass in a corrosion-resistant canister.

The chromium-nickel steel container in which the vitrified waste is delivered from the reprocessing plant is not accredited with any protective life of its own. Instead, the real protection is





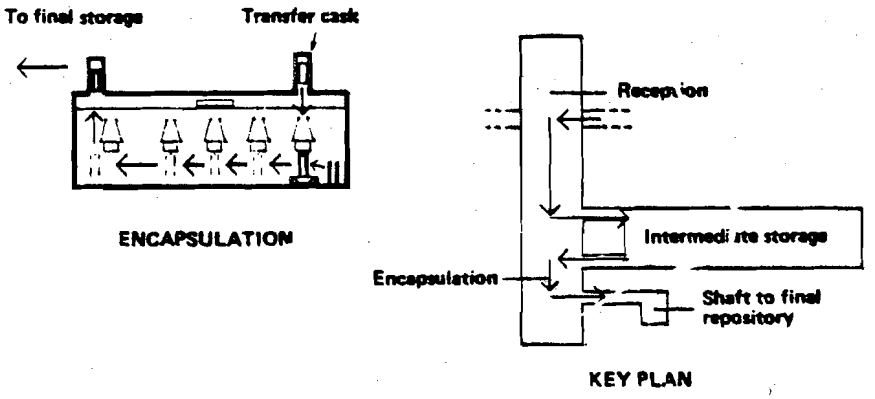


Figure 6-2. Reception and intermediate storage. The transport casks arrive in the reception section and the waste cylinders are unloaded. Damaged or contaminated cylinders are encased in an outer container of chromium-nickel steel. The waste cylinders are transferred to intermediate storage inside a transfer cask. After storage for at least 30 years, the cylinders are transferred to the encapsulation cell.

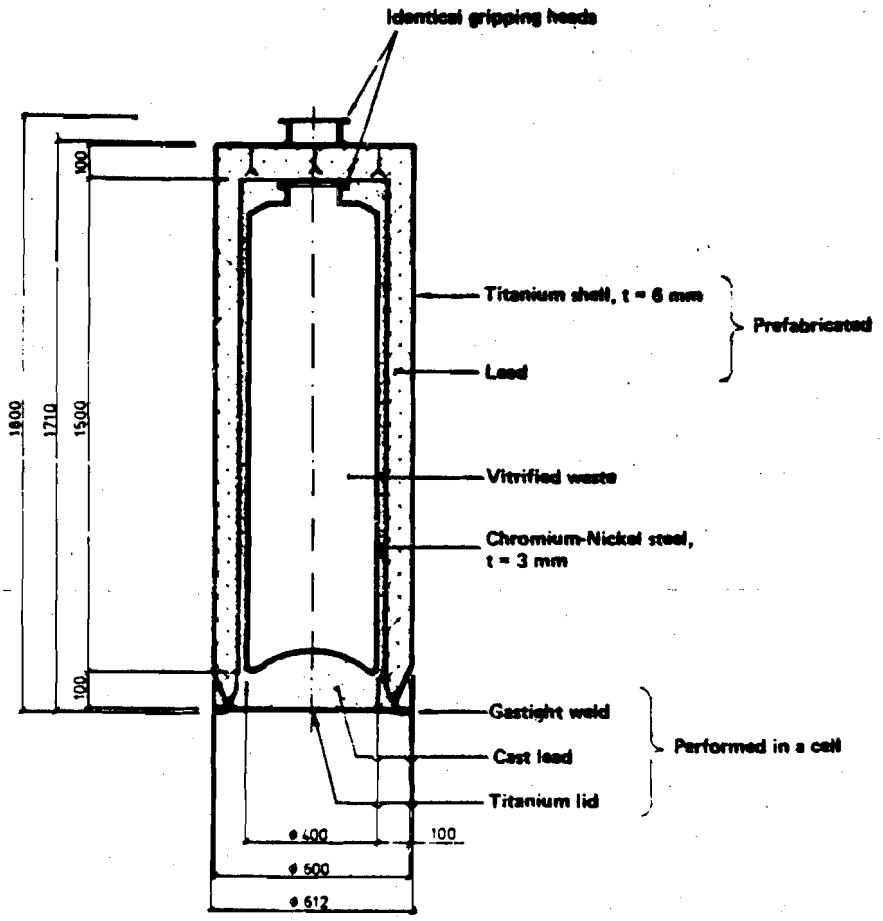


Figure 6-3. Lead-titanium canister. Waste cylinder with vitrified high-level waste in a canister of lead and titanium. Total weight approx. 3900 kg.

afforded by a canister made of lead and titanium, both of which materials possess good resistance to corrosion. The lead also serves as a radiation shield which reduces the radiation level and the radiolysis of the groundwater to a level which is negligible from the viewpoint of corrosion.

The corrosion resistance of the titanium casing derives entirely from the creation of a protective passivating layer. Under prevailing conditions, this passivating layer is self-healing when damaged. As long as this layer is intact, general corrosion of the material is extremely slow. Under the environmental conditions which are expected to prevail around the canisters in the final repository, local corrosion of titanium has not been observed at all. The titanium casing can be expected to remain intact for a very long period of time.

As far as the lead is concerned, general corrosion can be disregarded, since the lead is protected by the titanium casing. If

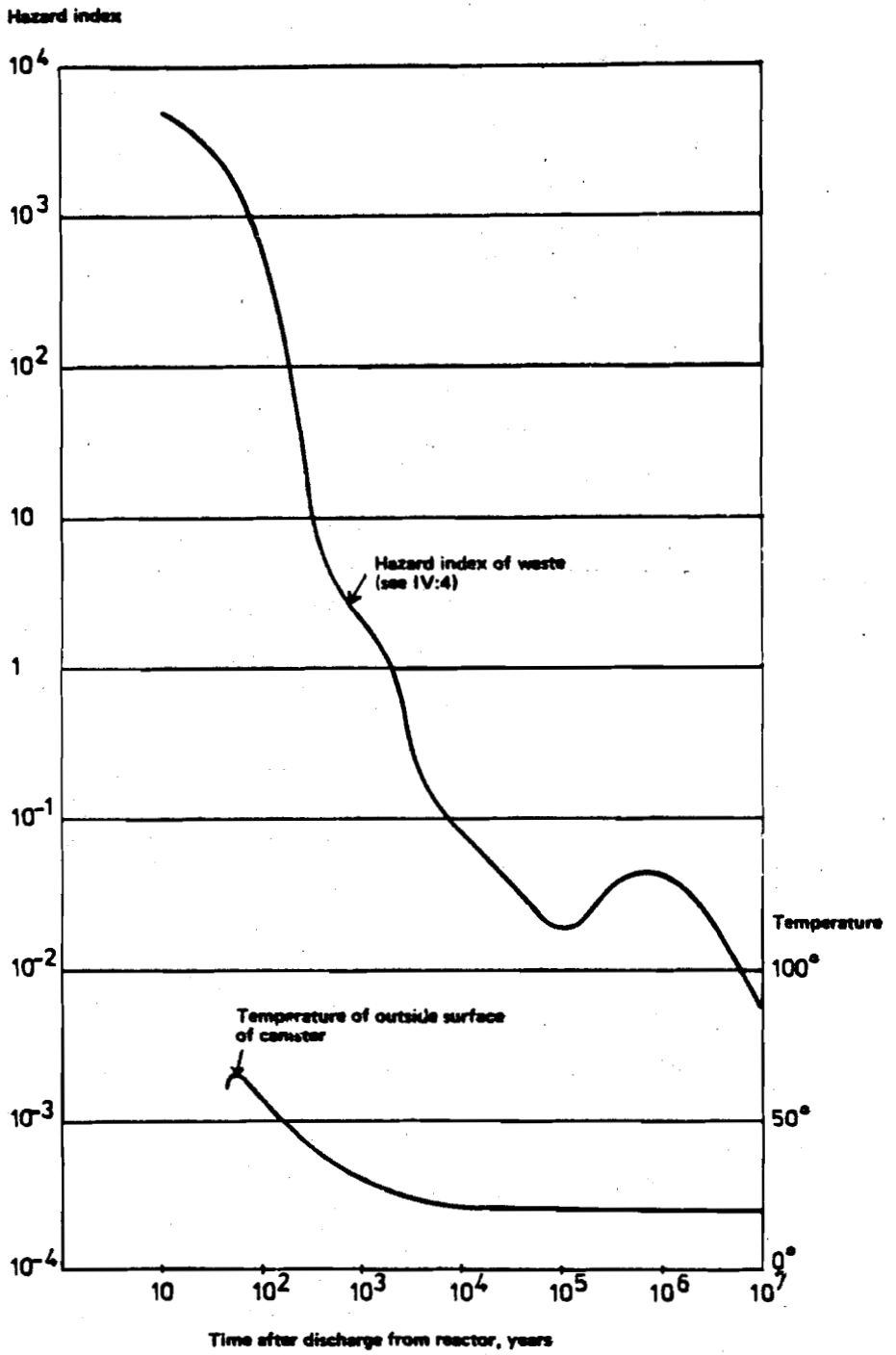


Figure 6-4. Graph showing how the toxicity of the vitrified waste (hazard index) and the external surface temperature of the canister vary with time. Note that the hazard index and time scales are logarithmic.

the titanium is penetrated, however, some pitting corrosion may be expected on the exposed surface. The quantity of lead which can then go into solution is estimated to be slightly more than 2 kg over a period of 1 000 years. The attack will penetrate down into the lead at a diminishing rate. It is tentatively estimated that pitting will penetrate the lead lining at the earliest about 500 years after the titanium casing has been penetrated, but this figure is probably grossly underestimated.

The Swedish Corrosion Research Institute and its reference group of specialists within the field of corrosion and materials was commissioned by KBS to examine the corrosion resistance of the proposed encapsulation materials. In a status report dated 27 September 1977, which is reproduced in KBS Technical Report No. 31, the life of the canister is estimated by some members to be at least 1 000 years and by others to be at least 500 years. However, a final assessment will not be forthcoming until the results of current in-depth studies are available.

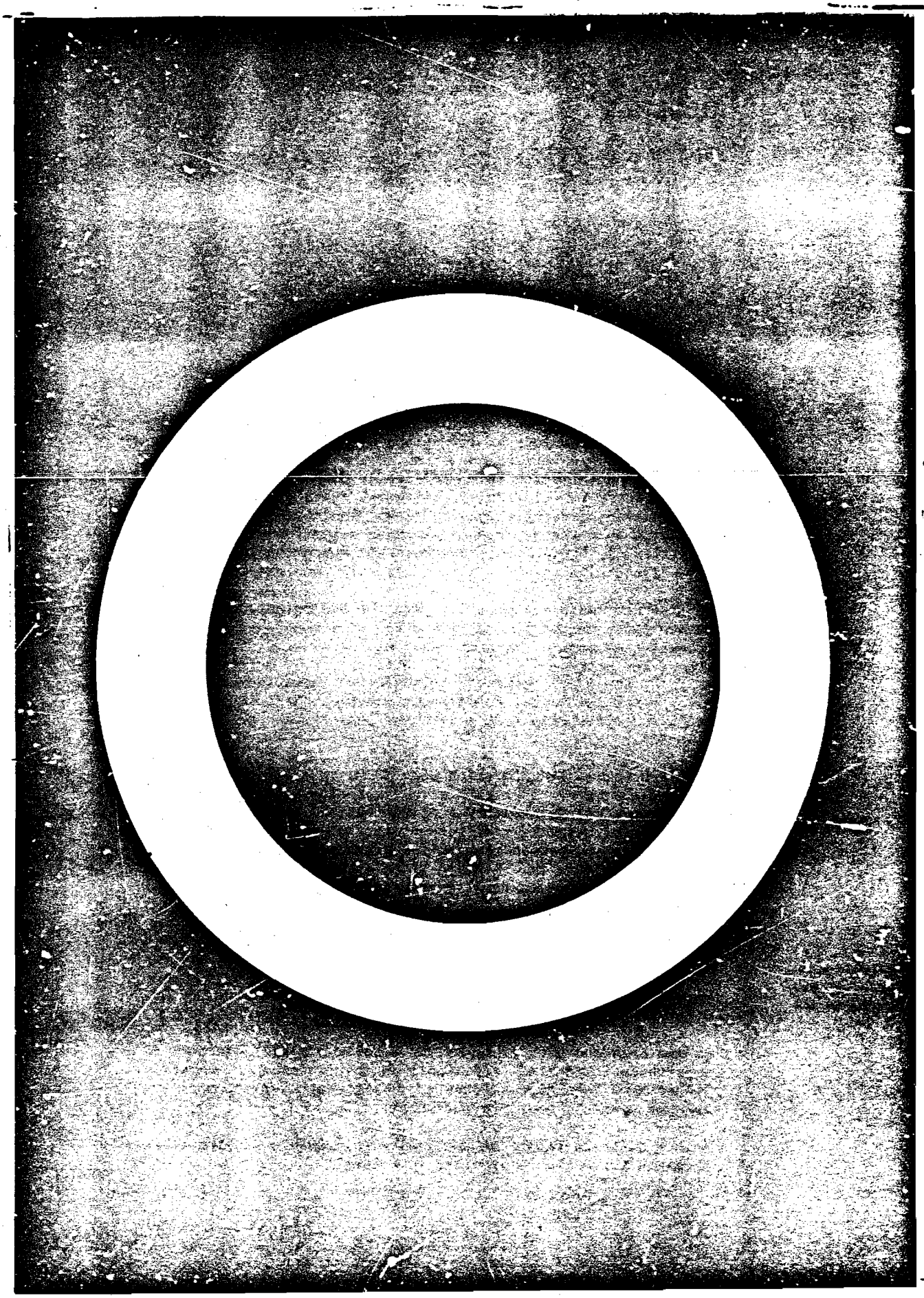
In one of the supplementary statements submitted by the members of the reference group, it is claimed that the estimates given in the status report are conservative and represent a lower limit for the durability of the encapsulation material. It is furthermore submitted that on the basis of existing knowledge, it is highly probable that further study will reveal a considerably longer life for the encapsulation material. KBS shares this opinion. See also III:5.3.

6.4 OPERATION OF FACILITY

Operation of the reception section and the encapsulation section is based on remote-controlled handling in closed cells. The operating personnel, whose main function is one of surveillance, are protected against radiation by thick concrete walls and radiation-shielding windows. If necessary, the equipment can be moved from the cells to prepared areas where it can be repaired and maintained. This technology has been proven and has been used for many years in a similar storage facility in Marcoule in France.

The plant and its operation will be under the supervision of the Swedish Nuclear Power Inspectorate and the National Institute of Radiation Protection in the same manner as a nuclear power station. The plant will be designed in compliance with the regulations which these authorities and the occupational safety authorities issue, following consultation with concerned personnel organisations. With regard to working environment and safety, see chapter I:10.

When the facilities for intermediate storage and encapsulation have served out their useful lives, decommissioning will be facilitated by their emplacement in rock. Decommissioning procedures are basically the same as those described for the central storage facility for spent fuel in section I:4.2.4.



7 GEOLOGY

7.1 GENERAL

The feasibility of safe terminal storage and disposal of high-level waste in geological formations has been under consideration for some time in various countries. Since it has generally been assumed that the waste is to be finally disposed of in the country where it is produced, different types of formations have come under consideration: salt, clays, shales, crystalline rock - depending upon the occurrence of these formations in different countries. In Sweden, interest has been concentrated on precambrian crystalline rock formations (granite, gneiss).

KBS has concluded an agreement with the Geological Survey of Sweden (SGU) concerning the execution of a comprehensive programme of geological field studies and theoretical investigations. The programme entails the drilling of 10 boreholes to a depth of about 500 metres, examination of cores and boreholes, water injection tests, groundwater analyses, hydrological tracer tests in crystalline rock and theoretical studies of groundwater movement.

Parallel to and in connection with SGU's investigations, various researchers have, under commission from KBS, carried out studies and surveys of the properties of the bedrock and potential movements in the bedrock in different parts of the country, patterns of movement and composition of the groundwater and various retardation effects on dissolved material as the groundwater passes through the buffer material and cracks in the rock.

An experimental station where observations and experiments can be carried out in a granite massif at a depth of 360 metres has been established in the Stripa mine.

Matters concerning geology and rock mechanics have been dealt with by an advisory group of experts.

On two occasions (in February and October of 1977), matters of importance for a final repository in the precambrian bedrock of Sweden have been discussed at special conferences attended by many of Sweden's geological experts.

A more detailed report on the geological surveys and the results obtained from them is provided in volume II.

7.2 OBJECTIVE

The geological study programme carried out by SGU for KBS was aimed at elucidating the bedrock and groundwater conditions which are determinant for the long-term safety of a storage facility in the crystalline basement rock of Sweden. The studies span a number of different disciplines. The bedrock at the site which is finally chosen must consist of a suitable type of rock of sufficient extent both horisontally and vertically. The occurrence of discontinuities and fracture zones can affect the design and safety of the rock repository. As regards the groundwater, information is required on its chemical composition, how much water can come into contact with the waste and for how long a time the water resides in the bedrock. It is also important to elucidate where the groundwater from a repository approaches the surface of the ground and how much it is diluted on its way to the surface, as well as how well the bedrock is able to retard and retain certain waste substances if they should escape into the groundwater.

Field studies have been conducted at five sites, three of which have been selected for further study. It should be emphasized that the present work was not aimed at finding a suitable site to be proposed for the location of a future rock repository at this point in time.

The geological surveys comprise a part of the work aimed at satisfying the requirement of the Conditions Act to demonstrate where an absolutely safe final storage of high-level waste can be effected. The selected areas are examples of sites where a final repository might possibly be located and where the natural conditions have been studied both from the surface and in depth. The studied areas contain our most common types of rock which are highly unlikely to be of interest for mining - namely gneiss, gneiss-granite and granite - and are each representative of many other parts of southeastern Sweden. Areas with less common types of rock which might offer special local advantages have not been studied.

7.3 STUDY AREAS AND RESULTS

The locations of the studied areas and of the experimental station in Stripa are shown in fig. 7-1. Studsvik, where certain field experiments were conducted, is also shown.

7.3.1 Karlshamn area

The studies in the Karlshamn area were conducted on the grounds of the Karlshamn oil-fired power plant. Of the different study areas, this is geologically the best-known. It is situated in a district of Sweden where the regional interrelationships between bedrock structures and groundwater conditions have been subjected to more scrutiny than anywhere else, and it is the only one of the KBS study areas where data is also available from existing rock caverns.

The study area is composed of a grey gneiss - Blekinge coastal gneiss. It contains few joints and little groundwater. Moreover,

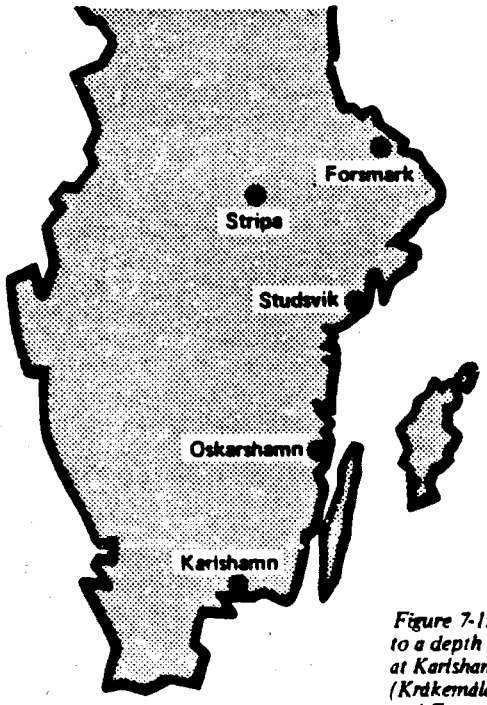


Figure 7-1. Map of study sites. Test drillings down to a depth of about 500 m have been undertaken at Karlshamn (Sternö), north of Oskarshamn (Krökemåla and Ävrö) and Forsmark (Finnsjö Lake and Forsmark). The KBS experimental station is located in the Stripa mine. Field experiments have also been conducted at Studsvik.

the directions of the fractures vary and do not exhibit any pronounced main orientation. These facts are reflected in the statistics on the existing oil storage caverns in the gneiss. Reported data on water inflow into the rock caverns, which total over 700 000 m^3, exhibit low values. The need for reinforcement following blasting has been remarkably low. The seepage data can be used to calculate the water permeability of the surrounding rock, which is expressed in m/s. Values of around 10^{-9} m/s are found, which is lower than normal for rock caverns at a depth of 30–50 m.

A core drilling within the area to a depth of 500 m shows good rock conditions at greater depths as well. These conditions stem from the fact that the gneiss, ever since its plastic folding more than 1 300 million years ago, has reacted as a rigid and highly resistant body in relation to the surrounding rock.

A special study in the area showed that displacements along existing fissures have been small for a very long period of time. The average rate of displacement is below 0.02 mm per million years. Existing cracks are largely mineral-filled. A fracture zone filled with swelling clay minerals was found north of the study area. No substantial shattered zones were found.

If the permeability of the rock and the slope of the free groundwater table is known, it is possible to calculate how much water flows through a given cross-section of the rock within a certain period of time. Due to the large differences in elevation in the

terrain in the Karlshamn area, the slope of the groundwater table is fairly steep, approximately 1:20. From the obtained data, the groundwater flow at a depth of 500 m can be calculated to be about 0.2 litres per m^2 and year. A level area with a similar bedrock would exhibit even lower flows.

7.3.2 Finnsjö area

The Finnsjö area is located 16 km west-southwest of the Forsmark nuclear power plant in northern Uppland. Geological and geophysical surveys have been conducted here. Rock conditions have been studied in depth in three core boreholes - one vertical to 500 m and two at inclinations of 50° to between 500 and 550 m vertical depth. The area is composed of older granite, which is a relatively uniform, weakly gneissified granodiorite rock. It is rather heavily fractured internally. But the cracks are mostly irregular, of varying direction, and largely filled with minerals. Small amounts of swelling clay minerals are found locally. On the east, the area is bounded by a fault which borders on an approximately 300 m wide belt with stronger fracturing. The central parts of the area, on the other hand, are distinguished by large blocks of little fractured bedrock with surface areas of up to 100 000 m^2. Between these blocks are fracture zones, some of which are filled with crushed material. The cores from the boreholes reveal sections of several hundred metres with permeabilities below 10^{-9} m/s, interrupted by a few zones with higher values. From the obtained data, the groundwater flow in large sections of rock at a depth of 500 metres can be calculated to about 0.1 litres per m^2 and year or less. Fracture zones with heavier flows are also found. Rock mechanics tests on drill cores show very good strength. The Finnsjö area represents a common type of bedrock in the Swedish crystalline basement rock and has been chosen as a reference area for some of KBS's studies.

7.3.3 Kråkemåla area

The Kråkemåla area is located 7.5 km north-northwest of the Oskarshamn nuclear power plant at Simpevarp, between the Baltic Sea and Götemaren Lake. Geological and geophysical surveys have been conducted here. Three core boreholes have been drilled, two vertial ones to 500 and 600 m, respectively, and one with a 50° inclination to a vertical depth of 570 m. The area is composed of a very uniform, undeformed granite, the Götemar granite. It is characterized to a large extent by a sparse but regular network of perpendicular, straight and long fissures. The walls of the fissures are lined with the minerals in the granite and with chlorite and calcite. Pyrite also occurs in the fissures, sometimes abundantly, as does fluorspar. Smectite, a swelling clay mineral with a good capacity to retard certain waste substances, occurs in small quantities. Drill cores from Kråkemåla exhibit considerably lower strength than those from Finnsjön.

In the boreholes, at a depth of between 300 and 500 metres, long sections are found with a water permeability of less than 10^{-9} m/s. However, these are surrounded by zones of higher water permeability and water content. The groundwater flow in the more impervious sections is estimated to be about 0.15 litres per m^2 and year or less. Considerably higher flows are found in fracture

zones, which is why rock wells in this granite often yield so much water.

7.3.4 Other areas

The two other areas where drillings were made are Ävrö, just north of Simpevarp, and Forsmark, approximately 3.5 km west of the Forsmark power station. After introductory studies, both of these areas were judged to have less favourable rock conditions than the three preceding areas, so the studies were discontinued.

7.4 GROUNDWATER CONDITIONS

7.4.1 Groundwater flows

Flows of 0.1–0.2 litres per m^2 and year have been calculated for the three study areas on the basis of model scudies calculated on the basis of potential field theory and available permeability data obtained from rock caverns at shallow depth and from direct measurements in boreholes. The borehole equipment did not permit determination of permeabilities below 10^{-9} m/s. In the Stripa mine, however, values down to 10^{-11} m/s have been measured in granite. For these reasons, the actual flows should be considerably less than the specified figures of 0.1-0.2 litres per m^2 and year. Thus, if nuclear waste is stored in any of the study areas, the low water flow rate can be expected to minimize both corrosion and leaching of the waste glass. However, time has not permitted the detailed studies required to verify these factors (see IV:6.2.2 and 6.3.10).

If the groundwater flow is known, it is possible to calculate how much the dissolved substances from the waste will be diluted on their way towards a recipient. For example, the substances released from a final repository at a depth of 500 m within the Finnsjö area during one year would be diluted in at least 500 000 m^3 of water in the fault which borders the area on the east. The portion of the dissolved substances which reached Finnsjön would be diluted in a water volume approximately 50 times greater.

7.4.2 The pattern of groundwater flow

The flow of the groundwater is determined by the precipitation and terrain features in the area as well as the nature of the ground and the bedrock. Computer programs have been developed which can be used to calculate the flow pattern for vertical sections through an area. Among the assumptions which must be made are that slopes run perpendicular to the plane of the section and for a large distance in this direction and that the permeability of the bedrock is constant or changes with depth in a regular manner. By varying the conditions it is possible to shed light upon the flow paths in a given area despite these limitations. This has been done by means of mathematical models which illustrate a few simple typical cases as well as models adapted to the conditions existing in the Finnsjö area.

The results show that, as expected, the groundwater flows down-

ward into the bedrock in elevated areas, after which it turns and flows upward again towards large adjoining valley floors, where it can reach the surface at points of groundwater inflow into lakes, waterways and springs. The influence of terrain features often extends down to a depth of several thousand metres. The longer the slopes are, the deeper their influence reaches. The surface areas where groundwater from great depths issues are small, and the upflow is accompanied by a very heavy dilution of the groundwater by water from higher levels.

One consequence of these general conditions is that the groundwater movements in an area lacking extensive, flat aquifers are divided into smaller flow cells and that groundwater transport is predominantly of a local character.

This effect is reinforced when the valleys follow steep fracture zones in the bedrock, where the vertical permeability is high. Models of the Finnsjö area show that the flow there is directed towards Finnsjö Lake and towards the fault valley in the east.

The calculations have been extended to include the upflow over a rock repository which is caused by the heat generation of the waste at the start of the storage period. In agreement with earlier American estimates, it was found that this heating leads only to an insignificant pertubation of the prevailing flow pattern in the vicinity of the final repository. The effect of drainage of the rock formation around the final repository during the construction and deposition period has also been investigated.

7.4.3 Groundwater age

The time during which the groundwater resides in the bedrock is of importance in view of the natural decay of the radioactive substances and their retardation and retention by the rock. In the same rock volume, the residence time for the water is least in the larger, water-bearing fracture zones. In the intervening bedrock blocks with low permeability, the residence time is many times greater.

Age determinations of water samples were carried out using the carbon 14 method, which tells how much time has passed since the water seeped down through the surface layers of the ground. Four water samples from the boreholes in Kråkemåla have been studied thus far. Ages of between 4 300 and 11 000 years have been founded.

The uncertainty inherent in these determinations is only ±100 years. Greater uncertainty in the age determinations is associated with the sampling and drilling procedures. During drilling, surface water was used for flushing, and heavy drainage pumping was carried out prior to sampling. Disturbances which lead to age underestimations may therefore have occurred. It appears most likely that the differences in age between the samples reflect differences in the permeability of the surrounding sections of rock.

Similar age data were previously obtained from a deep well in bedrock in Finland and a tunnel at a depth of about 300 metres at

Storjuktan, Sweden. With the support of such age data, the residence time of the groundwater at the depths in question can be estimated to be more than 10 000 years. When this figure is used as a basis for estimating the transit time of the water from a final repository to the biosphere, it is necessary to take into consideration the location of the repository and local topographical and hydrological features, which are dealt with in greater detail in volume II.

7.4.4 Groundwater chemistry

The chemical composition of the groundwater is of importance for the lifetime of the canisters in which the waste is enclosed. Retardation effects in the buffer mass and in cracks in the rock can also be affected by the composition of the groundwater. The level of chlorides and dissolved oxygen in the groundwater is of particular importance.

An evaluation of data from KBS and other studies shows that chloride concentrations of more than 300 mg/l have been found in groundwater only in rare cases. The occurrence of fossile groundwater in some areas might give rise to higher values. For this reason, analysis of the local groundwater chemistry constitutes an important part of the preliminary studies which precede the final choice of a site.

Both the known occurrence of bivalent iron in the groundwater and direct analyses have shown that groundwater at great depths can only contain extremely small quantities of dissolved oxygen, normally below the level which can be determined by standard analytical methods.

The pH values which have been measured in groundwater are only rarely less than of 7.2 or greater than 8.5. The proposed buffer material of quartz sand and bentonite (see chapter III:6) should stabilize the pH value to between 8 and 9. The buffer material does not affect the level of chlorides and dissolved oxygen in the groundwater.

7.5 RETARDATION OF WASTE SUBSTANCES

7.5.1 Retardation effects

Laboratory studies of buffer material and samples from the Swedish bedrock as well as field studies have been conducted in order to investigate the retardation of the radioactive waste substances in the bedrock. The measurements show, in agreement with the large body of data in the literature, that all of these substances, with the exception of iodine and technetium, are retarded to different degrees in relation to the movement of the groundwater.

Retardation factors have been calculated under the assumption that the groundwater moves in smooth-walled, plane-parallel, continuous cracks. Retardation factors calculated in this manner are in good agreement with results from field tests conducted in fractured rock at Studsvik.

The field test at Studsvik were conducted at a depth of 70 metres in fractured rock of high water content and permeabilities around 10^{-6} m/s. In an initial test series, the transit time for certain nuclides in the unconditioned rock was measured and compared with the transit time of the groundwater along the same flow path. The same rock section was then sealed by means of injection with bentonite, which is a natural material many millions of years old consisting primarily of smectite minerals. Smectite occurs frequently as natural crack filler in the Swedish bedrock and has also been found at Kråkemåla and Karlshamn. It is in chemical equilibrium with the groundwater and the other minerals in the bedrock. After this grouting, the measurements were repeated and are still in progress. Among other things, it has been found that, strontium added to the water has not, after 4 months, arrived at the metering point located 50 metres from the borehole where it was injected.

Retardation effects are discussed in greater detail in section I:13.4.2.

7.5.2 Retention of waste substances

During their residence time in the bedrock, the elements strontium-90, cesium-137 and americium-241 and -243 decay completely. Other elements participate in chemical reactions so that they are retained or retarded in the rock. Such a fixation of cesium has been demonstrated in laboratory experiments. Other experimental studies have shown that hydrogen sulphide or minerals containing bivalent iron can precipitate insoluble uranium dioxide from solutions of carbonate complexes of hexavalent uranium by means of reduction at room temperature. Theoretically, the same should occur with plutonium, neptunium and other transuranic elements.

The same reactions occur in nature. There are examples of large uranium ore deposits which have been formed by precipitation in this manner. In Sweden, uranium dioxide occurs as fissure filler in the crystalline basement rock in such areas as northern Uppland and at Pleutajokk in Norrbotten County. In both of these cases, the mineral has been retained in the rock for more than 1 500 million years. It has also been shown that naturally formed transuranium nuclides in the Oklo uranium field in Gabon have not been dissolved by the groundwater.

7.6 ROCK MOVEMENTS

7.6.1 Recent faults

A number of studies have been carried out in order to establish whether the safety of a rock repository can significantly deteriorate during the long storage period due to new fracturing and movements in the bedrock. A brief survey has been made of recent fault movements throughout the country, whereby it was found that these fault movements follow older fault zones to a great extent. This agrees with the results of a theoretical study of the process of deformation in fractured rock. It shows that even large deformational movements are distributed over existing cracks in a granitic rock of normal fracture content without the

creation of new cracks or any radical changes in individual cracks. By locating the final repository in an area without any major fault lines and by avoiding emplacing waste canisters in existing shear zones, canister damages resulting from rock movements can be avoided. It has also been found that only insignificant movement has taken place along fissures in southeastern Sweden over the past 570 million years, even during the large-scale deformations which led to the formation of the Caledonian mountain range. The current period in European geological history is characterized, as far as we know, by declining deformation. It can therefore safely be assumed that rates of dislocation during the required period of waste isolation will remain below the mean rates which have been determined in the KBS study and which are insignificant from a practical point of view.

7.6.2 Rock stresses

According to some measurements, the nearly horisontal shear stress in the Swedish bedrock rock is close to the strength limit of the rock just below the surface. Other studies give lower values. If the strength of the rock were exceeded, fracturing would occur. However, the shear strength of the rock increases with depth due to the increase in pressure. In fact, the risk of shear fractures decreases rapidly with increasing depth, since the shear stress at greater depths is probably much the same. Special calculations show that the changes in the rock induced by blasting and the estimated increase in temperature are very local and that the risk that new groundwater flow paths will be created as a result of the formation of new fractures is negligible.

7.6.3 Effects of a future ice age

Fracture formation in connection with the current land elevation and bedrock movements during a future ice age can be assessed on the basis of the present distribution of cracks in the bedrock. Permeability values from drill cores show that the fracturing is largely restricted to the top 100 or 200 metres of the bedrock, while deeper portions still possess good integrity after 10 to 20 Quaternary glaciations. One more ice age is not expected to alter the situation.

7.6.4 Tidal effects

A special study of the gravitional effects of the sun and the moon on the Swedish bedrock and other influences which affect the groundwater in the bedrock has been conducted. No detrimental effects for a rock repository have been identified.

7.6.5 Earthquakes

A review has been made of the statistics on earthquakes in Sweden through 1975 and of certain studies concerning earthquake-caused ground accelerations. The analysis shows that southeastern Sweden has a very low frequency of earthquakes and that the ground accelerations which can be expected will not cause damage to the repository or to the waste canisters.

7.7 SUMMARY EVALUATION

The above information, together with the safety analysis, provides the necessary basis for an evaluation of the three study areas of Karlshamn, Finnsjön and Kråkemåla. It shows that these areas fulfil the basic requirements for a safe rock repository for high-level waste, provided that the design of the facility takes into account the geometry of the low-permeable rock formations.

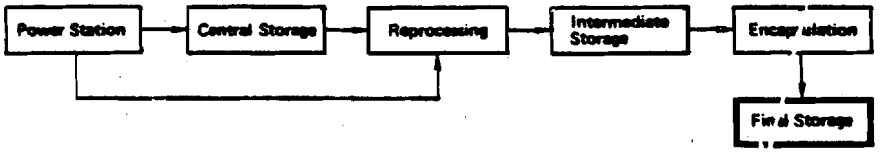
On the basis of existing knowledge, the coastal gneiss region of Blekinge is from a geological point of view, the most attractive area for a final repository.

The gneissified granitoid rock at Finnsjön also appears to offer large volumes of low permeability. However, existing internal fractures and crush zones may entail certain technical problems of the type which are normally encountered in tunnelling and rock cavern excavation. Compared with the Blekinge coastal gneiss, this type of rock permits greater freedom of choice in the location of a future rock repository, since similar rock conditions are found throughout large parts of southeastern Sweden.

The Götemar granite at Kråkemåla exhibits, despite sections of low permeability, a number of features which may require more extensive reinforcement and grouting during the construction phase. These features include lower strength, a regular fracture system with extensive horizontal fracture surfaces and high local groundwater flows.

The three study areas can be clearly arranged in order of priority: the Blekinge gneiss, the gneissified granodiorite in the Finnsjö area and the undeformed stocklike granite in the Kråkemåla area. This confirms previous experiences regarding the structural and water-bearing characteristics of these types of rocks. Against this background, other gneiss areas besides the one in Blekinge may be of interest.

8 FINAL STORAGE



8.1 GENERAL

The encapsulated waste will be sent to the final repository for final disposal.

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.

Whereas the waste in the intermediate storage facility is stored under dry conditions, requiring surveillance of e.g. the drainage system, the final repository is designed to be sealed and finally abandoned. The encapsulated waste will therefore be exposed to the action of the groundwater.

As mentioned in chapter I:5, the leaching rate for the vitrified waste is very low. In the encapsulation station, the waste cylinders are provided with a lead-titanium canister with high resistance to corrosion (chapter I:6). Finally, the storage holes, tunnels and shafts in the final repository will be backfilled with a buffer material of quartz sand and bentonite with low permeability and ionexchanging properties. Bentonite is a naturally occurring clay material.

Thus, vitrification and encapsulation of the waste, the buffer material and the rock constitute four barriers which prevent or greatly retard the migration of radioactive elements via the groundwater to the biosphere.

The final repository has been designed for a total capacity of 9 000 canisters and a deposition rate of 300 canisters per year.

For a more detailed description of the final repository, see chapter III:6 with appurtenant drawings.

8.2 DESCRIPTION OF FINAL REPOSITORY

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 (see fig. 8-1). The final repository covers an area of approximately 1 square kilometre. The geometric layout of the tunnel system will be adapted to the geological conditions prevailing on the selected site. Vertical holes drilled in the floor of the

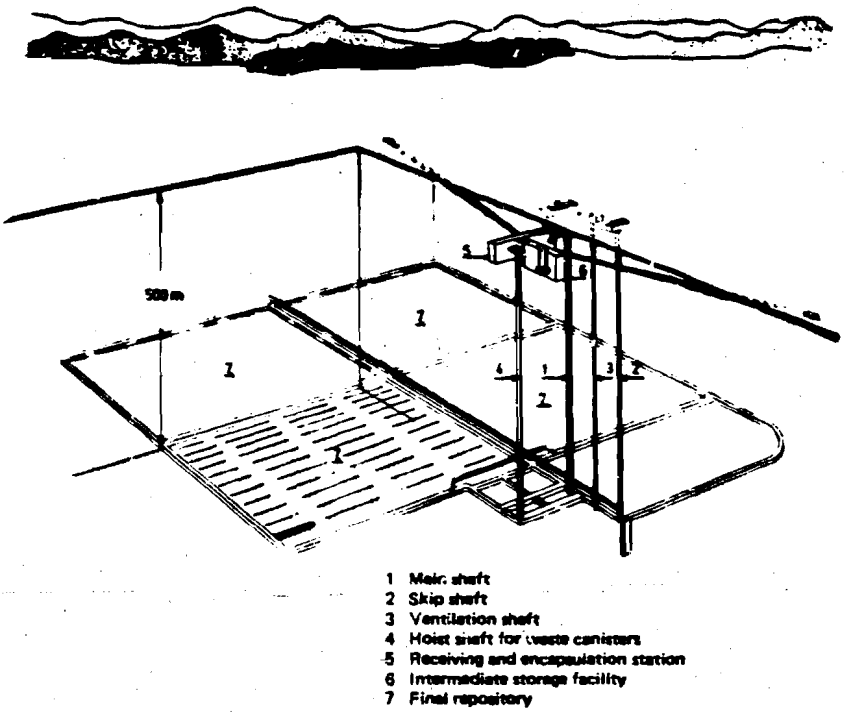


Figure 8-1. Perspective drawing of final repository with plant for intermediate storage and encapsulation. The final repository consists of a system of parallel storage tunnels situated 500 m below the surface.

storage tunnels constitute the final storage compartments for the waste canisters. Tunnels and shafts will be excavated by means of conventional mining and construction methods.

The centre-to-centre distance between the storage tunnels (25 m) and between the storage holes (4 m) has been determined on the basis of rock mechanics considerations and the effects of the heat released by the canisters (see fig. 8-2). The storage holes have a diameter of 1 m and a depth of 5 m. With the spacing selected, the increase in the temperature of the rock will not exceed 40°C. Studies show that this increase will not give rise to new cracks or new flow paths for the groundwater which could affect the safety of the final storage.

The various stages of the handling chain for encapsulated waste in the final repository are depicted in fig. 8-3.

The canisters are transferred from the encapsulation station to the final repository in a radiation-shielded transfer cask mounted on a wagon which runs on rails and is drawn by an electric tractor. They are then taken down to the storage tunnels in an elevator which travels in a vertical shaft. The elevator is equipped with safety devices which virtually eliminate the possibility of an accident.

When the transfer cask reaches the level of the repository, it is taken from the elevator through the tunnel system and placed in position over the hole in which the canister is to be deposited.

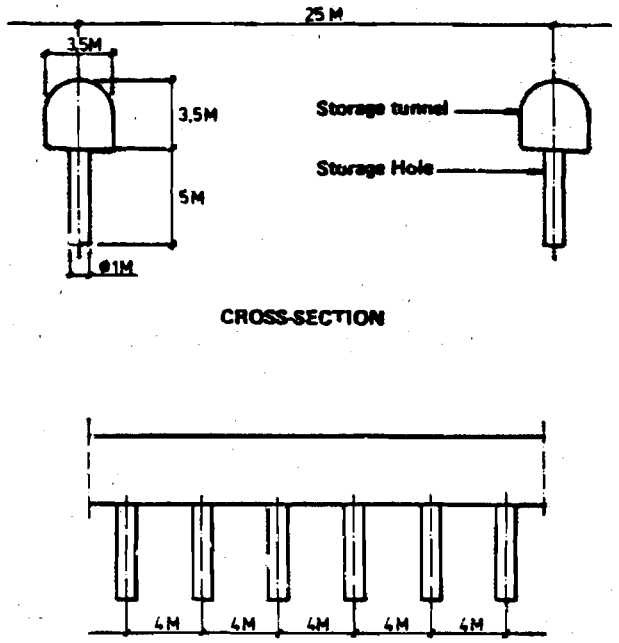


Figure 8-2. Cross-section and longitudinal section of storage tunnels in the final repository. Each storage hole is designed for one canister.

A mobile radiation shield at the top of the hole protects the personnel when the canister is lowered into the hole.

The canister is lowered into the hole onto a bed of sand/bentonite. The transfer cask and the mobile radiation shield are then removed and the hole is filled with sand/bentonite. The fill is compacted to give it good bearing capacity and low permeability. Finally, a lid is placed over the hole. The filler material provides sufficient radiation shielding for the personnel working in the storage tunnels. The properties of the quartz sand and bentonite mixture are described in section III:6.3.

8.3 OPERATION OF FINAL REPOSITORY

Deposition of the waste canisters begins when approximately one-fourth of the storage tunnels have been completed. The facility has been designed for complete physical separation of the construction work from the canister handling work.

The handling system for the canisters is similar to that which is used in the intermediate storage facility and is based on known technology. The method of applying sand/bentonite fill is based on the robot spraying technique which has been used for many years in tunneling work.

The facility and its operation will be inspected and supervised by such authorities as the Nuclear Power Inspectorate and the National Institute of Radiation Protection in the same way as a

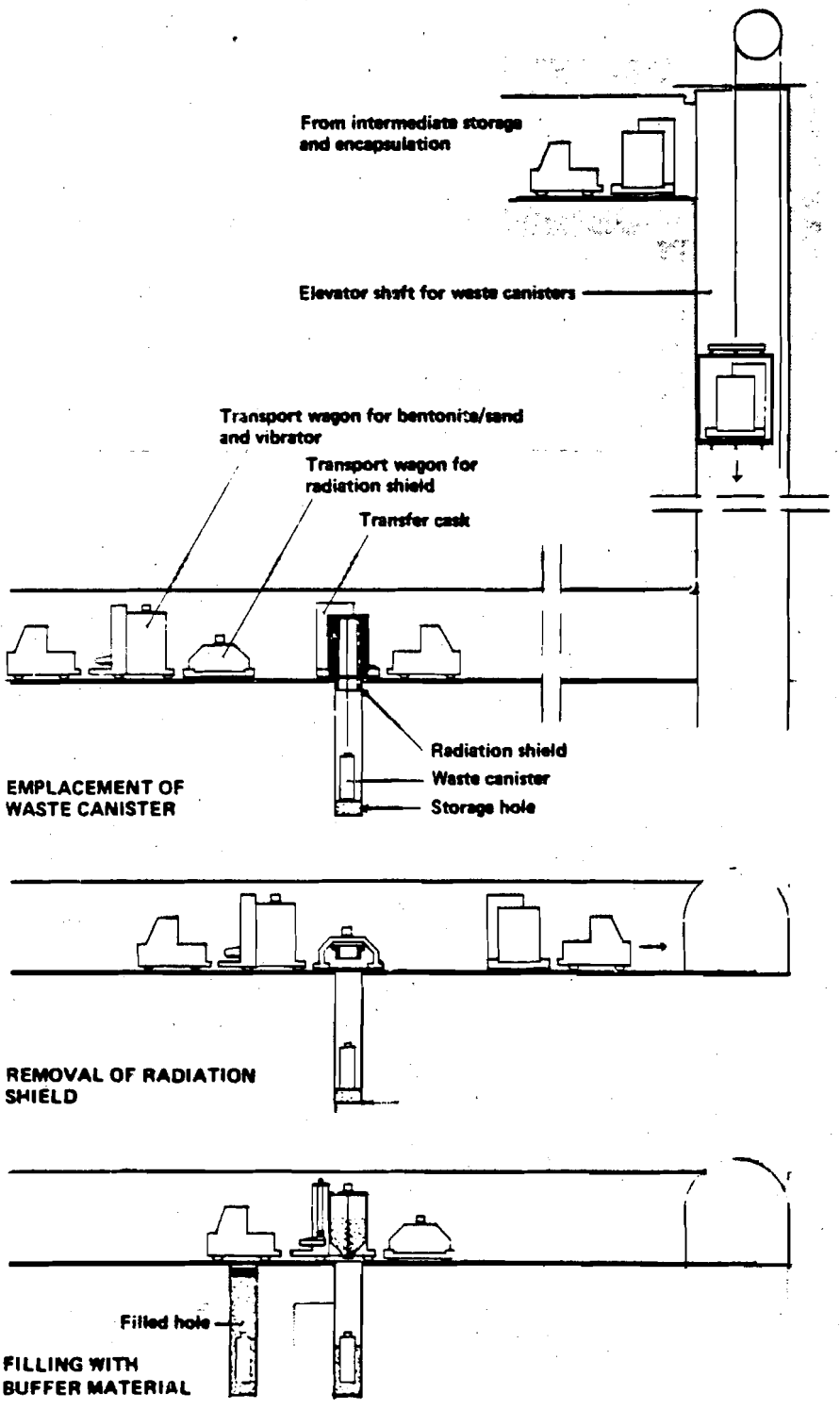


Figure 8-3. Handling of waste canisters in final repository.

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 occupational safety, see chapter I:10.

8.4 PERMANENT CLOSURE

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

When it is sealed, the tunnel system is filled with a mixture of quartz sand and bentonite similar to that used to fill the holes around the canisters (see fig. 8-4). The lower part of this fill is carried out using conventional earthmoving and compacting methods and the upper part by spraying. The application technique and the swelling of the bentonite as it absorbs water ensure that the tunnel section will be filled completely. A mixture of sand and bentonite is also used to fill vertical shafts. Boreholes which have been drilled to investigate the bedrock are filled with pure bentonite.

In this way, all cavities in the rock are filled with a material which is at least as impervious as the surrounding rock. In the storage hole, the fill material protects the canister from minor movements in the surrounding rock.

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 schedule for such activities will be drawn up in cooperation with the concerned authorities.

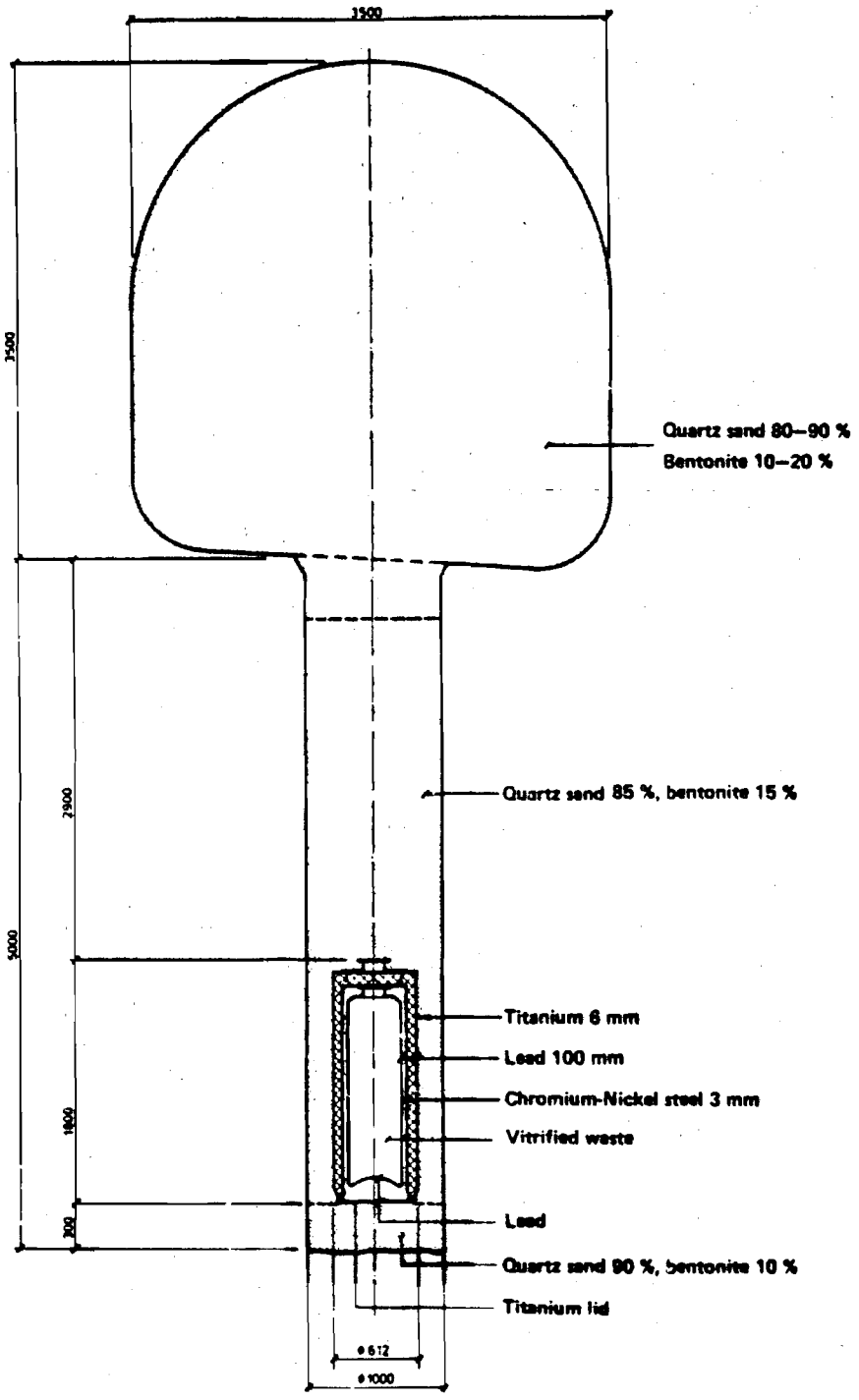
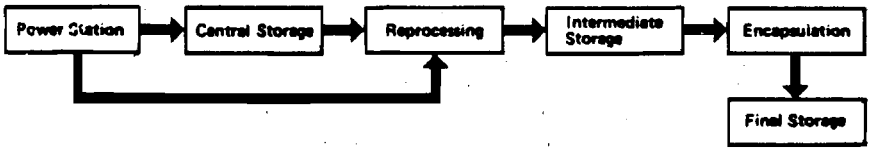


Figure 8-4. The sealed final repository. Tunnels and storage holes are completely filled with a buffer material consisting of quartz sand and bentonite.

9 TRANSPORTATION SYSTEMS



9.1 TRANSPORT CASKS, GENERAL

Relevant portions of the IAEA's transport regulations shall be observed in connection with the transportation of spent nuclear fuel and other radioactive material (see chapter I:12).

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 containers, which are described in greater detail in chapter III:2.

Every planned transport must be preregistered with the Nuclear Power Inspectorate, including specification of identification data for the selected fuel elements and a preliminary timetable 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.

The European transport casks which are 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 are currently operating a joint company called Nuclear Transport Limited (NTL) which more or less has a monopoly on the European market.

Figure 9-1 shows one of NTL's transport casks which is used for the transportation of spent fuel from the Oskarshamn plant to the reprocessing plant in Windscale, England.

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 burn-up fuel was transported, while in recent years transports have been carried out with high burn-up fuel (30 000 MWd/t) after only 6-9 months of cooling time at the reactor.

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.

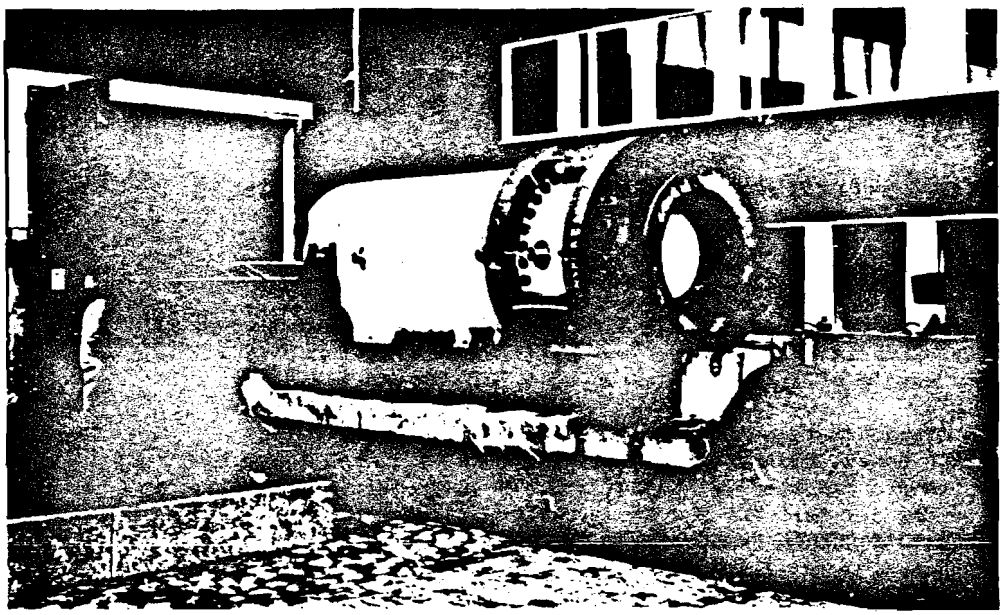


Figure 9-1. Trailer and transport cask outside of the Oskarshamn plant. This equipment has been used to transport spent fuel to the harbour for further shipment to the reprocessing plant at Windscale.

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

9.2 DESIGN OF TRANSPORT CASK

A transport cask consists of the following main components:

- An inner cask fitted with a neutron-absorbing substance usually made of a heat-conducting material.
- A heavy-duty gamma ray shield made of a heavy material such as lead or steel.
- A neutron shield to reduce neutron emission.
- Heat-dissipating flanges on the outside of the transport casks or an air-cooling system.
- A shock absorber to protect the transport cask's cover and its connections.

A transport cask for spent fuel or for high-level waste must meet the safety requirements of the IAEA transport regulations for TYPE B containers. 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.

Furthermore, the transport cask must meet the requirements imposed on TYPE A containers according to the IAEA regulations.

9.3 DESIGN OF A SWEDISH TRANSPORTATION SYSTEM FOR SPENT NUCLEAR FUEL

9.3.1 Preparatory work

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 reactors now in operation and on continued expansion to thirteen reactors.

In 1976, discussions were initiated with European and American organizations which work with the transportation of spent nuclear fuel for the purpose of examining 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. During 1978, NTL will put into use the largest transport cask ever available on the market - the NTL 12, which can transport up to 6 tons of nuclear fuel. A slightly smaller version called NTL 17 is in the design stage. It will be able to transport up to 3 tons of nuclear fuel. The NTL 11, which has already been put into operation, has the same capacity. The types of transport casks used by NTL are well-adapted to Swedish transportation requirements.

The American consultancy firm Nuclear Assurance Corporation (NAC) has designed four transport casks which go under the type designation NAC-1 and are now in routine operation in the USA. NAC is currently designing a transport cask with a maximum capacity of 3 tons of nuclear fuel. This cask is equally well-adapted to Swedish requirements.

SKBF is currently awaiting further developments on the transportation market. 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 is important that any transportations system which is adopted be compatible with any existing standard European system.

9.3.2 Scope of transports

Different alternatives have been studied in calculating the annual transport volume to the central storage facility. The required number of transport casks and the annual number of shipments by sea will depend on the following factors:

- Number of reactors in operation.
- Location of the central storage facility for spent nuclear fuel.
- Reception capacity of the central storage facility for spent nuclear fuel.

The annual discharge volume in the equilibrium state after expansion to 13 reactors will be approximately 1 400 fuel elements per year, corresponding to approximately 300 tons of uranium per year. A transport cask such as NTL11, NTL17 or the equivalent can transport max. 3 tons of nuclear fuel. When a state of equilibrium has been attained, i.e. after the fuel accumulated at the nuclear power plants has been transferred to the central storage facility, the number of casks to be transported each year will be approximately 100. 6-8 casks are required for this volume.

9.3.3 Transportation by sea

It is assumed that it will be possible to transport spent nuclear fuel to the central storage facility by sea. The construction of a ship especially adapted for that purpose is considered economically justifiable.

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. NTL11 or 17, at a time. Avaiable 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 must be equipped with particularly 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 either for conventional cargo handling or for roll-on roll-off. With conventional handling, the cargo is lifted directly down into holds by means of dock-based cranes. This method is used today at the nuclear power plants. With roll-on roll-off handling, the transport vehicle - the trailer - can drive both onto and off of the ships 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 anchored 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 hold 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 icebreaker will be required under difficult ice conditions.

The time for delivery of a vessel of the type described here from a Swedish shipyard is currently 1 1/2 to 2 years.

9.4 TRANSPORTATION OF VITRIFIED HIGH-LEVEL WASTE

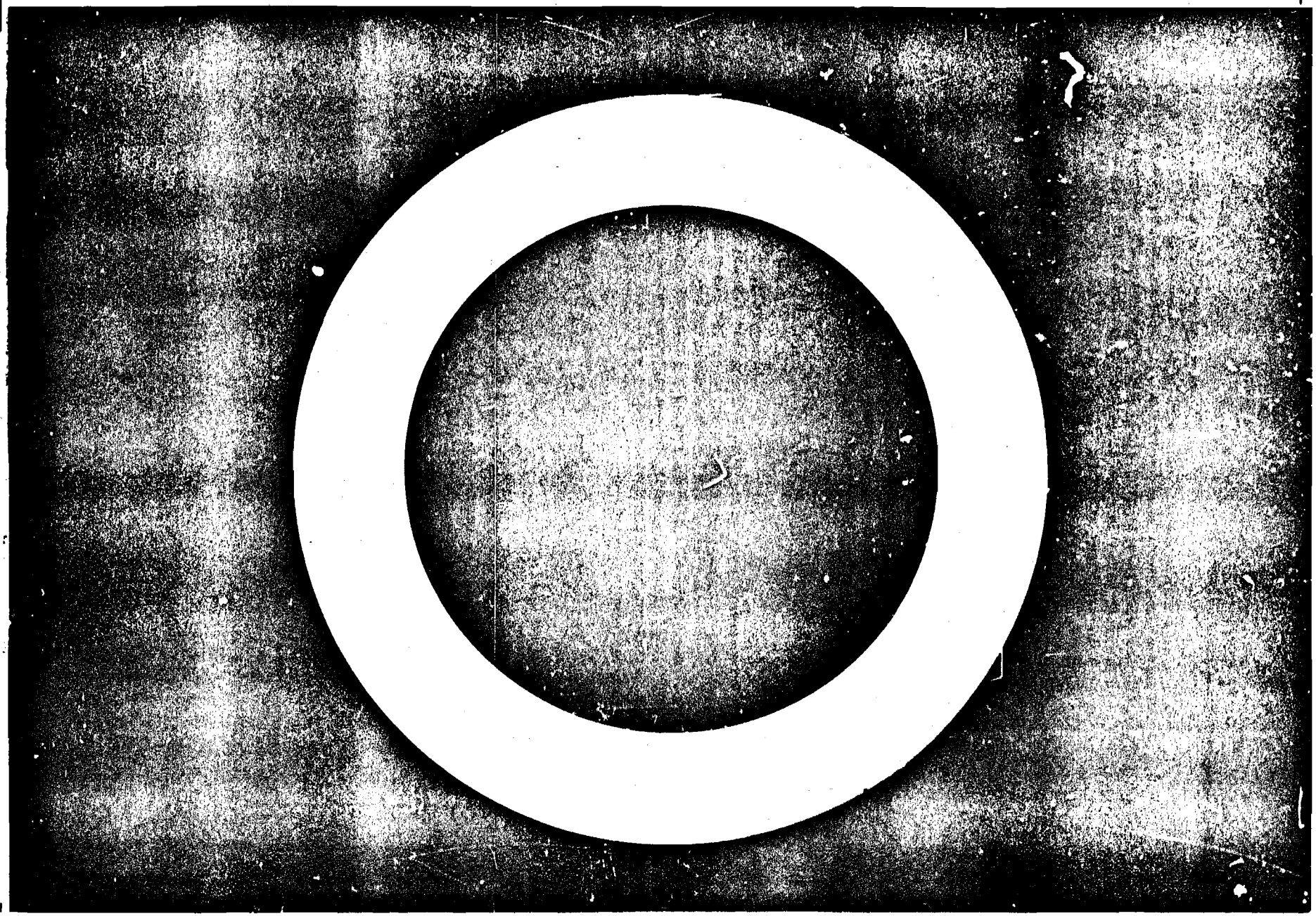
9.4.1 General

The transportation of the waste cylinders containing the solidified high-level waste from European reprocessing plants is handled by the reprocessing company or by a transport organization contracted by the company.

The waste cylinders will be transported from the reprocessing plant to Sweden in transport casks which are virtually identical to those used for spent nuclear fuel. NTL12 is one of the casks which may be used. It can transport up to 6 tons of nuclear fuel with a maximum permissible heat flux of 100 kW. Calculations carried out for this transport cask show that 15 waste cylinders can be transported. The heat flux is thereby 17 kW, which is far below the values permitted for the cask. The gamma ray and neutron shield is fully adequate to satisfy IAEA standards.

9.4.2 Scope of transports

After expansion to 13 reactors, the annual volume of fuel discharged from Swedish nuclear power plants will be about 300 tons of uranium, which corresponds to 300 waste cylinders. The ships which are used today have a cargo capacity corresponding to 6 transport casks of type NTL12. A transport ship will thus be able to carry a maximum of 90 waste cylinders, corresponding to 3-4 shipments per year if all nuclear fuel is reprocessed.



10 PROTECTION

The word "protection" is used as a collective term to cover working environment, rescue service, radiation protection, physical protection and wartime protection. These matters are dealt with in greater detail in chapter III:7. Matters of this nature are also dealt with in the descriptions of the design and functions of facilities and transportation systems. The working environment at plants for the reprocessing of nuclear fuel is dealt with in section III:4.1.4.

10.1 WORKING ENVIRONMENT

The Workers' Protection Act and various other statutes regulate matters of occupational hygiene in connection with the design, erection and operation of facilities. These matters shall be dealt with by inspection authorities and employee organizations before the facilities in question are erected. This report provides information on the nature of the working environment issues and on how the design and operation of the facilities are affected by attention to such matters.

10.2 RESCUE SERVICE

According to the Fire Protection Act, rescue service activities are aimed at minimizing damage to human beings, property or the environment in the event of fires, floods or other emergencies. Fires can cause severe damage to subsurface facilities, so special attention must be devoted to fire protection aspects in the design of such facilities. Responsibility for such matters in Sweden rests with county and municipal authorities.

10.3 RADIATION PROTECTION

Matters pertaining to the handling of radioactive waste and 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. A proposal has been submitted for certain alterations in the legislation. The recommendations of the International Commission on Radiological Protection (ICRP) comprise the basis for determining permissible radiation doses.

Principles and rules for the handling of radiation protection matters shall be examined by the National Institute of Radiation Protection. Data on individual doses shall be reported to the institute. The precautions which are required to achieve good radiation protection hygiene in connection with the transport, handling or storage of high-level waste are not expected to pose any particular difficulties.

10.4 PHYSICAL PROTECTION

The expression "physical protection" is a collective term for a series of safeguards against theft, sabotage and other acts of violence. The National Nuclear Power Inspectorate, with the support of the Atomic Energy Act, is the inspection authority in charge of the physical protection of fissionable material and nuclear power plants. The inspectorate issues directives and regulations and supervises and enforces compliance therewith. For the police activities which may be required in connection with such activities, the Nuclear Power Inspectorate cooperates with the National Police Board, which is responsible for keeping subordinate agencies up-to-date on current regulations.

The physical protection safeguards at operating nuclear power plants are gradually being augmented to comply with the tentative regulations issued by the Nuclear Power Inspectorate. At present, there are no corresponding regulations for facilities for the treatment and storage of high-level waste. These facilities are considerably less technically complicated than nuclear power plants and probably of less interest to saboteurs, so the regulations regarding their physical protection should be simpler.

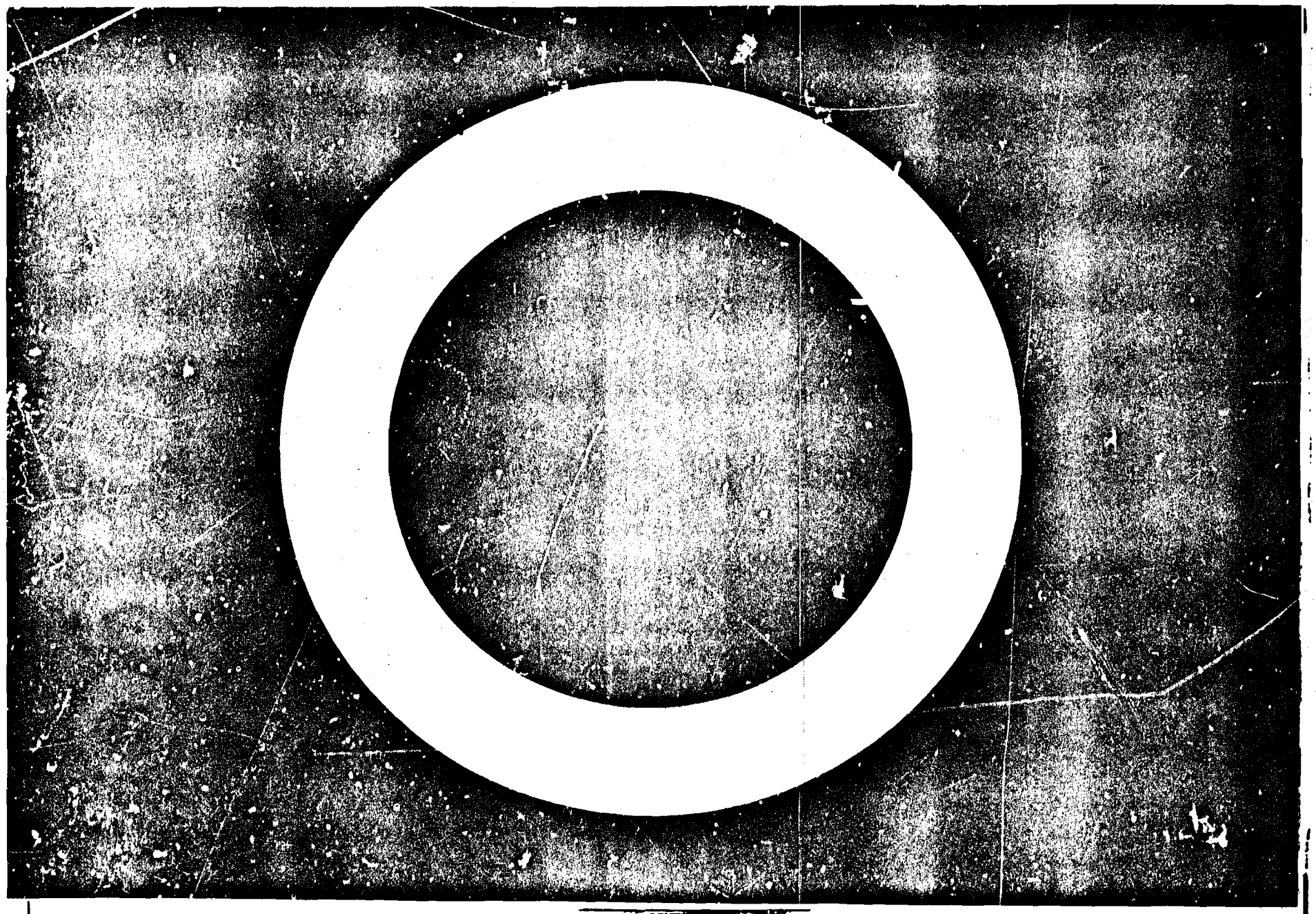
In the KBS study, however, it has been assumed that the protection at such facilities shall be largely equivalent to that at a nuclear power plant. This means that physical protection is divided into three main components: district or peripheral protection, shell protection (which is provided by robust building structures) and special protection for equipment included in vital safety systems. This last type of protection may include, for example, redundant auxiliary systems or administrative rules for authorized entry.

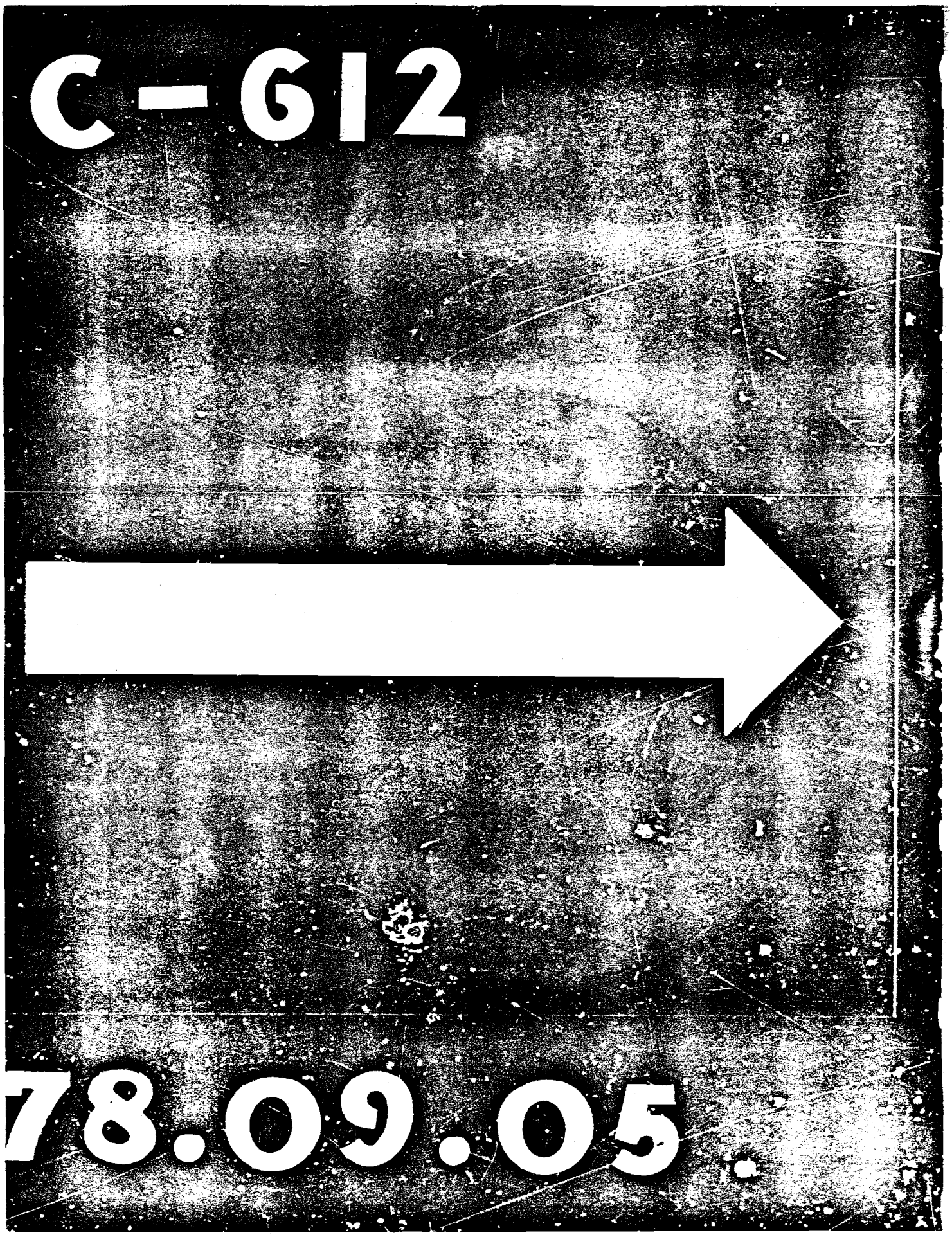
Physical protection in connection with the transportation of nuclear material shall comply with the guidelines issued by the Nuclear Power Inspectorate and already applied to the transportation of spent fuel.

10.5 WARTIME PROTECTION

The requirement for wartime protection is justified primarily by the fact that protection is required against damage which would lead to the escape of radioactive materials. It has therefore been deemed appropriate that the Nuclear Power Inspectorate, in consulation with the Commander-in-Chief of the Swedish Armed Forces and the National Institute of Radiation Protection, issue the directives and guidelines which may be considered necessary from the viewpoint of wartime protection. However, this question has not yet been formally regulated.

The emplacement of facilities for the handling and storage of high-level waste in rock permits a solution which adequately satisfies the demands for protection against conventional weapons. The final repository, with a rock coverage of 500 metres, provides adequate protection against nuclear weapons as well.

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11 SITING ASPECTS

11.1 SITE REQUIREMENTS

The transportation system comprises a vital link in the handling chain. Since all Swedish nuclear power plants, as well as the European reprocessing plants in question, are situated on waterways, all long-distance transports are expected to be made by sea. Only transports between harbours and plants will be made by land. The transportation system therefore requires access to suitable shipping lanes, channels and harbour sites. From the viewpoint of transportation, it is an advantage if the various facilities can be situated on or near the coast.

The central fuel storage facility described in chapter I:4 should, in view of the large concentration of radioactive material there, be provided with good protection against acts of war and sabotage. The storage section at least should therefore be situated in an underground rock vault. Site choice is determined primarily by proximity to the coast, availability of suitable bedrock, possibilities for coordination of manpower resources and service facilities with existing facilities and potential for future expansion.

A preliminary study carried out by PRAV has recommended Forsmark, Oskarshamn and Studsvik as suitable alternative sites.

The intermediate storage facility for waste cylinders described in chapter I:6 imposes basically the same site requirements as the central storage facility.

The encapsulation station described in chapter I:6 can also be located to advantage in a rock cavern. But in view of the limited quantities of radioactive material which are present in such a station at any one time, this should not be made as a requirement. Local conditions and coordination with other plant sections should determine whether the station is to be located in rock or on the surface. In the present study, the encapsulation station has been located in a rock cavern adjacent to the intermediate store.

The final repository described in chapter I:8 requires primarily that the site have a stable bedrock with small and slow groundwater movements. Due to limited time and resources, KBS has chosen to restrict its investigations to three areas with somewhat different types of bedrock, namely around Forsmark, Oskarshamn and Karlshamn. The studies of the final repository have been

applied to the geographical and geological conditions existing at Finnsjön near Forsmark. This site was chosen merely to give the study some geographical anchorage and is not necessarily more advantageous than other possible sites. Before a decision is made regarding the location of the final repository, comprehensive studies and investigations should be conducted over a number of years within possible areas. Existing knowledge of the characteristics of the bedrock in various parts of Sweden indicate that the coastal areas from Uppland to Blekinge contain many sections of rock which are well-suited for a final repository.

11.2 ENVIRONMENTAL IMPACT

The environmental impact of radioactive releases is dealt with in chapter I:13.

During the construction and operation phases, which partially overlap, the following environmental effects are possible:

- Interference with local residential, communication and industrial structures.
- Alteration of the landscape profile.
- Noise.
- Polluting discharges.

The extent of such impact can be limited by giving preference to sites which are of less interest for housing and industrial development. On the other hand, relatively closely situated housing accomodations and social services will be required for the construction personnel during the construction phase, so the distance to the nearest community should not be too great.

The landscape profile will be affected not only by the actual construction site, but also by access roads, power lines, harbour facilities etc. as well as stockpiles for surplus materials. If surplus materials from rockworks are used as backfill or as concrete aggregate or road-building material, the size of such stockpiles can be limited.

The noise control measures which are required to make the construction site acceptable from the viewpoint of occupational hygiene can also be assumed to be sufficient from the environmental viewpoint. The same applies to the control of dust from stone crushers and material stockpiles.

Warm ventilation air will be discharged from the intermediate storage facility and the final repository. But due to rapid dilution in the atmosphere, little environmental impact is foreseen.

In the encapsulation station, certain chemicals may be used for cleaning transport casks and glass cylinders. These chemicals shall be handled and disposed of in accordance with current regulations and government directives. Methods for this are known and have been used previously at nuclear power plants.

11.3 JOINT SITING

The question of how different facilities can suitably be jointly sited is contingent upon the following factors:

- Local conditions.
- Possibilities for rational coordination of activities.
- Deadlines for completion of the facilities.
- Time required to obtain the necessary information through studies.

In principle, the encapsulation station can be located at any point in the chain reprocessing - intermediate storage - final repository. One factor in favour of having encapsulation near the final repository in both time and space is the fact that technical developments during the intermediate storage phase can be exploited in the final design of the encapsulation process.

Theoretically possible siting alternatives are illustrated in fig. 11-1. The following comments can be made regarding the various alternatives.

Alternative 1

The comprehensive studies and licensing procedures which must precede a decision on the siting of the final repository cannot be completed by the time when the site of the fuel storage facility must be decided. There is therefore no reason to consider this alternative.

Alternatives 2 and 3

These alternatives are feasible provided that the site for the fuel store is chosen so that space and suitable bedrock is available for other required facilities as well. But alternative 3 should be given priority, since there are closer constructional and operational ties between encapsulation and final storage than between intermediate storage and encapsulation. The dates by which the various facilities have to be completed also speak for alternative 3.

Alternative 4

The present study is based on this alternative for the following reasons:

- Good possibilities for rational coordination of service and manpower resources.
- Only three external transport phases.
- The siting is not bound to the fuel storage facility

One disadvantage is that the decision on the location of the final repository must be made at the same time as the site for the intermediate storage facility is decided, i.e. at least 15-20 years earlier than is actually necessary (see I:14).

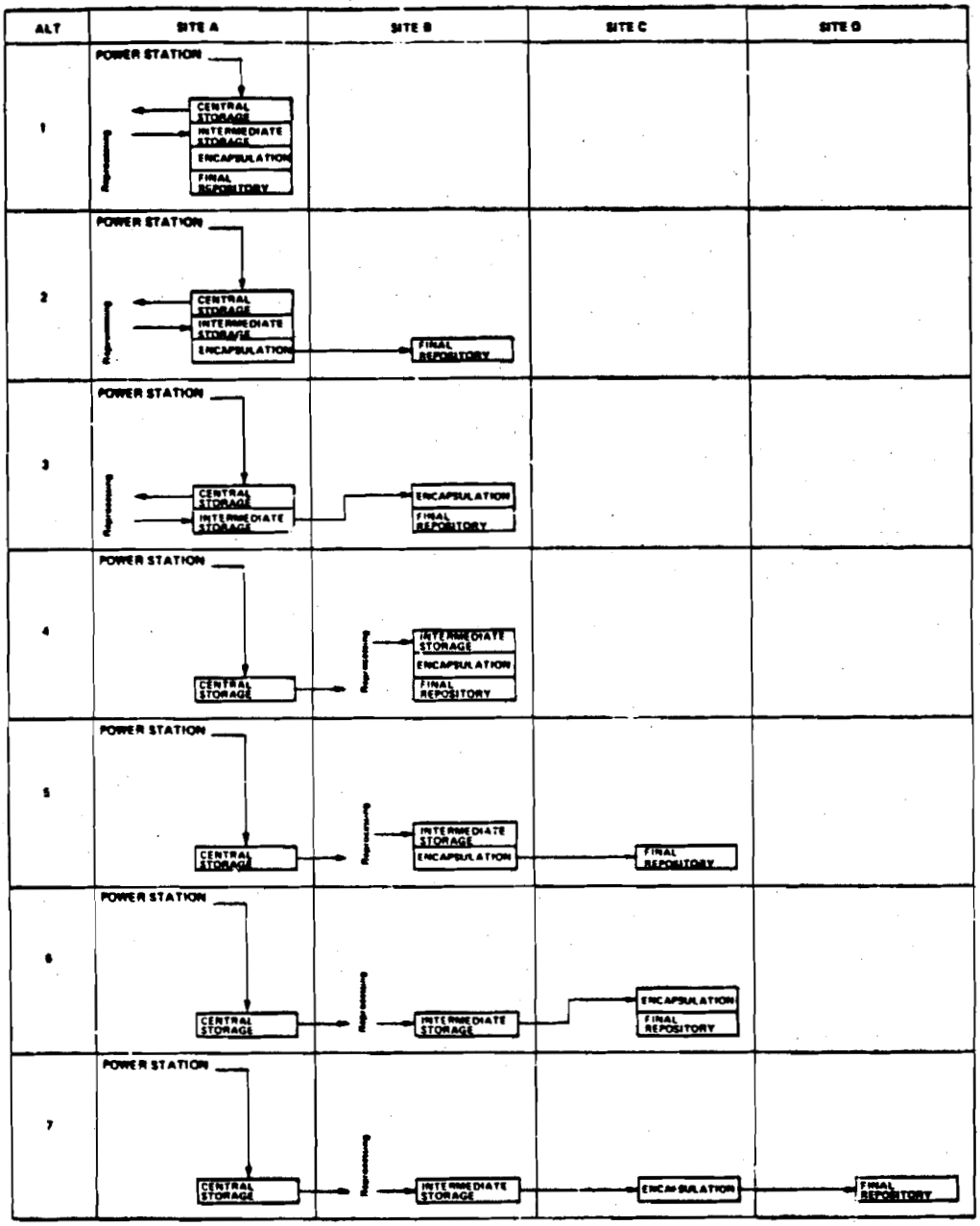


Figure 11-1. Possible siting alternatives for the various facilities for the storage of spent fuel and vitrified waste.

Alternatives 5 and 6

These alternatives provide poorer opportunities for coordination of resources than the previous alternatives, since they require the establishment of an additional site. Of the two, alternative 6 is preferable for the same reasons given for alternatives 2 and 3.

Alternative 7

This alternative provides the poorest opportunities for coordination and the most external transport stages and should therefore be discounted.

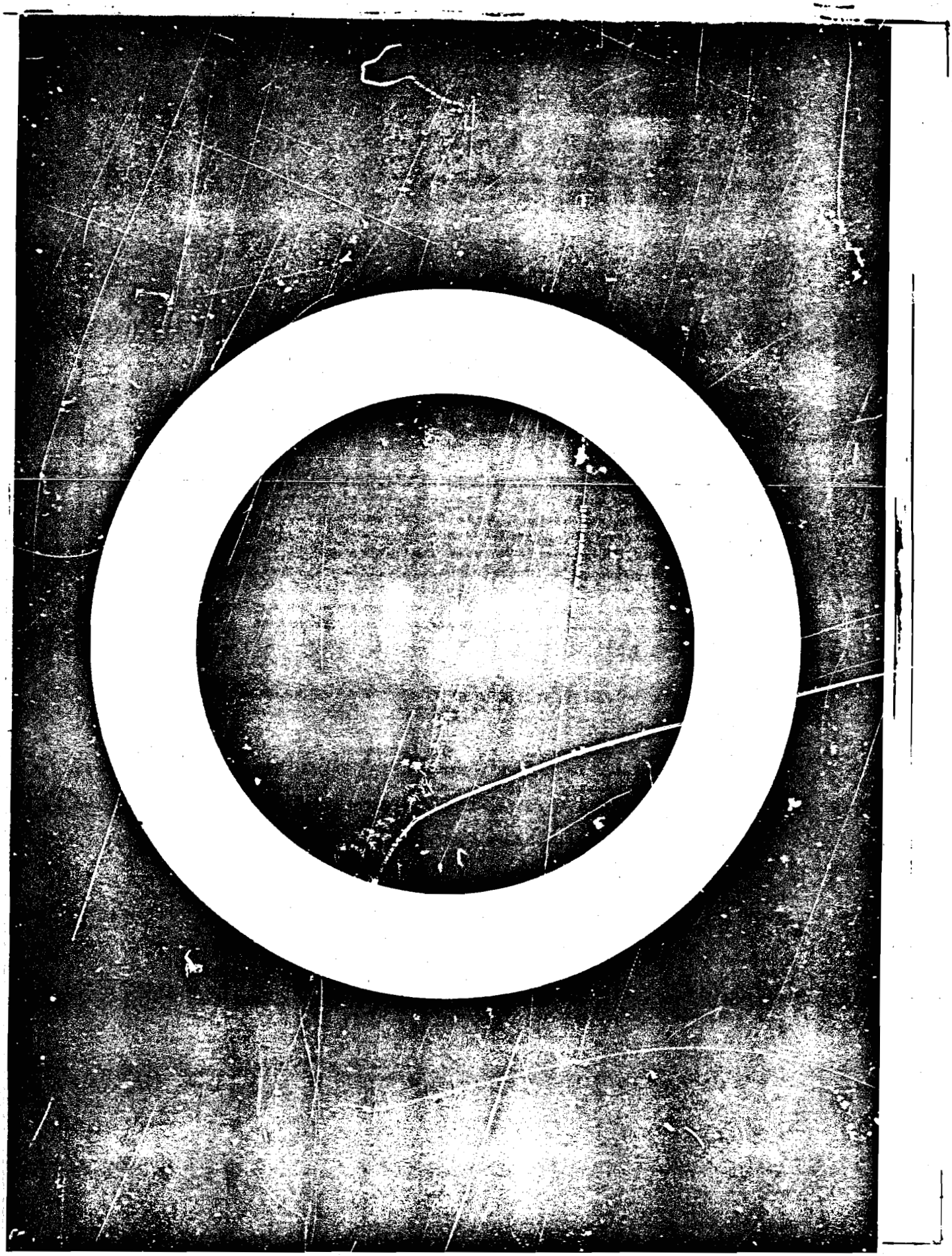
Summary

Of the siting alternatives shown in fig. 11-1, alternative 4 was chosen for presentation given in this report.

But it is possible that future studies will show that alternative 3 is preferable. Activities at the central fuel storage facility and the intermediate storage facility for waste cylinders are of the same character and considerable savings can be made at an early stage by coordinating service facilities and personnel. This will be especially true if there is little reprocessing of Swedish nuclear fuel in the near future. In this case, only a few waste cylinders will have to be disposed of in Sweden during an initial phase and the separate location of an intermediate storage facility would be less attractive.

11.4 COORDINATION WITH FACILITIES FOR OTHER TYPES OF RADIOACTIVE WASTE

In keeping with the provisions of the Conditions Act, KBS has only dealt with high-level waste from nuclear fuel. Methods and facilities for disposing of medium- and low-level waste are currently being studied by PRAV. In designing the requisite facilities, optimum coordination of the handling of all types of radioactive waste should be considered.



12 LAWS, STANDARDS, AND GROUNDS FOR EVALUATION

12.1 GENERAL

In general, very stringent safety rules are applied to nuclear power activities, as a result of which the risks to human health and the environment associated with nuclear power are very low. This high level of safety is a direct consequence of the extensive work on criteria and standards aimed at creating safe designs and the rigorous radiation protection recommendations for plant personnel and the general population issued by the International Commission on Radiological Protection (ICRP), whose principles have been accepted in all countries.

The work of other organizations, such as the World Health Organization (WHO) and the International Atomic Energy Agency (IAEA), as well as national authorities, has also contributed towards the safe and environmentally hygienic peaceful utilization of nuclear power.

Loss liability in the area of nuclear energy is regulated in most western Eueopean countries by the Paris Convention. The Paris Convention and its supplementary convention are complemented by the Brussels Convention, which regulates loss liability associated with the transport of nuclear fuel by sea. In Sweden, such liability is regulated by the Atomic Liability Act of 1968 (SFS 1968:45). which places the liability for an atomic accident with the plant owner. However, the plant owner's liability is limited to SKr 50 million per accident. Sums in access of this are paid by the State up to SKr 350 million, and after that up to approximately SKr 600 million by the states which have ratified the Brussels Convention.

Nuclear activities in Scandinavia must also comply with the Nordic Environmental Protection Convention. This convention was ratified by Sweden in 1976.

The dumping of wastes, including radioactive waste, in the sea is regulated by the London Convention. The Swedish Parliament, however, has prohibited the dumping of Swedish radioactive waste at sea al ogether.

12.2 LAWS AND REGULATIONS

In Sweden and other countries, nuclear activities are regulated by a number of mutually complementary laws aimed at ensuring

safety and protection for plant personnel, the general population and the environment. The central law within the field of nuclear power in Sweden is the Atomic Energy Act (SFS 1956:305), which requires permission from the Government or an authority appointed by the Government for the erection and operation of nuclear power plants or plants for the processing of nuclear fuel.

The supervisory authority designated by the Atomic Energy Act is the Swedish Nuclear Power Inspectorate, which is responsible for such matters as examining the safety of nuclear facilities and the design of various safety systems. Fissionable material safeguards and permits for the transportation of fissionable material also come under the authority of the Inspectorate.

The Radiation Protection Act contains provisions governing work with ionizing radiation, among other things. Permission for such activities is required from the inspection authority, the National Institute of Radiation Protection. Conditions and directives governing such activities are also issued by this authority. Matters pertaining to radiation protection in both the working environment and the external environment are also regulated by the Institute, which issues directives concerning:

- Maximum permissible occupational doses and the measurement and reporting of such doses.
- Maximum permissible releases of radioactive substances and how such radioactivity is to be measured and reported.
- Environmental monitoring with sampling and analyses of different types of samples as well as direct measurements.

Other Swedish laws which govern nuclear activities are:

- Environmental Protection Act
- Workers' Protection Act
- Building Act
- Emergency Planning Act

12.3 INTERNATIONAL RECOMMENDATIONS

International organizations such as the International Commission on Radiological Protection (ICRP), the United Nations' International Atomic Energy Agency (IAEA), the World Health Organization (WHO) and the OECD's Nuclear Energy Agency (NEA) agree on the following fundamental principles:

- No activity which entails the irradiation of personnel or population shall be accepted unless it can be demonstrated that its advantages outweigh its disadvantages from the viewpoint of society.
- The activity must be acceptable with respect to radiation risks.
- All radiation doses shall be kept within limits which are regarded as reasonable on the basis of economic and social considerations.
- No individual shall be subjected to radiation doses which exceed the dose limits recommended by the ICRP, either now or in the future.

The most recently updated edition of the ICRP recommendations

(12-2) was published in September of 1977. The intentions of this and previous relevant publications from the ICRP served as a basis for the work of this project.

The recommendations concerning maximum permissible radiation doses have not been changed. Thus, the following limits apply:

- Radiation dose to personnel in radiological work, 5 rems per year.
- Radiation dose to individuals in the population, 0.5 rems per year.

The term "weighted whole-body dose" has been introduced. The purpose of this is to weigh together all doses to different organs to arrive at a representative whole-body dose.

The term "dose commitment" has been introduced. By dose commitment is meant the sum of the annual radiation doses which are the results of releases over one year. This means that the annual radiation dose in a future postulated state of equilibrium is equal to the dose commitment from the releases over a single year. Dose commitment can also designate the total dose load from accidental releases.

The term "collective dose" refers to the sum of all of the individual doses within a given population. The purpose of setting a limit for the collective dose is to limit the future mean dose - and thereby the number of injuries - from a fully expanded nuclear power industry.

12.4 SWEDISH RADIATION PROTECTION STANDARDS AND CRITERIA

New regulations governing the release of radioactive substances from nuclear power plants were adopted by the government in 1977 on the basis of proposals from the National Institute of Radiation Protection (12-3). These regulations will enter into effect in 1981. Transitional regulations will apply until then.

The new regulations stipulate limits for whole-body doses to nearby residents and collective doses to the entire population. The values which are specified are:

- The sum of the weighted whole-body dose to nearby residents should be less than 10 mrems per year.
- The global weighted collective dose commitment should be less than 0.5 manrems per year and MW installed electrical output (MWe).

These limits are considerably more stringent than former limits. They have been determined on the basis of an evaluation of what is currently the lowest dose load which can reasonably be achieved.

If these requirements are fulfilled, acute injuries to any individual will be entirely eliminated. The safety margin to direct health effects is several orders of magnitude.

The limits stipulated by the new standards aim at minimizing the risk of delayed effects, both somatic and genetic. The reference

value of 10 mrems per year contributes less than 10% to the normal radiation environment.

The regulations governing releases of radioactivity from nuclear power plants also include provisions concerning:

- Countermeasures in the event of elevated release levels.
- Routines for inspection and reporting.
- Scope of environmental studies.

The transport regulations of the International Atomic Energy Agency, Regulation for the Safe Transport of Radioactive Materials (IAEA Safety Series No. 6), govern the transport of spent nuclear fuel and other radioactive material. There are also Swedish and international rules governing transports via various means. The supervisory authorities are the Nuclear Power Inspectorate and the National Institute of Radiation Protection.

12.5 DESIGN STANDARDS

Special rules for nuclear facilities other than nuclear power plants - such as central storage facilities for spent fuel, final repositories and intermediate storage facilities for high-level waste and encapsulated spent fuel - have not yet been issued in Sweden. But the general principles established for nuclear power plants should also be applicable to the safety measures and the evaluation of safety and environmental aspects in these areas. Some modifications may be necessary depending on the nature of the facilities and processes. General protection principles have already been established for the transportation of spent fuel and for the storage and treatment of such material.

Some design standards concerning the storage of spent nuclear fuel and radioactive waste have been established in the United States as well as in West Germany. These standards apply to temporary storage. Standardization work is currently underway in these countries concerning the later stages of the nuclear fuel cycle, including the final storage of high-level waste.

In the USA, the Nuclear Regulatory Commission (NRC) is pursuing a broadly-based programme aimed at establishing standards and licensing requirements with regard to the location, design and management of facilities for radioactive waste. This programme includes criteria aimed at ensuring the safety of the environment and of the personnel in the handling, transport, storage and final disposal of vitrified high-level waste. The initial results of this work are expected to be published during 1978. Work is also currently under way on design criteria for storage rooms for high-level waste. These criteria are expected to deal with the following points:

- Quality assurance measures for design and construction.
- Performance requirements for containment barriers.
- Requirements on nuclear safety.
- Compatibility between waste forms and containment media.
- Protection against mechanical damage.
- Security requirements.

These pending licensing requirements in the area of nuclear waste

have been given the working title 10 CFR 60 "Licensing of Radioactive Waste Management Facilities". More detailed design directives in this connection are expected to be presented by the NRC in Regulatory Guides.

A more thorough description of the situation concerning standardization work in the USA is provided in a KBS report /12-1/.

12.6 GROUNDS FOR EVALUATION WITH REGARD TO FINAL STORAGE

Safety criteria for final storage have not yet been established, but work is being pursued in this area in many different countries and in international cooperation. One of the requirements is that the long-term environmental load must be acceptable.

As regards the final repository, special attention must be paid to the ICRP's rule that no individual, either now or in the future, shall receive radiation doses which exceed the dose limits recommended by the ICRP. The current limit for individuals is 500 mrems/year from all activities which can give rise to radiation with the exception of medical uses of ionizing radiation. On the basis of considerations of what is technically feasible and economically reasonable on the one hand and improved protection on the other, national regulations for nuclear power plants have been issued which stipulate 10-50 mrems/year for nearby residents.

Thus, the release of radioactive substances from a final repository shall not give rise to more than a fraction of 500 mrems/year and person to nearby residents for all future time. Beyond this, the usual rule that all measures which are socially and economically acceptable shall be adopted to reduce the dose load shall apply.

In order to protect large population groups against genetic effects in the long run, a rule concerning the limitation of collective doses similar to the one which is currently applicable to nuclear power plants should also be applied to waste and fuel management facilities.

In this connection, Sweden and the other Nordic countries are campaigning for the adoption of a rule which would specify a dose limit per unit of electrical output for nuclear power as a whole, namely 1 manrem/MWe per year (12-4). Since 0.5 manrem/MWe per year has been allocated for the operation of nuclear power stations, 0.5 manrem/MWe remains for the other parts of the fuel cycle, including the final repository. All dose loads during the entire nuclear power era shall thereby be taken into consideration, whereby long-lived elements shall be totalled over 500 years. In the case of extremely longlived radioactive elements, the annual radiation doses shall remain low in relation to the natural radiation level. The choice of a level of 1 manrem/MWe is based on the goal of a maximum of 10 mrems/year and person and the assumption of an average global electrical power production from nuclear energy of 10 kW per person. This greatly exceeds the present total power consumption per inhabitant in the industrial countries and entails a large safety margin. As a comparison, it can be mentioned that electrical power consumption per person is

highest in Norway, with an average of more than 2 kW per year (1975). The corresponding figure for Sweden is 1.1 kW.

12.7 MANKIND'S CURRENT RADIATION ENVIRONMENT

Radioactive elements exist in nature and ionizing radiation from these elements is responsible for a portion of the natural background radiation level. Additional irradiation of man stems from a number of different sources, for example from building materials in residences and from the medical use of radiation.

The background radiation occurring in nature comes from cosmic radiation, radiation from radioactive elements in the bedrock and radiation from radioactive elements absorbed in the body. The natural background radiation level in Sweden is between 70 and 140 mrems/year (12-5). The body's natural absorption of potassium-40, uranium, thorium and radium along with their daughter products gives an average dose of 20 mrems/year in Sweden.

The radiation dose from building materials in buildings varies widely. Radiation doses between 20 and 200 mrads*/year in connection with uninterrupted presence indoors for the whole year are common (12-6). Values up to 700 mrads/year have been measures as external doses (12-7). The weighted internal whole-body dose caused by radon in our homes is between 10 and 1 000 mrems/year (12-8). Medical irradiation provides an extra annual dose of approximately 40 mrems per person.

Most products in our environment, both natural and manufactured, are weakly radioactive. Drinking water in Sweden contains, for example, radium-226 at levels which vary between 0.1 and 40 pCi/l (12-9). On the basis of the same calculating principles used for the final repository, this gives doses between 1 and 400 mrems/year. Natural waters also contain uranium at levels which are normally between 0.1 and 5 pCi/l, but extreme values of up to 1 500 pCi/l have been measured (12-10).

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* The absorbed dose is given in rads, while the biologically weighted dose is given in rems. Except in the case of alpha radiation, the rad and rem values are numerically equivalent.

13 SAFETY ANALYSIS

13.1 GENERAL

The handling chain for spent nuclear fuel and vitrified high-level waste described in chapters 3 to 9 has been thoroughly analyzed as regards safeguards against the dispersal of radioactive substances. The emphasis has been placed on those processes which are intended to be carried out in Sweden. Reprocessing and vitrification will be done abroad, and these stages have not been analyzed in detail by the project. For all of the other processes, including all transport operations, an analysis has been carried out in an attempt to assess the normal releases of radioactive substances which might occur and the probability and consequences of releases in connection with accidents.

Final storage has been analyzed with respect to the different phenomena which can contribute towards the slow dispersal of radioactive substances. The possibilities of various extreme events in connection with final storage have been surveyed.

In order to provide a basis for determining whether handling and final storage can be effected in an absolutely safe manner, a comparison is made with the grounds for evaluation described in chapter 12.

This chapter summarizes the detailed report on the safety analysis which is submitted in volume IV.

13.2 SAFETY IN CONNECTION WITH HANDLING, STORAGE AND TRANSPORTATION

13.2.1 Handling stages and methods

Safety in connection with the handling, storage and transportation of spent fuel and vitrified high-level waste is described in a brief summary. A more detailed treatment is provided in chapter IV:4.

The following handling stages are dealt with (see figure 13-1):

- Transportation of spent fuel elements from the reactor station to a central storage facility.
- Reception and storage in the central storage facility for 10 years.
- Discharge of spent fuel from the central storage facility

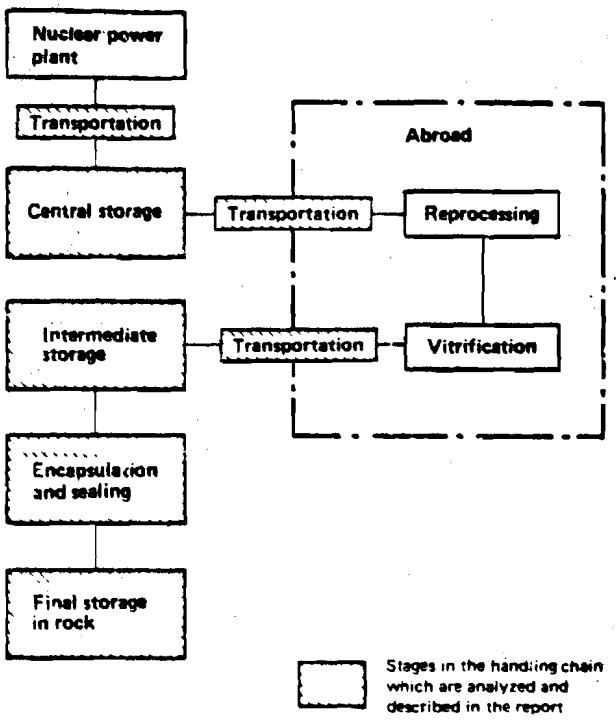


Figure 13-1. Handling chain for storage of spent fuel and vitrified waste.

and transportation to a foreign plant for reprocessing, where the high-level waste is vitrified.
- Transportation of vitrified waste back to domestic intermediate storage facility.
- Storage of waste for 30 years in intermediate storage facility.
- Encapsulation of vitrified waste in canisters of titanium and lead.
- Deposition of the encapsulated waste in a rock repository approx. 500 m down in the bedrock.

Personnel and the environment may be subjected to radiation in connection with both normal operation and incidents. Shielding and other normal radiation protection measures shall be instituted and implemented to such an extent that dose loads are limited in compliance with the recommendations and requirements of the ICRP and the National Institute of Radiation Protection. Protection for personnel engaged in the operation and maintenance of the facilities is dealt with in chapter III:7.

In the safety analysis, various safety precautions have been studied and normal radioactive releases have been evaluated. The analysis of accidents or failures has dealt with failures which can damage one or more of the barriers which protect against the escape of radioactive substances.

The failure analysis assesses the release of radioactive substances in connection with various failures and accidents and the probability that such inadvertent releases will occur.

The radiological consequences are described in the form of the individual dose for the most-exposed population group and the collective dose commitment, which is a measure of the long-term dose load on the population.

Adverse environmental effects due to accidents at plants and during transportation and from normal releases are judged to be much less serious than in the case of nuclear power plants, both as regards probability and consequences. This is due to the fact that pressure and temperature are lower and that the radioactive substances are constantly encased and isolated from the environment. The chances of a sudden and heavy release of radioactive substances are nill or negligible. Furthermore, such events develop much more slowly and allow more time for countermeasures.

13.2.2 Radioactive substances in spent fuel

Immediately after reactor shutdown, power generation in the fuel elements declines sharply, but some heat continues to be generated as a result of the decay of the radioactive elements formed during reactor operation. One minute after shutdown, heat generation has dropped to 5% of operating output and then continues to decrease rapidly. After a month, it is around 0.1%. Radioactivity declines at roughly the same pace.

Transportation of spent fuel from the reactor to a central storage facility takes place at the earliest 6 months after discharge and normally even later. The calculations in the safety analysis are based on the transport of one-year-old fuel to the central storage facility.

In the transport safety analysis, cesium-134 and cesium-137 are the main sources of radiation doses in the event of a failure. However, the only radioactivity which is available for leakage is that which has been released from the ceramic fuel and has accumulated in cavities inside the canning. This portion has been assumed to be 1% for cesium, which available information indicates is on the pessimistic side. Burn up has been assumed to be 33 000 MWd/ton uranium.

13.2.3 Central storage facility for spent fuel

The safety analysis was carried out for the type of central storage facility for spent nuclear fuel which is described in a preliminary study conducted by PRAV. Certain modifications may be made in the follow-up study which is currently being conducted within SKBF.

The facility is designed with a special emphasis on:

- Keeping the dose load on the personnel as low as possible.
- Preventing the release of radioactive substances which could escape to the environment.

The radiological safety of the personnel is guaranteed by a number of safeguards, such as:

- Amply dimensioned radiation shields in the form of e.g. concrete walls and water barriers.
- Monitoring instruments for checking direct radiation and airborne activity.
- Remote control of active components and systems.
- Dose monitoring of personnel.

The storage pools are located underground with a 30 m rock cover, which provides the stored fuel with an effective barrier against external forces.

The ventilation systems are sized to guarantee acceptable temperature and humidity. The areas for fuel discharge and other areas where there is a risk for airborne radioactivity are maintained at a lower pressure than the surrounding parts of the building. Radioactive substances are collected in filters before the ventilation air is discharged to the atmosphere.

The storage pools will be designed as free-standing thick-walled units lined with stainless steel. The external walls of the concrete structure are accessible for inspection and any leakage can be diverted via collection ducts behind the welded joints in the stainless steel plate and collected in a drainage system. This permits leakage to be detected at an early stage.

The pools do not have any low-level pipe connections. This prevents inadvertent emptying of the pools. The cooling system is designed for maintaining the water temperature at 25-30°C normally and below 50°C in the event of isolated component failures. If external cooling should fail completely, despite redundant components and a back-up power system, the temperature of the water in the pool will rise. If no countermeasures are adopted, it will take more than a week before the pools reach the boiling point. But the fuel can be cooled by make-up feed, so the fuel will not be damaged due to exposure.

The configuration of the fuel in the tanks has been analyzed for criticality. The calculations show a good safety margin to criticality, even for unirradiated fuel. It is improbable that the fuel will be redistributed due to an accident to a configuration which possesses higher reactivity than the normal storage configuration. Since the fuel in the central storage facility is spent, there is no risk for a critical configuration.

In order to minimize the fire hazard, the facility is divided into fire cells and equipped with an automatic fire alarm system, fire ventilation and fire extinguishing equipment adapted to the nature of the different areas. The fire load is low throughout.

The risk of rock collapse can be aliminated by means of conventional construction methods, such as rock reinforcement.

The 70-110 ton cask for spent fuel is transported from the harbour to the reception section on a trailer towed by a tractor vehicle. In order to reduce the risk of transport accidents, transport speed has been limited to 10 km/h. The grade in the tunnel is a maximum of 1:10 and long straightaways have been avoided.

The transport cask is moved horizontally within the reception section by means of an overhead crane over a reinforced floor section. Where vertical movements are required, lifting heights are limited as far as possible and the overhead cranes are equipped with extensive safety devices.

Many years of experience has been gained in the storage of spent fuel in water pools. There is normally some small leakage of radioactivity from the fuel to the water in the pool. This is disposed of by cleaning circuits in the same way as in nuclear power plants. Small quantities of krypton-85 and tritium are entrained in the ventilation air and released to the atmosphere. Traces of iodine and particulate-activity may also be released to the atmosphere. But most remains in the water and is collected by the ion-exchangers in the cleaning circuits.

Normal releases of radioactivity to the environment are very small and give rise to insignificant radiation doses to nearby residents (on the order of 0.0001 rems/year).

The central storage facility has been designed to reduce the probability of major failures to a very low level. The accidents which could nevertheless conceivably occur are restricted to minor incidents with moderate releases of activity.

The following accidents have been analyzed with respect to consequences and probability:

- Transport cask dropped
- Fuel cassette or other object is dropped
- Fuel element is dropped

These accidents would entail a release of max. 4 000 Ci krypton-85, which gives radiation doses less then 0.1 millirem. The probability of the maximum releases has been estimated to be about 0.0004 per year.

A central storage facility for spent nuclear fuel thus entails negligible radiation risks to the environment.

13.2.4 Transportation of spent fuel and vitrified, high-level waste

A description of the transportation system is provided in chapter I:9. Type B containers are used for the transports. These containers comply with the rules issued by the IAEA (International Atomic Energy Agency in Vienna) for design, construction and testing. The aim of these rules is to ensure that the transport cask for the radioactive material is designed to provide sufficient security against the escape of radioactive substances as a result of accidents. The intention is that all transports shall be possible with the use of conventional, generally available vehicles and without any special radiation surveillance. The rules therefore prescribe the following categorical tests, which are designed to simulate a serious accident:

- Fall from 9 m onto a hard, flat surface.
- Fall from 1 m onto a solid steel cylinder with a diameter of 15 cm.

- Heating for 30 minutes to 800°C.
- Immersion in water at a depth of 15 m for 8 hours.

The first three tests shall be carried out in sequence on the same cask. The cask shall pass these tests without any leakage. Specified requirements are also made on radiation shielding and cooling.

The cask shall be handled in accordance with detailed instructions. Transports outside of the plant area must be registered with and approved by the authorities in the concerned countries in advance.

Safety can be further enhanced by the use of specially-made ships. The ship which is planned for the purpose shall be equipped with special safety-enhancing equipment:

- Reinforcement for travel through ice.
- Monitoring equipment for radiation.
- Carbon dioxide system for fire extinguishing.
- Automatic sprinkler system.
- Device for tracking after sinking.
- Extra communication equipment.

Observations after actual transport accidents as well as special full-scale collision tests show that the casks can actually withstand greater stresses than those specified in the testing directives. But in order to investigate the potential environmental consequences of various transport accidents and failures, it has been assumed that the casks can be damaged in rare cases.

There are three different types of transports between plants:

- Transportation of fuel elements from the reactor station to the central storage facility.
- Transportation of fuel elements from the central storage facility to a foreign reprocessing plant.
- Transportation of vitrified waste from a foreign reprocessing plant to a Swedish intermediate storage facility.

The types of accidents and failures which have been considered are:

1 The transport cask is dropped in connection with loading or unloading.
2 The ship runs aground or sinks.
3 Ship collision.
4 Long-lasting fire onboard.
5 Collision and fire onboard.
6 Trailer collision with and without fire.

The most severe case is a ship collision with fire onboard. The probability of such an accident has been calculated to be 3×10^{-6} per year (three such accidents every million years). Some cesium and krypton-85 could then be released. The maximum collective dose to the population in the event of such an accident has been calculated to be about 30 000 manrems. This assumes that the accident takes place near a major city. Thus, the probability of this worst consequence is considerably lower than the actual average value given for this type of accident.

The safety analysis shows that transport of spent nuclear fuel entails very small risks for the escape of radioactive substances. Even in the event of such an improbable event occuring, the consequences of any release of radioactive substances would be slight.

13.2.5 Intermediate storage, encapsulation and deposition of vitrified waste

The intermediate storage facility is designed on the basis of experiences from the Marcoule plant in France. Design criteria similar to those for the central fuel storage facility have been used. Safety requirements played an important role in determining the design of the plant.

The emplacement of the intermediate storage facility in rock provides good protection against external forces, acts of war etc.

The waste cylinders are thoroughly decontaminated from surface activity but must be radiation-shielded in order to permit handling. The concentration of radioactive waste in the glass is approx. 9%. The waste has cooled for about 10 years before it is transferred to the intermediate store. As a result, heat flux has declined to about 1.2 kW/cylinder. The waste cylinders arrive protected by the transport cask. The waste cylinders are handled in closed rooms with thick radiation-shielding walls, constant underpressure and instrumental monitoring.

Transfer casks with built-in radiation shielding are used to move the waste cylinders. They are handled by overhead cranes and the cylinders are transported very close to the floor. The most important safety precaution is to maintain cooling of the storage section. Normally, cooling is provided by two fans connected in parallel. A third fan can be turned on when needed. There is also a fourth fan on the surface. A diesel generator provides an auxiliary power supply. Normally, the exhaust air temperature is about 80°C. If one fan fails, the back-up fan is automatically switched on. When only one fan is used for cooling, the temperature rises to about 110°C after 40 hours. If all fans should fail in an extreme case, a by-pass duct with an automatic damper permits air to circulate with natural convection. The temperature then rises to max. 340°C, which is attained after 40 hours. This does not lead to any release of activity.

The steel casing around the glass body is carefully cleaned during fabrication and all handling of waste cylinders is performed under dry conditions, so the risk for surface contamination is small. Since the waste is incorporated in the glass, which is in turn enclosed in tightly welded cylinders, no radioactivity is released during the storage period. Ten years of experience in the storage of high-level glass is available from Marcoule plant in France. During this time, no radioactivity has been detected in the ventilation filters.

The probability of failures in the plant has been limited by design precautions. Overheating to such high temperatures that volatile nuclides are vapourized does not occur even in the event of a total cooling failure as described above. The probability of

damage due to fire is reduced by a low fire load and adequate fire protection.

The possibility of mechanical damages which could lead to the release of airborne activity is deemed to be non-existent.

After approximately 30 years of storage, the waste cylinders are lifted up out of their storage positions and transferred in a transfer cask to a cell for encapsulation. The encapsulation section contains a number of work and inspection stations in a radiation-shielded cell. A lead shell with a titanium casing is placed over the cylinder and lead is poured into the space between the shell and the steel casing around the glass. The canister is then sealed with a titanium lid which is tightly welded. The encapsulation procedure is carried out in its entirety by means of remote control. The encapsulated waste is lifted into a transfer cask which is transferred on a wagon to the repository's elevator shaft.

The elevator which carries the encapsulated waste down to the final repository is equipped with at least two independent brake systems and the hoist cables can carry 10 times the design load. If an elevator should fail, the fall will be damped by the water pool at the bottom of the hoist shaft. Even if the canister should be damaged, the glass cannot shatter into particles of such small size that they can become airborne.

13.3 RADIOACTIVE SUBSTANCES IN HIGH-LEVEL WASTE GLASS

13.3.1 General

The core of a light-water reactor contains uranium fuel in the form of uranium dioxide enclosed in "cans" made of a zirconium alloy (nuclear fuel). The amount of uranium in a reactor core varies with the size and type of the reactor. The Swedish reactors have cores which contain between 70 and 126 metric tons of uranium. A boiling water reactor (BWR) has a lower power density and a larger quantity of fuel at a given size than a pressurized water reactor (PWR). The nuclear fuel is gradually replaced in connection with the annual shut-downs for fuel replacement and other maintenance. Each fuel element is irradiated between 3 and 5 years before it achieves full burnup. This is 25 000 - 28 000 megawatt-days per ton uranium (MWd/tU) for BWRs and 31 000 - 35 000 MWd/tU for PWRs. The typical composition of the spent fuel is specified in section 2.1.1.

During the burnup of the uranium fuel, radioactive waste is created by nuclear fission, which produces fission products, and by neutron capture, which produces activation products. The most important activation products in the fuel are elements which are heavier than uranium - the transuranium elements - neptunium (Np), plutonium (Pu), americium (Am), curium (Cm) etc. Other activation products are formed in the cans and structural components. These latter activation products constitute medium-level waste and are not discussed further here.

The high-level waste obtained from the reprocessing of the spent fuel contains most of all the fission products plus all of the

transuranium elements except for plutonium and small residual quantities of uranium and plutonium. Reprocessing and vitrification of the high-level waste has been described in chapter 5. In reprocessing, certain gaseous fission products are separated, mainly krypton-85 and iodine-129, but all others are present in the high-level waste.

13.3.2 Composition of high-level waste

As was mentioned above, the vitrified high-level waste contains almost all of the fission products and transuranium elements except for plutonium. The residual quantities of uranium and plutonium left in the waste glass may vary depending on the type and burnup of the fuel and on the detailed nature of the reprocessing process. Just as in the Aka Committee report (13-1), it has been assumed that 0.1% of the uranium and 0.5% of the plutonium in the spent fuel will be present in the high-level waste.

Recent French findings indicate that the uranium content can be expected to correspond to approximately 0.2% of the original uranium and the plutonium content to about 0.15% of the original plutonium. As noted in chapter IV:3, doubling the uranium content leads to an insignificant increase in calculated activity dispersal, while a reduction of the plutonium content to 30% leads to a considerable reduction of activity dispersal.

It should be noted that separation of plutonium from uranium and the precipitation and conversion of these elements entail further losses of uranium and plutonium. These lost quantities are present in the low- and medium-level waste and are normally slightly greater than the quantities in the high-level waste. According to guarantees from the reprocessing company, total losses will not exceed 3% plutonium and 2% uranium. Lower losses have been obtained in operating plants.

The most important fission products in the high-level waste as far as final storage is concerned are those with long and very long half-lives. The following table gives the half-lives of the important nuclides.

Nuclide	Half-life
hydrogen-3 (tritium)*)	12.3 years
selenium-79	65 000 years
krypton-85*)	10.8 years
strontium-90	28.1 years
zirconium-93	1.5 million years
technetium-99	210 000 years
antimony-126	100 000 years
iodine-129*)	17 million years
cesium-135	3 million years
cesium-137	30.0 years

*) tritium, krypton and iodine are removed in connection with reprocessing. However, it has been assumed in the safety analysis that 1% of the iodine-129 is present in the vitrified waste.

Nuclide	Halflife
prometium-147	2.62 years
samarium-151	87 years
europium-154	16 years

Of the nuclides listed in the table, strontium-90 and cesium-137 are the most important for evaluating safety during the first 500 to 1 000 years of final storage. For very long-term safety, technetium-99, iodine-129 and cesium-135 are of safety interest.

The residual quantities of uranium and plutonium, the other transuranics and the decay products of these elements comprise the heavy nuclides. The waste contains a hundred or so different heavy nuclides. A detailed list of these is provided in volume IV, chapter 3. The following table gives the heavy nuclides which are most important for evaluating the safety of the final repository.

Nuclide	Half-life	Parent nuclide
americium-243	7 650 years	
americium-241	433 years	plutonium-241
plutonium-241	14.6 years	
plutonium-240	6 760 years	
plutonium-239	24 000 years	
plutonium-238	89 years	
neptunium-237	2.13 million years	americium-241
uranium-238	4 510 million years	
uranium-235	710 million years	
uranium-234	247 000 years	plutonium-238, uranium-238
uranium-233	162 000 years	neptunium-237
thorium-230	80 000 years	uranium-234
thorium-229	7 300 years	uranium-233
radium-226	1 600 years	thorium-230

The column headed "parent nuclide" indicates that the nuclide in question is formed by radioactive decay of the parent nuclide. Thus, successive decay of plutonium-241 leads to the formation of americium-241, neptunium-237, uranium-233 and thorium-229.

The activity of radioactive elements in curies per ton uranium as a function of time is illustrated in figure 13-2. The unit "curie per ton of uranium" is roughly the same as curie per waste cylinder in this case.

The fission products cesium-137 and strontium-90 heavily dominate during the first 100 years. Technetium-99 then dominates among the beta-emitting fission products for over a million years, after which it is followed by cesium-135. Of the alpha-emitting heavy nuclides, americium-241 and -243 dominate in the beginning, followed by a period of plutonium-239 dominance and finally thorium-229 dominance. Since alpha-radiation is more dangerous than beta-radiation on a per-curie basis, the toxicity of the waste after approximately 300 years will be determined primarily by the latter heavy nuclides.

The quantities give in figure 13-2 apply for PWR fuel which has

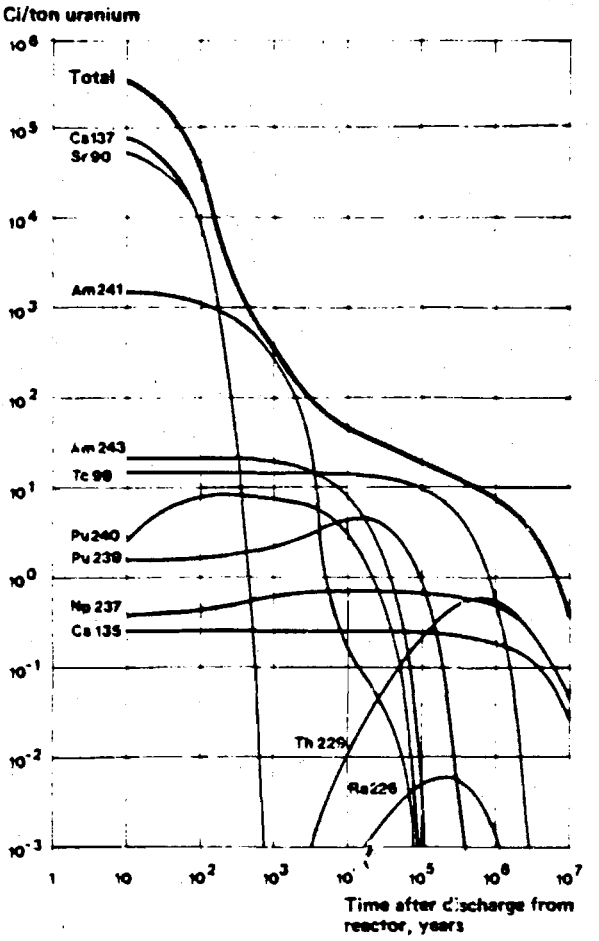


Figure 13-2. Radioactive elements in high-level waste. It is assumed that reprocessing takes place ten years after discharge of the spent fuel from the reactor.

been reprocessed ten years after full burnup has been acheived and the fuel has been discharged from the reactor. This time is of some importance for the activity of the high-level waste. Plutonium-241, which is a beta-emitter, decays with a half-life of 14.6 years to americium-241. The longer the wait for reprocessing, the more of this nuclide will be present in the waste. If the waste is only stored for three years before reprocessing, the amount of americium-241 given in figure 13-2 will decrease to about 40%. If a much longer time passes before reprocessing, the amount of americium-241 will increase to roughly double the value given in the figure. The daughter nuclide neptunium-237 is not as greatly affected, since a significant portion of this nuclide is formed directly in the reactor via other nuclear reactions.

The amount of plutonium formed in BWR fuel is somewhat lower than in the case of PWR fuel, and the portion of heavy nuclides is somewhat lower for the same amount of fission products.

It has been assumed throughout the safety analysis that the high-

level waste comes from PWR fuel which has been reprocessed ten years following discharge from the reactor and irradiated to 33 000 MWd/tU.

13.3.3 Decay heat in high-level waste

Figure 13-3 shows the decay heat in spent fuel and high-level waste obtained from reprocessing after two and ten years, respectively. The decay heat is given in watts per ton uranium, which is equivalent to watts per waste cylinder for the high-level waste. The curves are plotted for a residual uranium level of 0.5 %, but the other parameters are the same as in the preceeding section.

13.4 SAFETY IN CONNECTION WITH FINAL STORAGE

13.4.1 General background

In order to achieve a safe final storage of the high-level waste, the radioactive substances are enclosed in a number of consecutive barriers:

- chemical bonding to the low-soluble borosilicate glass
- encapsulation of the glass in a number of metal casings
- storage of the encapsulated waste cylinders in good rock at a depth of 500 m.

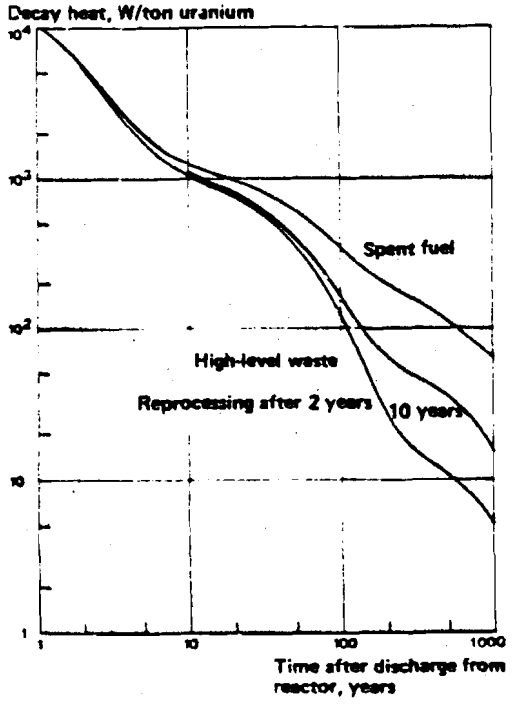


Figure 13-3. Decay heat in spent fuel and high-level waste from pressurized water reactor.

Each one of these barriers provides protection against dispersal. But each one possesses different protective properties and thereby also different protective functions which both reinforce and complement one another.

In the final repository, the waste cylinders are deposited in boreholes on the floor of the tunnel. A buffer mass between the canister and the rock keeps the waste cylinder fixed in position. Escape and dispersal of the radioactive substances is also retarded by sorption effects in the buffer mass and in fissures in the rock in the event the radioactive substances should penetrate through the inner barriers.

Requirements on isolation of the radioactive substances diminish as their radioactivity decays. Only a combination of flowing water and penetrated barriers can lead to a dispersal of radioactive substances from the final repository.

A distinction can be made between:

- Slow processes.
- Extreme events which lead to sudden dispersal of radioactive substances.

Slow dispersal of the radioactive substances is discussed in detail in volume IV, chapter 6, so only a brief summary will be provided here. The probability of extreme events which can break through the rock barrier and cause rapid dispersal of the radioactive substances is extremely low. The most important cases and consequences thereof are discussed in volume IV, chapter 7, and summarized here.

13.4.2 Factors involved in the slow dispersal of radioactive substances

The analysis of the slow dispersal of radioactive substances entails a series of calculations pertaining to various phenomena. In order for the results of the final calculation to reflect the most unfavourable situation which can be postulated, the assumptions and data used in the various stages of the calculation must be chosen with a considerable margin of safety. As more reliable data become available, these margins can be reduced. In the analysis reported here, a series of large safety margins have been stacked on top of each other. The calculated final result therefore provides a picture which is probably several orders of magnitude less favourable than what might actually be the case.

In order for the slow dispersal of radioactive substances to be possible at all, the metallic canisters around the glass bodies must be damaged in such a way that water comes into contact with the waste glass. If this happens, a slow leaching of the glass begins.

Canister damage

Two cases of canister damage have been studied in the safety analysis: a) a canister is damaged at the time of deposition, and b) the canister is slowly corroded.

The metallic canisters will be fabricated under very rigorous quality control. The canisters will also be inspected after fabrication for both leakage and weld flaws.

In the industrial fabrication of metallic components under high standards of quality control, defect frequencies on the order of 1 in 10 000 to 1 in 100 000 have been achieved. The canister in question is made in three metallic layers - stainless steel, lead and titanium. This should even further reduce the frequency of damages which completely penetrate the canister. The probability of initial canister damage is therefore judged to be considerably lower than 1 per 10 000 canisters. The total number of canisters will be approximately 10 000.

One reference case which is studied is a damaged canister in the final repository. This single waste body is dealt with in the analysis as if it were unencapsulated and the entire glass surface were accessible for leaching. This case is therefore also equivalent to damage to a number of canisters with only part of the glass surface exposed. Cases involving more than one or two initially damaged canisters are deemed to be so improbable that they do not have to be considered.

The encapsulation materials have been selected with the intention of achieving very good durability and service life under the prevailing conditions. The properties of the canister materials are dealt with in section 6.3. The question of the service life of the canister is also discussed in a report issued by the Swedish Corrosion Research Institute (KI) and a specially appointed reference group.

Some of the members of this reference group estimated the period of time during which the canister is completely intact to be at least 1 000 years, while others estimated its life to be at least 500 years.

It is clear from the KI report that these estimates assume that local corrosion could be determining for the time during which a canister is completely intact. However, local penetrations of the canister after 1 000 (or 500) years do not necessarily mean that the glass body in its entirety is exposed to the groundwater and its leaching action.

It was not possible to perform a systematic analysis of how long a time it can be expected to take before the entire glass body is exposed after the capsule has been breached. Considering the time it takes for the general corrosion of titanium and the corrosion rate for lead in relation to the total amount of lead in each canister, it would probably take tens of thousands or hundreds of thousands of years. However, in lieu of more precise data, it is necessary to make an analysis on the basis of hypothetical cases which can be regarded as being more unfavourable than actual cases in order to be on the safe side.

As a reference case, it has been assumed that the canisters are broken down gradually during the period from 1 000 to 6 000 years following the commencement of final storage.

Calculations show that it is relatively insignificant for the overall risk assessment whether the service life of the canister

is taken to be at least 500 years or at least 1 000 years (see chapter IV:6.9.4).

Glass leaching

If water comes into contact with the waste glass, a very slow leaching of the substances in the glass takes place. The leaching rate depends on a number of factors, the most important of which is the composition and temperature of the glass, the surface area with which the water comes into contact and the water flow rate. Other factors which can be of importance are chemical conditions and changes in the structure of the glass due to ionizing radiation from radioactive substances or due to variations in the fabrication of the glass. (See also sections 5.4 and IV:6.3).

Experimental studies in France have established a leaching rate of 2×10^{-7} per cm^2 and day at a temperature of $25^{\circ}C$ for typical waste glass produced by the French method. At a temperature of $70^{\circ}C$, this leach rate increases by a factor of 10, and at $100^{\circ}C$ by a factor of 35. The temperature of the waste glass reaches a maximum of about $70^{\circ}C$ and declines after 2 000 years to about $30^{\circ}C$.

The geometric surface area of a waste body is about 2 m^2. However, cracks may form during the manufacture and handling of the glass. As a result, the surface area which is theoretically accessible to water is greater than the geometric surface area. Normally, the surface area is increased by a factor of about 2-3, but in unfavourable cases, surface enlargement may be by a factor of 10. As an average value for the leaching surface, 10 m^2 per glass body has been used in the safety analysis - i.e. 5 times the geometric surface area.

Using these values for leached surfaces and leaching rate, a leached fraction of 1.7×10^{-5} (17 millionths) per year and glass body weighing approx. 420 kg is obtained. This would result in a complete dissolution of the glass after about 60 000 years assuming that the dissolved weight per year remains unchanged. In the calculations which have been carried out for one of the reference cases, a glass leaching period of 30 000 years has been used.

During the first 150-200 years after commenced final storage, the temperature of the glass is considerably higher than $25^{\circ}C$. A glass leaching time of 3 000 years is therefore used for the case involving an initially damaged canister.

At final storage, the canister is emplaced, packed in a buffer mass, in rock through which the water flow rate is very low. Geological studies have shown that 0.1-0.2 litres per m^2 and year is probable. In such an environment, the leaching rate will be limited by the water supply. This makes it difficult to carry out a precise analysis of the leakage of radioactive substances from the glass. Preliminary estimates indicate leaching rates which are only about 1% of those given above.

Transit time for dissolved substances

The transit time of the radioactive elements through the rock

from the final repository to the biosphere is dependent upon two factors. One factor is the time it takes for the water to flow from the final repository to the primary recipient. This time varies widely depending on local conditions and the properties of the rock in question. Age determinations of groundwater indicate flow times of several thousand years from a suitably located final repository. To be on the safe side, a transit time for the groundwater of 400 years in impervious rock has been used in the calculations carried out for the safety analysis. (See section 7.4.)

The second factor which determines the transit time for the radioactive elements is the retardation effect obtained through chemical reactions between these elements and the buffer material and the rock material. Different types of chemical reactions occur, mainly ion-exchange processes, ion absorption, reversible precipitation and mineralization. These processes come under the general heading of sorption.

Mineralization and precipitation are the most favourable processes from the viewpoint of safety. They lead to very low residual levels in the groundwater and effective retardation of the radioactive elements. It can be assumed on good grounds that many of the elements in the waste participate in mineralization and precipitation reactions - for example cesium (mineralization) and protactinium and americium (precipitation).

However, sorption has been treated purely as an ion-exchange process in the calculations of the transit time for radioactive elements. As a result, the magnitude of the retardation has been underestimated. The retardation is described in the calculations by means of a retardation factor which is defined as the ratio between the flow velocity of the water and the transport velocity of the element in question.

The retardation factors can be calculated on the basis of chemical equilibrium constants which have been determined experimentally. Within the project, such determinations have been carried out for various elements in contact with the groundwater and the buffer material (10% bentonite and 90% quartz sand) or granite. Comparisons have been made with foreign measurements. The following table gives the retardation factors for the most important elements in the waste, both for impervious rock with a permeability of 10^{-9} m/s and for pervious rock with a permeability of 10^{-5} m/s.

Retardation factors for certain radioactive elements

Element	Impervious rock	Pervious rock
Strontium	57	7
Zirconium	8 400	450
Technetium	1	1
Iodine	1	1
Cesium	840	90
Radium	700	76
Thorium	5 200	280
Uranium	43	3
Neptunium	260	15

Element	Impervious rock	Pervious rock
Plutonium	1 100	58
Americium	84 000	4 500

A retardation factor of 1 means that the element migrates at the same rate as water, while a retardation factor of 700 means that it takes 700 times as long for the element to move the same distance as the water. The given retardation factors are mean values. Different fractions of the dissolved quantity of a given element will have longer or shorter retardation periods. This is taken into account in the calculations.

The retardation factor for strontium in pervious rock (permeability approx. 10^{-5} m/s) given in the table has been confirmed by means of field tests conducted at Studsvik.

The retardation of the elements radium, thorium, uranium, neptunium, plutonium and americium is of great importance for the long-range safety of the final repository. The values of the retardation factors for neptunium and plutonium used in the calculations are probably too low by at least a factor of 10 (see chapter IV:6.5).

Primary recipient

Escape of the radioactive elements through the various barriers (canister, buffer material, rock) can eventually lead to contact with the biosphere. Since the elements are dispersed with the groundwater, such contact is achieved primarily in a receiving body of water.

Three main cases of primary recipients have been studied:

- deep-drilled well close to the final repository
- lake close to the final repository
- Baltic Sea

These main cases are illustrated schematically in figure 13-4.

The annual leached quantities of radioactive elements which reach the primary recipient will be diluted in a relatively large volume of water. In the case of the well, this has been calculated to be 500 000 m^3 and in the case of the lake 25 million m^3. The calculations are based on the local conditions at Lake Finnsjön near Forsmark. These conditions are judged to be relatively unfavourable in the case of the well, but representative for a larger number of sites in the case of the lake. (See section 7.4.1.)

When the radioactive elements have reached the biosphere via the primary recipients, they can reach man in basically two different ways. The elements can be ingested into the body either through food and water or through inhalation. As long as they remain in the body, they can give rise to so-called "internal irradiation". Knowledge concerning the transport and enrichment of the radioactive substances in the food chains is therefore of great importance for being able to calculate the dose load on man.

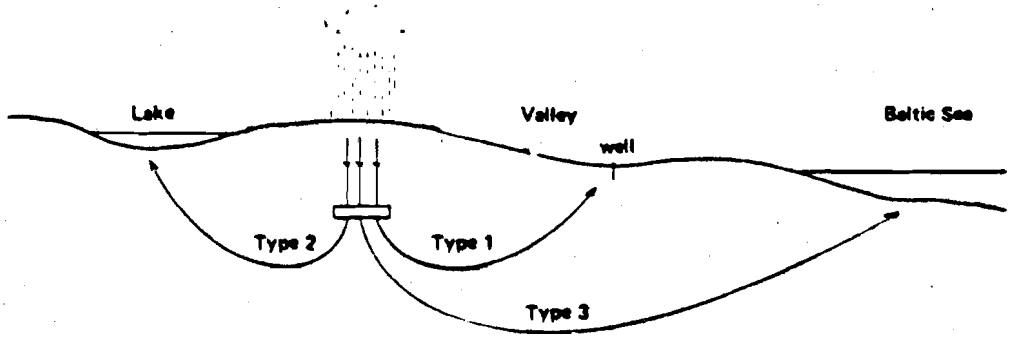


Figure 13-4. The three main cases of transport of radioactive substances to the biosphere.

Human beings can also be irradiated by radioactive elements outside of the body - "external irradiation". Figure 13-5 illustrates some of the paths through which radioactive elements in our environment can reach man. In order to establish the dose load, the radiation doses from inhalation and from the consumption of water and food have been calculated. Irradiation doses from the handling of fishing tackle and from ground deposits and water, for example in connection with bathing, have also been calculated.

A matter of primary interest is to establish which individuals may receive the highest radiation doses. These persons can be identified on the basis of the occupation, diet, living conditions etc. If this information is combined with knowledge regarding where the radioactive substances from the final repository may reach the biosphere, the so-called "critical groups" can be identified.

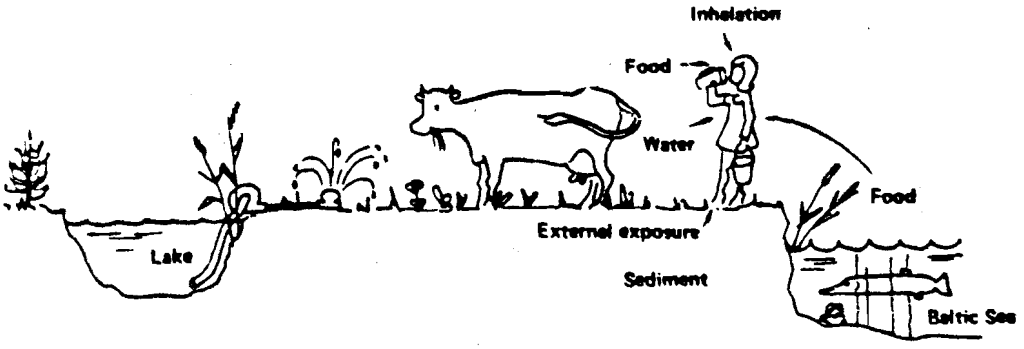


Figure 13-5. Paths of human exposure in the local ecosystem.

From the three outflow points - wells, lakes and the Baltic Sea - the radioactive substances disperse into the local ecosystem. In the model which is used in the safety analysis, this is assumed to comprise 0.25 km^2 of agricultural land plus a small lake such as Lake Finnsjön near Forsmark or Lake Götemaren north of Oskarshamn. Irrigation in the area employs well water or lake water at the rate of about 200 litres per day. It is assumed that the same supplies are used for both irrigation and drinking water.

The radioactive elements which enter the local ecosystem accumulate in the surface layer of the ground. They are transported first by the groundwater and then by the surface runoff. How fast the elements can enter into the natural water cycle depends on the sorption properties of the ground. Exposure has been calculated on the basis of the activity which reaches the local ecosystem via irrigation and the activity level which is obtained through long-term accumulation in the ground.

In the case where the outflow is into the Baltic Sea, the critical groups in the coastal zone are exposed through sea water, sediment and fish. The ecosystem is a 2 km wide and 30 km long coastal section where the radioactive substances from the rock repository enter the Baltic Sea. From the local ecosystem, they are further dispersed to other parts of the biosphere. How this happens is described in volume IV, chapter 6.7.

13.4.3 Consequences of the slow dispersal of radioactive elements

The consequences of a slow dispersal of the radioactive elements have been analysed for the following case:

- the encapsulation on the waste containers is penetrated after 1 000 years and all waste glass bodies are completely exposed to the groundwater after another 5 000 years
- the glass is leached at a rate which leads to complete dissolution in 30 000 years
- the transit time of the water in impervious rock from the final repository to the biosphere is 400 years
- as the radioactive substances pass through the rock, they are retarded in the manner described above.

These presumptions entail a number of conservative assumptions which lead to an overestimation of the calculated radiation doses.

The highest radiation dose in rems to a human being over a 30-year period has been calculated as a function of the time from the start of final storage. The period of 30 years was chosen in accordance with the common practice of counting this period as one generation and since only relatively small radiation doses come into question here.

Figure 13-6 shows a comparison between calculated maximum individual doses for the cases well, lake and Baltic Sea as primary inflow sources into the biosphere. As is evident from the figure, the well case gives approximately 15 times higher maximum doses than the lake case and approx. 1 500 times higher values than the Baltic Sea case.

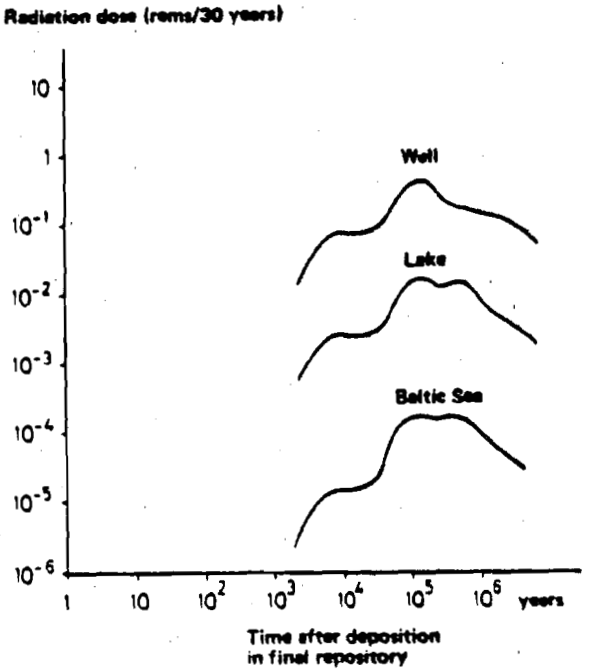


Figure 13-6. Calculated maximum individual doses to critical group (nearby residents) for various primary recipients.

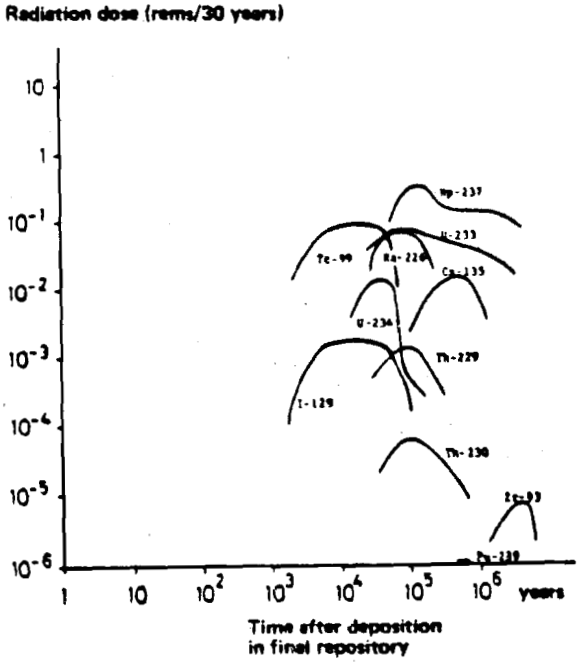


Figure 13-7. Calculated maximum individual doses to critical group (nearby residents) from different nuclides. The calculations were made for a well as the recipient.

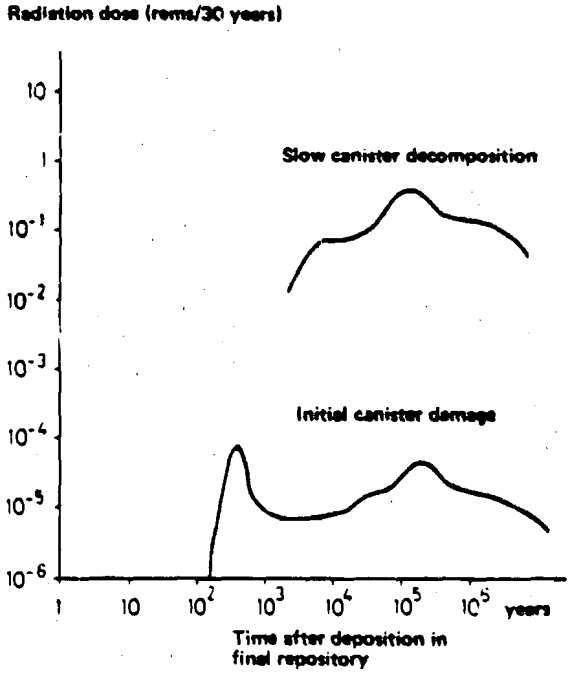


Figure 13-8. Calculated maximum individual doses to critical group (nearby residents) for the two main cases of canister damage. The cases are a) slow decomposition of all canisters during the period from 1000 to 6000 years, and b) initial damage to one canister. The primary recipient is a well.

Figure 13-7 shows the calculated maximum individual dose as a function of the time for different nuclides for the case where a well is the primary inflow recipient.

The results show that the dominant nuclides are neptunium-237, uranium-233, radium-226 and technetium-99. It is noteable that no radiation doses appear until after more than 1 000 years, due to the fact that the lead-titanium canister remains fully intact for at least 1 000 years and that the transit time for water is 400 years. Strontium-90 and cesium-137 decay completely during this time. Furthermore the nuclides of americium and plutonium are retarded for such a long time that they do not contribute significantly to the calculated radiation doses.

The dominant nuclides for the lake and Baltic Sea cases are cesium-135 and neptunium-237.

The dominant exposure paths in the well case are via drinking water. The dominant radiation doses in the lake and Baltic Sea cases are those due to fish consumption.

A comparison between slow decomposition of the canister and an initially damaged canister (treated as a glass body without any canister at all) is illustrated for the well case by figure 13-8. The latter case gives certain low radiation doses which appear after about 200 years, but the doses are far less than for the former case. The radiation doses due to slow decomposition of the

canisters are approximately 6 000 times higher than those from an initially damaged canister.

Since it has been shown that the case with a well as the primary recipient gives rise to the highest radiation doses, the other alternatives are of less interest in a discussion of maximum consequences for individuals. As mentioned above, the data and assumptions used in the chain of calculations have been selected with safety margins which are quite considerable in some cases. This leads to the radiation doses given by the upper curve in figure 13-9. If we instead use data and assumptions which can be regarded as more realistic, the calculations lead to the radiation doses within the area under the curve "probable conditions" in figure 13-9.

The dose limits applied to nuclear plants are also included in the same figure for purposes of comparison (see chapter 12). In addition, the range of variation for natural ionizing radiation in Sweden and for the radiation doses which can be obtained from natural drinking water in Sweden are also given. The latter values have been calculated on the basis of measured levels of radium-226, using the same dose conversion factors as in the other calculations. Only a few measured values are given for drinking water within the upper part of the variation range. The median value for all the 61 measured values corresponds to a radiation dose of 0.15 rem per 30 years.

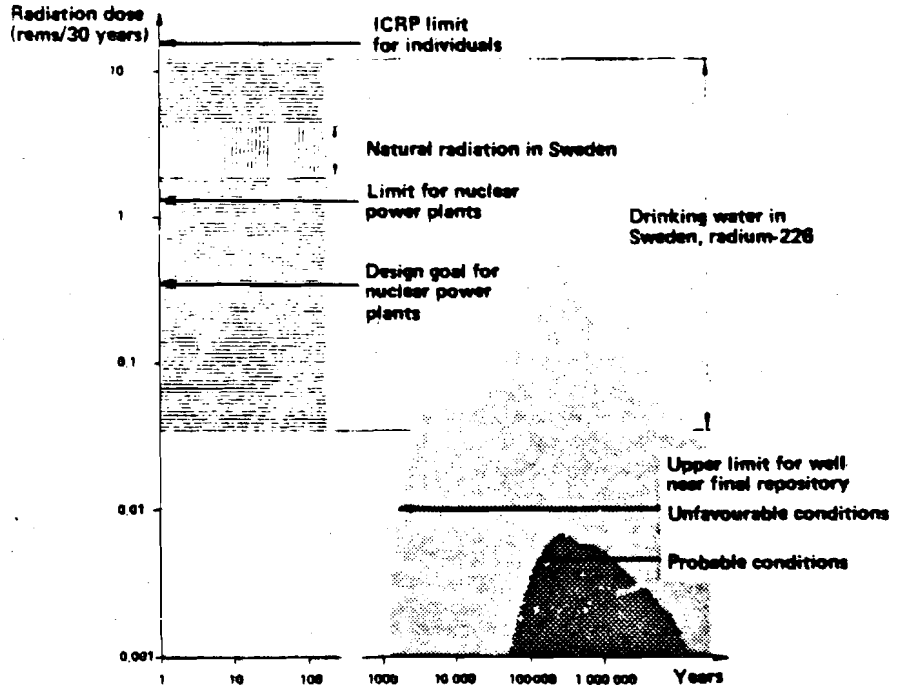


Figure 13-9. Calculated upper limit for radiation doses to people who live near the final repository (critical group). The calculations pertain to the slow decomposition of the canister with a well as the primary recipient. For purposes of comparison, the dose load from several natural radiation sources as well as a number of established dose limits have also been plotted in the diagram.

As is evident from the figure, the calculated radiation doses from radioactive substances which may escape from the final repository - even when large margins of safety are included in the calculations - are considerably below the limit recommended by the International Commission on Radiological Protection (ICRP). They are also well below the value used for nuclear power plants in Sweden, although slightly above the recommended value used as a design goal for new nuclear power plants. The ranges of variation for natural ionizing radiation and for radiation doses from natural drinking water are considerably greater than the calculated radiation dose from the final repository.

In addition to the maximum radiation dose to a human being, the collective dose to the earth's total population has also been calculated. This calculation is rather complicated (see volume IV for a more detailed description).

The collective dose to the earth's total population has been calculated both for the most unfavourable 500 year-period and for the first 10 000 years following the commencement of final storage. The results are largely independent of the path of entry into the biosphere. The collective dose from the final repository for the most unfavourable 500 year-period has been calculated to be about 2 000 manrems, i.e. 0.007 manrem per MWe and year of operation for 13 reactors with a combined output of 10 000 MWe which are operated for 30 years. These 500 years occur after several hundred thousand years. A collective dose of 30 000 manrems is obtained for the first 10 000 years, i.e. 0.1 manrem per MWe and year of operation. These collective dose values are based on the same unfavourable conditions as were specified above. Both values are clearly within the guideline limit set by the radiation protection authorities of 1 manrem per MWe and year of operation for the entire nuclear fuel cycle.

13.4.4 Consequences of extreme events

Certain types of extreme events might have a considerable effect on the escape of radioactive substances from a final repository. Such extreme events include e.g. movements in the bedrock in connection with earthquakes or the creation of new cracks due to other causes. Other events within this category include meteor impacts, acts of war, sabotage or some form of future human disturbance.

Rock movements, earthquakes

Rock movements could damage a final repository, either by creating new paths for groundwater flow or by damaging the canisters. However, limited damage to the canisters will not substantially change the assumptions of the safety analysis, since a case with initial canister damage has already been considered.

A number of studies have been conducted for the purpose of establishing the probability of bedrock movements which might affect the safety of a final repository.

The level of seismic activity in Sweden is very low, and very few earthquakes have caused damage to the ground surface.

The faults which have been observed in the Swedish bedrock are largely the result of the tectonic and seismic events of approximately 1 800 million years, where the movement from millenium to millenium is on the order of a few mm. However, larger fault movements have been observed and reported in areas with special zones of movement, for example in northwest Skåne and in Norrbotten County. The land elevation which followed the melting of the inland ice and is still proceeding is probably the primary cause of these recent bedrock movements.

Bedrock movements are discussed in greater detail in chapter II:7.

Studies of the occurrence of earthquakes in Sweden show that earthquakes have been concentrated to certain belts. Outside of these belts, there are large areas where no seismic activity at all has been observed. Magnitudes greater than 3.5 are rare, even within the most active areas.

The following relationship between magnitude and displacement has been estimated:

Magnitude	Displacement
3.5	0.3 cm
4.0	0.6 cm
4.5	1.5 cm
5.0	3.6 cm

The Swedish bedrock exhibits a pattern of fracture zones of varying size. Geological observations show that new fissures and faults will be located in already existent joint planes. In simple terms, this corresponds to the principle that it is the weakest link in a chain which breaks.

The probability that a final repository covering 1 km^2 will be affected by a fault movement has been estimated to be less than 10^{-9} per year for Sweden as a whole.

It has also been shown that vertical rock displacements must be several dm in order to jeopardize the sealing capacity of the clay material. However, stresses in the canister material can reach considerable proportions in the event of displacements of only a few cm.

In summary, studies carried out by KBS of bedrock movements which could have an adverse effect on the safety of the final repository have shown that:

- the probability of such movements in the Swedish bedrock is very low
- within areas which are surrounded but not intersected by fracture zones, the probability of new flow paths (cracks in the rock) opening is extremely low
- sections of rock which are found to have a high fracture content during the construction of the final repository should not be utilized for storage

- neither the proposed buffer layer nor the canister will be damaged, even if bedrock movements of considerable proportions by Swedish standard should occur in the final repository.

The risk of damage to part of the final repository as a result of bedrock movements is thus extremely low. If such damage should nevertheless occur, it will probably affect only a few percent of the canisters. The consequences of such damage are deemed to be of the same magnitude as those which have been calculated for slow canister decomposition.

Meteorite impact

If a meteorite should hit the surface of the earth directly above a final repository, a crater would be created which could weaken the geological barrier or, at worst, eliminate it completely.

Studies of meteorite impacts which have occurred over a period of 2 000 million years show that the probability of meteorite impact which would create a crater approximately 100 m deep is roughly 10^{-13} per year and km^2. Historical experience also confirms the assumption that a meteorite impact is not a risk which has to be considered in this context.

Acts of war and sabotage

In the long time perspective which must be applied to the final repository, acts of war cannot be considered to be "extreme events". On the other hand, the possibility that acts of war might lead to serious consequences for the safety of a terminally sealed final repository at a depth of some 500 metres in the Swedish bedrock must be considered to be remote.

Ground detonations of nuclear devices of 10-50 megatons create craters in the rock with a depth of roughly 110-180 m. Thus, the geological barrier would not be broken through, but may well be weakened. In such a situation, however, this would be of subordinate importance, since any release of radioactivity from the final repository would represent only a fraction of the radioactivity caused by the bomb, which would remain in the area for a long period of time.

Wartime damages to the final repository and the encapsulation station during the deposition stage are, naturally, conceiveable. But the probability is low, since these facilities are not likely to be primary targets for military actions. The consequences of bomb hits and similar occurrences will also be limited compared to the situations which would otherwise be a result of such acts of war.

Safeguards against sabotage as described in section 10.4 will be provided during the intermediate storage, encapsulation and final deposition stages. After the final repository has been closed and sealed, effective acts of sabotage are impossible.

Compared to other installations which experience has shown to be more likely targets for sabotage in terrorist actions, the faci-

lities described here are less attractive to potential saboteurs and are roughly comparable to other industrial plants where environmentally hazardous material is handled.

Future disturbance by man

It is conceivable that the knowledge of where the final repository is located may be lost in the distant future and that man at that time may, for some reason, perform drilling or rock work which results in contact with the waste. The final repository is situated in one of our most common types of rock which does not contain any valuable minerals which could conceivably be considered for profitable extraction. The depth and low water content of the impervious rock selected for this purpose makes it highly improbable that deep wells will be drilled for water in the future. No reason can be seen for seeking out such great depths for the construction of rock storage vaults or the like. Furthermore, the loss of the knowledge of the location of the final repository would presuppose that our current civilization would be destroyed as a result of some catastrophic event such as a global war of extermination or a new ice age. If the country is then repopulated again, the risks mentioned here would arise, but only after the new population had achieved a level of technological development which permitted advanced rock work. In such a case, it is probable that such a civilization would also possess the ability to detect the radioactivity in the final repository and act accordingly to avoid damage to the repository. A new glaciation of the country would not be expected to affect the integrity of the final repository. (See section II:7.7.)

13.4.5 Summary safety evaluation of final storage

The high-level waste from the reprocessing of spent nuclear fuel is vitrified and encapsulated in lead-titanium canisters after which it is emplaced in good rock at a depth of 500 m, where it is packed in a bed of buffer material (90% quartz sand and 10% bentonite). The safety analysis of such a final storage shows the following:

1 During the period of at least 1 000 years when the leadtitanium canister is completely intact, the elements strontium-90 and cesium-137 decay almost completely.

2 Initial damage to an individual canister would not lead to any measurable increase of the radiation level.

3 A slow breakdown of the canister could, after several thousand years, lead to a slight increase of the level of radioactivity in the environment. The elements plutonium and americium are retained in cracks in the rock etc. This slight increase comes primarily from neptunium-237, technetium-99, radium-226 and uranium-233 as well as cesium-135 and iodine-129.

4 In the most unfavourable case - a deep-drilled drinking water well near the final repository - the radiation dose to a human being in the future could increase by a maximum of 13 millirems per year. This increase is less than the local

variations which occur in the natural radiation at various places in Sweden. (See figure 13-10.)

5 The calculated maximum radiation doses are considerably lower than the maximum permissible radiation dose for individuals recommended by the ICRP.

6 The calculated maximum radiation doses are well below the Swedish limit for nuclear power plants, but of roughly the same magnitude as the guideline value which is applied as a design goal in the construction of new nuclear power plants. (See figure 13-10.)

7 The probable amount of additional radiation is less than 1% of the maximum value specified in point 4. This is due to the fact that the decomposition of canisters and leaching of glass at the low water flows which occur at a depth of 500 m in solid rock can be expected to take place at a considerably slower rate than what has been assumed in the calculations. Furthermore, these calculations were based on a retardation factor for neptunium and a water transit time which have both been conservatively selected.

8 The regional and global dose load on large population groups has been calculated over the most unfavourable 500-year period in the future. In the very long run, a maximum 500-year dose of approximately 2 000 manrems can be obtained, which corresponds to 0.007 manrem per MWe and year of operation.

9 Even in the most unfavourable case, with extremely conservative data in the calculations, the health hazards are extremely small, if any.

The calculated collective doses correspond to 0.4 cases of cancer and 0.4 cases of genetic defects for the population of the entire earth over a period of 500 years. The present frequency of mortalities due to cancer in Sweden is approximately 20 000 per year. Of all persons born, approximately 3% are afflicated with natural hereditary defects, which means that some 3 000 cases occur annually in Sweden. The given values for medical effects have been calculated on the basis of the internationally accepted relationships between radiation dose and maximum medical effects. Many factors indicate that these assumptions overestimate the medical effects at the low dose values and dose rates which are in question here.

10 The calculated increase of the level of radioactive elements in the recipients to which the waste products may conceivably spread is comparable to the natural levels of such substances. Neptunium-237 can be compared to uranium and cesium to potassium. The following table shows the ranges of variation for the level of certain elements in natural water and levels which have been calculated for the various primary recipients in the most unfavourable case.

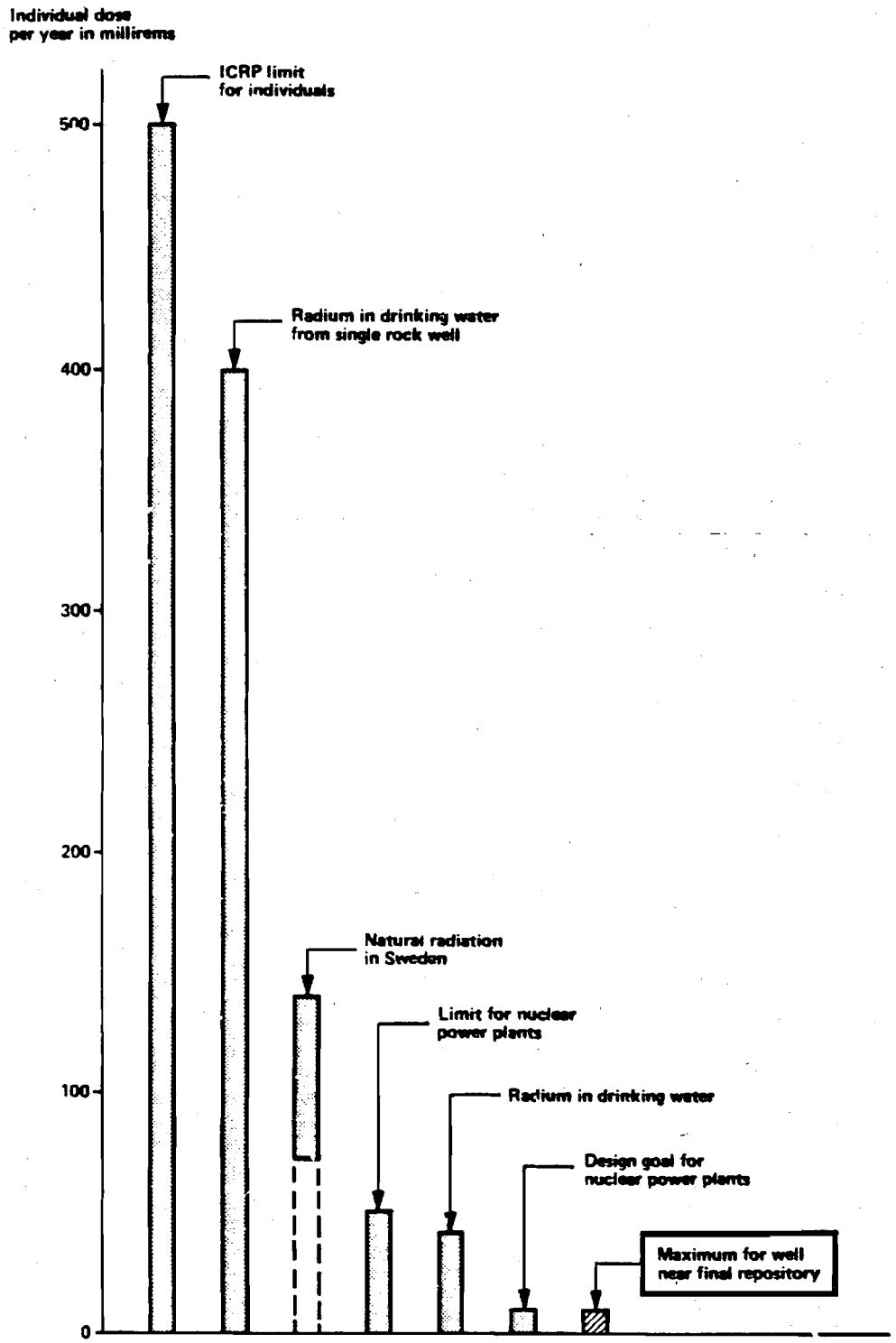


Figure 13-10. Bar graph showing the calculated maximum annual radiation doses which the final repository can give to a nearby resident and the annual dose to man from some natural radiation sources plus some established dose limits. The dose from drinking water comes from radium-226.

Radioactive element	Levels in natural water in Sweden (pCi/l)		Maximum calculated increase of level in primary recipients near the final repository[a]	
	Drinking water	Sea water[b]	Well	Lake
Radium-226	0.1-40	0.3	0.1	0.002
Uranium	0.1-1500[c]	3	30	0.6
Neptunium-237	-	-	90	2
Potassium-40[d]	ca 20	330	-	-
Cesium-135[d]	-	-	25	0.5

a) Expected maximum values are approximately 100 times less.
b) With 3.5% salt content.
c) Natural water (not necessarily drinking water).
d) Potassium-40 and cesium-135 are biologically comparable.

The radiation dose from radium-226 in drinking water is shown in the comparison illustrated in figure 13-10.

11 Even in the case where a number of unfavourable assumptions have been made, the calculated changes in the radiation environment are considerably less than normally occurring natural variations. These natural variations do not have any effects on either man or ecological systems which can be demonstrated today. The calculated maximum radiation doses due to leakage from a final repository are below the limit values for nuclear power plants which have been issued by the radiation protection authorities in Sweden. The proposed method for the final storage of high-level waste glass is therefore deemed to be absolutely safe.

14 SKELETON PLAN FOR CONTINUED WORK

The studies conducted under the auspices of the KBS Project have largely been concerned with the review and processing of existing knowledge and data, but the work in a number of areas has been of a development nature. The need for further research and study is urgent in a number of areas in order to gather sufficient information to serve as a basis for a technically/economically optimum design of the various phases in the handling and storage chain.

Thus, the work which has been done by KBS should be followed up, both by development work and by efforts aimed at gradually increasing the degree of specification for the facilities whose construction is most imminent. It is of the utmost importance that activities and developments in other countries be continuosly followed and that opportunities for co-operation be fully exploited.

The various plants for the handling and storage of high-level waste have to be completed at very disparate points in time, as shown in chapter I:3. The transportation system and the central fuel storage facility must be commissioned within a few years and an intermediate store by 1990, while the encapsulation station and final repository will not be required until around 2020.

The skeleton plan shown in figure 14-1 provides an approximate schedule for the completion of the various installations in various phases. The major feature of the activities pursued during the different phases are described below.

Phase 1 (1977-1978)

This phase encompasses a KBS activity period which is expected to continue to mid-1978. Efficient completion of the activities begun by KBS requires that the question of responsibility and organi must be decided before the end of this phase.

Phase 2 (1978-1984)

During this phase, the transportation system and the central fuel storage facility will be completed and commissioned.

Study and development work during this phase will be pursued in accordance with a multi-year plan aimed at a presentation of the total results at the end of the phase. The plan will encompass hydrogeological studies, refinement of models for the migration

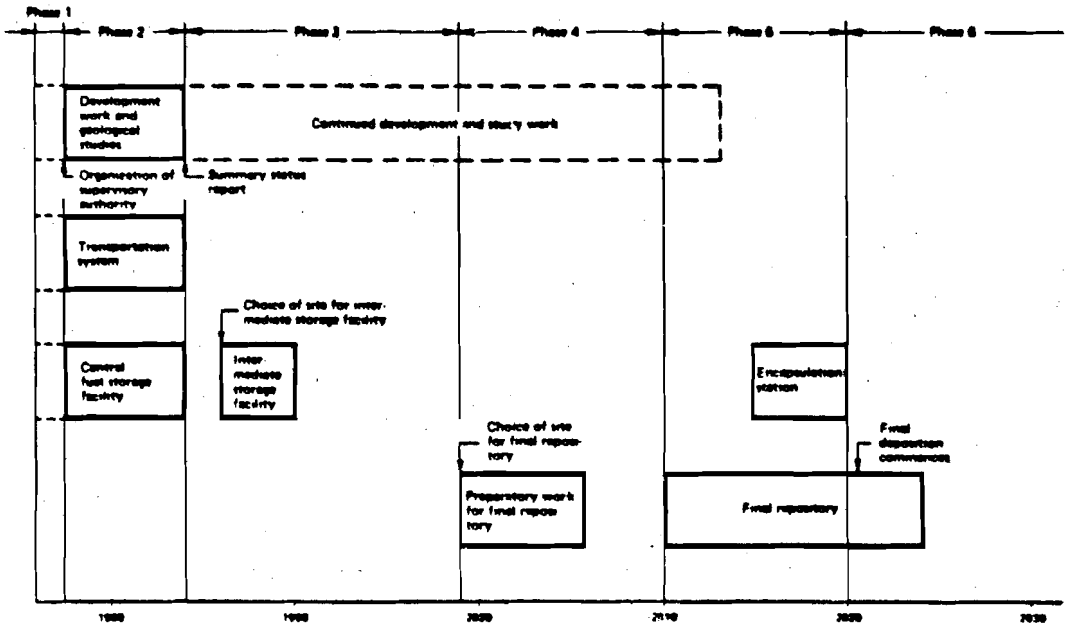


Figure 14-1. Skeleton plan for continued work on the final stages of the nuclear fuel cycle.

of nuclides in the groundwater, continued studies of buffer and encapsulation materials and design of the final repository.

During this stage, a major non-radioactive experiment should also be conducted whereby the function of various barriers can be studied on a relatively large scale under conditions which stimulate the actual environment in the final repository. The heating tests which are being conducted at Stripa in co-operation between the USA and KBS/SKBF will be completed during this phase and the results can be used in future tests.

A considerable international exchange of knowledge and experience will be pursued during this phase.

Phase 3 (1985–1999)

During this phase, the intermediate store for vitrified waste is expected to be completed and put into operation. The schedule for these activities will be contingent upon when the waste is returned to Sweden. According to an agreement between SKBF and COGEMA, this will be 1990 at the earliest. The question of the joint siting of the encapsulation station together with the intermediate store or the final repository (see chapter I:11) must be clarified before design work on the intermediate storage facility can commence.

Efforts with respect to encapsulation methodology and the design of the final repository will primarily entail a follow-up of the technical developments and some complementary internal efforts.

Phase 4 (2000-2010)

If a final repository is to be reday to accept high-level waste by 2020, the final choice of the site should be made at the beginning of this phase.

Full certainty as to whether the studies conducted from the surface have provided a correct picture of the actual conditions below the surface will not be obtained until certain shafts and tunnels have been built. A final verification that the rock in the selected area possesses the required characteristics should therefore be carried out in tunnels constructed at an early stage. If it is thereby found that the area is unsuitable, another area must be selected. Even though such a development is highly unlikely after the extensive preliminary studies which have been carried out, the schedule should take into account such a contingency.

During the 10-year period from the time when the first shaft has been sunk and the work on the main parts of the repository has been begun, there is sufficient time to build and study a pilot plant at the right depth and in the right environment. But it is doubtful whether such a pilot plant is economically justifiable.

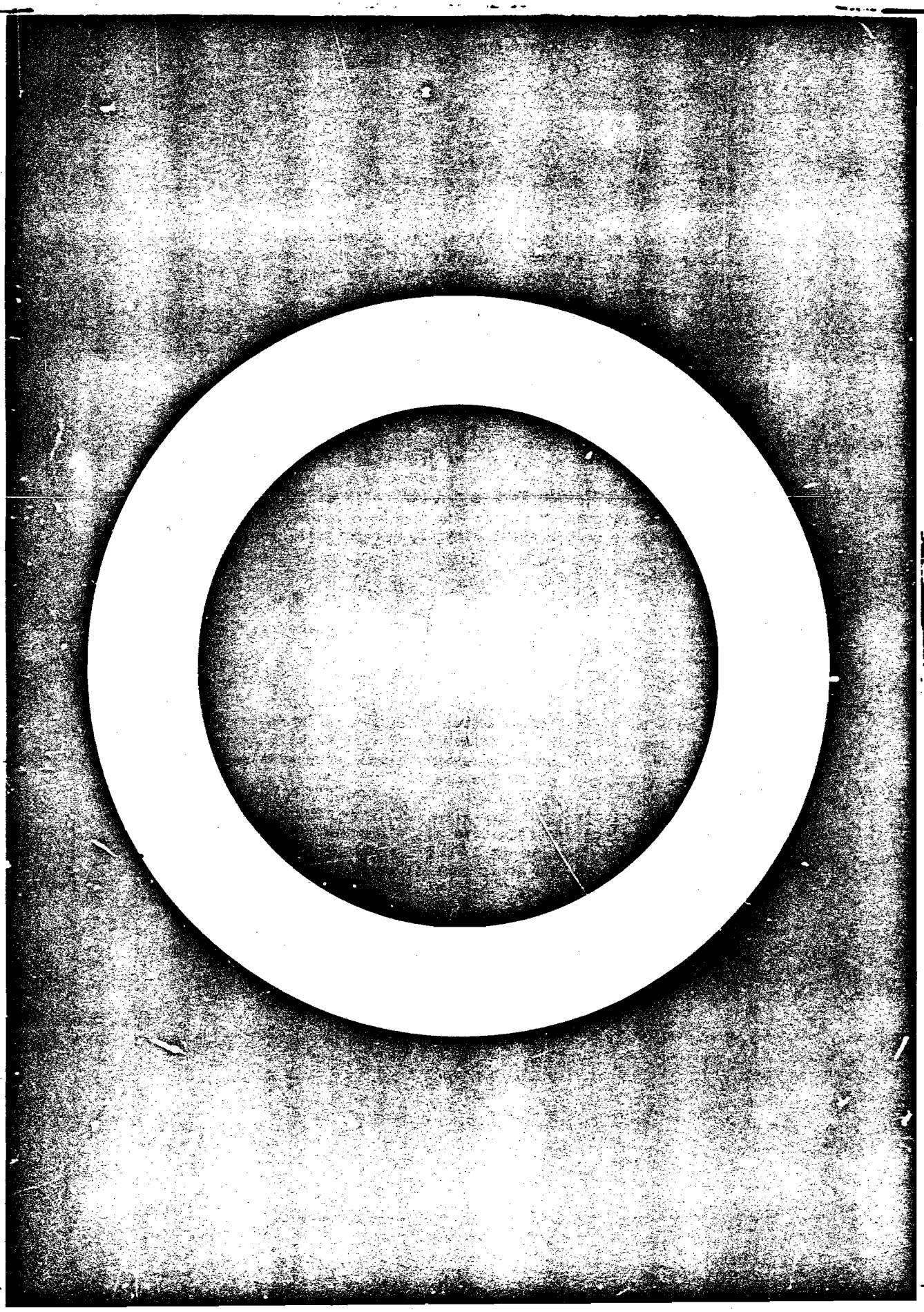
Phase 5 (2011-2020)

Final engineering and design work will be completed during this phase, along with construction of the encapsulation station and parts of the final repository.

Phase 6 (2021-?)

During this phase, the encapsulation station is in operation and the final repository is gradually filled with waste canisters. Construction of the final repository is completed simultaneously during the first part of the phase.

After the repository is filled with canisters, tunnels and shafts are sealed, surface installations are dismantled and the landscape is restored. The authorities can be expected to prescribe certain measurements and other control measures after the closure and sealing of the final repository.



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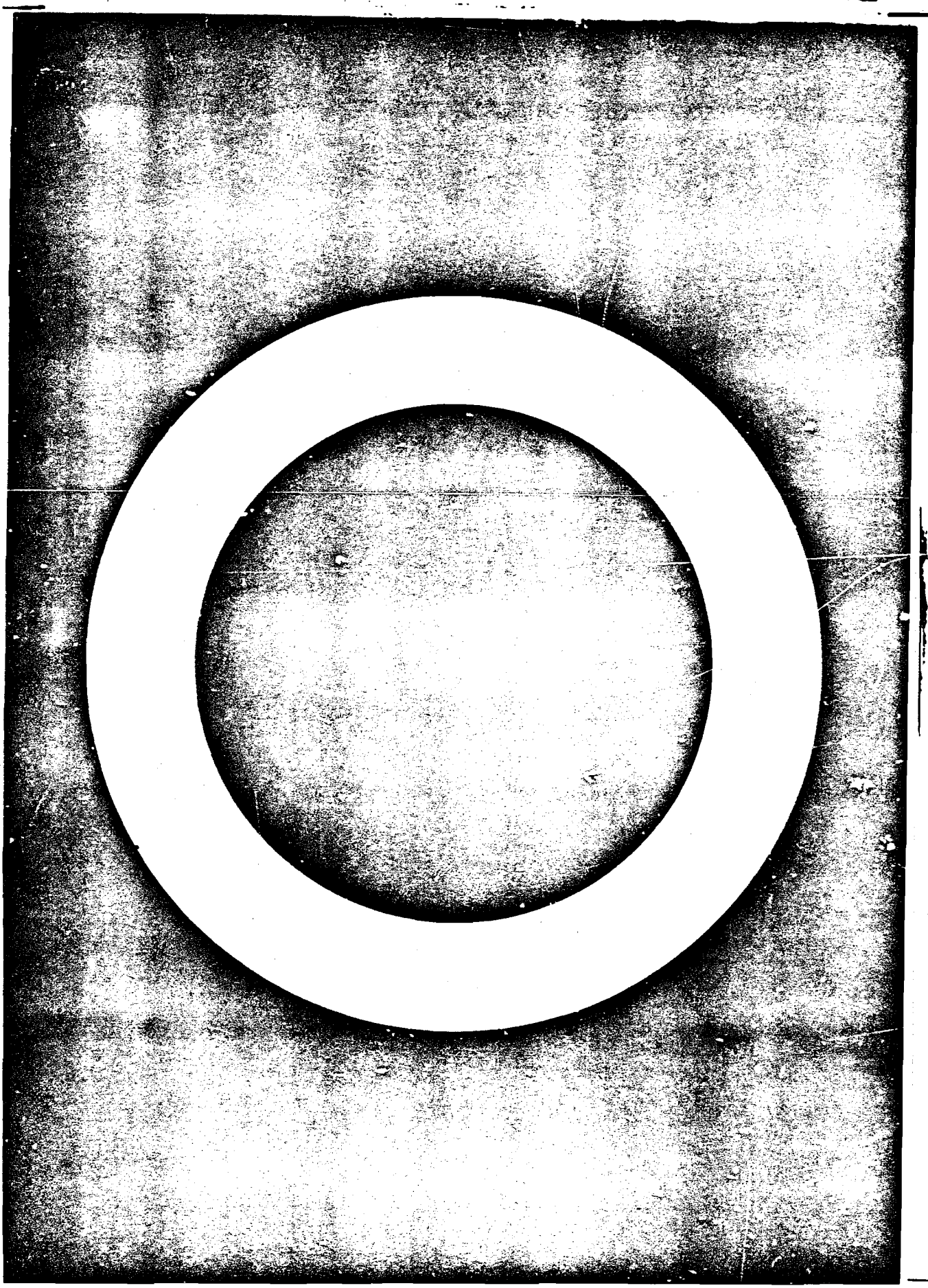
APPENDIX 1 STATUS REPORT ON DIRECT DISPOSAL OF SPENT FUEL

CONTENTS



STATUS REPORT ON DIRECT DISPOSAL OF SPENT FUEL

1 INTRODUCTION

Parallel with the present study of the handling and storage of vitrified waste from reprocessing, the alternative method of direct disposal of spent fuel has been studied. Following is a report on the status of these studies, which will be presented in greater detail later on in a separate report.

2 FUNDAMENTAL PRINCIPLES

In the case of direct disposal the spent fuel is placed directly in final storage without prior reprocessing.

As is the case of the reprocessing waste, and for the same reasons, deposition in the final repository will be preceded by a period of intermediate storage followed by encapsulation.

The intermediate storage facility will have water-filled pools in which the fuel elements will be stored dry in containers of stainless steel. This storage method is similar to the one used in the central fuel storage facility, except that the fuel does not come into contact with the water. The surrounding water provides the necessary cooling and radiation shielding. In this way, the intermediate storage facility will provide a natural continuation of the function of the central fuel storage facility. It has therefore been assumed that the intermediate storage facility will be located adjacent to the central fuel storage facility.

While the spent fuel is being kept in intermediate storage, it may be decided that reprocessing would be desirable in order to recover the remaining energy content of the fuel instead of depositing it directly in a final repository. The stainless steel container is therefore designed in such a manner as to permit removal of the fuel elements from the container.

As in the case of vitrification, a storage period of at least 30 years in the intermediate storage facility is foreseen. The fuel can then be deposited in a final repository of a design similar to the one proposed for vitrified reprocessing waste. Before deposition, the fuel will be encapsulated in a canister in an encapsulation station situated adjacent to the final repository.

The intermediate storage facility is designed for a capacity corresponding to 6 000 metric tons of uranium and the final repo-

sitory for 9 000 tons - the same capacity as for the reprocessing alternative. Since we do not know at this stage to what extent the two alternatives will be used, this report is based on the theoretical choice of "either or". In actuality, both alternatives may very well be employed, whereby the facilities will have to be adapted accordingly.

3 CHOICE OF ENCAPSULATION MATERIAL

3.1 GENERAL

As with the vitrified waste, the purpose of encapsulation is to provide the fuel with a corrosion-resistant casing to protect it from the groundwater in the final repository. Metallic encapsulation materials shall also provide radiation shielding to reduce radiolysis of the groundwater to a negligible level.

Since the toxicity of the spent fuel declines more slowly than that of the reprocessing waste, an inventory of possible encapsulation materials has been carried out aimed at finding materials with longer service lives than the combination of lead-titanium chosen for the vitrified reprocessing waste. Availability, economy and ease of fabrication have been taken into consideration.

On the basis of the results of this inventory, copper and two ceramic materials - alpha-aluminium oxide and a glassceramic material of the beta-spodume type - have been selected for closer study.

Alpha-aluminium oxide (corundum) is one of the most durable ceramic materials known. It was proposed by ASEA, who have developed a potentially suitable fabrication method based on high-pressure isostatic compaction.

Work is being done on the glass-ceramic alternative in collaboration with Corning Glass Works in the United States. But this work has not yet reached the point where any definite conclusions can be reported.

COPPER

Copper does not react at all with oxygen-free, pure water, which is verified by thermodynamic calculations. However, copper can react with oxidizing substances present in the groundwater, which may be present in concentrations of a few tenths of a milligram per litre. These oxidants are mainly oxygen, radiolysis products and sulphate and/or sulphide in combination with bacteria.

Groundwater experts are agreed that very little dissolved oxygen is present in the groundwater at the depths in question here. There is also little formation of oxidizing substances as a result of water radiolysis, due to the fact that the canister has thick walls.

The question of whether sulphate and/or sulphide can, in combination with bacterial action, cause corrosion to copper is under investigation.

The maximum attack on copper is limited to that caused by the oxidizing substances in the water which come into contact with the surface of the copper canister, whereby diffusion effects must be taken into consideration. By limiting the quantity of these oxidizing substances, it is possible to limit material losses due to copper corrosion.

Section 4.3 below describes a method of surrounding the canister with the same mixture of quartz sand and bentonite which is used for the reprocessing alternative. The canister is deposited horizontally in the storage tunnel, whereby it is enveloped in a thick bed of this material, which possesses very low permeability. If the corrosion attack on the canister is evenly distributed and if only small quantities of water can come into contact with the canister, it can be shown that corrosion is negligible, even over a period of millions of years.

Corrosion attacks on metals can, however, be of a local nature and may take the form of, for example, pitting. In pitting, erosion of the copper metal can be concentrated to a small part of the copper surface, whereby the time required for penetration is reduced proportionately. On the basis of earlier corrosion tests conducted on copper in various surface soils, the time required for the penetration of 20 cm of copper has been estimated to be at least 5 000 years. Continued study of pitting in the environment surrounding the canister will probably reveal a considerably longer canister life.

Detailed studies have shown that cracks are unlikely to occur in the buffer material surrounding the canister. If such cracks should nevertheless occur, there may be a local inflow of oxidizing substances to the surface where a crack is in contact with the canister, resulting in local corrosion.

Investigations of buffer materials other than quartz sand and bentonite are underway. The characteristics of compacted bentonite, asphalt and a mixture of MgO and SiO_2 are being studied.

A design proposal for a copper canister is shown in figure B1-1. The canister is fabricated from pure copper by the forging of a cast block which is then turned down on a lathe to the desired final external dimensions. After boring of the internal cavity, the opening end is turned to receive a lid. The lid is in three parts which are fastened by means of electron beam welding followed by helium leak-tracing on each part.

3.3 ALUMINIUM OXIDE

In aluminium oxide, aluminium is in its stable oxidation state, which means that no redox reactions will take place in an aqueous environment. The concentration of oxygen in the groundwater is therefore of no significance in this case, unlike in the case of copper.

But the oxide is not thermodynamically stable in water. A hydration takes place on the surface. At temperatures lower than 100°C, aluminium hydroxide is thereby obtained in amorphus or crystalline form. Crystalline aluminium hydroxide occurs in

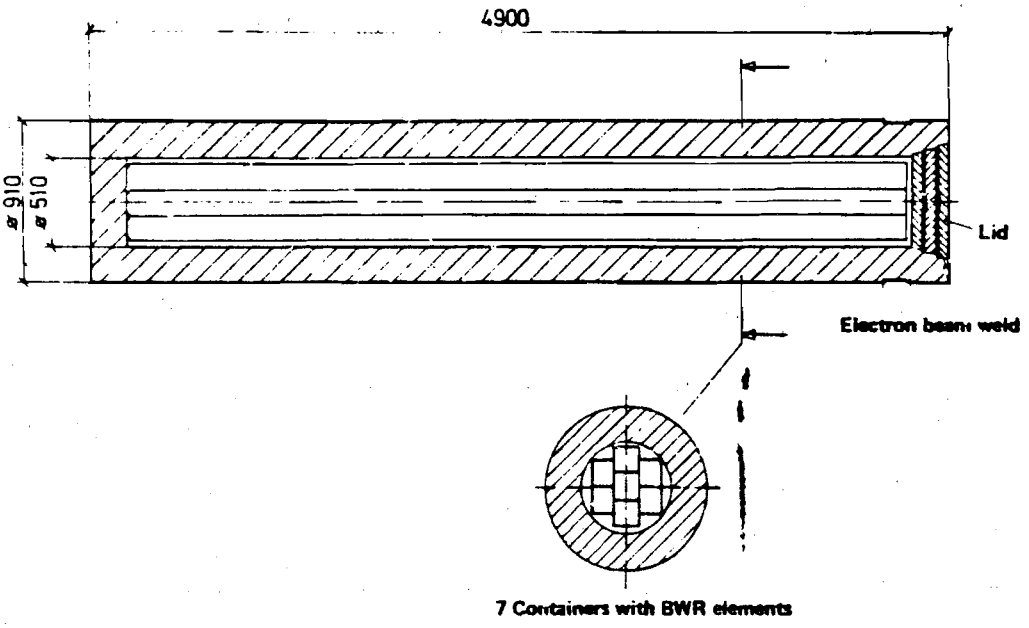


Figure B1-1. Copper canister for direct disposal of spent nuclear fuel. The canister has 7 containers with fuel elements from a boiling water reactor. The weight of the canister without fuel is about 20 metric tons.

nature as the mineral gibbsite and is more stable and less soluble than the amorphous form.

In freely flowing water at 90°C and PH 7, a corrosion rate of 0.2 µm/year has been measured. On the basis of other results, the corrosion rate at PH 9 is estimated to be 2 µm/year. Assuming a constant rate of corrosion, the latter value would correspond to a material loss of 20 mm in 10 000 years. In the case in question, much less corrosion is expected, due to a lower temperature, the presence of ions in the groundwater (which can produce a less soluble surface layer) and the slow rate of water flow. Under these conditions, corrosion can be expected to be practically negligible.

Local corrosion in the form of pitting and crevice corrosion does not occur on ceramic materials. Intercrystalline corrosion can also be disregarded in the case of aluminium oxide, providing the material is of sufficient purity. A factor which must be taken into consideration, however, is a form of stress corrosion cracking which can lead to delayed fracture. This form of stress corrosion cracking occurs in oxide-based ceramics in aqueous environments. Tensile stresses in the material lead to intensified corrosion at the apex of cracks, which gradually grow and can lead to fracture.

In order for delayed fracture to occur, the material must contain a defect of sufficient stress-raising character as regards depth, extent and form. In a single-phase material such as aluminium

oxide with small grain size, it is improbable that intercrystalline corrosion will lead to defects of such a critical magnitude.

In order to simplify canister fabrication, the canister should be shorter than what would be required if it were to accommodate full-length fuel rods. A method for rolling fuel rods encased in a tight metal casing into spiral form has therefore been developed. The rolls are placed on top of each other in the canister, whose length is determined by the number of rolls which it is to contain.

A design proposal for a ceramic canister is shown in figure B1-2. The capsule is fabricated at a pressure of 1 000 - 1 500 bar and a temperature of 1 350°C. The ceramic thermal barrier inside the canister is intended to keep the temperature of the fuel down when the semispherical lid is joined to the canister.

7.4 SUMMARY

In view of the fact that further investigation and development work remains to be done on ceramic canisters with respect to corrosion properties, risk of delayed fracture and fabrication technique, attention is being concentrated primarily on direct disposal of the spent fuel in copper canisters. The work with ceramic canisters will continue. The choice of encapsulation material and deposition technique may also be influenced by the fact

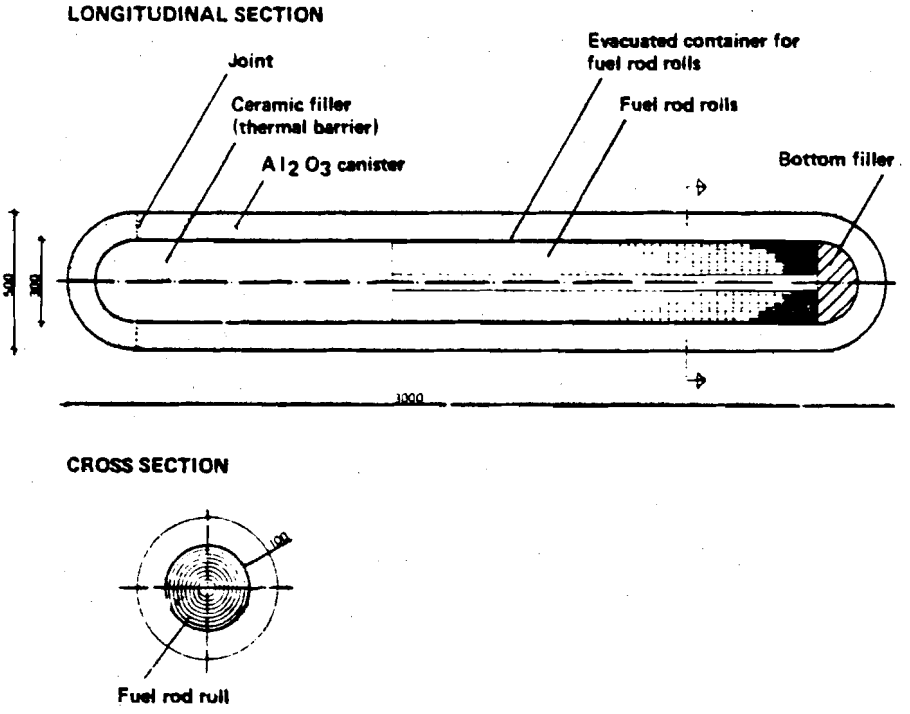


Figure B1-2. Ceramic canister of aluminium oxide. The fuel rods are rolled into spirals prior to placing in the canister.

that the buffer mixture around the canister may be replaced by some other material.

The Swedish Corrosion Research Institute and its reference group of specialists in the field of corrosion and materials has, under contract from KBS, studied the corrosion resistance of the proposed encapsulation materials. In a status report dated 27 September 1977 (and reproduced in KBS technical report No. 31), the following assessment of the service life of a copper canister and of an aluminium oxide canister is presented:

> "Copper is a relatively noble metal and is therefore thermodynamically stable in oxygen-free, pure water. The groundwater which comes into contact with the canister will probably contain oxidants which may cause some local corrosion. But a service life of at least 5 000 years is considered a realistic estimate, on the basis of our present state of knowledge.
>
> Some uncertainty exists, however, with regard to whether sulphate in the groundwater could, in combination with bacterial action, cause an attack on the copper. Such an attack would require access to organically bound carbon. In order to reduce the risk of such attacks, a low concentration of organic substances in the groundwater and in the buffer material is desirable.
>
> Local corrosion due to the action of sulphate should be studied more thoroughly.
>
> - - -
>
> Aluminium oxide is not a thermodynamically stable material under the conditions in question. A hydration of the surface layer and some dissolution takes place upon contact with the groundwater. But on the basis of currently available knowledge, both the dissolution and the growth of the hydrated zone appear to proceed very slowly.
>
> The risk of delayed fracture in this material cannot be excluded in theory, but it should be possible to fabricate and emplace the canister in such a manner that the risk of delayed fracture with the proposed design is negligible. Provided that it is possible to fabricate a canister of this material of sufficient purity and quality (as regards e.g. cracks and internal stresses), this alternative would appear to provide the necessary prerequisites for achieving a very long service life. Before a final evaluation can be made, however, more detailed investigations of corrosion in the environment in question would be desirable - especially with regard to hydration and delayed fracture."

The specialists in the Corrosion Research Institute's reference group are in unanimous agreement with these conclusions. Supplementary statements by some members of the reference group were appended to the Institute's status report.

In one of the supplementary statements, it is stated that the estimates given in the status report are conservative and represent 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.

4 DESIGN OF FACILITIES

4.1 INTERMEDIATE STORAGE

It is assumed that the intermediate storage facility will be located underground adjacent to the central fuel storage facility (see I:4).

In the central fuel storage facility, the fuel is stored in direct contact with the water in the spent fuel pools, which provides good cooling. After about 10 years of storage, the heat generation in the fuel has decreased to such a level that dry storage is possible without the fuel becoming excessively hot.

When the fuel is to be transported to the intermediate storage facility, it is conveyed underwater in a channel from the central fuel storage facility up to a position underneath a cell (see figure B1-3). The cell is enclosed in concrete of sufficient thickness to provide adequate radiation shielding. The fuel elements are handled in the cell via remote control. They are lifted up out of the water into the cell and allowed to dry. They are then placed in a stainless steel container with 2 mm thick walls,

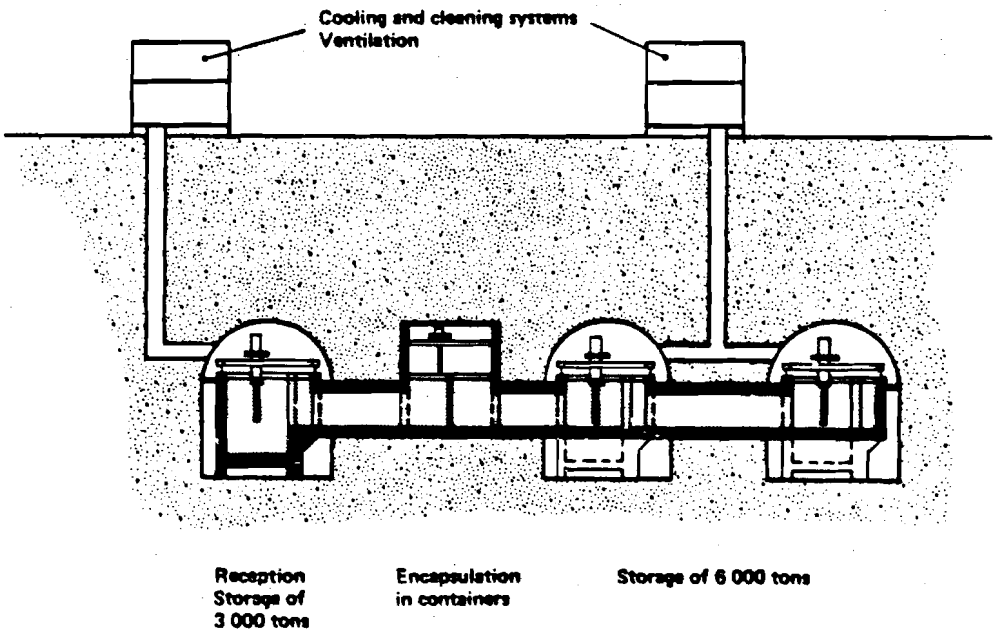


Figure B1-3. Intermediate storage facility for spent fuel.

one element in each container. The container is fitted with a lid, which is welded tightly. Following quality control, the container with the fuel elements is lowered into the water in the channel underneath the cell and placed in a cassette with room for a number of containers.

The cassette with the fuel containers is carried on a wagon which runs on tracks in a channel to the rock cavern in which the containers are to be stored. The rock cavern contains storage pools similar to those found in the central fuel storage facility. The cassette is placed in position in a pool by means of an overhead crane. The intermediate storage facility has two rock caverns, each with a storage capacity corresponding to 3 000 metric tons of uranium. The total capacity of the facility is thus 6 000 metric tons.

The central fuel storage facility will have a capacity of 3 000 metric tons. If the total quantity of fuel to be stored is 9 000 tons, these two pools may also be used for storing fuel containers, rendering further expansion of the intermediate storage facility unnecessary.

The fuel is stored dry in hermetically sealed containers. The water in the pools surrounding the containers provides adequate cooling and radiation shielding.

The advantage of dry storage is that the fuel is not exposed to the corrosive action of the water. The water in the pools is not contaminated by damaged fuel,so the demands on the filtering systems can be reduced.

After at least 30 years of storage, the fuel containers are checked for leakage and transferred to the encapsulation station at the final repository. The same transport casks which are used to transport fuel from the power station to the central fuel storage facility are used to transport the containers to the final repository.

If the fuel is to be sent away for reprocessing, the containers could be opened and the fuel elements removed in the same cell which was used to encase the fuel in the containers.

4.2 ENCAPSULATION STATION

In the encapsulation station, which is located above ground adjacent to the final repository, the fuel containers are encased in a copper canister before being deposited in the final repository.

The reason why a surface location has been chosen in this case is that the facility requires a relatively large building volume. Locating the facility underground would involve restrictions on, among other things, maximum spans, making it more difficult to optimize the design of the plant. The quantity of fuel which can be handled in the station at any one time is relatively small. Furthermore, the fuel is continuously enclosed, either in the container in which it was placed in the intermediate storage facility or, at a later stage, by the copper canister as well.

The design of the facility is shown in figure B1-4.

The transport cask arrives from the central fuel storage facility at the station's receiving section, where it is lifted from its trailer, cooled and washed. It is then lowered in two stages into a pool and placed on a wagon which takes it to an unloading position. Here, the fuel containers are lifted out of the water into a cell, where they are dried, inspected and placed in a copper canister. The canister shown in fig. B1-1 can hold 7 BWR elements. A canister which is 100 mm larger in diameter is used for PWR elements. It holds 4 PWR elements.

The filled canister is transferred via a lock, where it can be washed down with water, to an encapsulation cell. In this cell, the canister is fitted with a lid which is fastened by means of electron beam welding. The lid is in three parts, due to the penetration limit of existing equipment for this type of welding. It will probably be possible in the future to weld thicker material, in which case the lid can be made in two parts.

The weld is inspected ultrasonically and by means of helium leakage tracing. The canister is then ready to be transferred to the final repository.

4.3 FINAL REPOSITORY

As in the case of the reprocessing alternative, the final repository consists basically of a system of parallel storage tunnels situated approximately 500 metres below the surface, with appurtenant transport and service tunnels and shafts (see fig. B1-5). The storage tunnels, however, are of greater height - 4.9 m -and the repository occupies a slightly larger area - 1.2 km^2.

The repository has been designed for horizontal deposition, i.e. the canisters are emplaced horizontally in the longitudinal direction of the tunnel (see figs. B1-6 and B1-7). Because the canister is so long (4.9 m), deposition in vertical holes - as in the vitrification alternative - is less convenient and requires considerably greater tunnel height. Horizontal deposition also permits a much thicker layer of buffer material (sand/bentonite) around the canister than is possible with a vertical, drilled hole. As was noted in section III:6.3, a compacted sand/bentonite fill possesses such low permeability and other properties that only very small quantities of water can come into contact with the canister.

The quartz sand and bentonite filler is deposited as follows: first, a bed of material is laid down and compacted in the same manner as the core of an earth dam for a hydroelectric power station. The copper canister, which is transported from the encapsulation station down to the level of the final repository via an elevator and then on to the deposition site via a specially designed vehicle, is then deposited in a channel in the bed (see fig. B1-7). The channel is formed by compacting the upper part of the bed around a dummy canister with the same dimensions as a real canister. When all canisters have been emplaced in this man-

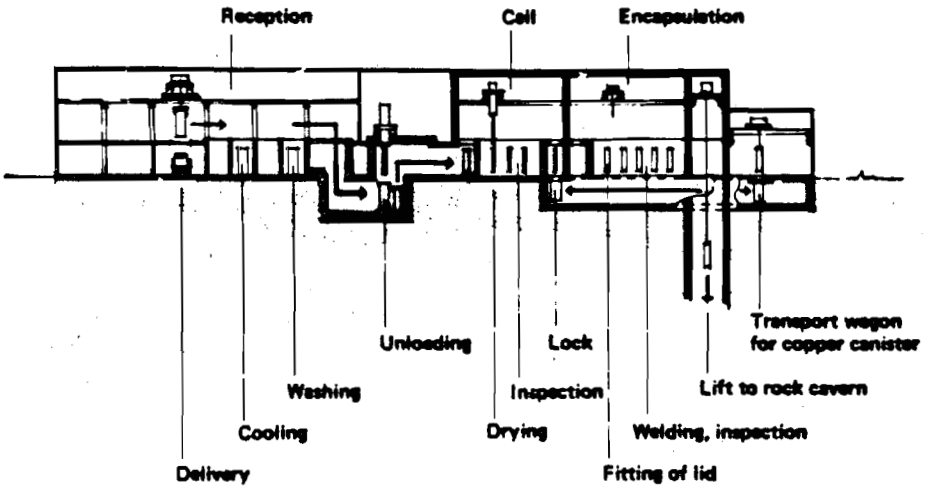
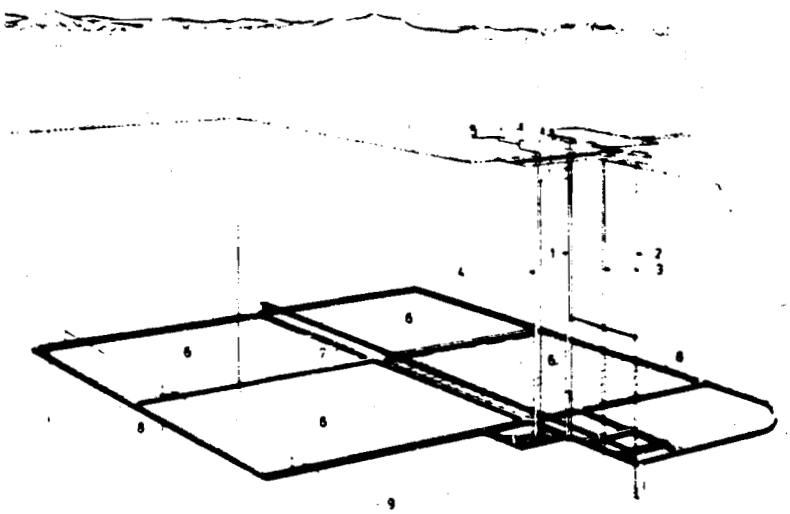


Figure B1-4. Encapsulation station for direct disposal of spent fuel.



1 Central shaft
2 Skip shaft
3 Ventilation shaft
4 Hoist shaft for waste canisters
5 Receiving and encapsulation station
6 Final repository
7 Central access tunnel
8 Outer access tunnel
9 Access and storage tunnels

Figure b1-5. Final repository for direct disposal of spent fuel.

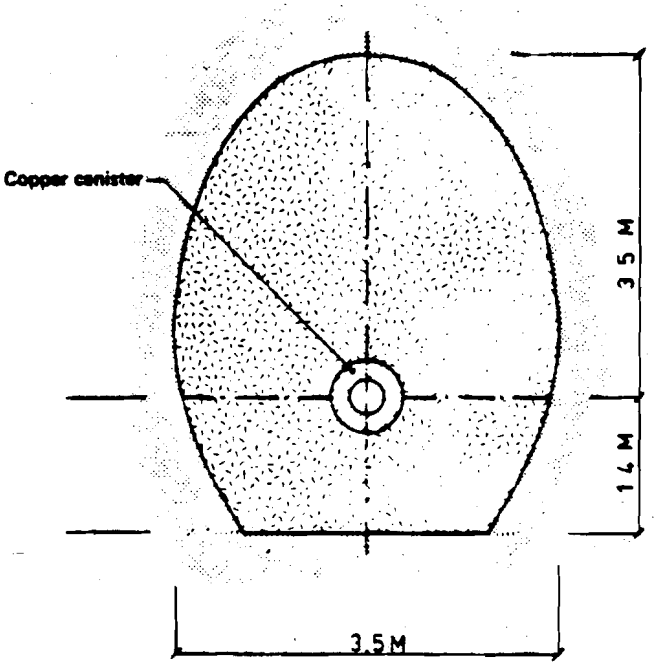


Figure B1-6. Storage tunnel for direct disposal of spent fuel. The copper canister is placed horizontally. Buffer material is placed in layers to a height of 1.4 m in the tunnel and compacted. The top part of the tunnel is filled by spraying of buffer material. (Cf. figure B1-7).

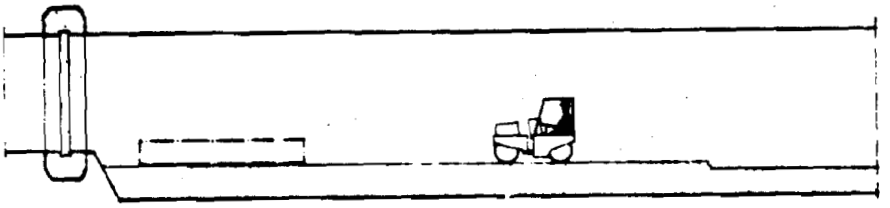
ner in a storage tunnel, the tunnel is filled completely with sand/bentonite, which is applied by spraying. The ends of the storage tunnel are sealed with concrete walls.

All work associated with the deposition of canisters and the spraying of buffer material is performed with equipment in which the personnel are protected against radiation. A temporary radiation shield can be positioned above the canisters when they are emplaced in order to permit persons to enter the tunnel without radiation protection. This temporary protection is then removed when the canisters are to be covered with buffer material.

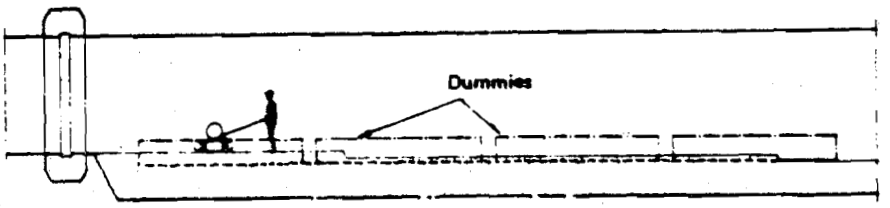
When the entire repository has been filled with canisters, transport tunnels, shafts and other cavities in the rock are sealed in the same way as was described for the vitrification alternative in section III:6.7.

5 TRANSPORTATION SYSTEM

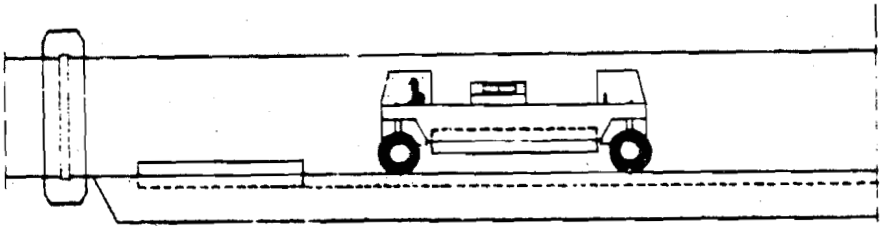
The same transportation system as is used for transporting spent fuel from the nuclear power plants to the central storage facility can be used for transporting the fuel from the intermediate storage facility to the encapsulation station at the final repository. The required number of transport casks and sea transports is also the same.



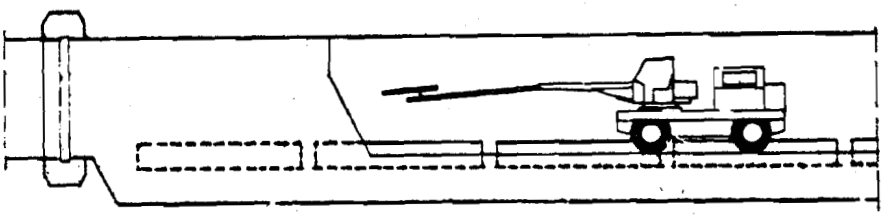
Laying of under bed



Laying of side-bed



Placement of canisters



Spraying of buffer material

Figure B1-7. Filling of storage tunnels for spent fuel canisters with buffer material.

6 SAFETY ANALYSIS

6.1 GENERAL

The main differences between the direct disposal of spent fuel and the final storage of vitrified high-level waste which may be pertinent to safety considerations are:

- The amounts of uranium and plutonium which are deposited as waste in direct disposal are 200 times greater than in the case of high-level vitrified waste. The waste also contains a number of other radioactive products which would otherwise be separated in reprocessing.
- The first barrier consists of the relatively insoluble fuel and its cladding instead of the borosilicate glass.
- The canister is made of copper or a ceramic material.
- The canister is deposited horizontally in the storage tunnel and embedded in a buffer material of considerably greater thickness than in the case of vitrified reprocessed waste.

6.2 RADIONUCLIDE INVENTORIES

In the direct disposal of spent fuel, uranium and plutonium are deposited as waste. One secondary effect is that radium is formed as a daughter product of uranium. Krypton-85 and the tritium and carbon-14 which is left in the fuel are also deposited.

The radionuclide inventories for the two alternatives are reported in chapter IV:3.

6.3 THE FUEL AS A BARRIER

The uranium dioxide fuel possesses very low solubility in water. 90-99.9% of the fission products are present in the uranium dioxide itself and are therefore inaccessible for leakage in the event of canister failure. Practical experience is available from many years of storing damaged fuel canisters in pools of water. This experience shows that such storage entails little release of radioactivity.

Special experimental studies are being conducted at Batelle Northwest Laboratories in Richland in the USA aimed at determining the leaching rate of irradiated fuel. Preliminary results indicate a leaching rate which is comparable to that for borosilicate glass. A special experimental study is also being conducted at Studsvik with a modified technique which also permits comparisons.

6.4 THE CANISTER

Encapsulation of the fuel in a 20 cm thick copper container provides a highly durable barrier against penetration by water. The copper shell can only corrode in contact with oxygenated water. Water at great depth usually contains only small quantities of oxygen. Low levels of oxidants can be formed by radiolysis. The

importance of this phenomenon from the viewpoint of corrosion is being studied. Equilibrium levels are considerably lower than the solubility of oxygen and hydrogen. This means that no gas will be evolved.

The thick canister reduces the radiation field on the outside and provides good mechanical protection. The canisters are sealed by several lids in order to guard against weld defects. The probability of initial canister damage and the migration of radiolysis products to nearby canisters is currently under investigation.

Encasement in an aluminium oxide canister provides a highly corrosion-resistant barrier. Technical fabrication considerations make it desirable to limit the length of the canister to about 3 metres, which means that the fuel must be converted into shorter units prior to encapsulation. This is to be done in a special handling process. In order to counteract the risk of dispersal and release of radioactive substances, this handling is performed in concrete cells with special ventilation systems and filters. A special safety analysis for this process will be carried out.

6.5 FINAL STORAGE

Emplacement in the centre of the tunnel section in a metre-thick bed of quartz sand and bentonite provides protection against possible fault movements and contributes towards the sorption of escaping nuclides.

APPENDIX 2 CONTRACTED AND CONSULTING COMPANIES, INSTITUTIONS AND EXPERTS

INDUSTRIAL AND CONSULTING COMPANIES

AB Atomenergi (= Atomic Energy Company of Sweden)
Ahlsell AB
ASEA
ASEA-ATOM AB
ASEA-Hafo AB
ASEA-Kabel AB
Avesta Jernverks AB
Forsgren Produktion AB
The Mining Industries' Work Study and Consultancy Programme
Hagby Bruk AB
H Folke Sandelin AB
Hagconsult AB
Hydroconsult AB
IFÖ Electric Högspänning AB
IPA-Konsult
Kemakta Konsult AB
Orrje & Co AB, Ingenjörsfirman
Salénrederierna AB
Stabilator AB
The National Swedish Institute for Materials Testing
Ställbergs Grufveaktibolag
The Swedish Academy of Engineering Sciences
Teleplan AB
AB Vattenbyggnadsbyrån
WP-system AB

INSTITUTIONS OF HIGHER LEARNING

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University of Gothenburg

Division of marine microbiology — Professor K Gundersen

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Department of inorganic chemistry	Professor N-G Vannerberg

Royal Institute of Technology

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Department of agricultural hydrotechnics	Professor Y Gustafsson Professor I Larsson
Department of nuclear chemistry	Dr. T E Eriksen
Department of inorganic chemistry	Professor I Grenthe
Department of engineering electrochemistry and corrosion science	Professor G Wranglén

Luleå Institute of Technology

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Division of geotechnics	Professor R Pusch A Jacobsson, Ph.D.

OTHER INSTITUTIONS

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National Defence Research Institute, Stockholm	G Walinder
Glass Research Institute, Växjö	B Simmingsköld, Lic. Eng. T Lakatos, Ph.D.

Institute for Metals Research	G Eklund, D.Eng.
Corrosion research Institute, Stockholm	Professor E Mattsson L Ekblom, Lic.Eng.
Swedish College of Agriculture, Uppsala	Professor L Fredriksson
Swedish Silicate Institute, Gothenburg	R Carlsson, Lic. Eng.
Tandem accelerator laboratory in Uppsala	Professor A Johansson
Geological Survey of Sweden	Dr. O Brotzen U Thoregren, Ph. L.

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Claes Helgesson, D. Eng.

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KJELLER, Norway

Frauenhofer-Gesellschaft
Institut für Silicatforschung
WÜRZBURG, West Germany

Lehrstuhl für Glas und Keramik
Institut für Steine und Erder
CALUSTAHL-ZELLEFELD, West Germany

Saint Gobain Techniques Nouvelles
PARIS, France

COGEMA
PARIS, France

University of Western Ontario
Professor W Fyfe
LONDON, Ontario, Canada

Corning Glass Works
CORNING, New York, USA

Electric Power Research Institute
PALO ALTO, California, USA

Department of Energy
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Dr. Ralph E Grim
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Lawrence Berkeley Laboratory
University of California
BERKELY, California, USA

Nuclear Regulatory Commission
WASHINGTON D.C., USA

Office of Waste Isolation
Union Carbide Corp.
OAK RIDGE, Tennessee

Personnel from the Swedish State Power Board, Oskarshamnsverkets Kraftgrupp AB (OKG), the Swedish Nuclear Fuel Supplies Company (SKBF), the Swedish Central Power Supply Board (CDL) and the National Council for Radioactive Waste Management (PRAV) have also contributed their services to KBS. The project has organized reference and working groups within the following areas:

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Dr. Rudolf Hiltscher	Saltsjö-Boo
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Dr. Thorbjörn Thedéen	University of Stockholm
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Göran Eklund, D. Eng.	Institute for Metals Research
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Arvid Jacobsson, Ph. D.	Luleå Institute of Technology
Professor Ivars Neretnieks	Royal Institute of Technology
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PROGRAMME GROUP P21/22

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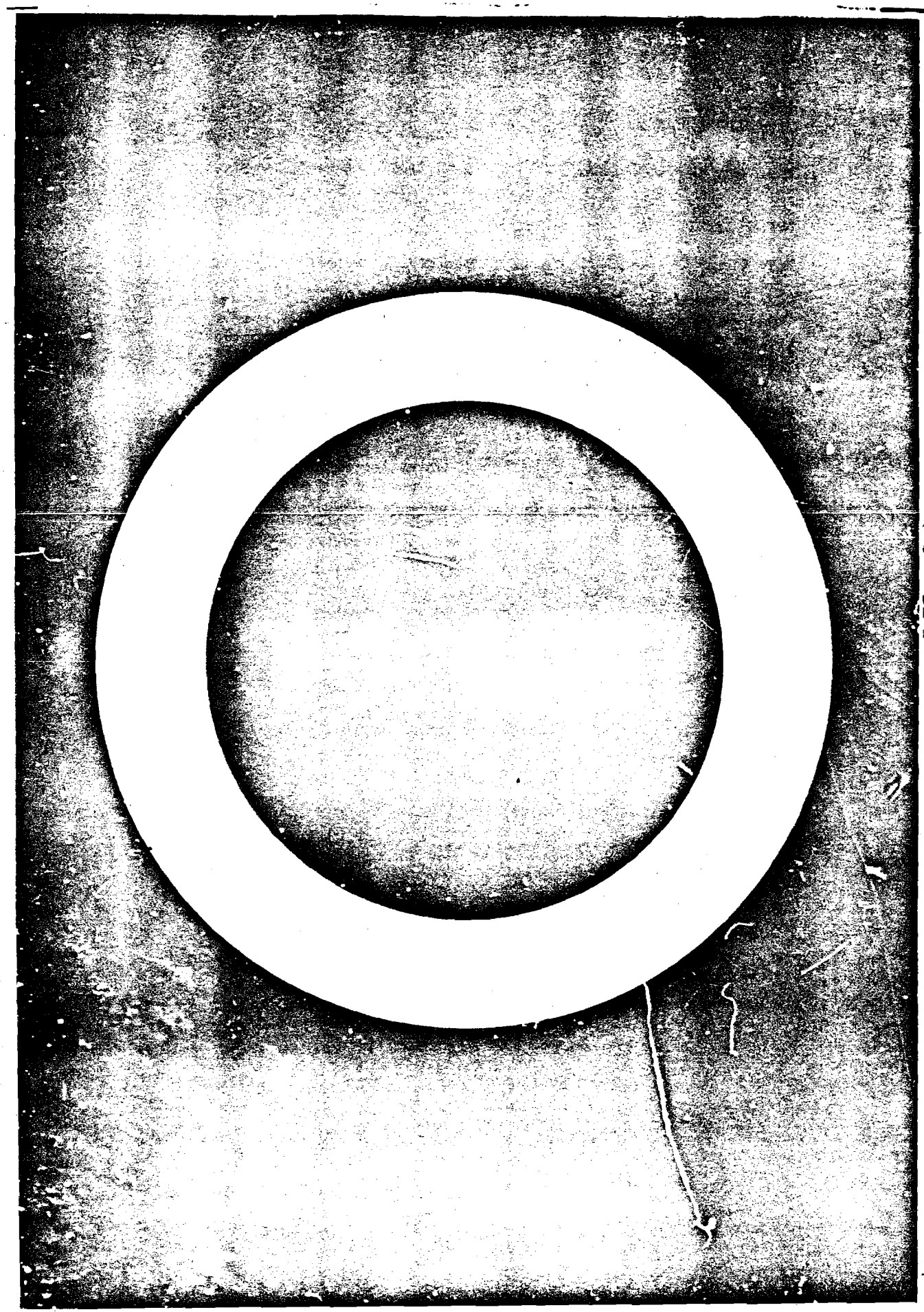
Bertil Mandahl, Eng.	OKG



APPENDIX 3 LIST OF KBS TECHNICAL REPORTS

01 Källstyrkor i utbränt bränsle och högaktivt avfall från en PWR beräknade med ORIGEN
("Emission rates in spent fuel and high-level waste from a PWR, calculated using ORIGEN")
Nils Kjellbert
AB Atomenergi, 77-04-05

02 PM angående värmeledningstal hos jordmaterial
("Memorandum concerning the thermal conductivity of soil")
Sven Knutsson
Roland Pusch
Luleå Institute of Technology, 77-04-15

03 Deponering av högaktivt avfall i borrhål med buffertsubstans
("Deposition of high-level waste in boreholes containing buffer material")
Arvid Jacobsson
Roland Pusch
Luleå Institute of Technology, 77-05-27

04 Deponering av högaktivt avfall i tunnlar med buffertsubstans
("Deposition of high-level waste in tunnels containing buffer material")
Arvid Jacobsson
Roland Pusch
Luleå Institute of Technology 77-06-01

05 Orienterande temperaturberäkningar för slutförvaring i berg av radioaktivt avfall, Report 1
("Preliminary temperature calculations for the final storage of radioactive waste in rock")
Roland Blomqvist
AB Atomenergi, 77-03-17

06 Groundwater movements around a repository, Phase 1, State of the art and detailed study plan
Ulf Lindblom
Hagconsult AB, 77-02-28

07 Resteffekt studier för KBS ("Decay power studies for KBS")
Del 1 Litteraturgenomgång ("Part 1 Review of the literature")
Del 2 Beräkningar ("Part 2 Calculations")
Kim Ekberg
Nils Kjellbert
Göran Olsson
AB Atomenergi, 77-04-19

08 Utlakning av franskt, engelskt och kanadensiskt glas med högaktivt avfall
("Leaching of French, English and Canadian glass containing high-level waste")
Göran Blomqvist
AB Atomenergi, 77-05-20

09 Diffusion of soluble materials in a fluid filling a porous medium
Hans Häggblom
AB Atomenergi, 77-03-24

10 Translation and development of the BNWL-Geosphere Model
Bertil Grundfelt
Kemakta Konsult AB, 77-02-05

11 Utredning rörande titans lämplighet som korrosionshärdig kapsling för kärnbränsleavfall
("Study of suitability of titanium as corrosion-resistant cladding for nuclear fuel waste")
Sture Henriksson
AB Atomenergi, 77-04-18

12 Bedömning av egenskaper och funktion hos betong i samband med slutlig förvaring av kärnbränsleavfall i berg
("Evaluation of properties and function of concrete in connection with final storage of nuclear fuel waste in rock")
Sven G Bergström
Göran Fagerlund
Lars Rombén
The Swedish Cement and Concrete Research Institute, 77-06-22

13 Urlakning av använt kärnbränsle (bestrålad uranoxid) vid direktdeponering
("Leaching of spent nuclear fuel (irradiated uranium oxide) following direct deposition")
Ragnar Gelin
AB Atomenergi, 77-06-08

14 Influence of cementation on the deformation properties of bentonite/quartz buffer substance
Roland Pusch
Luleå Institute of Technology, 77-06-20

15 Orienterande temperaturberäkningar för slutförvaring i berg av radioaktivt avfall
("Preliminary temperature calculations for the final storage of radioactive waste in rock")
Report 2
Roland Blomquist
AB Atomenergi, 77-05-17

16 Översikt av utländska riskanalyser samt planer och projekt rörande slutförvaring
("Review of foreign risk analyses and plans and projects concerning final storage")
Åke Hultgren
AB Atomenergi, August 1977

17 The gravity field in Fennoscandia and postglacial crustal movements
Arne Bjerhammar
Stockholm, August 1977

18 Rörelser och instabiliteter i den svenska berggrunden
("Movements and instability in the Swedish bedrock")
Nils-Axel Mörner
University of Stockholm, August 1977

19 Studier av neoteknisk aktivitet i mellersta och norra Sverige, flygbildsgenomgång och geofysisk tolkning av recenta förkastningar
("Studies of neotectonic activities in central and northern Sweden, review of aerial photographs and geophysical interpretation of recent faults")
Robert Lagerbäck
Herbert Henkel
Geological Survey of Sweden, September 1977

20 Tektonisk analys av södra Sverige, Vättern - Norra Skåne
("Tectonic analysis of southern Sweden, Lake Vättern - Northern Skåne)
Kennert Röshoff
Erik Lagerlund
University of Lund and Luleå Institute of Technology, September 1977

21 Earthquakes of Sweden 1891 - 1957, 1963 - 1972
Ota Kulhánek
Rutger Wahlström
University of Uppsala, September 1977

22 The influence of rock movement on the stress/strain situation in tunnels or boreholes with radioactive canisters embedded in a bentonite/quartz buffer mass
Roland Pusch
Luleå Institute of Technology, 1977-08-22

23 Water uptake in a bentonite buffer mass
A model study
Roland Pusch
Luleå Institute of Technology, 1977-08-22

24 Beräkning av utlakning av vissa fissionsprodukter och aktinider från en cylinder av franskt glas
("Calculation of leaching of certain fission productr and actinides from a cylinder made of French glass")
Göran Blomqvist
AB Atomenergi, 1977-07-27

25 Blekinge kustgnejs, Geologi och hydrogeologi
("The Blekinge coastal gneiss, Geology and hydrogeology")
Ingemar Larsson — Royal Institute of Technology
Tom Lundgren — Swedish Geotechnical Institute
Ulf Wiklander — Geological Survey of Sweden
Stockholm, August 1977

26 Bedömning av risken för fördröjt brott i titan
("Evaluation of risk of delayed fracture of titanium")
Kjell Pettersson
AB Atomenergi, 1977-08-25

27 A short review of the formation, stability and cementing properties of natural zeolites
Arvid Jacobsson
Luleå Institute of Technology, 1977-10-20

28 Värmeledningsförsök på buffertsubstans av bentonit/pitesilt
("Thermoconductivity experiments with buffer material of bentonite/pitesilt")
Sven Knutsson
Luleå Institute of Technology, 1977-09-20

29 Deformationer i sprickigt berg
("Deformations in fissured rock")
Ove Stephansson
Luleå Institute of Technology, 1977-09-28

30 Retardation of escaping nuclides from a final depository
Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1977-09-14

31 Bedömning av korrosionsbeständigheten hos material avsedda för kapsling av kärnbränsleavfall.
("Evaluation of corrosion resistance of material intended for encapsulation of nuclear fuel waste".) Status report, 1977-09-27, and supplementary statements.
Swedish Corrosion Research Institute and its reference group

32 Long term mineralogical properties of bentonite/quartz buffer substance
Preliminary report, November 1977
Final report, February 1978
Roland Pusch
Arvid Jacobsson
Luleå Institute of Technology

33 Required physical and mechanical properties of buffer masses
Roland Pusch
Luleå Institute of Technology 1977-10-19

34 Tillverkning av bly-titan kapsel
("Fabrication of lead-titanium canister")
Folke Sandelin AB
VBB
ASEA-Kabel
Swedish Institute for Metals Research
Stockholm, November 1977

35 Project for the handling and storage of vitrified high-level waste
Saint Gobain Techniques Nouvelles, October 1977

36 Sammansättning av grundvatten på större djup i granitisk berggrund
("Composition of groundwater deep down in granitic bedrock")
Jan Rennerfelt
Orrje & Co, Stockholm, 1977-11-07

37 Hantering av buffertmaterial av bentonit och kvarts
("Handling of buffer material of bentonite and quartz")
Hans Fagerström, VBB
Björn Lundahl, Stabilator
Stockholm, October 1977

38 Utformning av bergrumsanläggningar
("Design of rock cavern facilities")
Arne Finné, KBS
Alf Engelbrektson, VBB
Stockholm, December 1977

39 Konstruktionsstudier, direktdeponering
("Design studies, direct deposition")
ASEA-ATOM
VBB
Västerås, Sweden

40 Ekologisk transport och stråldoser från grundvattenburna radioaktiva ämnen
("Ecological transport and radiation doses from groundwater-borne radioactive substances")
Ronny Bergman
Ulla Bergström
Sverker Evans
AB Atomenergi

41 Säkerhet och strålskydd inom kärnkraftområdet.
Lagar, normer och bedömningsgrunder
("Safety and radiation protection in the field of nuclear power. Laws, standards and grounds for evaluation")
Christina Gyllander
Siegfried F Johnson
Stig Rolandson
AB Atomenergi and ASEA-ATOM

42 Säkerhet vid hantering, lagring och transport av använt kärnbränsle och förglasat högaktivt avfall.
("Safety in the handling, storage and transportation of spent nuclear fuel and vitrified high-level waste")
Ann Margret Ericsson
Kemakta, November 1977

43 Transport av radioaktiva ämnen med grundvatten från ett bergförvar
("Transport of radioactive elements in groundwater from a rock repository")
Bertil Grundfelt
Kemakta, November 1977

44 Beständighet hos borsilikatglas
("Durability of borosilicate glass")
Tibor Lakatos
Glasteknisk Utveckling AB

45 Beräkning av temperaturer i ett envånings slutförvar i berg för förglasat radioaktivt avfall
("Calculation of temperatures in a single-level final repository in rock for vitrified radioactive waste")
Report 3
Roland Blomquist
AB Atomenergi, 1977-10-19

46 Temperaturberäkningar för använt bränsle
("Temperature calculations for spent fuel")
Taivo Tahrandi
VBB

47 Teoretiska studier av grundvattenrörelser
("Theoretical studies of groundwater movements")
Preliminary report, October 1977
Final report, February 1978
Lars Y Nilsson
John Stokes
Roger Thunvik
Department of agricultural hydrotechnics, Royal Institute of Technology

48 The mechanical properties of Stripa granite
Graham Swan
Luleå Institute of Technology, 1977-08-29

49 Bergspänningsmätningar i Stripa gruva
("Measurements of rock stresses in the Stripa mine")
Hans Carlsson
Luleå Institute of Technology, 1977-08-29

50 Lakningsförsök med högaktivt franskt glas i Studsvik
("Leaching trials with high-level French glass at Studsvik")
Göran Blomqvist
AB Atomenergi, November 1977

51 Seismotectonic risk modelling for nuclear waste disposal in the Swedish bedrock
F Ringdal
H Gjöystdal
E S Husebue
Royal Norwegian Council for scientific and industrial research

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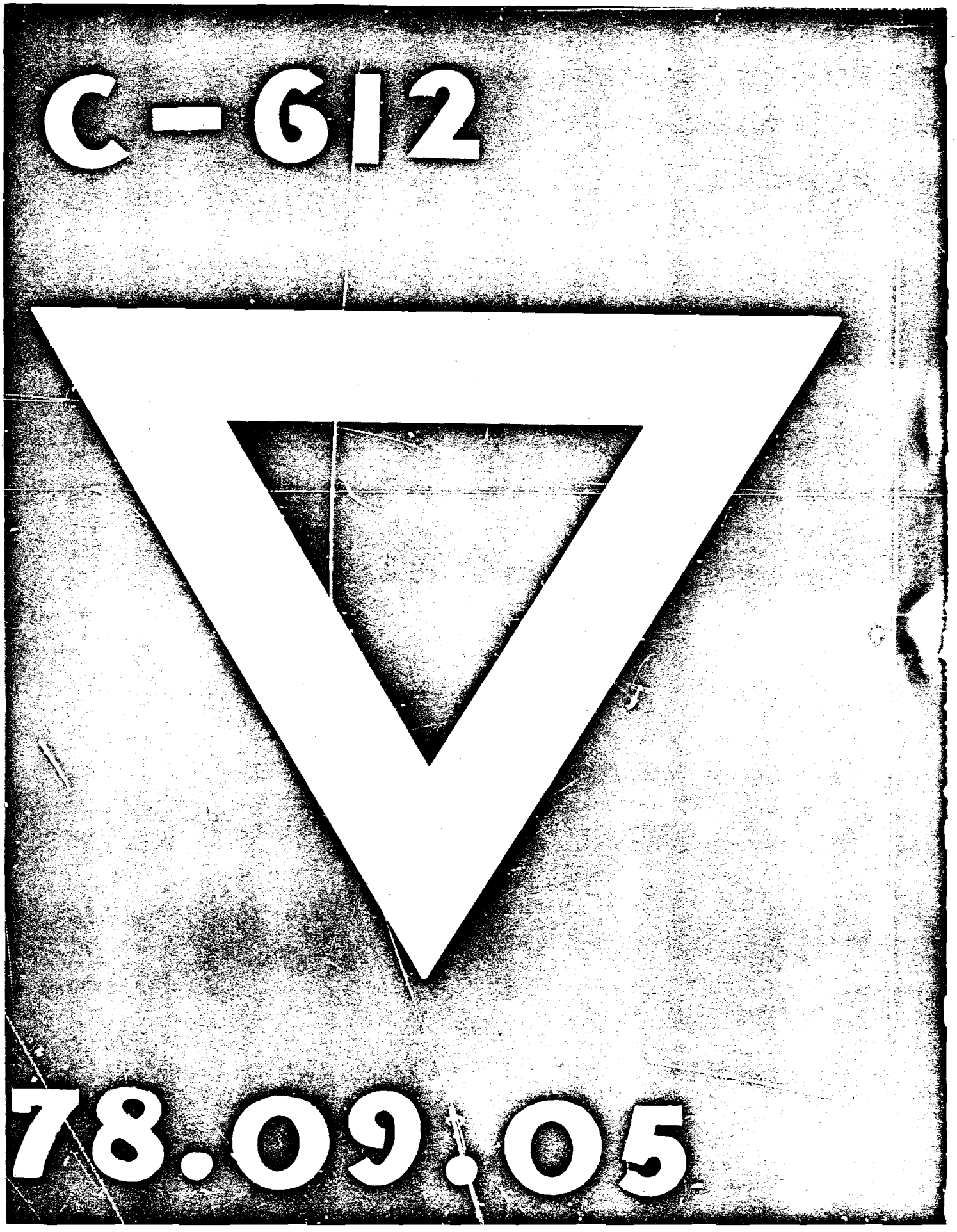
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Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste

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

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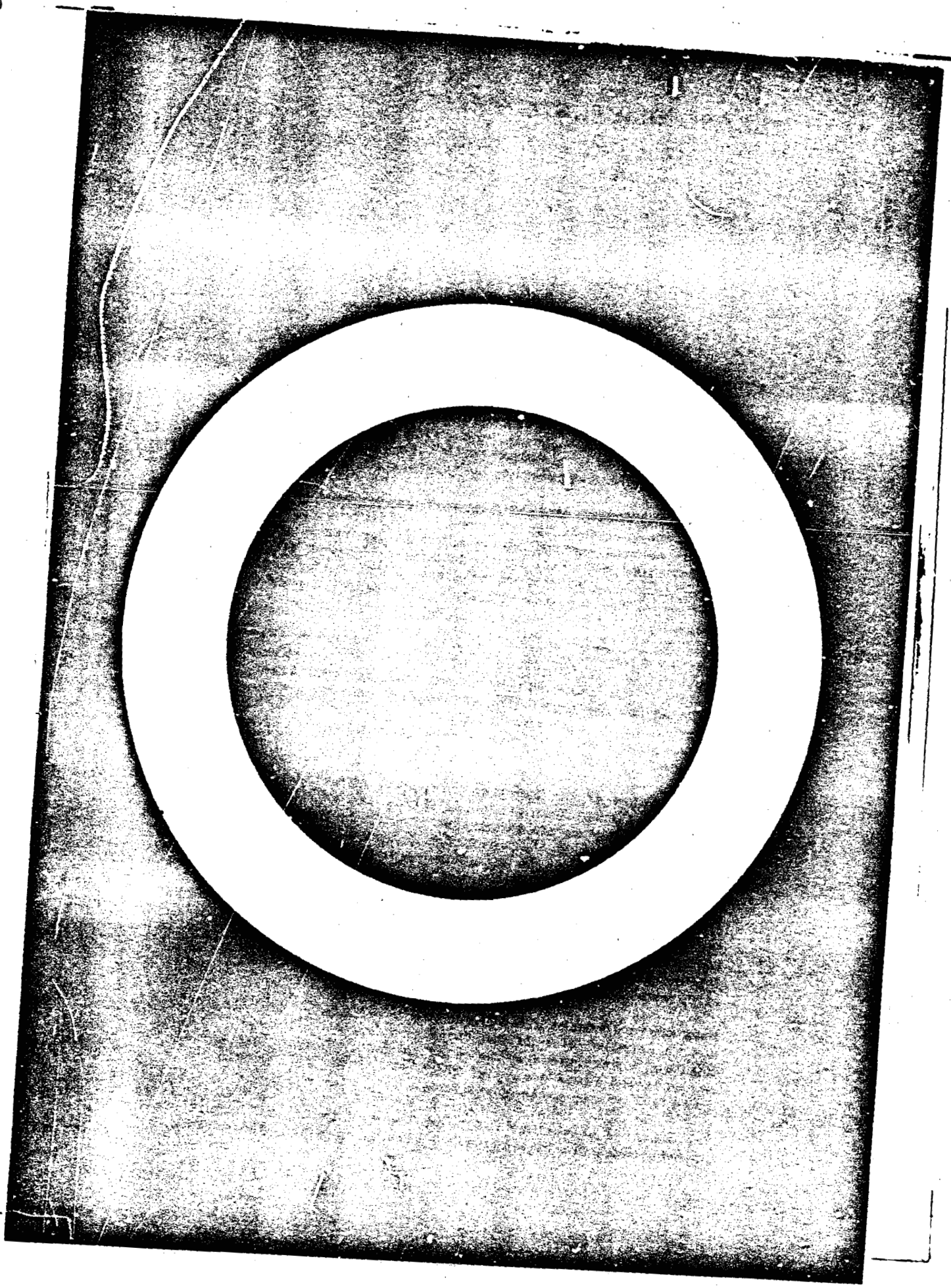
1 BACKGROUND

The possibility of disposing of high-level waste in geological formations has been under discussion for many years. It has generally been assumed that the waste is to be disposed of in the country where it was produced, and different types of formations have come under consideration in different countries, depending on their occurrence (salt, clay, shale, crystalline rock). In Sweden, interest has been concentrated on Precambrian bedrock formations (gneiss, granite).

The geological studies conducted for the AKA Committee (The Government Committee on Radioactive Waste) have been supplemented by studies performed by the National Council for Radioactive Waste Management (PRAV). Equipment for water injection testing and geophysical borehole logging as well as a special borehole pump have also been developed by PRAV. Since the KBS project commenced in early 1977, the geological investigations have been concentrated on this project.

In February of 1977, an agreement was reached between KBS and the Geological Survey of Sweden (SGU), which is the central government agency for geological matters, concerning a geological study programme aimed at ascertaining the feasibility of a final storage of high-level waste in the Swedish bedrock. A summary of the contents and principal results of the programme is provided in this chapter. More detailed accounts of the methods and results of the investigations are provided in the reports referred to under section A of the list of references.



2 THE GEOLOGICAL STUDY PROGRAMME

2.1 PURPOSE

The goal of the geological work programme is to carry out studies within several geographical areas in order to obtain basic data on the bedrock and groundwater conditions which determine the long-term safety of a final repository for high-level waste. These studies are complemented by theoretical studies.

The work is aimed at establishing whether the bedrock is composed of a uniform, suitable type of rock of sufficient extent down to depths of several hundred metres. This is important, since inferior conditions may exist at the boundaries between different types of rock. The occurrence of fissures and fracture zones with may influence the design or safety of the repository must also be elucidated.

As regards the groundwater, information is needed on how much water may come into contact with the waste. This requires measurements of the permeability of the rock and theoretical calculations of how the water flow in the rock decreases with depth. Such calculations also provide a basis for determining the final dilution of the water which has been in contact with the waste canisters.

When the waste or the canisters come into contact with the groundwater, some dissolution may occur. The extent of this dissolution depends on the properties of the materials and the chemical composition of the water. The programme therefore includes sampling and analysis of the groundwater at the depths in question.

If the waste substances go into solution, it is important to know their residence time in the rock. If this time is long, certain radioactive elements will decay before they reach the biosphere. Information on the residence time can be derived from the age of the groundwater.

Most waste elements are retarded on their way through the bedrock due to sorption effects and chemical reactions. Such retardation has been investigated by means of laboratory work, theoretical analyses and field tests, of which the latter fall within the framework of the geology programme.

The question of whether and to what extent geological conditions of importance for the safe final storage of high-level radioac-

tive waste may change in the future must be answered. This includes the study of the geographical and chronological distribution of movements in the bedrock.

The geological programme has concentrated primarily on gathering information to serve as a basis for the siting and design of an absolutely safe rock storage facility adapted to the conditions which prevail in Swedish bedrock. Time has not yet permitted a more fundamental analysis of the data.

2.2 SCOPE

SGU has been commissioned by KBS to perform the following main studies:

- Geophysical ground measurements, mapping of outcrops and joints, drilling, evaluation of drill cores, borehole logging and TV examination of boreholes.
- Water injection tests and calculations, water sampling for chemical analysis and age determination.
- Theoretical studies of groundwater movements (carried out through the Department of Land Improvement and Drainage at the Royal Institute of Technology in Stockholm).
- Field tests using tracer elements in fissured rock before and after injection (previously begun by PRAV).

The total cumulative length of drilled core boreholes amounts to slightly more than 5 000 m, distributed among five study areas, three of which have been chosen for further study - namely, Sternö near Karlshamn, Kråkemåla near Oskarshamn and Finnsjö near Forsmark. The bedrock in these areas varies and the choice of study areas was determined partially by the fact that knowledge was desired on the characteristics of the different types of rock.

In addition to the above studies by SGU, KBS has commissioned the following:

- a compilation and supplementation of known data on the Blekinge coastal gneiss,
- a mathematical model study of groundwater movements and rock stresses in and around a final repository,
- a theoretical mathematical study of the expected formation of new fractures when a rock mass is subjected to simple shearing,
- studies of the chemical composition of the groundwater,
- studies of sorption effects which may be encountered when various waste substances are transported with the groundwater in buffer material and rock fissures,
- studies of post-precambrian rock movements and recent earthquakes.

The results of these studies will complement the results of SGU's own studies.

In early February of 1977, KBS and PRAV invited a large number of geologists to a conference for a discussion of questions of importance concerning a final repository for high-level waste in the Swedish bedrock. Among other things, the probability that

movements in the bedrock would jeopardize the safety of the repository was discussed. A number of proposals for studies were submitted by the conference participants and many of these proposals led to investigations sponsored by KBS. In early October of 1977, the results were reported and discussed at a second conference arranged by KBS.

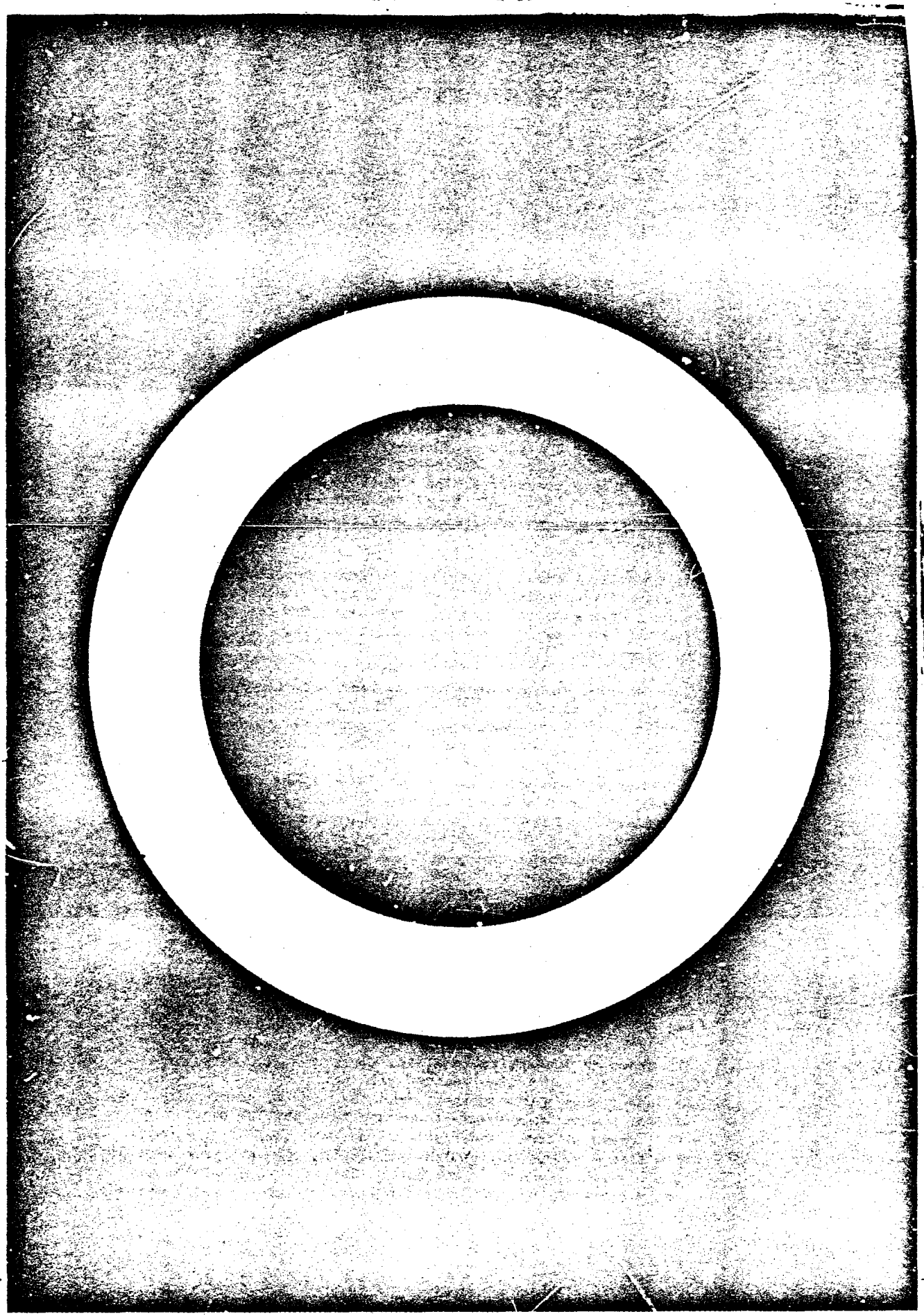
A group of specialists called the "Geogroup" has been established within KBS. Its function is to:

- serve as a forum for the discussion of questions concerning geology, hydrogeology and rock engineering,
- participate in the formulation of plans for investigations and experiments,
- assist in the follow-up and evaluation of results of investigations and experiments.

The report submitted in the following chapters 2 through 8 has been prepared by SGU. It is based not only on SGU's own studies, but also on the above-mentioned special investigations carried out on commission from KBS.

2.3 STRIPA EXPERIMENTAL STATION

The experimental programme being conducted by KBS at the Stripa mine is described in chapter 9. Mining operations were recently discontinued at the mine and the opportunity was offered to conduct practical experiments in a granite massif at a depth of about 350 m. These experiments are aimed at studying the properties of granite at this depth, both in the unconditioned state and after heating, and at studying fracturing resulting from the blasting of tunnels. An agreement has been concluded with the US Energy Research and Development Administration (ERDA) concerning cooperation in the execution of a large-scale heating experiment. The results of this experiment will not be available for a couple of years. KBS' own experiments also require such a long time that the most relevant results cannot be reported here. But the experiments are of such a nature that they will not affect the basic conclusions, but will rather primarily serve as a basis for an optimization of the detailed technical design of a final repository.



3 CHOICE OF STUDY AREAS

The preliminary work for the studies has been concentrated on finding areas with suitable bedrock of sufficient extent near the east coast of Sweden between Uppland and Blekinge. On the basis of the selected design capacity - waste from the operation of 13 reactors over a period of 30 years - an area of about 1 km^2 is required. Proximity to the coast is desirable in order to avoid long overland transports. The locations of the nuclear power plants and the desirability of avoiding seismically active areas has restricted the preliminary work to the east coast between Uppland and Blekinge.

In order to be able to complete in-depth drillings and studies in the short time which was available, it was necessary to exclude a number of geologically promising areas where the ownership of the land are complex or where it was not possible to obtain the permission of the proprietor.

Flat areas with much exposed rock have been sought after. The gradient of the water table in such areas is generally low, resulting in a low potential for groundwater movements. Another factor of importance is that the fracture zones in the bedrock in such areas are also normally widely spaced and narrow, with large intervening volumes of good rock. Between the large fracture zones, the bedrock should contain relatively widely spaced, small and irregular fissures, so that the groundwater permeability of the rock is low. This can be studied where the rock is exposed in outcrops. Outcrop mapping also shows whether the bedrock is uniform and consists of some common type of rock - granite or gneiss - which is of little value and therefore unattractive for future mining. The final areas were chosen to provide examples of the conditions in a massive granite which postdates orogenic upthrusting and folding, a gneissic granite and a clearly folded gneiss. Together with the vein gneiss previously studied for the AKA Committee investigation, this makes a total of four different typical types of precambrian bedrock which have been investigated.

In order to find suitable areas, topographical, economical and geological maps - supplemented with satellite and aerial photographs - have been examined. This has made it possible to compare terrain, ownership relation and the distribution of different types of rock in different areas and to chart the major fracture lines in the bedrock. Following this analysis promising areas were inspected in the field. In order to determine the features of the bedrock in unexposed parts and in depth as well,

geophysical measurements were carried out, primarily using electromagnetic methods which detect the groundwater-bearing zones in the bedrock on the basis of their increased electrical conductivity.

In certain cases, seismic methods were used to measure the velocity of shock and shear waves in the bedrock in order to gain some insight of its elasticity constant and fissure content.

Three areas were finally chosen for further study and drilling (see fig. 3-1).

- Karlshamn, i.e. the area around the Karlshamn power plant.
- Finnsjön, an area east of the northern part of Finnsjö Lake, some 16 km WSW of the Forsmark nuclear power plant in northeastern Uppland.
- Kråkemåla, an area 1.5 km NW of Kråkemåla and 7.5 km NNW of the Oskarshamn nuclear power plant at Simpevarp in eastern Småland.

In addition, certain studies were conducted at the following places:

- Ävrö, about 1.5 km N of Simpevarp
- Bussvik, about 4.5 km NNW of Simpevarp
- Forsmark, about 3.5 km W of the Forsmark nuclear power plant.

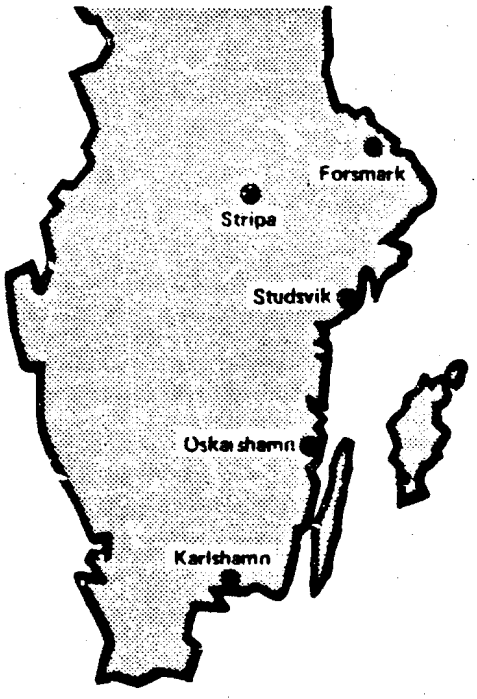


Figure 3-1. Map showing study areas. Test drillings to a depth of about 500 metres were undertaken at Karlshamn (Sternö), north of Oskarshamn (Kråkemåla and Ävrö), and at Forsmark (Finnsjön and Forsmark). The KBS experimental station is located in the Stripa mine. Field studies were carried out at Studsvik.

graphs. Flat cracks of varying direction and dips up to 30° generally comprise the top surfaces of the rock slab. The varying direction of the steep cracks is shown in the fracture diagram (see fig. 4-4), which shows broad and low peaks in the NW and NE. The peak at N30°W corresponds to the direction of the Finnsjö line and the fault valley which borders the study area on the east. The peaks around N60°W and N45°E correspond to more regular, straight and intersecting fractures. N60°W characterizes the metabasites in the study area, but also an important group of partially open, wide and very long fracture zones which are spread over this entire part of the country from Singö and Forsmark to the Storvik region /see Svedmark 4-8, Lundegårdh 4-9/. They are characterized by the fact that the openings are often lined with beautiful crystals of quartz and calcite. Small quantities of bitumenous substances, known as "rock pitch", are also found. The fractures which run in this direction in the outcrops in the Finnsjö area are filled with the same material. A horizontal longitudinal displacement of 0.3 m has been found for one of these faults. But the faults here are few and insignificant and are generally less than 1 cm wide. It must instead be assumed that whatever major fracture zones may exist are not visible in the outcrops, but are rather located in the soil-covered zones between them. The aerial photographs have therefore been examined carefully to determine the location of any fracture lines (see fig. 4-5). Ground examination has shown that these lines for the most part correspond to the edges of outcrops, bog lines and zones of heavy plant growth. Their direction coincides with the direction of the aforementioned shear zones. The only really clear line from the study area runs in a north-south direction

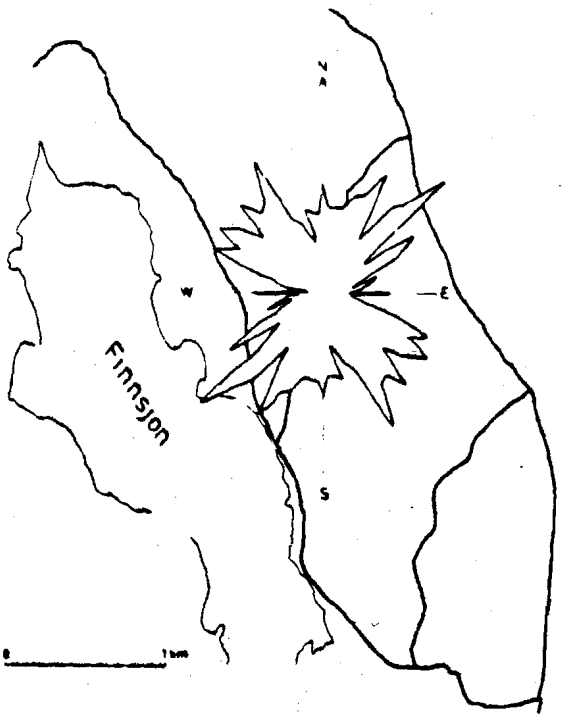


Figure 4-4. Diagram showing the direction of 448 steep fractures within the study area at Finnsjö Lake. The fractures are irregularly oriented in widely varying directions. (Bedrock Bureau, Geological Survey of Sweden).

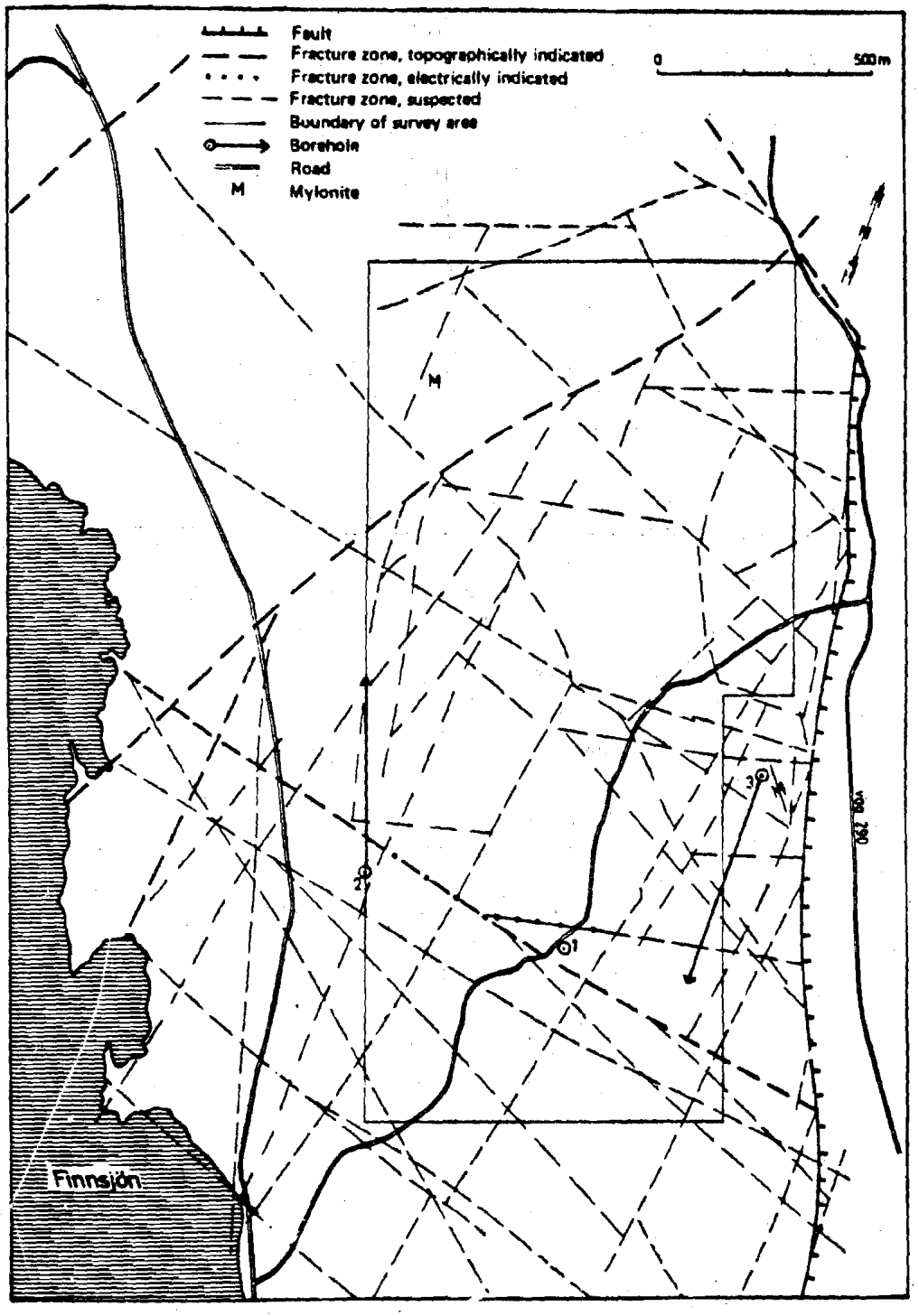


Figure 4-5. Map of the Finnsjö Lake area showing the distribution of fracture lines and the location of boreholes. The indicated survey area was examined by means of magnetic and electrical methods. (Bedrock Bureau, Geological Survey of Sweden).

and exhibits a "horsetail" towards the south. In order to gain further knowledge of the structure of the rock, the area has been covered by geophysical measurements and examined by means of drilling.

4.3.3 Geophysical ground measurements

An area nearly 2 km^2 in size has been surveyed with a magnetometer and an electromagnetic method (slingram). These magnetic measurements do not indicate any distinctive structures. The slingram measurement shows that there is one distinct zone of higher electrical conductivity. It is located in the southern half of the area and runs with some interruptions from near its western edge for approximately 500 metres in an easterly direction and seems to coincide with the photogeologically most clearly indicated east-west line within the area. Other lines have not given any magnetic loop indications.

4.3.4 Drillings

Finnsjö 1 Core Borehole - length 500 m, diameter 56 mm - sunk vertically in good rock, approx. 50 m from the interpolated core line for the electrically conductive zone indicated by slingram measurement.

In brief, the borehole shows that the bedrock is uniform and consists of granodiorite with insignificant variations down to a depth of 500 m. It contains insignificant pockets of isolated, thin pegmatites and metabasites. Below 85 m, the fissure content of the rock is low. More high-fissured zones are found in the sections 214-228 m, 336-362 m and more generally between 432 and 500 m. To a great extent, the fissures are filled with chlorite, quartz (SiO_2) and calcite ($CaCO_3$). The zones of disturbance exhibit a general transformation with reddish coloration of the feldspars.

The permeability of the rock is low, around 10^{-9} m/s or lower. Higher values are found in the surface layer and below 432 m, where a slight increase is noted in connection with a rising fissure content and chlorite content. This is probably connected with the fact that the borehole at this point approaches an electrically conductive zone, as indicated by slingram measurement from the surface. Permeability and RQD diagrams are shown in fig. 4-6.

Finnsjö 2 Core Borehole - diameter 56 mm, length 698 m, depth approx. 525 m - sunk at a 50° angle towards N20°W. It is situated at the western edge of the survey area and crosses the eastwest fracture zones, including the extention of the electrical disturbance zone towards the west. In this way, information is obtained on the boundary of the low-fissured central bedrock blocks towards Finnsjö Lake. At the far north, the hole was also expected to contact the southern branches of the major zone of disturbance which runs through the measuring area in a northeasterly direction without giving rise to any electrical indications.

Bedrock conditions in this borehole are very similar to those in borehole 1. Between 36 and 110 m, however, there is a general

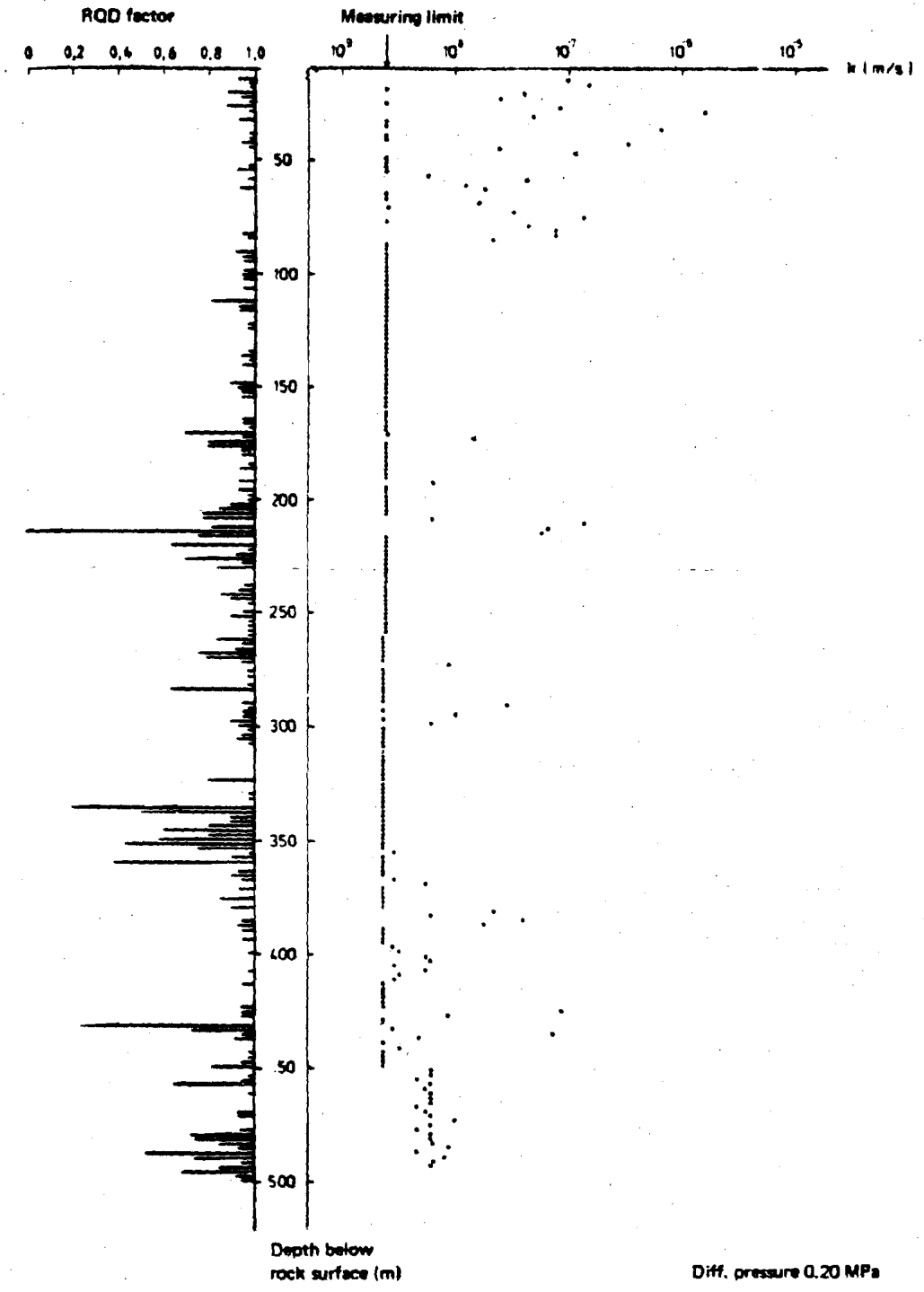


Figure 4-6. Diagram of rock permeability (right) and fissure content expressed as RQD factor (left) in core borehole 1 at Finnsjö Lake (Bedrock Bureau, Geological Survey of Sweden).

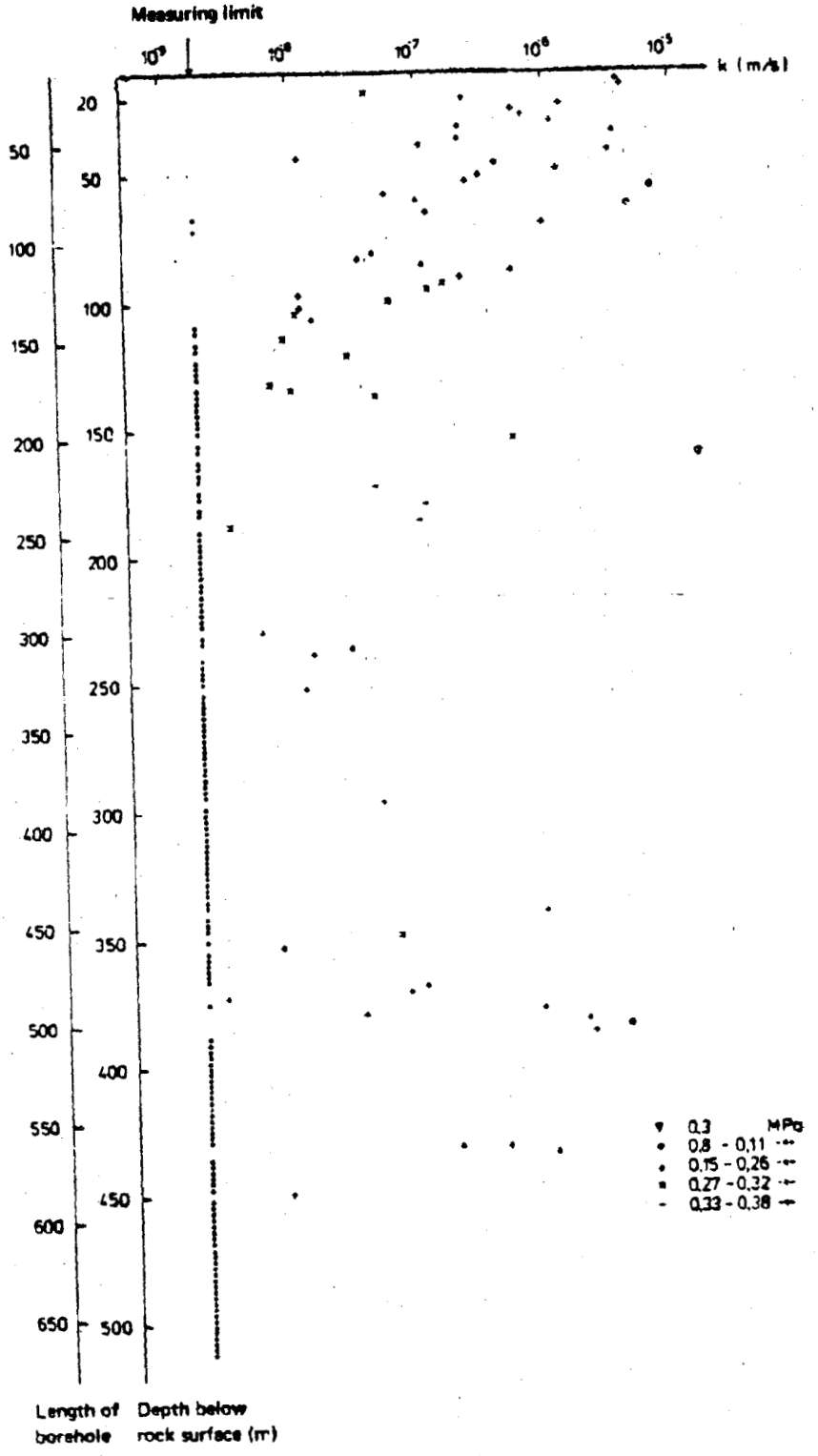


Figure 4-7. Diagram of rock permeability in core borehole 2 at Finnsjö Lake. The borehole was sunk at an angle of 50° to the horizontal. (Bedrock Bureau, Geological Survey of Sweden).

fissuring which is accompanied by reddish colouration and the formation of chlorite, calcite and quartz - both in the rock and as crack filler. Between 102.6 and 103.6 m, a lamellar chlorite dyke containing some smectite was found, which caused both core and water losses. The dyke was followed by a zone of intensive crushing. This location coincides well with a vertical projection of the extension of the electrical disturbance zone and the corresponding photogeologically indicated fracture zone.

At 688 m, a crush zone which was filled with a finegrained crush product between 688.4 and 688.9 m was drilled into at an acute angle (approx. 20°). The material from this 50 cm-wide zone filled 4.5 m of the core tube. This remarkable increase in volume indicates the presence of swelling clay minerals which have a low water content to start with and begin to swell when they come into contact with the flush water and the groundwater which collects in the borehole. A corresponding "dry" filler, also in the north-easterly main zone, would explain why no electrical indication of this zone was obtained. Alternatively, this zone may be composed of mylonite.

The borehole was stopped at 698 m (approx. 525 m vertical depth), still in broken rock. Between 110 and 680 m the bore core consists of predominantly good rock. Those cracks which do occur are largely filled with chlorite, calcite, quartz and prehnite, which, like the conditions in the Finnsjö 1 borehole, provide good imperviousness (see fig. 4-7).

Samples of granodiorite and the chlorite dyke at a depth of 103 m and the crushed material from 688.4 m have been examined by means of X-ray diffraction in order to determine what minerals the groundwater is in contact with. The results show that the rock has some illite, that the chlorite dyke contains smectite and perhaps some mixed strata mineral and that the crushed material contains swelling minerals of the mixed strata type. This means that both the rock and the disturbance zones contain ion-exchanging minerals and that the disturbance zones contain swelling minerals which can have some self-sealing effect.

Finnsjö 3 Core Borehole - diameter 56 mm, length 700 m, depth 550 m - sunk at a 50° angle towards the south at the eastern edge of the survey area in order to obtain information on its relatively highly fissured border zone towards the fault valley in the east. The rock here is also uniformly granodioritic, but is more intensively fractured and therefore exhibits reddish coloration and an elevated quartz content. Here as well, the fissures are filled to a large extent with minerals. Despite many small fracture and crush zones, no large dyke zones have been encountered here.

Due to the angle of the borehole towards the fault valley, its distance to the valley increases with increasing depth. The results obtained so far indicate that the zone of impaired rock quality extends some 300 m in from the valley side. It is worth nothing that all boreholes in the Finnsjö area have shown that the zones of disturbance are surrounded by reddish-colour rock with mineral-filled cracks. This means that when a rock repository is being planned and built, warning will be obtained in plenty of time that one is approaching such zones. Virtually no core losses were recorded in connection with drilling through

these zones. Nor are they expected to give rise to any special petrological problems.

4.4 KRAKEMÅLA AREA

4.4.1 Location and topography

The study area in Kråkemåla is located approximately 7.5 km NNW of the Oskarshamn nuclear power plant at Simpevarp and approx. 1.5 km NW of the village of Kråkemåla between the Baltic Sea and Lake Götemaren (see fig. 4-8).

The Kråkemåla area is located near the transition between the subcambrian peneplain along the Baltic coast in the south and the fractured countryside which characterizes the coastal regions of northern Småland and Östergötland. The countryside is characterized by flat landscape broken by pronounced fracture valleys running primarily in north-south, east-west and northwest directions (see fig. 4-9).

The study area comprises the eastern portion of a local watershed between the Baltic Sea and Lake Götemaren, whose average water level is only 1 metre above sea level. The nearly horizontal surface of the area falls from 20 m above sea level in the west to 15 m in the east, where it is bounded by a north-west fracture valley with a pass point about 10 m above sea level between Bussviken Bay (in the Baltic) and the outlet of Lake Götemaren.

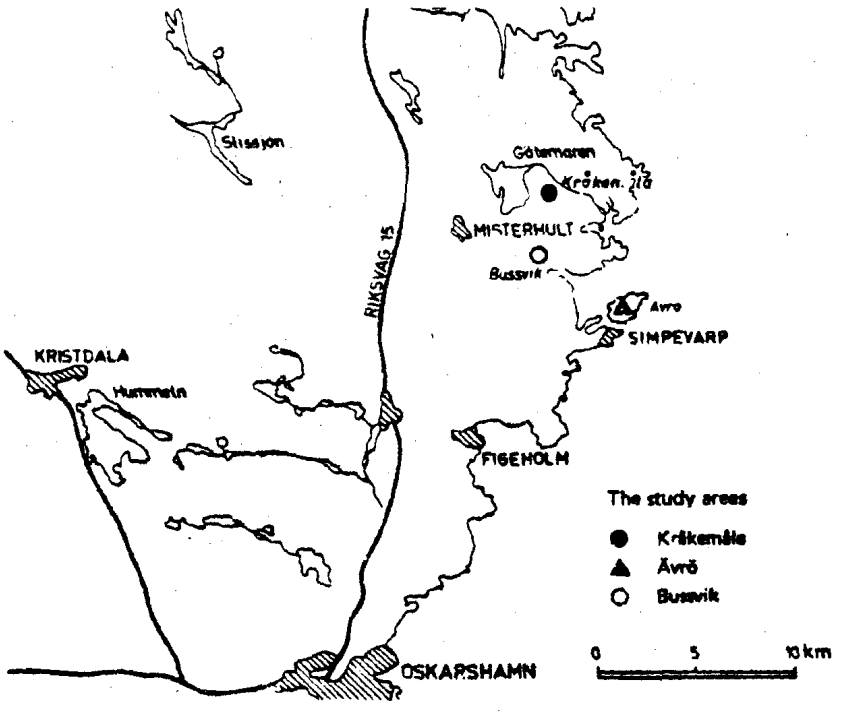


Figure 4-8. Map showing the study areas of Kråkemåla, Ävrö, and Bussvik

The fracture valley is partially filled with sand and gravel deposits. The bottoms of some very small gravel pits reach the local watertable. The soil cover is generally thin and large areas consist of exposed rock.

4.4.2 Bedrock conditions

For an overall picture of bedrock conditions in the area, see the geological map-sheet for Oskarshamn /4-10/. The region is composed of Småland granites while the actual study area is located within the Götemar massif - a body of young granite, the Götemar granite, with a circular outcropping about 9 km in diameter. Its age has been determined at about 1 380 million years /Åberg 4-11/. The massif was recently described by Kresten and Chyssler /4-12/.

The Götemar granite is composed of four subtypes which occur in different parts of the massif and which differ primarily with respect to grain size.

A coarse-grained red granite with a grain size of around 15 mm dominates in the study area. This granite consists primarily of microcline (60-75%), quartz (20-35%), biotite and accessory minerals. In the western part of the area, it borders on medium-grained granite with a grain size of less than 10 mm, often somewhat lighter and muscovitebearing but otherwise of nearly identical composition. Chemical analyses of the rocks in the Kråkemåla area are reported in table 4-3.

Flat, horizontal bodies of pegmatite in which the grain size can reach around 10 cm are encountered frequently in both types of granite. They are seldom more than 0.5 cm thick and a metre or so long.

The general pattern of fractures in the region has been analyzed by Asklund /413/ and Nordenskjöld /4-14/.

The Götemar massif is very uniform with respect to rock structure. It exhibits four pronounced fracture directions. Steep vertical cracks parallel to the circumference of the massif make up a concentric pattern which is complemented by a radial system which, together with the concentric set, produce a very regular, nearly perpendicular pattern of fractures in each local section (see fig. 4-10 and 4-11). In addition to these, there are diagonal fractures and widely dispersed, nearly horizontal fractures areas which are also responsible for the flat rock surface within the study area. The various types of granite in the massif are characterized by varying fissure density (see fig. 4-12). The coarse granite especially is of unusually low fissure content at the surface, which was one of the reasons for choosing this study area.

The same fracture pattern recurs in the surrounding Småland granite, where it is filled with granite and pegmatite veins which are associated with the Götemar granite. This, along with the deposits of primary minerals such as muscovite on the fracture surfaces, show that these cracks belong to a late phase in the formation of the Götemar granite and are therefore just as old, while more recent deformation is of much less importance.

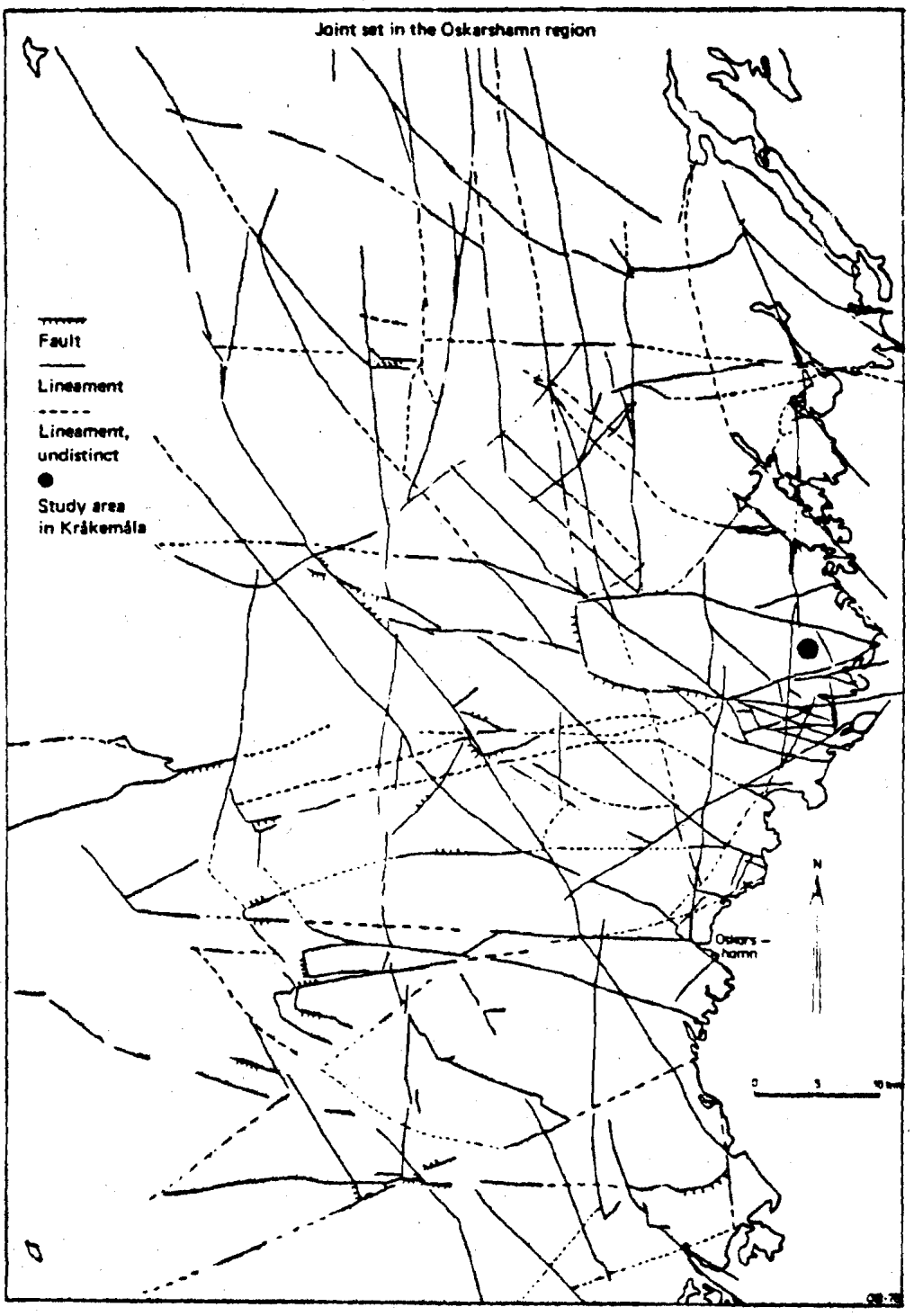


Figure 4-9. Map of the joint set in the Oskarshamn region. (Bedrock Bureau, Geological Survey of Sweden).

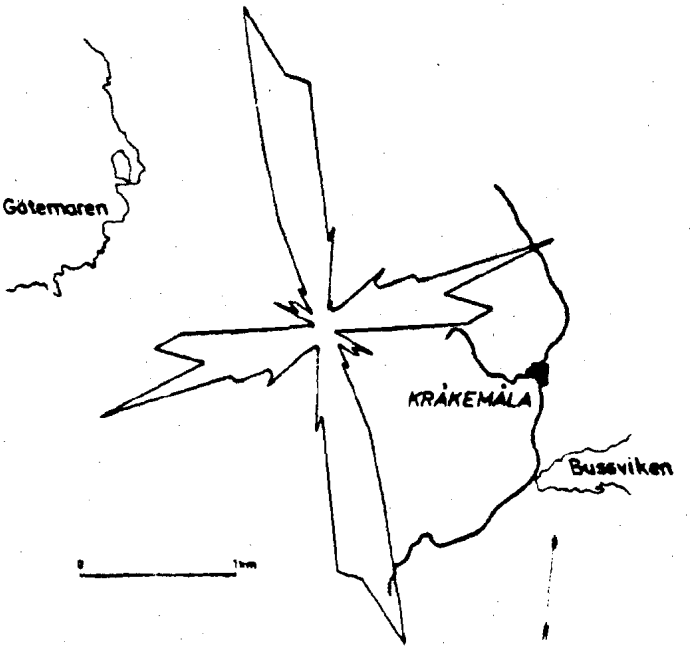


Figure 4-10. Diagram showing the direction of 327 steep fractures within the study area at Kråkemåla. The cracks are regularly oriented and deviate little in terms direction. (Bedrock Bureau, Geological Survey of Sweden).

An early tensional event with a probable age of around 1 200 million years is represented by a diabase vein which cuts through the granite in a NNW direction. East-west fractures, which are undeformed and filled with sandstone of precambrian age, indicate the proximity of the subcambrian peneplain.

More recent post-Cambrian movements can be discerned locally. These include the north-south fault which runs through the massif approximately 5 km west of the study area. The east-west fault at Mönsterås, which constitutes the northern limit of the precambrian sandstone there, can be seen at a greater distance.

4.4.3 Geophysical ground measurements

Two square kilometers within the Kråkemåla area were previously surveyed magnetically, electrically and seismically by Eriksson /4-15/ on commission from PRAV. The measurements showed, in agreement with the fracture pattern, that the coarse Götemar granite possesses very low electrical conductivity, but is subdivided into blocks bounded by zones of higher conductivity. The elastic properties of the rock were determined seismically. The following values were obtained: modules of elasticity 45-55 GPa, modules of shear 22.18 GPa, Poisson's ratio 0.25.

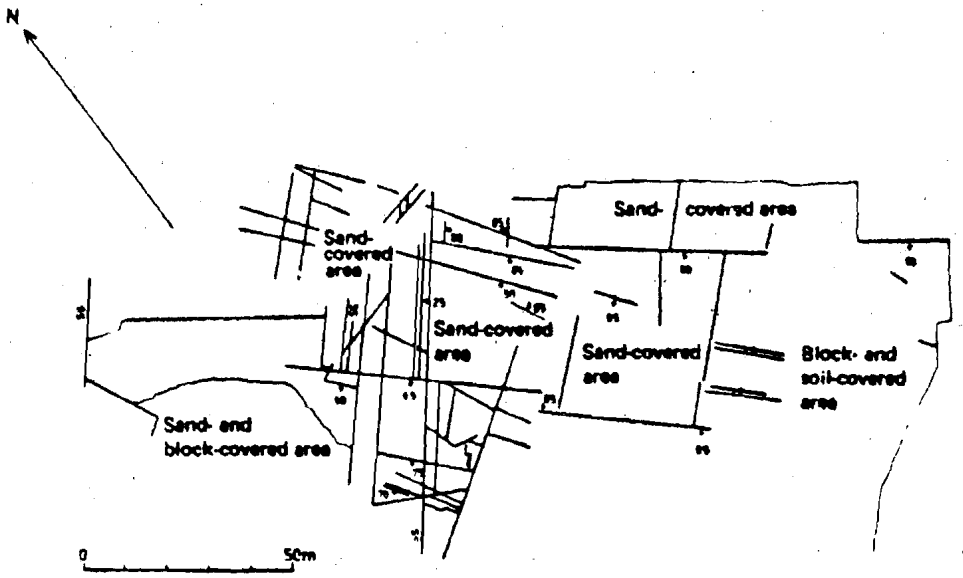


Figure 4-11. Joint map of stone quarry north of Kråkemåla (location, see figure 4-12). It illustrates the regular pattern of cracks. SS marks cracks filled with 550 million-year-old Cambrian sandstone. (Bedrock Bureau, Geological Survey of Sweden).

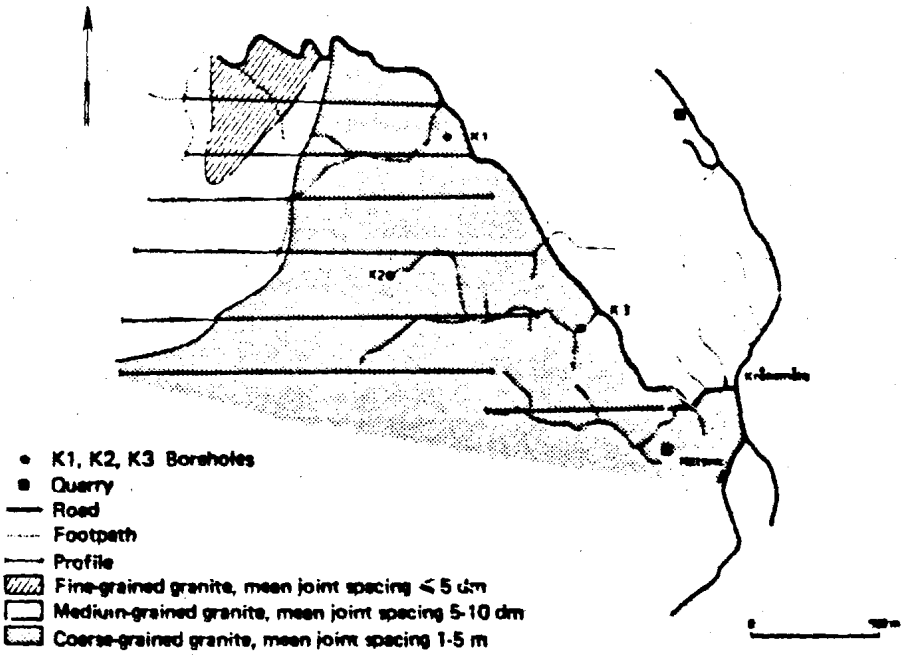


Figure 4-12. Map of study area at Kråkemåla showing the variations in the grain size of the granite. The 3 core boreholes are marked. The mean joint spacing in the different types of granite varies between 0.5 and 5 m. (Bedrock Bureau, Geological Survey of Sweden).

Table 4-3. Chemical composition of the rock in the Kråkemåla area, after Kresten and Chyssler, 1976.

	1	2	3	4	5	6	7	8	9	10	11	12	13
SiO_2	73.2	71.9	75.9	72.2	72.4	76.1	74.4	71.7	74.9	73.8	73.3	76.1	67.1
TiO_2	0.41	0.42	0.15	0.36	0.36	0.08	0.13	0.05	0.13	0.07	0.26	0.14	0.80
Al_2O_3	13.7	13.8	12.6	13.2	13.2	12.6	14.8	15.9	12.7	14.3	13.0	12.5	14.5
Fe_2O_3	0.8	0.9	0.1	0.8	0.7	<0.1	0.1	<0.1	0.2	0.1	0.5	0.1	1.5
FeO	0.6	0.9	0.5	0.8	1.1	0.3	0.1	0.3	0.6	0.3	0.8	0.4	2.3
MnO	0.04	0.06	0.03	0.03	0.05	0.02	0.01	0.18	0.07	0.06	0.04	0.04	0.07
MgO	1.2	1.2	0.8	1.3	1.2	0.5	0.3	0.10	0.6	0.10	1.1	0.5	1.1
CaO	0.6	0.6	0.08	0.33	0.36	0.02	0.2	0.02	0.08	0.4	0.25	0.06	1.9
Na_2O	3.1	3.7	3.6	3.7	3.6	4.2	2.9	6.1	3.9	4.0	3.5	4.0	3.1
K_2O	4.3	5.3	5.0	5.1	5.1	4.4	4.1	4.0	4.6	5.2	5.1	4.5	5.1
H_2O+	-	-	0.4	0.4	0.3	0.3	-	0.3	0.5	0.5	0.3	0.2	0.6
H_2O-	-	-	<0.1	0.1	0.1	<0.1	-	<0.1	0.1	0.1	0.1	<0.1	0.1
P_2O_5	-	-	0.01	0.08	0.07	<0.01	-	<0.01	0.02	<0.01	0.05	0.01	0.22
CO_2	-	-	0.03	0.01	0.11	0.04	-	0.03	0.03	0.02	0.19	0.01	0.12
F	0.49	0.56	0.56	0.59	0.51	0.31	0.06	0.41	0.55	0.24	0.51	0.32	0.18
S	-	-	<0.02	<0.02	<0.02	<0.02	-	<0.02	<0.02	<0.02	<0.02	<0.02	0.04
BaO	0.08	0.08	0.02	0.08	0.08	0.01	0.02	<0.01	0.01	0.01	0.05	0.01	0.13
Sum	98.32	99.32	99.78	99.11	99.26	98.88	97.02	99.09	98.99	99.20	99.05	98.99	98.86
-O for F, S	0.21	0.24	0.24	0.25	0.21	0.13	0.03	0.17	0.23	0.10	0.21	0.13	0.09
Total	98.11	99.08	99.54	98.86	99.05	98.75	96.99	98.92	98.76	99.10	98.84	98.86	98.77
Norm:													
Q	35.7	27.2	33.7	29.1	29.8	32.9	40.0	19.4	32.9	28.1	31.7	33.7	26.9
Ab	28.7	33.7	32.9	34.0	33.0	38.4	27.1	54.8	35.9	36.5	32.1	36.6	26.6
Or	23.3	29.0	27.3	27.6	27.3	25.0	24.5	22.8	25.9	30.7	27.7	25.6	26.8
An	3.0	3.0	0.2	1.2	1.3	0.1	1.0	0.1	0.2	2.0	0.9	0.2	6.8
Bi	4.6	4.5	3.7	5.2	5.6	2.4	1.1	1.3	3.2	0.9	4.9	2.4	6.0

1. GÖT 11 Coarse-grained granite. 2. GÖT 24 Coarse-grained granite. 3. SL31A Coarse-grained granite. 4. GG 14 Coarse-grained granite. 5. GG 3 Coarse-grained granite. 6. GG 8 Medium-grained granite. 7. GÖT 26 Medium-grained granite, pale pinkish. 8. SL36B Fine- to medium-grained granite, white, with garnet and topaz. 9. SL1A Fine-grained granite, porphyritic. 10. G4a Medium-grained granite. 11. GG 2 Granite porphyry, dyke north of the massif. 12. GG 7 Porphyritic granite, eastern margin of the massif. 13. GG 1 Småland granite, reddish grey, porphyritic variety. Wall-rock to the north.

4.4.4 Drilling

The first hole was drilled in order to determine whether the fundamental characteristics of the rock as regards vertical extent and uniformity were suitable. When this was found to be the case, further holes were drilled. Borehole 2 illuminates the western boundary of the low-fissued, coarse-grained granite, where the medium-grained granite begins. Borehole 3 runs through the border zone of the study area towards the fracture valley which constitutes its boundary towards the east.

Kråkemåla 1 Core Borehole - diameter 56 mm, length 504.65 m - was sunk vertically in good rock and runs throughout nearly its entire length through uniform, coarse-grained, red massive granite. There are, however, five bands of fine-grained aplitic granite with a combined thickness of 12.2 m. Coarse-grained granite thus comprises 97.6% of the core. At a depth of between 60 and 76 m, the granite exhibits scattered grains of pyrite and molybdenite.

The distribution of fractures is depicted by the RQD diagrams - see fig. 4-13. The majority consist of fresh fractures straight through the bore core, and many were created during drilling.

The water injection tests show that the permeability of the rock down to 50 m is distributed around 10^{-7} m/s (see fig. 4-13), while a clear division into high and low values is found at the deeper levels. The high values here are also around 10^{-7} m/s, while the low values are at or below the measuring limit, i.e. no water loss could be measured by means of the equipment which was used. This means that the permeability of the rock is less than 1.9×10^{-9} m/s at a pressure of 0.2 MPa and less than 8×10^{-10} m/s at a pressure of 0.6 MPa. Only four 2-metre sections along the entire section between 320 and 496 m exhibited measurable water loss. These sections contain smooth, deposit-filled fractures while the sections containing the more common fresh cracks do not give rise to any measurable water loss.

The fractures in the Götemar granite exhibit varying mineral content. Sample scrapings from the walls of the cracks in the bore core from Kråkemåla 1 were therefore subjected to closer study. Aside from the usual minerals in the granite and general crack minerals such as chlorite and calcite, the cracks also contained sulphur pyrite and lead glance as well as fluorspar, kaolinite and smectite. Kaolinite was found with certainty in only a single sample near the surface, while smectite was found in four samples down to 326 m, although in little quantity.

Kråkemåla 2 Core Borehole - diameter 56 mm, length 604.8 m - was sunk vertically in good rock near the western boundary of the coarse-grained red granite. This is reflected in the core by many (25) bands of fine-grained granite, which together constitute 14% of the entire length of the core, and a generally higher fissure content - see fig. 4-14. This borehole also exhibits good imperviousness between 330 and 495 m, with values for water loss which lie below the measuring limit. At greater depths, fissure content increases again. It is therefore realistic to assume that this borehole marks the western boundary of the volume of rock which is suitable for housing a rock repository.

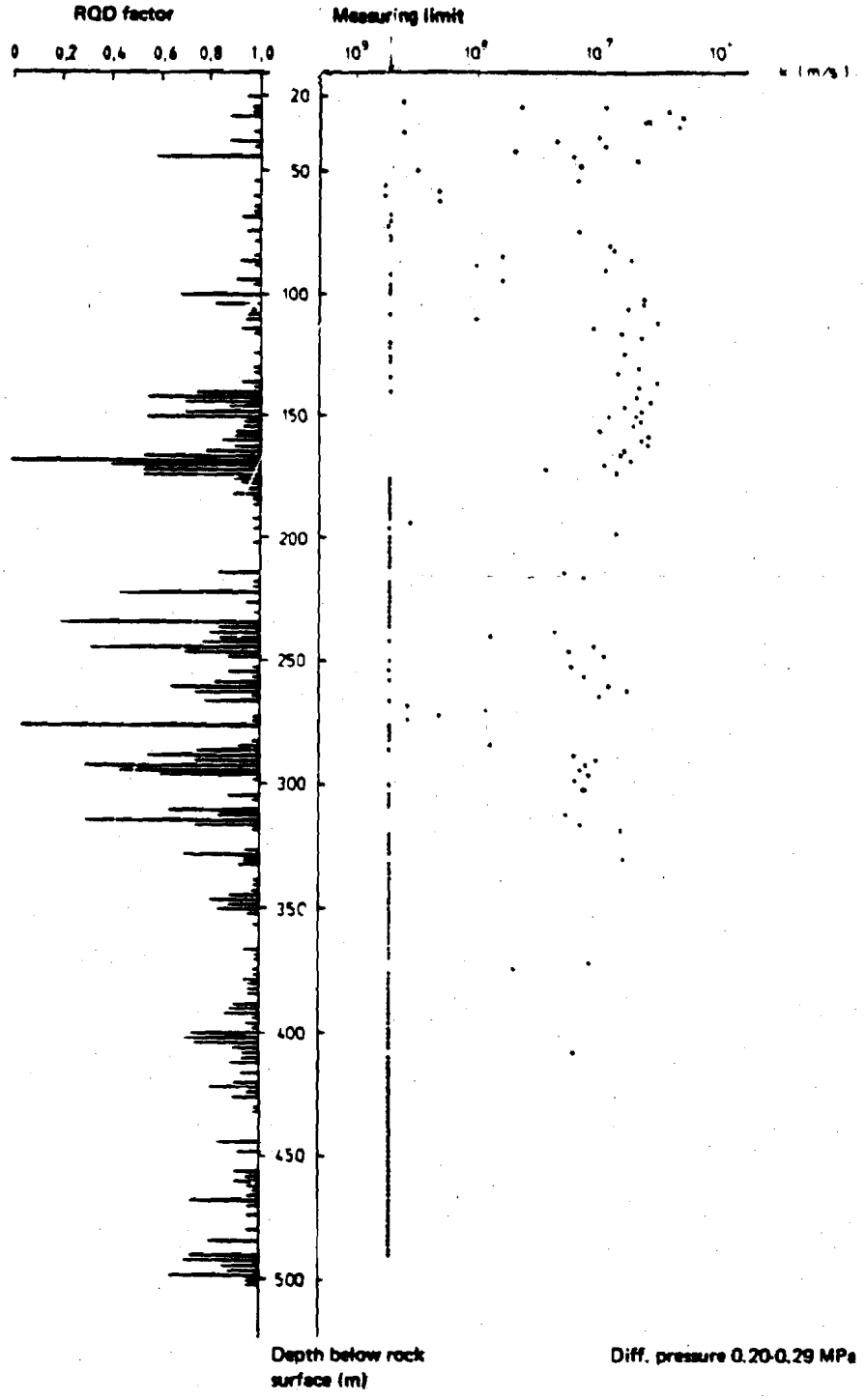


Figure 4-13. Diagram of rock permeability (right) and fissure content expressed as RQD factor (left) in core borehole 1 at Kråkemåla. (Bedrock Bureau, Geological Survey of Sweden).

Kråkemåla 3 Core Borehole - diameter 56 mm, length 760 m, depth approx. 560 m - was sunk with a 50° angle towards the WNW in good rock on Solberget, which is a low-fissured rib between two northwesterly fracture valleys east of the geophysically studied area. Bedrock conditions in this borehole are very similar to those in Kråkemåla 1. Despite the frequency of fissures and narrow crush zones in the upper parts, the fissure content of the rock decreases with depth. This indicates that fissuring of the rock around the topographically prominent valleys in the Kråkemåla area is limited.

4.5 OTHER AREAS

4.5.1 Åvrö

Åvrö is located 1.5 km north of Simpevarp, and most of the land belongs to Oskarshamns Kraftgrupps AB (see fig. 4-8). Topographically, it is an island with many small fracture valleys. The bedrock consists of red to grey, medium-grained and unstratified to weakly gneissic Småland granite. An eastwest steep diabase was observed in one exposed rock slab. Geophysical measurements indicate that the entire area is divided into blocks with intervening, slightly electrically conductive zones. A seismic study revealed the following data: modulus of elasticity 25-43 GPa, modulus of shear 10-17 GPa, Poisson's ratio 0.25.

Åvrö 1 Core Borehole - diameter 56 mm, length 502.2 m - was sunk vertically in good rock in an area of high resistivity. The core shows red granite, which, despite considerable fissure content, has permeabilities below 10^{-7} m/s. Diabase was encountered in four sections, but the lengths of the sections are probably much greater than the thicknesses of the diabases, due to their steep angles. Below 400 m, the granite in this borehole is heavily crushed and highly permeable. The studies were therefore not carried to completion.

4.5.2 Bussvik

Bussvik Bay is located 4.5 km northwest of Simpevarp (see fig. 4-8) and was only studied geologically and seismically from the surface. The area is characterized by large, relatively low-fissured surface slabs of Småland granite. Good seismic values were measured: modulus of elasticity 50-60 GPa, modulus of shear 20-24 GPa, Poisson's ratio 0.25. No drillings have yet been carried out in Bussvik.

4.5.3 Forsmark

Forsmark is located about 3.5 km west of the Forsmark nuclear power plant (see fig. 4-3) and within an area around the power plant which has been surveyed by means of geophysical aerial measurements. Surface geology and geophysical measurements indicate that the area comprises a single coherent bedrock block with high resistivity and a low fracture frequency. The bedrock is composed of medium-grained, weakly gneissic grey quartz diorite, which borders on leptitic gneiss on the south with a northwesterly

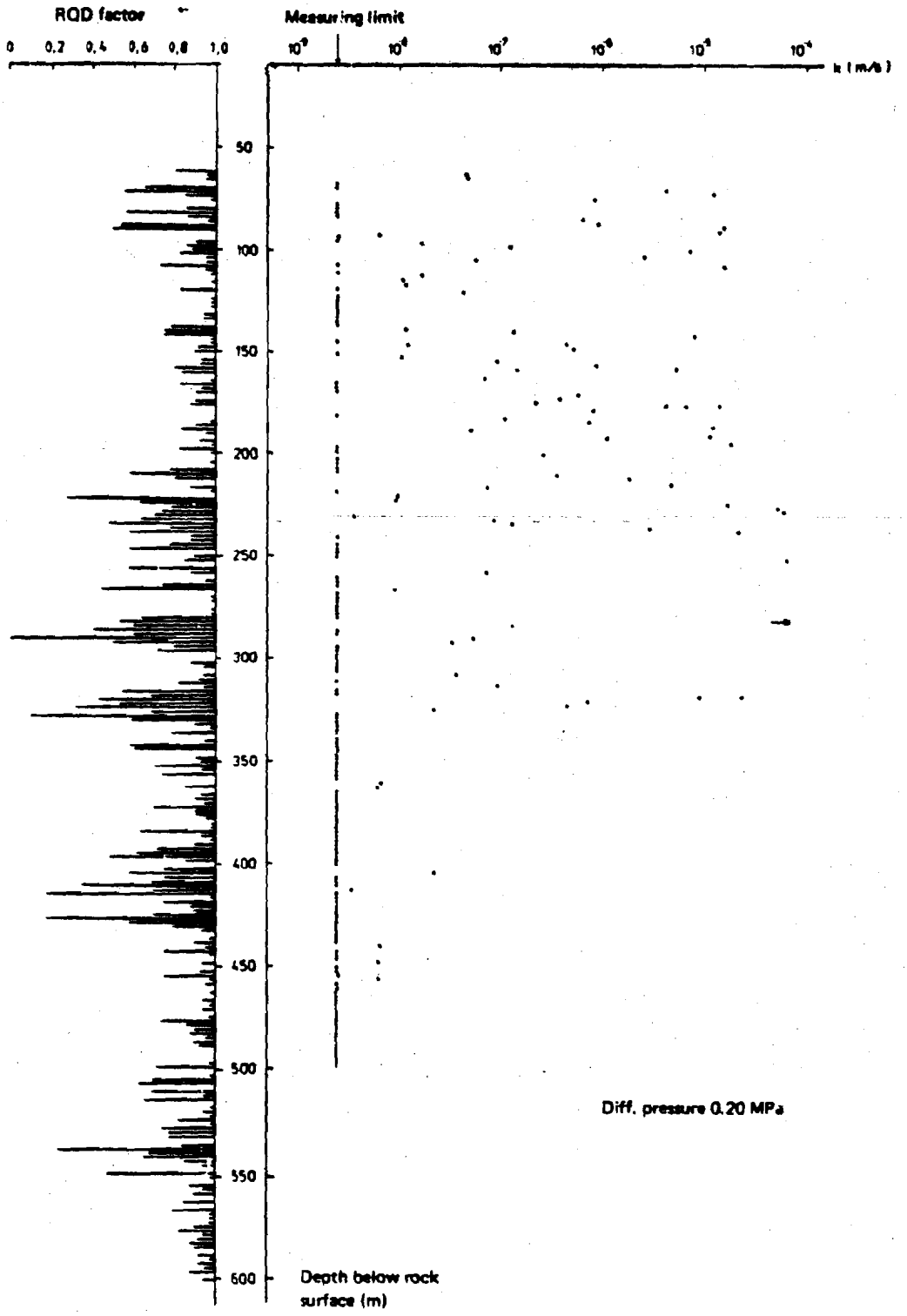
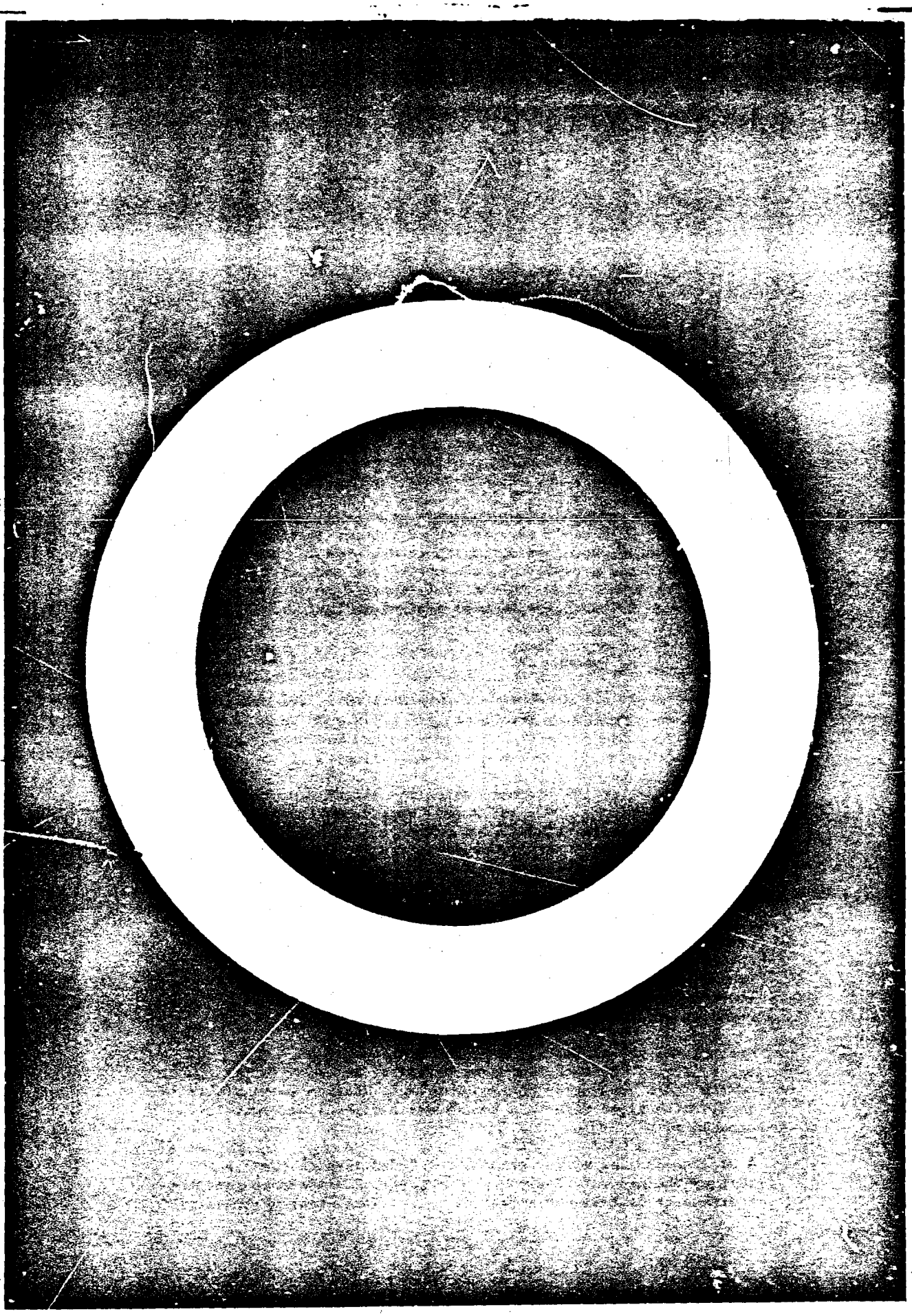


Figure 4:14 Diagram of rock permeability (right) and fissure content expressed as RQD factor (left) in core borehole 2 at Kråkemåla. (Bedrock Bureau, Geological Survey of Sweden).

strike. The leptite is partially folded and has a predominantly northeasterly dip.

Forsmark 1 Core Borehole - diameter 56 mm, length 478.3 m - was sunk vertically in good rock. Down to 219 m, the hole runs through a rather low-fissured diorite with a varying content of hornblende and biotite, and occasional layers (2 m thick) of pegmatite and aplite. At increasing depth, a nearly horizontal banding becomes increasingly pronounced, and below 375 m, banded, light, partially leptitic gneisses predominate. At the same time, the quality of the rock deteriorates considerably. Drilling was therefore terminated at 478.3 m. The drilling results indicate that the study area describes a U-shaped convolution and that the diorite is not sufficiently deep. The area must therefore be regarded as less suitable for a deep rock repository.



5 GROUNDWATER CONDITIONS

5.1 GROUNDWATER HYDROLOGY

Radioactive waste which is stored deep down in the bedrock can be dispersed only via the groundwater. The magnitude of the groundwater flow in the areas under consideration, as well as its velocity, retention time and pattern of movement, are therefore of great interest.

The metallic or ceramic material which is used to encapsulate the waste can be subjected to corrosion attack when it comes into contact with the groundwater. The nature and rate of the corrosion attack depend upon the chemical composition of the groundwater, which is therefore another crucial factor.

5.1.1 Permeability of the bedrock

As was related in chapter 4, the permeability of the bedrock was measured in a number of boreholes in 2 m (in some cases 3 m) long sections from the surface of the rock to the bottom of the borehole. The results can be summarized as follows.

The upper part of the bedrock, which may extend down to a depth of anywhere between 20–30 and a few hundred metres, is often characterized by relatively high permeability, owing to an extensive and coherent network of fissures. The upper sections of the bedrock correspond most closely to the model for fissured rock developed by Snow /5-1/ on the basis of a large number of drillings and permeability determinations down to a depth of 100 m. With increasing depth, the abundance of sections of very low permeability increases, and there is a transition to conditions characterized by large formations of predominantly impervious rock, interrupted by narrower water-bearing fracture zones. The lower sections therefore exhibit the conditions for crystalline rocks at great depth described by Webster et al. /5-2/.

Most of the groundwater flow in the bedrock takes place in the upper part of the rock, where permeability is often between 10^{-5} and 10^{-7} m/s. Hydraulic coherence in this section is generally good, which gives rise to a continuous and level water table /see Larsson et al. 5-3/.

A smaller portion of the groundwater flows through the deeper part of the bedrock, where its movement is for the most part restricted to certain water-bearing zones. Water-bearing zones of

high saturation have been found in Swedish mines down to a depth of 900 m. Intervening sections of rock have a permeability of less than 10^{-9} m/s. 5×10^{-11} m/s has been measured in the granite at Stripa /5-4/. Hydraulic coherence between the individual fissures at great depth appears to be severely restricted, as is evidenced by the fact that no measurable water flow was found in sections where both the drill core and TV examination indicate the existence of fissures. Considerable differences in the chemical composition and age of the water also indicate that hydraulic coherence between the waterbearing zones in the same boreholes can be limited at these depths. But there is always some hydraulic coherence via the more permeable upper part of the bedrock.

5.1.2 Groundwater flow

The rate of groundwater flow is determined by the profile of the water table, the permeability of the bedrock and depth below the water table. The water table follows the contours of the landscape, with some smoothing-out.

In order to calculate the groundwater flow, a large number of two-dimensional models with different water table profiles and permeability conditions have been simulated by means of special computer programs /5-4, 5-5/. Figure 5-1 shows the groundwater flow underneath an island, from its centre outwards. The calculation is based on the fact that permeability is known near the surface and decreases with depth in a regular manner. This diagram has been used as a model for the Karlshamn area, which is situated on a peninsula. At a depth of 500 m, a subsurface permeability of 10^{-9} m/s and a water table slope of 0.05, a flow of about 0.2 litres per m^2 and year is obtained.

Figure 5-2 shows the groundwater flow underneath a kilometre-long slope. In the calculation, it is assumed that permeability does not vary with depth, which leads to groundwater flows down to great depths. This model has been applied to the areas at Finnsjö Lake and Kråkemåla. For the Finnsjö area, a flow of 0.1 litres per m^2 and year is obtained at a depth of 500 m, with a permeability of 10^{-9} m/s and a water table slope of 0.008. For the Kråkemåla area, a flow of 0.15 litres per m^2 and year is obtained at the same depth and permeability with a water table slope of 0.012.

The flow values calculated above are probably much higher than the actual values, since the average permeability of the rock is lower than the value of 10^{-9}, which was the measurability limit.

5.1.3 The pattern of groundwater flow

The path of the groundwater through the bedrock can be illustrated by means of the same type of diagram as is shown in figures 5-1 and 5-2, provided the calculations are adjusted to the actual elevations within a given area. For two-dimensional calculations, it is assumed that slopes and other landscape contours are oriented perpendicular to the plane of the figure and extend far in this direction. In order to be realistic, the calculations must be made with regard to a plane which is perpendicular to the direction of the dominant valley. Furthermore, the permeability of

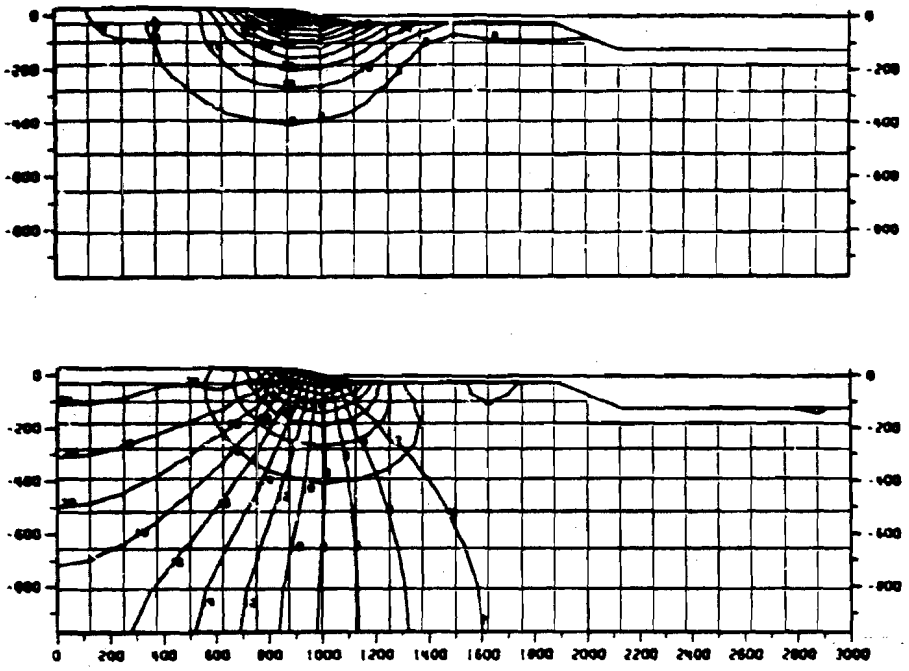


Figure 5-1. Diagram of the calculated groundwater flow under an island from the centre outwards. The height and length scales are in metres. The top chart shows lines for equal flow expressed in 10^{-9} m^3/s and a cross-sectional area of 1 m^2. The lower chart which shows flow lines and equipotential lines, assumes an impenetrable bottom surface at a depth of 1000 metres and a superficial permeability of 10^{-6} m/s. Permeability then decreases exponentially to $5 \cdot 10^{-8}$ m/s at a depth of 1000 metres /5-5/.

the rock within the area must be assumed to be constant or change with depth in a regular manner. The influence of individual zones of higher permeability, which are responsible for much of the flow in the deeper parts of the bedrock, can therefore not be simulated by the model. By varying the assumptions for the calculations and thereby distributing the effects of the individual zones over greater volumes, it is nevertheless possible to shed light upon the general flow conditions. This has been done for the Finnsjö area (see fig. 5-3 and 5-4).

The diagrams show, as was already known, that the groundwater flows downward into the bedrock in elevated areas, after which it turns and flows upward again towards large adjoining valley floors, where it can reach the surface at points of groundwater inflow into waterways. The influence of terrain features often extends down to depths of several thousand metres. The longer the slopes are, the deeper their influence reaches. The surface areas

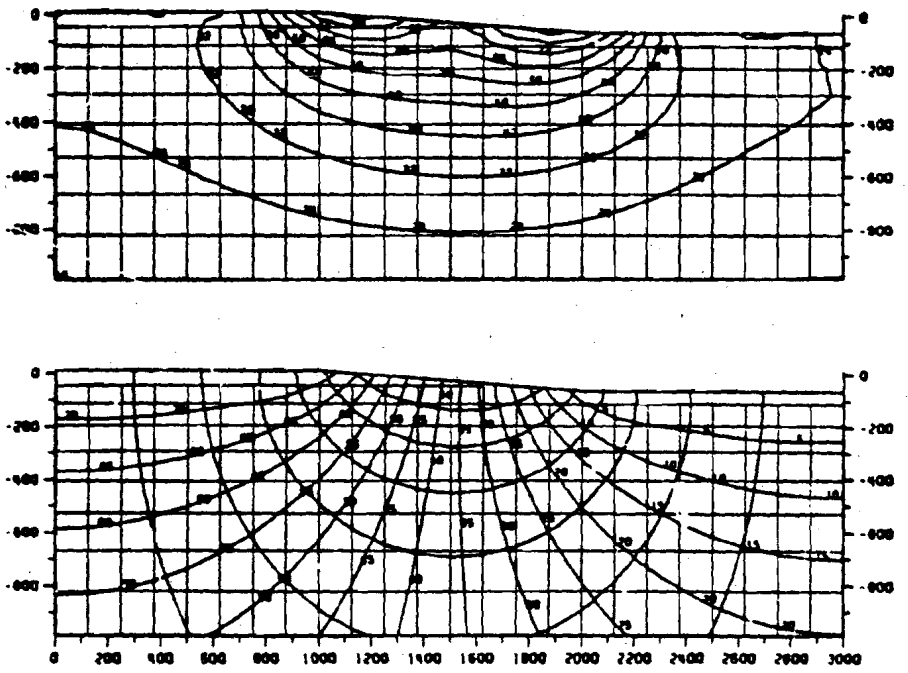


Figure 5-2. Diagram of calculated groundwater flow under a slope. The height and length scales are in metres. The top chart shows lines for equal flow expressed in 10^{-9} m^3/s and a cross-sectional area of 1 m^2. The lower chart shows flow lines and equipotential lines. No bottom surface at a finite depth is assumed in this case. Permeability is constant at 10^{-6} m/s. /5-5/.

where groundwater from great depth issues are small, and the upflow is accompanied by a very heavy dilution of the groundwater.

One consequence of these general conditions is that the groundwater movements in an area lacking extensive, flat aquifers are divided into smaller flow cells and that groundwater transport is predominantly of a local character. This pattern becomes more pronounced if the valleys follow fracture zones in the bedrock where permeability is high.

The diagrams for the Finnsjö area show that the flow there is directed towards Finnsjö Lake and towards a valley approximately 2.5 km northeast of the lake. There is probably also some upward flow in the fault valley which borders on the study area towards the northeast. No cold springs have been found here or in other parts of the area, which indicates that none of the deep groundwater reaches up to the surface. The groundwater should therefore follow roughly the pattern illustrated by the diagram.

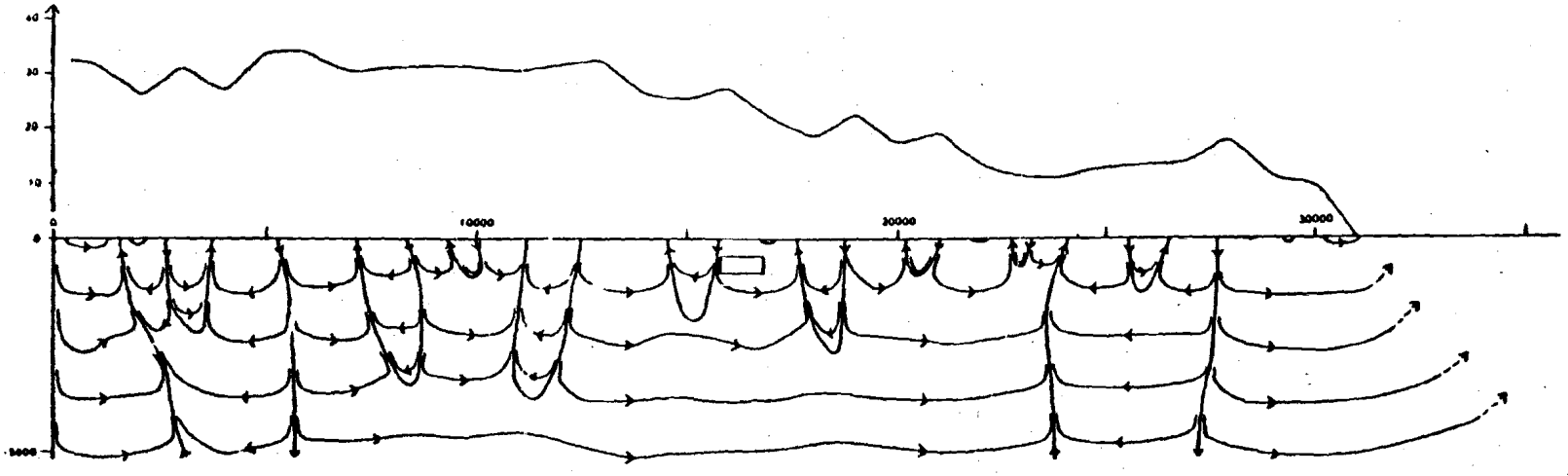


Figure 5-3. Diagram of groundwater flow pattern in a northeasterly section through the area at Finnsjö. An impenetrable bottom surface at finite depth is not assumed in this case. Permeability is assumed to decline by 50% for every 100 metres. The flow lines are chosen so that the flow decreases with depth by a factor of 10 000 between two flow lines. /5-5/.

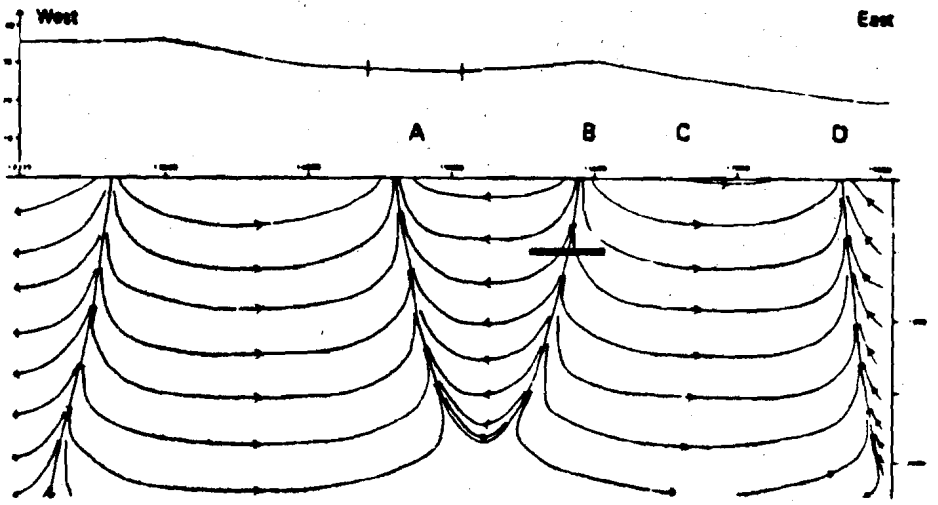


Figure 5-4. Detail from figure 5-3. The flow lines are chosen so that the groundwater flow decreases with depth by a factor of 10 between two flow lines. A marks Finnsjö Lake, B marks the groundwater divide, C marks a fault valley, and D marks the outflow area. The repository is marked with a heavy line under B. Note that the groundwater flows through the repository in a downward direction and does not reach above a depth of 500 metres until it reaches the outflow areas. (The figure was taken from /5-5/, but the repository is shown in another position).

5.1.4 Dilution effects

The diagram in figure 5-4 can serve as a basis for an assessment of the dilution of the groundwater which could come into contact with the waste canisters in a rock repository in the Finnsjö area. Assume that the repository is located at a depth of 500 metres in the middle of the downward groundwater flow at point B in the figure. The repository contains 9 000 canisters with a length of 1.8 m and a diameter of 0.4 m. The cross-sectional area of a canister is thus 0.7 m^2. The flows to Finnsjö Lake and towards the northeast each come into contact with half of the number of canisters, with an area of about 3 000 m^2, and amount to 3 000 x 0.1 = 300 litres per year.

As is evident from the diagram, the flow at the repository is in a downward direction and then remains for the most part at a depth greater than 500 m, until it swings upward underneath Finnsjö Lake (A in figure 5-4) and under the valley in the

northeast (D in figure 5-4). It is thus out of reach of normal rock wells along its path.

In Finnsjö Lake, the groundwater is diluted by surface runoff and groundwater from the Finnsjö catchment area, which is about 100 km^2. With an annual precipitation of 550 mm and an evaporation of 300 mm, a total water volume of 2.5×10^7 m^3 is obtained. As was mentioned above, a groundwater flow of max. 300 litres per year is expected to come into contact with the waste canisters and then flow into Finnsjö Lake. Its dilution there will then be approximately 1 part in 8×10^7.

A similar dilution of the water from a catchment area of 10 km^2 takes place under the valley at D. The dilution is then 1 part in 8×10^6.

As regards the fault which borders the Finnsjö area towards the northeast at C in figure 5-4, it is conservatively assumed that an upward flow takes place. It is even more conservatively assumed that this upward flow is so heavy that it leads to a complete and thorough mixing of the groundwater down to the level of the final repository. Dilution takes place by the groundwater from an infiltration area of 2 km^2, whose volume is calculated to be 0.5×10^6 m^3. Runoff within the area is regarded as negligible. The dilution ratio is then about 1 part in 1.7×10^6, which applies to the water which may be taken out from a rock well. If the upward flow does not reach the surface and mixing of the groundwater in the fault zone is incomplete, the well will tend to be filled by groundwater closer to the surface which has not been in contact with the waste. The dilution ratio will then be considerably greater.

5.1.5 Effects of waste heat generation

Calculations have also been carried out in order to elucidate the effect of the upward flow over a rock repository which may be caused by the heat generated by the waste during the early phase of the storage period. In agreement with previous American estimates by the National Academy of Science /5-6/, this factor has been shown to give rise to an insignificant perturbation of the prevailing pattern of flow in the vicinity of the rock repository.

5.1.6 Lowering and recharging of the aquifer

The effects of the drainage pumping of a rock repository during the construction and deposition phases have also been investigated /5-4, 5-7/. These calculations are of importance for future planning and engineering design work. The recharge time for the aquifer around the final repository, after it has been sealed, can be expected to be lengthy on the basis of experience from abandoned mines. During this period, there can be no outflow from the area.

5.1.7 Short-term variations in the groundwater level

A review of the literature and Swedish studies of short-term variations in the groundwater level has been carried out /5-8/. It shows that the groundwater level in the Swedish bedrock exhibits certain short-term variations which occur without the water content of the rock changing. The level is affected by such factors as tidal movements in the bedrock due to the influence of the sun and the moon and changes in air pressure. Severe earthquakes at distant points on the globe, for example off Portugal and in Japan, also affect the groundwater level. All of these variations are small in comparison with seasonal and precipitational variations and no effects with any appreciable bearing on the conditions in and around a rock repository have been found.

5.1.8 Groundwater age

The time during which the groundwater resides in the bedrock is of importance in view of the natural decay of the radioactive elements and their retardation and retention by the rock. In the same rock volume, the residence time for the water is least in the larger, water-bearing fracture zones. In the intervening bedrock blocks of low permeability, the residence time is many times greater. Residence time in the fracture zone was ascertained by means of age determinations carried out on water samples from the boreholes.

Age determinations on water samples were carried out using the carbon 14 method, which tells how much time has passed since the water became isolated from the atmosphere. Four water samples from the borehole in Kråkemåla have been studied thus far. 70 litres of water are required for each determination, as a result of which samples could only be obtained from zones with high water flow.

Two samples from depths of 291 and 510 m in borehole No. 2 gave ages of 4 400 and 4 275 years respectively. Two samples from 407 and 493 m in borehole No. 1 gave 11 055 and 8 205 years, respectively. The determinations were carried out by the laboratory for radioactive dating. Correction for exchange with carbonate mineral was made on the basis of carbon-13 content.

The analytic uncertainty inherent in these determinations is about ±100 years. However, larger errors in the age determinations may result from various measures undertaken in the boreholes prior to sampling. During drilling, large quantities of "younger" surface water were pumped down into the holes for flushing purposes, and heavy draiage pumping was carried out prior to sampling. These disturbing influences resulted in water of different ages being mixed, presumably resulting in age underestimations. Judging from the chemical composition of the water samples, the sample of the highest age is the one which is least disturbed by surface water.

The difference between the results 4 275 and 4 400 years is within the analytical error limits. The differences between the age results of 4 000 years on the one hand and 8 000 and 11 000 years on the other can probably be explanined by the different positions and permeability conditions of the boreholes, insofar

as they are not the result of the aforementioned disturbances. Borehole No. 1 is located on a nearly horizontal plateau-like part of the study area - not far from the fracture valley which comprises the boundary of the area on the northeast. The water table is therefore nearly level. And since the core from this borehole shows a large portion of impervious rock, the rate of groundwater flow is low, the residence time of the water long and its age high. The differences between the ages and compositions of the water samples from the fissures in this hole clearly show that the fissures have little hydraulic coherence.

Borehole No. 2 in Kråkemåla is located closer to the local water divide between Lake Götemaren and the Baltic Sea, and the water table here is located about 10 m below the surface of the ground. This would indicate that the groundwater here is flowing downward, which, together with the predominantly higher permeability of the rock, could explain the lower age of the water.

The Kråkemåla area is certainly not an exceptional case with respect to groundwater age. The general groundwater turnover rate here is, in fact, probably higher than in the other two study areas. Only a few age determinations on groundwater from great depths in the bedrock from other locations are known. One age determination of 4 010 years on mixed water from a depth of 136 m from a well in the bedrock in Finland /Donner and Jungner, 5-9/, and another determination of 9 785 years on a mixed water from a depth of about 300 m at Storjuktan carried out by the Geological Survey of Sweden on behalf of the National Council for Radioactive Waste Management, show that the age determinations from Kråkemåla are not exceptional. The sampling conditions in both of these cases were slightly different, but mixture with younger water should have led to underestimations of age.

The age determinations show that it takes at least 11 000 years for the water to travel from the inflow area to one of the sampling points in the Kråkemåla 1 borehole. Since the dated sample was taken from a water-bearing fissure at a depth of about 400 m, it may be concluded that groundwater which passes through a deep rock repository located in the inflow area in a rock formation of low permeability should require even more time to return to the surface.

The residence time of the groundwater in the rock has been calculated for a number of typical cases with the aid of the flow models mentioned previously. The results vary widely, however, depending on the choice of conditions. As a rule, the calculated residence times are lower than the ages determined by means of the carbon 14 method. This is due, among other things, to the fact that the measured permeability values which were used in the calculations are higher than the actual values, and that their directional dependence, as well as the hydraulic factors which prevail at great depth, are not yet sufficiently well understood.

5.2 GROUNDWATER COMPOSITION

The chemical composition of the groundwater plays a decisive role in determining the durability of the waste and encapsulation materials and in determining the ability of the bedrock to retard and retain the waste substances. The decisive factor which deter-

mines the composition of the groundwater is the fact that it is in long-term contact with the minerals in the bedrock, which are solid phases of relatively constant composition. The fissure-filling minerals are especially important in this connection. This contact results in chemical equilibriums which give rise to a given groundwater composition, regardless of local or random disturbances /Garrels 5-10, Eriksson and Khunakasem 5-11, Jacks 5-12, Eriksson and Holtan 5-13/. An example of such equilibrium reactions is that groundwater which can locally become acidic will rapidly return to a pH of around 8 as a result of reactions with the bedrock. If the water should locally become oxidizing due to oxygen infusion, this oxygen will quickly be consumed by reactions with the bivalent iron present in the rock mineral. This has been illustrated by several introductory experiments performed in the presence of atmospheric oxygen:

Samples from the Finnsjö area consisting of granodiorite and chlorite plus crushed material from a crush zone were finely ground in air and mixed with distilled water with a pH of 5.6 and a redox potential of Eh + 610 m V. Following centrifugation in sealed tubes, the Eh of the water dropped to +0.315, +0.267 and +0.084 m V, respectively, while the pH rose to 8.9, 9.2 and 9.4 respectively.

Groundwater analyses from rock wells in Sweden have not been systematically compared. But an extensive body of data is available from Finland, which has the same type of bedrock and groundwater as Sweden. Laakso /5-14/ reviews these data, which specify mean concentrations and ranges of variation for most of the principal substances in water from some 1 100 analyses of samples taken all over Finland, and Lahermo /5-15/ presents a selection from the coastal region of southeastern Finland. However, these data generally apply to shallow wells. The only analyses of water samples from depths of around 500 m which are available apply to mine water, which for many reasons cannot be regarded as being representative in this context.

Water samples from the boreholes in Kråkemåla and Forsmark were analyzed. In Forsmark, samples were taken from a depth of 450 m by means of an apparatus designed to prevent contact with atmospheric oxygen /5-16/. These analyses show that the oxygen content of the groundwater is less than 0.01 mg/l (which is the measurability limit) and that all dissolved iron is bivalent, plus that the water has a sulphide content (in the form of hydrogen sulphide) of 5 mg/m.

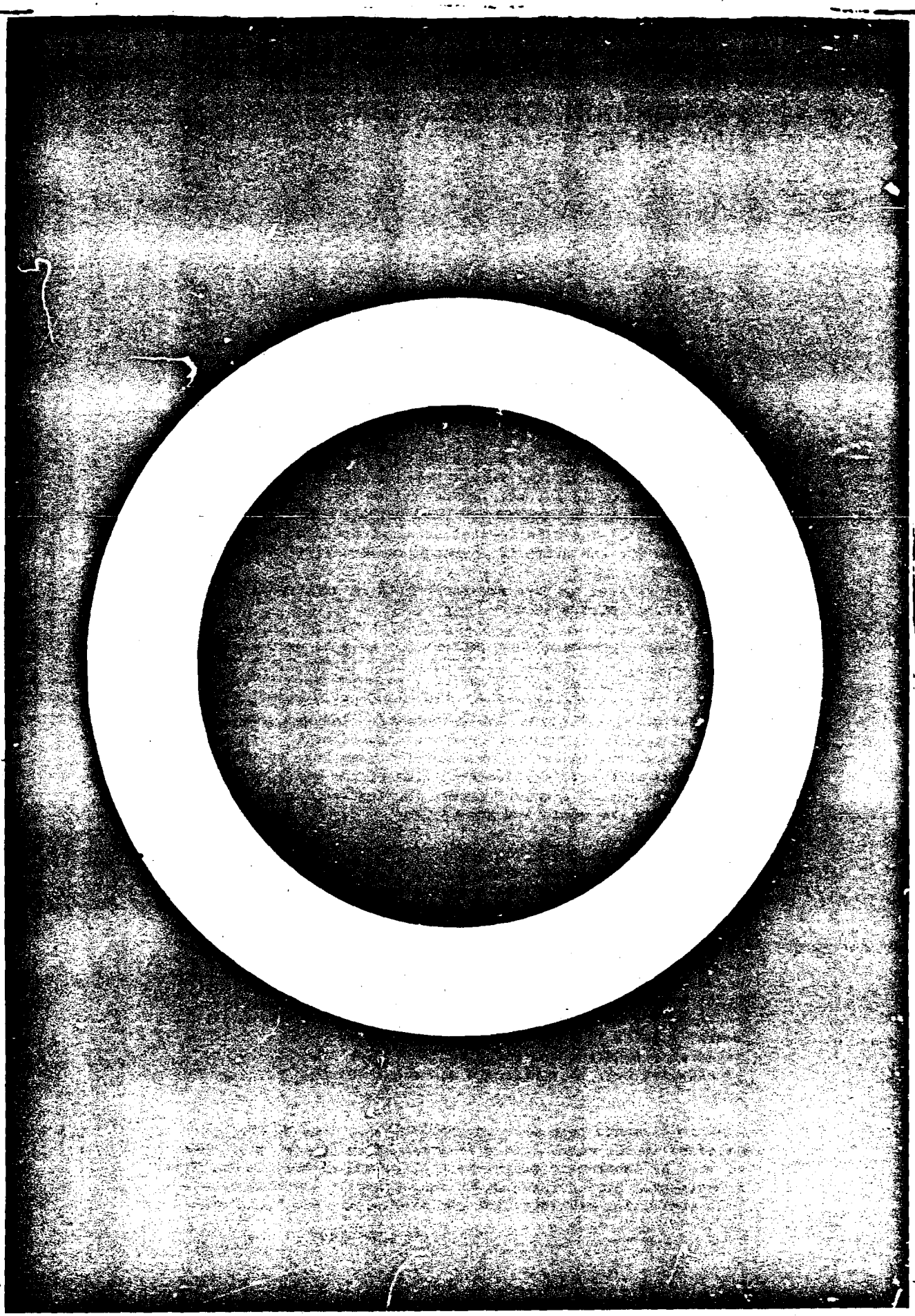
Table 5-1, which is taken from /5-16/, gives the results of analyses of water from a number of wells and boreholes in northern Uppland. The table also indicates the estimated range of variation of the analysis values.

Table 5-1. Analyses of natural water from northern Uppland and probable analysis values of groundwater.

Analysis	Unit	Forsmark area (KBS-TR 36 ref 5)	Borrstadina mine 78 m	Hallstavik 70 m 1963	Hallstavik 175 m 1977	Forsmark 1 450 m 77-09-26 x)	Forsmark 1 450 m 77-10-05	Surface water Forsmark		Groundwater, bedrock, probable analysis range	max xx)	Brackish well in Uppsala area, 100 m	Leach water bentonite sand 95°C 8 days
Conductivity	µS/cm		580	504		440	460	121	154	400-600	1100	1920	
pH		7.1 - 7.5	7.1			8.1	7.2	7.0	6.9	7.2 - 8.5	9.0 min 6.5	8.0	
Colour	Pt mg/l		50	40				95	85				
$KMnO_4$ consum.	mg/l	16-32	33	41		32	33	73	68	5-35	50	-	140
COD_{Mn}	O_2 mg/l		8							1.2-9	12.5	-	35
Ca^{2+}	mg/l		97 as Ca	35 as Ca				9	16	20-60	100	-	
Mg^{2+}	"							16	19	15-30	150	-	
Na^+	"									(~20-40)	200	-	
K^+	"												
Fe - total	"	0.4 - 0.7	7.9	0.12	2.0	29	15	0.26	0.22	1-20	30	0.13	
Fe^{2+}	"				0.5	11	15			0.5 - 15	30		
Mn^{2+}	"	0.1 - 0.4	1.1	0.08		0.30	0.17	0.05	0.06	0.1 - 0.5	3	0.05	
HCO_3^-	"	>90	381	246		390	390	53	55	150-600	500	92	
CO_2	"	0-14	(9 aggr)	(1 aggr)		27	21	7 (aggr)	10 (aggr)	0-25	50	0	
Cl^-	"	30-60	18	27		40	45	9	11	20-100	400	558	
SO_4^{2-}	"		<1	22		10	9	8.8	7.6	20-60	100	40	
NO_3^-	"		0.38	2		0.24	0.23	0.22	0.29	0.1 - 2	10	0.01	
PO_4^{3-}	"		0.11	0.1				<0.01	0.17	0.1 - 0.6	1	0.19	
F^-	"					1.0	1.0	-	-	0.5 - 3	8	-	
SiO_2	"		19			20	22	2.87	17	15-40	60	14	
HS^- (total)					<0.1	5	5	-	-	<0.2 - 5	30	-	
NH_4	"		0.02	<0.1		0.04	0.14	0.10	0.04	0.1 - 0.4	5	0.02	
NO_2	"		<0.001	0.01		0.075	0.11	0.00	0.0	0.01 - 0.1	0.5	0.00	
°dH	°dH	7-14	13.6	4.9		7.1	7.3	5.0	6.6	6-15	50	12.8	
O_2	mg/l				(<0.6)	<0.01	<0.01			<0.01	1		
TU	-												
Age	Years												

x) The groundwater may be contaminated with surface water.

xx) Estimated probability that the max. value will not be exceeded is 95%.



6 RETARDATION AND RETENTION OF WASTE SUBSTANCES

Various sorption effects and other chemical processes generally lead to a retardation of the substances dissolved in the groundwater in relation to the movements of the groundwater. Laboratory studies and field tests have been conducted in order to shed light on these factors by Allard /6-1/, Landström et al. /6-2/ and Neretnieks /6-3/. The results are generally in agreement with what is reported in the literature /Burkholder et al. 6-4/.

On the basis of the experimental data, retardation factors can be calculated for the different substances /Grundfelt 6-5/.

These and related matters are dealt with in greater detail in volume IV, chapter 6.5.

Field tests concerning such retardation effects were conducted at Studsvik for the National Council on Radioactive Waste Management (PRAV). In these tests, tracers were injected into boreholes at a depth of 70 m in fissured rock with heavy water flow and permeabilities around 10^{-5} m/s. The groundwater flow was accelerated by means of pumping in another hole and samples were taken from a hole between the injection hole and the pump hole as well as from the discharge water. The transmit time of the water was determined with the aid of a water tracer. The test confirmed the retardation effect on strontium and cesium.

In a later study, which was commissioned by KBS, the tests were repeated after the same rock sections had been sealed by means of bentonite grouting. Bentonite is a commonly occurring natural mineral many millions of years old consisting primarily of smectite minerals. Smectite occurs frequently as a natural crack filler in the Swedish bedrock and has also been found at Kråkemåla and Karlshamn. It is in chemical equilibrium with the groundwater and the other minerals in the bedrock /Garrels, 5-10/. The tests are still in progress, but it can be noted that strontium added to the water has not yet (after 4 months) arrived at the metering point located 50 metres from the borehole where it was injected /Landström and Klockars, 6-6/. The transit time of the groundwater over this distance was about 10 hours prior to sealing.

Certain elements take part in chemical reactions so that they are retained in the rock. Such a fixation of cesium has been demonstrated by laboratory experiments /Levi and Miekely, 6-7/. Other experimental studies have shown that hydrogen sulphide or minerals containing bivalent iron can precipitate insoluble uranium

dioxide from solutions of carbonate complexes of hexavalent uranium by means of reduction at room temperature /Rafalsky, Miller, 6-8, 6-9/. Theoretically, the same should occur with plutonium and other transuranium elements.

Many examples of such reactions are found in nature. Thus, extensive uranium ore deposits have been formed by precipitation in this manner /Adler, 6-10, Dahl and Hagmaier, 6-11/. In Sweden, uranium dioxide occurs as fissure filler in the crystalline basement rock in such areas as northern Uppland and at Pleutajokk in Norrbotten county /Adamek and Wilson, 6-13/. In both of these cases, the mineral has been retained in the bedrock for more than 1 500 million years. It has also been shown that naturally formed transuranium nuclides in the Oklo uranium field at Gabon have not been dissolved or carried away by the groundwater /6-14/.

7 FUTURE BEDROCK MOVEMENTS

Special studies have been devoted to the question of whether current rock conditions may significantly deteriorate during the long period of waste storage in a rock repository due to new fracturing and future movements in the bedrock. The formation of new fissures could then lead to higher permeability in the rock. Future displacements could also cause damage to the vitrified waste bodies and their canisters. These questions are further explored below.

7.1 ROCK MOVEMENTS AT KARLSHAMN

The exposed rock at the Karlshamn power station provide a good opportunity for examining the extent of the displacements and dislocations which have occurred due to fracturing in a selected area of bedrock. There are recently blasted vertical road cuts as well as large, smooth and clean sections of rock outcrops along the shore with a combined area of many thousands of square metres. The bedrock contains numerous slices of light pegmatite. At points where these slices of light pegmatite are crossed by fractures, the magnitude of the vertical (in the road cuts) or horizontal (in the shore outcrops) displacement in each individual fracture since the formation of the pegmatite can be measured. The pegmatites were formed $1.45 \cdot 10^9$ years ago /Welin and Blomqvist, 7-1/.

Two independent observers have studied the area. There is no significant difference between their reports. The results are summarized in the following table:

Displacement (mm)	1	1-2	2-5	5-10	10-20
Number, observer I	(44)[a]	25	11	5	1
Number, observer II	37	16	15	4	
Percent	51	26	16	6	1

a) Calculated in proportion to the number of cracks exhibiting major displacement.

The studied areas of rock have been exposed to glacial action frost bursting and other disintegration processes (blasting in the road cuts) which have acted on the exposed top surface of the rock. Rock elements at greater depths would have been subjected to uniform constraint from all sides by the surrounding rock. It

can therefore be concluded that the total vertical and horizontal movements caused by similar fractures at greater depths are less than 20 mm. The resulting average displacement has thus been less than 0.02 mm per million years. Similar observations at Finnsjö Lake indicate a value 15 times greater. The time dependence of the movements will be discussed later on.

7.2 FAULT MOVEMENTS IN THE VICINITY OF THE STUDY AREAS

Besides internal fissuring, the bedrock also exhibits continuous fracture lines, sometimes many miles in length. These fracture lines divide the bedrock into blocks of varying size which have been displaced more or less in relation to each other. Such fracture zones are called faults. The magnitude of the vertical displacements (throws) in the subcambrian peneplain (see 4.3.1) can be established both in the region around Finnsjö Lake and around Kråkemåla. In northern Uppland, such displacements are generally less than 15 m, while the total movement in the Oskarshamn district is around 30 m. This gives an average vertical displacement in these zones of movement of approximately 6 cm per million years or less during the 570 million years which have passed since the peneplain was formed.

Of special interest is the north-south fault which runs through the Götemar granite west of the study area at Kråkemåla. The subcambrian peneplain here has been displaced some 25 m, while the total movement is estimated on the basis of crystalline bedrock geology to be around 500 m /Kresten and Chyssler, 4-12/. This shows that about 95% of the movement took place more than 570 million years ago, and that the average speed at that time was about 10 times higher than afterwards.

7.3 FAULTS IN SKÅNE

One of Europe's major zones of movement runs through Skåne and can be followed towards the southest all the way to the Carpathians /Yanshin et al., 7-2/. In Skåne, this zone marks the boundary of the Scandinavian Precambrian shield towards the younger bedrock to the south. The boundary zone is characterized by faults which have exhibited major movements during the past 570 million years as well.

Röshoff and Lagerlund /7-3/ have studied two of these faults in relation to the sedimentary rocks of Skåne and their ages, and were thereby able to establish the vertical displacements which have taken place during four different geological periods: over 180, 200, 130 and 65 million years, respectively. The result obtained for the larger fault was a total vertical throw of nearly 2 000 m and an average vertical throw of 3.4 m per million years. Average displacements of similar magnitude were noted for the different periods - the variation is between 4.9 m per million years and 2.6 m per million years. The smaller fault exhibits a total vertical throw of around 240 m and its average rates vary in a similar manner between 0.59 m and 0.23 m per million years. Röshoff and Lagerlund emphasize that these data are mean values taken over long periods of time and that it is assumed that the

movements took place in rapid steps separated by long periods of little or no movement.

It is also of interest that the younger rock formations in Skåne exhibit fewer faults than the older underlying Precambrian bedrock. The number of faults in chalk deposits is only about 20% of the number in the bedrock. Approximately one-third of the faults in the Silurian strata and half of the faults in the Triassic strata do not extend down into the Precambrian bedrock. Conditions are similar in many other areas with edimentary bedrock, and have also been simulated in instructive model studies /7-4/.

7.4 RECENT ROCK MOVEMENTS OF THE FAULT TYPE

Rock movements which occurred in Sweden during or since the ice age and are still in progress have been known for a long time, but have often been regarded with some doubt. The AKA Committee's report made special mention of the occurrence of such movements in south-eastern Sweden. Since that time, new and more definitive observations have been made. A review of these observations and a general regional inventory of recent fracture zones in southern Sweden has been carried out by Röshoff and Lagerlund /7-3/. A similar study for central and northern Sweden is reported by Lagerbäck and Henkel /7-5/.

According to these reviews, the most important examples of recent faults are to be found at Kullaberg in Skåne and in certain recently geologically surveyed parts of Norrbotten and Västerbotten counties. An area exhibiting indices of recent fissuring of another character has also been noted on the Fulu massif in Dalarna. The irregularities in the land elevation process noted by Mörner /7-6/ should also be mentioned here.

The inventory of fracture zones has included studies of satellite pictures and a review of topographical maps and aerial photographs. No certain, previously unknown, recent faults have been found, with the possible exception of a fracture zone at Ekorva on the Åsheda topographical map-sheet.

This is no guarantee that recent faults have not occurred in other areas, but it does indicate that large parts of Sweden lack clear examples of such zones of movement. In particular, it should have been possible to discover recent faults, if such existed, in the very flat areas where KBS bedrock studies were conducted.

Further studies are required to obtain a complete picture of the formation and importance of reported recent rock movements. But an important factor which recurs in the cited reports is that these recent movements are associated with older zones of movement in the Precambrian bedrock. It is said that the faults at Kullen in Skåne were initiated more than 570 million years ago, and that subsequent movements have taken place during several geological periods, including since the ice age. It is also noted that the recent faults in Norrbotten and Västerbotten counties conform largely to older faults. Mörner also shows that the irregularities which he reports in shorelines and in the land elevation rate are caused by the fact that movements are still taken place in the contacts between different types of rock and

in the fault lines in the bedrock. This agrees with the basic geological observation that large cracks and fissure zones are very old and have been reactivated in recent periods of deformation /Cloos, 7-7/. This has also been confirmed by a large-scale study of the entire structure of Eurasia by Yanshin et al. /7-2/ and in Scandinavia by Tuominen et al. /7-8/ as well as Strömberg /7-9/. Knowledge of existing zones of fracture is of great importance in planning a rock repository. By locating the repository in an area without major fault lines, and in a block of bedrock which is bounded by joint planes where any existing stresses can be released, it is possible to isolate the repository from the effects of recent fault movements.

7.5 ROCK MECHANICS STUDIES

One reason why recent fracture and fault movements follow older fracture lines is that such fracture lines represent already existing joint planes where stresses can be released more easily. This has been elucidated by means of rock mechanics calculations by Stephansson /7-10/. The results show that the risk of fissuring decreases as the number of existing fissures increases and the distance between them decreases. A sample calculation shows that a displacement of 10 m/km in rock with a fissure interval of 2 metres does not give rise to new fissures. Instead, the movement is distributed over existing fissures, and the change in fissure width, and thereby in permeability, is negligible.

In this context, it should also be noted that certain rock mechanics parameters have been determined in the laboratory for drill core samples from the areas at Finnsjö Lake and Kråkemåla. Tests on materials from Karlshamn are still in progress. The results obtained thus far are presented in table 7-1.

No measurements of the internal stresses in the bedrock have been undertaken within the study areas. However, a relatively large number of stress measurements have been carried out in Scandinavian bedrock at varying depths down to about 900 m over the years 1951-1976. The results show a very wide spread both geographically and locally. Some of this spread seems to be a result of the fact that earlier measurements /7-11/ indicated considerably greater stresses than recent measurements /7-12, 7-13/ using improved and more well-defined measuring methods. There are, nevertheless, undoubtedly genuine wide regional and local variations in the internal stresses in the rock due to various geological factors, topography etc. /7-13/.

In general, it can be concluded that the measurements indicate the existence of larger horizontal stresses than could be expected on the basis of purely theoretical elasticity considerations. The same applies for the maximum shear stress, which determines the risk of fracture in the rock mass. However, the shear strength of the rock increases almost linearly with increasing depth due to the pressure of overlying rock masses /7-14/. Judging from the rock stress measurements which have been made, it can therefore be concluded that, apart from isolated cases, adequate safety margins against rock fracture exist at the depths in question due to the internal stresses of the rock. The same conclusion can be drawn on the basis of the results of the earlier

Table 7-1. Rock mechanics parameters from drill cores, determined at the Division of Rock Mechanics at the Luleå Institute of Technology.

	Compressive strength	Tensile strength	Modulus of elasticity, 50% breaking load	Poisson's ratio
Kråkemåla 1 (6 samples)	MPa	MPa	GPa	
M (Mean value)	188.2	8.92	61.4	0.20
d (Standard deviation)	17.8	0.94	4.3	0.03
Kråkemåla 2 (6 samples)				
M	152.7	6.77	57.1	0.21
d	18.7	1.59	6.7	0.05
Finnsjön 1 (6 samples)				
M	252.5	13.47	81.5	0.18
d	7.7	2.30	3.6	0.01
Finnsjön 2 (6 samples)				
M	228.8	13.50	83.6	0.21
d	11.2	1.50	2.4	0.02

measurements. This conclusion is also in full agreement with practical experience from mining operations at various depths.

Naturally, measurements of rock stresses will be included in the arsenal of preliminary study methods which will be used to determine a suitable location for a final repository.

7.6 FRACTURING FORECAST

It is possible to forecast the formation of new fractures and fracture movements during the storage period on the basis of average values and their expected deviation. Such considerations can be based on the age of the bedrock and the local frequency of fractures in each area. The existing fractures in the bedrock constitute a natural record of all previous occasions on which permanent cracks were formed.

As an example, consider a 1 000 m long section of a rock formation which is 1 000 million years old and where the number of cracks is also 1 000 (in other words, the average distance between the cracks is 1 metre). This means that an average of one new fissure was formed every million years. Assume that this

section is representative for the rock surrounding a rock repository. A storage period of one million years with an average amount of rock movement would then lead to an increase in the fissure content of the rock, and thereby its permeability, by a factor of 0.001 of the present value. A more qualified forecast of this type of future has been worked out by Ringdahl et al. /7-15/. Such a forecast is dependent upon whether the rock movements during the forecast period are greater or less than average. Another factor which must be taken into consideration is that the risk that a deformation will generate a new fracture diminishes as the distance between existing fissures decreases.

The question of whether the rock movements which will occur during the storage period will correspond to an average frequency for a very long period of time can be clarified by considering the rock movements in their chronogeological context. It is important to begin by noting that the most of the fracturing of the Precambrian bedrock outside of Skåne took place more than 570 million years ago, i.e. prior to the formation of the subcambrian peneplain. In the Karlshamn area, the fracturing took place primarily about 900 million years or more ago. The fracture pattern at Kråkemåla is for the most part 1 300 million years old. The general mineralization of the fractures in the Finnsjö area and their proximity to the Jotnian rock formations at Gävle indicate a similar or greater age /see Wiman 7-16, Welin 6-12, Gorbatschev 7-17 and Welin and Lundqvist 7-18/.

The generally small amount of movement in the Precambrian bedrock over the past 570 million years is evident in the large extent and limited deformation of the subcambrian peneplain and in the largely undisturbed stratification of the overlying aluminous slates and limestones. Disturbances in the peneplain consist of an elevation of the land surface of up to several hundred metres which can be followed from the west coast to Skellefteå and which comprises the western boundary of the peneplain. Then there are also the previously mentioned faults, which divide the bedrock into blocks. The faults are normally steep and the displacement is predominantly vertical.

All of this shows that fracturing and movements in the precambrian bedrock over the past 570 million years have been small and below the average for a longer period of time. The above-cited data from the Götemar fault show that the aggregate deformation there was about 20 times less, and the deformation rate between 5 and 10 times lower, during this period than previously. Similar conditions have also been reported for other parts of the country by Röshoff and Lagerlund /7-3/. This abating tendency in rock movements indicates that the previous forecast is based on excessively high average values. Thus, change in the rock during the storage period will be considerably less than calculated.

However, movements in the bedrock have not been constant during the past 570 million years either. Peaks in the fissuring process can be established on the basis of the general connection which exists between the fissuring of the earth's crust and volcanic activity. The following age determinations have been reported for volcanic formations in Sweden which are younger than the Precambrian bedrock /see Klingspor 7-19, Byström et al. 7-20, Kresten et al. 7-21/:

- 540 million years, alkaline rocks from Alnön and Åvike near Sundsvall.
- 450 million years, bentonite from Kinnekulle in Västergötland.
- 295-280 million years, basaltic rocks in northern Skåne and Västgötaberg, alkaline rocks in Särna.
- 167 million years, basalts, northern Skåne.
- 108 million years, basalts, northern Skåne.

No more recent basaltic volcanism has occurred in Sweden, but faults in the chalk deposits in Skåne indicate movements which have taken place during the past 65 million years.

These data show that the past 570 million years, with their lower average fracture movement data, have also had periods of higher activity. The concentration of these events in Skåne is clear.

Age determinations make it possible to consider movements in the Precambrian bedrock in a larger context. Three major and extensive deformation periods during the past 570 million years have led to extensive orogeny, granitization, fracture movements and volcanism in the areas between the Mediterranean Sea and the North Atlantic. During all of this period, the Nordic bedrock has remained an extremely stable area, whose low level of deformation is proved by the subcambrian peneplain and superimposed rock strata. It may be added that many other precambrian rock areas in the world exhibit a similar appearance, while areas with more active rock movements and volcanism are also characterized by different conditions with respect to age, rock types and geological structural features. Probing deeper, it can be noted that basalt volcanism in Sweden and associated break-up of the subcambrian peneplain approximately 290 million years ago coincides with similar activities in Greenland, Europe, East Africa, Madagascar and Western Australia /see Kent 7-22, Turner and Verhoogen 7-23/. Similarly, more recent basaltic volcanism and fault movements in Skåne can be chronologically associated with other areas, of which the North Atlantic and the continental part of Europe north of the Alps are of the greatest interest as far as our forecast is concerned.

A survey of bedrock movements and their age relationships in the North Atlantic was recently submitted by Bott /7-24/. Volcanic activity in the area is summarized by Turner and Verhoogen. The geological development which is described began some 180 million years ago and culminated about 40 million years ago. At that time, volcanic activity extended from Greenland, Jan Mayen and Spitzbergen all the way to Ireland, at the same time as northern Europe separated from North America. The rock movements in Skåne can be regarded as a marginal part of this progression, where volcanism comprised an introductory phase. The same progression also affected other parts of northwestern Europe, although with much less intensity. This probably explains the weak upwarping which constitutes the western boundary of the subcambrian peneplain in Sweden, the much greater upfaulting which gave rise to the steep Atlantic coast of Norway and the subsidence of the previously forested parts of the Baltic Sea, which are the source of Baltic amber. The studies in the North Atlantic make it possible to follow developments over the past 100 million years as well, although observations in Sweden concerning this period of time are very incomplete. The decrease in rock movements over

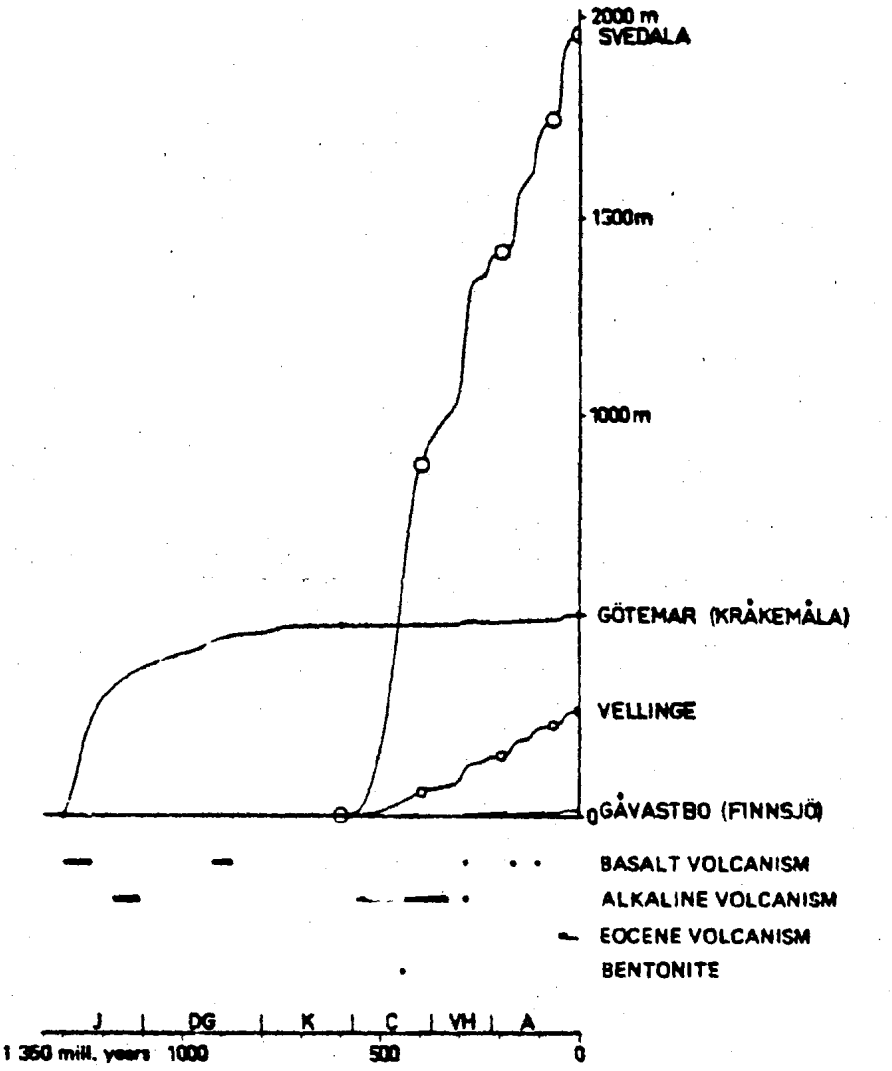


Figure 7-1. Diagram of fault movements, deformation periods and volcanism over the past 1 350 million years. The diagram shows the elevation in metres of the upthrown rock block in relation to the subsided block at four studied faults. The curves connecting the points were drawn on the basis of the peneplain formation of the Precambrian rock (flat portions), volcanism (deep portions) and the character and thickness of the sedimentary bedrock. The values on the flatter portions of the curves for the elevations of the faults in Svedala and Vellinge are approximate between the measured points. The large movements between 400 and 600 million years ago in Skåne (Svedala and Vellinge) have only insignificant counterparts within the Precambrian rock area. Only the Götemar fault can be followed back farther than 600 million years. Bentonite indicates extensive volcanism in the area of the Scandinavian mountain chain (the Caledonians). Eocene volcanism took place outside of Scandinavia. The thickness of the vertical axis corresponds to the next 4 million years.

The following deformation periods are marked on the time axis:

J	*= Jotnian*	*C*	*= Caledonian*
DG	*= Dalsland–Grenville*	*VH*	*= Variscan–Herrynian*
K	*= Katanga*	*A*	*= Alpine*

the past 40 million years, as well as the fact that volcanism is currently restricted for the most part to Iceland, show that the entire region is now in a period of insignificant and abating rock movement.

A similar picture is obtained of the continental part of Europe north of the Alps /Rutten 7-25/. This region once contained a somewhat fragmented volcanic province which extended from France to Poland. This volcanic activity has been connected with later phases in the formation of the Alps. As in the North Atlantic, volcanism in this province reached a maximum during the Tertiary period and has diminished steadily since that time. The most recent volcanic eruption occurred 11 000 years ago in southern Germany. The rock movements in Scandinavia can also be regarded as a part of this phase.

The movements which took place in connection with certain faults in Sweden during various periods and which can be derived from geological observations are presented in figure 7-1 in order to depict schematically the progression of movement and the difference between the Precambrian shield and adjacent areas.

Thus, extensive regional observations support the conclusion that movements in the Swedish bedrock constitute part of an extensive and long-range process which is clearly in slow decline. Realistic estimates of how soon new fractures can be expected to occur must therefore be well below the mean values obtained from forecasts based on the age of the bedrock and the local fracture frequency alone. From this it is evident that changes in the fissure content and permeability of the rock due to rock movements during the storage period will be so small that they cannot have an adverse effect on the function of a rock repository which is suitably situated in relation to existing fracture and crush zones.

7.7 LAND ELEVATION AND GLACIATION

The current land elevation in Sweden has been studied on the basis of field observations /Mörner 6-7/ and on the basis of gravitational considerations /Bjerhammar 7-26/. Despite the differences in basic data and analysis techniques, quite similar results were obtained. But there are certain discrepancies which should be the object of further study.

In summary, it can be said that the land elevation following the ice age reached a maximum of nearly 1 000 m on the coast of Ångermanland. For the most part, the land elevation represents a rebound of the land following its depression by the weight of the inland ice. According to Mörner, this rebound ceased 2 000 - 3 000 years ago, and the current land elevation has other causes. A rebound from the elevation of western Scandinavia and subsidence of the Baltic Sea, which was dealt with in the previous section, could possibly explain this process. According to Bjerhammar's analysis, the land is still rebounding from the depression caused by the inland ice.

Fracturing and movements in the bedrock in connection with the land elevation and in connection with a future ice age can be assessed on the basis of the present distribution of fractures in the bedrock. Permeability values from the study boreholes show

that fissuring is limited for the most part to the uppermost 100 or 200 m of the bedrock. Deeper portions which have undergone the same movements still possess good imperviousness. This shows that the land elevation and the preceding depression did not affect the permeability of the bedrock. Furthermore, there have been a total of 10 to 20 Quarternary glaciations /Kukla 7-27/ and the present state of the bedrock reflects the cumulative effect of all of these. This leads to the conclusion that one more glaciation would not disturb a deep rock repository.

7.8 EARTHQUAKES IN SWEDEN

The effects of earthquakes on a rock repository are discussed in recent studies by Dowding /7-28/ and Yamahara et al. /7-29/. The latter concentrated primarily on Japanese conditions. He notes in his introduction that no serious earthquake damage has been reported from Japan's many mines and tunnels. He also shows by means of rock mechanics calculations that earthquake movements in rock abate rapidly with increasing depth. At a depth of about 100 m, rock movements are only about 1/4 to 1/3 of what they are at the surface. The greatest induced rock stress at a depth of 100 metres when the surface acceleration is 2.24 m/s^2 is only 1.2 MPa. Even the severest conceivable earthquake in Japan would give rise to a maximum stress which is estimated to be only 3.0 MPa. These stresses are negligible in relation to the strength of the types of rock in question in Sweden. Dowding uses practical cases to show that no damage was caused to rock caverns by earthquakes where acceleration on the surface was 1.9 m/s^2 or less. This corresponds to an intensity of VII - VIII on the Modified-Mercalli Intensity scale (MM), which is a measure of the effects of earthquakes on the surface. Minor damage, falling stones and cracking were observed at accelerations up to 5 m/s^2, corresponding to an intensity of VIII-IX. The severest known earthquake in Fennoscandia, at Oslo in 1904, had an intensity of VII-VIII. The tunnels and storage holes in the repository are filled with compacted soil material, rendering minor damage to superficial rock insignificant.

Earthquakes are rare and weak in Sweden. The earliest known earthquake occurred in the year 1497. Since 1891, systematic statistics have been kept on earthquakes /Båth 7-30, 7-31/. Since 1951, earthquakes have been registered by means of sensitive instruments which can detect quakes at sea and in uninhabited regions as well. A review of known observations up to 1972 was published in a study by Kulhanek and Wahlström /7-32/. A map of Swedish earthquakes during the period 1951 - 1976 with comments and other material was obtained from Båth /7-33/.

The frequency of earthquakes exhibits wide variations during the observation period. The geographical distribution of earthquakes in Sweden and adjoining areas has, however, remained relatively unchanged.

Southeastern Sweden exhibits extremely few earthquakes. On the preliminary map in figure 7-2 provided by Båth /7-33/, only 3 quakes near the Roxen-Motala fault zone are shown for the period 1951 - 1976. Most of the quakes are instead located in a belt extending from the west coast, across the Lake Vänern district - where they are relatively numerous - towards Gävle and then along

the coast of the Gulf of Bothnia. From the northernmost part of the Gulf of Bothnia, the belt turns towards the northwest and then again towards the southwest in the North Sea and runs along the coast of Norway, thereby forming a circle around central Scandinavia /7-34/. The distribution of the quakes is thus related to bedrock movements on and off the Norwegian coast, and in Sweden a connection can be seen with the fault lines in the Vänern-Vättern region, the western boundary of the subcambrian peneplain and the areas of recent rock movement in Skåne, Västerbotten and Norrbotten. This would indicate - as is maintained by Kvale /7-35/ - that the Scandinavian earthquakes are related to bedrock movements which are independent of the ice age.

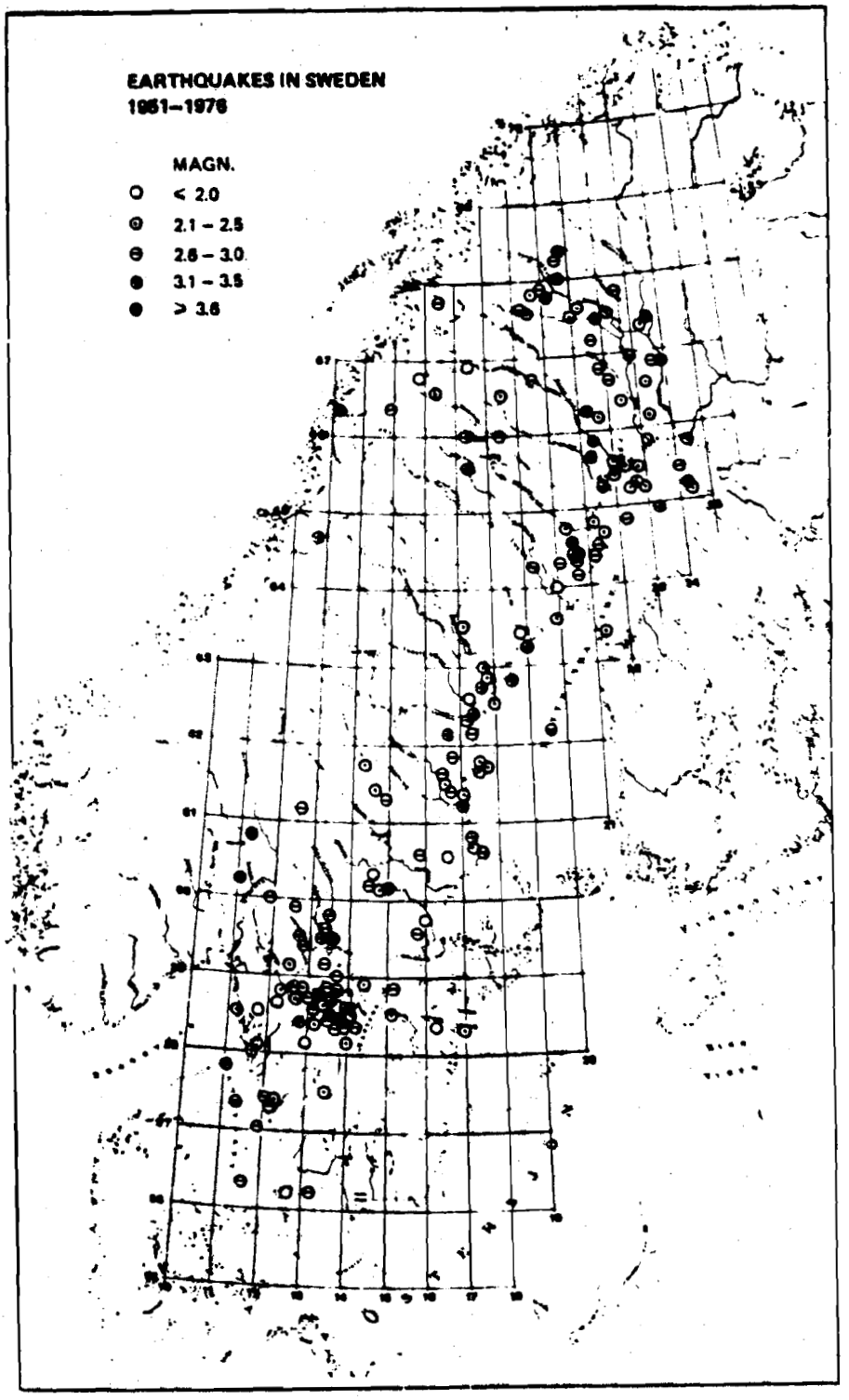


Figure 7-2. Map of earthquakes in Sweden registered by instruments during the years 1951–1976. Note the low number of quakes in southeastern Sweden. (Professor Markus Båth, Department of Seismology at the University of Uppsala).

8 SUMMARY AND EVALUATION

Bedrock and groundwater conditions have been studied in five areas: at Karlshamn, Finnsjö Lake, Kråkemåla, Ävrö and Forsmark.

In the first three areas, large volumes of uniform and sound rock with a permeability of around 10^{-9} m/s or less have been found.

The Karlshamn area exhibits the least occurrence of crush and fracture zones.

The groundwater flow in sound rock at depths of around 500 m within the areas has been calculated at 0.2 litres per square metre and year or less.

Determinations of groundwater age show that the transit time of the groundwater up to the surface of the earth from a rock repository in an area of inflow can amount to several thousand years.

Deep groundwater samples have been found to be oxygenfree and weakly alkaline with a pH of between 7 and 9. They contain bivalent iron in solution. Sulphide, determined as hydrogen sulphide, has been found in the groundwater at Forsmark.

The radioactive waste elements, with the exception of iodine and technetium, are retarded in the bedrock in relation to the movements of the groundwater.

Uranium, plutonium and other transuranium elements can be precipitated in the form of insoluble oxides of bivalent iron and sulphide or in the form of hydrogen sulphide in solution.

The effects of future rock movements in rock formations without major zones of fracturing and movement can be neglected.

The changes around a rock repository which are caused by blasting and the heat generated by the waste are of a highly local nature. The risk that new flow paths for the groundwater will be created in this manner is negligible.

The above conclusions, together with the safety analysis, provide a basis for the judgement that the three closely studied areas of Karlshamn, Finnsjö Lake and Kråkemåla fulfil the basic requirements for a safe rock repository for high-level waste. This assumes that the rock repository is properly situated in relation to the geometry of the existing formations of low permeability. The study areas on Ävrö and at Forsmark have been found to be

less suitable and the studies there have not been carried to completion.

On the basis of current knowledge, the Blekinge coastal gneiss area is the most attractive from a geological point of view of all the studied areas for a rock repository.

The slightly gneissic granite at Finnsjö Lake also offers large volumes of good imperviousness. Existing internal fracture and crush zones, however, may present certain technical problems of the type which are normally encountered in tunneling and rock cavern excavation. Compared to the Blekinge coastal gneiss, this type of rock permits greater freedom of choice in the location of a future rock repository, since similar rock conditions are found throughout large parts of southeastern Sweden.

The Götemar granite at Kråkemåla exhibits, despite sections of very low permeability, a number of features which may require more extensive reinforcement and grouting during the construction phase. These features include lower strength, a regular fracture system with extensive horizontal fracture surfaces and high local groundwater flow, which may also be associated with radon problems.

The three study areas can be clearly arranged in order of priority: the Blekinge gneiss, the gneissic granodiorite in the Finnsjö area and the undeformed stocklike granite in the Kråkemåla area. This confirms previous experiences regarding the structural and water-bearing characteristics of these types of rock. Against this background, other gneiss areas may also be of interest.

9 STRIPA EXPERIMENTAL STATION

9.1 OVERVIEW

9.1.1 Reasons for an experimental station

Only a very limited quantity of basic data on rock at great depths below the ground surface (500 - 1 000 m) are available today. Additional data should be obtained on which to base decisions regarding the siting, design and construction of a final repository for high-level waste. For this purpose, an experimental station at great depths is of great value. It also provides an opportunity for the demonstration of working methods and the design of the various sections of the final repository.

When KBS was organized in late 1976, it was decided that an experimental station should be established in the Stripa Mine, 15 kilometres north of Lindesberg. The ore in the mine was nearly played out and iron ore mining was scheduled to be discontinued in early 1977. Immediately adjacent to the mine is a granite massif which is directly accessible at a level 350 m below the surface. Since the personnel and equipment required for immediate commencement of the rock work was available, considerable savings in time and costs could be made in comparison with the alternative of constructing a new experimental station somewhere else.

An experimental station in rock permits the following studies and tests to be conducted:

- testing and demonstration of working methods for a final waste repository,
- detailed characterization and surveying of a rock massif a great depth,
- analyses and comparisons between the results of various measuring methods and actual conditions in order to evaluate the accuracy of the methods to be used in future rock studies,
- studies of how blasting, heating and grouting affect the rock and its characteristics,
- analysis of groundwater movement and groundwater composition at great depth,
- studies of the properties of the materials which may be used in a final repository.

A number of researchers and institutions with special experties in the field of petrology were contacted for assistance in plan-

ning appropriate studies. An advisory group of experts, the "Geo-group", was formed and instructed to propose suitable studies at Stripa. The group also participates in the evaluation of the test results.

Studies within the following subject areas have been initiated at Stripa:

- The effect of blasting on surrounding rock
- Rock characterization
- Rock stress measurements
- Material properties of the Stripa granite
- Permeability of the rock at different pressures and temperatures
- Thermal stresses
- Injection studies
- Water analyses

Most of the results obtained from KBS experiments at Stripa will be reported during the first quarter of 1978.

9.1.2 Construction work

Blasting work for the test station started in December of 1976. The final appearance of the excavated rock caverns is illustrated in figure 9-1. The tunnels were blasted using a technique known as smooth blasting and with nearly circular or oval sections in order to minimize stresses on the rock. Cross-sectional tunnel

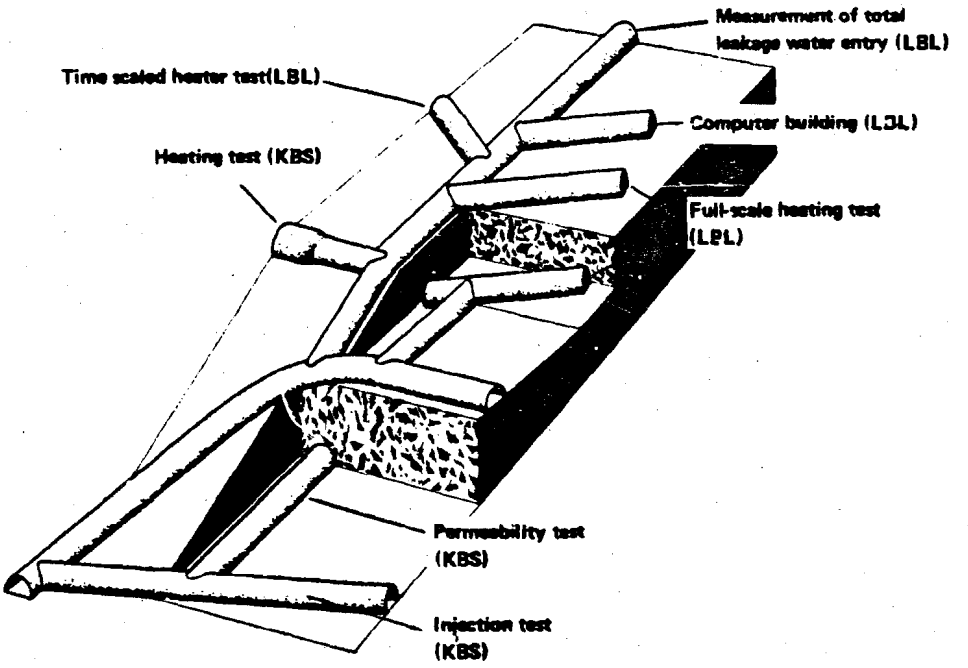


Figure 9-1. KBS experimental station at Stripa. The layout of the excavated rock caverns and the sites of the various tests are illustrated in the figure.

area varies from 12 m^2 to 26 m^2, depending on the space required for construction machines or testing activities.

The quality and strength of the Stripa granite is very good. Reinforcement bolting has not been considered necessary and no rock fall has been observed.

In order to permit the execution of planned experiments and tests, a large number of holes of varying length and diameter have been drilled from the tunnels. A total of approximately 1 500 metres of holes have been drilled for KBS's own tests, beyond those required for the blasting work. Most of the holes have been diamond drilled to a diameter of 56 mm.

Blasting and drilling work has been in progress for about 10 months and has employed around 20 men.

9.1.3 Cooperation with the US ERDA

KBS's experimental programme in Stripa has attracted international interest, and in the spring of 1977, KBS was contacted by a group from the USA in order to discuss the possibilities of a joint research project concerning waste storage in crystalline rock. The conditions for such a joint project at Stripa were clarified during the early summer, and an agreement was signed between SKBF (Swedish Nuclear Fuel Supplies Inc.) and the US ERDA (United States Energy Research and Development Administration). According to this agreement, SKBF shall be responsible for holding the mine open through October of 1979 and for all rock blasting work, rock drilling (approx. 3 800 m) and certain services on the worksite. The US ERDA will be responsible for and defray the costs of the actual study work. KBS is responsible for Swedish commitments during its period of activity.

The US ERDA has contracted Union Carbide to administer and execute an extensive development project concerning the final deposition of high-level waste. A special unit has been organized for this purpose within Union Carbide - OWI, Office of Waste Isolation. The execution of the Stripa programme has in turn been entrusted to a group of researchers at the Lawrence Berkely Laboratory, LBL, at the University of California.

The research programme was drawn up by LBL, but SKBF/KBS are kept continuously informed concerning the planning and execution of the tests and experiments and also have a voice in planning the experimental programme. The results of the tests will be the common property of the parties.

9.2 STUDIES CONDUCTED UNDER THE AUSPICES OF KBS

9.2.1 Rock characterization

In order to be able to evaluate the alternative sites for a repository for high-level waste, reliable and economically feasible methods are required to establish the properties of the rock. The experimental station at Stripa provides an opportunity for the testing and evaluation of such testing methods. For this purpose,

the properties of the Stripa granite must be well-known. For this reason, a rock characterization must be carried out. This includes:

- assembling all available geological information on the area in question,
- mapping of the network of fractures on the rock surfaces which have been exposed for the tests,
- core surveys and TV examinations of diamond boreholes near the tunnels before and after blasting,
- studies in a vertical deep borehole from the bottom of the mine (from the 410 m level to the 900 m level),
- water injection tests in all boreholes. In cases where blasting has been carried out near the hole, this test is performed both before and after blasting.

The dominant type of rock in the experimental area in the Stripa mine is a red to grey, medium-grained, unstratified granite. The granite contains a few narrow, steep veins of pegmatite and younger diabase.

The grain size of the essential materials in the granite varies between 1 and 5 mm. A typical sample exhibits the following mineral composition:

quartz	44%
plagioclase (partially altered)	39%
microline	12%
chlorite	3%
muscovite	2%

The granite at Stripa can be said to be representative of a large group of younger granites in central Sweden.

Despite its relatively high fracture frequency, the granite is highly impervious, with permeability values around 10^{-10} m/s. This can be explained by the high degree of crack filling.

9.2.2 Rock stress measurements

The purpose of the tests is to establish the primary stress state in the Stripa granite. This information is required for certain other tests and as a basis for theoretical calculations of stress and flow conditions.

Three-dimensional stress conditions were measured at 19 points along a 20 m long borehole from a side drift. Measurement cells based on Leeman's method were used in the procedure. The measurements showed that the maximum main stress is 10 MPa and is oriented parallel to the strike of the granite. The intermediate main stress, 5.7 MPa, is nearly horizontal and is oriented perpendicular to the contact. The minimum main stress is 2.7 MPa. The measured vertical stress, 9.8 MPa, is of the same order of magnitude as the theoretically calculated stress of 9.2 MPa. A detailed report of obtained results is provided in KBS technical report No. 49 /9-1/.

9.2.3 Properties of the Stripa granite

In order to interpret the obtained data and carry out theoretical calculations, information is required on the mechanical and physical properties of the rock. Theoretical calculations and practical test results can then be compared in order to provide a more reliable basis for the evaluation of planned areas for waste disposal.

The following data have been obtained:

- Modulus of elasticity, Poisson's ratio and uniaxial compressive fracture stress at 25°C, 50°C, 100°C and 200°C.
- Compressive fracture stress and elasticity properties as a function of normal stress.
- Coefficient of linear expansion as a function of radial load.
- Brazilian tensile fracture stress.
- Residual shear strength as a function of normal stress.
- Anisotropy ratios in strength and elasticity properties.
- Coefficient of thermal conductivity.

The following data have been calculated for the Stripa granite and do not exhibit any significant deviations from normal data for central Swedish granite.

Poisson's ratio	0.21
Young's modulus	69.4 GPa
Uniaxial compressive strength	207.6 MPa
Tensile strength	15.0 MPa
Thermal conductivity	3.0-3.6 W/C°m

A detailed account of obtained results is provided in KBS technical report No. 48 /9-2/.

9.2.4 Permeability of the rock at various pressures and temperatures

This test is intended to provide information on variations in the permeability of the rock at different temperatures.

The test equipment is shown in fig. 9-2. The leakage water flow into a 300 mm diameter and 10 m long vertical borehole was measured in the test. In a circle at a distance of approx. 1 m from the 300 mm hole, 16 3" holes were drilled. Water was pumped into these holes at a given pressure and temperature. The rock mass was heated to the desired temperature by circulating hot water in the boreholes. By changing the water pressure at different temperatures and measuring the quantity of water leaking out of the centre hole, permeability can be determined as a function of pressure and temperature. The results of the tests are then compared with theoretical calculations.

Measurements of the permeability of the rock carried out thus far have given values which are approximately 10 times lower than results from conventional "packer test" measurements. The test used here probably gives more correct values of the actural permeability of a large rock formation. This indicates that the permeability values obtained from boreholes in the field studies may be too high. Preliminary test results have also shown that permeabi-

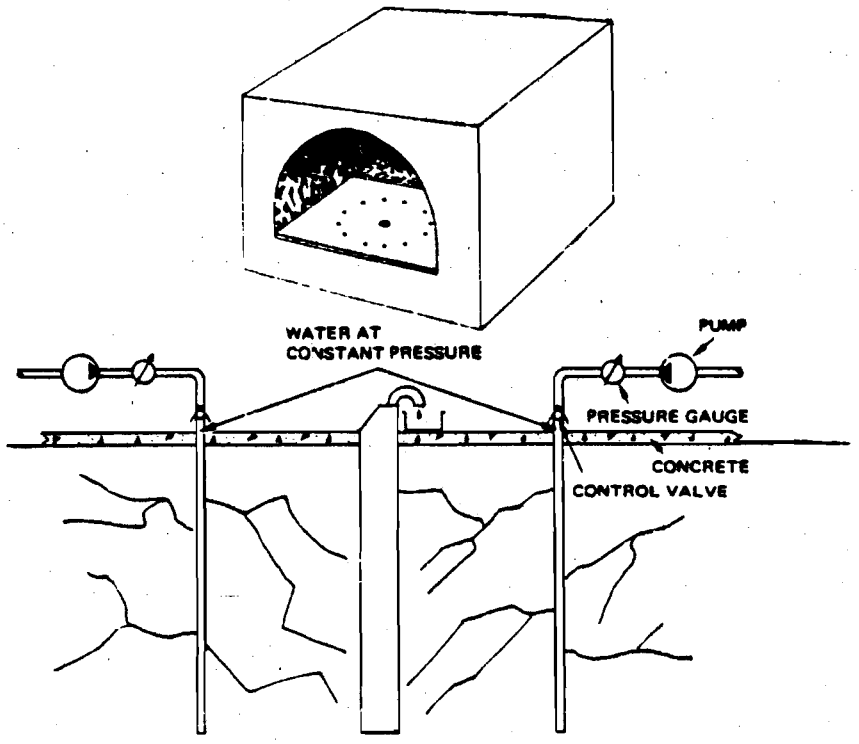


Figure 9-2. Permeability measurements. By varying the water pressure and the temperature, the manner in wich the permeability of the rock varies with pressure and temperature is measured.

lity decreases to about half when the rock is heated from 10° to 40°C.

9.2.5 Thermal stresses

This test is aimed at elucidating changes in existing rock stresses and fracture conditions which occur in a rock formation in connection with local heating. Owing to the low thermal conductivity of the rock, a relatively long test period is required for such a test.

The test set-up is illustrated in figure 9-3. The primary 3-dimensional rock stresses in the rock adjacent to the test drift are measured. The heaters and measuring holes are oriented so that their direction coincides with the direction of one of the main stresses.

The heat source consists of a specially designed 5 kW electric heater which is lowered into a 300 mm borehole. Three 1 kW auxiliary heaters are lowered around this centre hole at a distance of about 1.25 m. Temperature and rock stresses are then measured continuously in ten boreholes parallel to the centre hole and located in three directions and at a distance varying between 2 m and 6 m from the centre hole. The instruments also register changes in the width of major fissures within the test area.

The test is also aimed at recording conditions during a cooling-

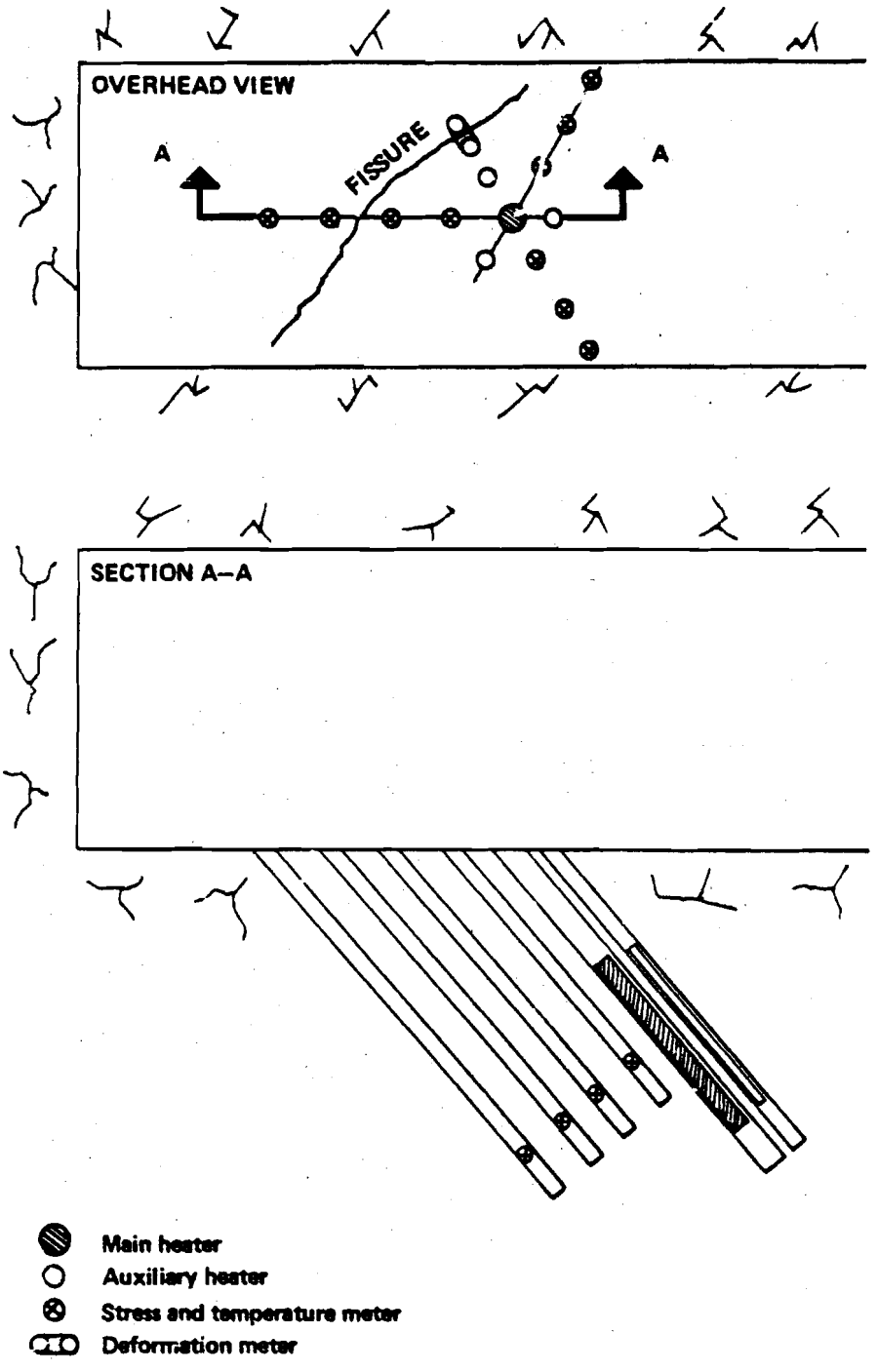


Figure 9-3. Measurement of changes in rock stresses and fissure widths in connection with local heating.

off period after a max. temperature of 60°C has been reached in the central portions of the test area. The obtained results will be compared with theoretical calculations.

9.2.6 Grouting

A final repository for radioactive waste should be located in rock which is as impervious as possible. Nevertheless, allowance must be made for the fact that some local rock volumes will be of higher permeability. It is then possible to inject such formations with material which will remain intact over a very long period of time.

A suitable grouting material seems to be silica (quartz). This is a highly durable material and is also available in very fine-grained form so that it can penetrate into very small fissures. However, during the preparatory work for a planned experiment, the permeability of the rock was found by hydraulic testing to be so low that grouting was not considered necessary.

9.2.7 Water analyses

The chemical composition of the groundwater is of importance for the rate of corrosion of the encapsulation and buffer materials and for the leaching rate of the waste glass.

Relatively few data are available on the composition of groundwater at great depths.

At Stripa, the groundwater is accessible at various levels from 350 m down to about 900 m below the surface. Since the mine has been drained for a long time - the mine was opened about 500 years ago - the groundwater conditions are disturbed, especially down to the level of the lowest point in the mine (-490 m). The samples which have been collected are therefore not representative of groundwater at corresponding levels under undisturbed conditions.

The following available results from analysis of the groundwater are worth mentioning:

- its age 340 m below the surface of the ground is about 15-20 years,
- its chemical composition shows low mineral and salt contents and
- the pH of the water is about 8.5.

9.3 STUDIES UNDER COOPERATION WITH US ERDA

A large-scale programme of tests and studies is planned by LBL at Stripa. The content of this work can be summarized in the following points:

1 Investigation of how the properties of the rock (pressure, expansion, thermal conductivity etc.) are affected by local heating.

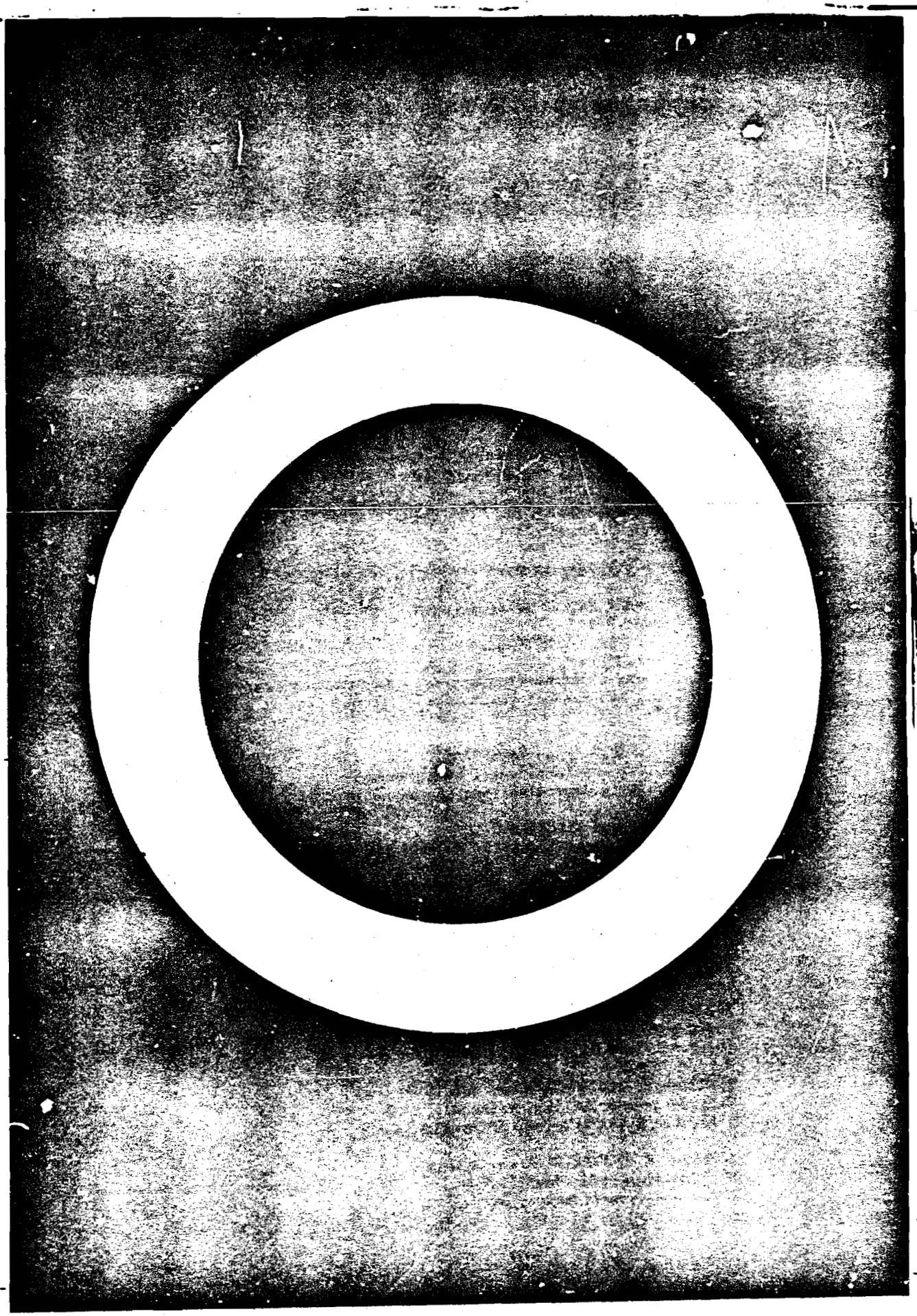
2 Mapping of the fissures in the rock (extent, size, direction contents etc.) with the aid of borehole studies and geophysical surveys.

3 Laboratory determination of various material data for the rock (microcracks and permeability as a function of pressure and temperature).

4 Measurement of the total leakage of water into a rock cavern for determination of the permeability of the rock.

5 Measurement of rock pressure by forcing water into boreholes to the point of fracture (hydraulic fracturing).

As previously mentioned, these experiments are planned to be carried out during 1978 and 1979. The first results of the tests will therefore probably not be available until 1978. Consequently, no results can be reported in KBS's main report, and only incomplete results will be available when KBS concludes its activities in the middle of 1978.



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

I General
II Geology
III Facilities
IV Safety analysis
V Foreign activities

KÄRN-
BRÄNSLE-
SÄKERHET

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

III Facilities

KÄRN-
BRÄNSLE-
SÄKERHET

teleplan AB Teleplan SOLNA Sweden 1978/Ultrasoftset AB SOLNA

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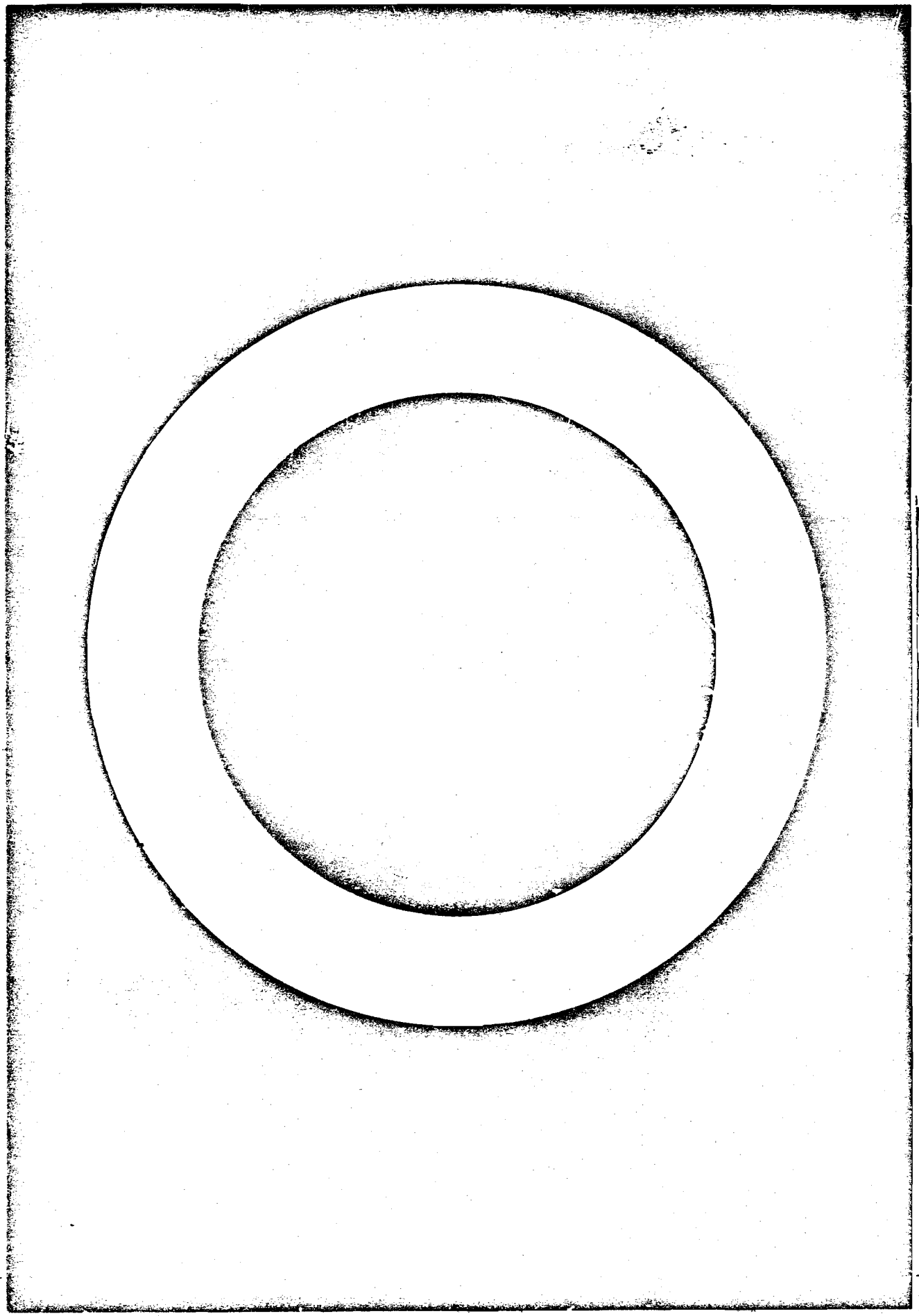


1 INTRODUCTION

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.

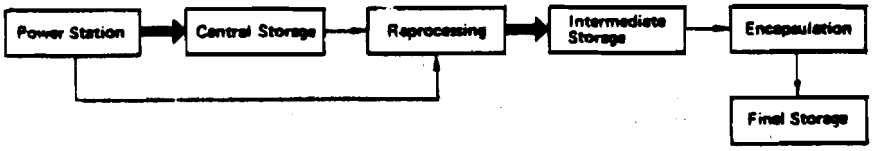
As in the AKA Committee (Government Committee on Radioactive Waste) study it is assumed that the final disposal of the high-level waste will be accomplished by deposition in crystalline bedrock possessing the appropriate properties. The calculations and evaluations in this report are based on a repository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

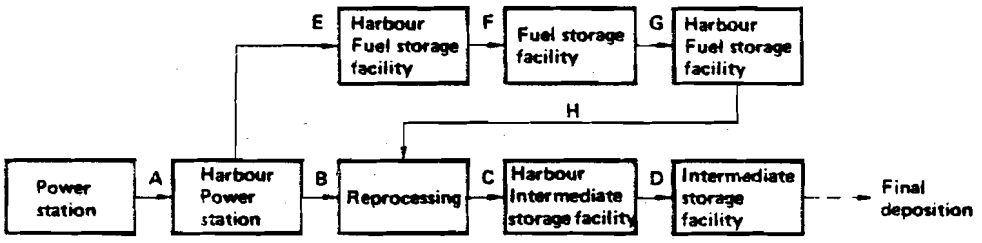


Figure 2-1. Flow chart of transportation within the back end of the fuel cycle. B packaging is required for transport stages A–H. Transports A, D, F and G go by road.

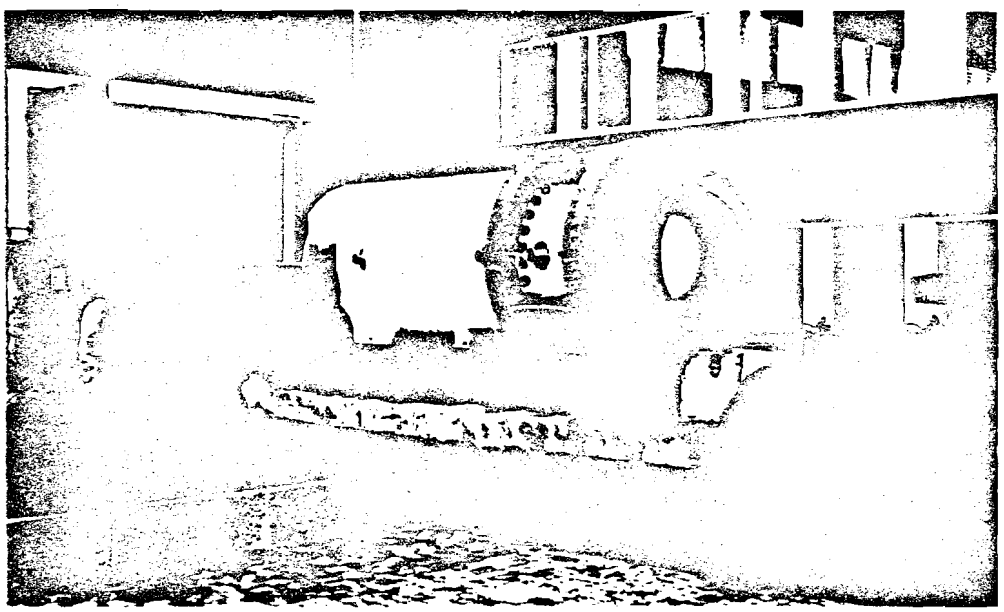


Figure 2-2. Trailer and transport cask (B packaging) outside of Oskarshamn plant. This equipment has been used to transport spent fuel to the harbour for further shipment to the reprocessing plant at Windscale.

reprocessing plant, responsibility for the transport cask is assumed by the reprocessing company's transportation organization. The Swedish transportation orgarization 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	4/200	7/200	7/200	7/200	3/125	-	12/230	7/215	12/215	5/230	5/215	5/215
BWR	9/114	-	19/114	12/140	-	7/140	-	17/140	30/140	-	14/140	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)	3875	3380	4370	4675	4280	4520	5050	4630	4580	5160	4674/4776	4887
diam. (mm)	440	864	864	864	3x230	474	1220	914	1220	914	864	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 Excellox 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-

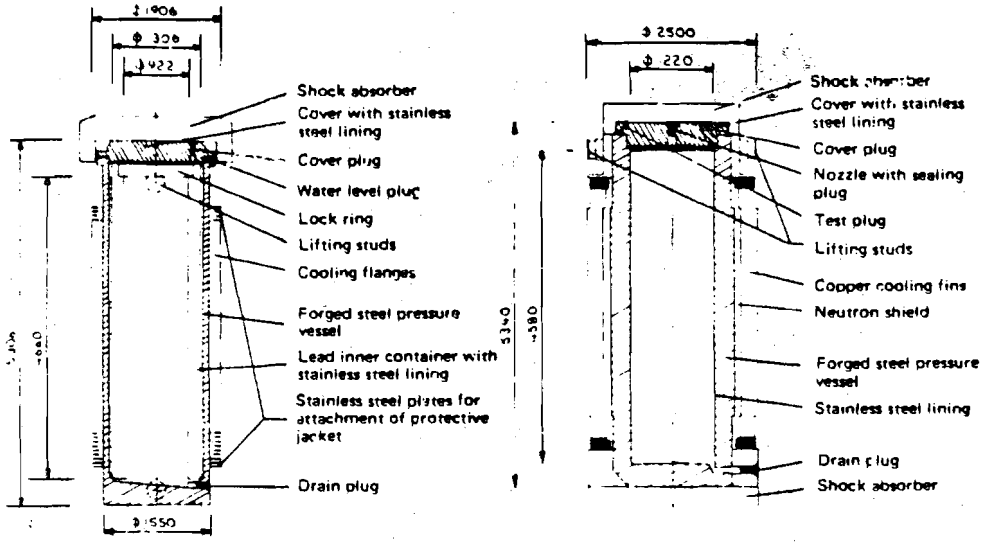


Figure 2-3. NTL 11 transport cask for spent fuel (left).

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.

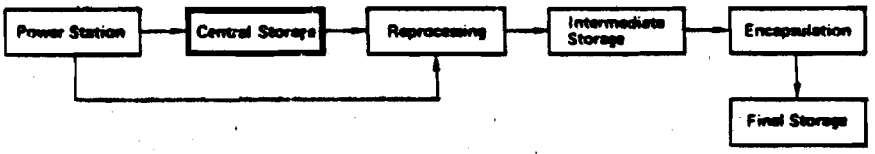
Waste cylinders will be transported from the reprocessing plant to Sweden in transport casks which are in principle identical to those which are used for spent nuclear fuel. NTL 12 is one of the types which may be used. This cask can transport up to 6 tons of

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.

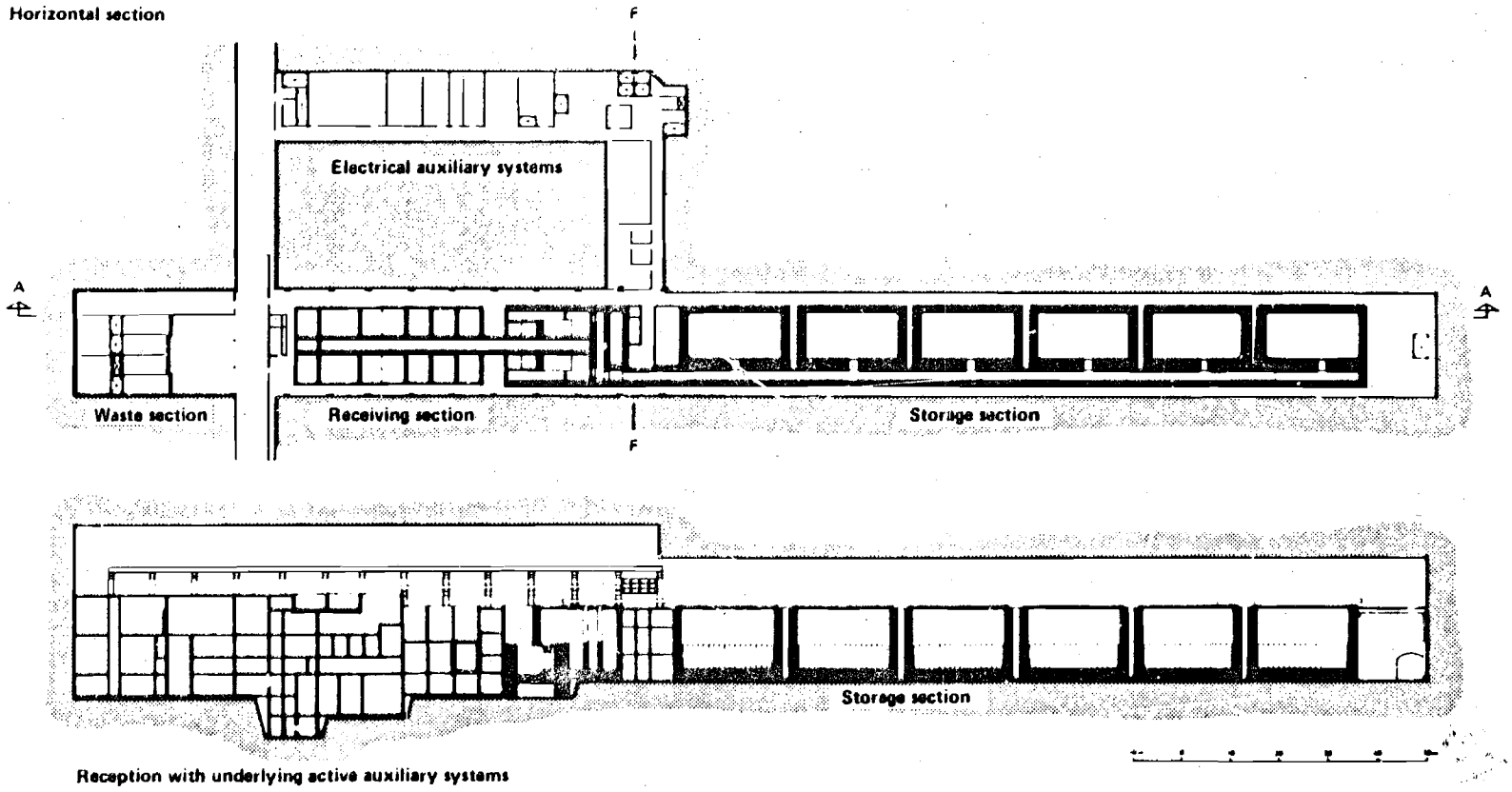


Figure 3-1. Horizontal and longitudinal sections of central storage facility for spent fuel. (From conceptual study by National Council for Radioactive Waste Management).

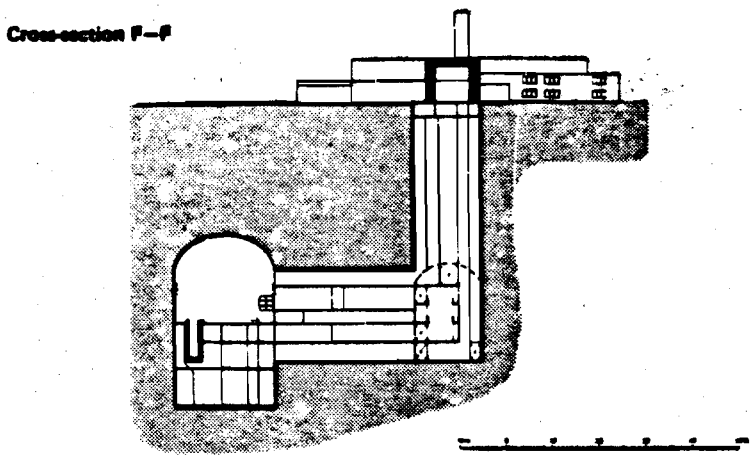


Figure 3-2. Cross-section of central storage facility for spent fuel. (From conceptual study by National Council for Radioactive Waste Management).

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.

5.3.2 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.

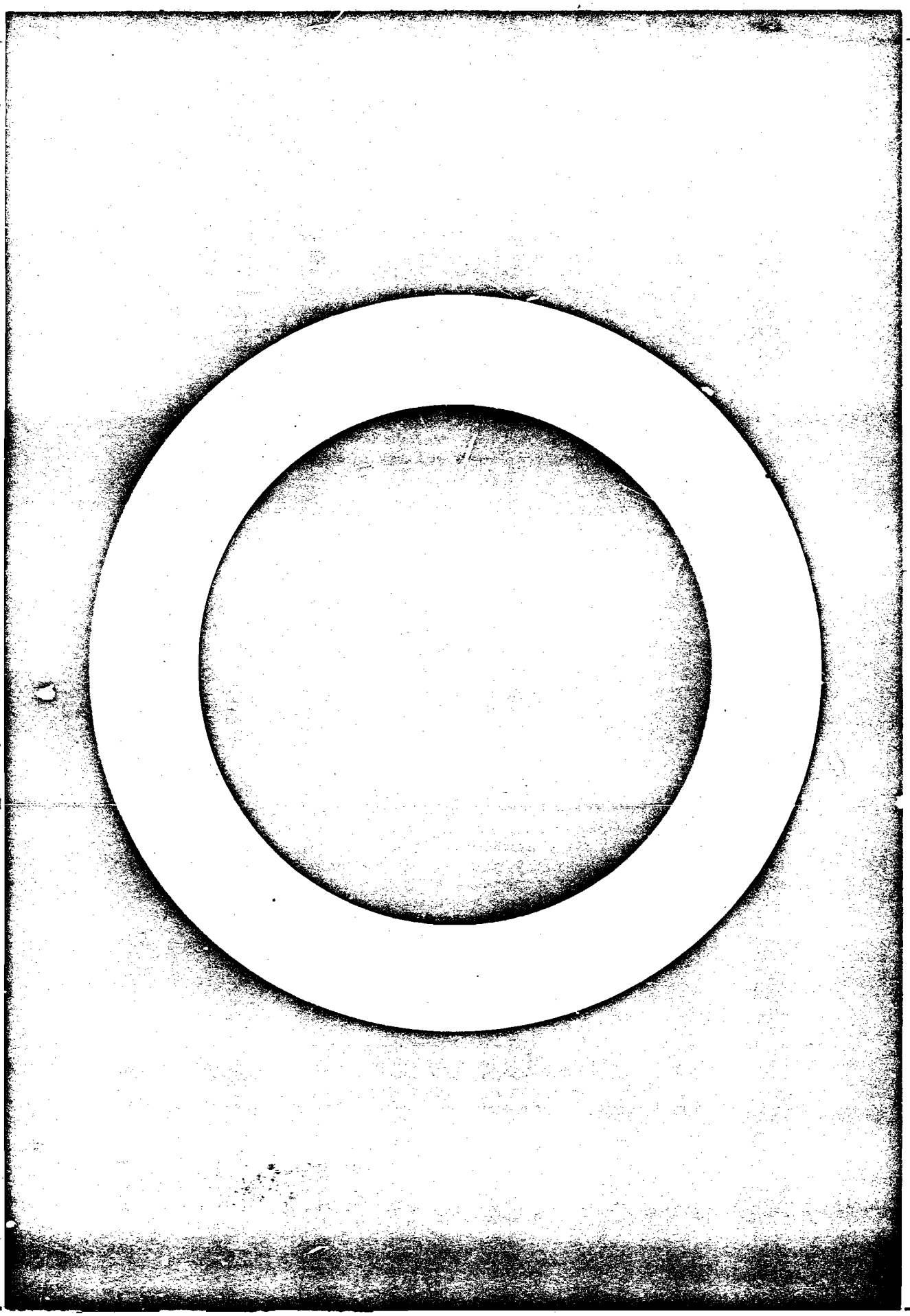
The decommissioning of a central fuel storage facility poses fewer problems than the decommissioning of a nuclear power plant. This is primarily due to the fact that the central storage facility does not contain heavy equipment or permanent installations which are highly radioactive.

3.5 OPERATION OF FACILITY

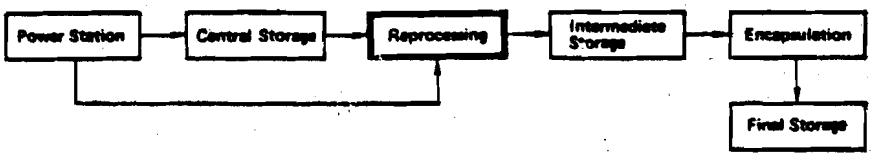
The central storage facility for spent nuclear fuel will be under the supervision of the same authorities as a nuclear power plant, namely the National Nuclear Power Inspectorate, the National Institute of Radiation Protection etc. These authorities issue directives and regulations governing both the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

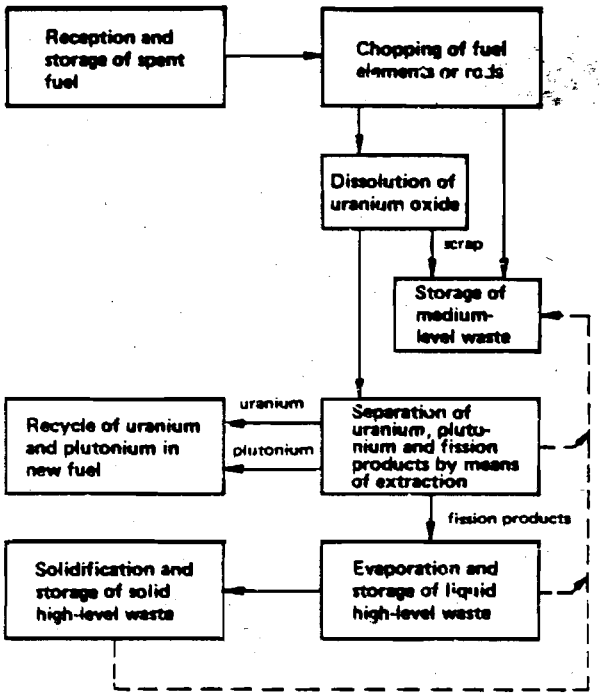


Figure 4-1. Process scheme for reprocessing of spent fuel from light-water reactors.

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 iodine, 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-thick 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 ie 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 burn-up.

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.

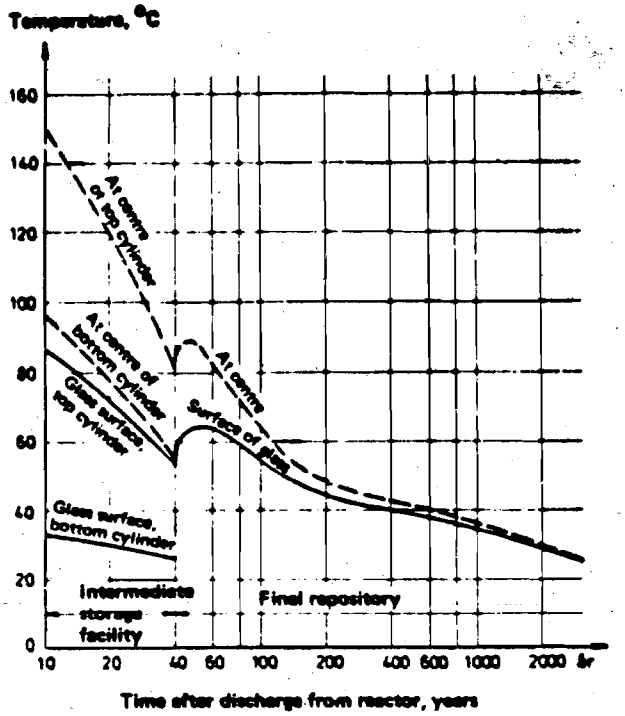


Figure 4-2. Temperature of the vitrified waste in the intermediate storage facility and the final repository.

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	Developed He 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

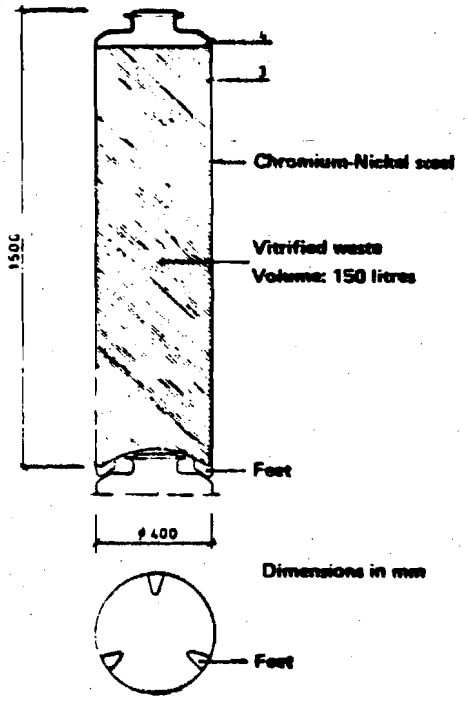
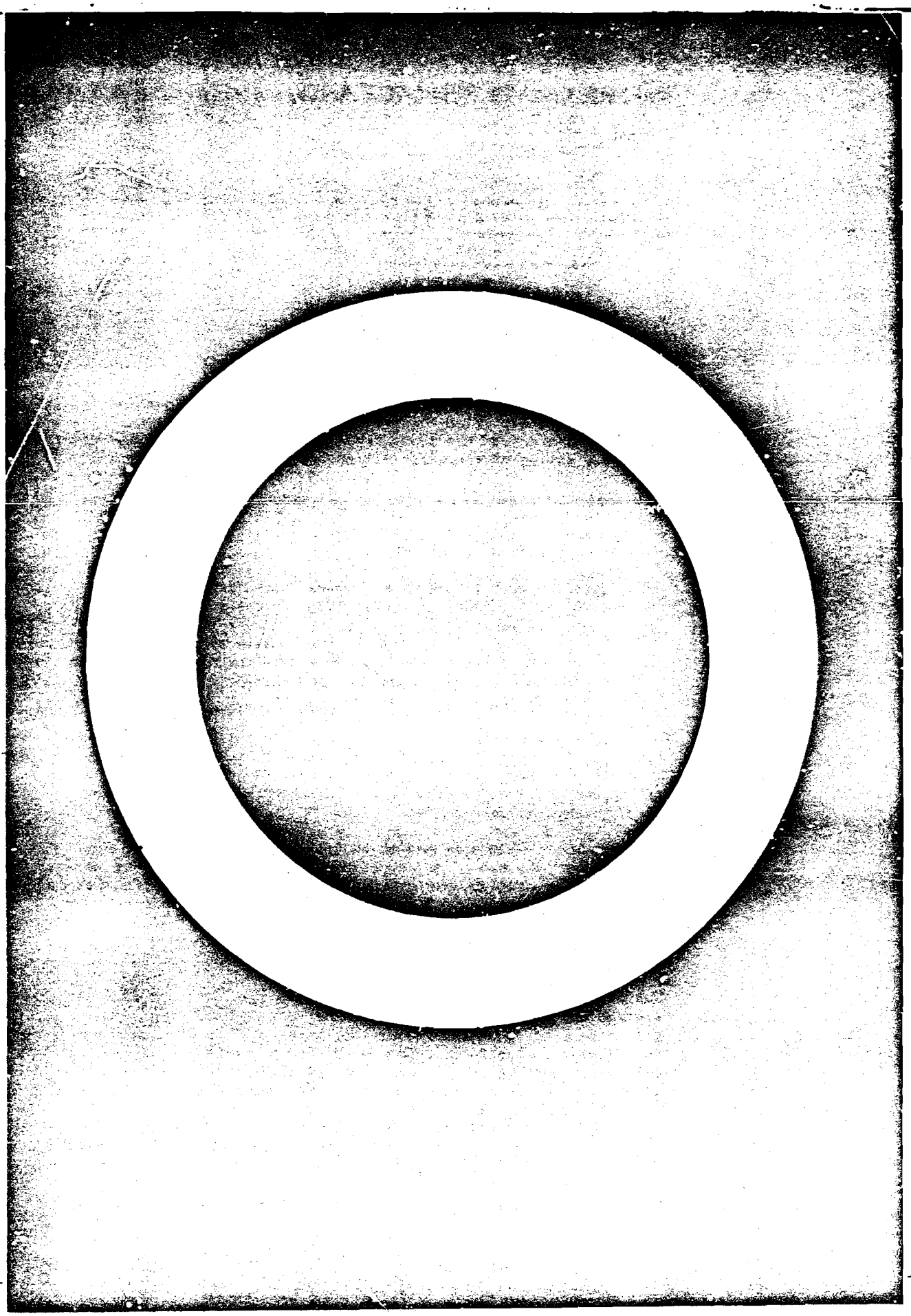
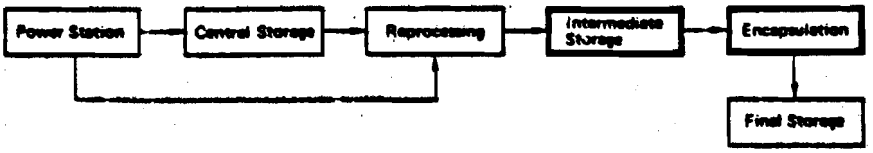


Figure 4-3. Waste cylinder. The vitrified waste is cast in a container made of chromium-nickel steel. The container is sealed with a welded-on lid. The feet enable the waste cylinders to be stacked on top of each other.



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.

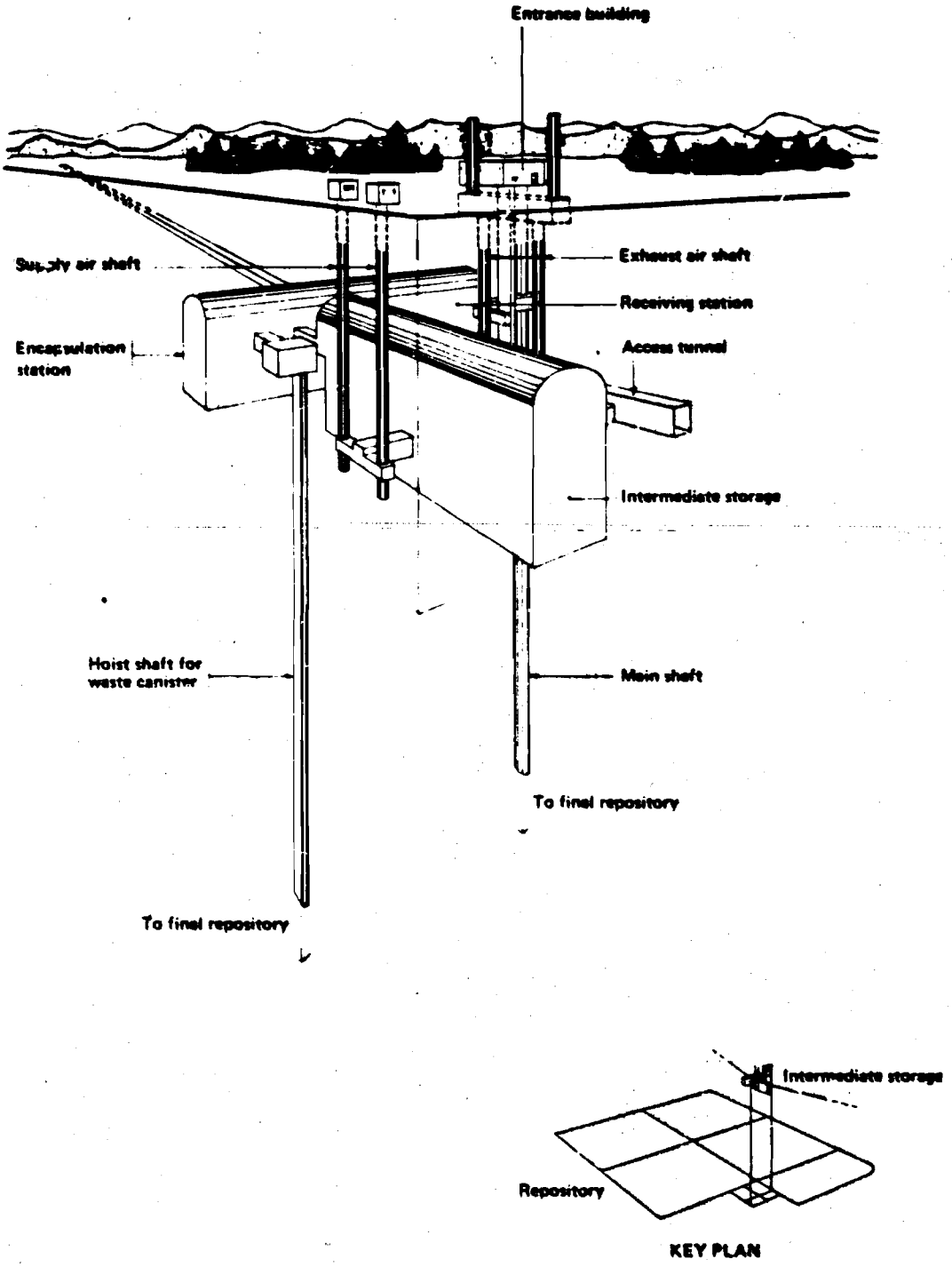


Figure 5-1. Perspective drawing of plant for intermediate storage and encapsulation. It is located underground with a rock cover approximately 30 metres thick. The plant is located above the final repository.

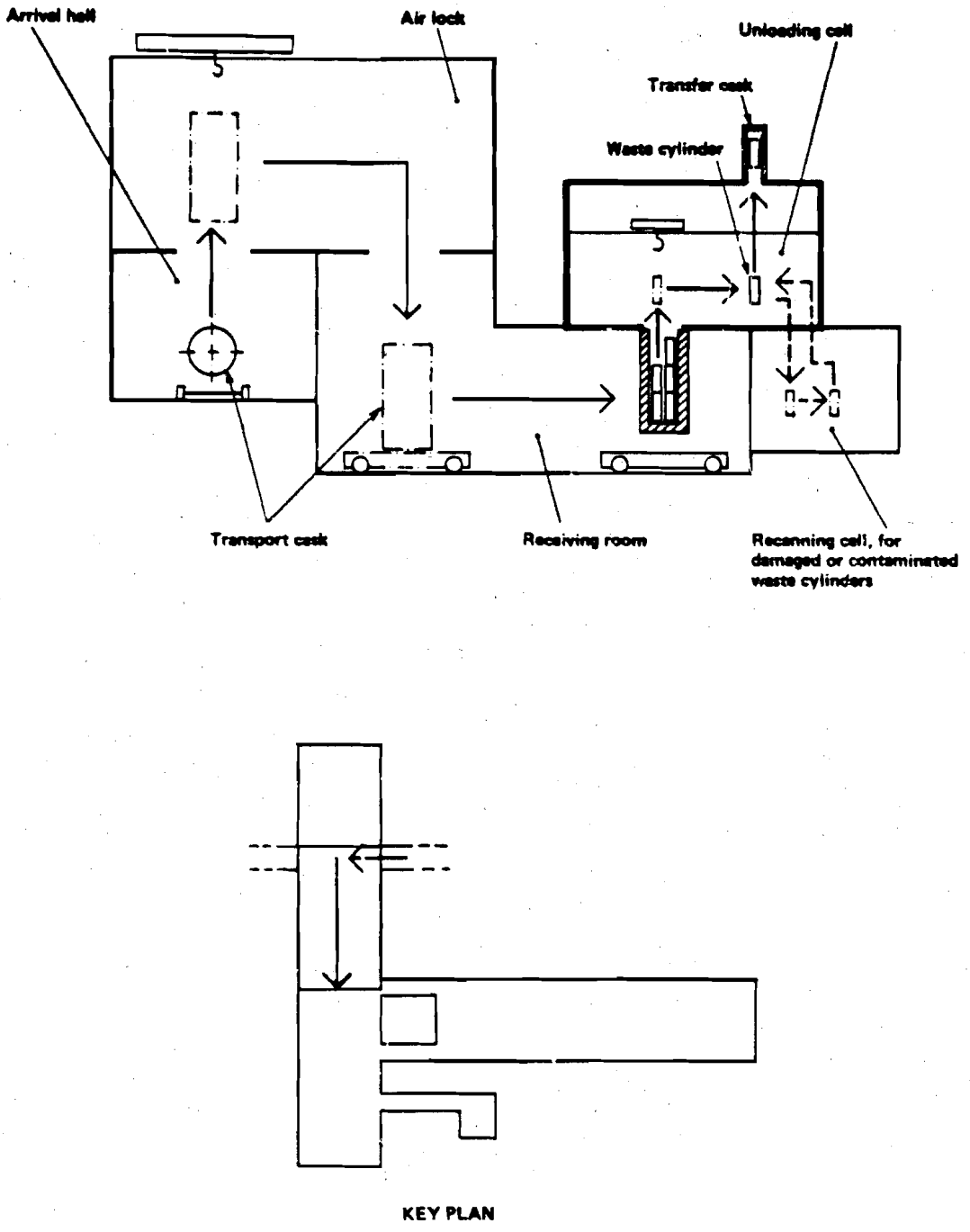


Figure 5-2. Schematic diagram of reception. In this section, transport casks are received and the waste cylinders are unloaded. Damaged or contaminated cylinders are encased in an outer container of chromium-nickel steel.

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

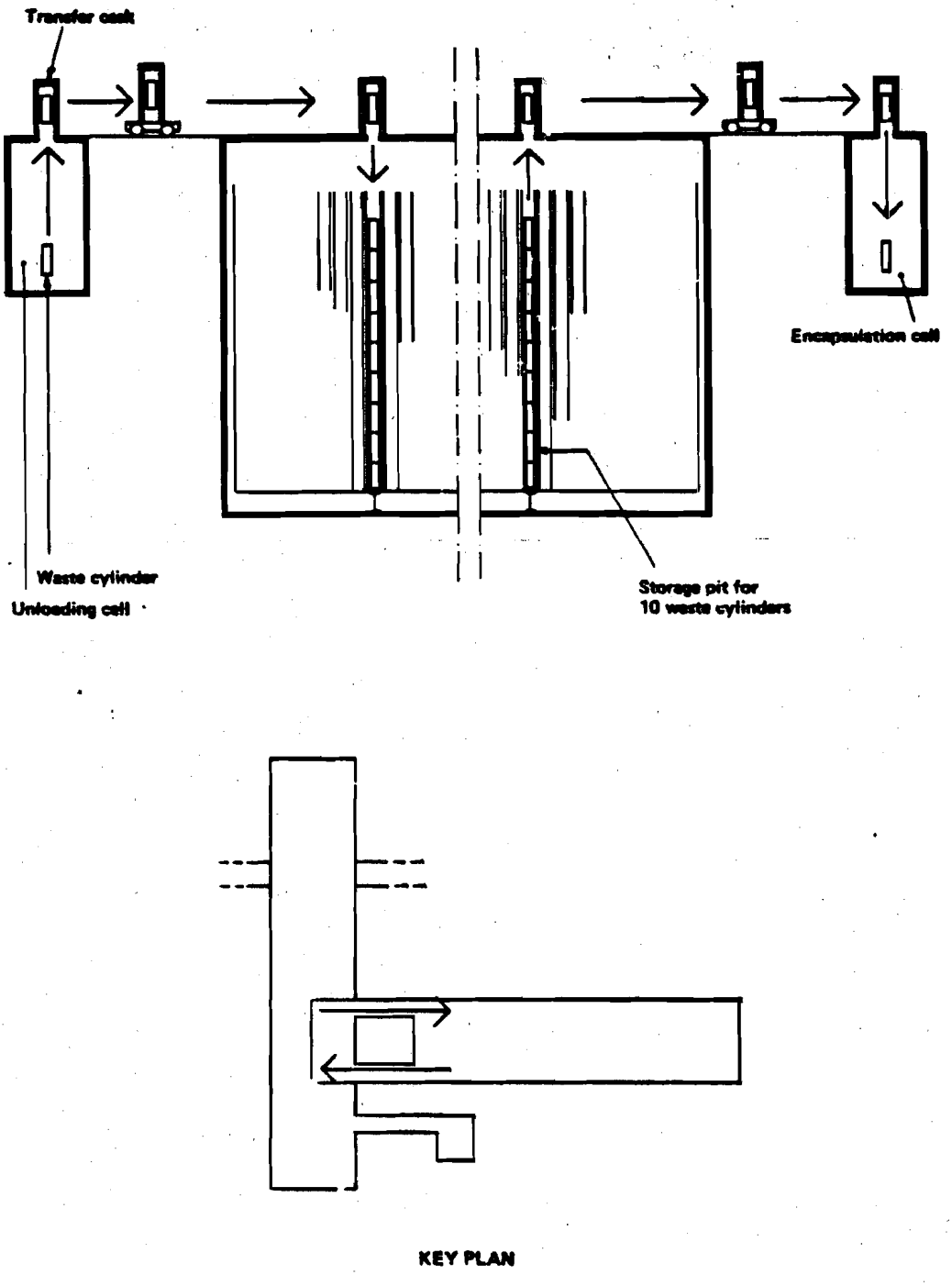


Figure 5-3. Intermediate storage. The waste cylinders are transferred to the intermediate storage section inside a transfer cask. After storage for at least 30 years, the cylinders are transferred to the encapsulation cell.

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.

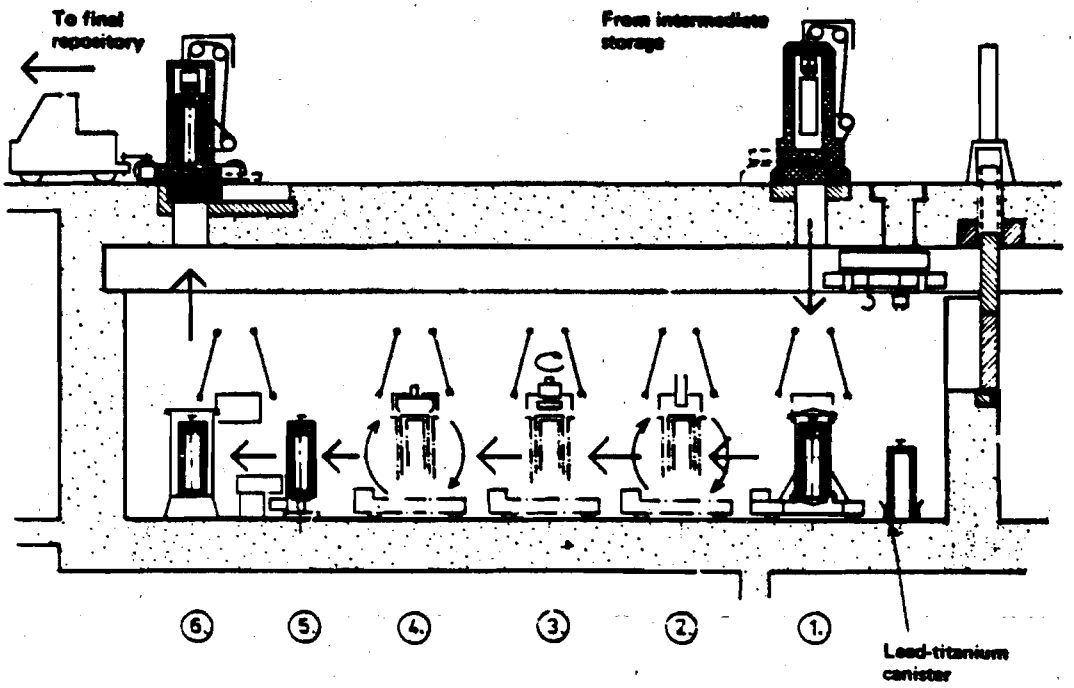
At the first station, a prefabricated part of a lead-titanium canister is placed over the waste cylinder. This part of the canister (which is fabricated outside of the facility) is brought into the cell through an opening in the roof. The opening is sealed by a concrete plug when it is not being used.

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.

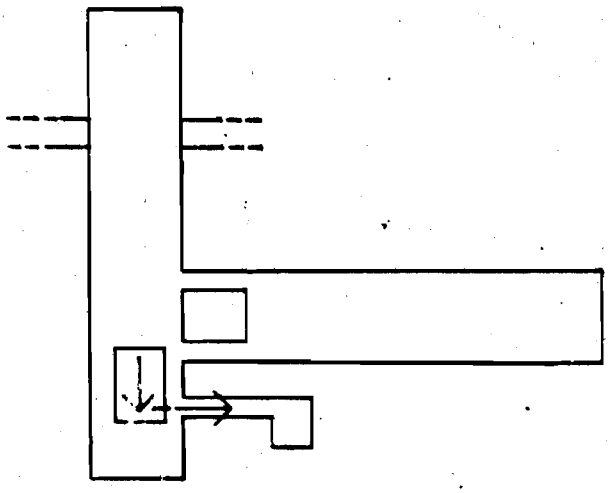
At the third station, the surface of the lead which was pured at the preceding station is machined (following cooling) in order to facilitate the attachment of a titanium lid.

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

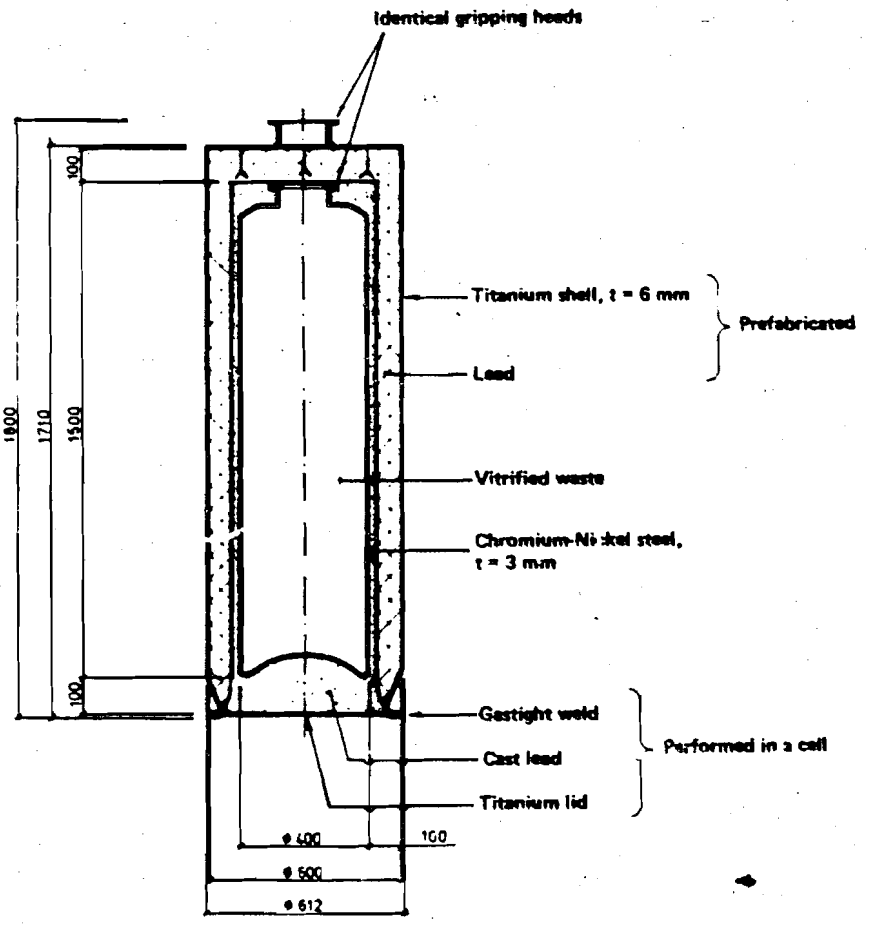


Figure 5-5. Lead-titanium canister for waste cylinders with vitrified waste. Total weight approx. 3.9 ton.

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, drainage etc.) in the event of an external power failure. In the event of malfunctions of pumps for the groundwater drainage 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°C /5-2/ when the temperature of the supply air is 20° 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,

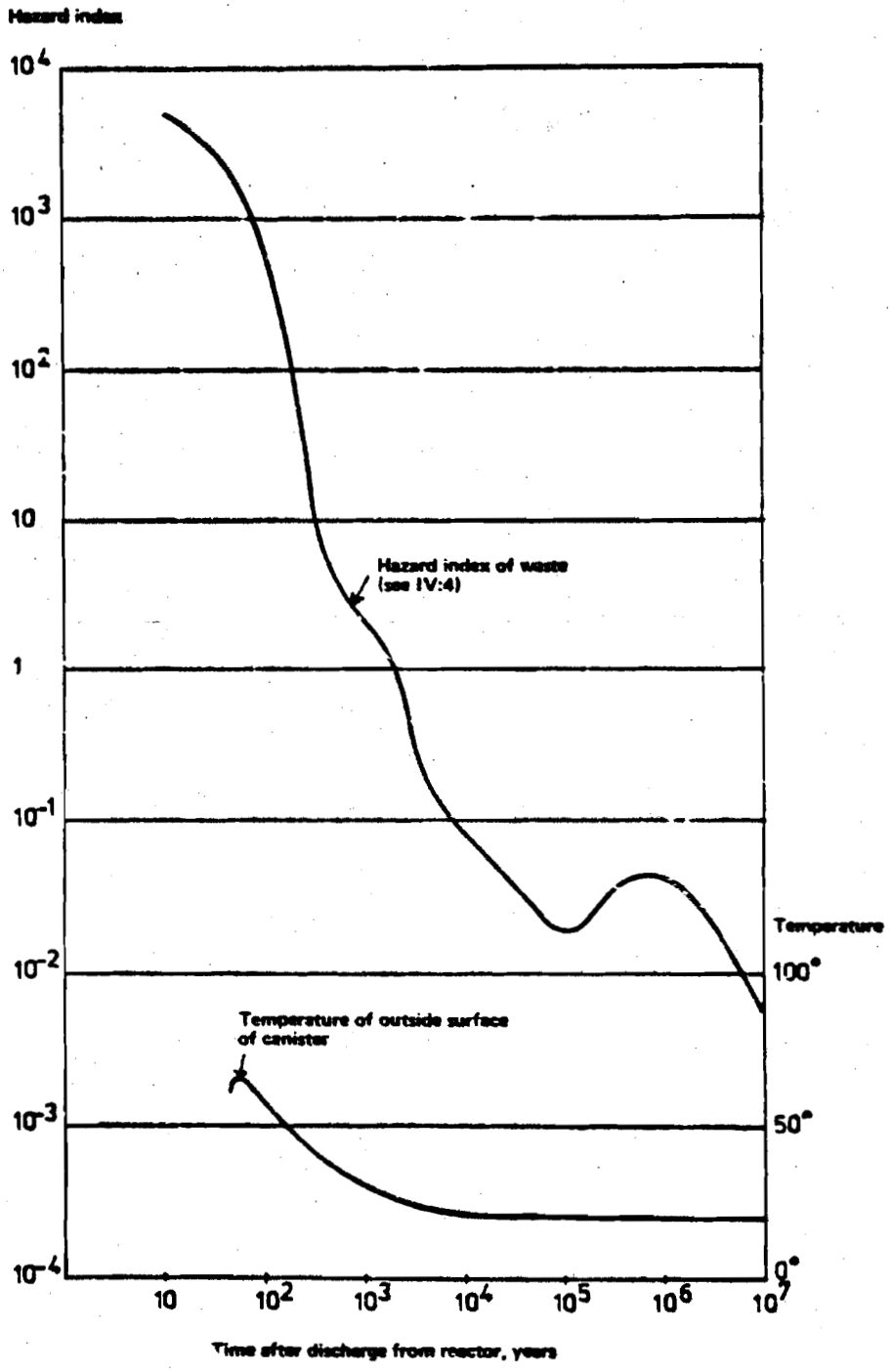


Figure 5-6. Graph showing how the toxicity of the vitrified waste (hazard index, see IV:3) and the external surface temperature of the canister vary with time. Note that the hazard index and time scales are logarithmic.

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

- general corrosion, i.e. 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 um/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-

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

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

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

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

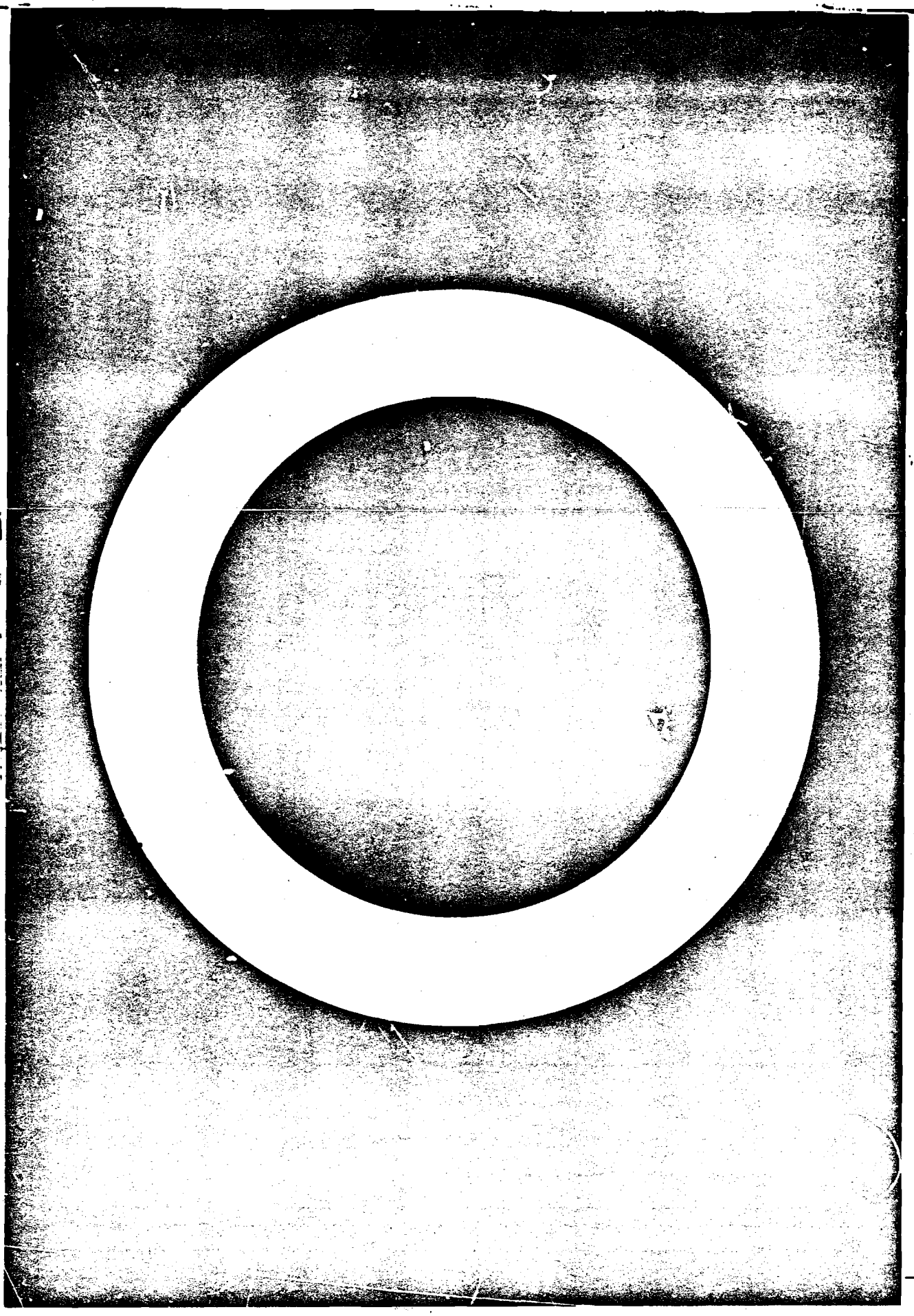
Some of these quality control procedures may take the form of random sample tests, the frequency of which shall be determined on the basis of the probability of defects.

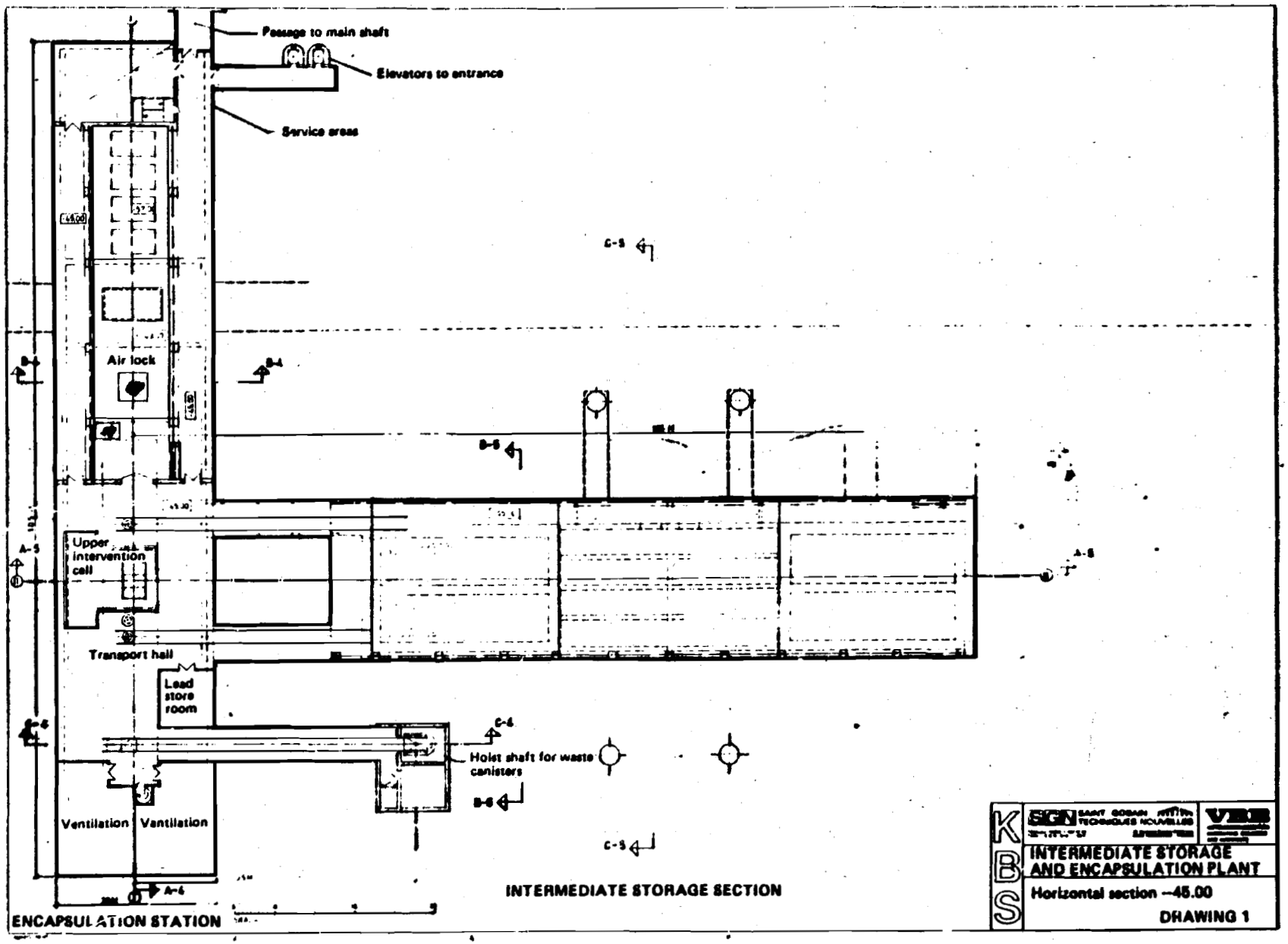
Quality control which is related to the glass body and the chromium-nickel cylinder will be performed at the foreign reprocessing plant. The extent to which quality control and quality assurance shall be carried out by the manufacturer, an independent quality control institution or the owner shall be 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.

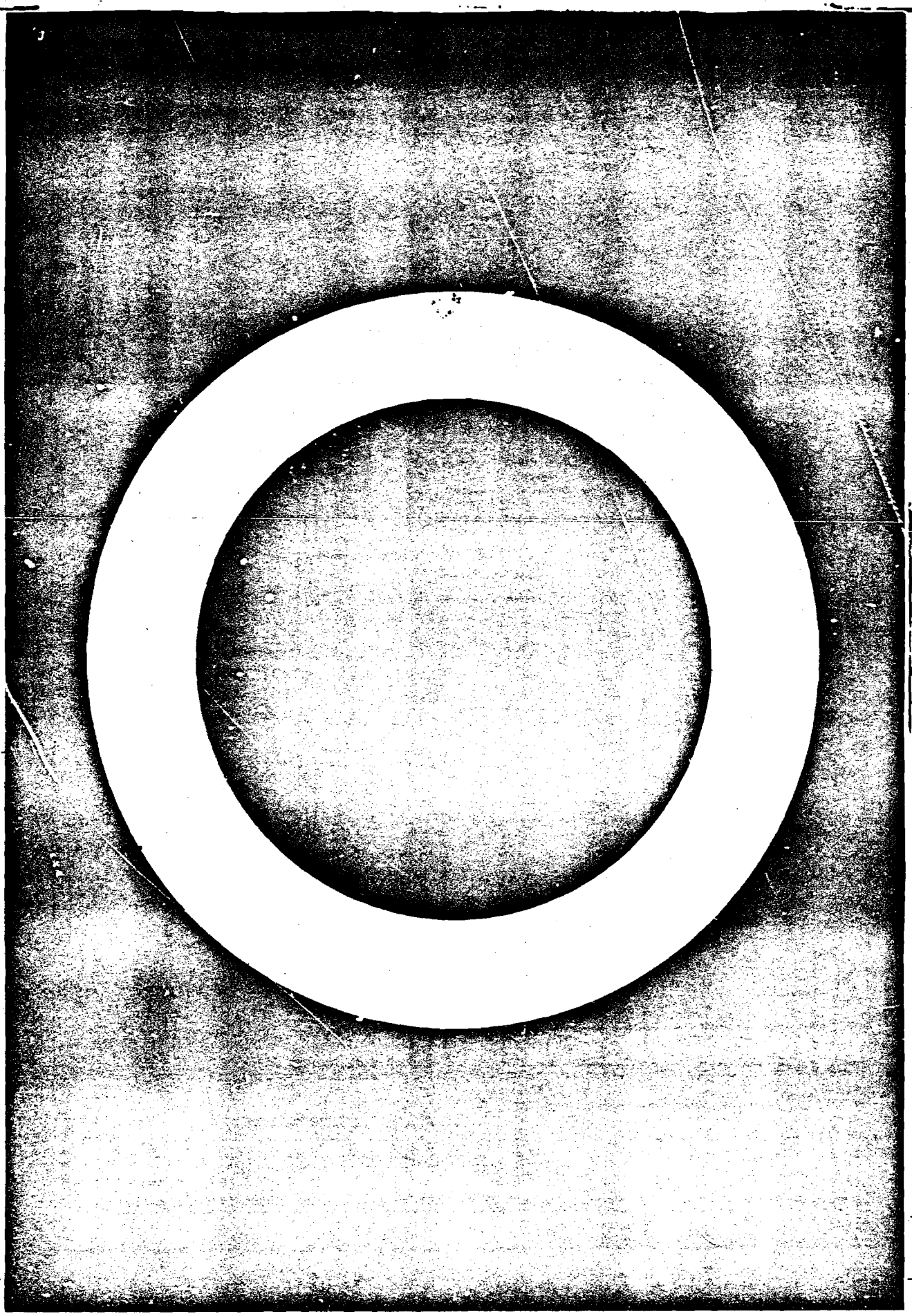
5.6 DECOMMISSIONING

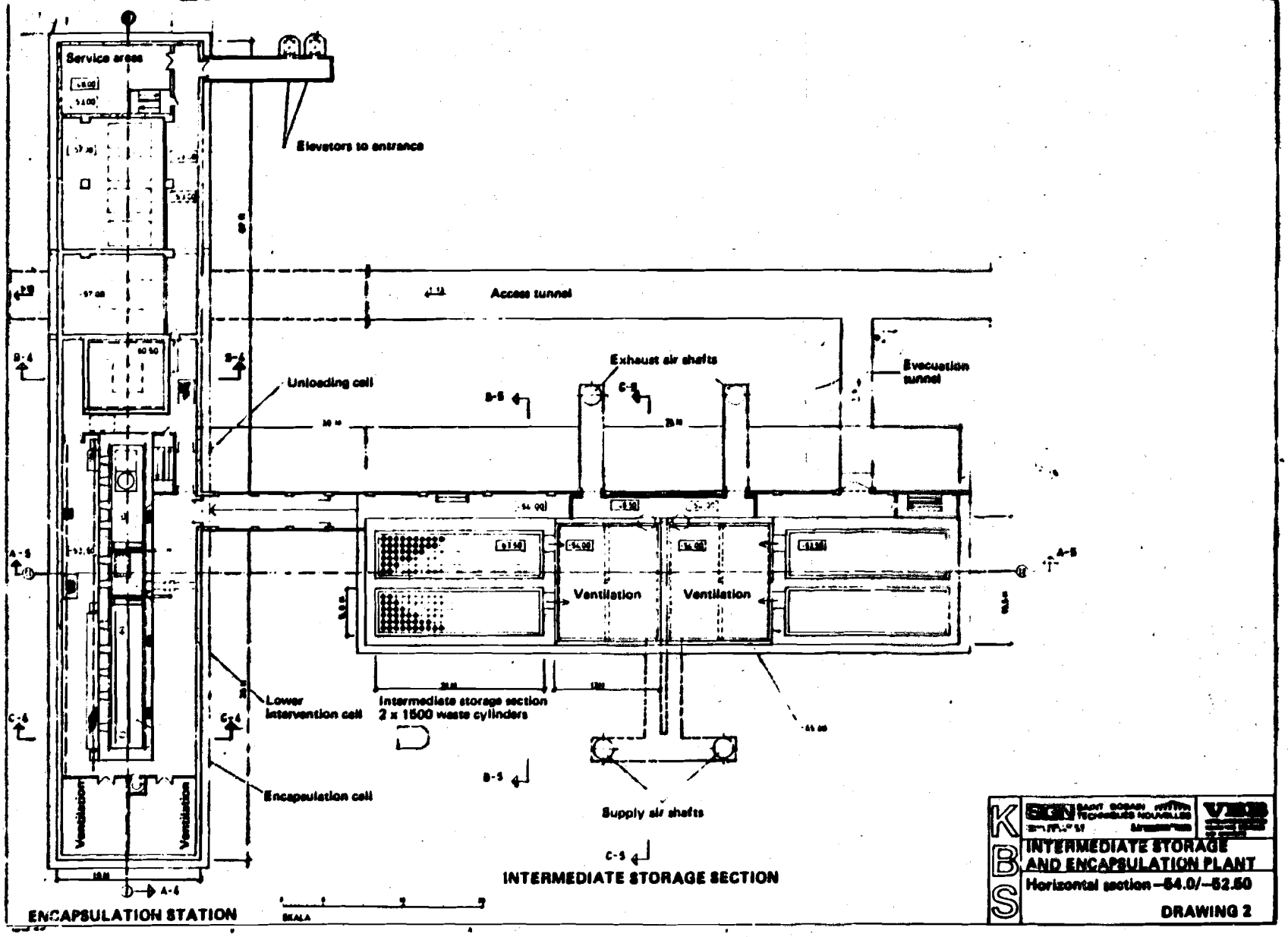
When the facility is no longer required and there are no waste cylinders or canisters left in it, the facility shall be decontaminated and all "own" radioactive waste, contaminated scrap and building materials shall be taken away to facilities which are equipped to receive and treat such materials. The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

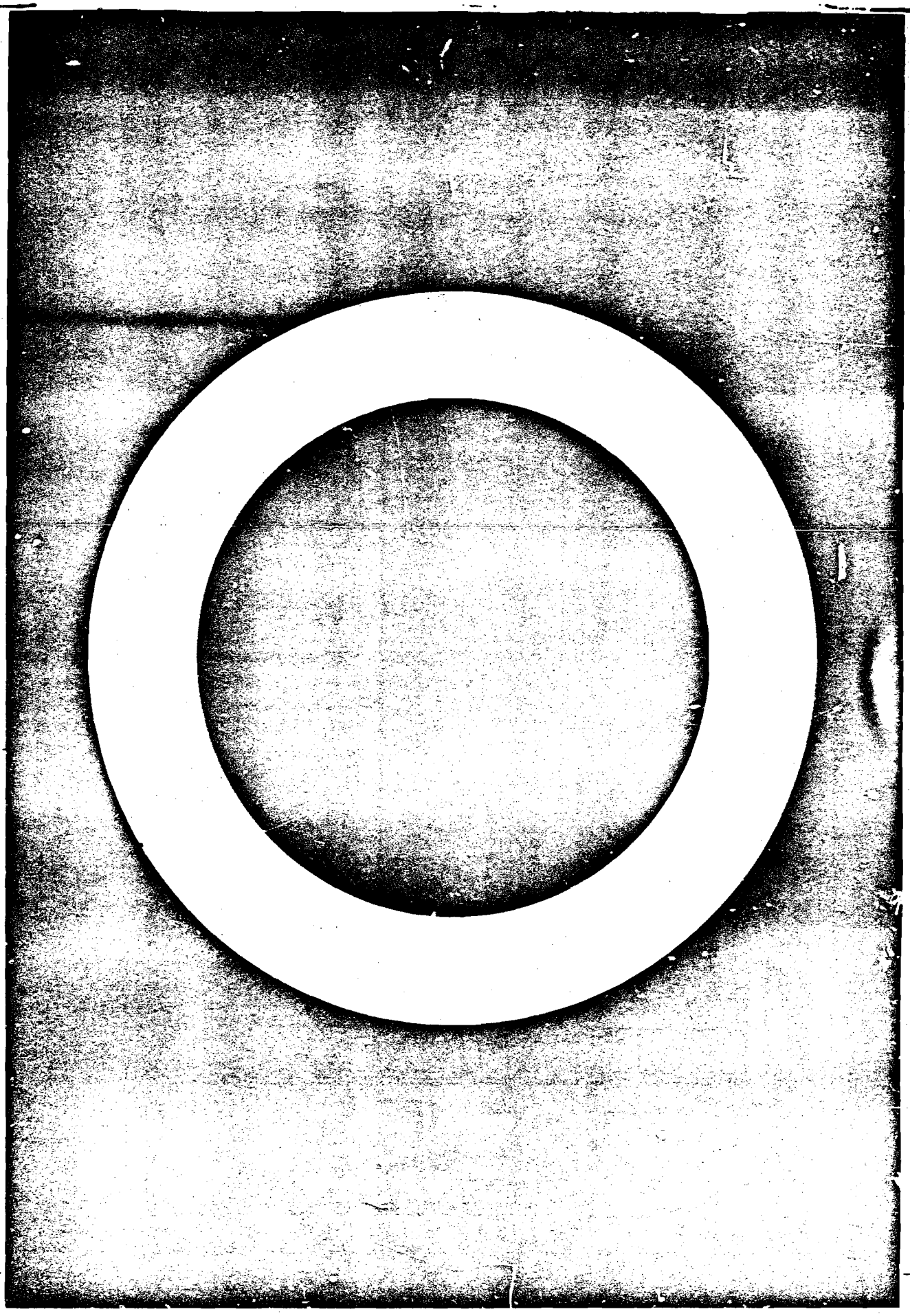
In general, very little contamination can be expected, so decommissioning should not present any difficult problems.

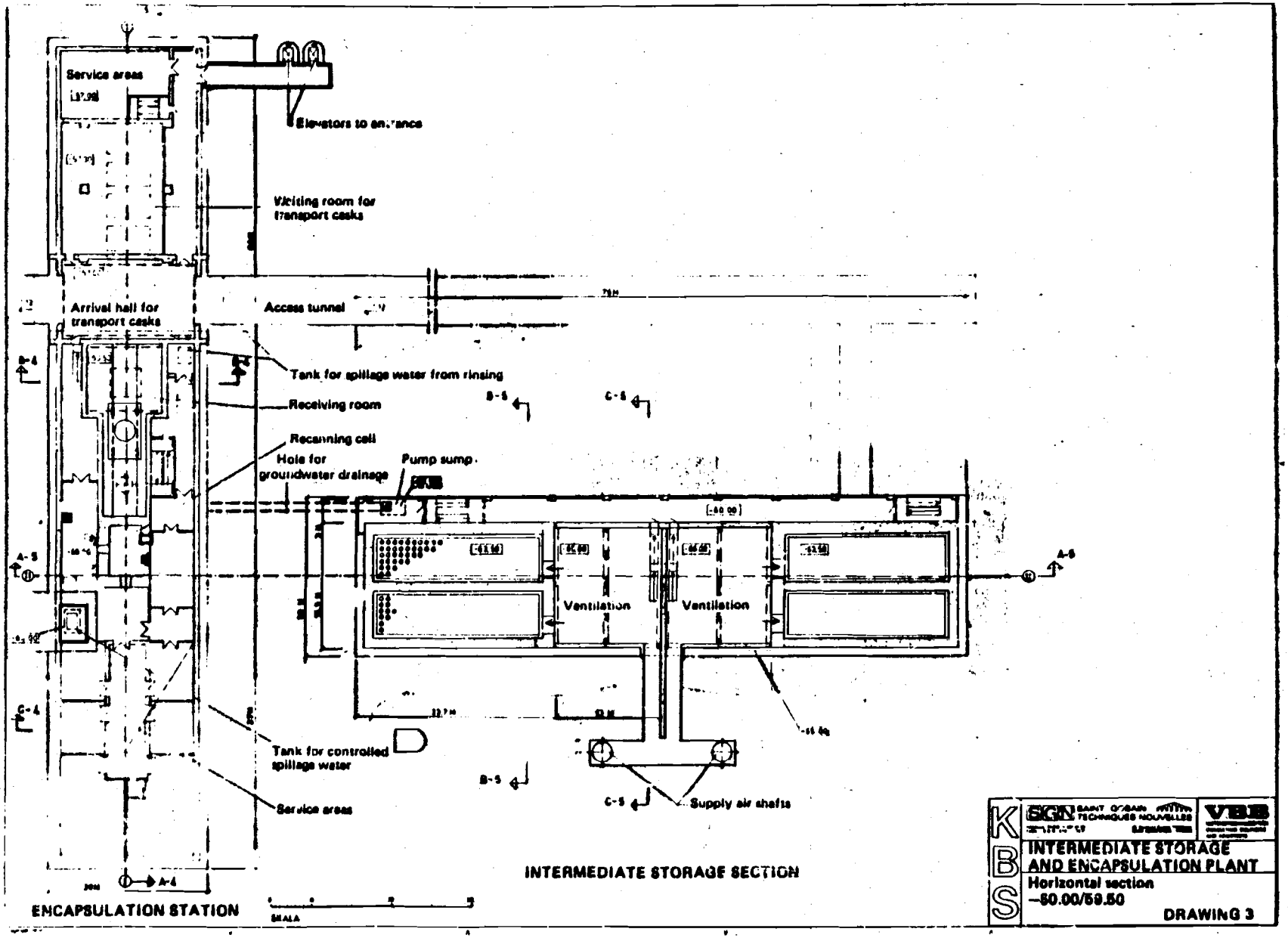


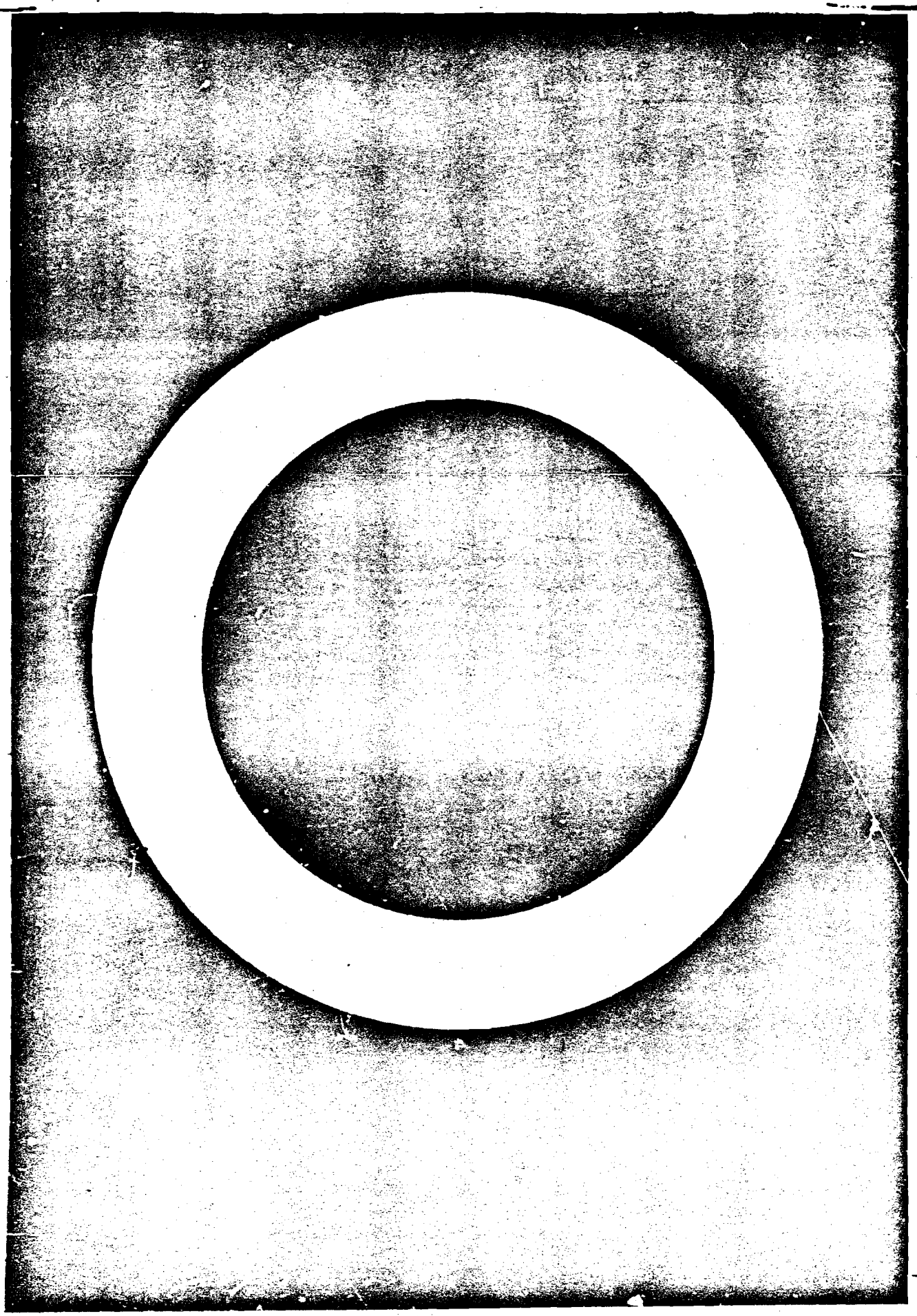

Passage to main shaft
Elevators to entrance
Service areas
Air lock
Upper intervention cell
Transport hall
Lead store room
Hoist shaft for waste canisters
Ventilation
Ventilation
ENCAPSULATION STATION
INTERMEDIATE STORAGE SECTION
KBS
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT
Horizontal section --45.00
DRAWING 1

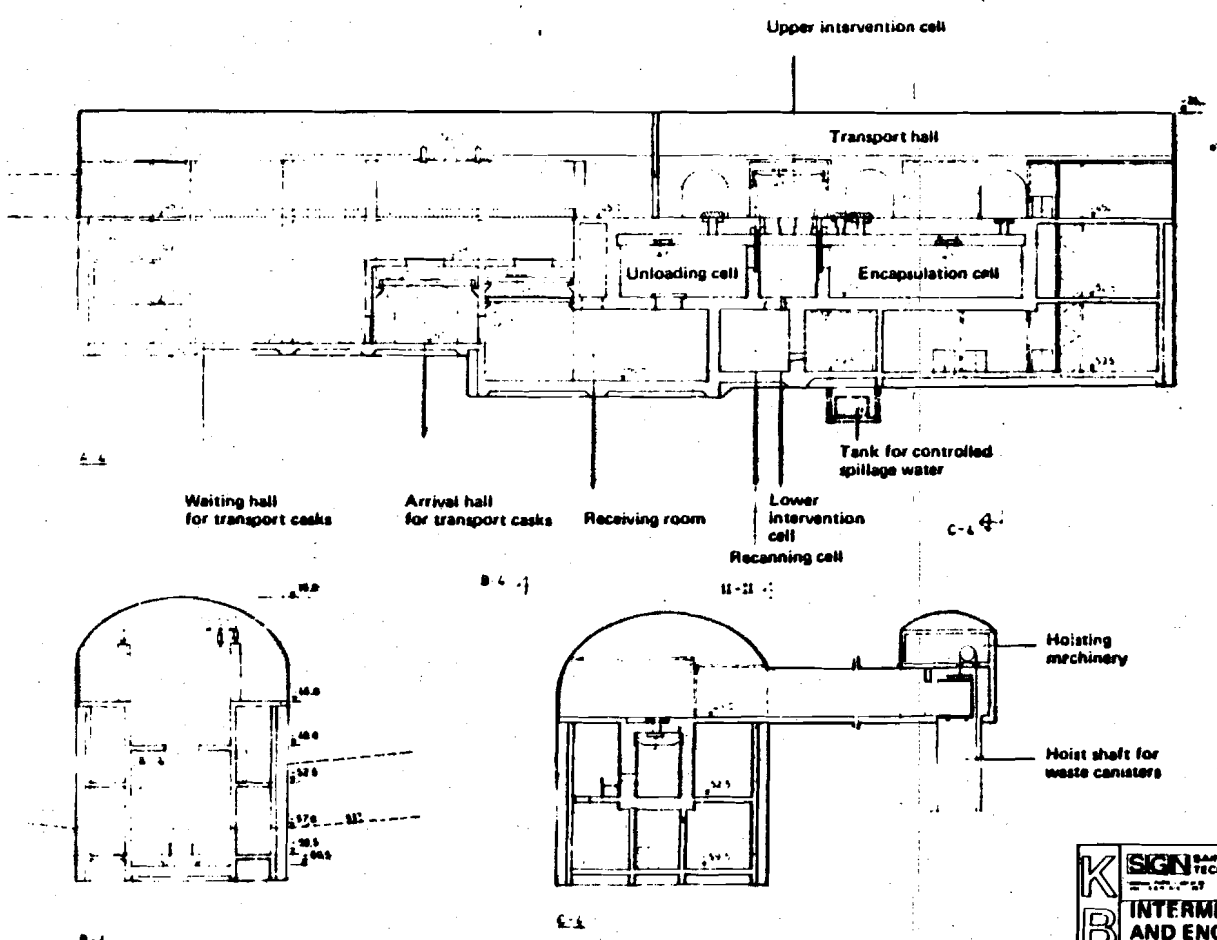



Service areas
Elevators to entrance
Access tunnel
Exhaust air shafts
Evacuation tunnel
Unloading cell
Ventilation
Ventilation
Lower intervention cell
Intermediate storage section
2 x 1500 waste cylinders
Encapsulation cell
Supply air shafts
Ventilation
Ventilation
ENCAPSULATION STATION
INTERMEDIATE STORAGE SECTION
SCALA
KBS
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
Tank for spillage water from rinsing
Receiving room
Recanning cell
Hole for groundwater drainage
Pump sump
Ventilation
Ventilation
Tank for controlled spillage water
Service areas
Supply air shafts
ENCAPSULATION STATION
INTERMEDIATE STORAGE SECTION
KBS
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT
Horizontal section
-60.00/59.50
DRAWING 3





KBS

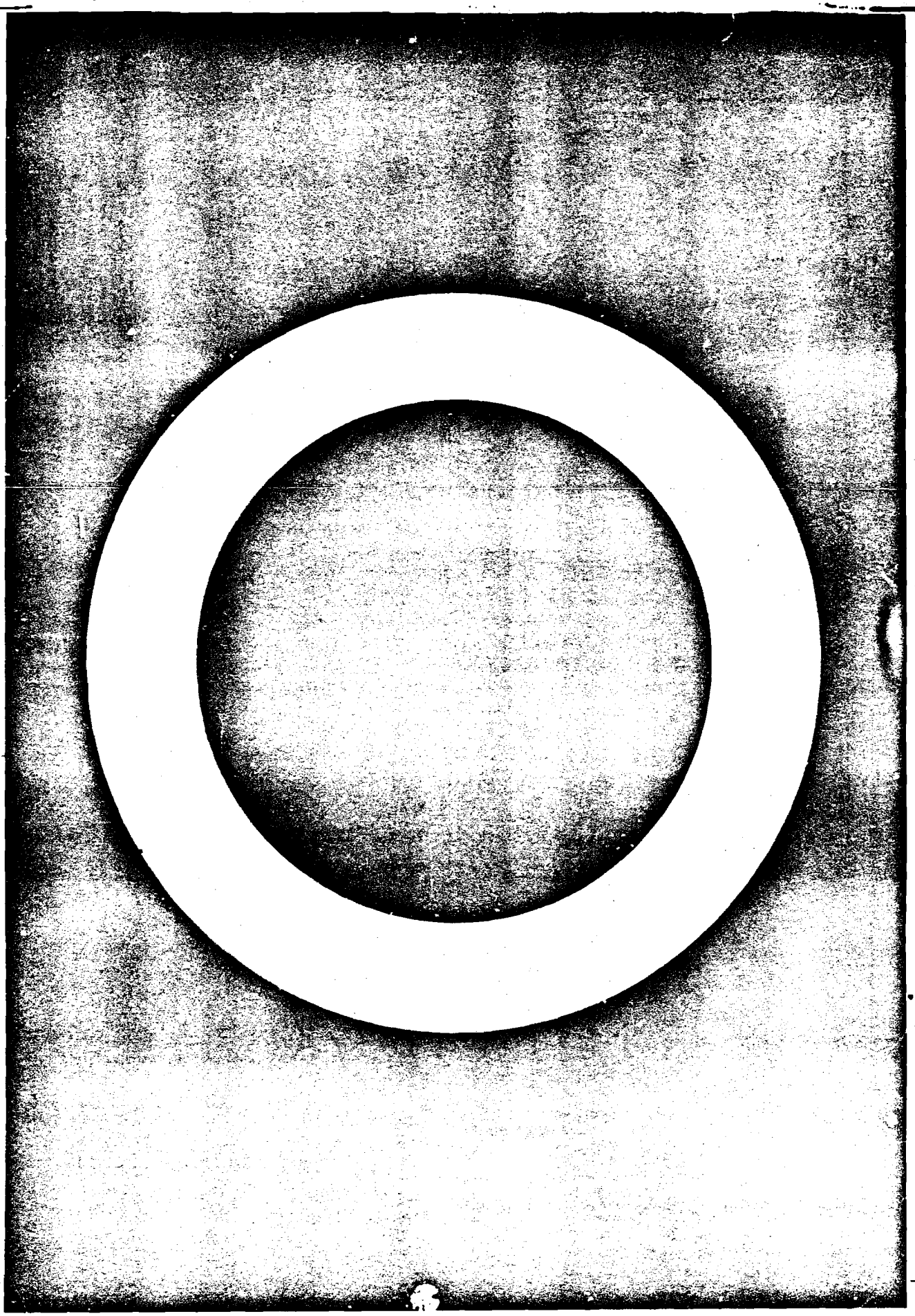
SGN SAINT GOBAIN TECHNIQUES NOUVELLES

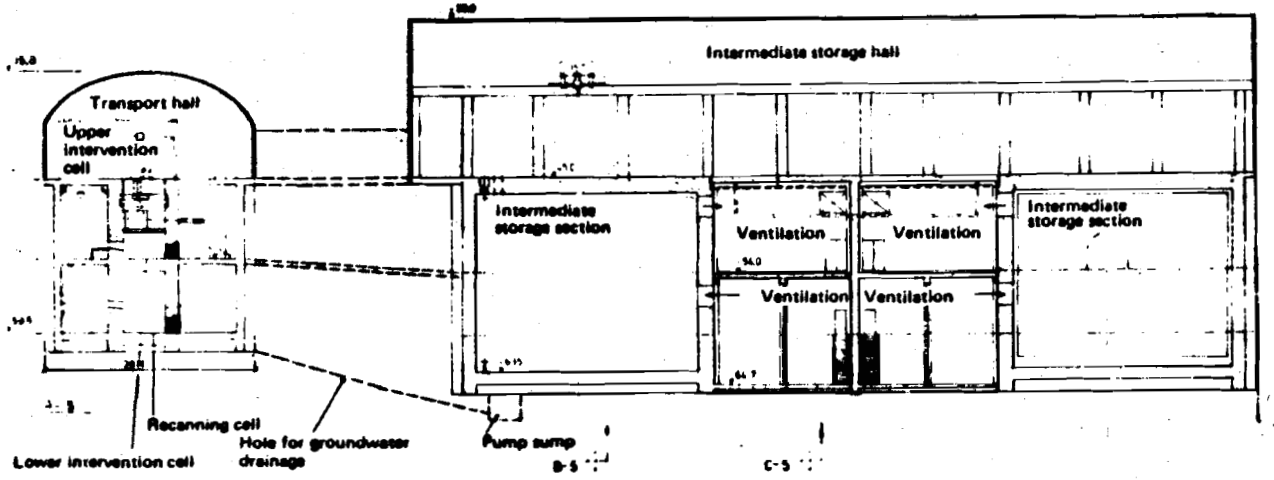
VBR

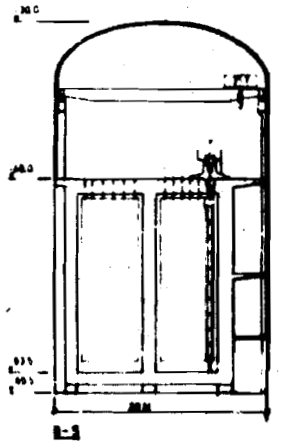
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT

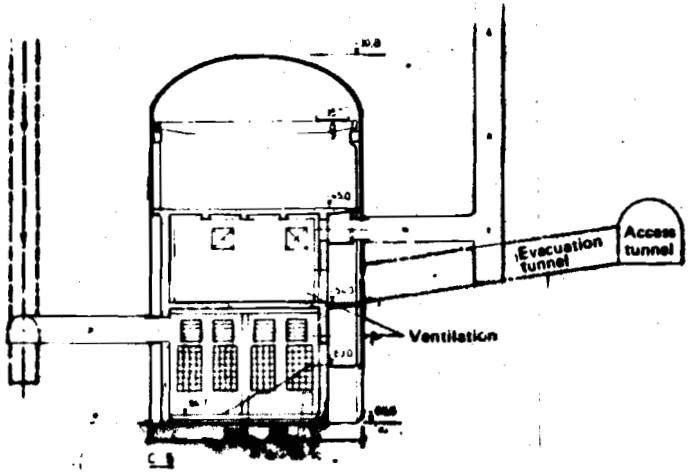
Longitudinal section
I–I and cross-sections

DRAWING 4



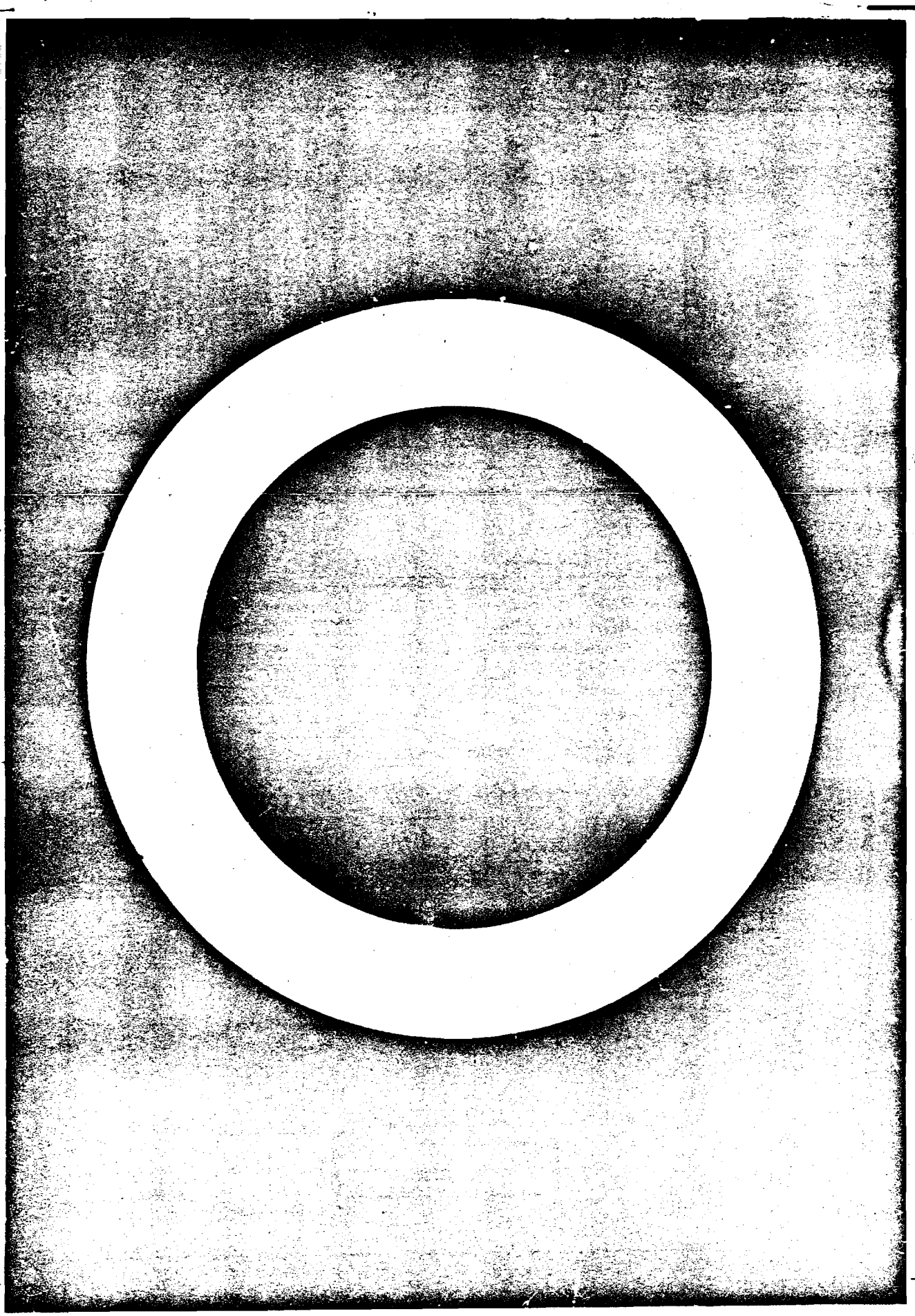


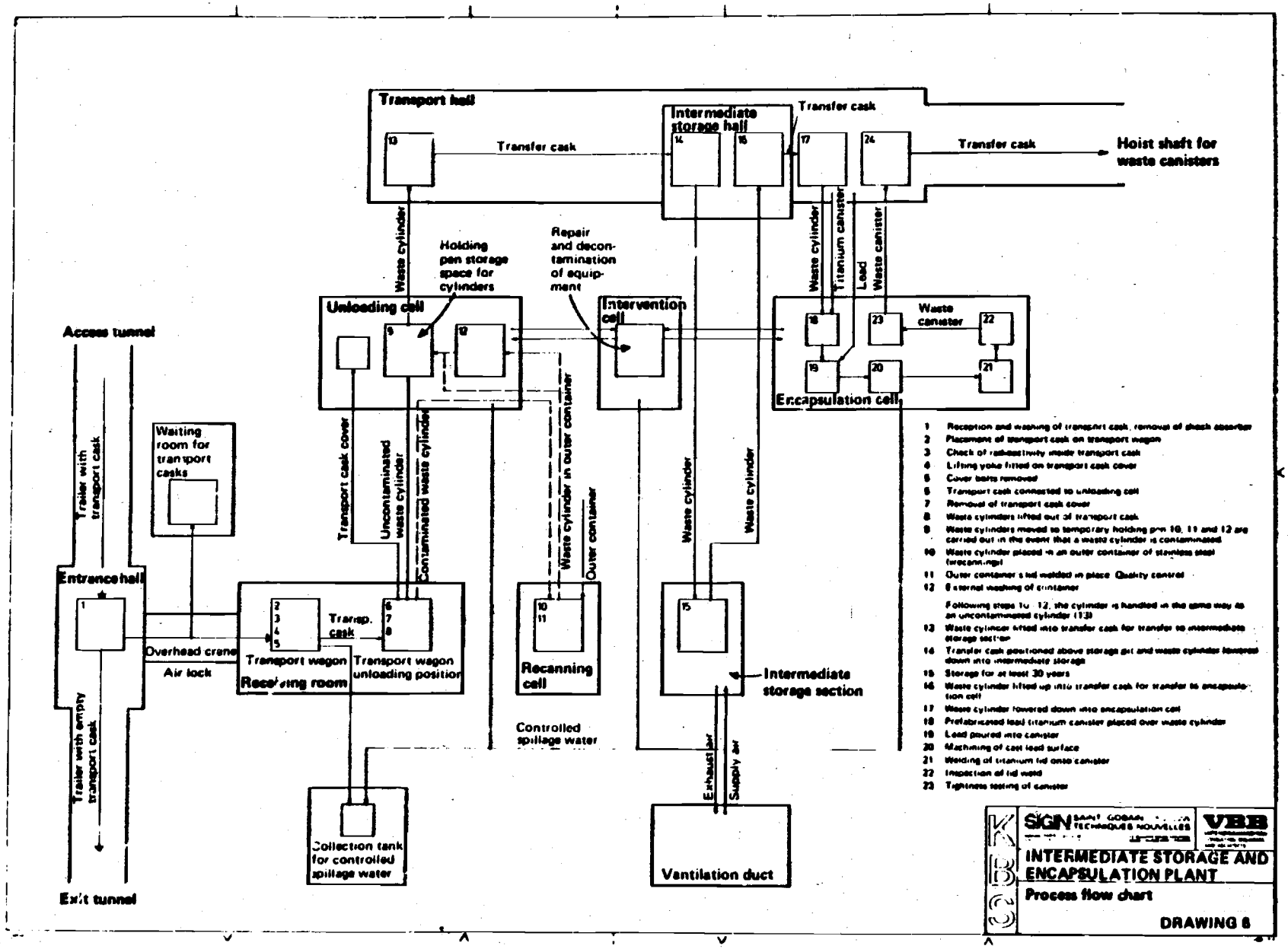


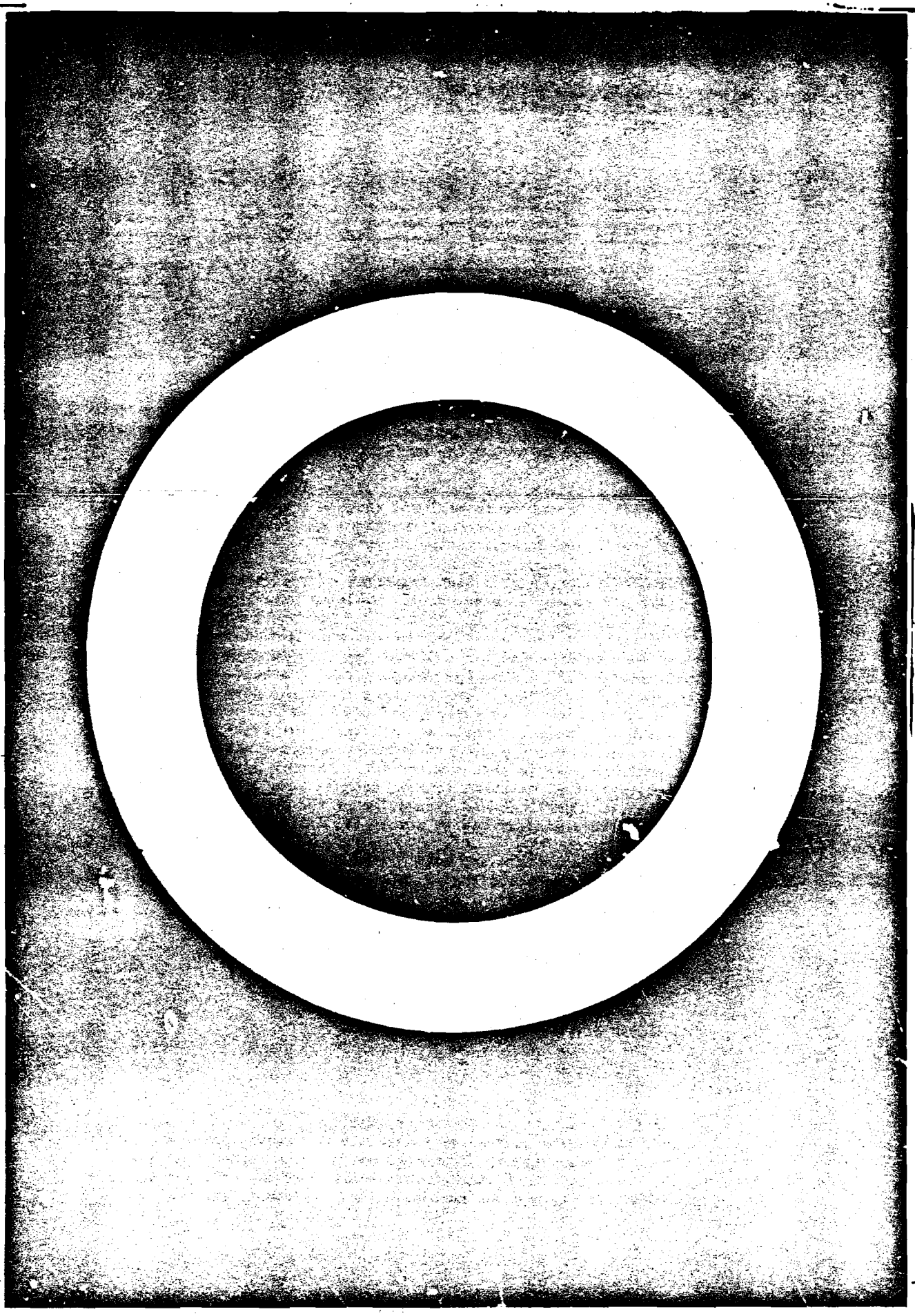


KBS	SGN SAINT GOBAIN TECHNIQUES NOUVELLES	VBB

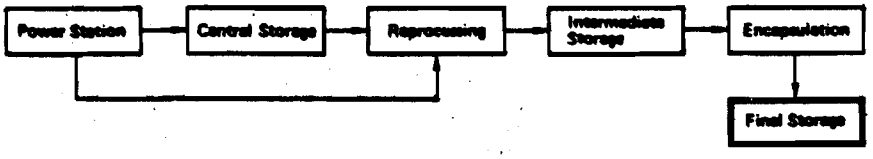
INTERMEDIATE STORAGE AND ENCAPSULATION PLANT
Longitudinal section II–II and cross-sections
DRAWING 5




Transport hall
Intermediate storage hall
Transfer cask
Transfer cask
Transfer cask
Hoist shaft for waste canisters
Waste cylinder
Holding pen storage space for cylinders
Repair and decontamination of equipment
Unloading cell
Intervention cell
Waste cylinder
Titanium canister
Lead
Waste canister
Waste canister
Encapsulation cell
Access tunnel
Waiting room for transport casks
Trailer with transport cask
Transport cask cover
Uncontaminated waste cylinder
Contaminated waste cylinder
Waste cylinder in outer container
Outer container
Waste cylinder
Waste cylinder
Entrance hall
Overhead crane
Air lock
Transport wagon
Receiving room
Transp. cask
Transport wagon unloading position
Recanning cell
Intermediate storage section
Controlled spillage water
Trailer with empty transport cask
Exhaust air
Supply air
Collection tank for controlled spillage water
Ventilation duct
Exit tunnel
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 (precanning)
11 Outer container is 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
SIGN SAINT GOBAIN TECHNIQUES NOUVELLES
VBB
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-

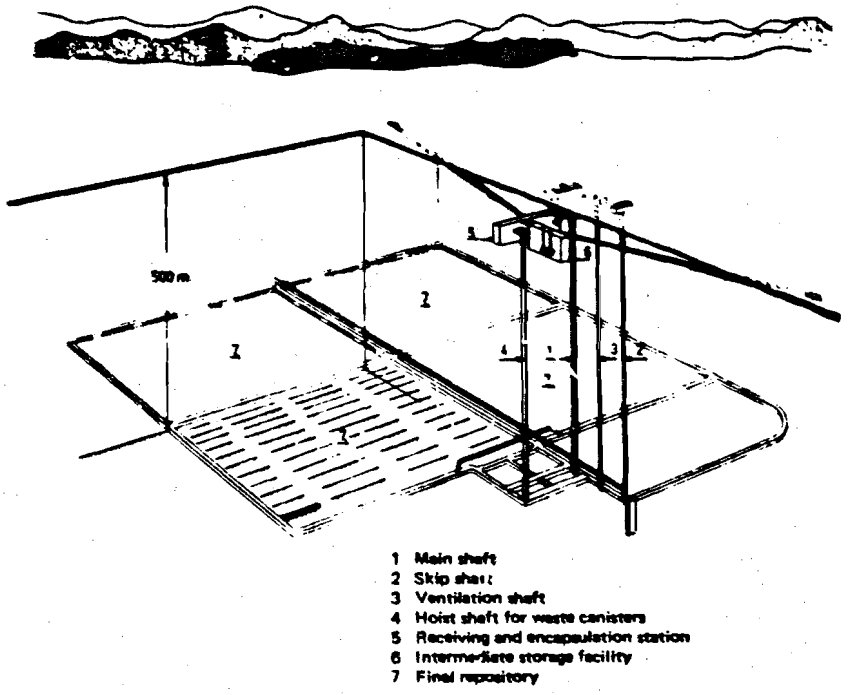


Figure 6-1. Perspective drawing of final repository with plant for intermediate storage and encapsulation. The final repository consists of a system of parallel storage tunnels situated 500 m below the surface.

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/.

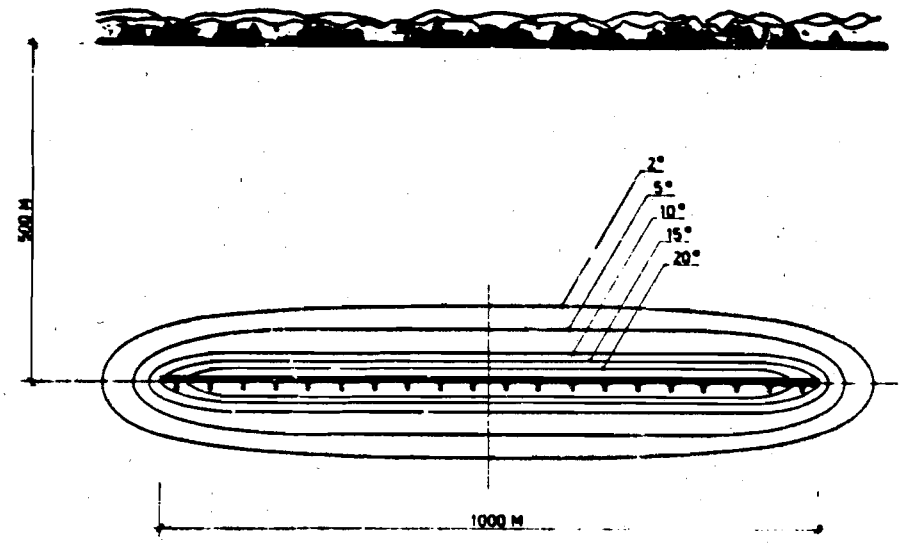
See I:14 for the schedule for the construction of the facility.

6.2.3 Deposition of waste canisters

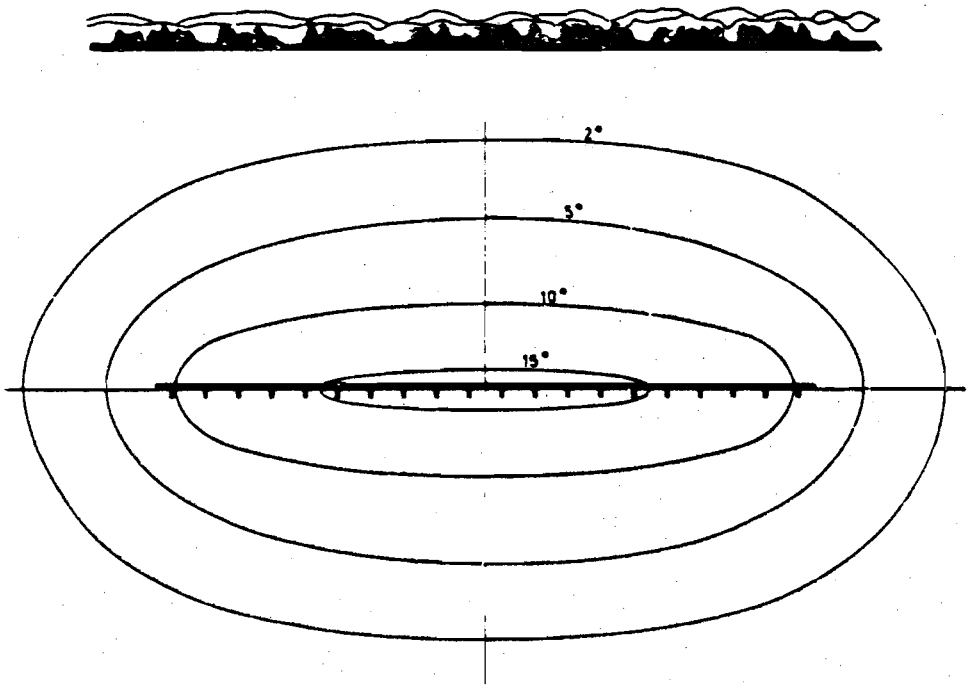
When a waste canister is to be transferred from the facility for intermediate storage and encapsulation to the final repository, it is first lifted out of the encapsulation cell into a radiation-shielded transfer cask by means of a procedure similar to that used for handling of the waste cylinders in 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.

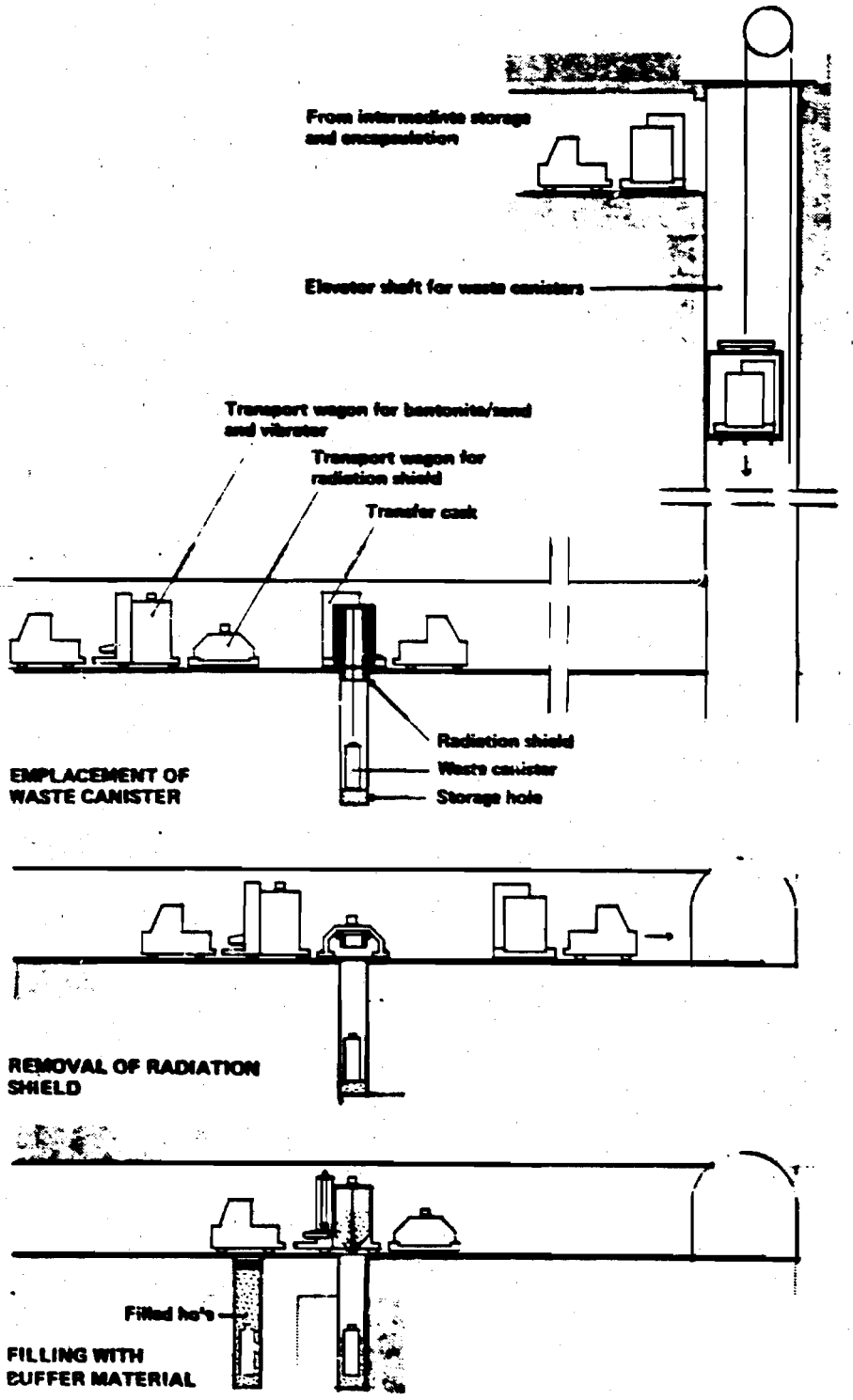


Figure 6-3. Handling of waste canisters in final repository.

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

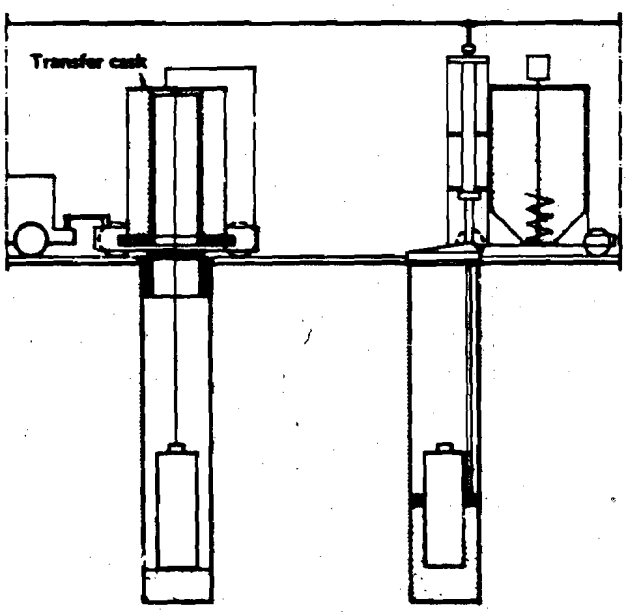


Figure 6–4. Deposition of canister (at left) and filling of deposition hole (at right). The filler material consists of a mixture of quartz sand and bentonite.

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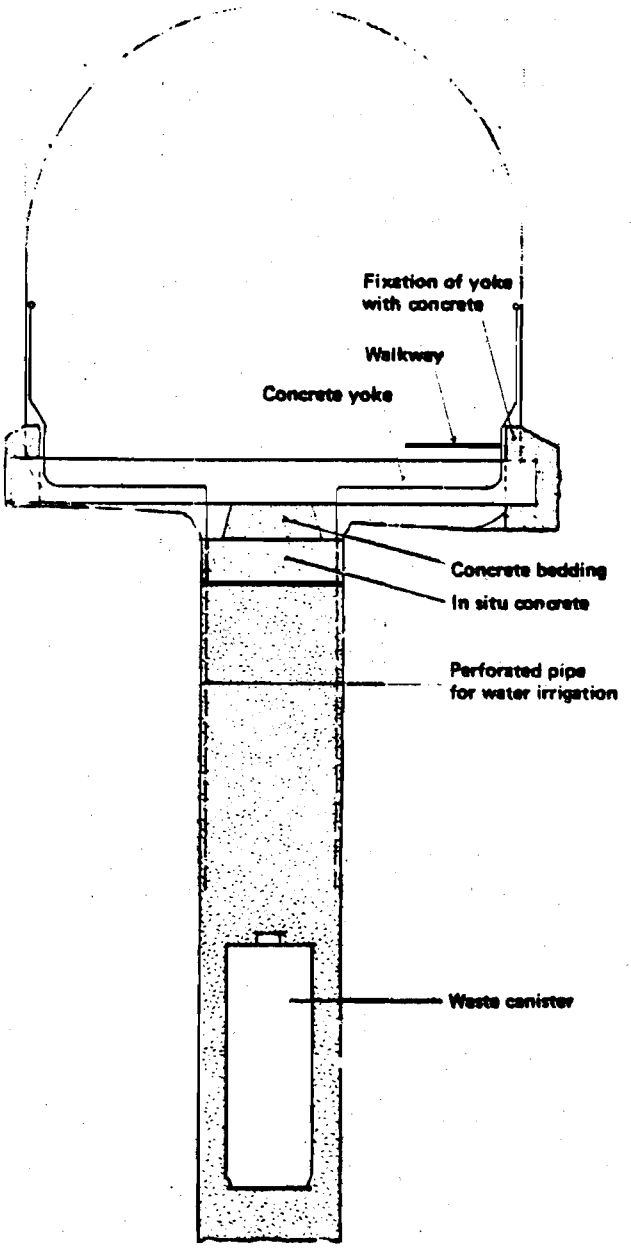


Figure 6.5. After backfilling with a mixture of quartz sand and bentonite, the deposition hole is sealed with a lid of cast-in-situ concrete. A concrete yoke is anchored above the lid.

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, compressed 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.

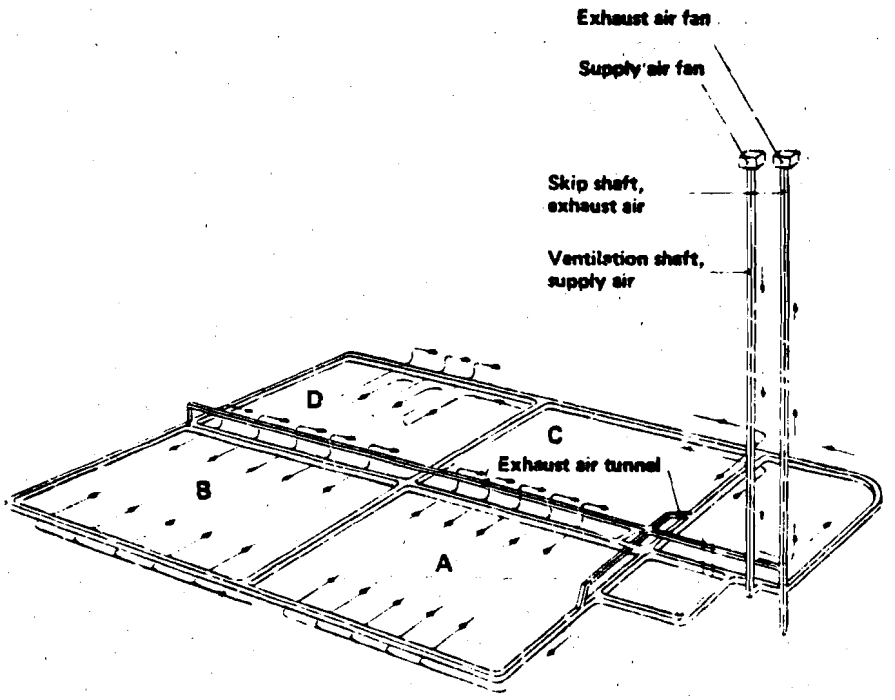


Figure 6-6. Perspective drawing of ventilation system in final repository. Deposition has been concluded within area A and is in progress in area B. Blasting has been concluded within area C and is in progress within area D.

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.

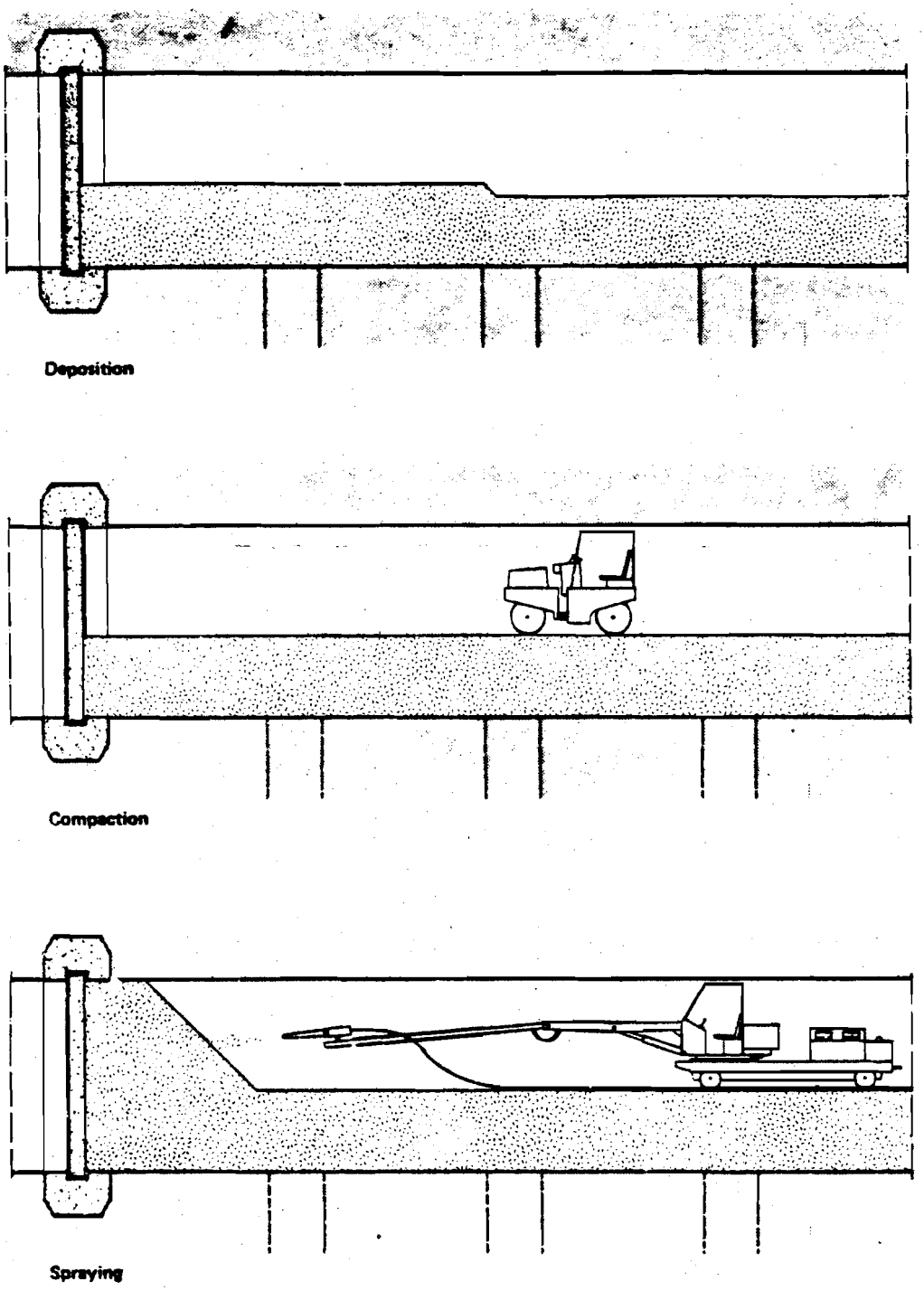


Figure 6-7. When the final repository is sealed, the tunnels are filled with a mixture of quartz sand and bentonite. The lower part is placed by tractors and vibrorolled. The upper part of the tunnel is filled by spraying.

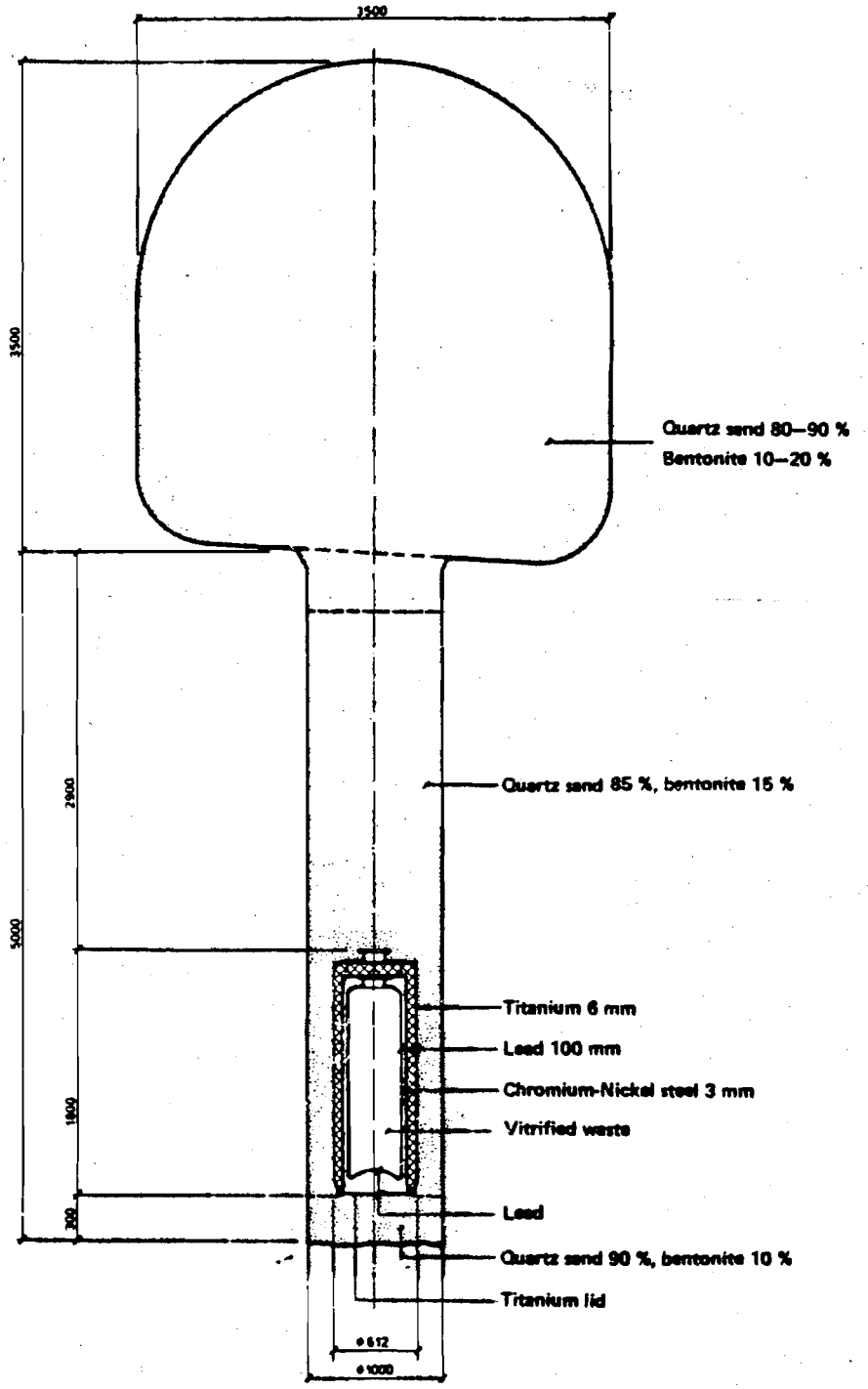
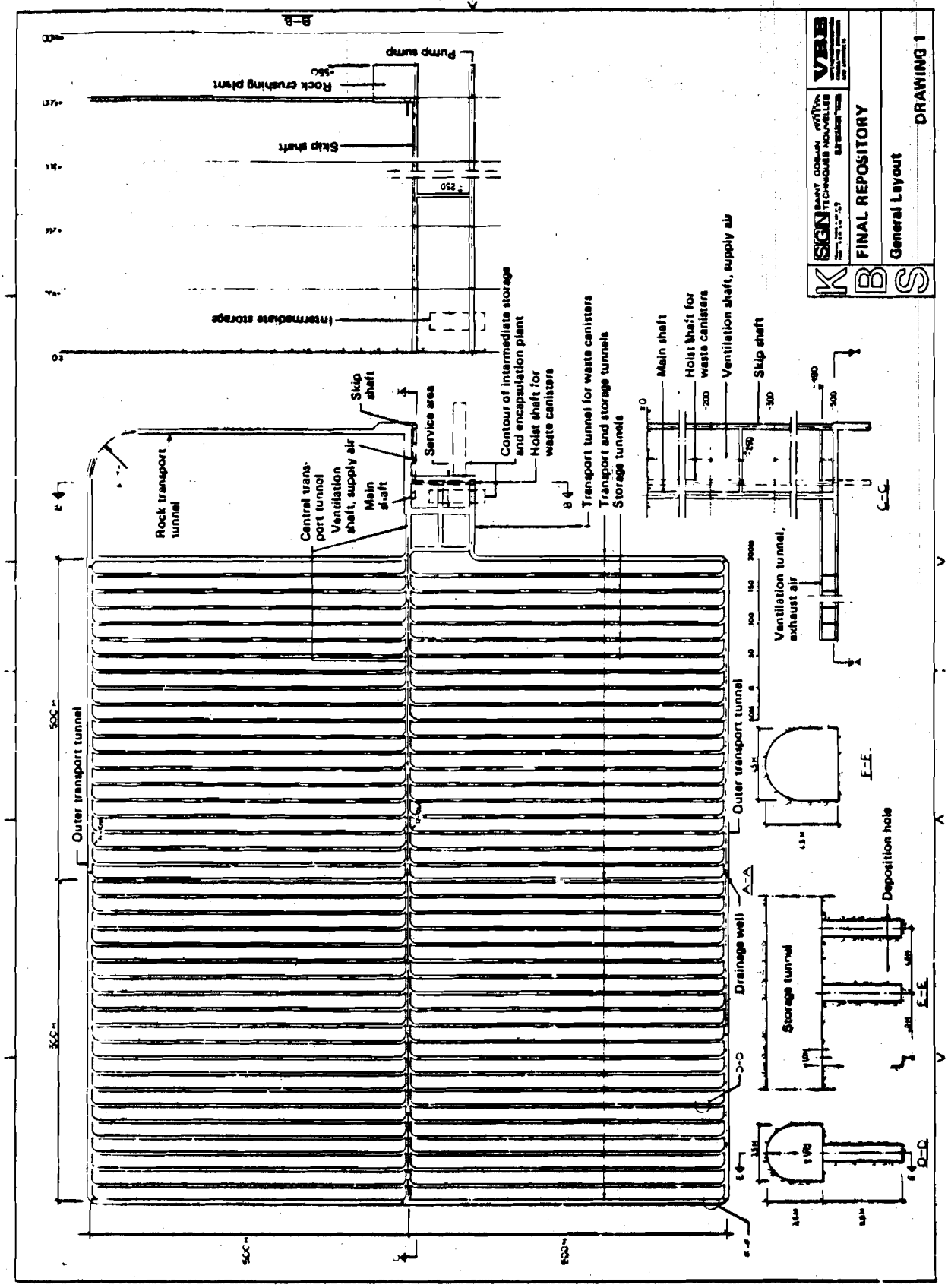
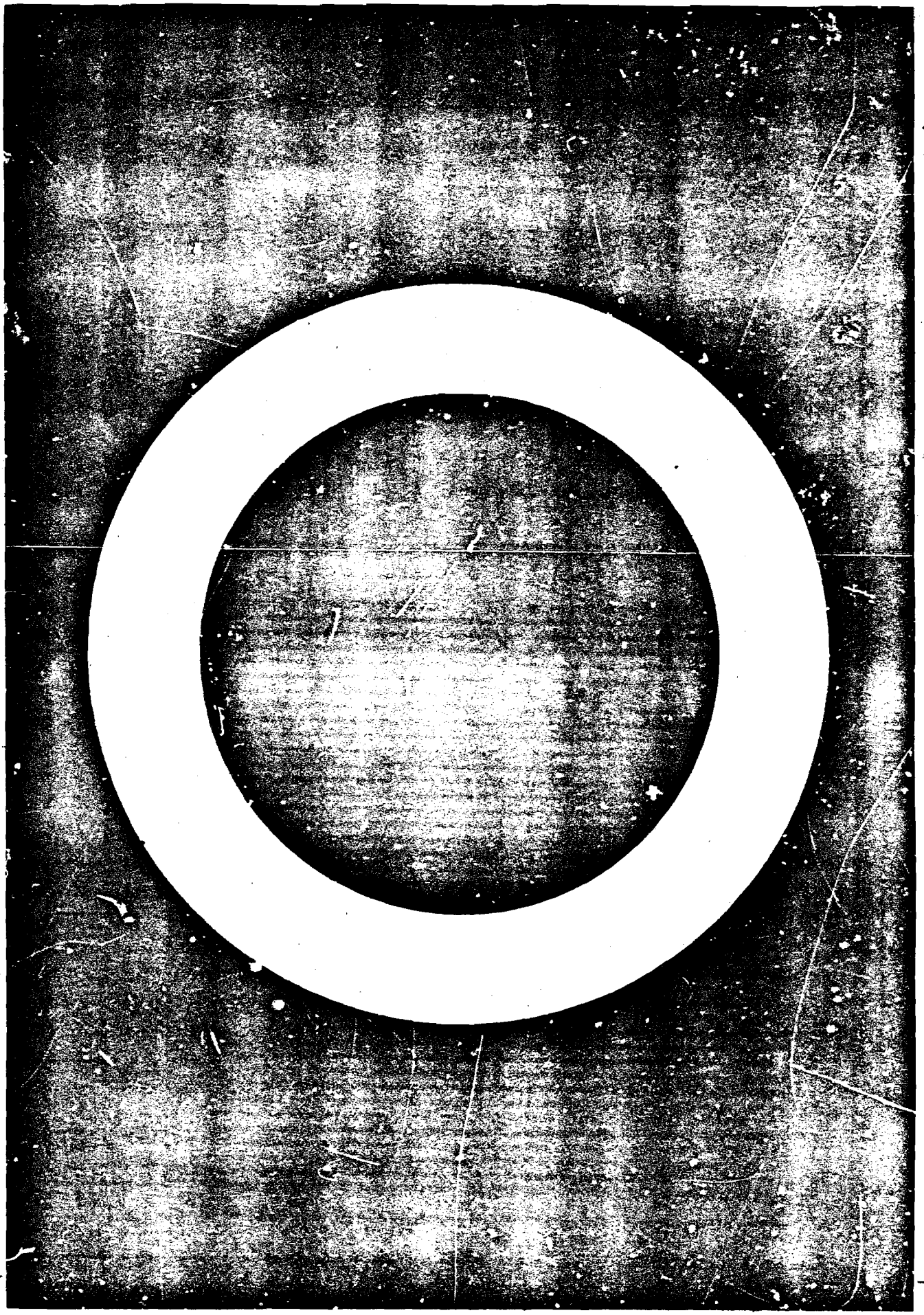


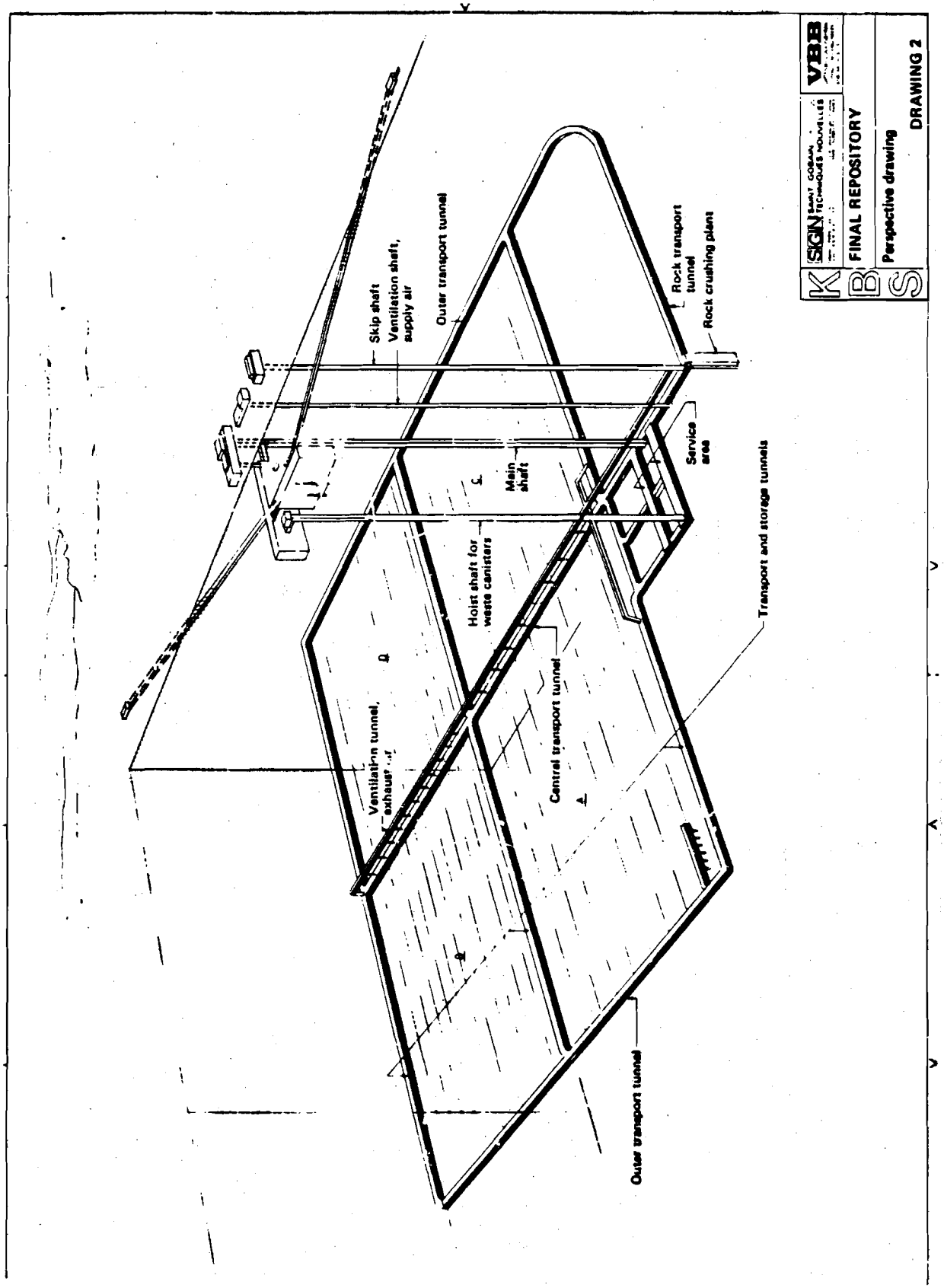
Figure 6-8. Sealed final repository.

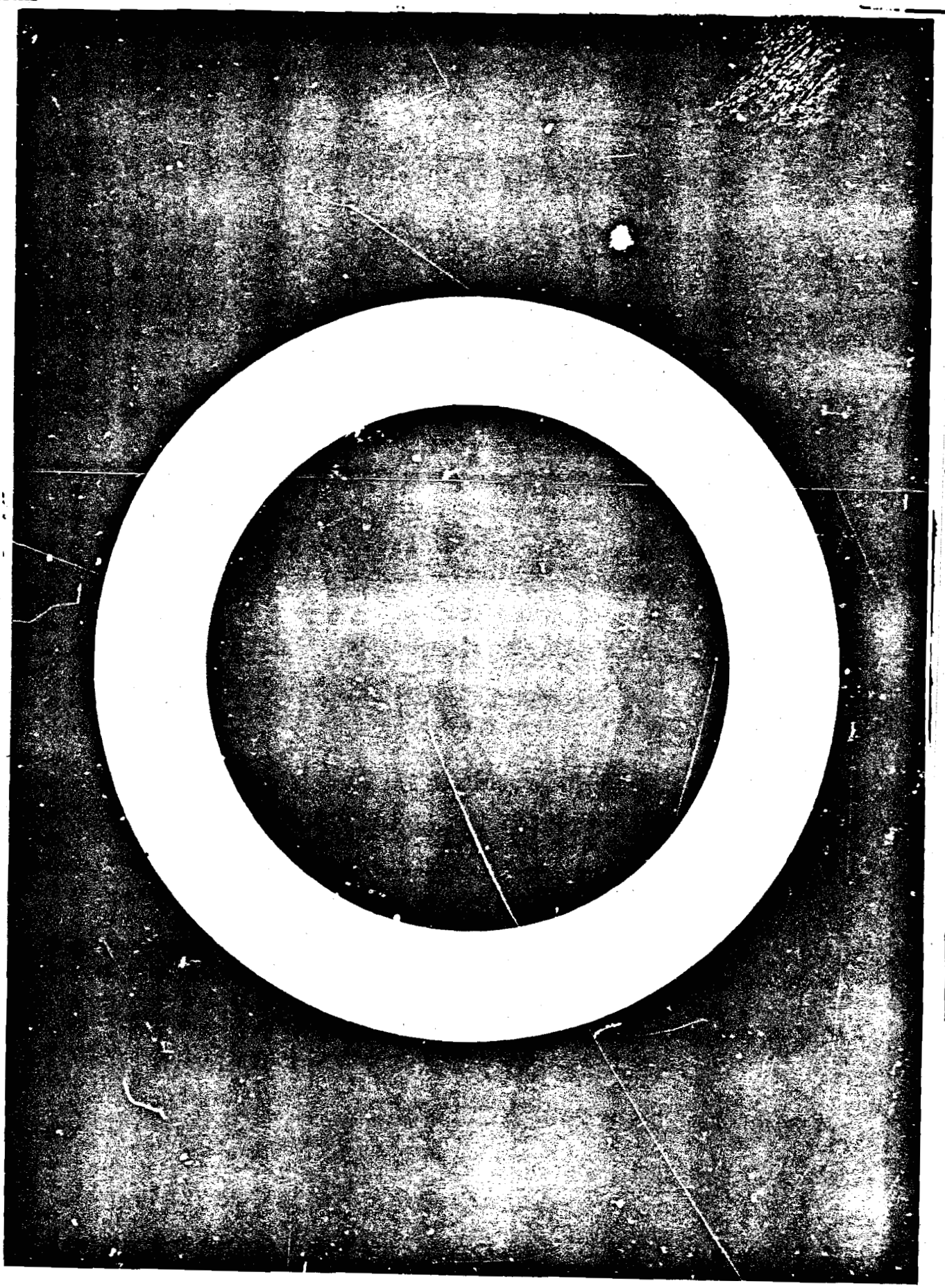
6.7 DRAWINGS

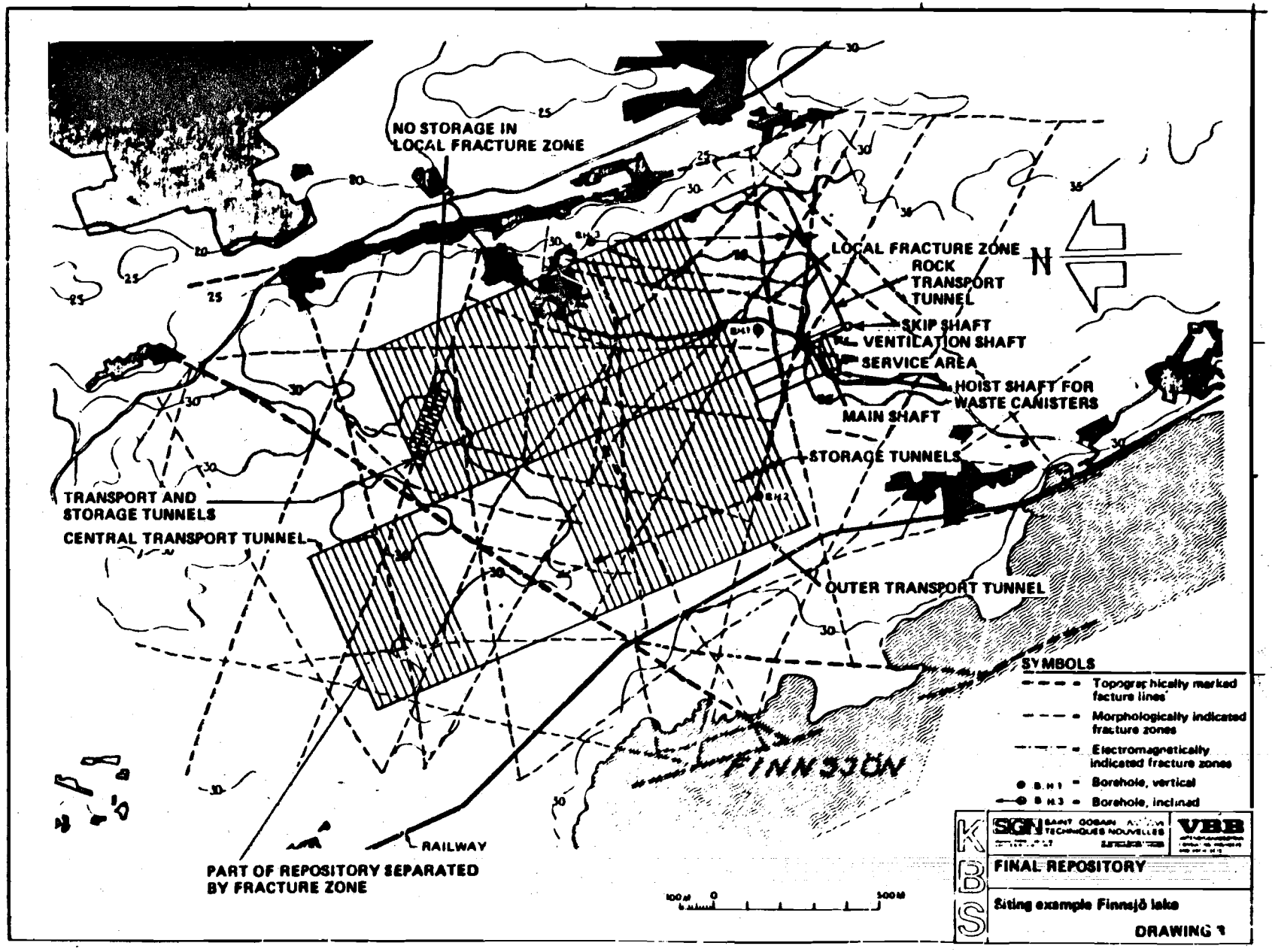
DRAWING	1	GENERAL LAYOUT
"	2	PERSPECTIVE DRAWING
"	3	SITING EXAMPLE
"	4	CONSTRUCTION STAGES
"	5	TRANSPORT ROUTES
"	6	VENTILATION
"	7	STORAGE TUNNELS
"	8	SEALED REPOSITORY
"	9	TRANSPORT OF WASTE CANISTER FROM INTERMEDIATE STORAGE TO FINAL REPOSITORY
"	10	TRANSPORT AND DEPOSITION OF WASTE CANISTER IN FINAL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

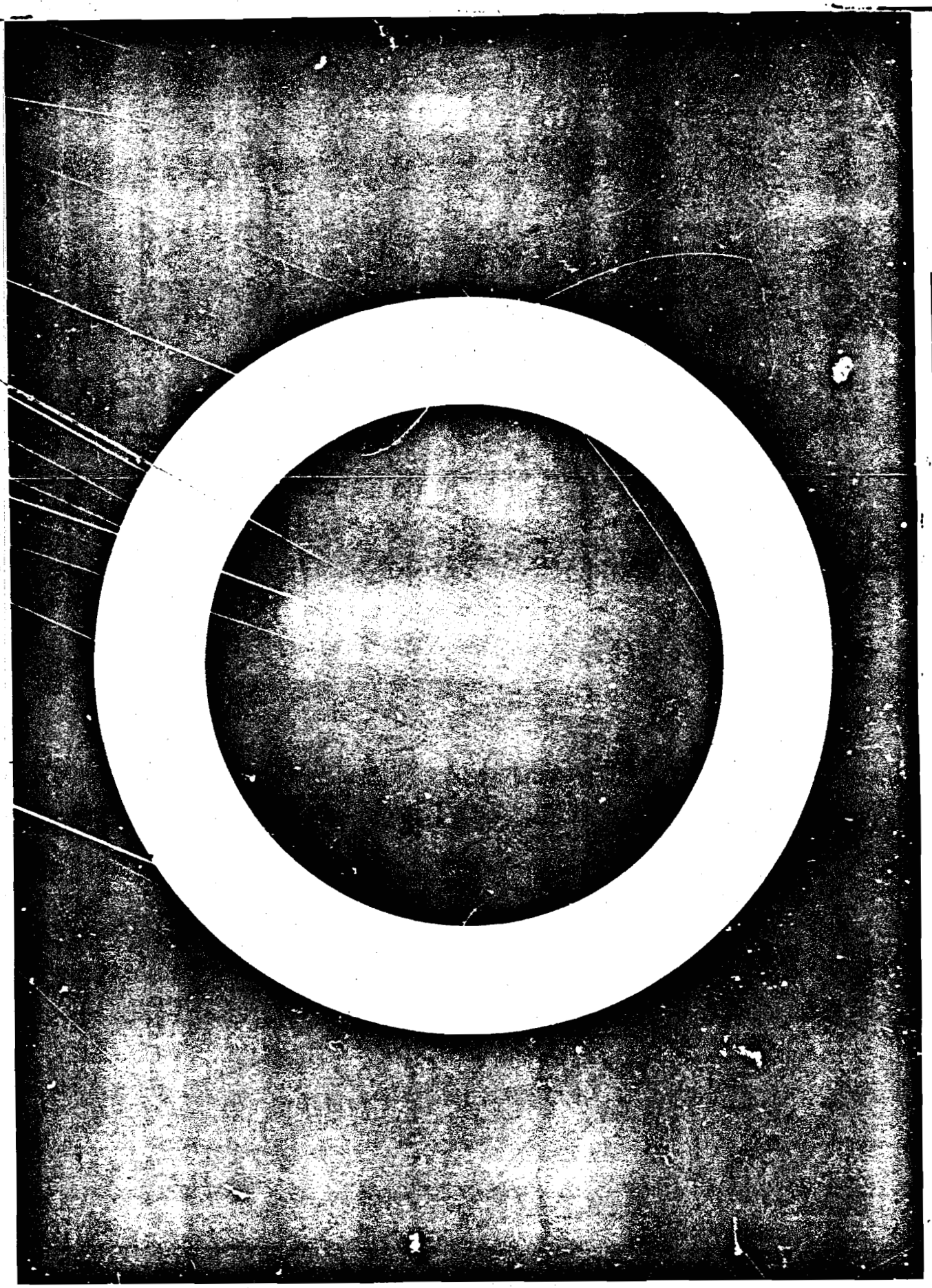

KBS
SKN
VBB
FINAL REPOSITORY
General Layout
DRAWING 1
Pump sump
Rock crushing plant
Skip shaft
Intermediate storage
Skip shaft
Service area
Contour of intermediate storage and encapsulation plant
Hoist shaft for waste canisters
Transport tunnel for waste canisters
Transport and storage tunnels
Storage tunnels
Rock transport tunnel
Central transport tunnel
Ventilation shaft, supply air
Main shaft
Main shaft
Hoist shaft for waste canisters
Ventilation shaft, supply air
Skip shaft
Ventilation tunnel, exhaust air
Outer transport tunnel
Outer transport tunnel
Drainage well
Storage tunnel
Deposition hole
B-B
C-C
E-E
E-E
A-A
D-D
Q-Q

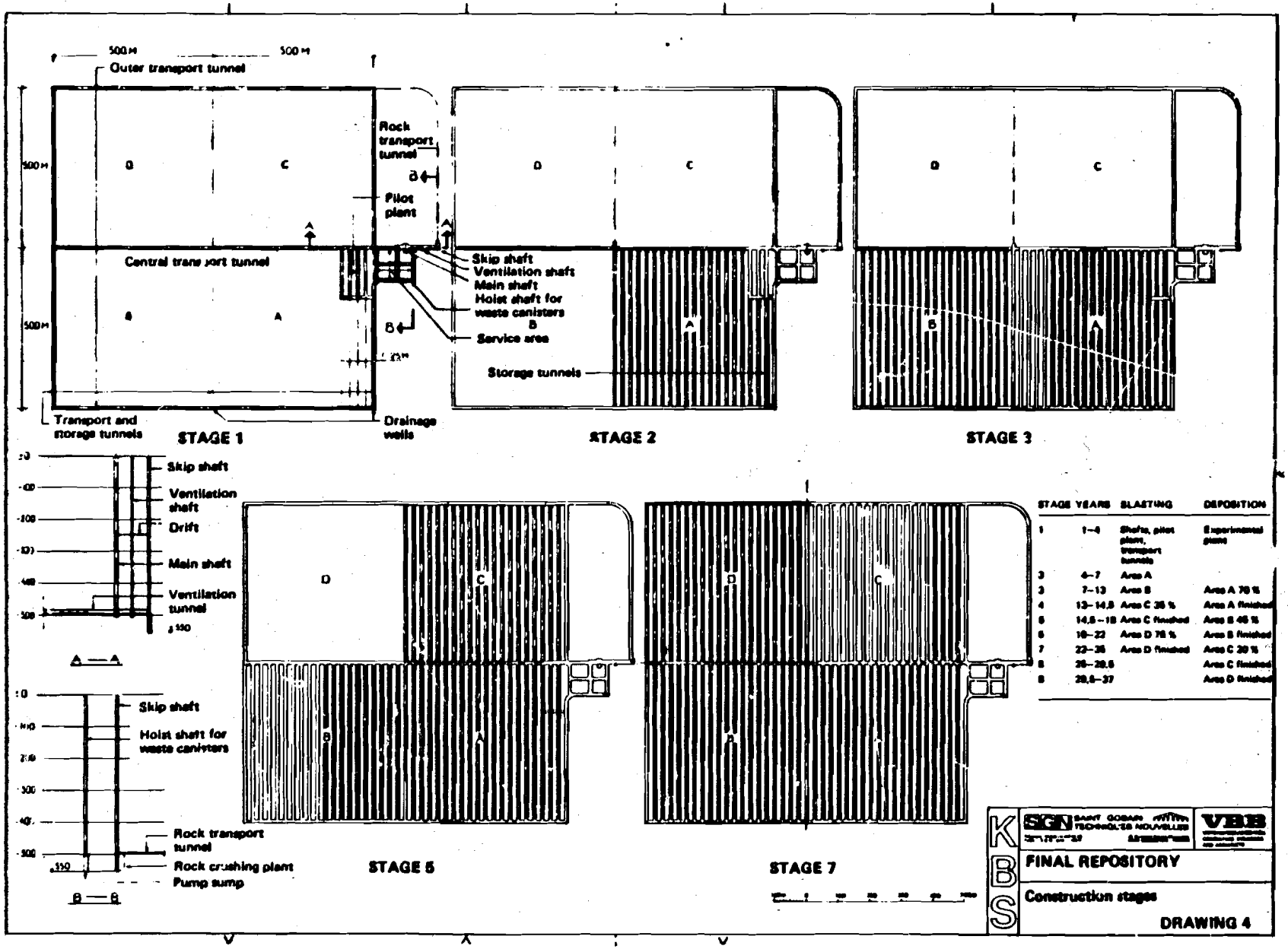


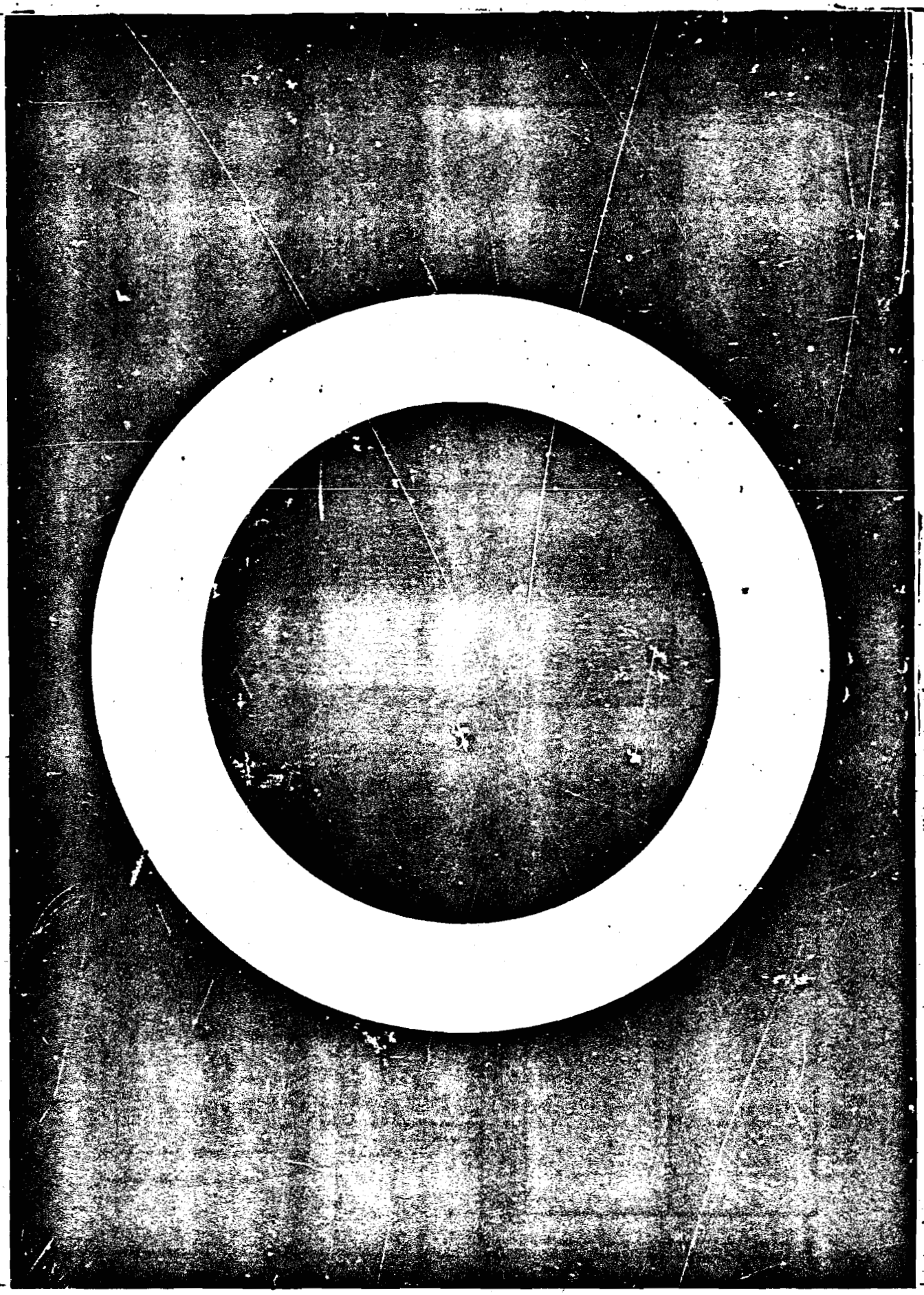

Skip shaft
Ventilation shaft, supply air
Outer transport tunnel
Rock transport tunnel
Rock crushing plant
Service area
Main shaft
Hoist shaft for waste canisters
Central transport tunnel
Transport and storage tunnels
Ventilation tunnel, exhaust air
Outer transport tunnel
SKB
FINAL REPOSITORY
Perspective drawing
DRAWING 2

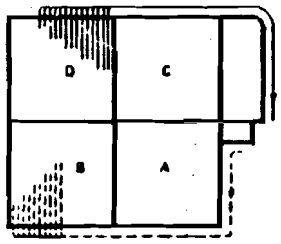



NO STORAGE IN LOCAL FRACTURE ZONE
LOCAL FRACTURE ZONE
ROCK TRANSPORT TUNNEL
SKIP SHAFT
VENTILATION SHAFT
SERVICE AREA
HOIST SHAFT FOR WASTE CANISTERS
MAIN SHAFT
STORAGE TUNNELS
TRANSPORT AND STORAGE TUNNELS
CENTRAL TRANSPORT TUNNEL
OUTER TRANSPORT TUNNEL
FINNSJÖN
RAILWAY
PART OF REPOSITORY SEPARATED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

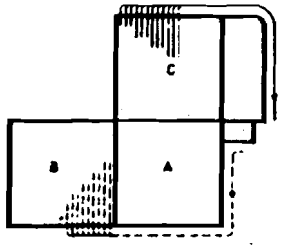



500 m
500 m
Outer transport tunnel
500 m
500 m
D
C
B
A
Rock transport tunnel
Pilot plant
Central transport tunnel
Skip shaft
Ventilation shaft
Main shaft
Hoist shaft for waste canisters
Service area
Storage tunnels
Transport and storage tunnels
Drainage wells
STAGE 1
STAGE 2
STAGE 3
Skip shaft
Ventilation shaft
Drift
Main shaft
Ventilation tunnel
A — A
Skip shaft
Hoist shaft for waste canisters
Rock transport tunnel
Rock crushing plant
Pump sump
B — B
STAGE 5
STAGE 7
STAGE YEARS BLASTING DEPOSITION
1 1–6 Shafts, pilot plant, transport tunnels Experimental plant
3 6–7 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 70 % Area B finished
7 22–35 Area D finished Area C 30 %
8 35–36,5 Area C finished
9 36,5–37 Area D finished
KBS
SGN
VBB
FINAL REPOSITORY
Construction stages
DRAWING 4

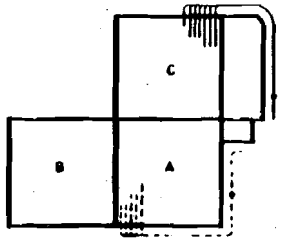




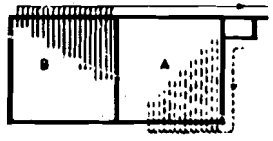
Stage 6



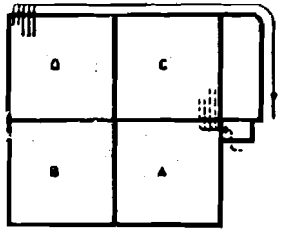
Stage 5



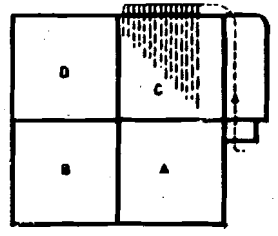
Stage 4



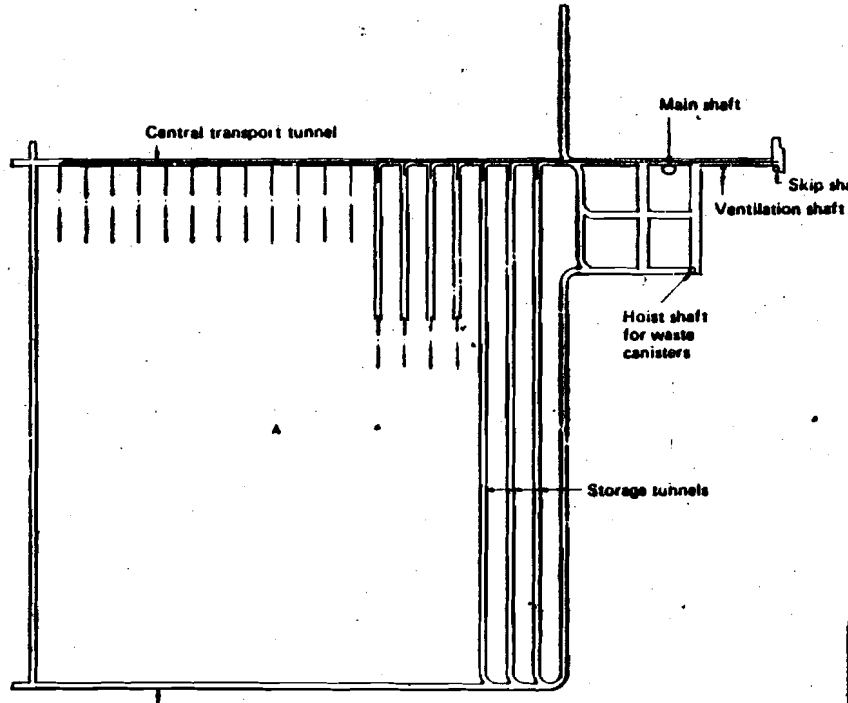
Stage 3



Stage 7



Stage 8



Stage 2

Direction of rock transport

Direction of waste transport

Note: Construction stages as per drawing 1

KBS

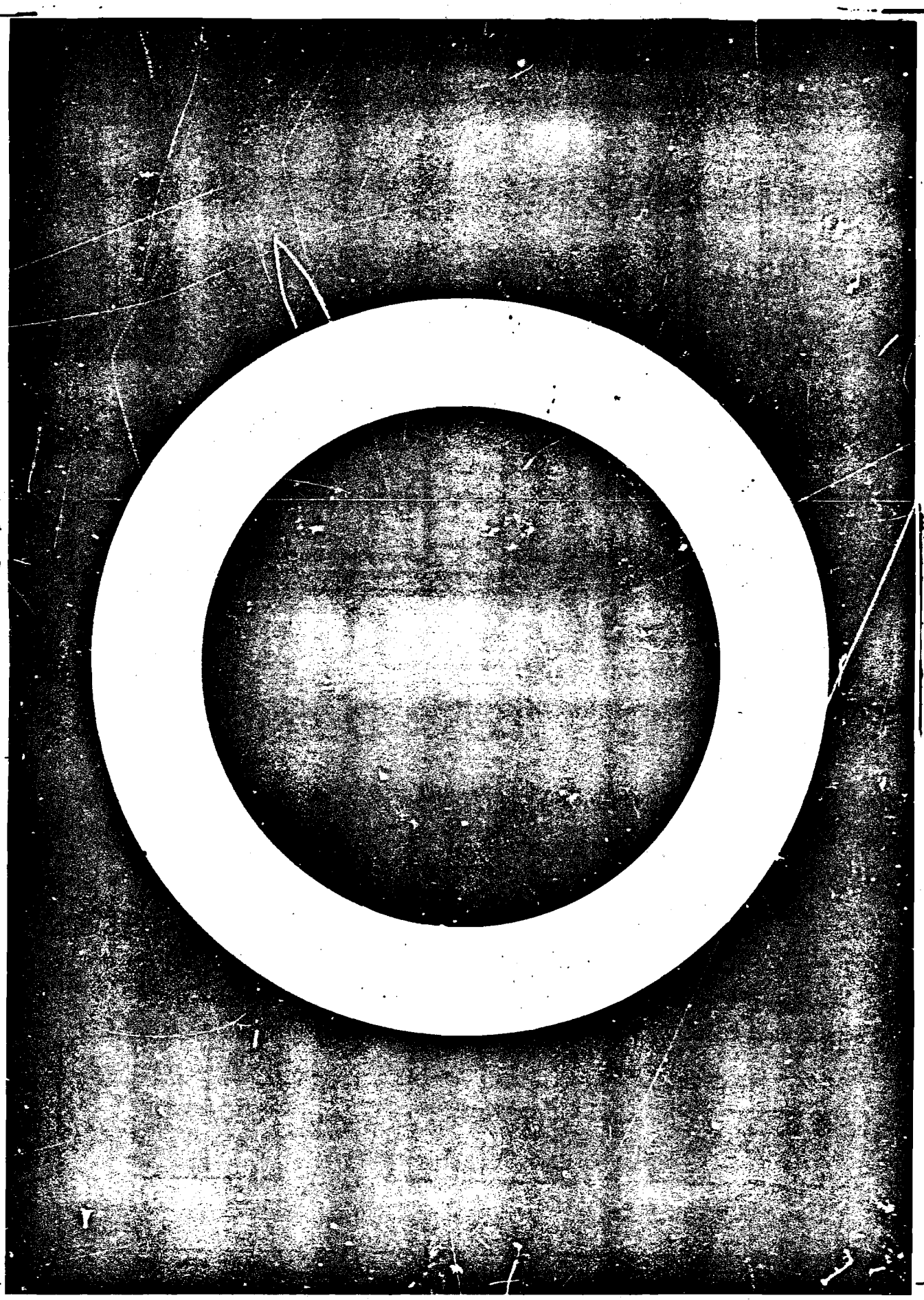
SGN SAINT GOBAIN TECHNIQUES NOUVELLES

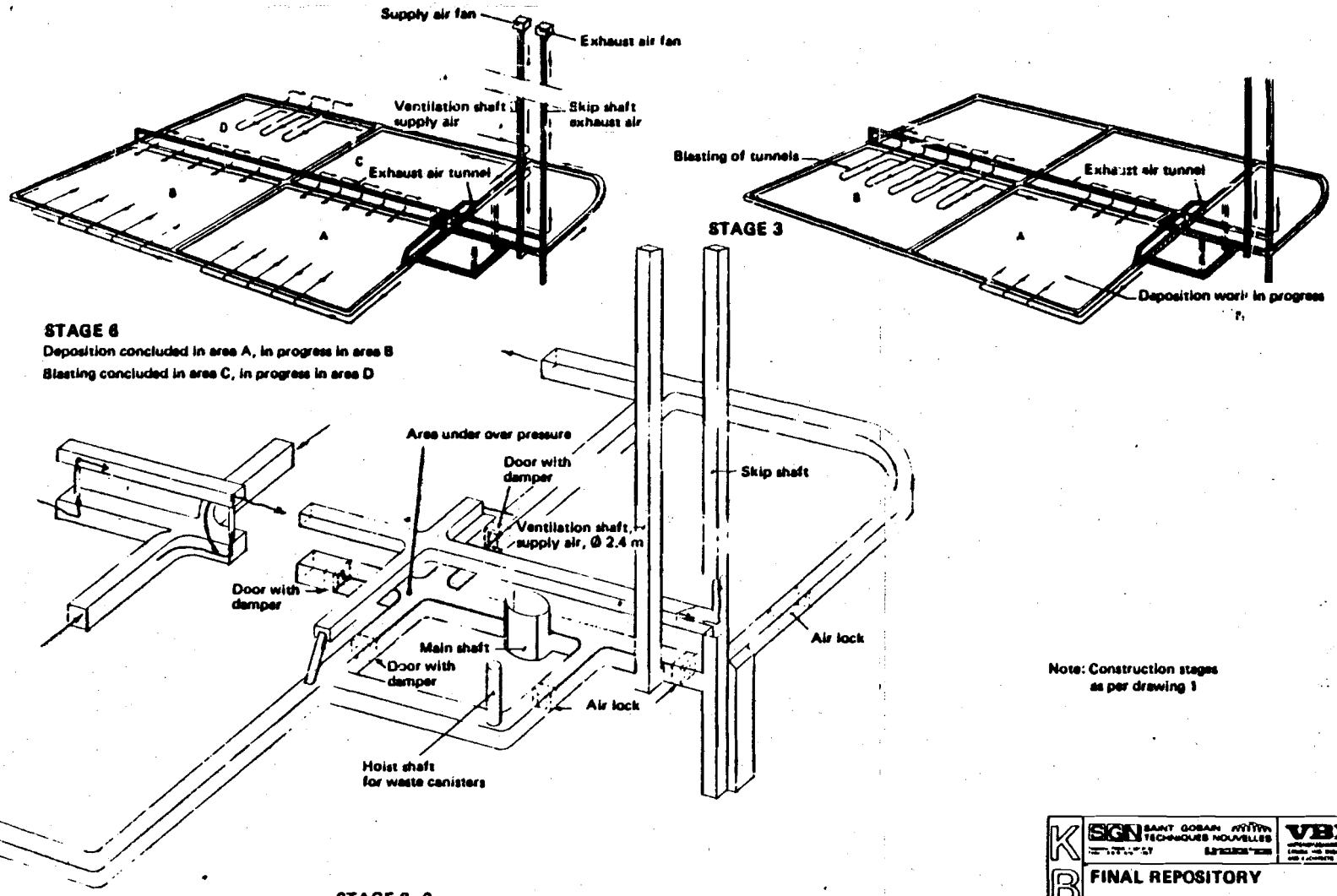
VBB

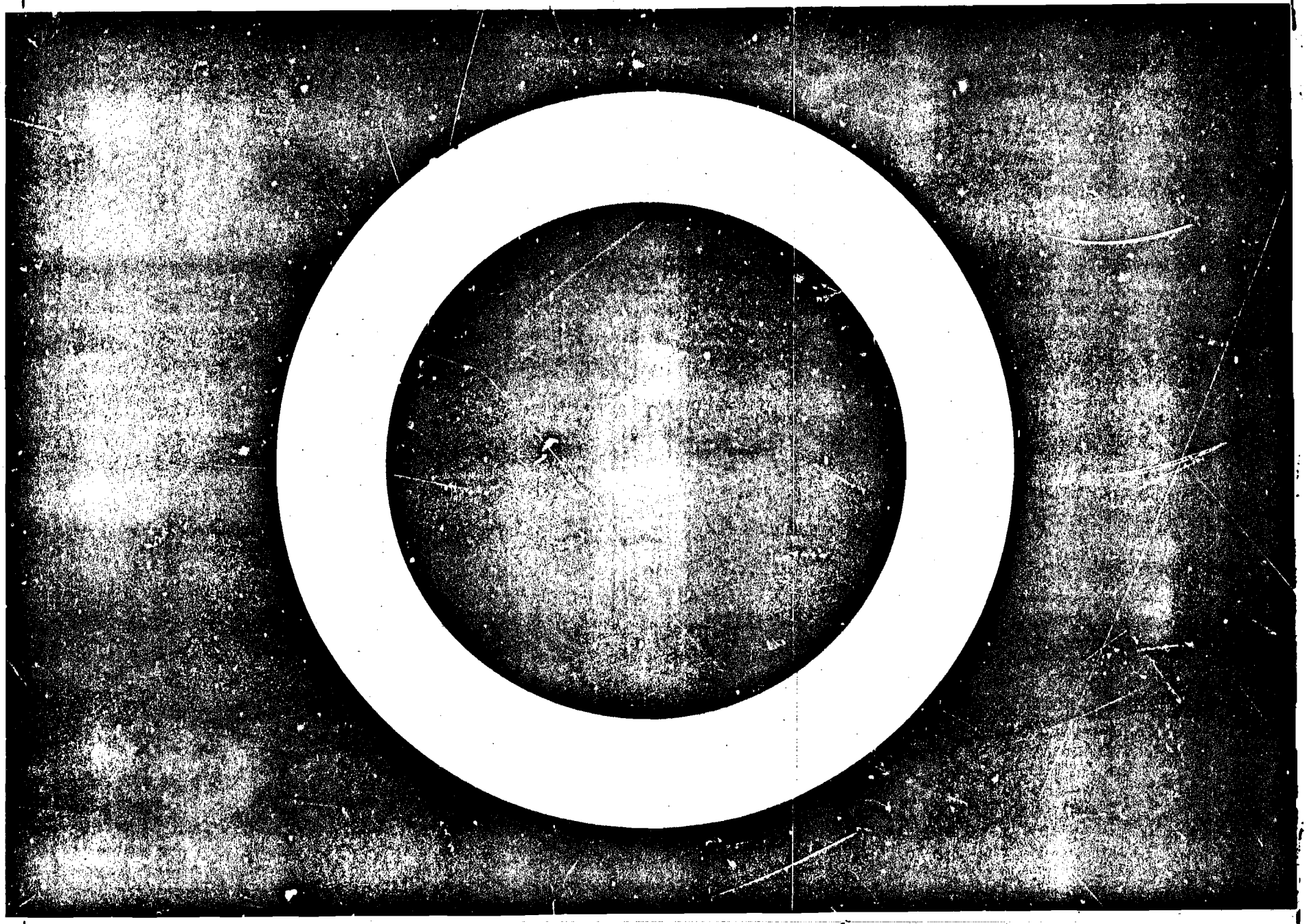
AL REPOSITORY

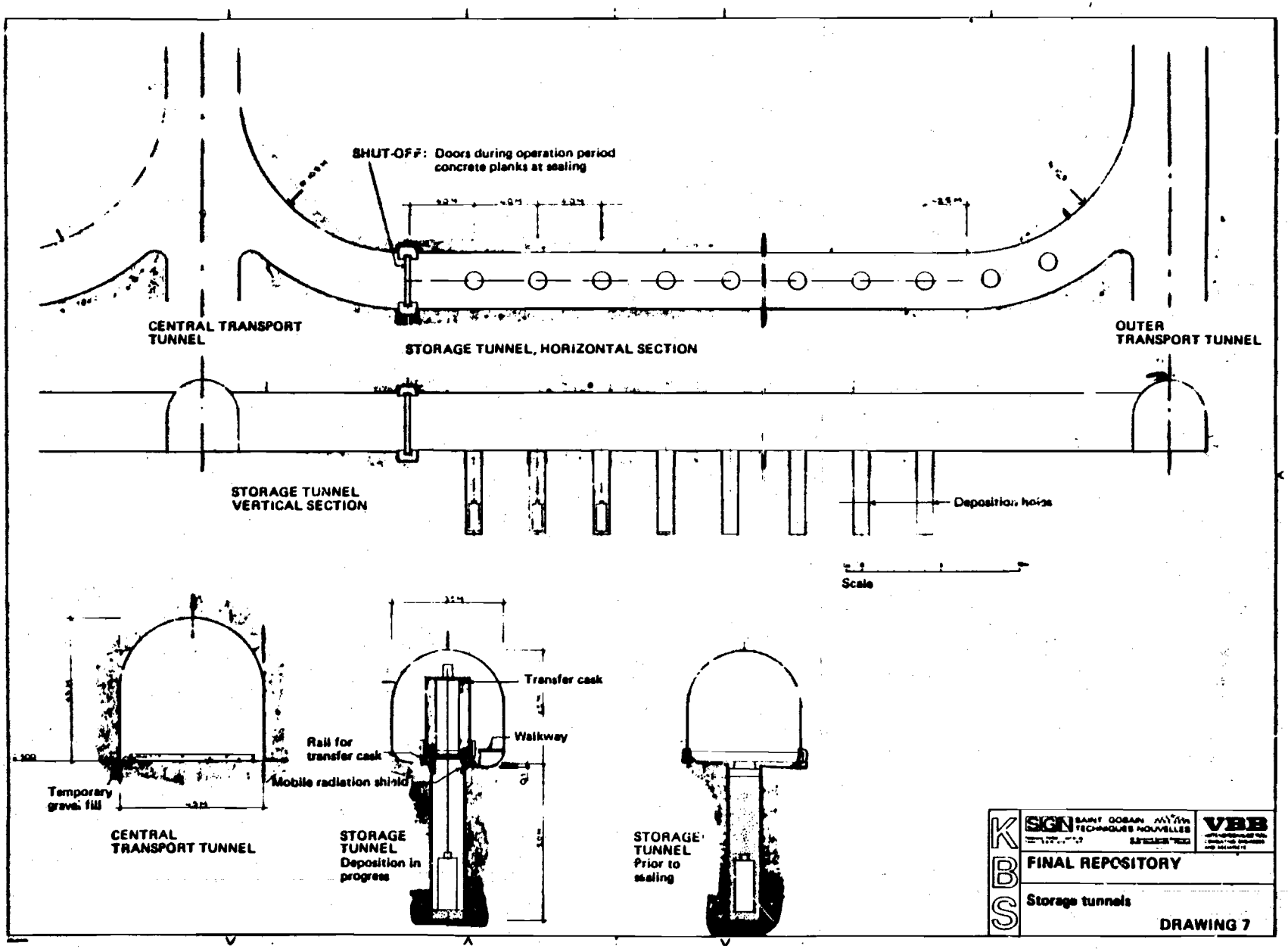
Transport routes

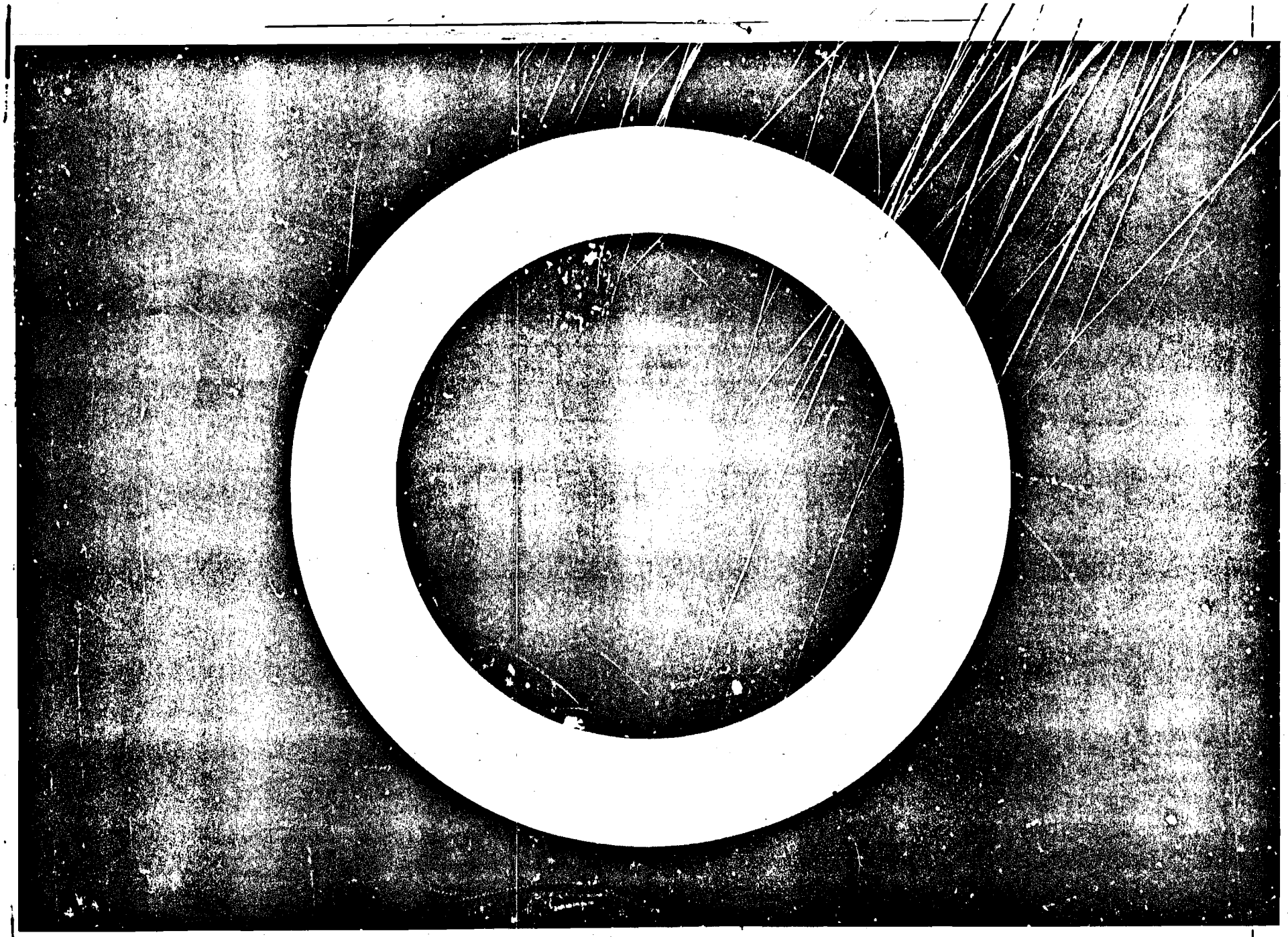
DRAWING 5

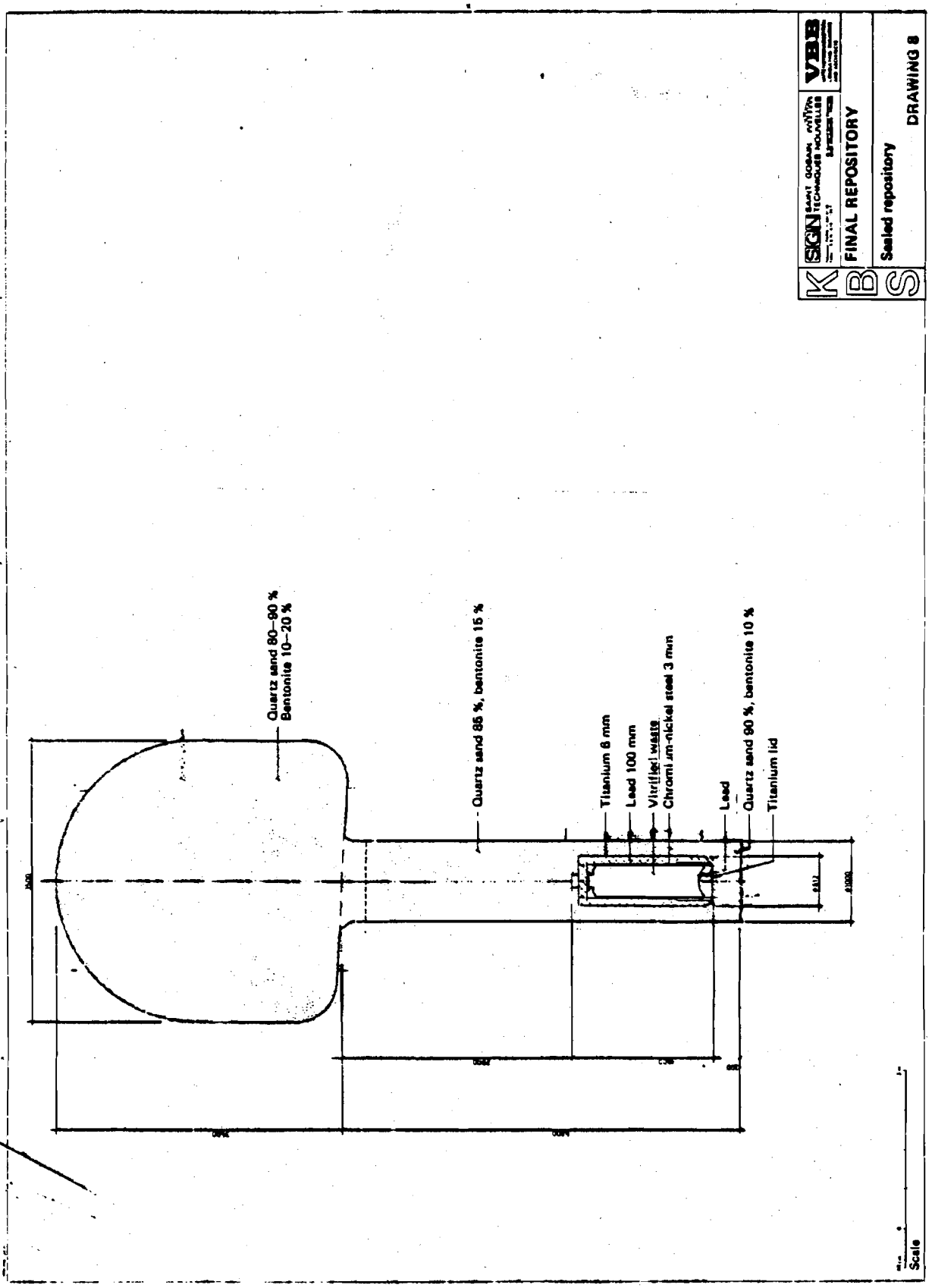


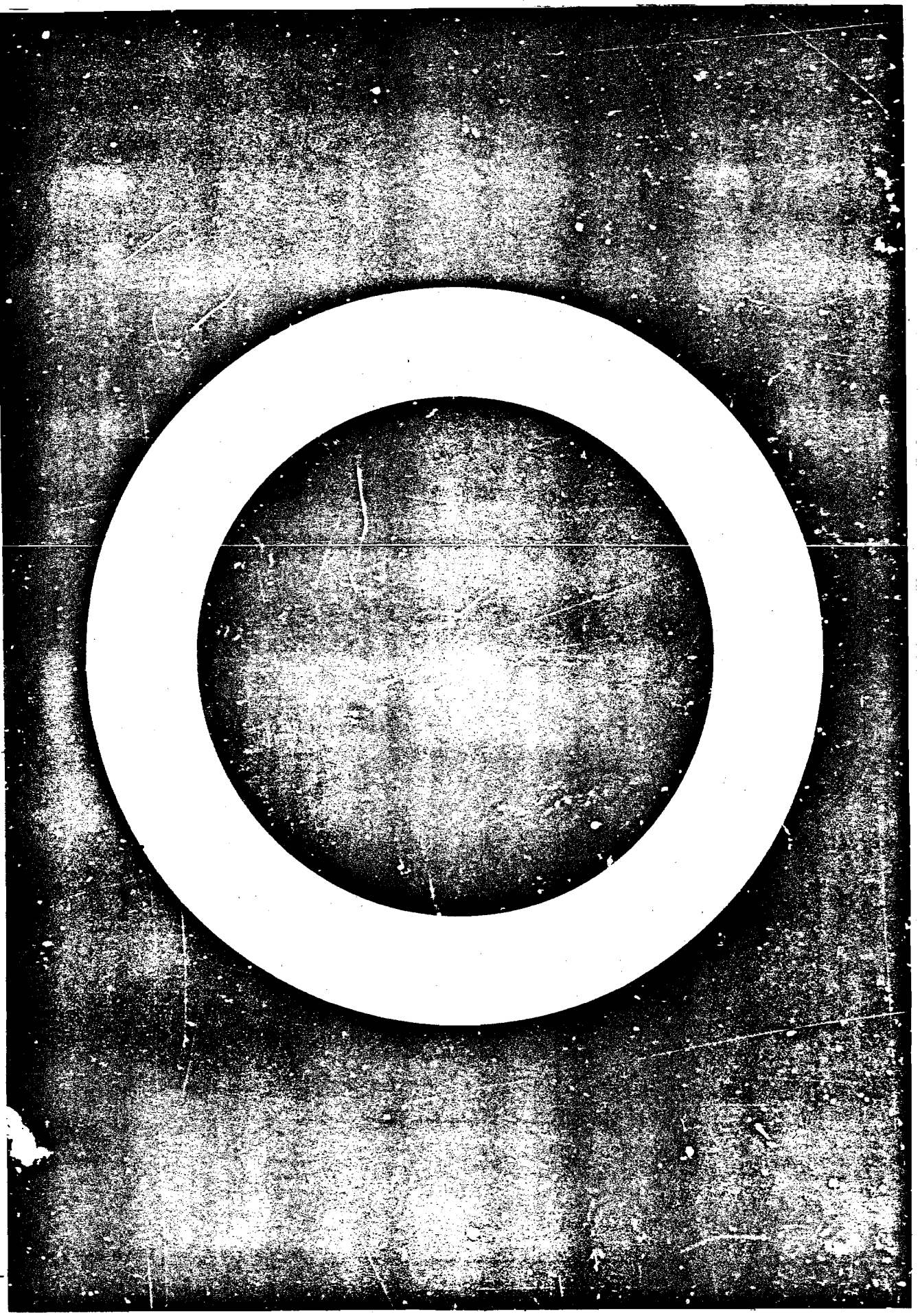

Supply air fan
Exhaust air fan
Ventilation shaft supply air
Skip shaft exhaust air
D
C
B
A
Exhaust air tunnel
STAGE 6
Deposition concluded in area A, in progress in area B
Blasting concluded in area C, in progress in area D
Blasting of tunnels
Exhaust air tunnel
STAGE 3
Deposition work in progress
Area under over pressure
Door with damper
Skip shaft
Ventilation shaft, supply air, Ø 2.4 m
Door with damper
Main shaft
Door with damper
Air lock
Air lock
Hoist shaft for waste canisters
STAGE 8–9
Note: Construction stages as per drawing 1
KBS
SGN SAINT GOBAIN TECHNIQUES NOUVELLES
VBB
FINAL REPOSITORY
Ventilation
DRAWING 6

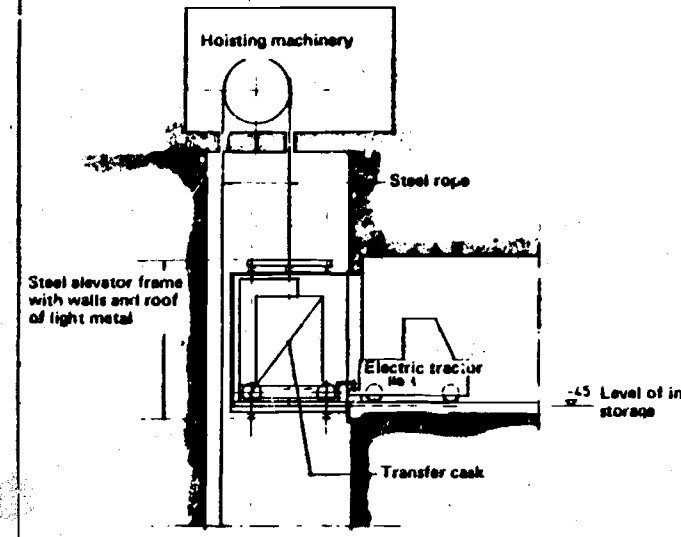


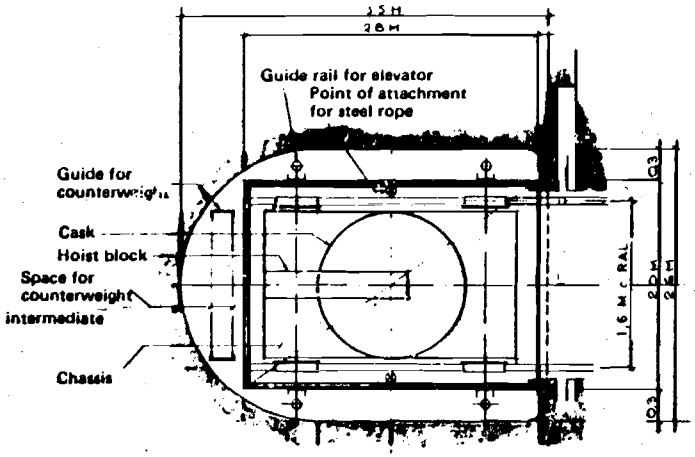

SHUT-OFF: Doors during operation period
concrete planks at sealing
CENTRAL TRANSPORT TUNNEL
STORAGE TUNNEL, HORIZONTAL SECTION
OUTER TRANSPORT TUNNEL
STORAGE TUNNEL VERTICAL SECTION
Deposition holes
Scale
Transfer cask
Walkway
Rail for transfer cask
Mobile radiation shield
Temporary gravel fill
CENTRAL TRANSPORT TUNNEL
STORAGE TUNNEL Deposition in progress
STORAGE TUNNEL Prior to sealing
KBS
SGN SAINT GOBAIN TECHNIQUES NOUVELLES
VBB
FINAL REPOSITORY
Storage tunnels
DRAWING 7



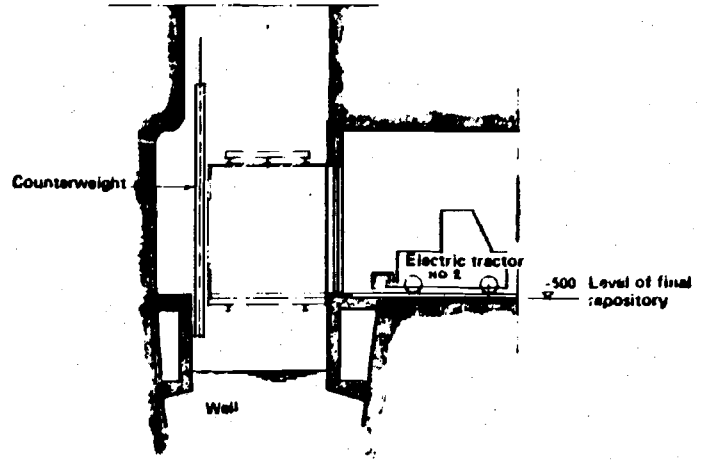

Quartz sand 80–90 %
Bentonite 10–20 %
Quartz sand 85 %, bentonite 15 %
Titanium 6 mm
Lead 100 mm
Vitrified waste
Chromium-nickel steel 3 mm
Lead
Quartz sand 90 %, bentonite 10 %
Titanium lid
Scale
SGN
VBB
FINAL REPOSITORY
Sealed repository
DRAWING 8





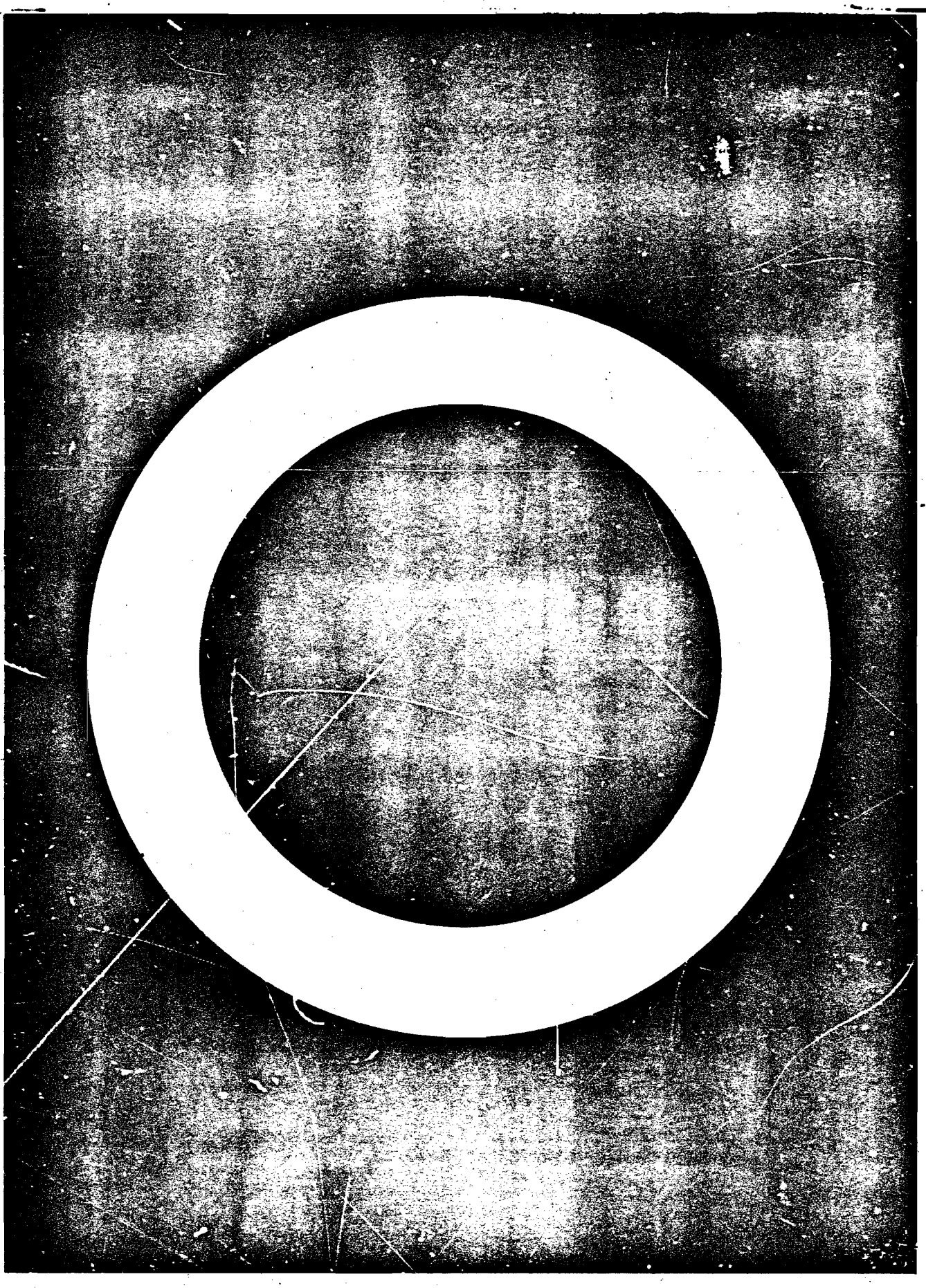


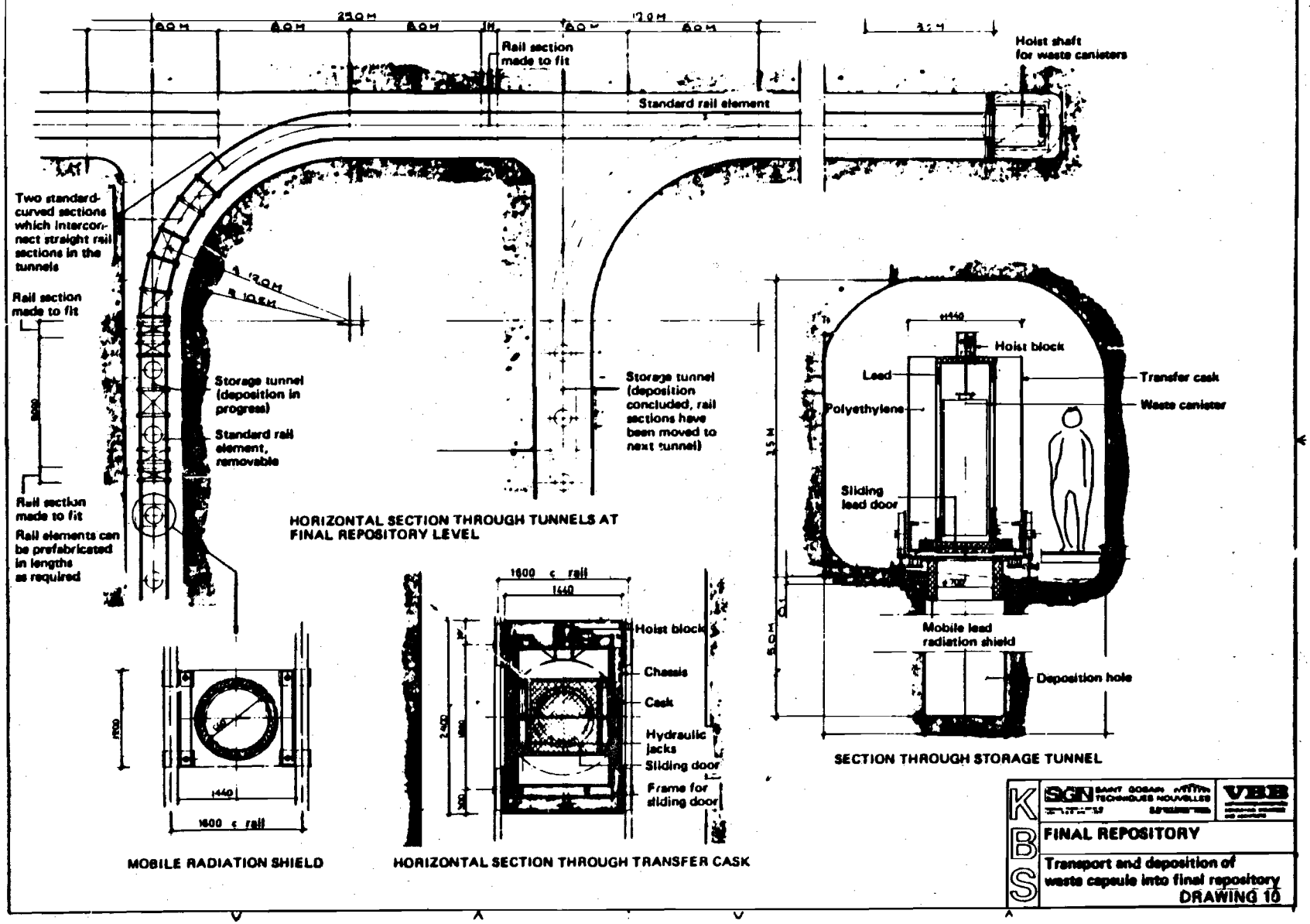
MAIN SECTION THROUGH HOIST SHAFT

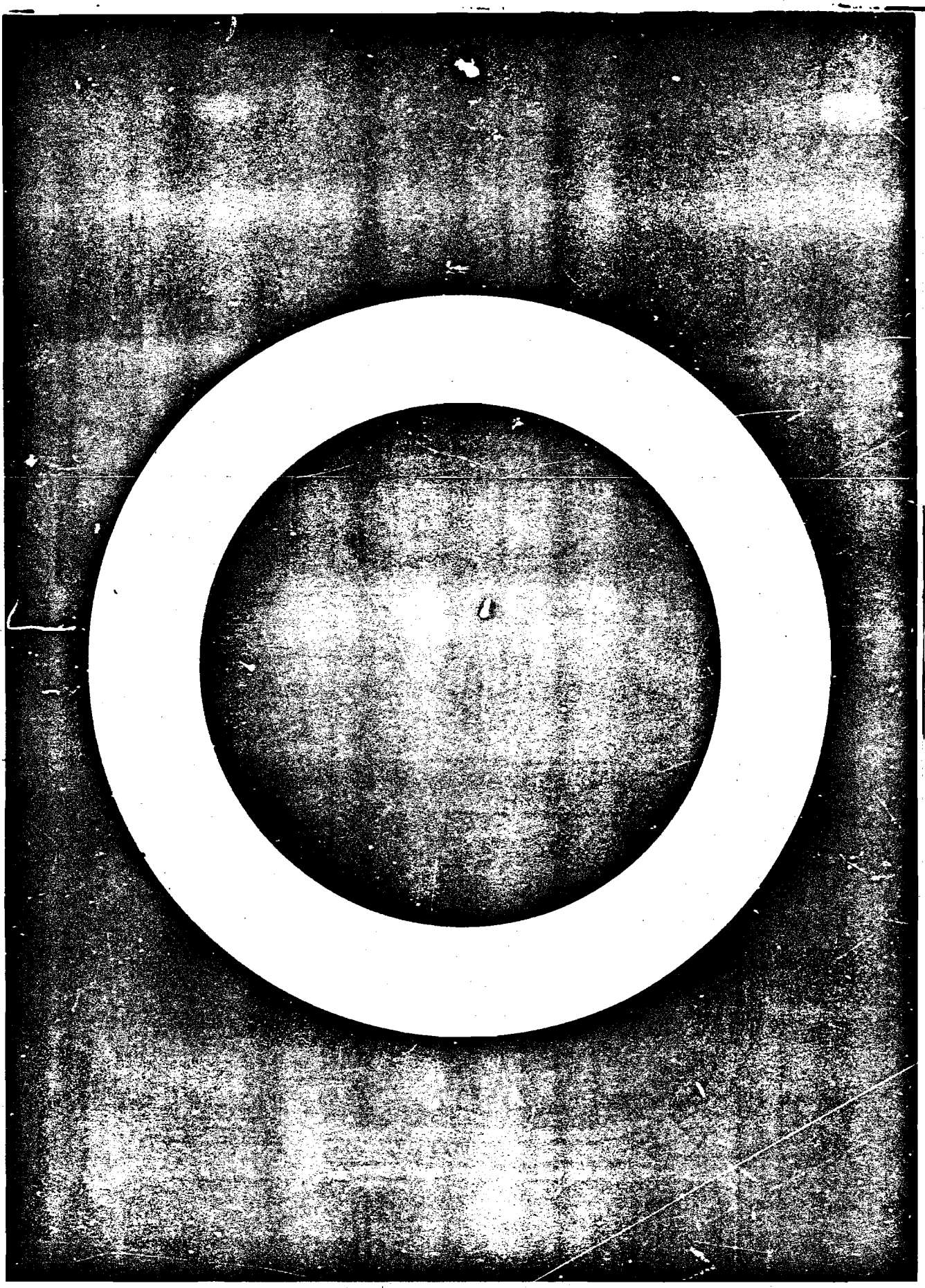


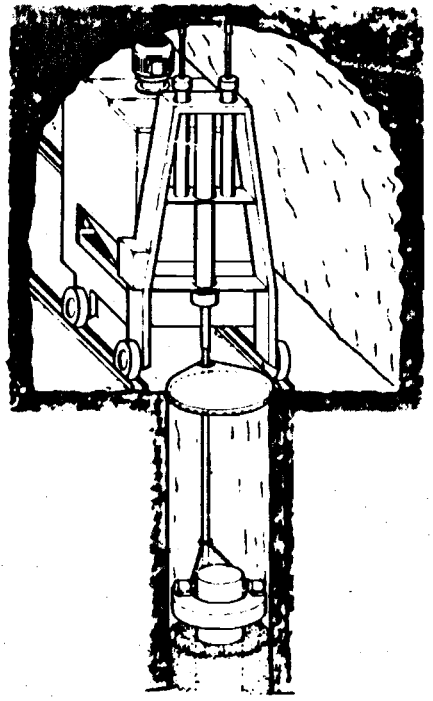
VERTICAL SECTION THROUGH HOIST SHAFT

KBS	SGN SAINT GOBAIN TECHNIQUES NOUVELLES	VBB
	FINAL REPOSITORY	
	Transport of waste canister from intermediate storage to final repository	DRAWING 9

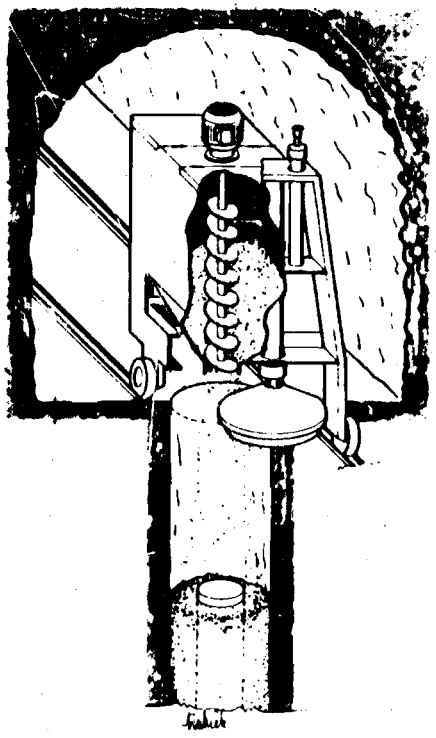



Rail section made to fit
Standard rail element
Hoist shaft for waste canisters
Two standard-curved sections which interconnect straight rail sections in the tunnels
Rail section made to fit
Rail section made to fit
Rail elements can be prefabricated in lengths as required
Storage tunnel (deposition in progress)
Standard rail element, removable
Storage tunnel (deposition concluded, rail sections have been moved to next tunnel)
HORIZONTAL SECTION THROUGH TUNNELS AT FINAL REPOSITORY LEVEL
Hoist block
Lead
Transfer cask
Polyethylene
Waste canister
Sliding lead door
Mobile lead radiation shield
Deposition hole
SECTION THROUGH STORAGE TUNNEL
1600 ℄ rail
1440
MOBILE RADIATION SHIELD
Hoist block
Chassis
Cask
Hydraulic jacks
Sliding door
Frame for sliding door
HORIZONTAL SECTION THROUGH TRANSFER CASK
KBS
SGN
VBB
FINAL REPOSITORY
Transport and deposition of waste capsule into final repository
DRAWING 10

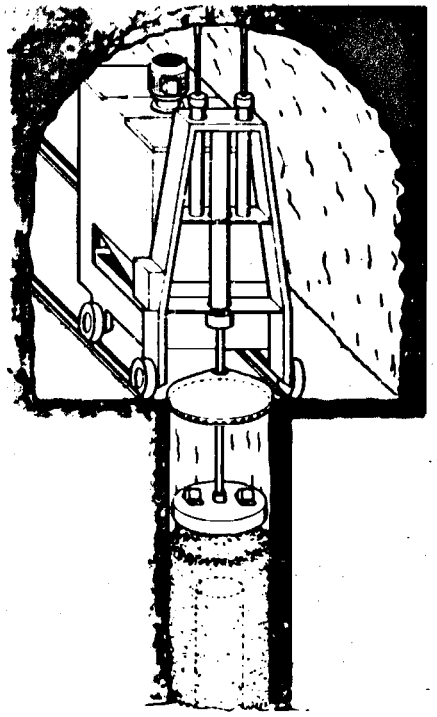




Compaction of side bed



Backfill with buffer material



Compaction of top bed

KBS

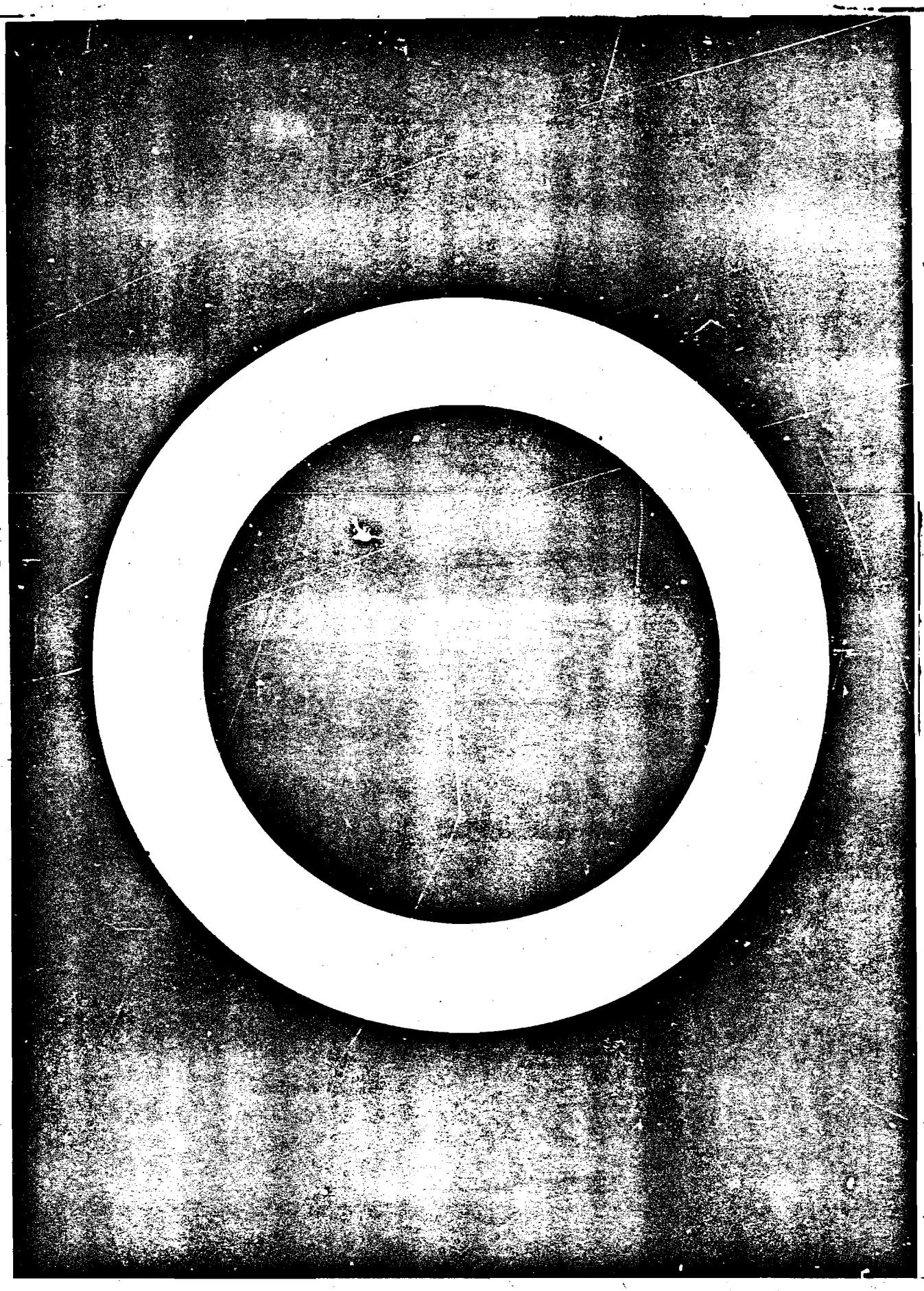
SGN SAINT GOBAIN TECHNIQUES NOUVELLES

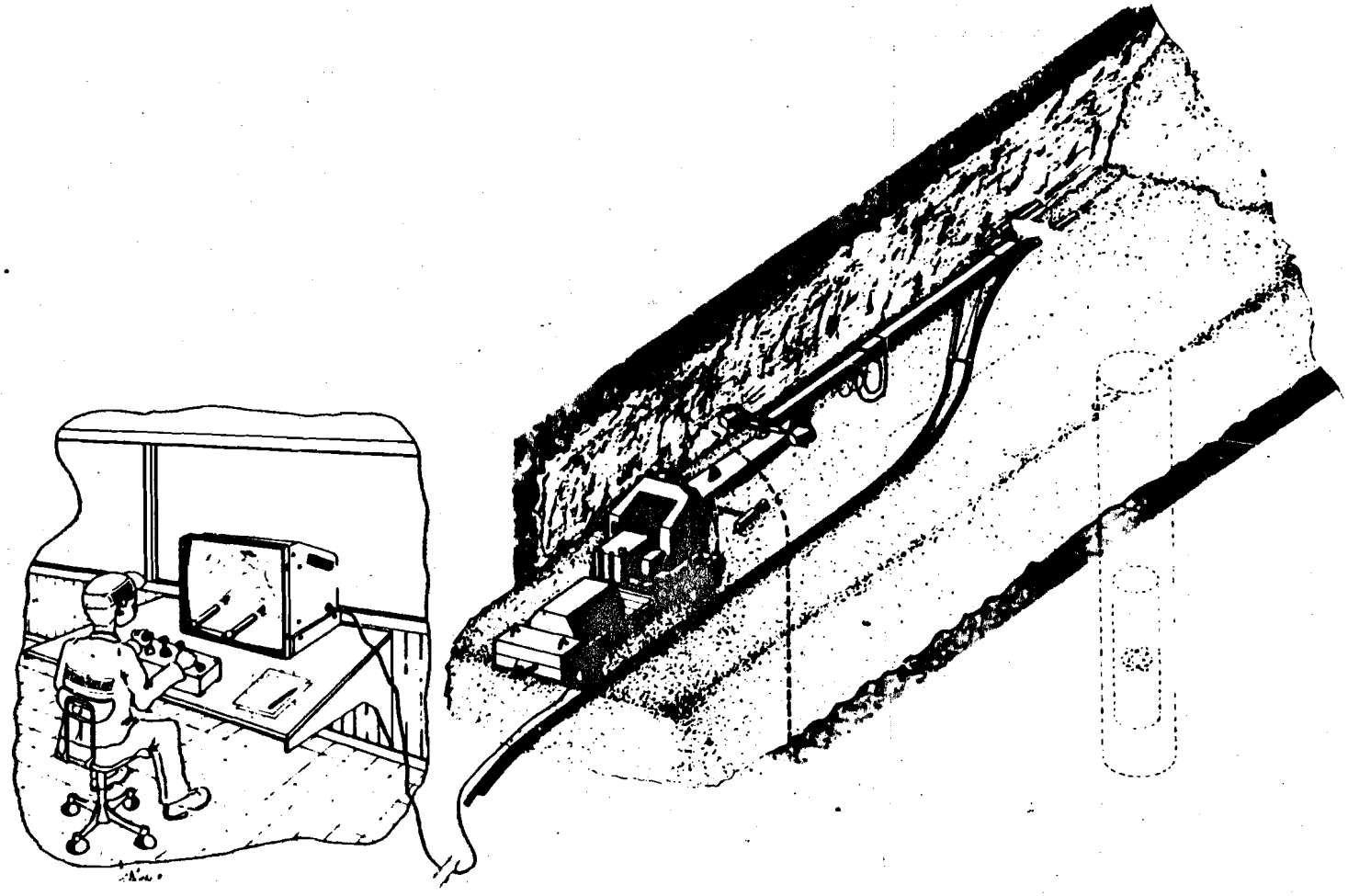
VBB

FINAL REPOSITORY

Sealing of deposition holes

DRAWING 11





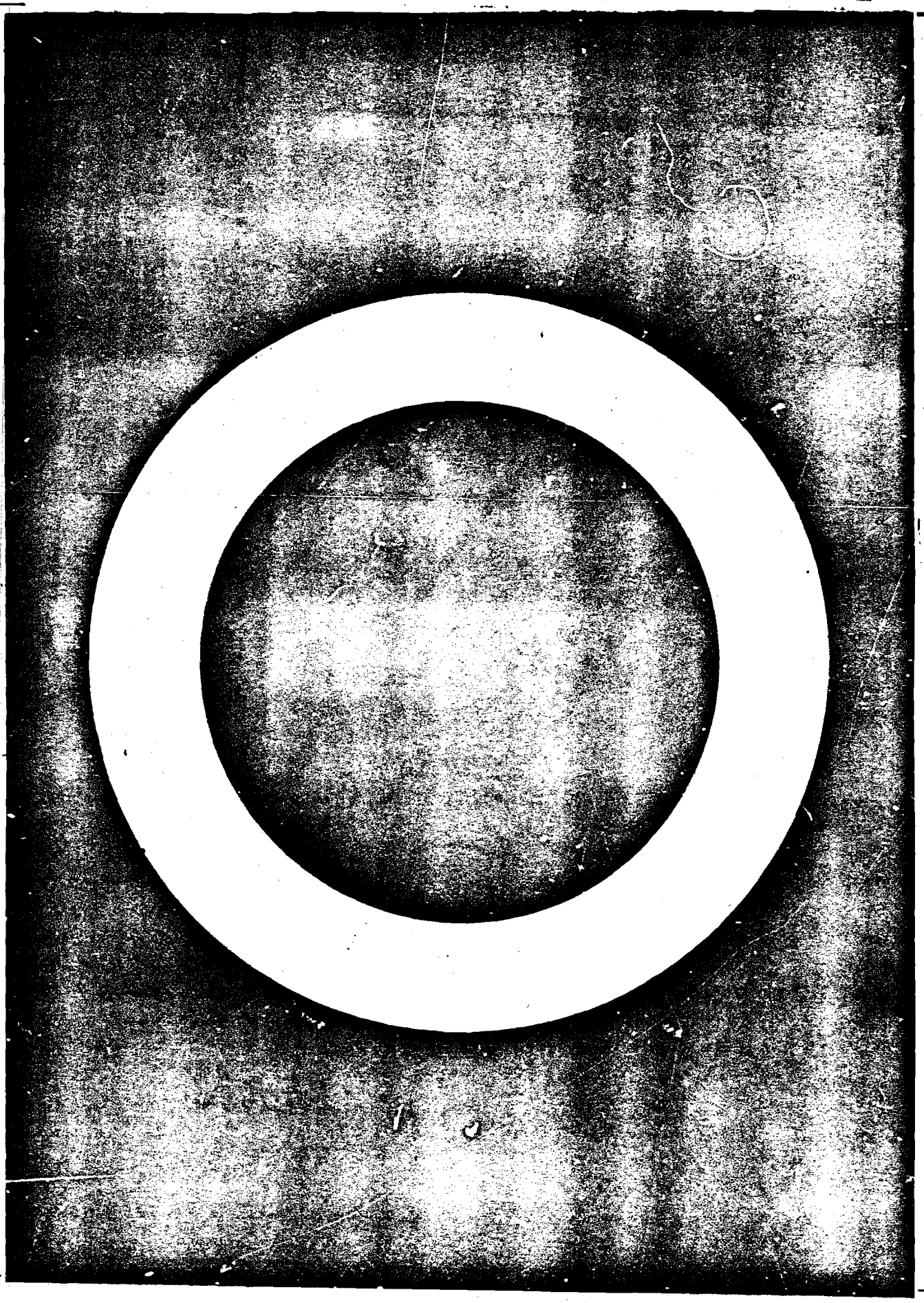
Spraying of buffer material

KBS | SGN SAINT GOBAIN TECHNIQUES NOUVELLES | VBB

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 groundvibrations 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.

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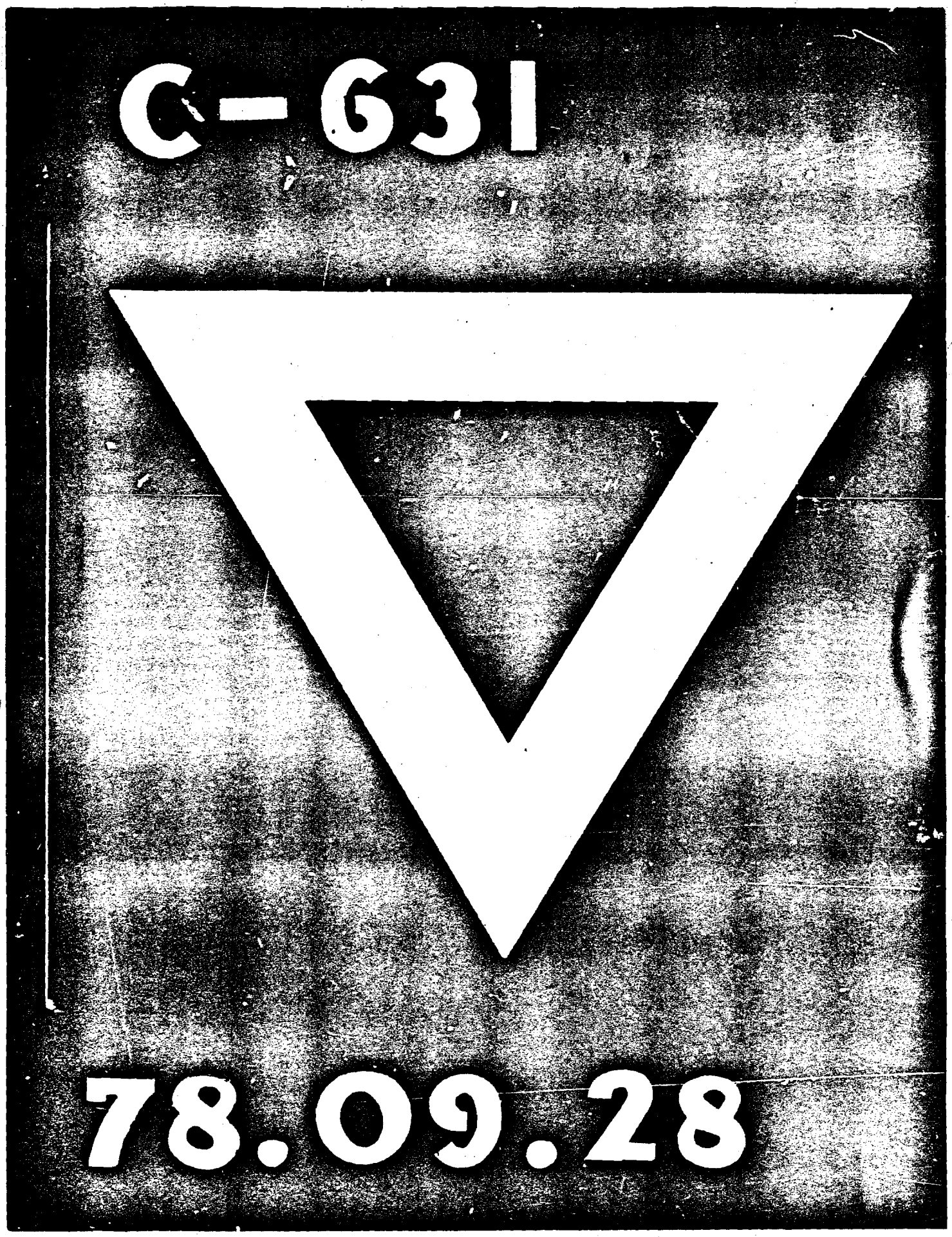
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