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

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


## 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

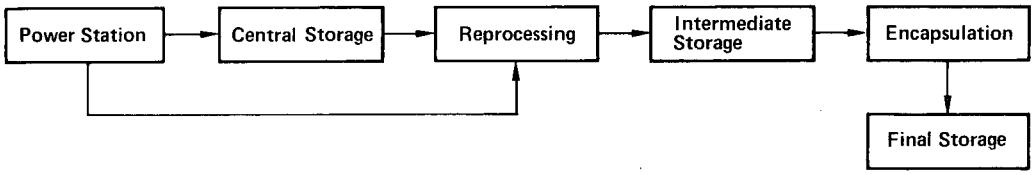
In order to provide a basis for the report, KBS has carried out a great number of technical-scientific investigations and surveys. The results of these are published in KBS Technical Reports. 56 volumes of these reports have so far been published, (see volume I, appendix 3.)

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.

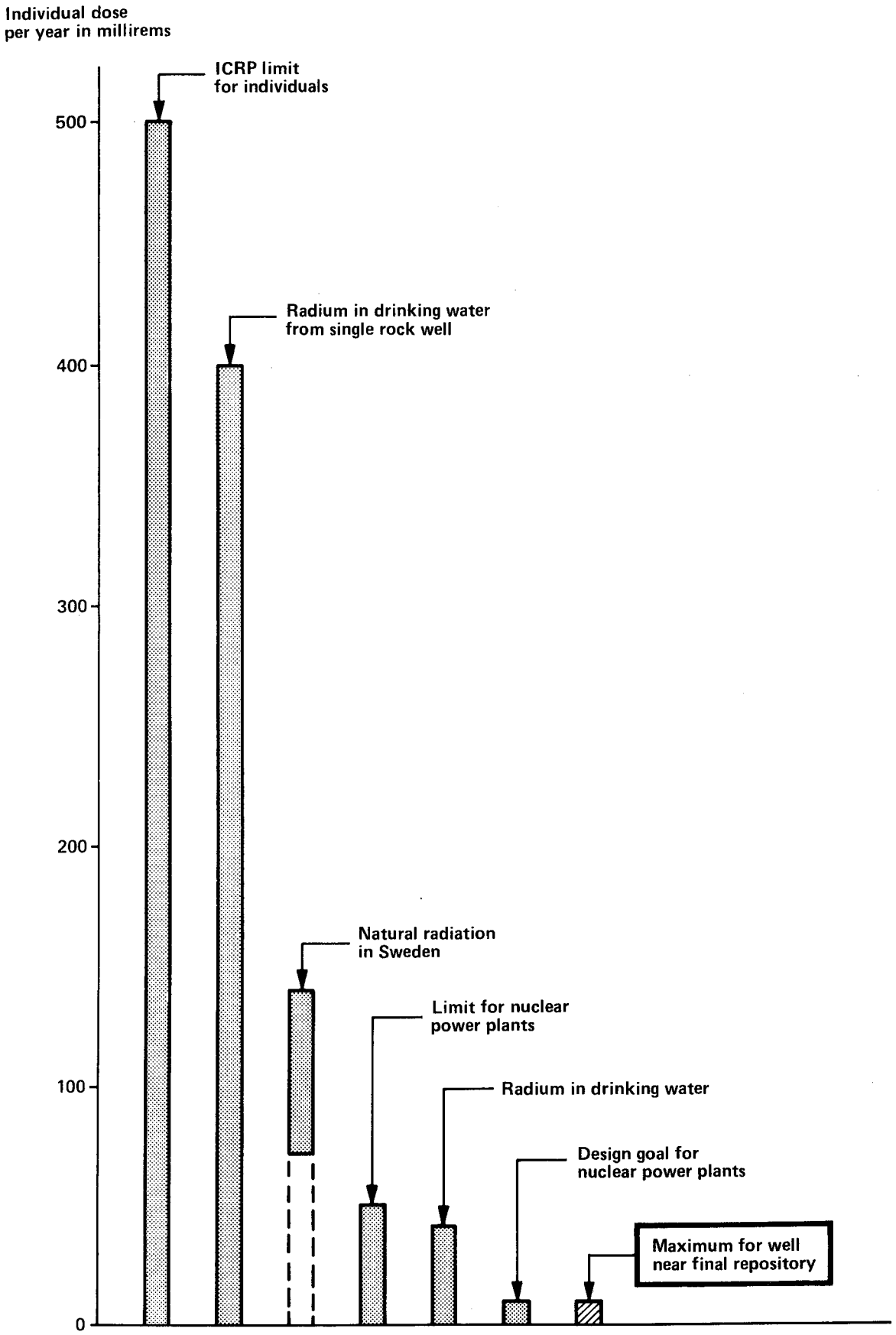
As mentioned above, the assumptions and data used in the safety analysis were selected with safety margins. It is considered probable that the dosage will be approximately 1/100th of the maximum value of 13 millirems per year given above. One reason for this is that the very low rate of water flow in 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.

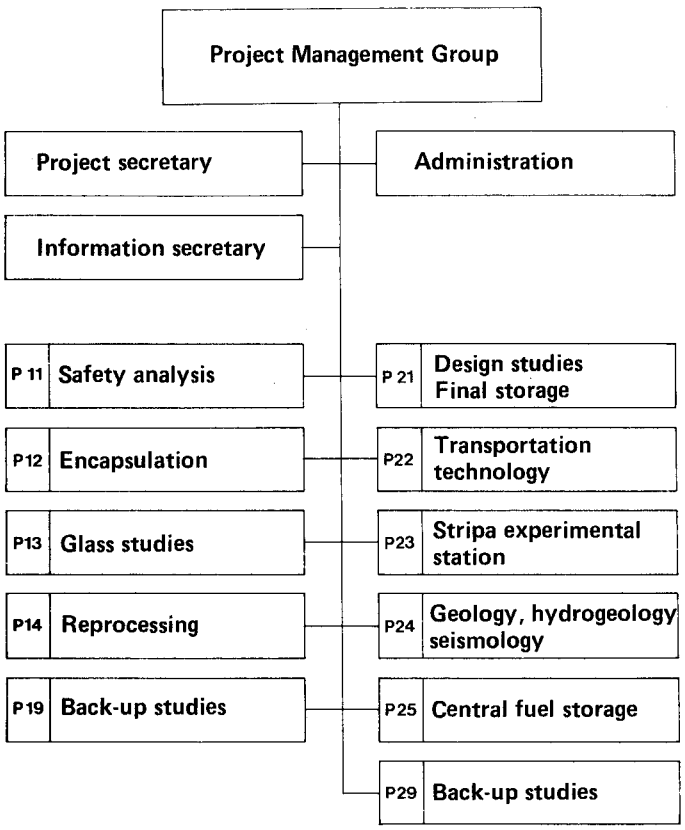



*Figure 1-1. Organization plan for the Nuclear Fuel Safety Project (KBS).*

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.

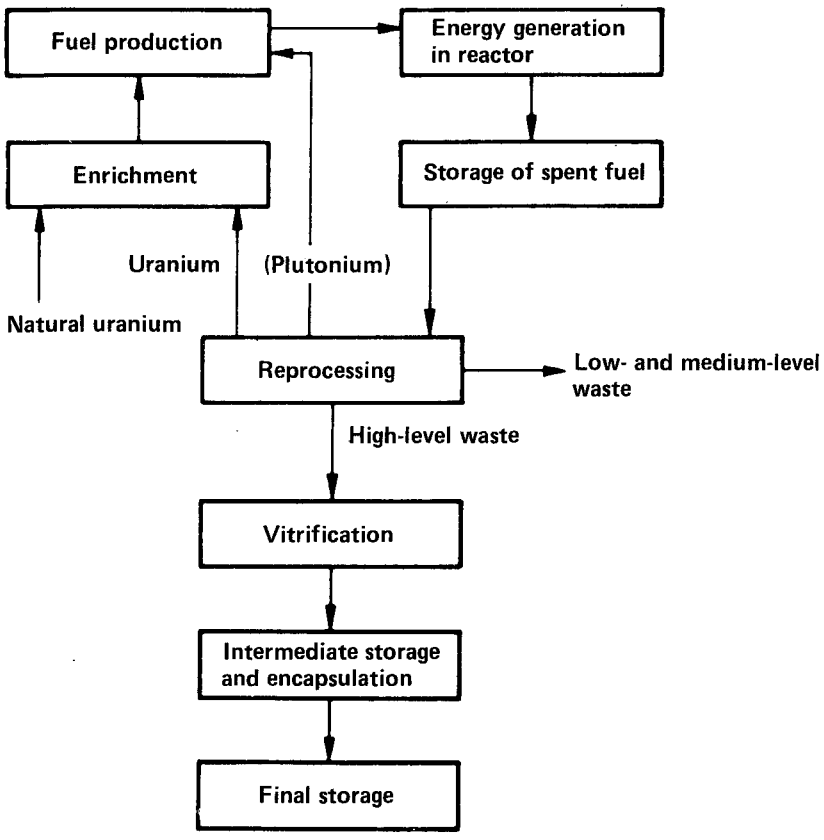



*Figure 2-1. The reprocessing alternative. Flow scheme for the fuel cycle with reprocessing of spent fuel and vitrification of the high-level waste.*

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

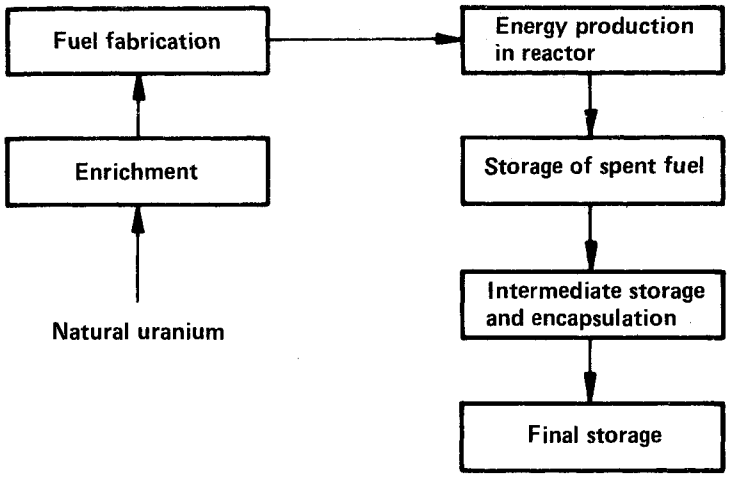



*Figure 2-2. The direct deposition alternative. Flow scheme for the fuel cycle with direct storage of the spent fuel without reprocessing.*

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

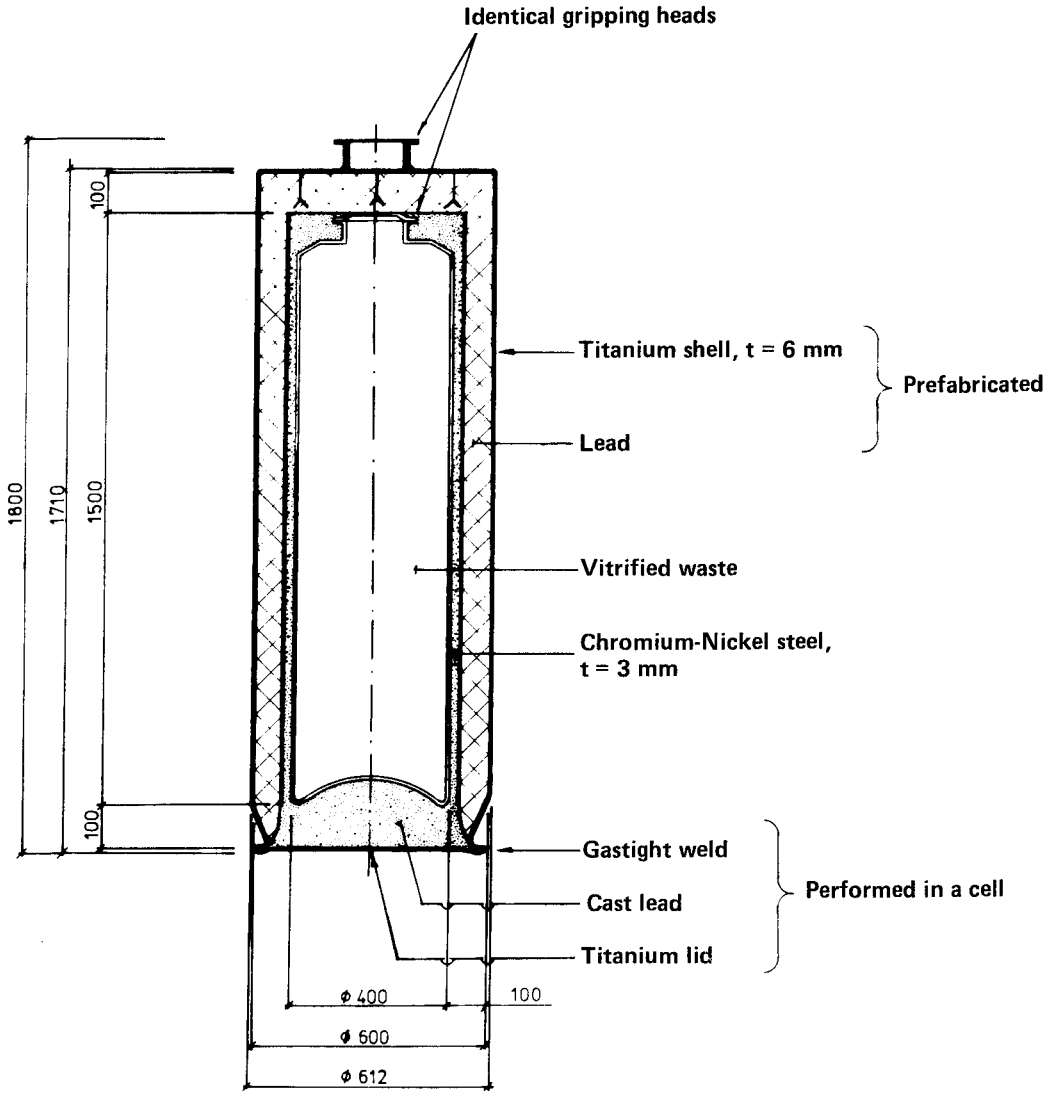
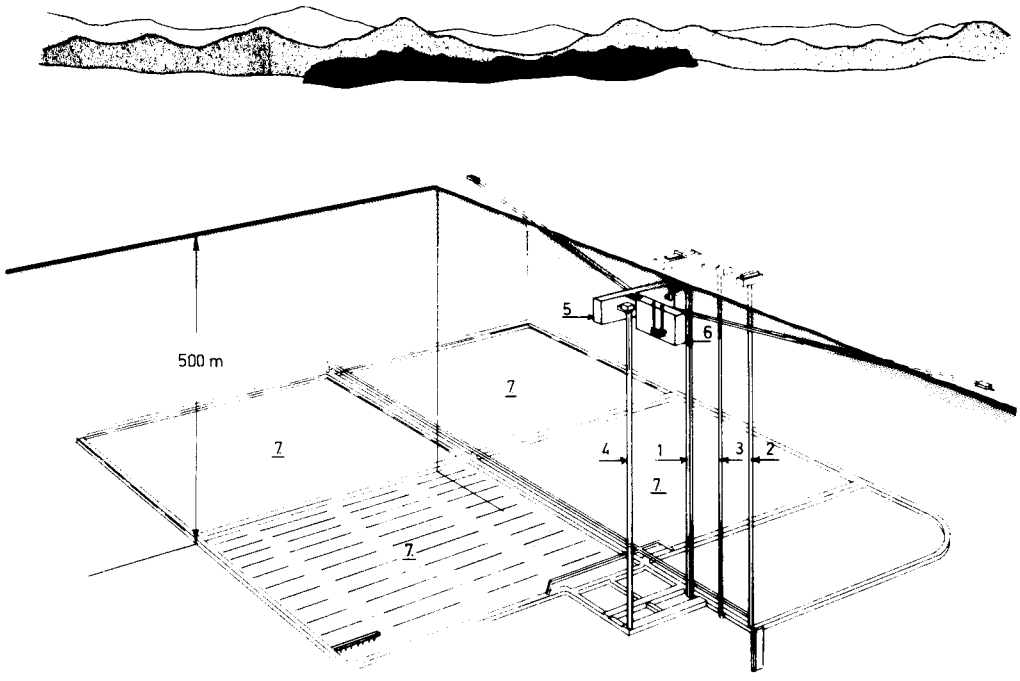



*Figure 3-1. Lead-titanium canister. Waste cylinder with vitrified high-level waste enclosed in a canister of lead and titanium. Total weight approx. 3900 kg.*

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




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

*Figure 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.*

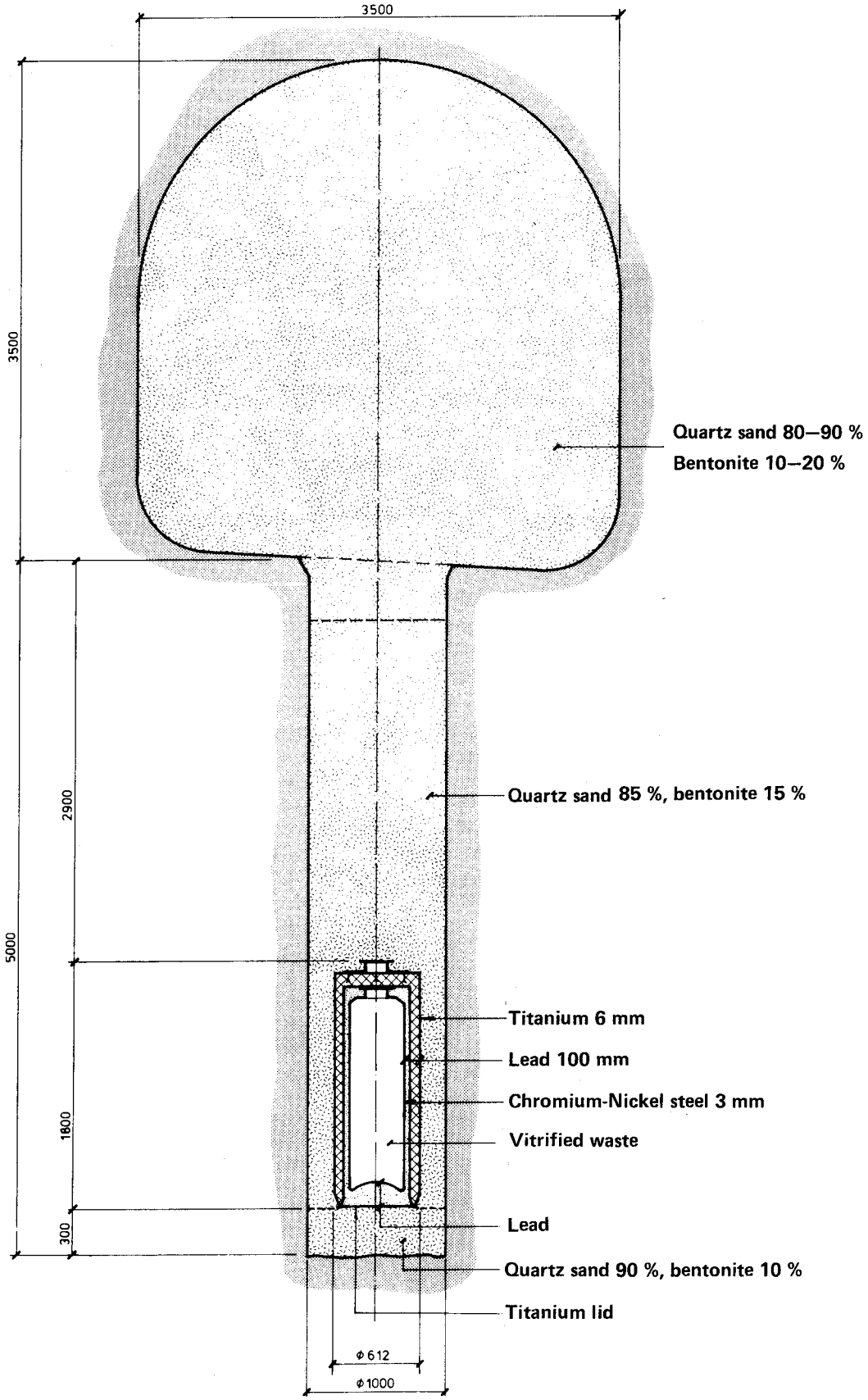



*Figure 3-3. The sealed final repository. Tunnels and storage holes are completely filled with a buffer material consisting of quartz sand and bentonite.*

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.

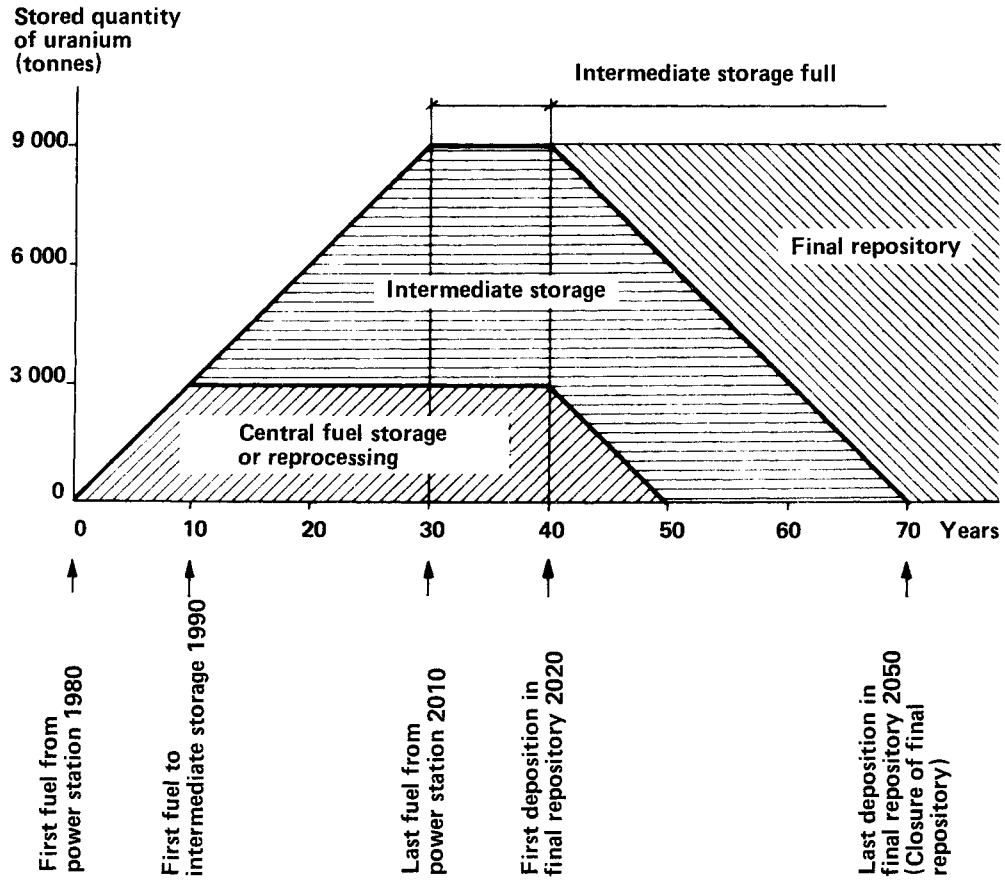



*Figure 3-4. Diagram showing capacity requirements for the central fuel storage facility and final repository at different points in time.*

### 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.

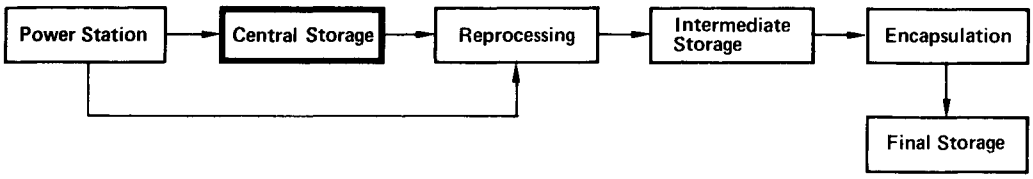
One advantage of being able to postpone a final decision on the question of large-scale reprocessing is the fact that new methods for converting the waste into solid form may be developed. These new methods may make final storage simpler and less expensive.

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

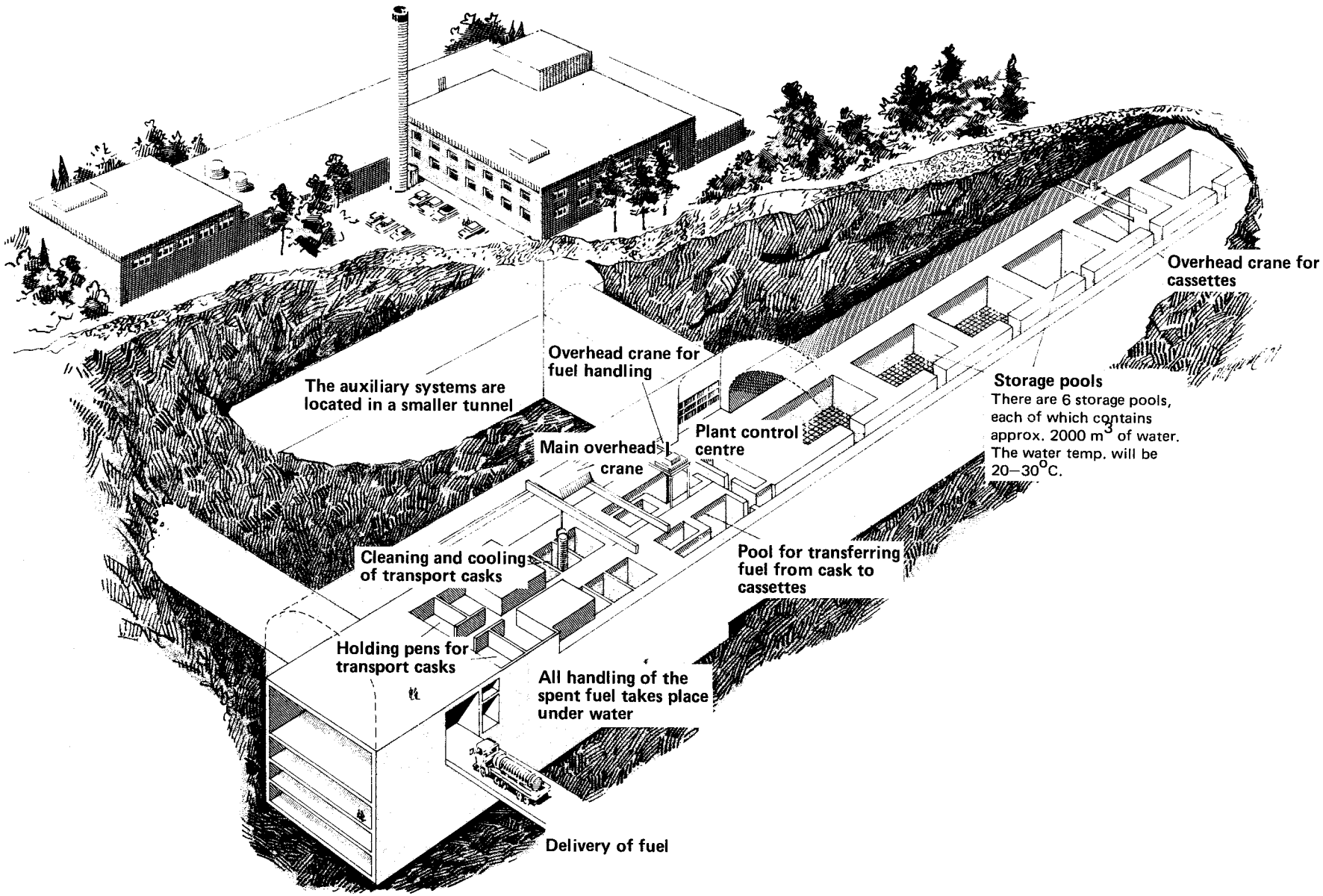



*Figure 4-1. Perspective drawing of the central storage facility for spent fuel.*

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 thorougly 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.

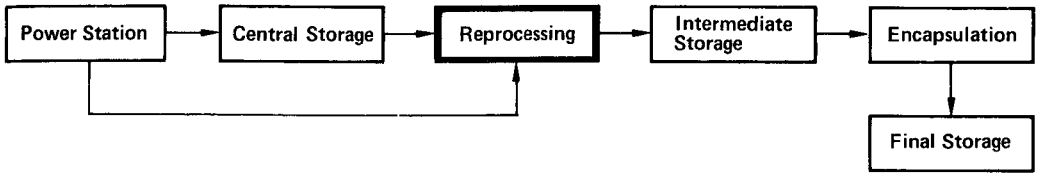
## 4.3 OPERATION OF FACILITY

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 autohorities 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 in 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.

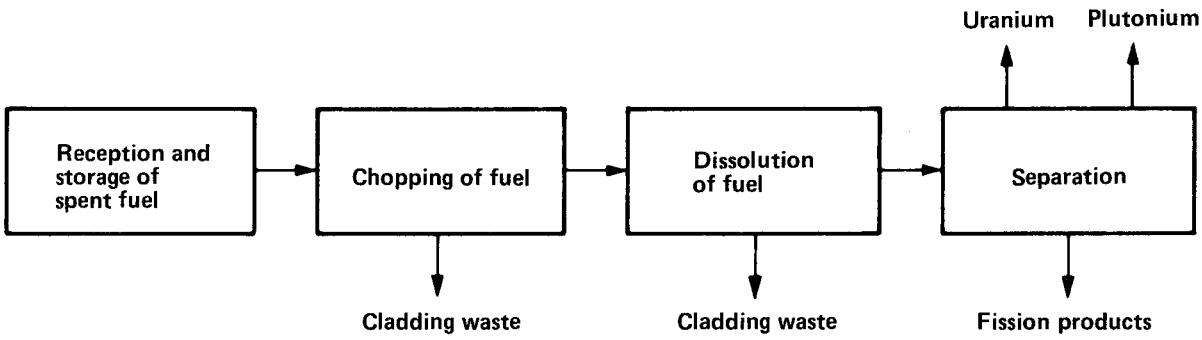



*Figure 5-1. Flow scheme of the various stages of the reprocessing of spent nuclear fuel from light-water reactors.*

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,

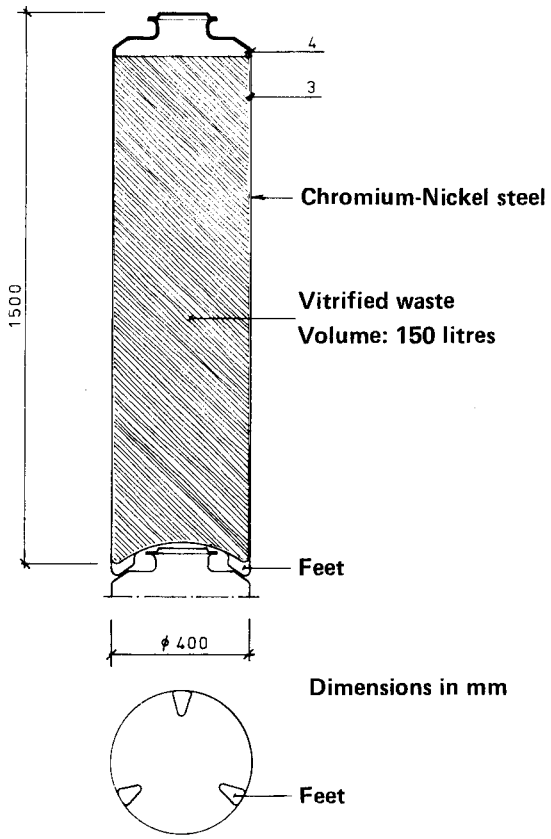



*Figure 5-2. Waste cylinder. The vitrified waste is cast in a container made of chromium-nickel steel. The container is sealed with a welded-on lid. The feet enable the waste cylinders to be stacked on top of each other.*

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.

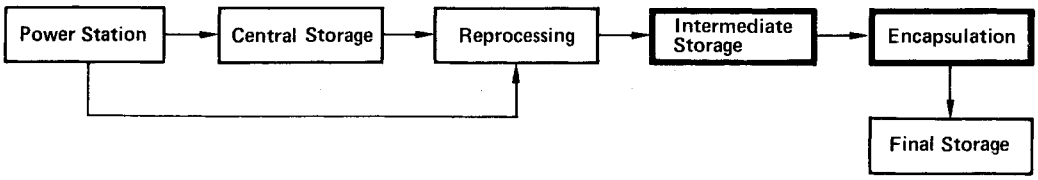
### 5.4.5 Resistance of borosilicate glass to radiation

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 onehalf. 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).

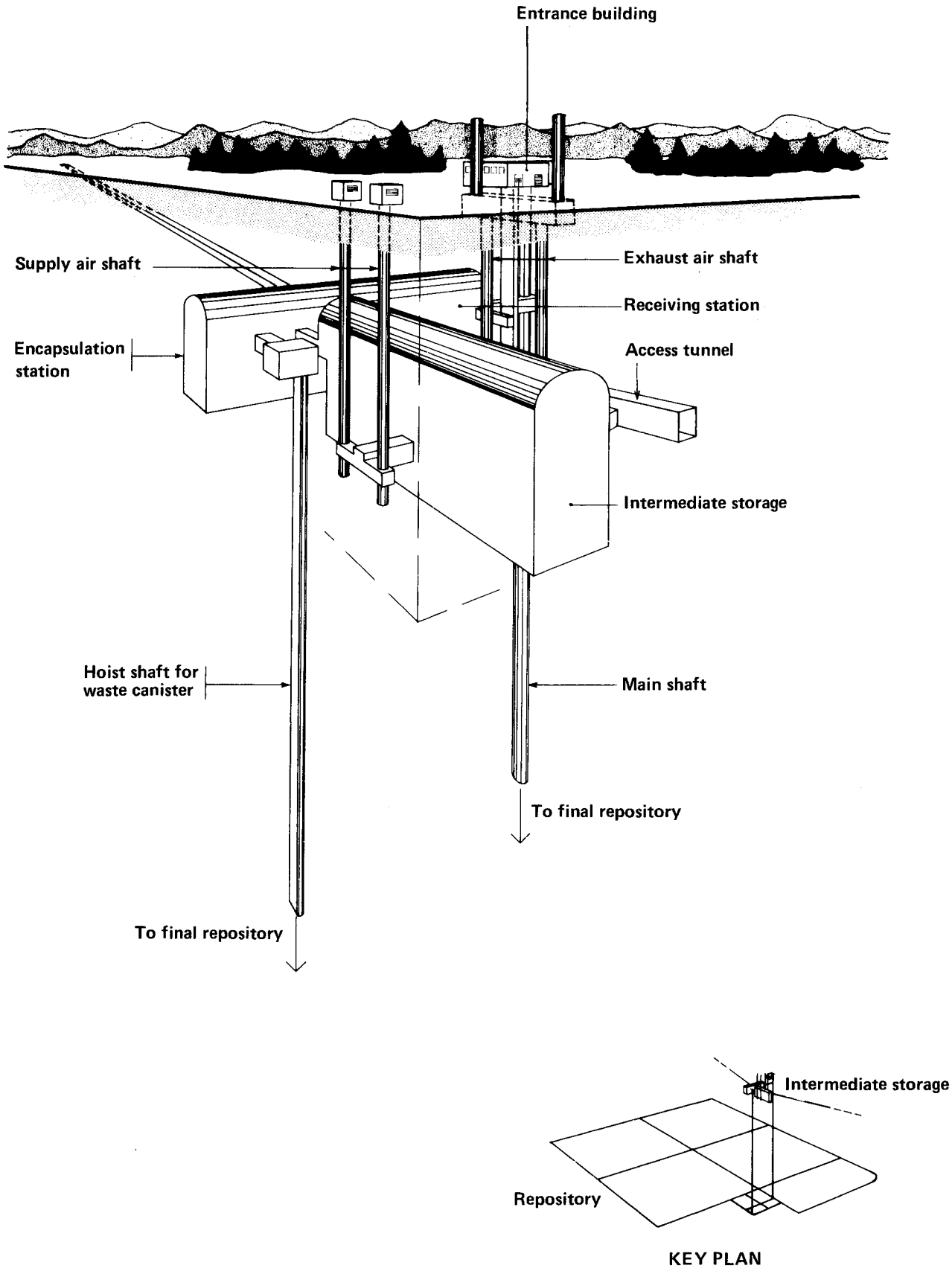



*Figure 6-1. Perspective drawing of plant for intermediate storage and encapsulation. It is located underground with a rock cover approximately 30 metres thick. The plant is located above the final repository.*

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

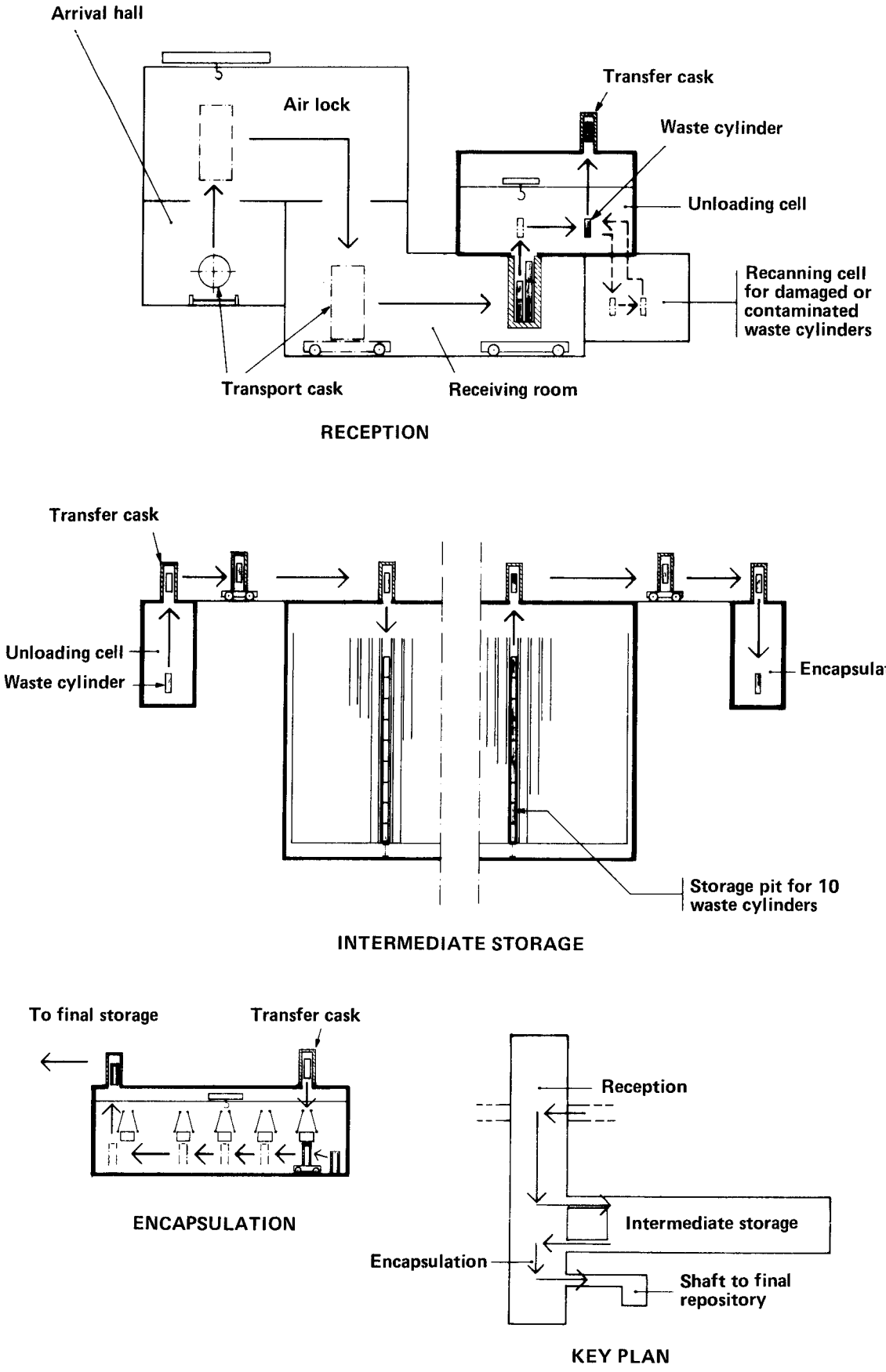



*Figure 6-2. Reception and intermediate storage. The transport casks arrive in the reception section and the waste cylinders are unloaded. Damaged or contaminated cylinders are encased in an outer container of chromium-nickel steel. The waste cylinders are transferred to intermediate storage inside a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

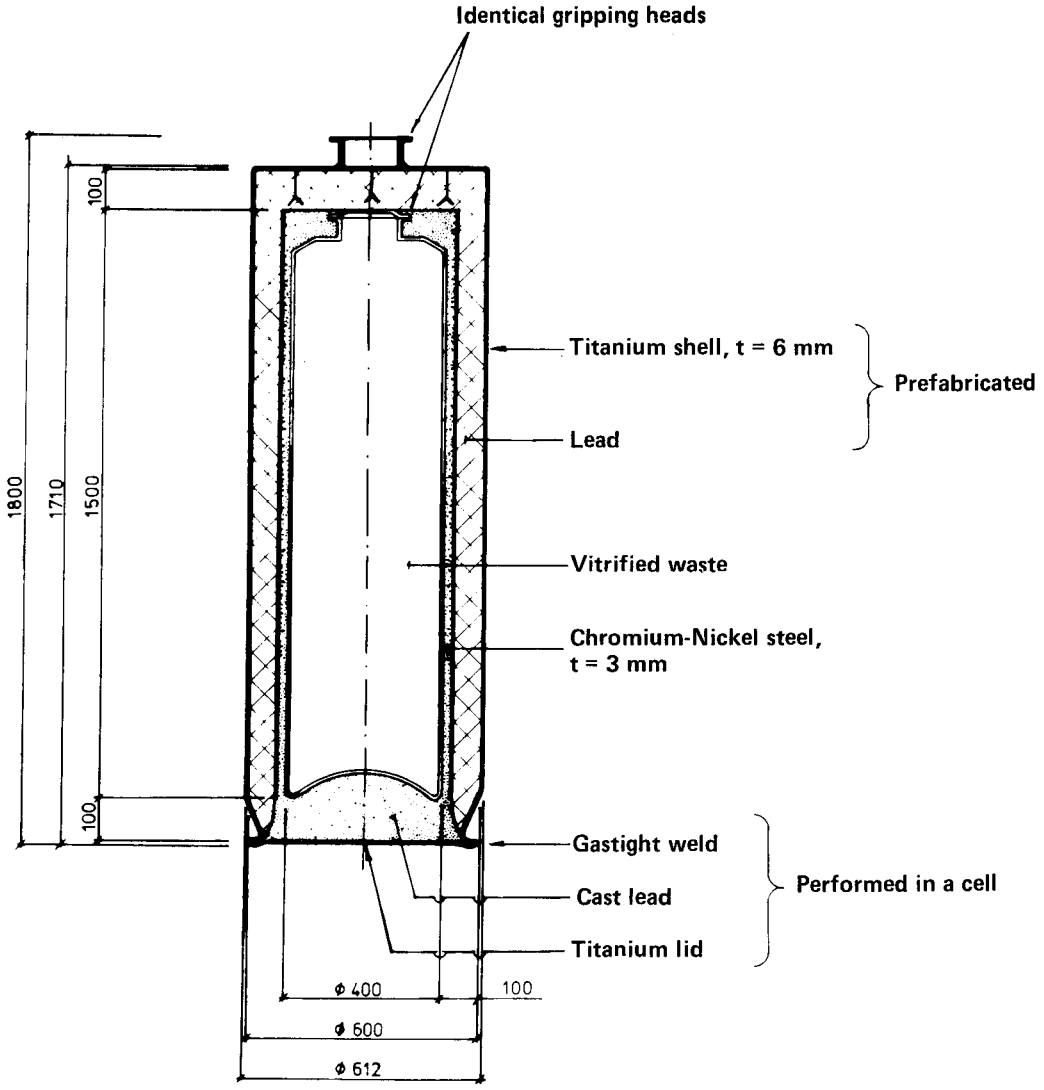



*Figure 6-3. Lead-titanium canister. Waste cylinder with vitrified high-level waste in a canister of lead and titanium. Total weight approx. 3900 kg.*

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

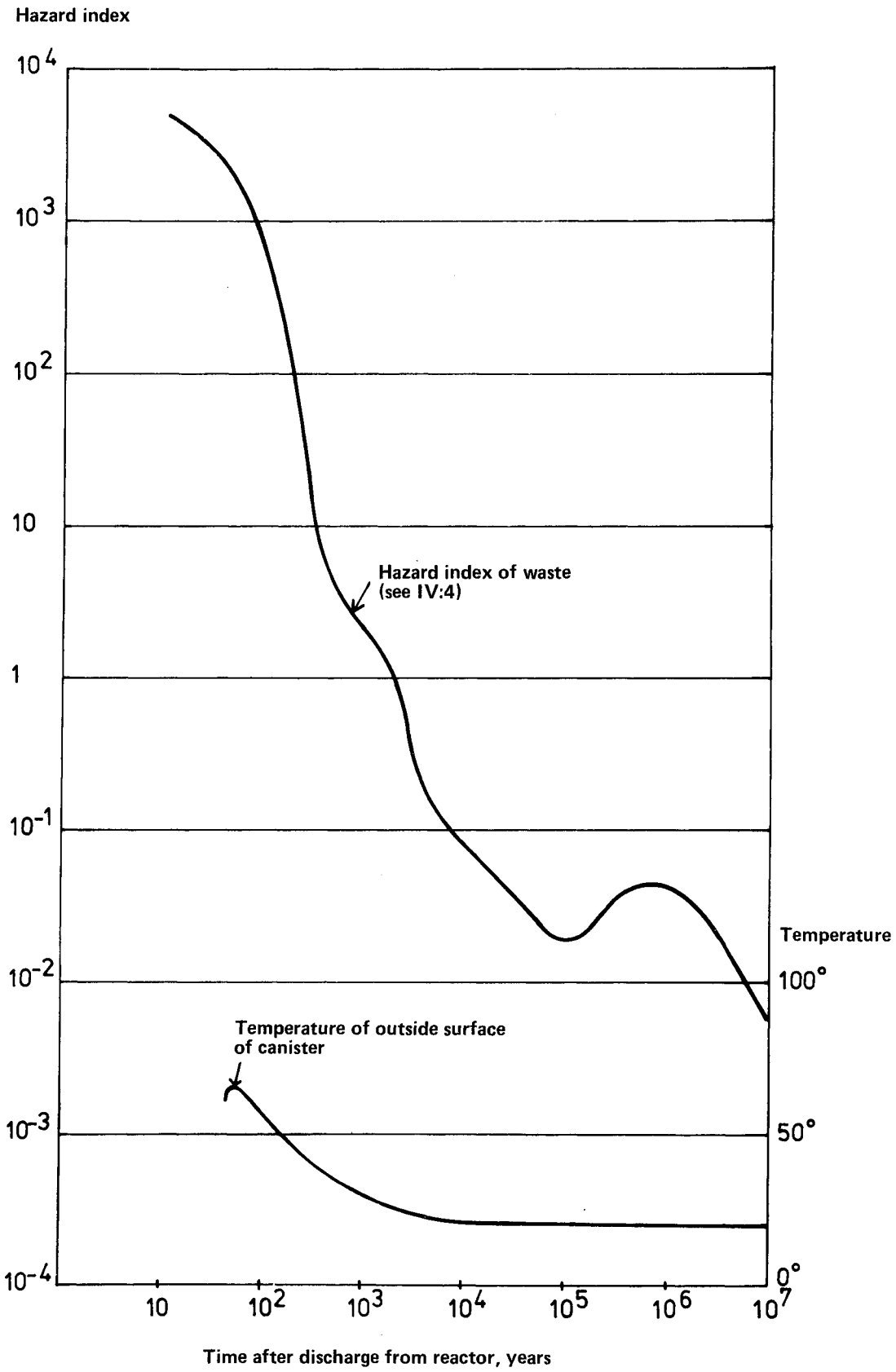



*Figure 6-4. Graph showing how the toxicity of the vitrified waste (hazard index) and the external surface temperature of the canister vary with time. Note that the hazard index and time scales are logarithmic.*

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,

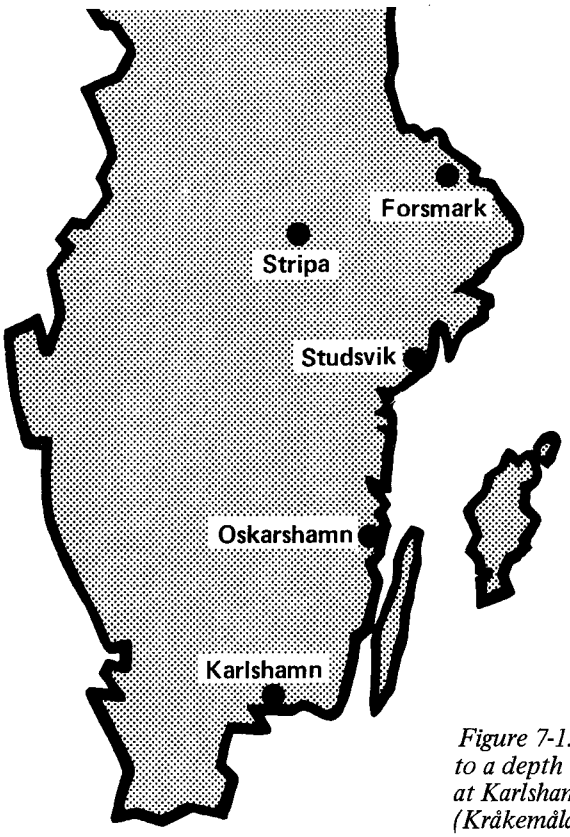



*Figure 7-1. Map of study sites. Test drillings down to a depth of about 500 m have been undertaken at Karlshamn (Sternö), north of Oskarshamn (Kråkemåla and Ävrö) and Forsmark (Finnsjö Lake and Forsmark). The KBS experimental station is located in the Stripa mine. Field experiments have also been conducted 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 $50^{o}$ 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 $50^{o}$ 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 studies 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.

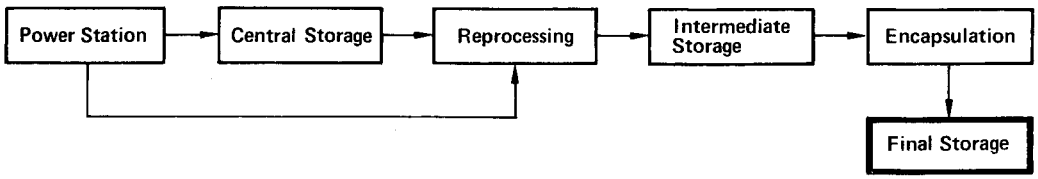
On the basis of existing knowledge, the coastal gneiss region of Blekinge is from a geological point of view, the most attractive area for a final repository.

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.

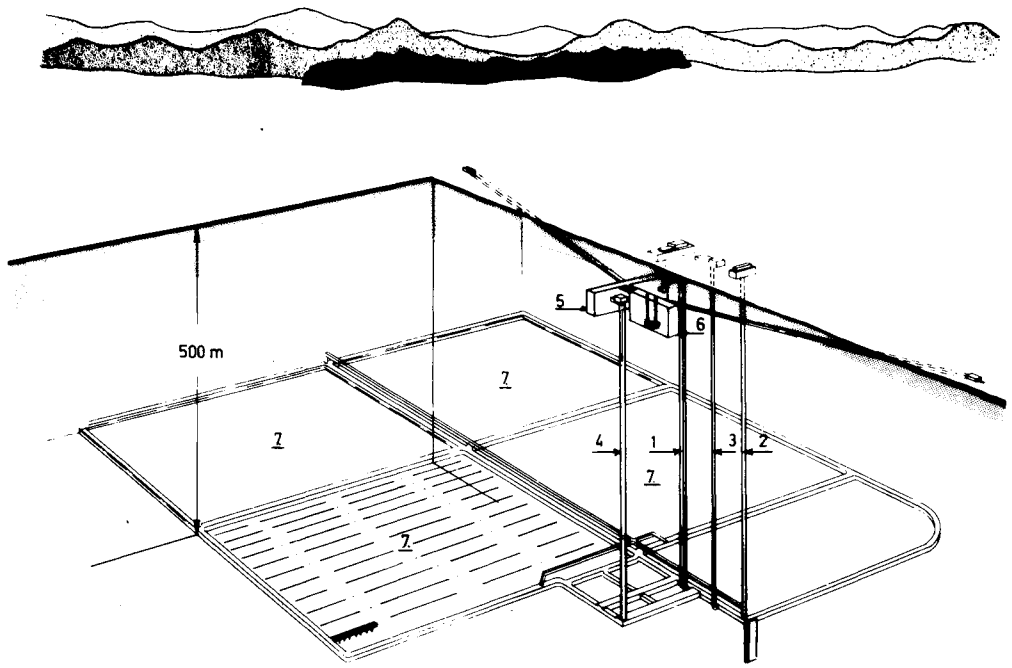
Thus, vitrification and encapsulation of the waste, the buffer material and the rock constitute four barriers which prevent or greatly retard the migration of radioactive elements via the groundwater to the biosphere.

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




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

*Figure 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.

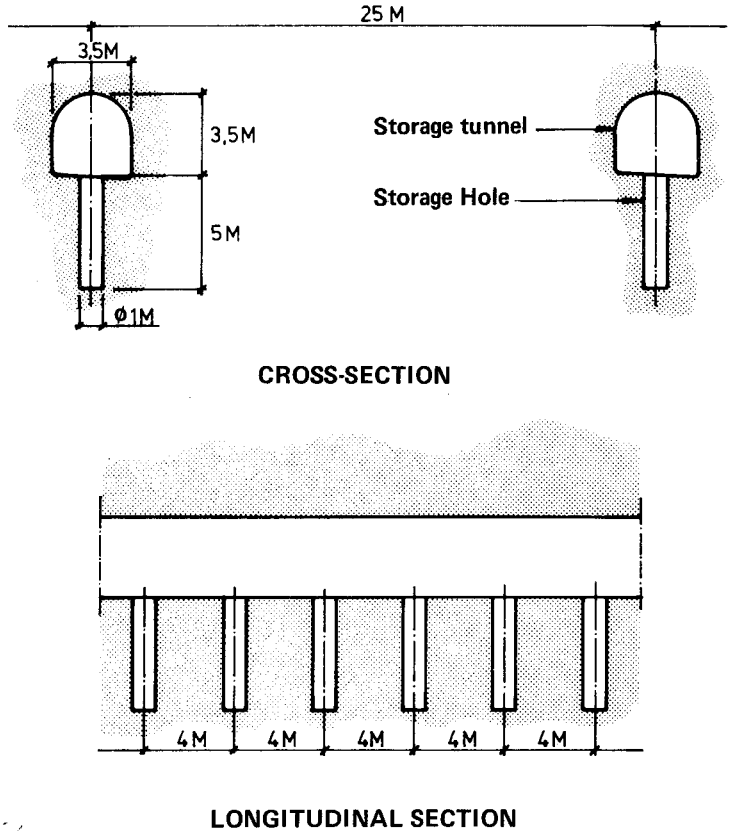



*Figure 8-2. Cross-section and longitudinal section of storage tunnels in the final repository. Each storage hole is designed for one canister.*

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

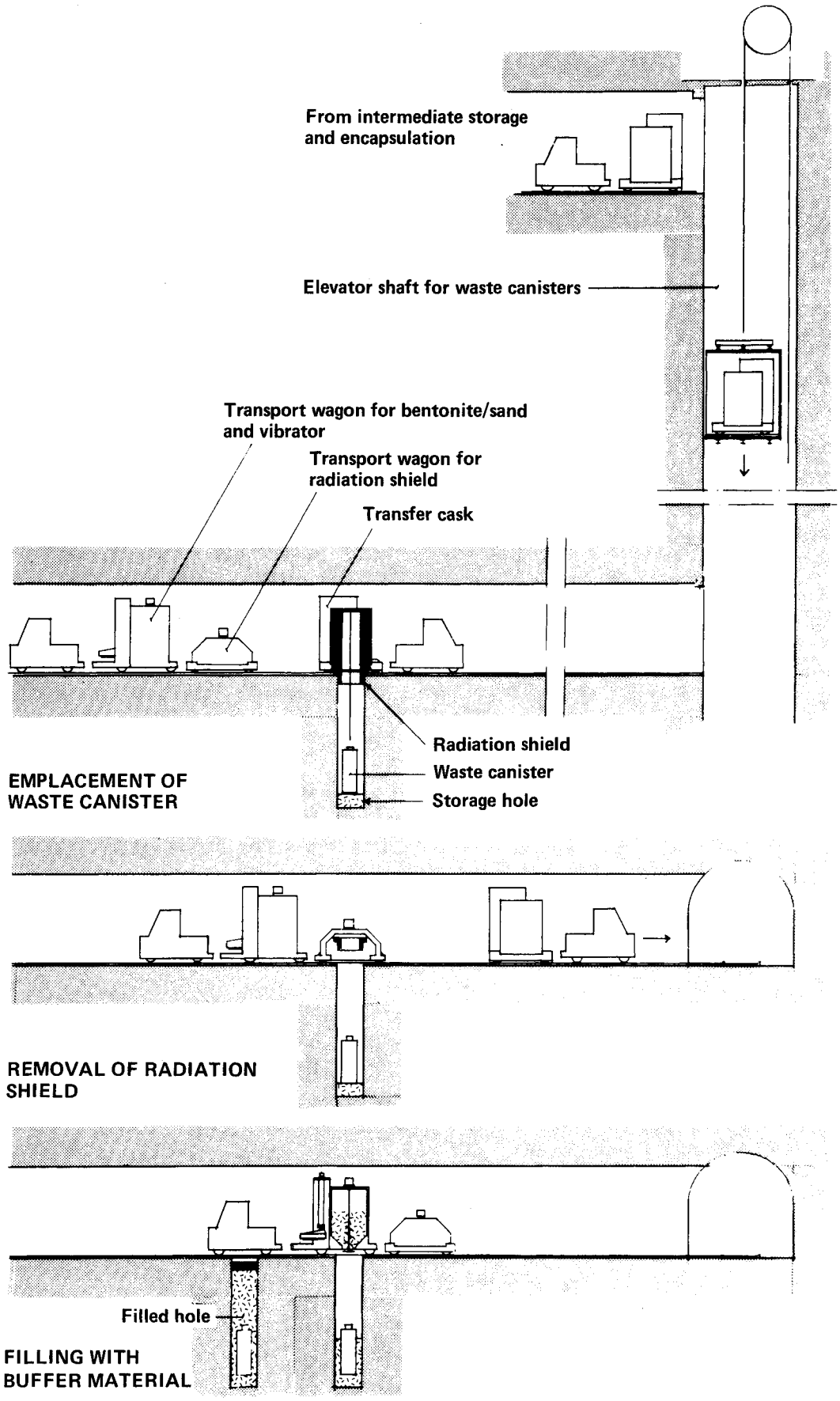



*Figure 8-3. Handling of waste canisters in final repository.*

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.

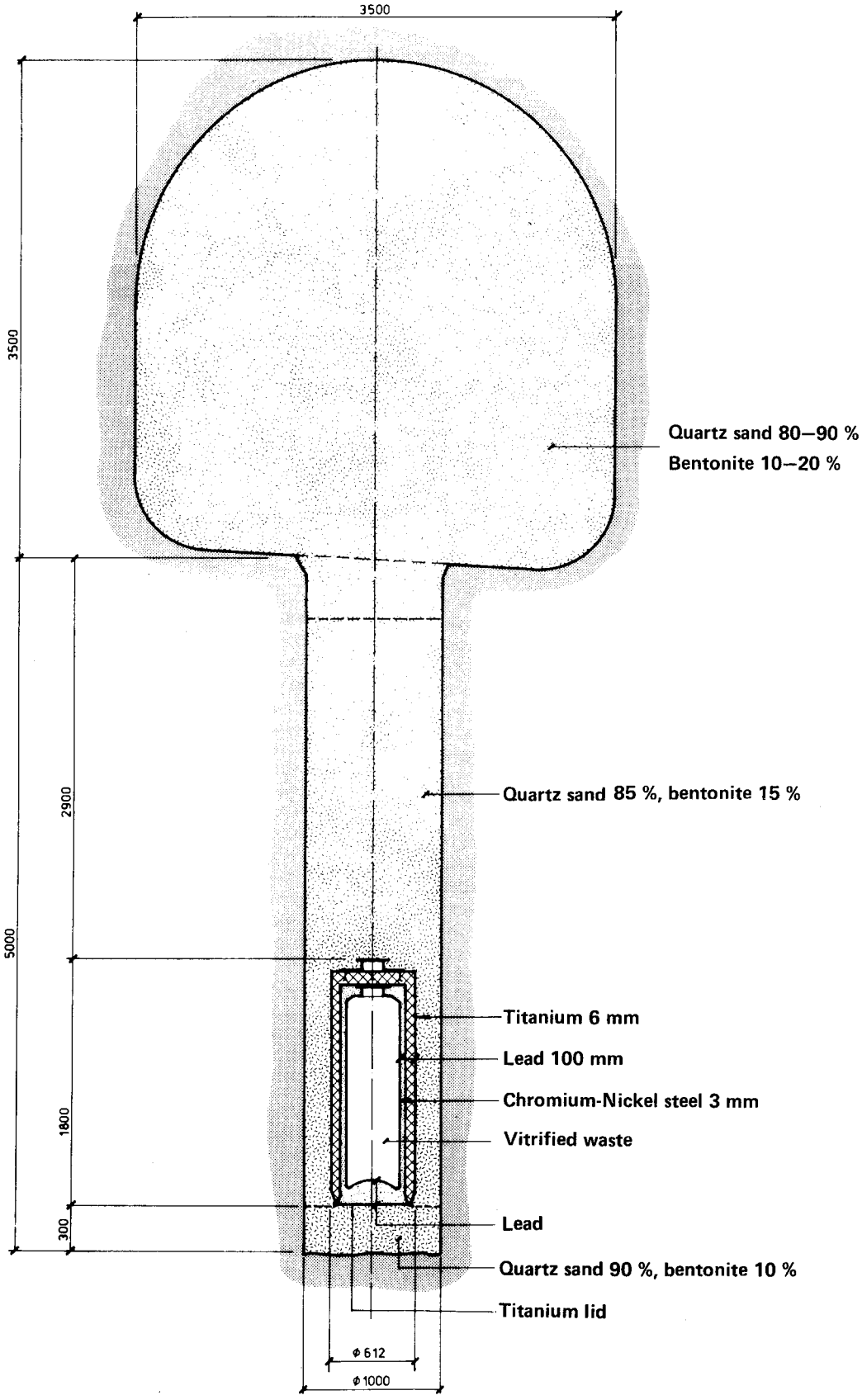
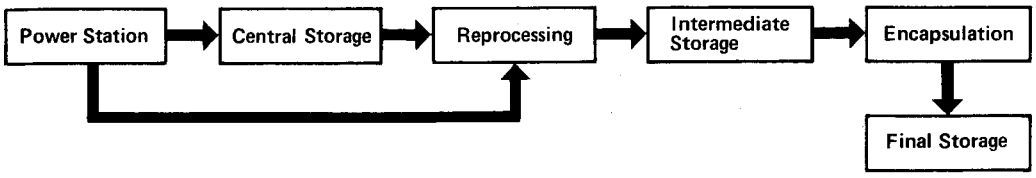



*Figure 8-4. The sealed final repository. Tunnels and storage holes are completely filled with a buffer material consisting of quartz sand and bentonite.*

# 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.

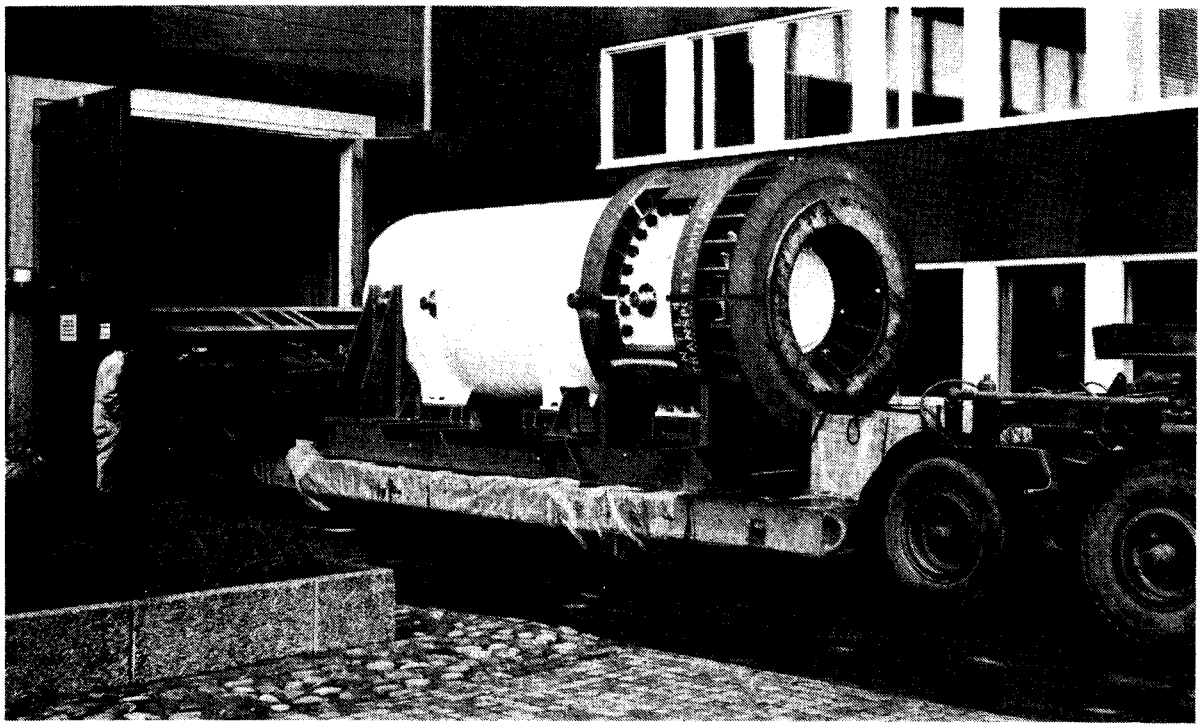


*Figure 9-1. Trailer and transport cask outside of the Oskarshamn plant. This equipment has been used to transport spent fuel to the harbour for further shipment to the reprocessing plant at Windscale.*

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.

# 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).

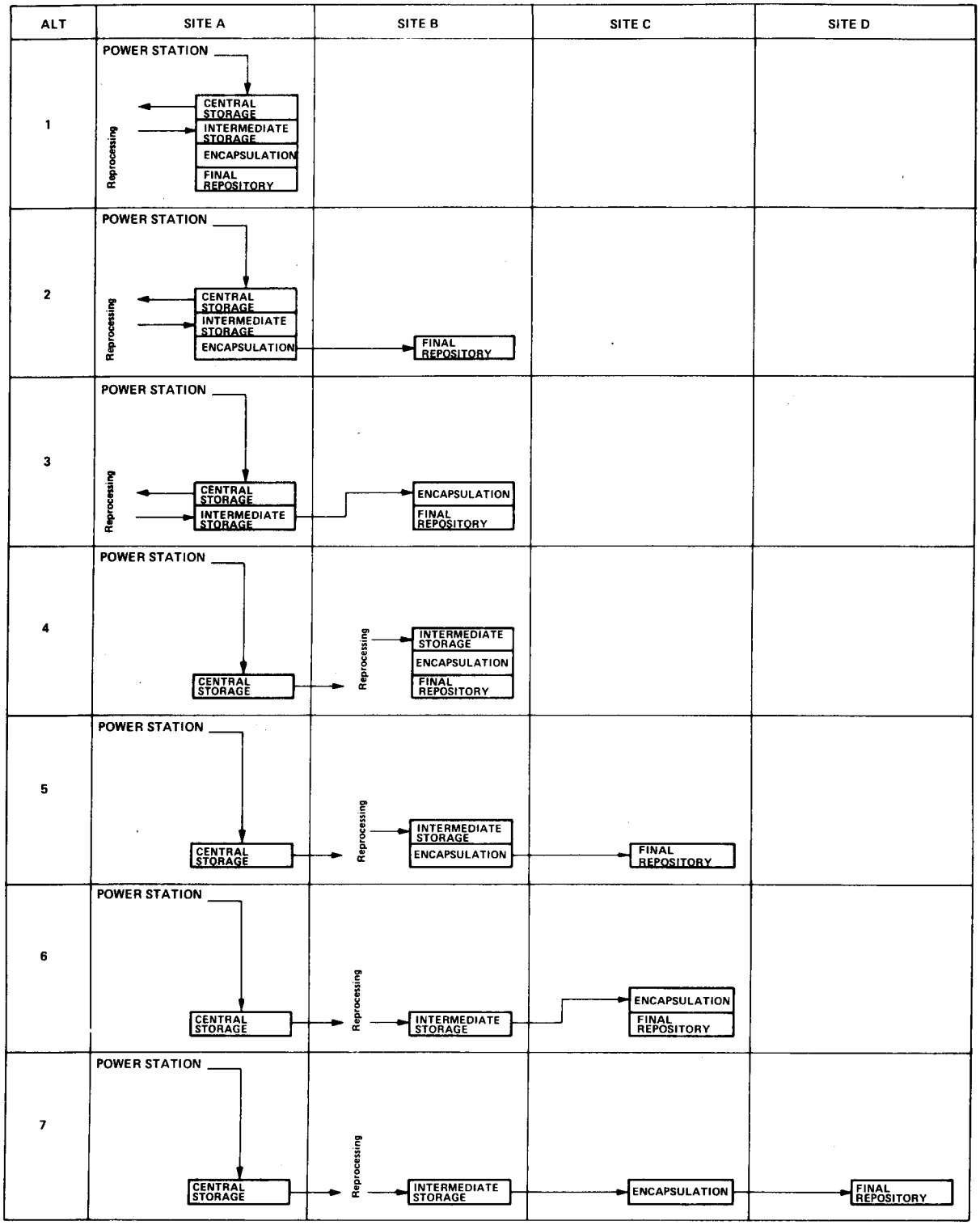



*Figure 11-1. Possible siting alternatives for the various facilities for the storage of spent fuel and vitrified waste.*

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 altogether.

## 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

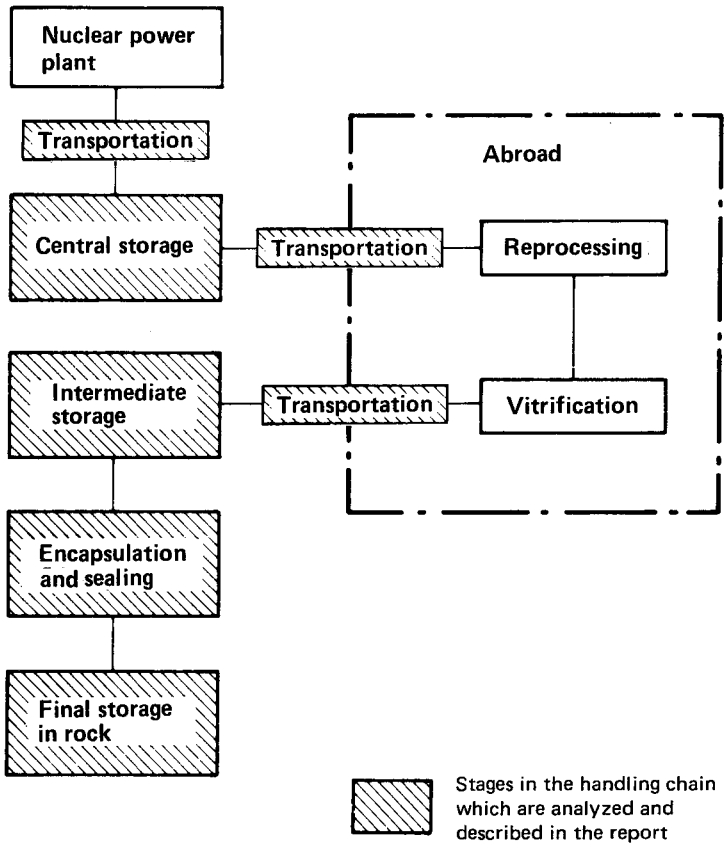


*Figure 13-1. Handling chain for storage of spent fuel and vitrified waste.*

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$^{o}$C normally and below 50$^{o}$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 $3 \times 10^{-6}$ 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

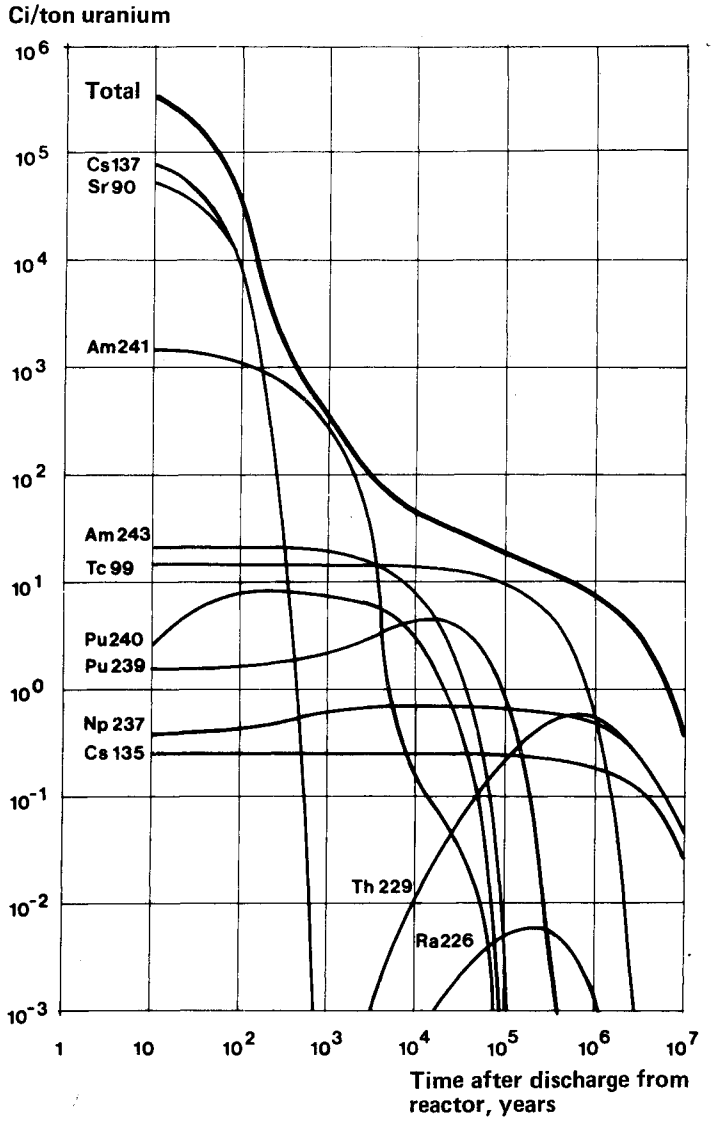


*Figure 13-2. Radioactive elements in high-level waste. It is assumed that reprocessing takes place ten years after discharge of the spent fuel from the reactor.*

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.

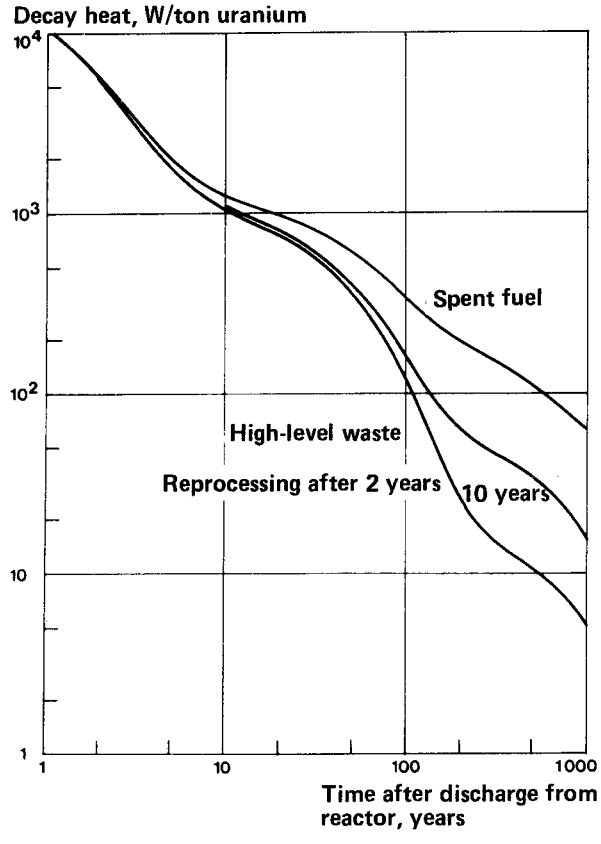


*Figure 13-3. Decay heat in spent fuel and high-level waste from pressurized water reactor.*

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 $2 \times 10^{-7}$ 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 $1.7 \times 10^{-5}$ (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.

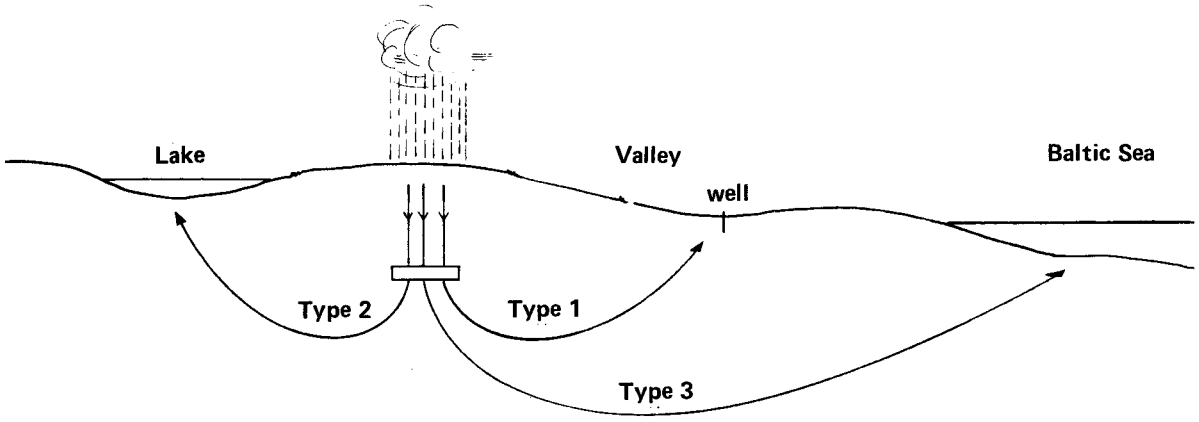


*Figure 13-4. The three main cases of transport of radioactive substances to the biosphere.*

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.

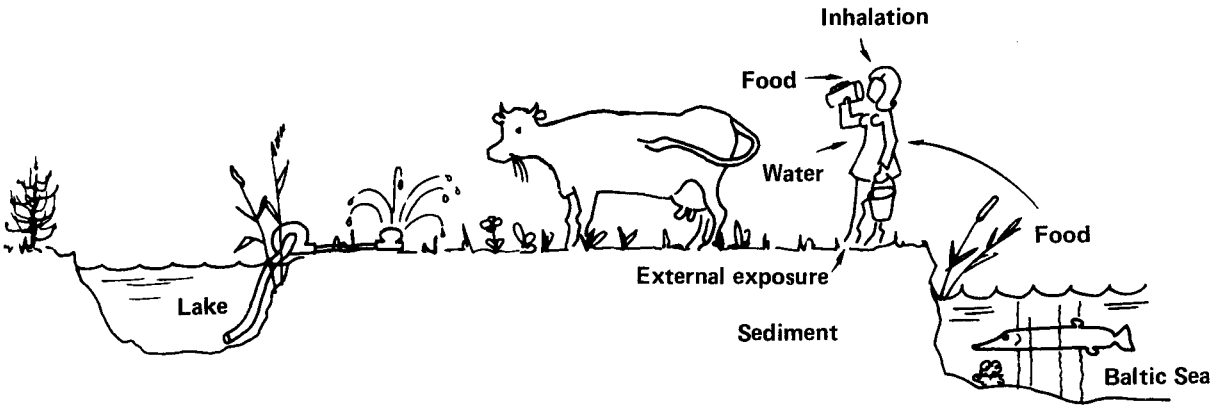


*Figure 13-5. Paths of human exposure in the local ecosystem.*

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.

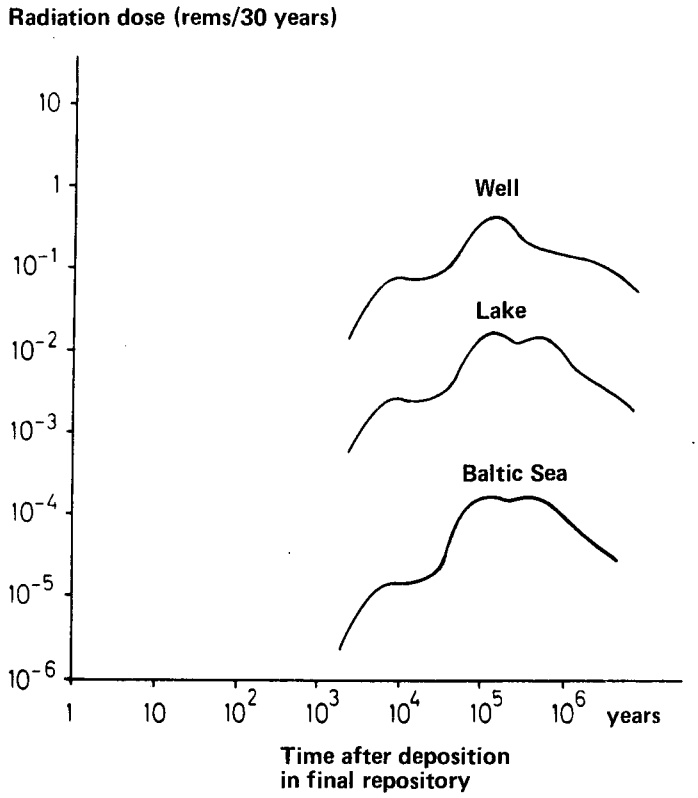


*Figure 13-6. Calculated maximum individual doses to critical group (nearby residents) for various primary recipients.*

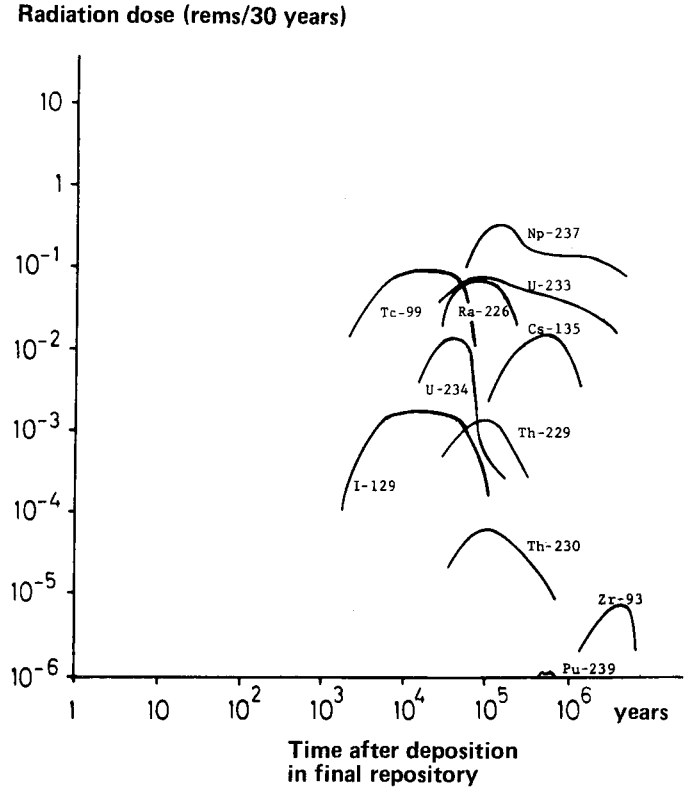


*Figure 13-7. Calculated maximum individual doses to critical group (nearby residents) from different nuclides. The calculations were made for a well as the recipient.*

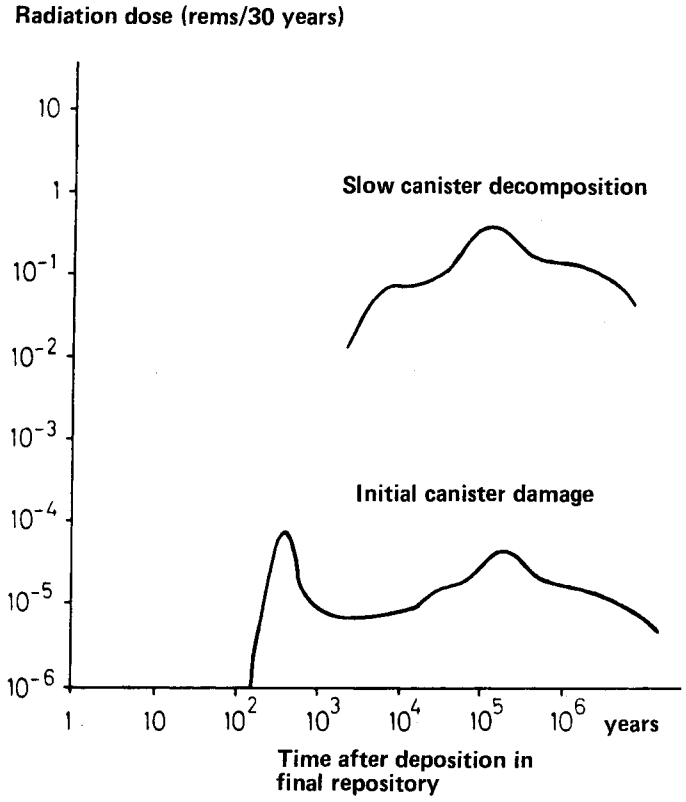


*Figure 13-8. Calculated maximum individual doses to critical group (nearby residents) for the two main cases of canister damage. The cases are a) slow decomposition of all canisters during the period from 1000 to 6000 years, and b) initial damage to one canister. The primary recipient is a well.*

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.

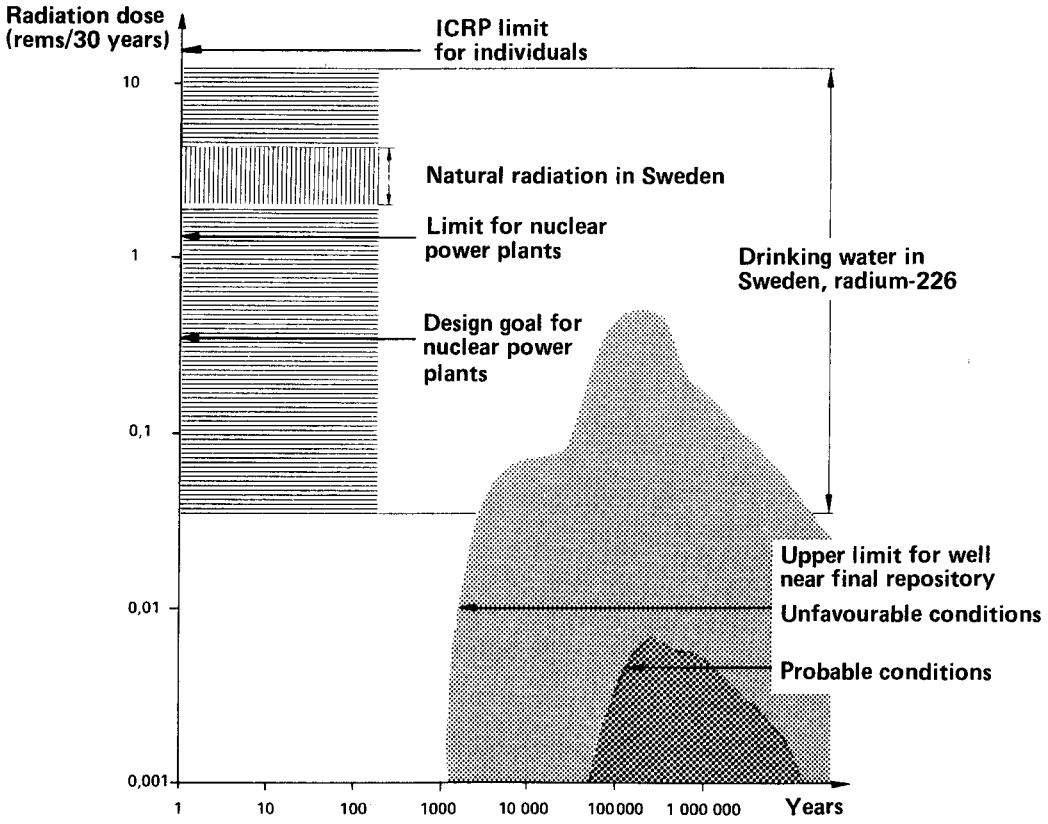


*Figure 13-9. Calculated upper limit for radiation doses to people who live near the final repository (critical group). The calculations pertain to the slow decomposition of the canister with a well as the primary recipient. For purposes of comparison, the dose load from several natural radiation sources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

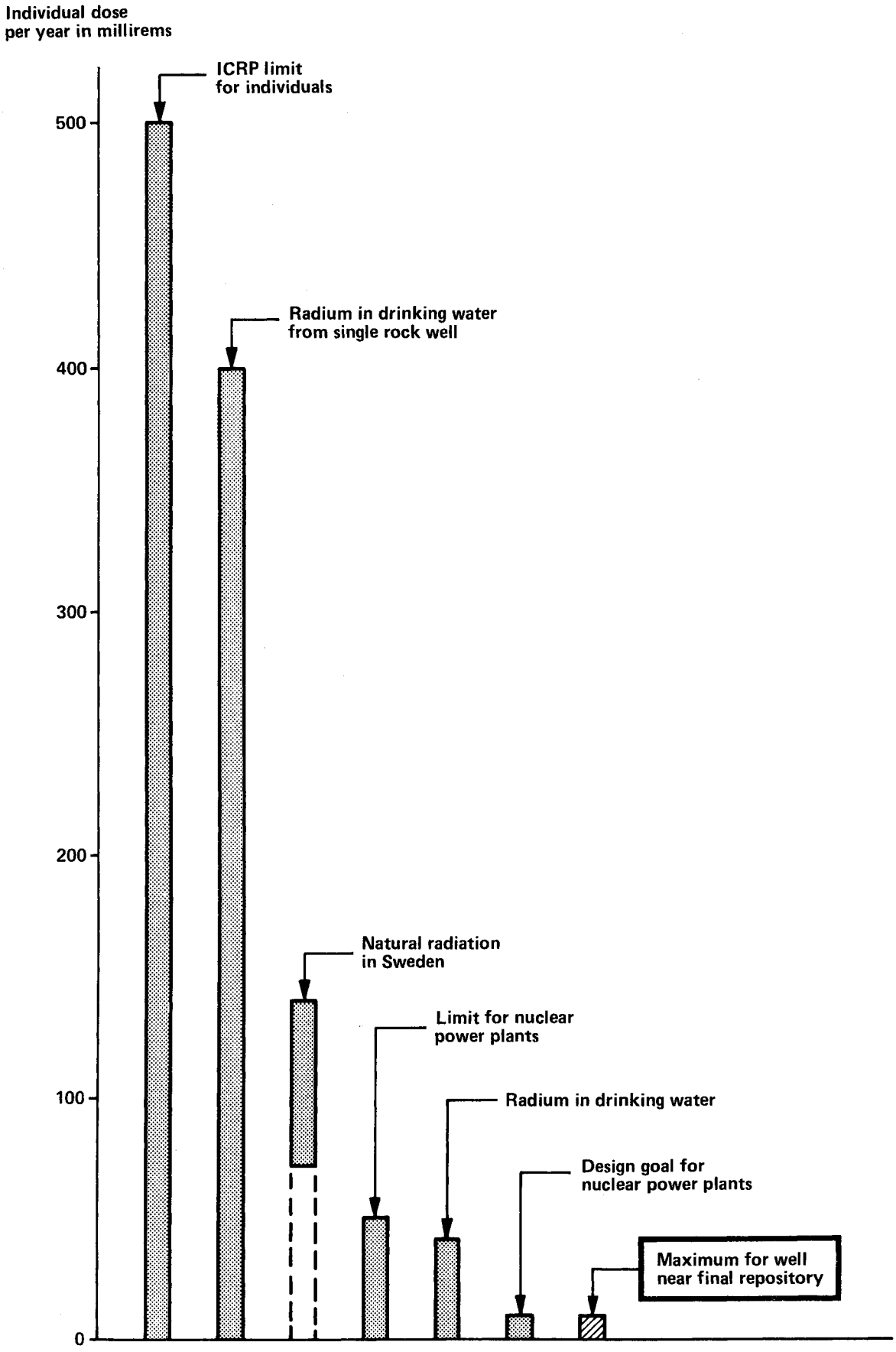


*Figure 13-10. Bar graph showing the calculated maximum annual radiation doses which the final repository can give to a nearby resident and the annual dose to man from some natural radiation sources plus some established dose limits. The dose from drinking water comes from radium-226.*

| 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 organization 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

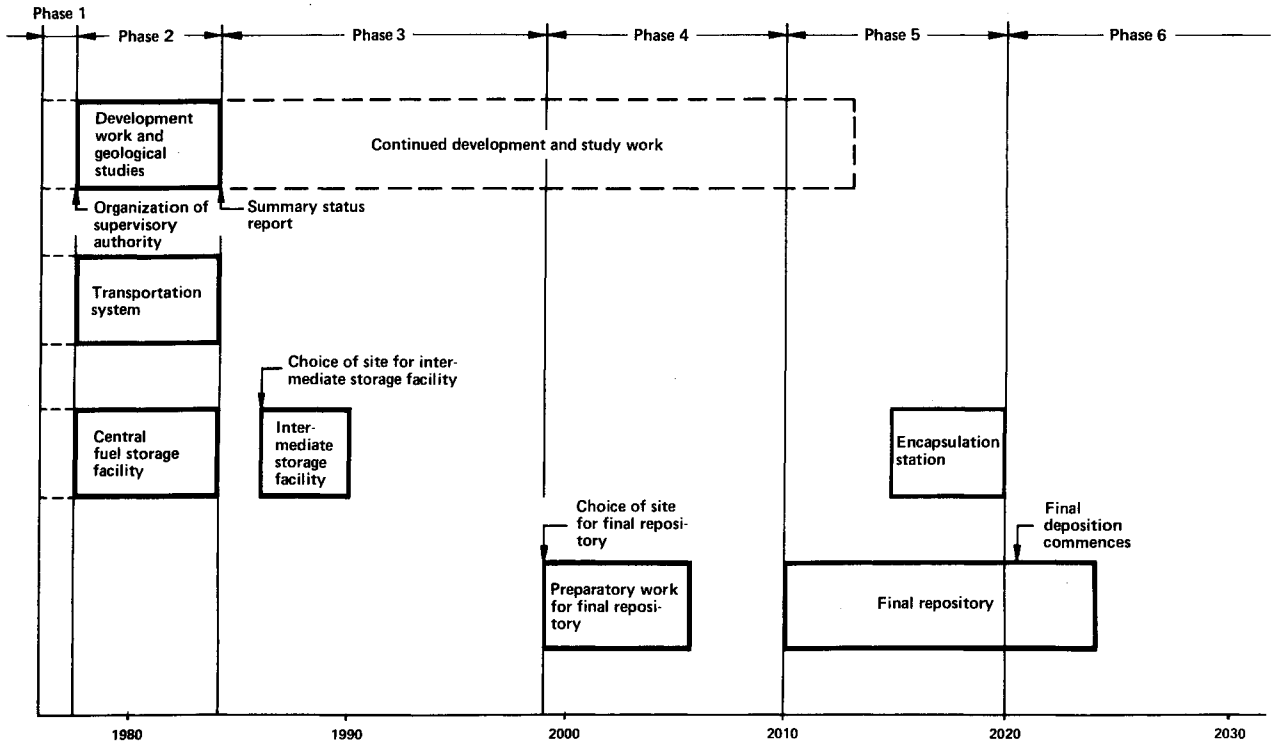


*Figure 14-1. Skeleton plan for continued work on the final stages of the nuclear fuel cycle.*

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.

### 3.2 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 $100^{\circ}C$, aluminium hydroxide is thereby obtained in amorphus or crystalline form. Crystalline aluminium hydroxide occurs in

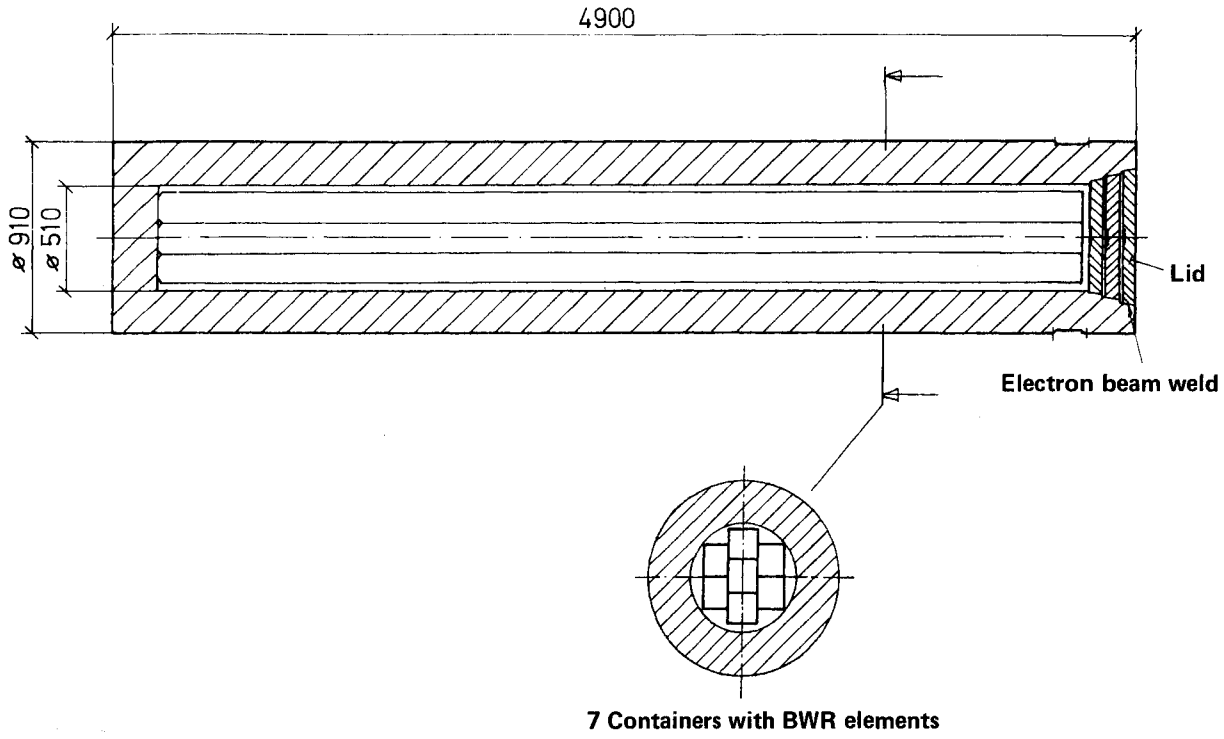



*Figure B1-1. Copper canister for direct disposal of spent nuclear fuel. The canister has 7 containers with fuel elements from a boiling water reactor. The weight of the canister without fuel is about 20 metric tons.*

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.

## 3.4 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

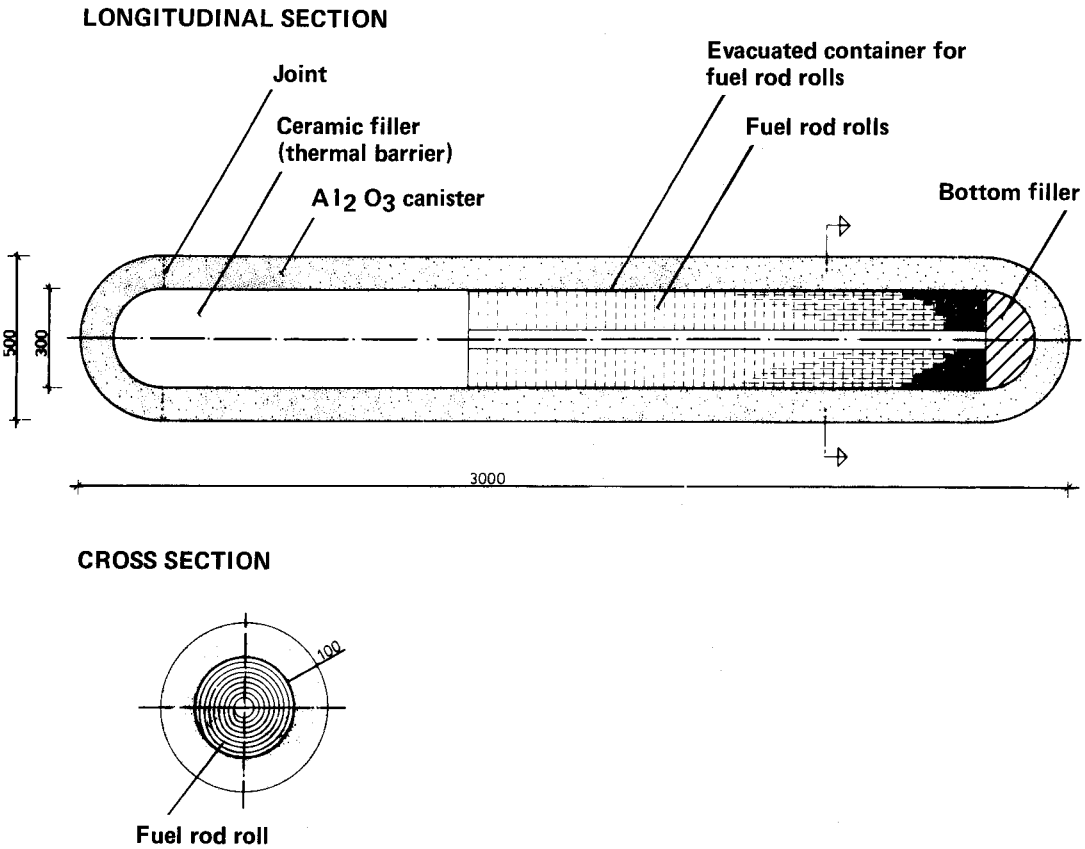



*Figure B1-2. Ceramic canister of aluminium oxide. The fuel rods are rolled into spirals prior to placing in the canister.*

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,

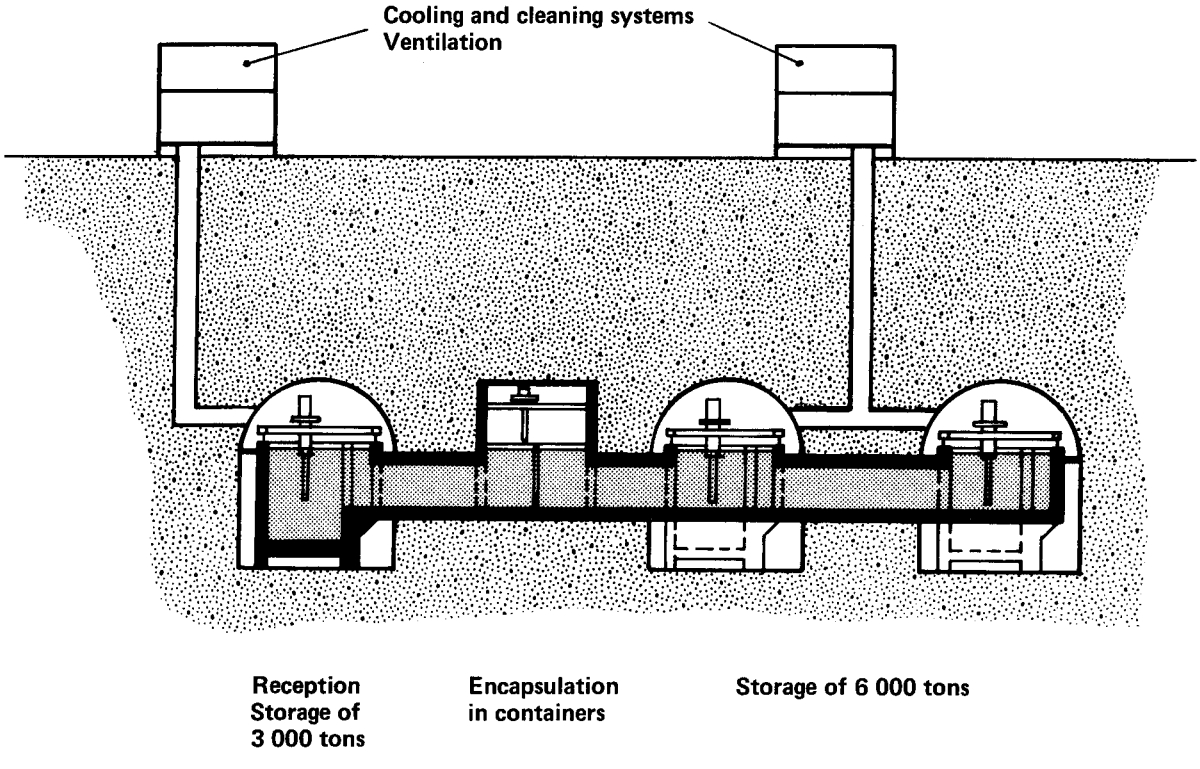



*Figure B1-3. Intermediate storage facility for spent fuel.*

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-

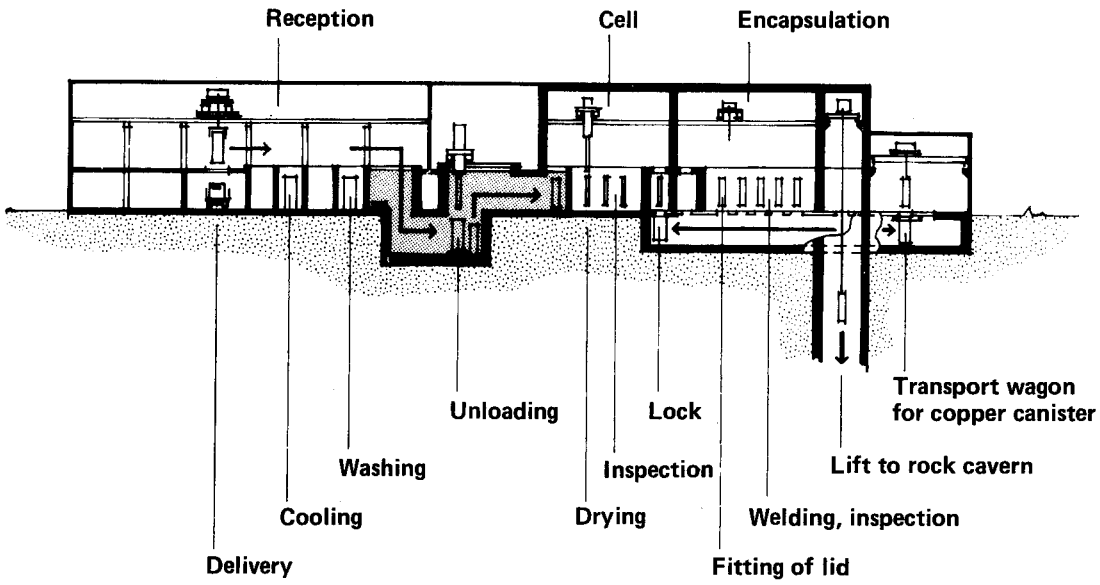
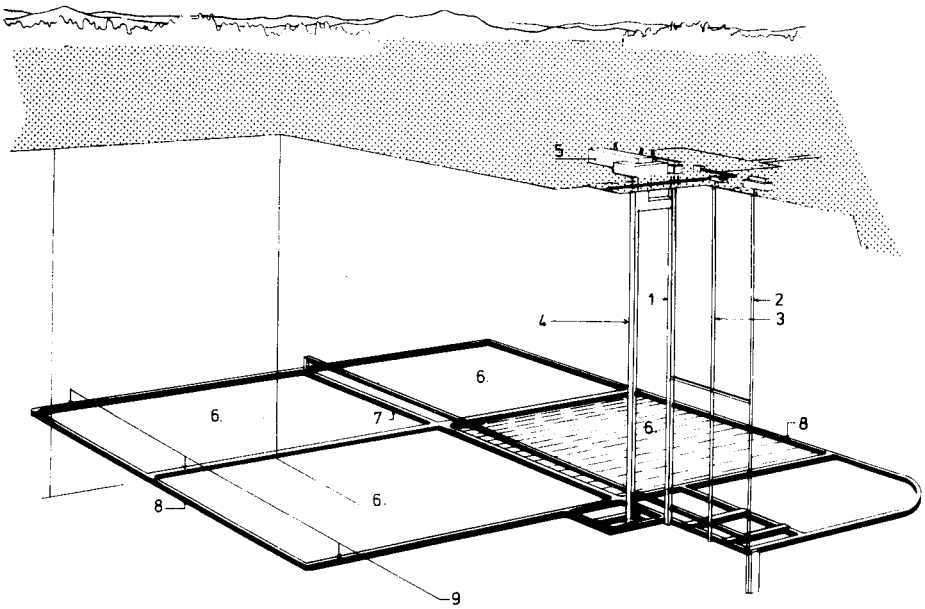



*Figure B1-4. Encapsulation station for direct disposal of spent fuel.*




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.*

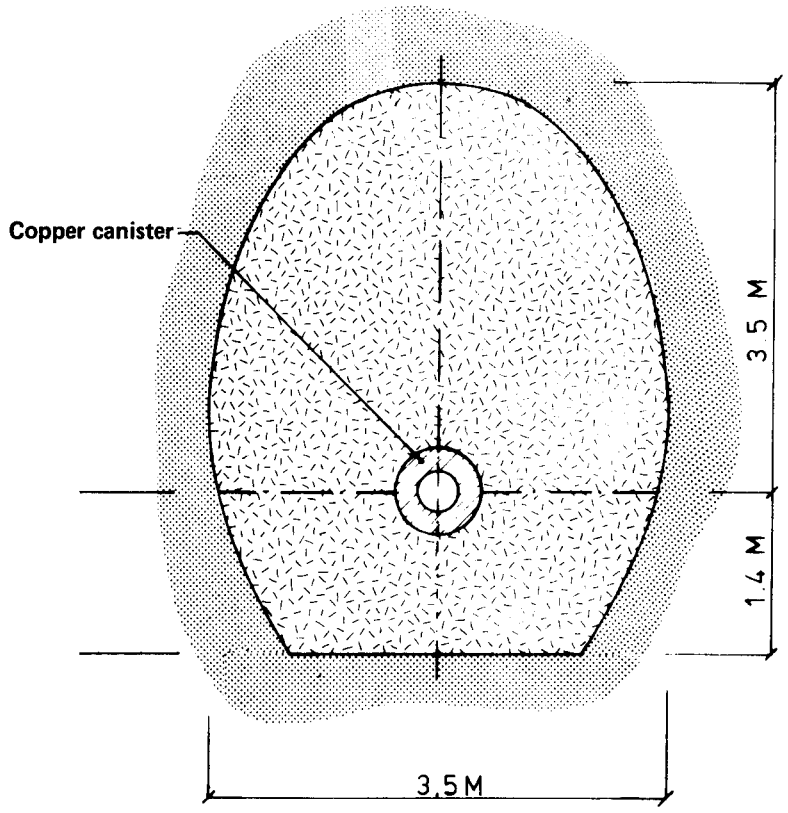



*Figure B1-6. Storage tunnel for direct disposal of spent fuel. The copper canister is placed horizontally. Buffer material is placed in layers to a height of 1.4 m in the tunnel and compacted. The top part of the tunnel is filled by spraying of buffer material. (Cf. figure B1-7).*

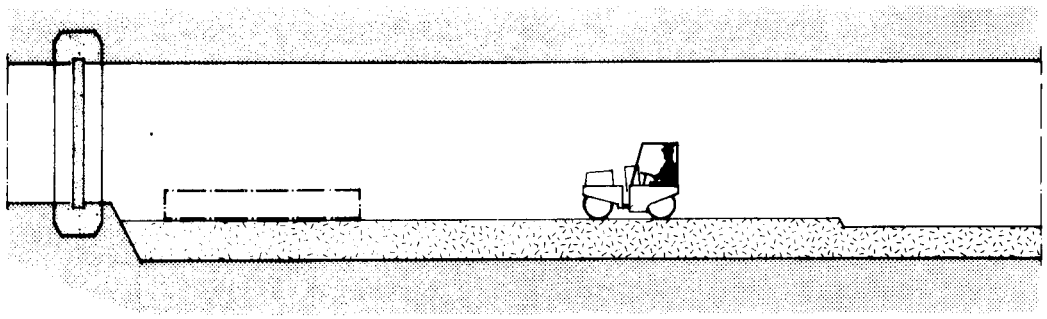
ner in a storage tunnel, the tunnel is filled completely with sand/bentonite, which is applied by spraying. The ends of the storage tunnel are sealed with concrete walls.

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.

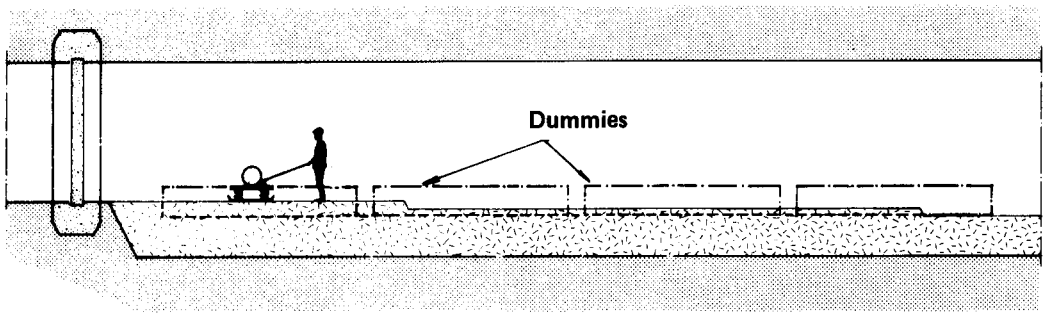
When the entire repository has been filled with canisters, transport tunnels, shafts and other cavities in the rock are sealed in the same way as was described for the vitrification alternative in section III:6.7.

## 5 TRANSPORTATION SYSTEM

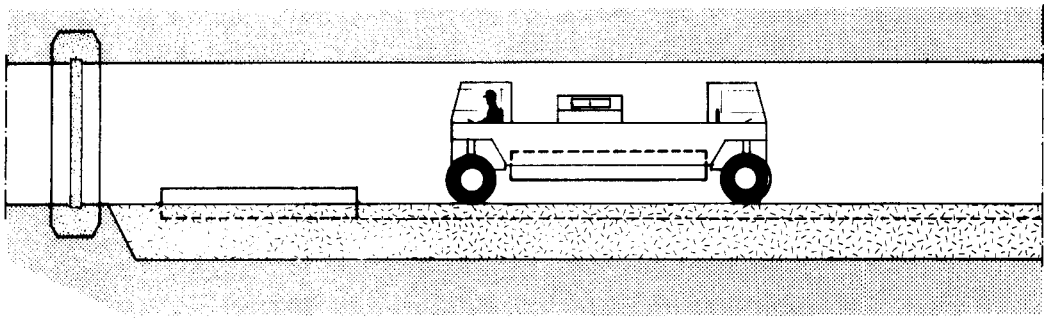
The same transportation system as is used for transporting spent fuel from the nuclear power plants to the central storage facility can be used for transporting the fuel from the intermediate storage facility to the encapsulation station at the final repository. The required number of transport casks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



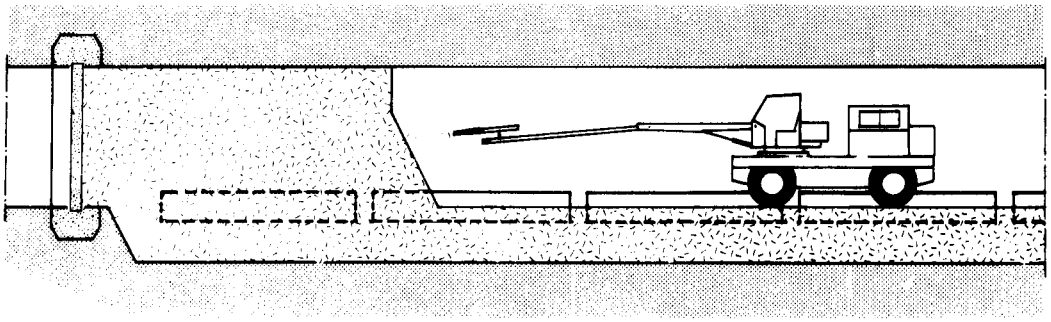
Laying of under bed




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 fision 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 çanister 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

University of Lund

| | |
|---|---|
| Division of quarternary geology | E Lagerlund, Ph.D. |

University of Gothenburg

| | |
|---|---|
| Division of marine microbiology | Professor K Gundersen |

University of Stockholm

| | |
|---|---|
| Department of geology | Dr. N-A Mörner<br>T Flodén, Ph.L. |
| Department of microbial geochemistry | Dr. R Hallberg |
| Department of statistics | Dr. T Thedéen |

University of Uppsala

| | |
|---|---|
| Department of Seismology | Dr. O Kulhánek |

Lund Institute of Technology

| | |
|---|---|
| Department of inorganic chemistry | Dr. Sten Andersson |

Chalmers University of Technology

| | |
|---|---|
| Department of geology | L Bergström, Ph.D. |
| Department of nuclear chemistry | Professor J Rydberg<br>B Allard, D.Eng. |
| Department of engineering metals | Dr. I Olefjord |
| Department of inorganic chemistry | Professor N-G Vannerberg |

Royal Institute of Technology

| | |
|---|---|
| Department of rock engineering | Professor I Janelid |
| Department of physical chemistry | Dr. C Leygraf |
| Department of geodetics | Professor A Bjerhammar |
| Department of chemical engineering design | Professor I Neretnieks |
| Department of agricultural hydrotechnics | Professor Y Gustafsson<br>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

| | |
|---|---|
| Division of rock mechanics | Professor O Stephansson<br>Dr. K Röshoff |
| Division of geotechnics | Professor R Pusch<br>A Jacobsson, Ph.D. |

OTHER INSTITUTIONS

| | |
|---|---|
| Cement and Concrete Research Institute, Stockholm | Professor S G Bergström |
| National Defence Research Institute, Stockholm | G Walinder |
| Glass Research Institute, Växjö | B Simmingsköld, Lic. Eng.<br>T Lakatos, Ph.D. |

| | |
|---|---|
| Institute for Metals Research | G Eklund, D.Eng. |
| Corrosion research Institute, Stockholm | Professor E Mattsson<br>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<br>U Thoregren, Ph. L. |

EXPERTS AND RESEARCH WORKERS

Dr. Sten G A Bergman
Claes Helgesson, D. Eng.

FOREIGN ORGANISATIONS AND COMPANIES

The Risö Research Station
ROSKILDE, Denmark

Royal Norwegian Council for
Scientific and Industrial Research
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
WASHINGTON D.C., USA

Dr. Ralph E Grim
URBANA, Illinois, USA

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:

GEOGROUP

| | |
|---|---|
| Dr. Sten G A Bergman | Stocksund |
| Dr. Otto Brotzen | Geological Survey of Sweden |
| Professor Y Gustafsson | Royal Institute of Technology |
| Dr. Rudolf Hiltscher | Saltsjö-Boo |
| Professor Ingvar Janelid | Royal Institute of Technology |
| Dr. U Lindblom | Hagconsult |
| Professor Roland Pusch | Luleå Institute of Technology |
| Professor Ove Stephansson | Luleå Institute of Technology |

REFERENCE GROUP FOR SAFETY ANALYSIS

| | |
|---|---|
| Stig Bergström, M. Eng. | AB Atomenergi |
| Professor Arne Bjerhammar | Royal Institute of Technology |
| Professor Lars Fredriksson | Ultuna College of Agriculture |
| Tore Nilsson, M. Eng. | Swedish Nuclear Power Inspectorate |
| Professor Jan Rydberg | Chalmers University of Technology |
| Bo Simmingsköld, Lic. Eng. | Glass Research Institute |
| Jan Olof Snihs, Ph. L. | National Institute of Radiation Protection |
| Dr. Thorbjörn Thedéen | University of Stockholm |
| Dr. Gunnar Walinder | National Defence Research Institute 2 |

REFERENCE GROUP FOR CORROSION

| | |
|---|---|
| Roger Carlsson, Lic. Eng. | Swedish Silicate Research Institute |
| Chief Engineer Thomas Eckered | Nuclear Power Inspectorate |
| Lars Ekbom, Lic. Eng. | Corrosion Research Institute |
| Göran Eklund, D. Eng. | Institute for Metals Research |
| Professor Ingemar Grenthe | Royal Institute of Technology |
| Dr. Rolf Hallberg | University of Stockholm |
| Sture Henriksson, M. A. | AB Atomenergi |
| Professor Einar Mattsson | Corrosion Research Institute |

| | |
|---|---|
| Professor Nils-Gösta Vannerberg | Chalmers University of Technology |
| Professor Gösta Wranglén | Royal Institute of Technology |

BUFFER GROUP

| | |
|---|---|
| Bert Allard, D. Eng. | Chalmers University of Technology |
| Kåre Hannerz, Lic. Eng. | ASEA-ATOM |
| Arvid Jacobsson, Ph. D. | Luleå Institute of Technology |
| Professor Ivars Neretnieks | Royal Institute of Technology |
| Professor Roland Pusch | Luleå Institute of Technology |

PROGRAMME GROUP P21/22

| | |
|---|---|
| Alf Engelbrektsson, M. Eng. | AB Vattenbyggnadsbyrån (Swedish Hydraulic Engineering Company Ltd.) |
| A-M Eriksson, M. Eng. | Kemakta Konsult AB |
| Göran Fröman, M. Eng. | AB Vattenbyggnadsbyrån (Swedish Hydraulic Engineering Company Ltd.) |
| Bo Gustafsson, Eng. | SKBF |
| 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 products 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

52 Calculations of nuclide migration in rock and porous media penetrated by water
H Häggblom
AB Atomenergi, 1977-09-14

53 Mätning av diffusionshastighet för silver i lera-sand-blandning
("Measurement of rate of diffusion of silver in clay-sand mix")
Bert Allard
Heino Kipatsi
Chalmers University of Technology, 1977-10-15

54 Groundwater movements around a repository

54:01 Geological and geotechnical conditions
Håkan Stille
Anthony Burgess
Ulf E Lindblom
Hagconsult AB, September 1977

54:02 Thermal analyses
Part 1 Conduction heat transfer
Part 2 Advective heat transfer
Joe L Ratigan
Hagconsult AB, September 1977

54:03 Regional groundwater flow analyses
Part 1 Initial conditions
Part 2 Long term residual conditions
Anthony Burgess
Hagconsult AB, October 1977

54:04 Rock mechanics analyses
Joe L Ratigan
Hagconsult AB, September 1977

54:05 Repository domain groundwater flow analyses
Part 1 Permeability perturbations
Part 2 Inflow to repository
Part 3 Thermally induced flow
Joe L Ratigan
Anthony Burgess
Edward L Skiba
Robin Charlwood
Hagconsult AB, September 1977

54:06 Final report
Ulf Lindblom et al
Hagconsult AB, October 1977

55 Sorption av långlivade radionuklider i lera och berg
("Sorption of long-lived radionuclides in clay and rock")
Part 1 Determination of coefficients of distribution
Part 2 Review of the literature
Bert Allard
Heino Kipatsi
Jan Rydberg
Chalmers University of Technology 1977-10-10

56 Radiolys av utfyllnadsmaterial
("Radiolysis of filler material")
Bert Allard
Heino Kipatsi
Jan Rydberg
Chalmers University of Technology 1977-10-15

57 Stråldoser vid haveri under sjötransport av kärnbränsle
(Radiation doses in the event of a failure during the transport of nuclear fuel by sea")
Anders Appelgren
Ulla Bergström
Lennart Devell
AB Atomenergi

58 Strålrisker och högsta tillåtliga stråldoser för människan
("Radiation hazards and maximum permissible radiation doses for human beings")
Gunnar Walinder
AB Atomenergi

A number of additional reports are in preparation.