Handling and Final Storage of Unreprocessed Spent Nuclear Fuel

Volume

I General

II Technical

KÄRN -
BRÄNSLE -
SÄKERHET

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TABLE OF CONTENTS FOR GENERAL VOLUME

TABLE OF CONTENTS FOR TECHNICAL VOLUME

1 SUMMARY

Background

In April 1977, the Swedish Parliament passed a law called the "Stipulation Law" which stipulates that new nuclear power reactors may not be charged with fuel unless the owner is able to demonstrate that terminal storage of either the high-level waste from the reprocessing of spent fuel or unreprocessed spent fuel can be accomplished in an absolutely safe manner. In order to investigate these and related matters, the Swedish power utilities formed the Nuclear Fuel Safety Project (KBS), which submitted its first report entitled "Handling of spent nuclear fuel and final storage of vitrified high-level reprocessing waste" in December of 1977. On the basis of that report plus an agreement for the reprocessing of nuclear fuel, the power industry has, with reference to the Stipulation Law, applied to the Government for permission to charge two reactors (Ringhals 3 and Forsmark 1) with nuclear fuel.

The present second report from KBS describes how the safe final storage of spent unreprocessed nuclear fuel can be implemented.

The requirements of the Stipulation Law concerning description of the final storage method

According to the Stipulation Law, the owner must specify in which form the waste is to be stored, how final storage is to be effected, how the waste is to be transported and all other aspects of fuel handling and storage which must be taken into consideration in judging whether the proposed final storage method can be considered to be absolutely safe and feasible. Thus, the description must go beyond general plans and sketches.

The description is therefore relatively detailed, even concerning those parts which are less essential for evaluating the safety of the waste storage method. For those parts of the handling chain which are the same for both alternatives of the Stipulation Law, the reader is referred in some cases to the preceding report.

Both of the alternatives of the Stipulation Law may be used in the future. Handling equipment and facilities for the two storage methods are so designed that a combination in the desired proportions is practically feasible.

Outline, contents and experimental basis of the report

The report is divided into a General Volume (I) and a Technical Volume (II). The general volume presents premises and data, a description of the various steps of the handling procedure, a summary of dispersal processes and a safety analysis.

The technical volume contains, as did volumes II-IV of the preceding report, detailed sections on geology and the facilities in the handling chain. Buffer material, canister material, leaching and dispersal are dealt with in separate chapters and in greater depth than in the preceding report. The technical volume is concluded with an analysis of the safety of the various handling and storage steps.

The report is based on material obtained from the literature and from institutions in Sweden and abroad which are active within fields related to the handling and storage of spent fuel. KBS has also commissioned a large number of technical-scientific studies and investigations of its own, which are described in KBS Technical Reports. At the time of the publication of the first KBS report, 56 such reports were available. The number is now 120 (see Volume II, Appendix 2). These technical reports are available, but are not included in the present report.

The organization of the KBS project was described in the first report and has remained largely unchanged since then.

Companies, institutions and experts engaged by the KBS project are listed in an appendix to this volume. They have done work within their own special fields and have reported their results in KBS technical reports. Some have also participated in the preparation of sections of the present report. KBS is, however, fully responsible for the contents of this and the preceding report.

Handling procedure for spent nuclear fuel

The handling of spent nuclear fuel is similar in some respects to the procedures described in the KBS report on vitrified waste from reprocessing. Those parts of the handling chain which pertain to the reprocessing of the fuel and the intermediate storage of the vitrified waste are, however, not applicable. Otherwise, the handling chain differs primarily with respect to the encapsulation procedure and the composition of the buffer material which surrounds the canister in the final repository. The reason for this is that the quantity of heavy elements with very long-lived radioactivity is considerably greater in spent fuel than in vitrified waste. Demands on the long-term isolation of the spent fuel are therefore greater.

After the spent fuel has been stored for some time (at least six months) in the nuclear power station's storage pools, it is transported to a central storage facility for spent nuclear fuel, where it is stored for 40 years. During this period of time, the heat emitted by the fuel declines to such a degree that the temperatures in the rock can be kept to a level for which experience exists (e.g. from oil storage in rock caverns).

The length of the period of supervised storage - 40 years - prior

to final storage is the same as the length of the corresponding period in the reprocessing alternative. It is technically feasible and permissible from the viewpoint of safety to shorten or lengthen the storage period. The final choice of storage period is a later question related to optimization of the handling chain.

After the storage period, the fuel is transported to an encapsulation station. There, the fuel is disassembled and the fuel rods are separated from the metal components of the fuel assemblies.

The fuel rods are encapsulated in containers of pure copper with a wall thickness of 20 cm. The space between the rods is filled with lead. The filled copper canisters, which weigh about 20 tonnes each, are then transferred to a final repository consisting of a system of tunnels approximately 500 metres down in the bedrock. The canisters are deposited in vertical boreholes and surrounded with a buffer material composed of highly compacted bentonite. Bentonite is a naturally occurring clay material which swells when it absorbs water. When the repository is filled with canisters, it is sealed, the tunnels and shafts being filled with a mixture of quartz sand and bentonite. Figure 1-1 shows a cross-section through the vertical storage hole with canister and buffer material after sealing.

The metal components of the fuel assemblies, which are also radioactive, are embedded in concrete cubes. The cubes are deposited in tunnels in a separate final repository in rock at a depth of about 300 metres. The storage tunnels are filled with concrete.

According to the KBS proposal, final storage shall not commence until some time early in the next century. In the meantime, current development work will undoubtedly lead to alternative methods for final storage. Alternative proposals for encapsulation, among other things, have been studied within the KBS project. Such an alternative is described in the present report (Volume II, Appendix 1) in the form of a canister made of aluminium oxide which appears to possess excellent long-term stability. This type of container may comprise an alternative to the copper canister.

Geological premises

Comprehensive studies and investigations carried out to determine the suitability of the Swedish bedrock for a final repository were described in the preceding KBS report. Work has continued in the three areas judged to be the most suitable. This work has confirmed that the chosen areas comprise suitable possible sites for the location of a final repository. The results of the continued work have verified the order of priority presented in the preceding report, namely Karlshamn, Finnsjön and Kråkemåla in that order. Continued studies can be expected to show that numerous other areas in the country also meet the conditions imposed on a final repository.

No reasons have been found to question the previous conclusions that a final repository could be located at a depth of about 500 metres in the Swedish bedrock.

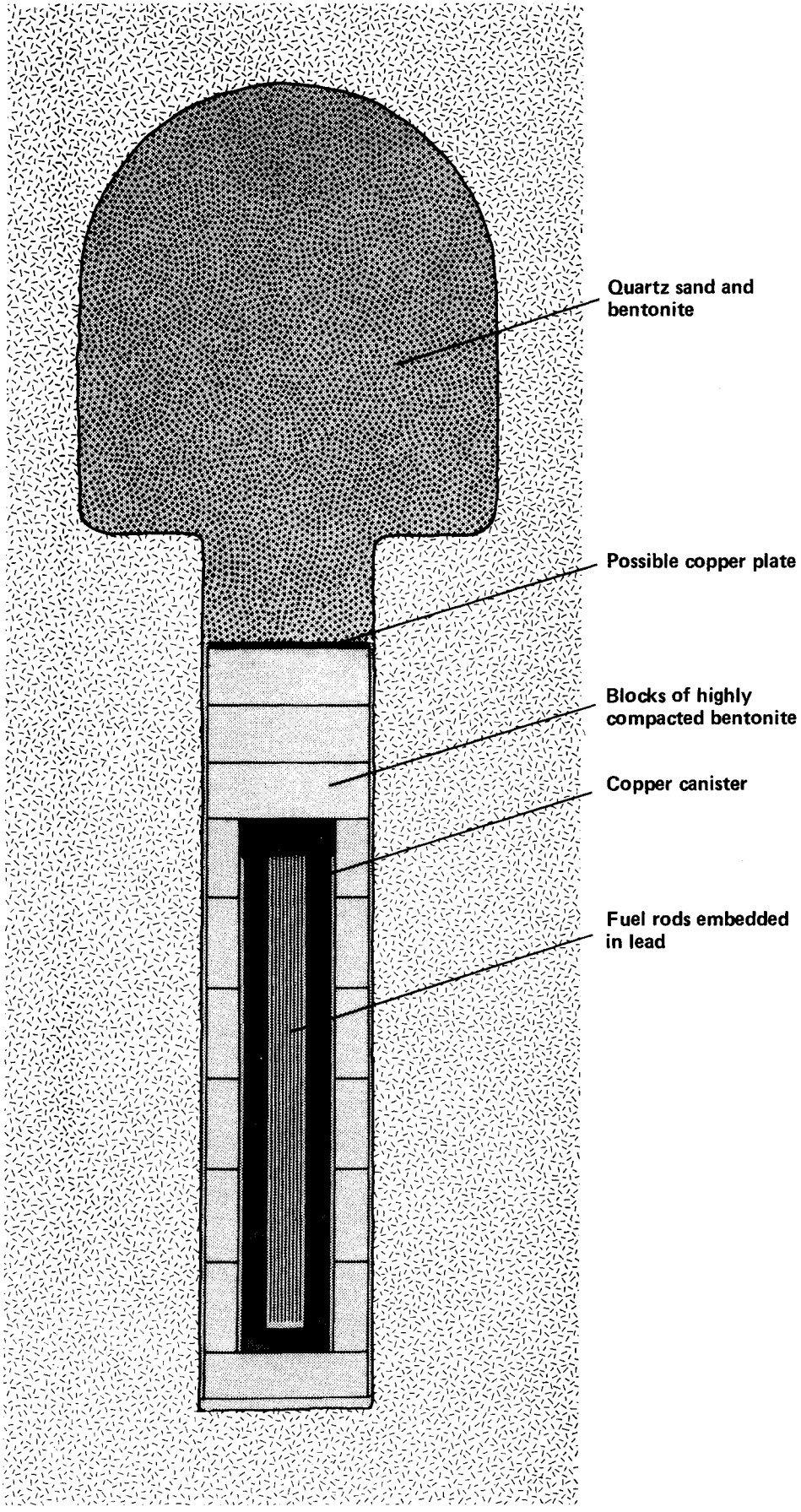


Figure 1-1. The sealed final repository. The canister is surrounded in the storage hole by highly compacted bentonite. The gaps are filled with bentonite powder. The tunnel is filled with a mixture of quartz sand and bentonite. A copper plate can, if desired, be placed on top of the bentonite blocks to serve as a diffusion barrier.

For all practical purposes, radioactive substances from the final repository can only be dispersed with the groundwater. It is therefore of vital importance for the assessment of the safety of the final repository that the rock possess high imperviousness and that forces which would set the groundwater in motion are small. Continued hydrogeological investigations have shown that the groundwater flows in the study areas are very low and that the groundwater's current pattern of flow will not change for a very long time in the geologically stable areas which have been investigated.

Theoretical studies have shown that the flow time of the groundwater from the final repository to the surface of the ground is very long in high-quality rock. Through numerous age determinations, the flow time of the groundwater from a suitably situated repository at a depth of 500 metres to the surface of the ground has been found to be about 3 000 years or more.

Measurements of the oxygen content and redox potential of Swedish groundwater from considerable depth as well as mineralogical and geochemical observations show that reducing conditions generally prevail in the Swedish bedrock. This is essential for the limited capacity of the groundwater to dissolve and disperse the radioactive substances.

Encapsulation and leaching

The copper canister which encases the spent fuel comprises a long-lasting barrier against the dispersal of radioactive substances in the fuel. The canister also comprises a radiation shield which reduces the radiolysis of the groundwater to a level which is insignificant from the viewpoint of corrosion.

The copper canister is very resistant to corrosion. Corrosion attack can be caused by oxygen, certain sulphur compounds and microbiological action. The groundwater and the buffer material, for instance, can contain substances which lead to corrosion. Calculations based on available quantities of oxidizing material show that the service life of the copper canister should be in encess of one million years. An expert group assembled by the Swedish Corrosion Research Institute has arrived at the conclusion that it is "realistic to anticipate a service life of hundreds of thousands of years for a copper canister with a wall thickness of 200 mm". One of the members, however, has submitted a dissenting statement.

The copper canister can also be subjected to mechanical stresses during its lifetime due to external overpressure and other forces. In order to prevent damage to the canister, the space between the fuel rods and the canister is filled completely with lead. Mechanical stresses which could shorten the service life of the canister to less than one million years cannot be foreseen.

In the final repository, the copper canisters are surrounded by highly compacted bentonite. The bentonite will gradually absorb water, whereby it swells and closes up all cracks and small cavities which might be created in connection with deposition. As a result of this swelling, the bentonite becomes highly impervious and virtually impenetrable by flowing water. The transport of

corrosive agents, corrosion products and radioactive elements through the bentonite layer can therefore only take place by means of diffusion. The highly compacted bentonite thereby comprises a second encapsulation of the waste.

In order for the groundwater to come into contact with the radioactive substances in the fuel, the copper canister, the lead which surrounds the fuel rods and the fuel's zircaloy cladding must be penetrated. Canister damage of such extent that water can come into contact with a large portion of the fuel can only occur after a very long period of time. If this happens, various constituents of the fuel may be leached out and eventually transported away with the groundwater. Owing to the stable and chemically reducing environment which prevails in the Swedish bedrock and owing to the limited water flow, this leaching process will be extremely slow. Even under less favourable conditions, it will take many hundreds of thousands of years to leach out the uranium content of a canister.

The action of the groundwater will also gradually lead to the breakdown of the concrete around the radioactive metal components. The metal components will then come into contact with the groundwater, leading to the dispersal of, among other elements, radioactive nickel. However, even with unfavourable assumptions, the quantity of radioactive material which will then be dispersed will give rise to insignificant radiation doses.

Retardation and dispersal

The majority of radioactive elements which are dissolved from the fuel will, owing to various reactions, migrate at a slower rate than the groundwater. Retardation factors have been determined for various nuclides by means of experimental studies.

The work which has been done on geochemical conditions in the bedrock since the publication of the preceding KBS report has greatly affected the evaluation of the geological barrier. Studies and measurements have verified the fact that reducing chemical conditions prevail at the depths in question in the type of rock in which the repository is proposed to be located. Swedish observations as well as studies reported in the literature, including those from Russian and Czechoslovakian uranium deposits, show that uranium and uranium compounds possess extremely poor solubility in the chemical environment in Question. This is entirely in agreement with and confirms the results obtained from laboratory tests. In addition to low solubility, the reducing environment entails considerably greater retardation factors than those which were used in the preceding report.

The dispersal of the radioactive elements in the biosphere through various food chains must be clarified in order to permit the dose load to mankind to be calculated. Thirteen different paths of exposure have been studied in a manner similar to that described in the first KBS report. In order to provide a concrete basis for the calculation, a repository has been postulated at Finnsjö lake.

Safety in the handling chain

The safety analysis has shown that the risk of a dispersal of radioactive substances in connection with normal operation or in the event of accidents in the various stages of the handling chain is insignificant. Analyses of the transportation and storage of spent nuclear fuel were described in the preceding KBS report. The present report analyzes other parts of the chain, of which the encapsulation procedure especially is particular to spent fuel. It is assumed that the detailed demands on control, documentation and surveillance which will be made by the concerned Swedish Government authorities when they examine the design of and grant permission for the construction and operation of the facilities included in the handling chain will be similar to those currently imposed on nuclear power plants.

Safety of the final repository

In order to isolate the radioactive elements in spent nuclear fuel in the final repository for very long periods of time, the fuel is surrounded by a number of barriers. Each barrier provides protection against the dispersal of radioactive substances, but the different barriers also possess different properties and functions which both reinforce and complement each other.

These barriers, which have been mentioned above, are:

- the uranium dioxide, which is in itself a poorly soluble substance,
- the canister, made of a highly durable material,
- packing of the canister in an impervious buffer material,
- final storage in stable bedrock with a low groundwater flow,
- chemical barriers against dispersal of the radioactive elements.

As in the case of vitrified waste, the calculations show that the group of people which could be exposed to the greatest radiation effects are persons who, in a distant future, obtain their drinking water from a rock well drilled in the vicinity of the repository.

The safety analysis shows that, with conservatively chosen assumptions (the so-called "main case" in the calculations), these persons will be subjected to a maximum additional irradiation (individual dose) of about 10 millirems per year, which will not occur until after about 70 million years. This dose is on a level with the Swedish design goal for new nuclear power plants and is well below the limit of 100 millirems per year recommended by the International Commission on Radiological Protection (ICRP) for exposure over a number of years. As an extreme case, the individual dose assuming extremely short water transit times from the final repository to the surface of the ground an lower retardation factors for the dispersal of radioactive substances to the geosphere has been calculated. The individual dose would then amount to about 70 millirems per year and would occur after about 1 million years. This value is also lower than the limit of 100 millirems per year recommended by the ICRP, but slightly higher than the 50 millirems per year permitted in Sweden as the highest radiation dose to persons living near a nuclear power plant.

A dominant contribution to the radiation dose is provided by radium-226, which also occurs in nature. The calculated elevated levels of this element due to release from the final repository lie within the interval which has been measured for natural drinking water in Sweden.

The probable sequence of events on the basis of completed analyses is that the copper canister will remain intact for at least one million years and that subsequent leaching of the uranium fuel will take millions of years. The dose load will hereby be less than in the main case discussed above and the maximum dose rate will occur at a later time.

Over a period of one million years, most of the radioactive elements in the spent fuel will decay. During the following period of time, the radiotoxicity of the fuel is dominated by decay products from uranium, primarily radium-226. In this case, the consequences of the final storage of the spent fuel will not be greater than the consequences of the storage of a corresponding quantity of uranium dioxide which has never been irradiated in any reactor. One exception is for the dispersal of iodine-129, but this does not give rise to any appreciable dose increments.

The radiation doses from a final repository are presented in fig 1-2, which also presents the dose limits which apply to nuclear power plants in Sweden as well as the natural radiation levels in the country.

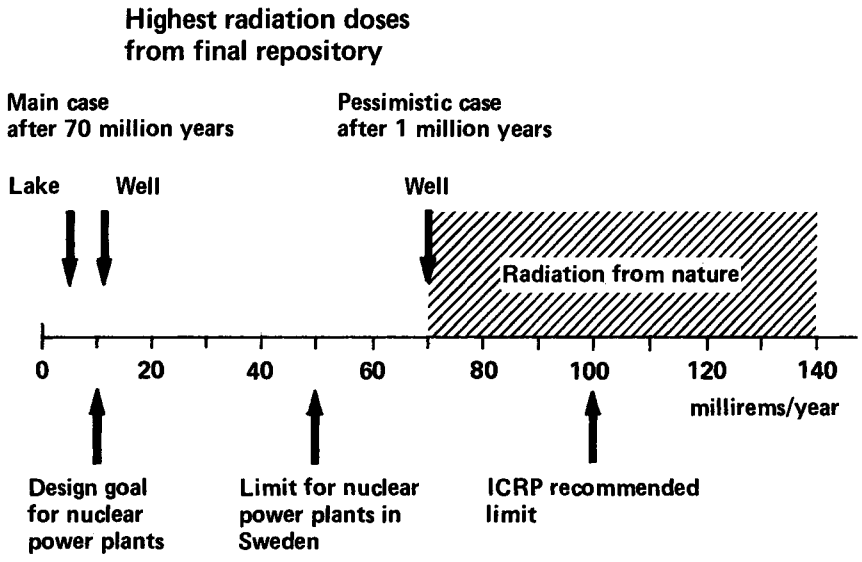


Figure 1-2. The calculated maximum radiation doses from the final storage of spent fuel to an individual compared with some limit values. ICRP is the International Commission on Radiological Protection. Radiation from nature varies from one place to another in Sweden and lies within the hatched area.

Time perspective

The time spans which are dealt with in the safety analysis are of such an order of magnitude that they can scarcely be conceived of in our normal frames of reference. A somewhat better overview of the situation may be obtained by considering the anticipated future sequence of events divided into different phases.

It is hereby assumed that the earth will remain the abode of some form of human life throughout this sequence of events. Otherwise, the discussion of the effects of the final repository on ecological systems on the earth is without interest.

Several thousand years in the future

During this phase, the copper canisters will remain completely intact, with the possible exception of a few isolated canisters which may have been defective from the start. The quantity of radioactivity which can leak out from these defective canisters and reach the biosphere gives rise to negligible doses.

For several thousands of years to come, the final repository can therefore be regarded as being "absolutely safe", regardless of how this concept is defined.

From several thousands of years to several hundreds of thousands of years in the future

Calculations show that there is no chance of any release of radioactivity to the biosphere during this period of time, even if pessimistic assumptions are made concerning the condition and function of the various barriers. During this phase, it is not impossible that the canisters could begin to be penetrated and groundwater could come into contact with the spent fuel. However, the chemical conditions which prevail in the buffer mass and in the rock comprise an effective barrier to any dispersal of radioactivity. From a geological and geochemical viewpoint, conditions during this period of time are not difficult to foresee; nor could major climatic changes affect a final repository at a depth of 500 metres in the Swedish bedrock.

It is therefore asserted that the final repository will fulfil the requirement of being "absolutely safe" during this time period as well.

From several hundreds of thousands of years onward

The safety analysis shows that, with pessimistically chosen assumptions, some release of radioactivity can occur to the biosphere after about 1 million years. The dose increment which would result from this release has been calculated to be of the same order of magnitude or lower than the limits which are applied today and lower than the doses which stem from natural radiation. However, in this time perspective of millions of years, it is not considered to be meaningful or reasonable to discuss in any greter depth the impact of a final repository in relation to present-day standards. Nor has any attempt at such an

evaluation been made within other areas of human endeavour which could have long-term environmental impact - not even for time perspective which corresponds to the first phase dealt with above.

In view of the long-term geological stability exhibited by the bedrock in those parts of the country which could be considered for the location of a final repository, the possible impact of a final repository would be very limited. The impact of radiation on ecological systems and their evolution would be dominated by natural radiation, in comparison with which the impact of the final repository would be small locally and negligible globally.

2 INTRODUCTION

2.1 REQUIREMENTS TO BE FULFILLED BY THE NUCLEAR POWER INDUSTRY

In April of 1977, the Swedish Parliament passed the "Stipulation Law" (Law concerning special permission for charging nuclear reactors with fuel, etc; SFS 1977:140).

For new reactors, the Law provides:

> "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, unreprocessed nuclear fuel can be effected."

In order to fulfil the requirements stipulated in the Stipulation Law which pertain to the handling and final storage of spent nuclear fuel or high-level waste, the Nuclear Fuel Safety Project, KBS, was formed by the four nuclear power utilities Statens Vattenfallsverk (the Swedish State Power Board), Oskarshamnverkets Kraftgrupp AB (OKG), Sydkraft AB and Forsmark Kraftgrupp AB (FKA).

KBS was organized as an independent project within Svensk Kärnbränsleförsörjning AB (SKBF) - Swedish Nuclear Fuel Supply Company. The work has been 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.

In December of 1977, KBS presented its initial report entitled "Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste", a report on how and where an absolutely safe final storage of the high-level waste obtained from reprocessing can be effected.

The present second report, entitled "Handling and Final Storage of Unreprocessed Spent Nuclear Fuel", is a report on how and

where an absolutely safe final storage of spent, unreprocessed nuclear fuel can be effected.

The accountability requirements imposed on reactor owners are specified in greater detail in the Stipulation Law Bill (Bill 1976/77:53). The requirements pertaining to unreprocessed spent nuclear fuel 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 will be handled, and b) that the handling method will provide adequate safeguards against harmful effects.

3 The basic premise must be that the spent nuclear fuel is 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 spent nuclear fuel will be stored.
- How the storage site will be arranged.
- How the spent nuclear fuel 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 spent nuclear fuel for as long a time as is required for the radioactivity to diminish to a harmless level.

The risk of release of the 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.

2.2 OUTLINE OF THE REPORT

The report is divided into two volumes:

I General
II Technical

Where the general volume can be read independently. The technical volume is a more detailed technical description of the facilities which are required, the barriers which prevent or retard the dispersal of radioactive substances from the final repository to the biosphere and the safety analysis which has been carried out on the described handling and storage methods.

Chapter 3 of volume I (I:3) defines the term "spent fuel" and gives data on quantities and storage requirements, after which chapter 1:4 summarizes the proposed chain of handling, treatment and final storage of the spent nuclear fuel.

Chapter 1:5 gives a general description of the necessary facilities. A more detailed description, including drawings, is provided in chapter II:2.

The geological background material and an account of the feasibility of locating the proposed final repository in Swedish bedrock are presented in chapters I:6 and II:3.

Chapters 7-10 of volume I describe the function and properties of the various barriers, as do chapters 4-7 of volume II.

The concluding chapter I:11 summarizes the safety analysis and safety evaluation which is reported in greater detail in chapter II:8.

An appendix listing the companies, institutions and experts contracted or consulted by the KBS project has also been attached to the General volume.

The development status of an alternative encapsulation method - an aluminium oxide canister proposed by ASEA - is described in Appendix II:B1 to the Technical volume. A second appendix, II:B2, contains a list of all technical reports KBS has published. As is evident from the list of references in Volume II, these reports constitute the greater part of the background material for this report.

3 PREMISES AND DATA

3.1 DATA FOR SPENT NUCLEAR FUEL

3.1.1 Composition

The uranium fuel with which a reactor is charged consists of uranium dioxide which, compared with naturally occurring uranium, has a higher content of the fissile isotope uranium-235. Fuel for lightwater reactors contains about 3% uranium-235. Natural uranium contains 0.7%.

The composition of the fuel changes during reactor operation. The spent fuel which is taken out of the reactor after having reached the intended burnup has an average composition as follows:

	Boiling water reactor (BWR)	Pressurized water reactor (PWR)
Uranium-235	0.7%	0.9%
Uranium-236	0.4%	0.4%
Uranium-238	95.2%	94.1%
Fissile 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 ionizing radiation. The radiation from the spent fuel comes mainly from fission products and diminishes as the elements decay.

Radioactive nuclides decay with different half-lives and the radioactivity of the fuel declines with time (the fuel "cools"). Figure 3-1 shows how the radioactivity of PWR fuel corresponding to 1 tonne of uranium declines after discharge from the reactor. The figure also shows which nuclides dominate the radioactivity at different points in time after discharge from the reactor.

3.1.2 Decay heat output in spent fuel

Immediately after shutdown of the reactor, heat generation in the fuel elements declines sharply, but a certain amount of decay heat is still generated as a result of the disintegration of the

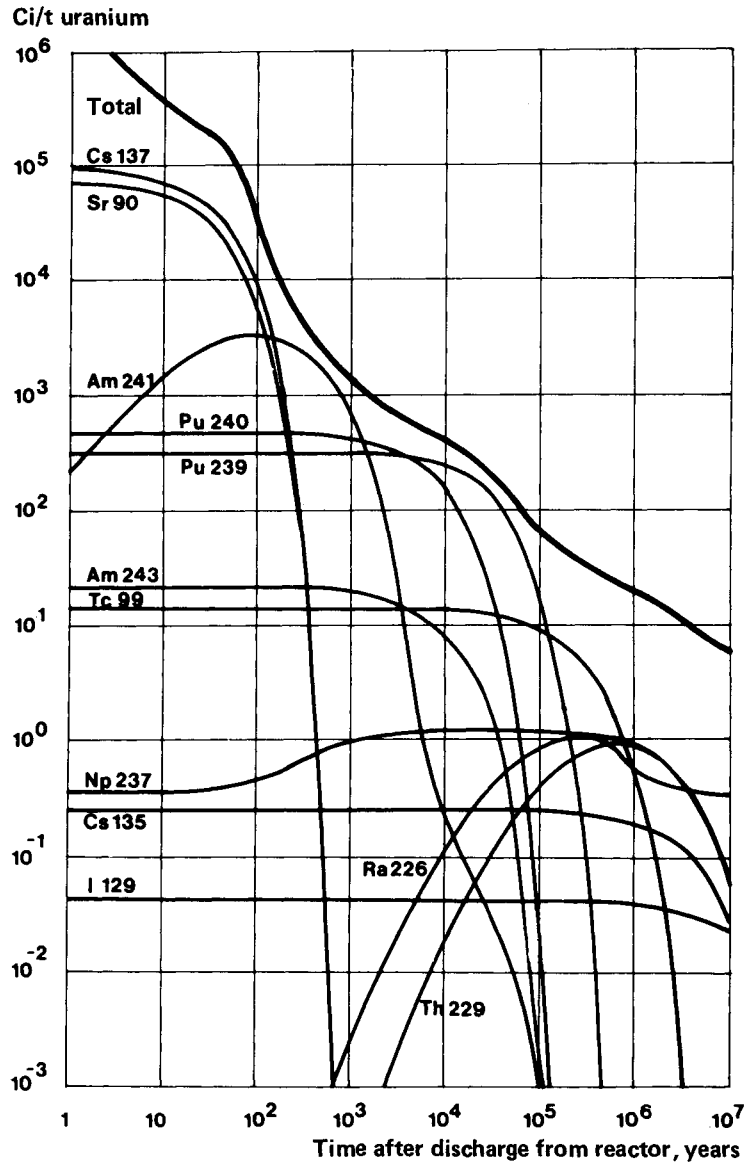


Figure 3-1. The graph shows how the radioactivity of the radioactive elements in spent PWR fuel declines with time.

radioactive elements formed in the process. One minute after shutdown, decay heat output is 5% of thermal power output during operation and continues to decline rapidly. After a month, it is around 0.1%. Radioactivity declines at approximately the same rate. Figure 3-2 shows how decay heat output declines during the time period between 1 and 10 000 years after discharge of the fuel from the reactor. A more detailed account of the radioactive elements and the decay heat generation in spent fuel is given in II:8.2.

3.2 DIFFERENT TYPES OF WASTE FROM SPENT FUEL

The Stipulation Law requires as an alternative an account of the final storage of "spent, unreprocessed nuclear fuel".

The fuel for a nuclear reactor consists of cylindrical pellets of uranium dioxide enclosed in tubes ("cans") made of a zirconium

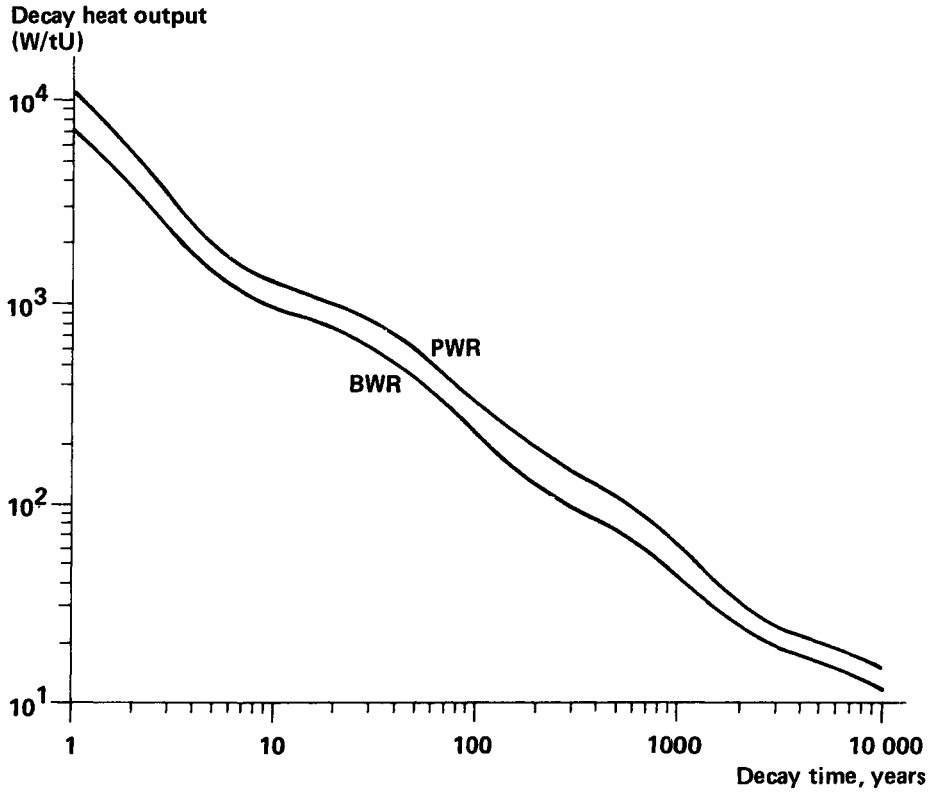


Figure 3-2. The graph shows how heat generation per tonne of uranium (decay heat output) in spent PWR and BWR fuel declines with time.

alloy. The cans with their pellets are called fuel rods. The fuel rods are held together by structural elements in square bundles called fuel assemblies (see figure 5-1). Fuel assemblies are handled as units in connection with fuel replacement. The fuel assemblies in the first core in a PWR reactor may also contain special rods with neutron-absorbing substances to reduce the reactivity in a fresh core.

In a boiling water reactor (BWR), the fuel assemblies are encased by square fuel channels, known as boxes, in order to prevent a cross-flow of reactor water in the core. In a pressurized water reactor (PWR), special control rod guide tubes are installed in the fuel assemblies. Through these guide tubes, the control rods are lowered down into the core in order to control thermal power generation.

In the encapsulation procedure preceding the final storage of spent nuclear fuel in accordance with the proposed handling sequence (see I:4), the fuel assemblies are to be dismantled, after which the fuel rods are placed in a copper canister which is deposited in the bedrock.

Any fuel which is spilled in connection with the handling of the fuel assemblies is encapsulated in smaller copper cans, which are in turn placed in a copper canister of the same size as those for spent fuel and deposited in the bedrock.

The metal components which are separated from the fuel rods dur-

ing dismantling contain radioactive elements formed by neutron absorption, but no fission products or actinides. The same applies to fuel boxes and special rods with neutron-absorbing elements. These components are collectively referred to here as the fuel's metal components. These metal components are placed (after chopping and compaction) in prefabricated cubicle concrete boxes, moulds, which are filled with mortar and placed in separate rock tunnels.

All of these three types of waste - fuel rods, fuel spill and metal components - are included in this report by the use of the term "spent nuclear fuel" and their handling and final storage is described.

Handling of spent nuclear fuel also gives rise to other wastes of the same type as the low- and medium-active waste which is obtained from the operation of nuclear power reactors. The handling and final storage of this waste is not dealt with in this report. It has a lower radioactivity concentration and a lower decay heat output, which greatly lowers the requirements on the barriers which are to isolate the waste from the environment in comparison with those for the waste dealt with here.

3.3 QUANTITIES OF SPENT FUEL FROM SWEDISH REACTORS

Table 3-1 gives the expected accumulated quantity of spent fuel

Tabell 3-1

Accumulated quantities of spent fuel in tonnes 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

obtained from the operation of the 13 reactors established by the 1975 Swedish Parliament as the framework for Sweden's nuclear power plant construction programme up to 1985. The table also shows the accumulated quantity from the six reactors 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 the energy availability factor for all blocks will be 60% during the first three years and 70% thereafter.

The annual quantities of spent nuclear fuel discharged from the currently operative Swedish reactor blocks are given in table 3-2.

Tabell 3-2

Annual quantities of spent fuel from operative Swedish reactors in tonnes 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					

The nuclear power stations have some storage capacity in existing fuel pools. This capacity can be expanded by the acquisition of new fuel racks which permit a denser placement of fuel elements. Table 3-3 gives the dates on which spent fuel must have been removed from the specified stations, assuming an expansion of pool storage capacity and retained stand-by capacity to empty the reactors of fuel.

Tabell 3-3

Dates for first necessary transport of spent fuel from plant.

Reactor	First transport 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

In order to meet the need for additional storage space for spent nuclear fuel, the Swedish Nuclear Fuel Supply Company is conducting a design study for a central storage facility for spent fuel. A siting application for such a facility was submitted to the Government on November 30th, 1977. The facility is planned to have a storage capacity equivalent to 3 000 tonnes of uranium, with potential for expansion.

The present proposal for the handling and final storage of spent nuclear fuel presumes a 40-year storage of the fuel prior to encapsulation and desposition in the final repository. With 30 years of operation of 13 reactors, a storage capacity requirement corresponding to 9 000 tonnes of uranium is obtained. A tripling of the planned capacity of the central fuel storage facility would therefore be required, see chapter I:5.

4 HANDLING PROCEDURE FOR SPENT NUCLEAR FUEL

4.1 GENERAL

The handling of spent nuclear fuel is similar in some respects to the procedure described in the KBS report on vitrified waste from reprocessing. However, aside from the fact that those parts of the handling chain which pertain to the reprocessing of the fuel and the intermediate storage of the vitrified waste are not applicable, handling of unreprocessed spent fuel also differs with respect to the encapsulation procedure and the composition of the buffer material which surrounds the canisters in the final repository. The reason for this is that the amount of heavy elements with very long-lived radioactivity is considerably greater (on the order of 100 times greater) in spent fuel than in vitrified waste. The requirements on the long-term isolation of the spent fuel are therefore greater. The design of the final storage method is also influenced by the fact that some of the radioactive elements in spent fuel exist in volatile or easily soluble form. Finally, the final storage of spent fuel also includes the handling and disposal of the radioactive metal components in the fuel assemblies.

4.2 PROPOSED HANDLING SEQUENCE

The handling sequence for spent nuclear fuel which is not to be reprocessed is illustrated by the block diagram in figure 4-1. In brief, it involves the following stages.

1 After discharge from the reactor, the spent nuclear fuel is stored in the nuclear power station's storage pools for at least six months. During this time, the decay heat declines considerably and short-lived fission products (e.g. iodine-131) disintegrate completely. The risk of a serious accident in connection with the handling of the fuel is thereby greatly reduced.

2 The fuel is then transported to a central storage facility for spent nuclear fuel. The transport system is described in the KBS report on vitrified waste from reprocessing.

3 The fuel is stored in the central fuel storage facility for a period of 40 years. As at the power stations, the fuel is stored here in water-filled pools. If all of the fuel from 30 years of operation of 13 reactors is to be stored in this manner, the central storage facility which is currently

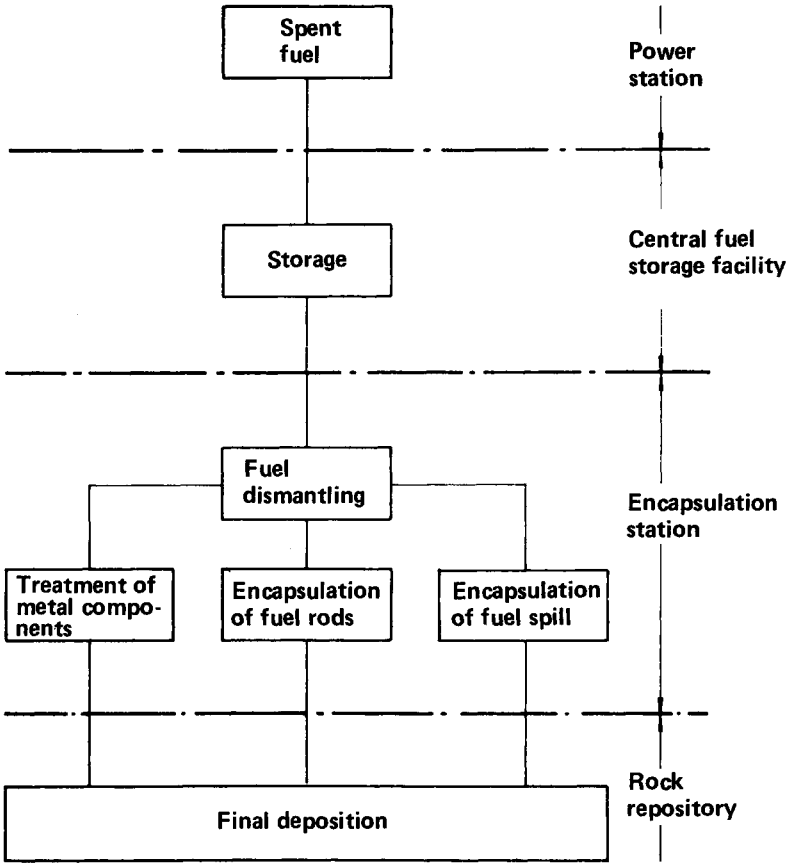


Figure 4-1. Flow scheme illustrating the handling chain for the spent fuel from the power station to the final repository.

planned must be expanded to triple its capacity, 9 000 tonnes (see section 5.2). Alternatively, several central storage facilities of similar design can be built at different sites.

4 After 40 years of storage, the fuel is transported to an encapsulation station. This transport is identical to that used to transfer the fuel from the power station to the central fuel storage facility.

5 The encapsulation station is situated above ground in connection with the final repository. In the encapsulation station, the fuel is dismantled, whereby the fuel rods are separated from the metal components of the assemblies, see section 5.3.

The fuel rods are enclosed in copper canisters with walls 200 mm thick, see figure 5-7. Each container is 77 cm in diameter and 470 cm long, weighs about 16 tonnes and can hold 498 or 636 rods, depending on the type of fuel. The space between the fuel rods is filled with lead. Fuel spill from handling of the fuel assemblies is encapsulated in a similar manner. A total of some 7 000 copper canisters are required for the spent fuel from 30 years of operation of 13 reactors.

The metal components of the fuel assemblies are embedded in concrete cubes (moulds), 1.6 m on a side. In all, some 1 200

concrete cubes with a total volume of about 5 000 m^3 are obtained from 30 years of operation of 13 reactors.

6 The copper canisters are transferred to a final repository approximately 500 m down in the bedrock. The repository is designed as a system of tunnels 3.7 m in width and 4 m in height and spaced at a distance of about 25 m. Storage holes 1.5 m in diameter and 7.7 m deep are drilled in the floors of the tunnels. One waste canister is stored in each hole. The centre-to-centre distance between the storage holes is 6 m. The layout of the tunnel system and the storage holes is described in section 5.3 and illustrated by figures 5-8 and 5-9.

In the storage holes, the copper canisters are surrounded by blocks of highly compacted bentonite. Bentonite is a clay which swells considerably when it absorbs water. This material has been chosen in view of its good mechanical stability, very low water permeability and good long-term stability. It also possesses good ion-exchanging capacity. The bentonite is compressed under high pressure into blocks which are stacked on top of each other under, around and on top of the copper canister. Bentonite powder is packed into the gaps next to the canister and at the surface of the rock. See figure 4-2. When the bentonite absorbs water, it swells and forms a homogeneous extremely impervious clay mass. The only transport process of significance through this dense, tightly-packed clay is diffusion, which takes place at an extremely low rate.

7 After the final repository has been filled to design capacity with canisters, the facility can be kept open and supervised as long as surveillance is considered to be desirable. The facility is then sealed, whereby all tunnels and shafts are filled with a mixture of quartz sand (80-90%) and bentonite (10-20%). At some points, highly compacted bentonite is used to seal extra tightly.

8 The concrete cubes containing the metal components from the fuel assemblies are deposited in a separate final repository. This repository is designed with two storage tunnels 7.6 m wide, 6.5 m high and 250 m long situated at a depth of 300 m in impervious bedrock, see figure 5-13.

The concrete cubes are stacked in the storage tunnels on a concrete floor, two in height and three abreast. The storage tunnels are sealed in stages by being filled with concrete. The transport tunnels and shafts are filled with sand/bentonite in the same manner as in the final repository for copper canisters.

The timetable for the above-described handling chain is illustrated by figure 4-3, under the assumption that all of the fuel from 30 years of operation of 13 reactors is to be disposed of without reprocessing. The date for the first shipment of fuel from a power station to the central fuel storage facility has thereby been taken to be 1980.

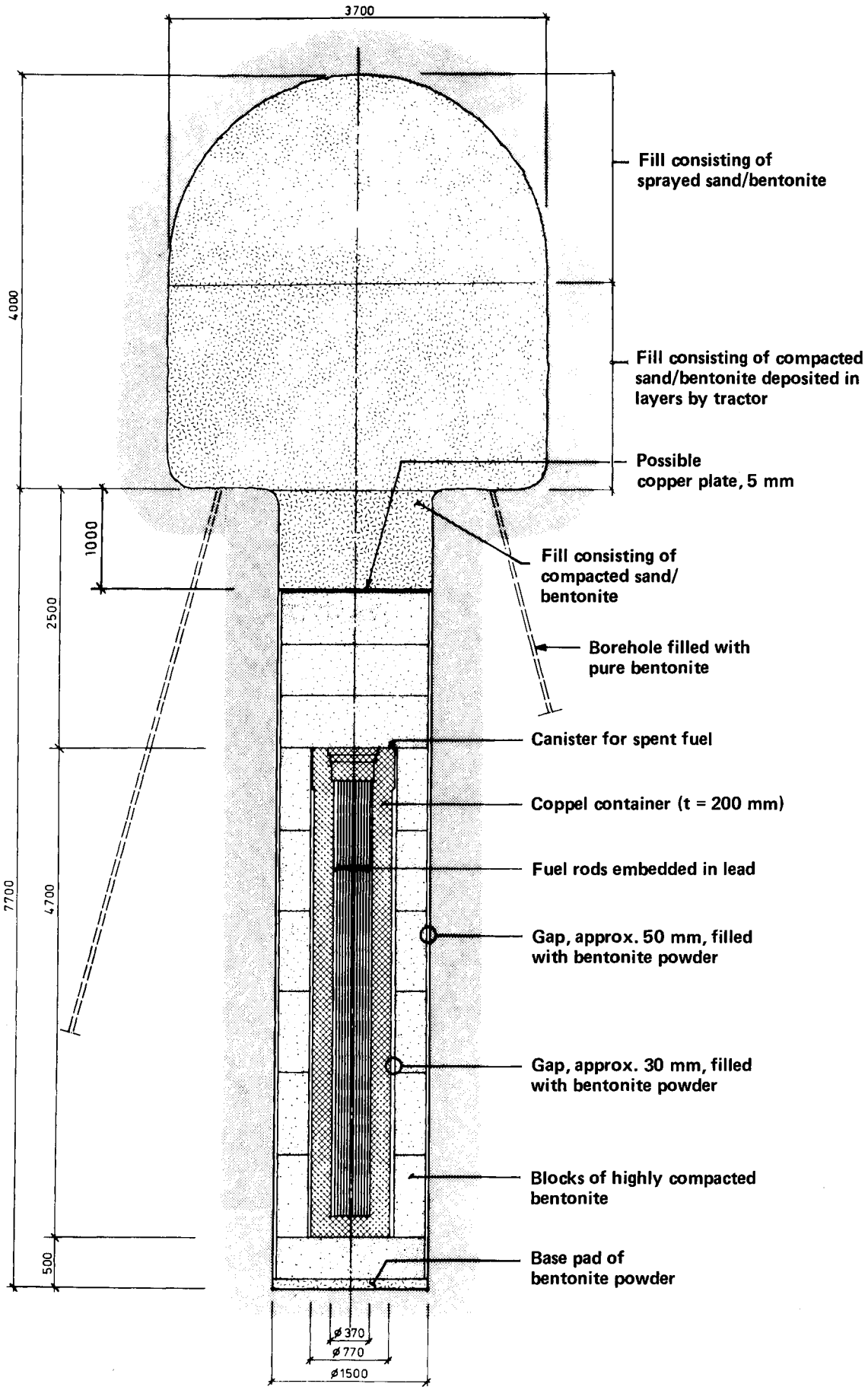


Figure 4-2. The sealed final repository. The canister is surrounded in the storage hole by highly compacted bentonite. The gaps are filled with bentonite powder. The tunnel is filled with a mixture of quartz-sand and bentonite. A copper plate can, if desired, be placed on top of the bentonite block to serve as a diffusion barrier.

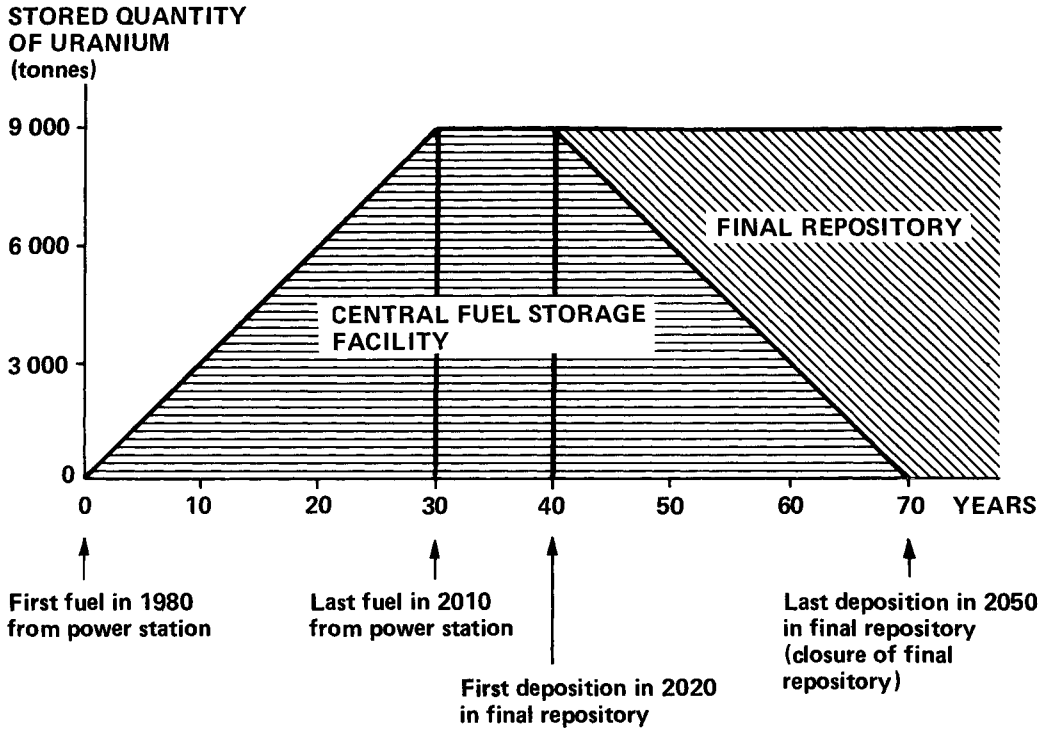


Figure 4-3. Diagram illustrating the capacity requirements for the central fuel storage facility and the final repository at different points in time.

4.3 BARRIERS AGAINST DISPERSAL OF RADIOACTIVE SUBSTANCES FROM FINAL REPOSITORY

The proposed handling chain provides for the safe handling and final storage of the spent nuclear fuel. This is further demonstrated in chapter 11. In the final repository, the dispersal of the radioactive substances is prevented or retarded in the following manner.

- The radioactive elements are, for the most part, bound in the ceramic fuel or in the metal components of the fuel assemblies. Both the fuel and the metal components are poorly soluble in water.

- The fuel rods are enclosed in copper canisters with walls 200 mm thick. The spaces between the rods are filled with lead. The ceramic fuel is enclosed in zircaloy cans (which are a part of the fuel rods). In the environment in the final repository, there is virtually no corrosion of the copper. The lead and zircaloy also possess excellent resistance to corrosion in this environment.

  It is therefore improbable that water will come into contact with the fuel for several hundreds of thousands or even millions of years following deposition (see chapter 8).

- The buffer material which surrounds the canister possesses very low water permeability and the transport of radioactive

substances through the material is controlled by diffusion. This takes place at such a slow rate that it will probably take millions of years to dissolve the fuel, if it is dissolved at all (see chapter 9).

- The buffer material and the rock have a chemically retarding effect on most radioactive elements in the fuel. Due to ion exchange, precipitation and mineralization, the radioactive elements which have been dissolved in the water are dispersed at a much slower rate than that at which the water flows (see chapter 10).

- The rock formation in which the final repository is to be situated is chosen with care. Groundwater movements shall be small and shall be in such a direction that it takes a long time for the water to flow from the final repository to areas in contact with the ecological systems. Studies carried out by the Geological Survey of Sweden (SGU) on behalf of KBS have shown that there are bedrock areas in Sweden which possess the desired properties (see chapter 6).

4.4 FLEXIBILITY AND DEVELOPMENT POSSIBILITIES

The handling chain proposed here, with 40 years of supervised storage of the fuel prior to final storage, permits considerable flexibility with regard to future alternatives and in anticipation of further technical development.

Such flexibility is desirable in order to permit an adaptation of methods and facilities to future technological advances. Current development work will probably produce methods which simplify and reduce the costs of final storage without sacrificing safety. Intermediate storage of the spent fuel also makes it possible to decide in favour of reprocessing at a later date and thereby make use of the latent energy in the spent fuel.

Alternative solutions to e.g. encapsulation have been studied within the framework of the KBS project. Certain ceramic materials have been found to possess promising properties. According to a method developed by ASEA in collaboration with KBS, the fuel rods are enclosed in canisters of aluminium oxide fabricated by means of hot isostatic high-pressure compaction. This material appears to be equivalent to copper with respect to potential service life under the conditions which exist in the final repository. A status report from this development work is presented in section 8.4 and Appendix 1 to Volume II.

5 FACILITIES

5.1 GENERAL

The fuel for a nuclear reactor consists of cylindrical pellets of uranium dioxide enclosed in zirconium alloy (zircaloy) cladding tubes. The tubes with pellets are called fuel rods. They are bound together in fuel assemblies, which are handled as units.

The fuel assemblies are of varying design, depending on the type of reactor for which they are intended (see section 2.1.1, Volume II). Figure 5-1 shows a fuel bundle intended for a BWR reactor. A BWR assembly consists of such a bundle plus a surrounding fuel channel, also called a box. A PWR assembly has more rods and lacks a box. The assemblies also contain structural components made of various metal alloys.

The facilities which are required for the handling, intermediate storage and final disposal of unreprocessed spent fuel are:

- a central fuel storage facility in which the fuel can be stored while awaiting final disposal,
- an encapsulation station in which the fuel and the metal components of the fuel assemblies are enclosed in a durable canister prior to final disposal,
- a final repository,
- a transportation system for transports of the fuel from the nuclear power stations to the central fuel storage facility and from there to the encapsulation station at the final repository.

The central fuel storage facility and the transportation system are described in the appendix to the siting and licensing application for the central fuel storage facility submitted by the Swedish Nuclear Fuel Supply Company (SKBF) in November of 1977 to the Government. During transport, the fuel is contained in special transport casks which fulfil the requirements stipulated by current international regulations - see also chapter 2, Volume III of the KBS report on vitrified waste from reprocessing.

The facilities are designed to handle spent fuel from 30 years of operation of 10 BWR and 3 PWR reactors. The total quantity of spent fuel is estimated to be about 40 000 BWR assemblies containing 2.5 million fuel rods and about 4 700 PWR assemblies containing 1.1 million fuel rods, equivalent to a total of about 9 000 tonnes of uranium.

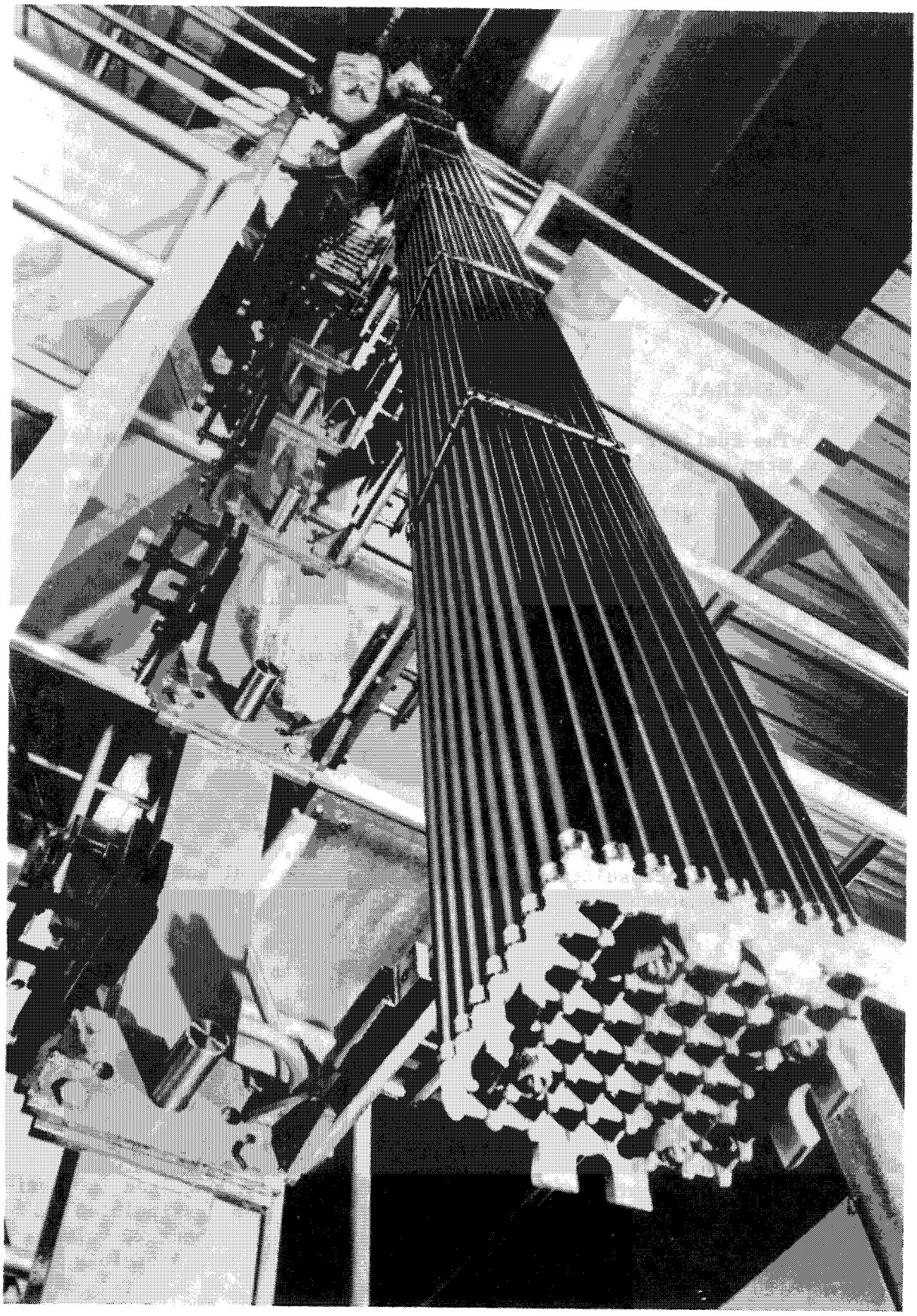


Figure 5-1. Fuel bundle for a BWR assembly. (Photo from ASEA-ATOM).

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

With regard to working environment and protection, see 5.5.

5.2 CENTRAL FUEL STORAGE FACILITY

Regardless of whether the spent fuel is to be reprocessed or disposed of without reprocessing, additional storage capacity is required for spent fuel. The reason for this is that the storage spaces which are presently available at the nuclear power stations and which only correspond to spent fuel from a few years of operation will be full before sufficient reprocessing capacity or facilities for the final disposal of spent fuel are available. For economic reasons, a central storage facility is preferable to an expansion of storage capacity at the individual nuclear power plants.

The design of the central fuel storage facility which is described in the aforementioned siting and licensing application is now being further refined by SKBF. The goal is a completed facility by 1984. Planned storage capacity will be equivalent to 3 000 tonnes of uranium, which covers needs up to 1990. The storage period will be up to 20 years.

The age of the fuel at the time of deposition in the final repository is an important factor in the design of a final repository for spent fuel. The longer the fuel has been stored prior to final deposition, the lower is its heat flux.

In the final repository, the fuel is enclosed in a copper canister. The amount of heat emitted by the fuel determines how much fuel can be stored in each canister in order that a given maximum canister temperature is not exceeded and how densely the canisters can be deposited in the rock in order that a given maximum temperature increase in the rock around the final repository is not exceeded.

An excessively high canister temperature can have an adverse effect on the buffer material which surrounds the canister. A large temperature increase in the rock can lead to undesirable rock stresses and groundwater movements. In order to limit these effects, the KBS project has made it an essential design prerequisite that the heat load be kept at a low level.

In the proposal presented here, it has been assumed that the fuel is stored for 40 years prior to deposition in the final repository. However, the storage period can be varied, and its length is largely a technical-economical optimization question. A shorter storage period requires more canisters and a larger final repository and vice versa. The proposed length of the storage period is not based on detailed studies, but rather on a general judgement that it entails a reasonable optimization of handling and final storage.

The storage principle for the spent fuel in the central storage facility is the same as at the power stations, i.e. the fuel assemblies are stored in water pools, whereby the water provides sufficient cooling and radiation shielding. The integrity of the fuel is thereby dependent upon the durability of the material (zircaloy) which is used for the cladding on the fuel rods. Zircaloy-clad fuel has been used since the 1950s and experiences from the storage of such fuel in water pools are good. No degradation mechanisms which could affect the integrity of the fuel within a period of 40 years have been identified.

There is thus no reason to depart from the storage principle which is applied at the power plants and in the proposed central storage facility if the storage period is 40 years. An extension of this storage period is possible. If all fuel from 13 reactors, each operating over a period of 30 years, is to be directly deposited without reprocessing - which is the hypothetical premise for this report - a storage capacity of about 9 000 tonnes is required. A trebling of the planned capacity of the central fuel storage facility would thereby be required, whereby the facility would have a design as illustrated by fig. 5-2.

For a more detailed description of the design of the facility, see the siting and licensing application referred to under 5.1.

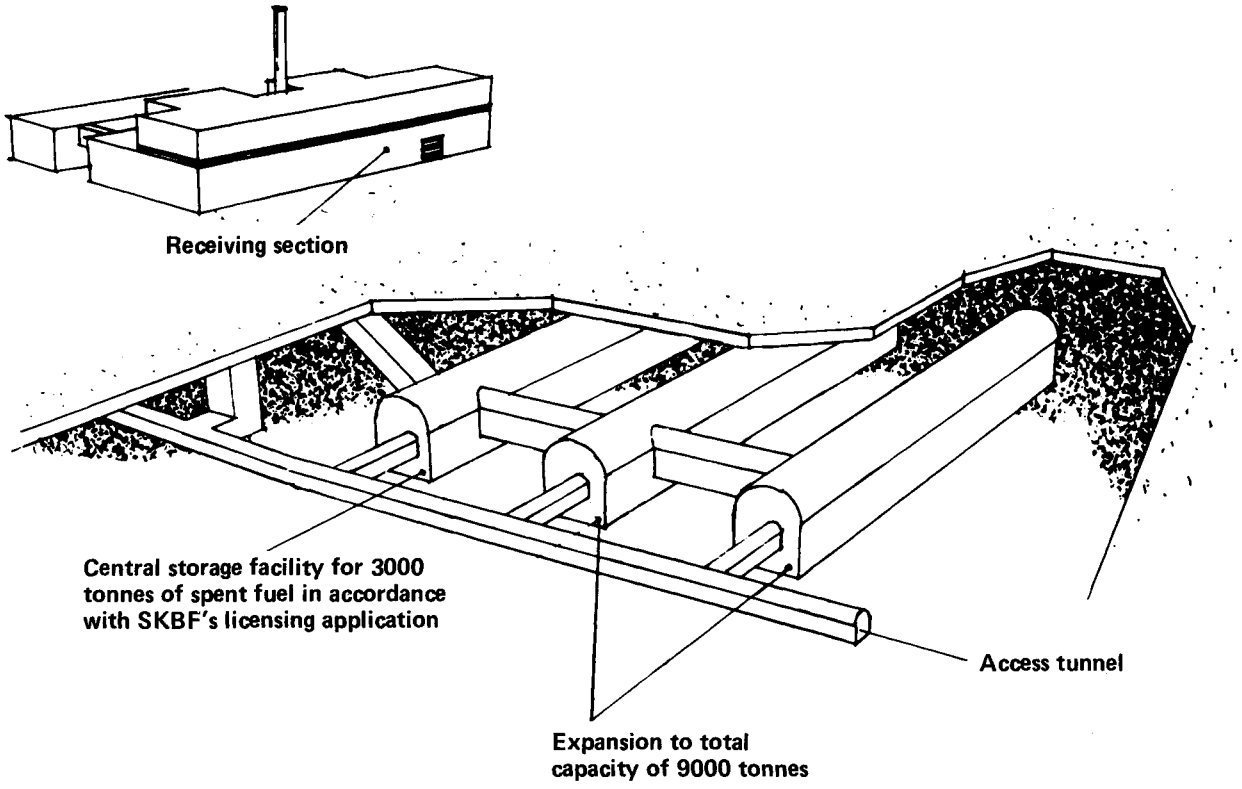


Figure 5-2. Central fuel storage facility expanded to capacity of 9 000 tonnes.

5.3 ENCAPSULATION STATION

5.3.1 General

An encapsulation station will be built in connection with the final repository. The facility will consist of process and service buildings located on the surface, see fig. 5-3.

The spent fuel arrives at the encapsulation station following storage in the central fuel storage facility. Here, the fuel rods are enclosed in copper canisters and the metal parts of the assemblies in concrete moulds prior to final disposal. The copper canister provides a long-term protection against the groundwater in the final repository. It also affords radiation shielding which simplifies handling and reduces radiolysis of the groundwater around the copper canister to a low level, which is important in limiting corrosion. The concrete also affords radiation protection, which simplifies handling. In addition, it provides protection against the effects of the groundwater, primarily by raising the pH of the water, which limits the rate of dissolution of Ni-59, which is the most important isotope in the metal components of the fuel assemblies from the viewpoint of safety.

The facility has a capacity of 8 canisters per week, each containing fuel corresponding to approximately 1.4 tonnes (BWR) or 1.1 tonnes (PWR) of uranium, which provides a good margin to a deposition pace which corresponds to the flow of fuel from 13 reactors (approx. 300 tonnes/year).

Prior to encapsulation, the fuel assemblies are dismantled in order to permit better utilization of the canister's cavity. Only the fuel rods and fuel spill are sealed in copper canisters, while boxes, spacers, tie plates and other parts of the fuel assemblies are cast in concrete moulds in a part of the facility specially equipped for this purpose.

The total number of canisters is about 7 000 and the number of concrete moulds is 1 200.

For a more detailed description of the encapsulation station, see section 2.3 of Volume II with appurtenant drawings.

5.3.2 Description of facility

The layout of the process building is shown in fig. 5-4. It can be functionally divided into a receiving section with associated storage and fuel dismantling stations, an encapsulation section including a casting cell, a cooling cell and a welding cell and an auxiliary systems section. It also contains equipment for handling metal components etc. from the fuel assemblies and for fuel spill.

Connected to the process building are buildings for administration and service. The process building, which is classified as a controlled area, is entered via the entrance building. From here, a connection is provided to the final repository via a hoist shaft.

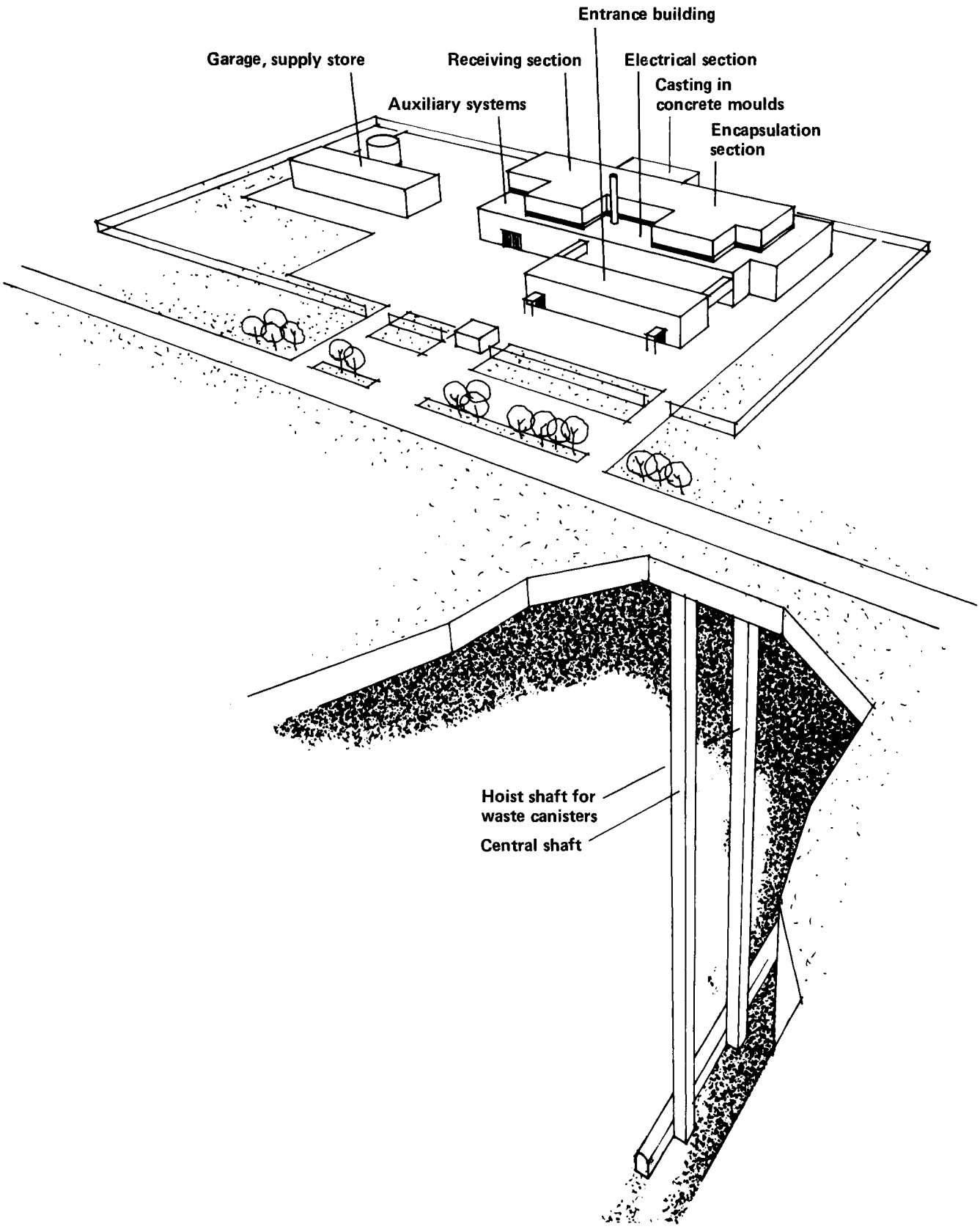


Figure 5-3. Encapsulation station for spent nuclear fuel. The facility is located at ground level above the final repository.

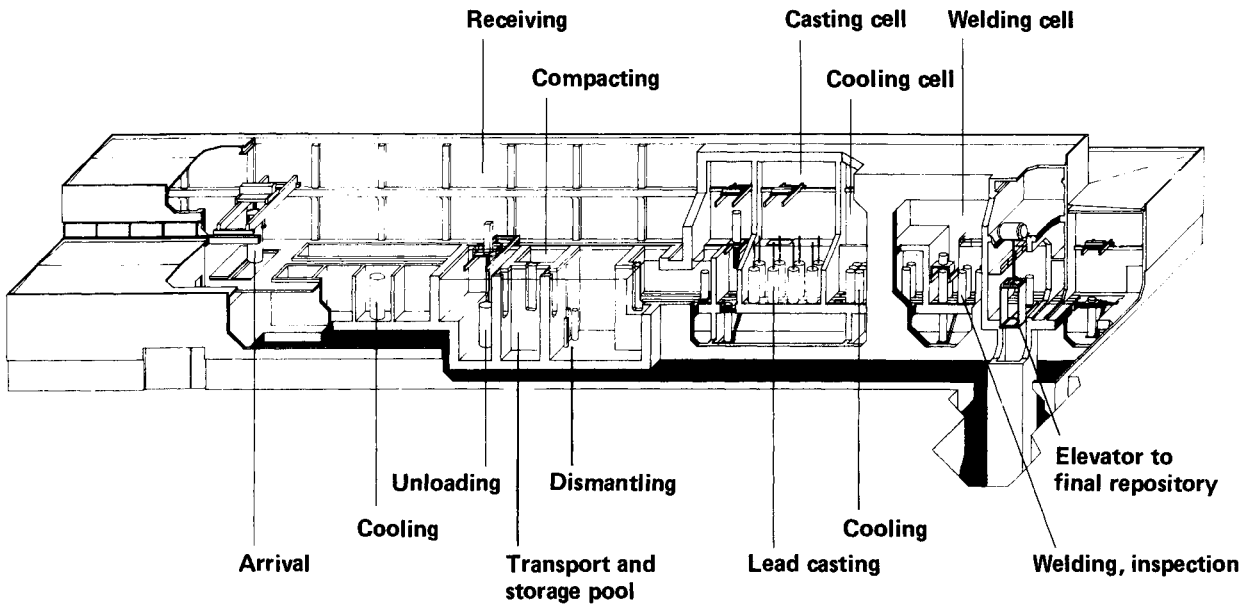


Figure 5-4. Perspective drawing of encapsulation station's process building.

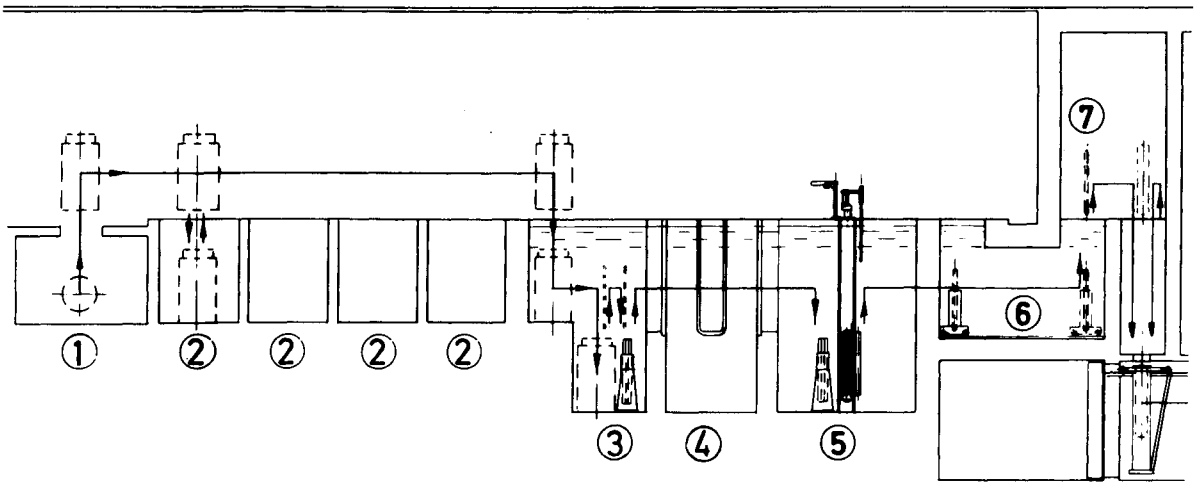
The fuel arrives at the facility in a transport cask on a trailer (or on a railway wagon). A cask contains the equivalent of about 3 tonnes of uranium. The flow of fuel from 13 reactors corresponds to about 100 casks per year.

The various operations included in the handling sequence in the receiving section are illustrated in fig. 5-5.

The transport cask containing the spent fuel arrives at the station's receiving section, where it is lifted off 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 assemblies are lifted out of the transport cask and placed in cassettes. The fuel cassettes are transported under water by an overhead crane either directly to the dismantling pool or to the intermediate storage pool to await dismantling.

In the dismantling pool, the fuel assemblies are dismantled and the fuel rods are moved to a copper rack. Each rack holds 498 or 636 rods, depending on the type of fuel. Other components of the fuel assemblies (spacers, bottom and top tie plates, boxes etc.) are transferred to the compacting pool in the receiving section to await further processing as described below.

When a rack has been filled with fuel rods, it is transferred to a transfer carriage for transport to the encapsulation section. Up to this point, the fuel has been handled and stored under water with sufficient water coverage to provide adequate radia-



1 **Arrival hall**
2 **Cooling and washing**
3 **Unloading of transport casks**
4 **Transport and storage pool**
5 **Dismantling of fuel assemblies**
6 **Transfer carriage**
7 **Drying of fuel rack. Drying and compacting of fuel spill**

Figure 5-5. Schematic illustration of handling procedure in receiving section of process building.

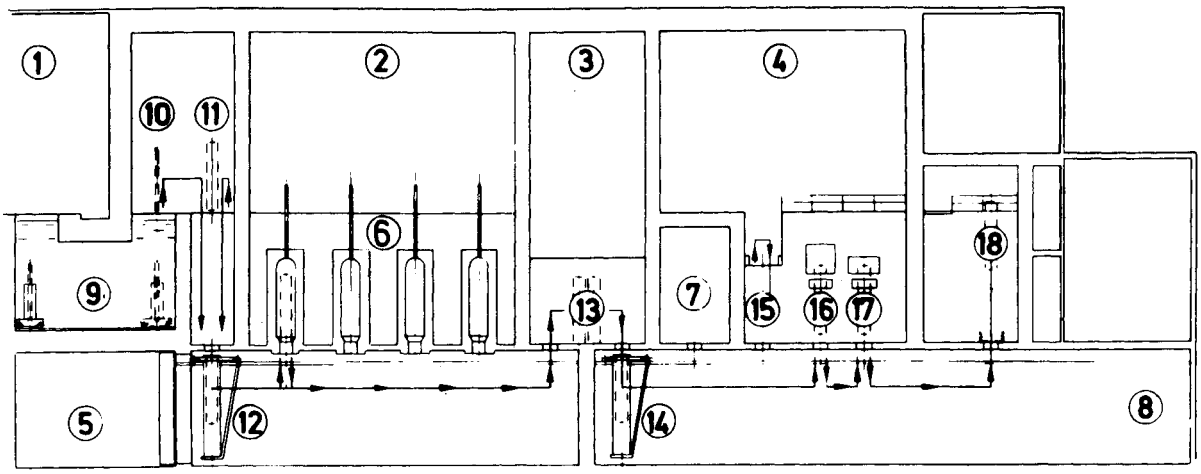
tion shielding. From now on, the fuel is handled in air in cells via remote control, whereby the personnel are shielded from radiation by thick concrete walls.

Handling in the encapsulation section is illustrated by fig. 5-6.

After the rack with the fuel rods has been lifted up from the transfer carriage and allowed to dry in the air, it is placed in a copper canister, which stands in a transport wagon. The canister is then taken to a casting cell, where the spaces between the fuel rods and between the rods and the canister are filled with molten lead. The filled canister is then moved via a cooling cell to the welding cell. In the welding cell, the canister is fitted with a lid which is joined by means of electron beam welding. The weld is tested by means of ultrasonic inspection and helium leakage tracing. The finished canister, fig. 5-7, is then transferred to a wagon for transport down to the final repository. A filled canister weighs about 20 tonnes.

Fuel spill, which can be shaken loose from damaged fuel rods during handling in the encapsulation station, is collected in copper cans which are then enclosed in copper canisters by means of a procedure similar to that described above. Based on extensive experience from the handling of fuel, it is estimated that one canister could hold all spill from the fuel for which the final repository is designed.

Metal components etc. from the fuel assemblies are cast, after



1	Receiving section	11	Arrival of copper canister from store
2	Casting cell	12	Transport wagon
3	Cooling cell	13	Cooling station
4	Welding cell	14	Transport wagon
5	Service area for transport wagon 12	15	Mounting station for lid
6	Furnaces for filling canister with lead	16	Welding station
7	Cell for opening sealed canister	17	Inspection station
8	Service area for transport wagon 14	18	Dispatch station
9	Transport lock		
10	Drying of fuel rack. Drying and compacting of fuel spill		

Figure 5-6. Schematic illustration of handling sequence in encapsulation section of process building.

compacting, in concrete to form cubical moulds measuring 1.6 metres on a side. One mould weighs about 10 tonnes, and a total of about 1 200 moulds are required. The facility where this operation is performed is located in connection with the compacting pool in the receiving section and is connected with this pool via a water lock.

5.3.3 Operation of facility

All handling of fuel and fuel assembly components is done via remote control, either under water, whereby the water provides sufficient radiation shielding, or in radiation-shielded cells. Activities in the cells are supervised through lead glass windows. Known technique and experiences from similar systems in existing facilities are applied in the facility.

When the encapsulation station is no longer required and there is no fuel left in it, the facility will be decontaminated and all "home" radioactive waste, contaminated scrap and building material will be taken to facilities which are equipped to receive and process such material. The facility can then be modified for other use or demolished.

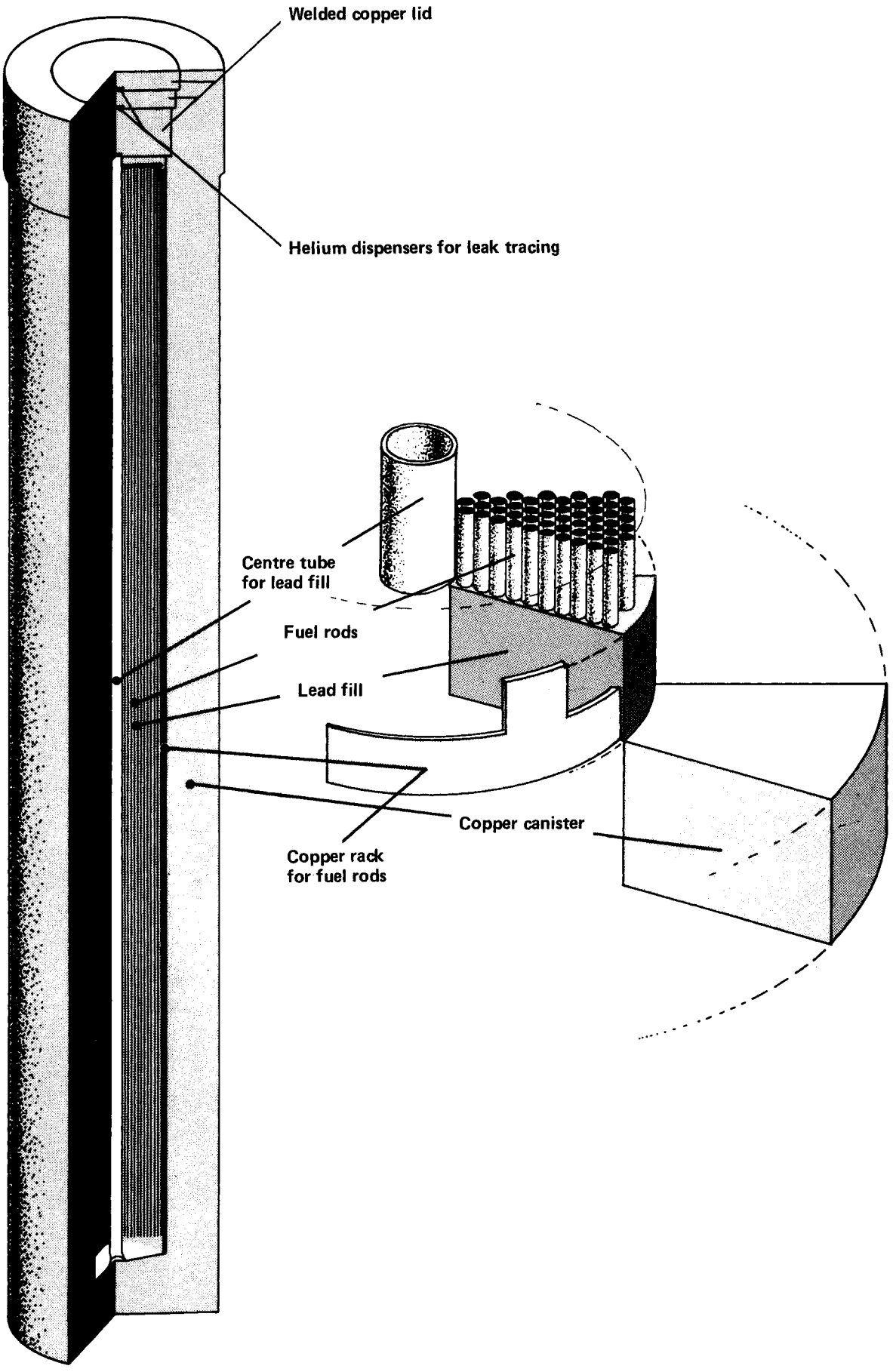


Figure 5-7. Cutaway drawing of copper canister filled with fuel rods. After dismantling of the fuel assemblies, the fuel rods are placed in a cylindrical rack whose perforated copper shell encloses the fuel rods. The filled rack is covered with a perforated lid and lowered into the canister. The centre tube is used for lifting and lead casting.

5.4 FINAL REPOSITORY

5.4.1 General

In the final repository, the encapsulated waste is received for final deposition. The facility is situated in rock under-neath the encapsulation station at a depth of about 500 metres below the surface, see fig. 5-8. (With regard to the final storage of concrete moulds containing radioactive metal components, see 5.4.5 below.)

The design of the final repository is similar to that described in the KBS report on vitrified waste from reprocessing, with the principal difference that the canister in the storage hole is surrounded by highly compacted pure bentonite instead of a sand/ bentonite mixture. The reason for this and for the fact that the fuel is encapsulated in copper instead of lead and titanium is that the radioactivity of the spent fuel decays at a much slower rate than that of vitrified waste from reprocessing. The design of the canister ensures the longer life which is commensurate with the higher demands on the durability of the containment which isolates the unreprocessed spent fuel from the biosphere.

The final repository has been designed for a total capacity of some 7 000 canisters containing a total quantity of fuel equivalent to approximately 9 000 tonnes of uranium.

For a more detailed description of the final repository, see section 2.4 of volume II with appurtenant drawings.

5.4.2 Description of facility

The final repository for the fuel canisters consists of a system of parallel storage tunnels located within a roughly square area of slightly more than 1 km^2 at a depth of about 500 metres below the surface, with appurtenant transport and service tunnels and shafts. The geometric configuration of the tunnel system will be adapted to the geological conditions prevailing at the selected site.

Vertical holes (diameter 1.5 m, depth 7.7 m) drilled in the floors of the storage tunnels constitute the final storage compartments for the waste canisters, see fig. 5-9. Each hole is intended for one canister.

The various operations included in the handling procedure in the final repository are illustrated in fig. 5-10.

The canister is transferred from the encapsulation station to the final repository in a transport wagon which takes it down to the level of the storage tunnels via an elevator in a vertical hoist shaft. The elevator is equipped with extensive safety devices.

Outside of the elevator, at the level of the repository, the canister is transferred to a special vehicle which then transports the canister to the place where it is to be deposited. The storage hole is lined with blocks of highly-compacted bentonite. The canister is lowered into the hole by the equipment on the vehic-

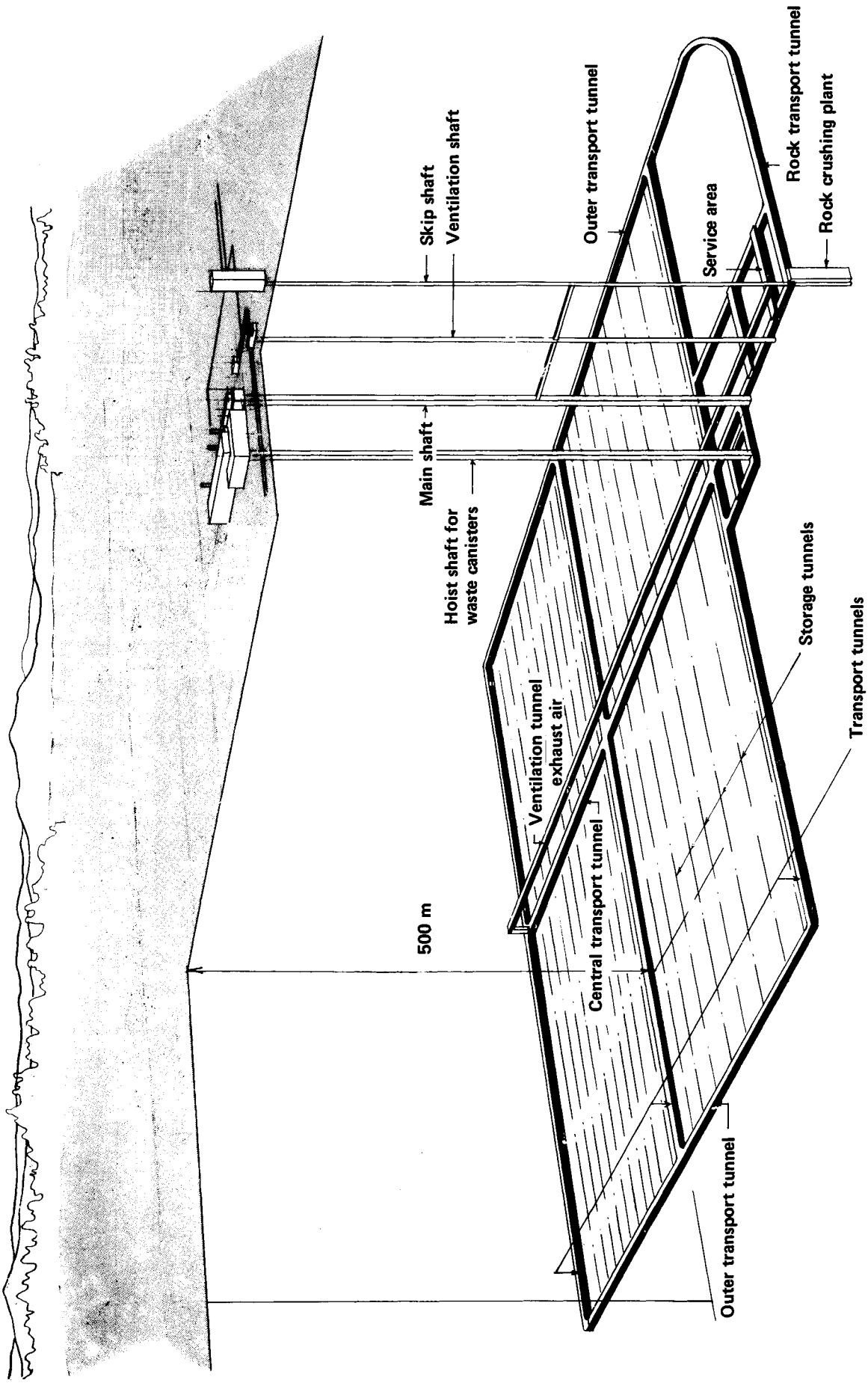


Figure 5-8. Perspective drawing of final repository. The encapsulation station is located at ground level. The final repository consists of a system of parallel storage tunnels situated 500 metres below the surface.

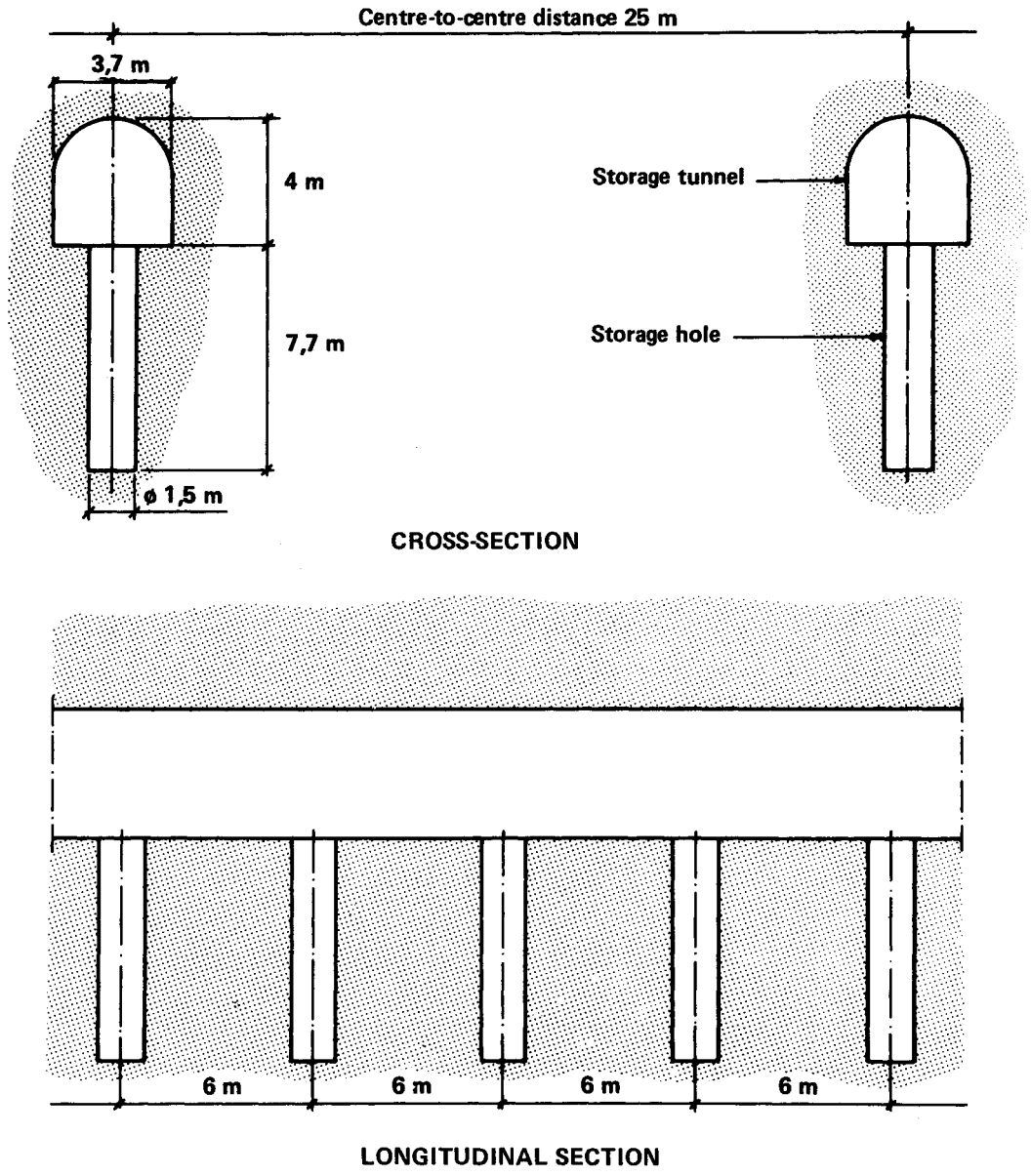


Figure 5-9. Cross-section and longitudinal section of storage tunnels in the final repository. Each storage hole is designed for one canister.

le. The hole is then filled with additional bentonite blocks. The spaces between the bentonite blocks and the rock and between the blocks and the canister are filled with bentonite powder. Finally, a temporary concrete lid is laid in place as a protection until the tunnel is backfilled, see fig. 5-11.

Bentonite is a naturally occurring clay of volcanic origin which is characterized by a high swelling capacity when it absorbs water. It also possesses a high ion exchange capacity. The bentonite blocks which are used in the storage holes are produced by compressing pure bentonite under very high pressure. The highly-compacted bentonite which is thereby obtained possesses extremely low water permeability and good bearing capacity. Since swelling of the bentonite in the hole is restrained, a swelling pressure is obtained when the bentonite absorbs water. As a result, water-bearing fissures cannot open in the material, and any fissures which may exist in the walls of the hole at the time of deposi-

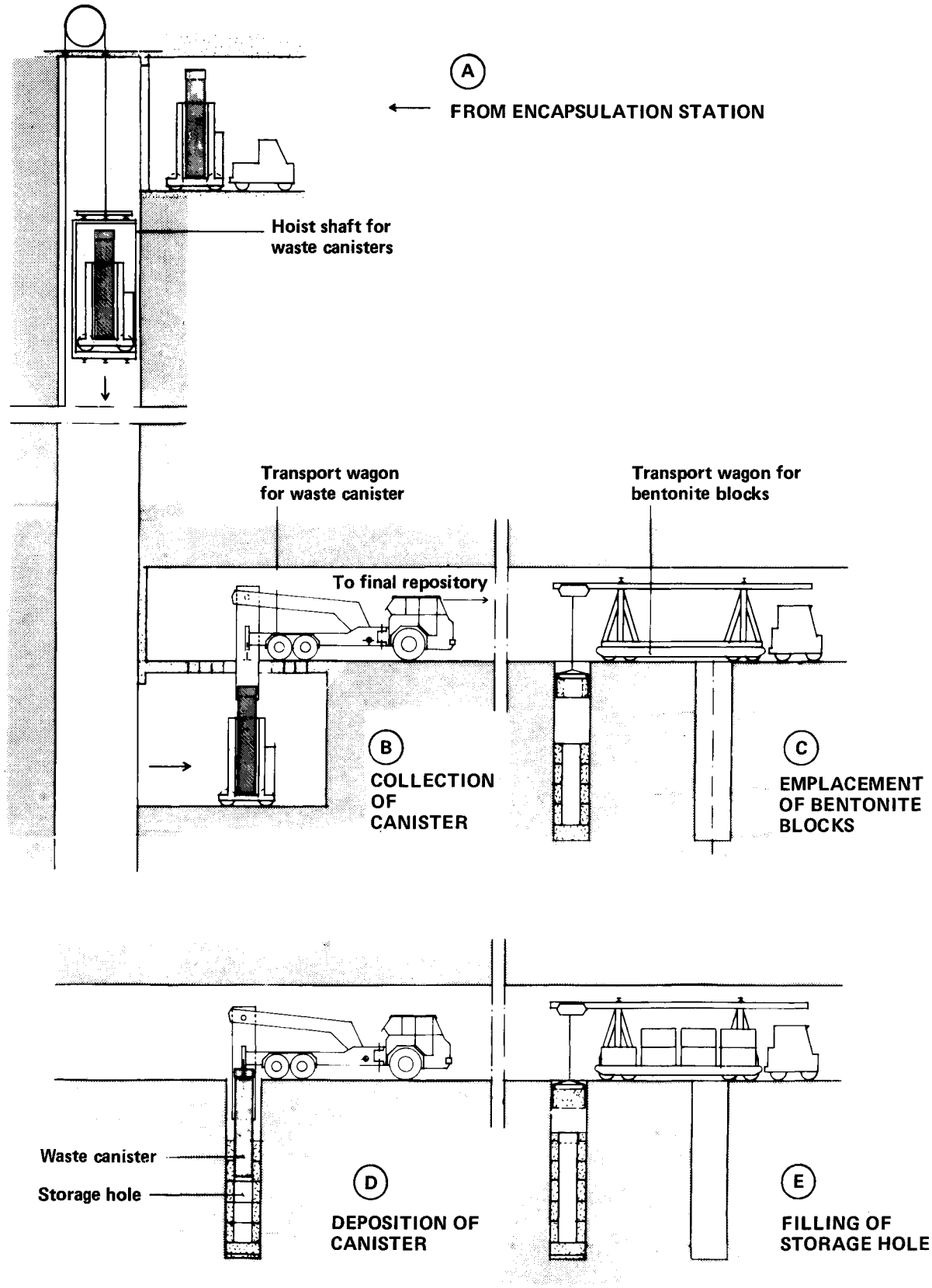


Figure 5-10. Handling of waste canister in final repository.

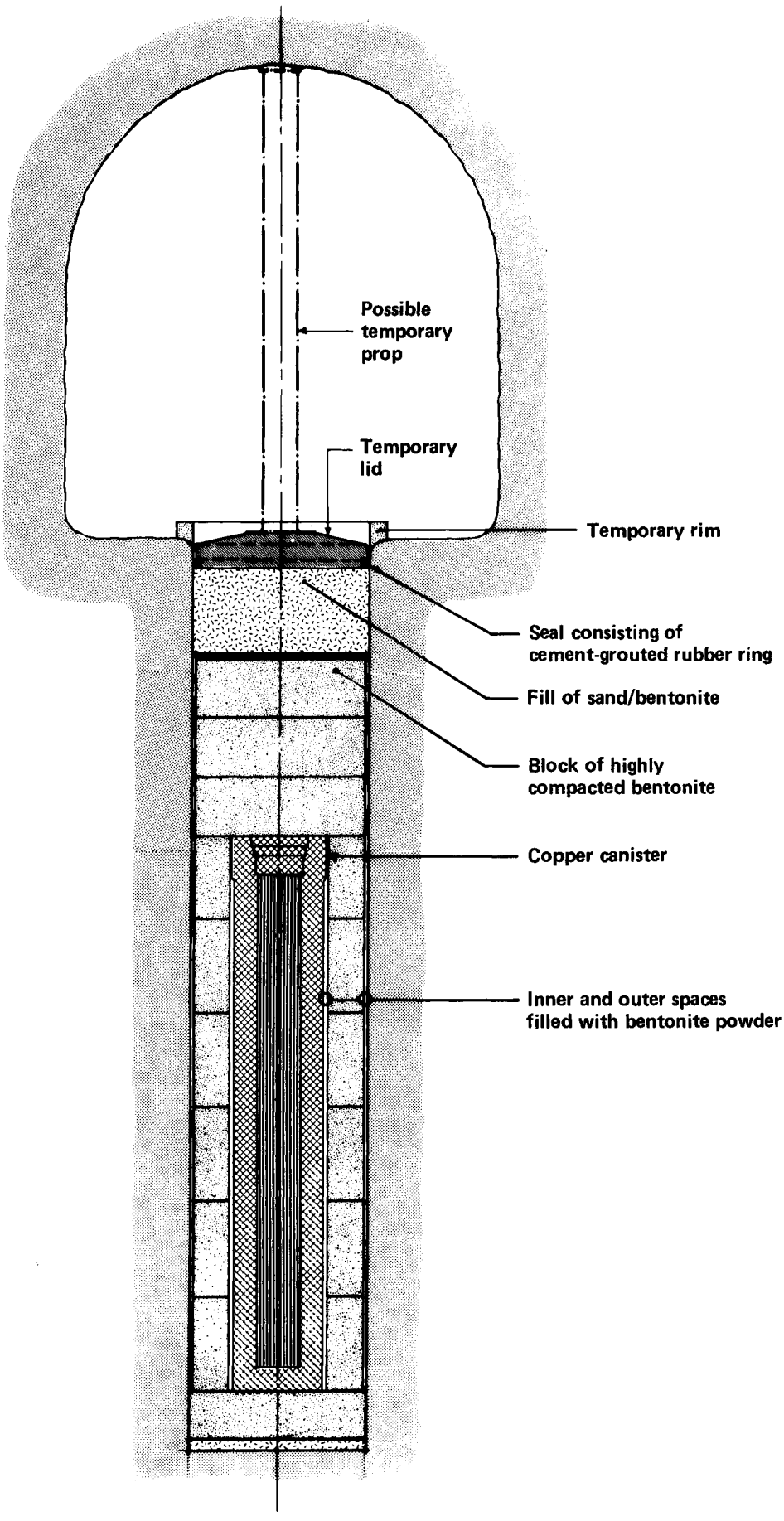


Figure 5-11. After deposition, the storage hole is sealed with a concrete lid. The lid can be propped against the rock roof to counteract any swelling of the bentonite.

tion or which may be created at a later time are sealed. For a more detailed account of the properties and function of the highly-compacted bentonite, see chapter 7.

5.4.3 Operation of facility

Deposition of the waste canisters begins when approximately one-fourth of the storage tunnels have been completed. The facility is designed in such a manner that the continued construction work is completely separated from the canister handling work.

The canisters are handled by means of remote control and with radiation shielding. However, the copper canister itself provides such good radiation protection that brief work can be carried out near the canister without any additional shielding.

The storage tunnels in which the canisters have been deposited can be inspected and measurements of rock stresses, temperatures, inflow of groundwater etc. can be carried out all the way up until the time when the final repository is to be sealed. During this time, the inflow of water into the storage holes is so small that the bentonite will not become water-saturated. Therefore, only a low swelling pressure can build up prior to sealing. This pressure can be absorbed by the temporary lid, which can, if required, be supported by a temporary prop against the tunnel roof.

5.4.4 Sealing of facility

When the final repository has been filled with canisters to design capacity, the facility can be kept open and inspected as long as surveillance is considered desirable. The facility can then be sealed and finally abandoned.

Prior to sealing, the temporary lids on the deposition holes and the concrete rims are taken away. A copper plate can than be placed on top of the fill in the hole to serve as a diffusion barrier. Even if the temporary lid is affected by the swelling pressure from the fill in the storage hole, the fill will not swell up when the lid is removed. In order to swell in this manner, the bentonite would have to absorb additional water, which is a very slow process.

When the repository is sealed, tunnels and shafts are filled with a mixture of quartz sand (80-90%) and bentonite (10-20%). Certain sections, as well as test holes, are filled with pure compacted bentonite.

For a description of the properties of the sand/bentonite mixture, see section 6.3, Volume III of the KBS report on vitrified waste from reprocessing.

When the sand/bentonite mixture used as a tunnel fill is being prepared, 0.5% ferrophosphate is added to serve as a so-called "oxygen-getter" (see chapter 8).

The lower part of the tunnels is filled using conventional earth-moving and compacting methods. The upper part is filled using a

spray technique which has long been employed for spraying concrete. The fill application technique and the swelling properties of the bentonite permit the tunnel section to be filled completely with a high degree of compaction, fig. 4-2.

In this manner, all cavities and voids in the rock are filled with a material which possesses at least as low permeability as 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 consultation with concerned authorities.

5.4.5 Final storage of radioactive metal components etc.

The concrete moulds with radioactive metal components and other components from the fuel assemblies are also intended to be deposited in final storage in rock caverns. However, since the radioactivity of this type of waste is considerably lower than that of the fuel, demands on the encapsulation and the buffer material are lower, and the material does not have to be stored at as great a depth below the surface. Furthermore, owing to the limited quantities involved and the negligibly low heat flux of the material, the final repository requires a relatively small volume.

It is proposed that the facility be designed with two parallel tunnels approximately 250 metres in length and approximately 50 metres from each other. These tunnels are connected with the surface via a hoist shaft and a main shaft, see fig. 5-12. The tunnels have a cross-sectional area of about 50 m^2. The repository is situated approximately 300 metres below the surface.

The final repository for the concrete moulds is shown here as a separate facility, but it could also conceivably be situated in connection with the final repository for the fuel canisters or with the planned facility for final storage of medium-active waste.

The moulds are transferred to the final repository in a radiation-shielded motor-driven wagon which is brought down to the level of the repository in an elevator. They are stacked two-high and three abreast in the storage tunnels on a concrete floor. They are then covered with concrete planks on top and on the sides. The planks constitute a supplementary radiation shielding which facilitates the work with the sealing of the storage tunnel. The tunnel is sealed in stages by filling of all spaces between and around the moulds with concrete as deposition proceeds. When the entire repository is full, transport tunnels and shafts are sealed with sand/bentonite in the same manner as the final repository for fuel canisters.

5.5 PROTECTION

The word "protection" is used as a collective term to cover working environment, rescue service, radiation protection, physical

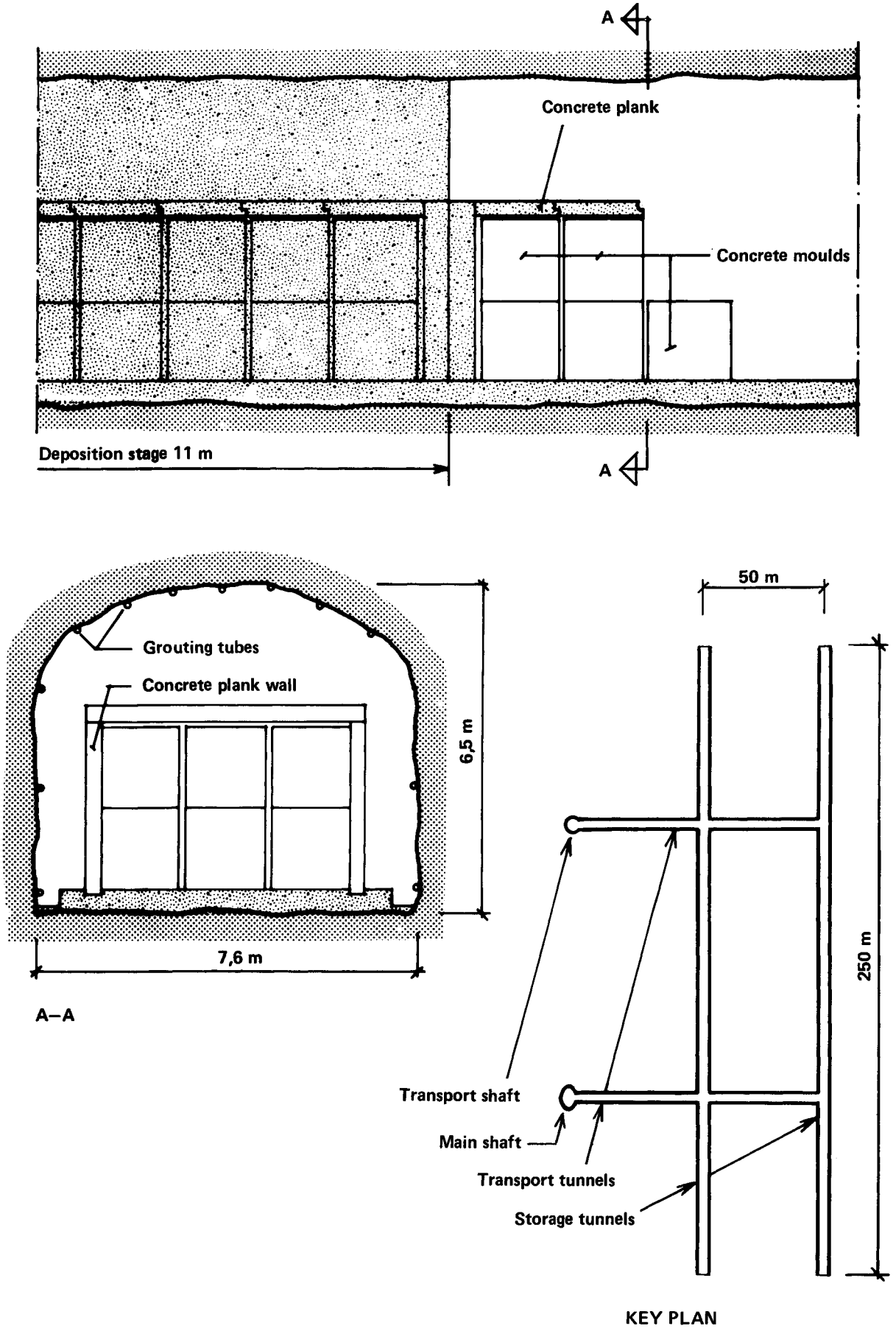


Figure 5-12. Final repository for concrete moulds containing radioactive metal components from the fuel assemblies. The repository is sealed in stages of 11 metres by filling of the tunnel with concrete. In order to guarantee good contact between the concrete and the rock, concrete is grouted through tubes along the rock wall.

protection and wartime protection. These matters have been dealt with in the KBS report on the handling and final storage of vitrified waste from reprocessing (Volume III, chapter 7). That report also applies in principle to the final storage of unreprocessed spent fuel. The differences which exist and which affect the question of protection are mainly:

- that the dry intermediate storage of the vitrified waste from reprocessing has been replaced by an extended period of storage in pools in an expanded central storage facility,
- that the receiving section and a large portion of the auxiliary systems for the central fuel storage facility plus the entire encapsulation station have been placed on the surface,
- that the encapsulation procedure involves the handling of spent fuel instead of vitrified waste, entailing relatively complicated work operations, of which, however, some experience has been gained from the operation of nuclear power plants.

6 GEOLOGY

6.1 BACKGROUND

The general background for the geological and hydrogeological studies presented in the KBS report for vitrified waste from reprocessing applies in all essential respects to the alternative method of final storage of unreprocessed spent nuclear fuel as well. Since the first report was compiled, the Geological Survey of Sweden, SGU, has concluded the work programme for which is was commissioned by KBS. The present report is therefore based on a more comprehensive body of background material than the previous one.

The aim of the work has been to elucidate the bedrock and groundwater conditions which determine the long-term safety of a storage facility in Swedish Precambrian bedrock. The studies span a number of disciplines. The bedrock at the site which is eventually selected must consist of a suitable type of rock of sufficient extent both horizontally and vertically. The existence of fracture and crush zones can influence the design and safety of the rock repository. As regards the groundwater, information is needed on its chemical composition, how much water can come into contact with the waste and how long a time the water resides in the bedrock. It is also important to determine the manner in which the groundwater from the storage site approaches the surface of the ground and how diluted it becomes on the way there, as well as the capacity of the bedrock to retard and retain various waste substances if they should escape into the groundwater.

Test drillings have been undertaken at five sites, of which three - at Forsmark, Oskarshamn and Karlshamn - have been selected for further study, see fig. 6-1.

An experimental station was established at Stripa in a granite massif at a depth of 360 metres. Measurements of rock stresses, permeability and thermal conductivity have been carried out. Data on the composition of the groundwater have been collected. A full-scale heating experiment has been commenced in cooperation with the Lawrence Berkeley Laboratory and is expected to continue until the end of 1979.

Field tests have been conducted at Studsvik in order to study the retardation of the radioactive elements as they are transported with the groundwater through fissured rock.

The chemical environment in and around a rock repository for

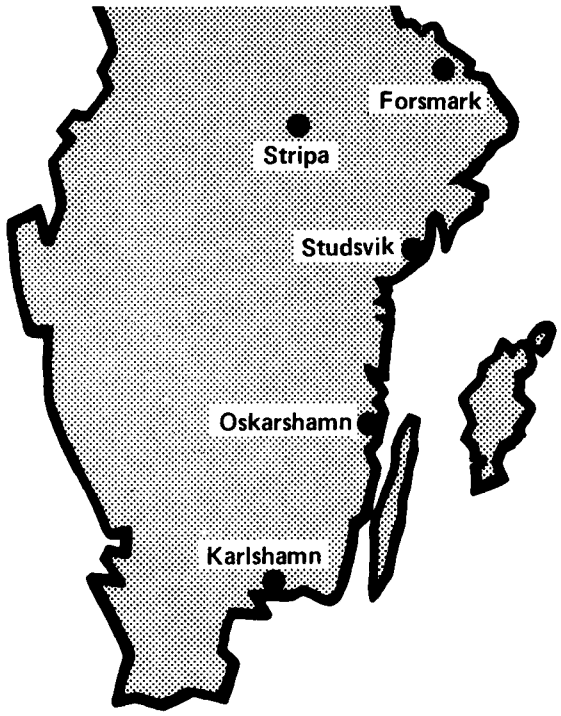


Figure 6-1. Map showing study areas. Test drillings to a depth of about 500 metres were undertaken at Karlshamn (Sternö), north of Oskarshamn (Kråkemåla and Ävrö), and at Forsmark (Finnsjö Lake and Forsmark). The KBS experimental station is located in the Stripa mine. Field studies were carried out at Studsvik.

high-level waste is an important factor in determining the durability of the canister and the buffer material as well as the possibilities of dispersal of the waste substances. These questions can be explored with the aid of chemical equilibrium calculations, based on the composition of the groundwater and knowledge of the natural occurrence of the various materials involved. In some cases, it can be shown that their natural state of occurrence has remained unchanged for many millions of years, thereby providing answers to questions concerning the durability of the materials.

Continued work on behalf of KBS has further elucidated the chemistry of the groundwater by means of studies and measurements of important equilibria and new analytical data. Special attention has been devoted to questions concerning how uranium, plutonium and other long-lived actinides behave in the groundwater. New laboratory measurements and measurements of Swedish groundwater, as well as analytical and theoretical studies of the minerals in the bedrock, show that these elements will be virtually insoluble in the deep groundwater. This will prevent their dispersal to the surface of the earth. These results are further verified by conditions at some foreign uranium deposits where the ores have been subjected to groundwater flow for many millions of years.

The geological material is presented in greater detail in chapter II:3.

6.2 EVALUATION OF BEDROCK CONDITIONS

Within the framework of KBS's geological study programme, the natural conditions for a deep rock repository for vitrified high-level waste or spent nuclear fuel have been studied in three separate areas. The obtained data are summarized and evaluated in the following sections.

The borehole at Karlshamn shows that the local rock type, Blekinge coastal gneiss, which is observed on the surface extends down to a depth of at least 500 m. The borehole also shows that this rock possesses extremely low permeability and that the vertical distance between water-bearing zones in the rock exceeds several hundred metres. Rock caverns within the area and in its vicinity show, over large areas and volumes, that the infiltration from existing fissures, even at shallow depths, is unusually low. In addition, water-bearing zones within this area are so widely spaced horizontally that a number of large rock caverns could be built without any problems with water infiltration. The stability and low reinforcement costs reported for these caverns show that the Blekinge gneiss is favourable from the constructional point of view, that any existing rock stresses here do not give rise to any engineering problems and that the effects of previous ice ages on the bedrock in the area has been limited to surface features and the uppermost surface layer of the rock. Existing rock caverns for the storage of heavy oil at elevated temperature serve as a large-scale, long-range experiment which demonstrates in actual practice that the bedrock here can withstand the heat load arising from a waste storage programme in accordance with the concept proposed by KBS.

These observations prove that a deep-lying rock repository could be constructed here. The natural rock conditions are such that each individual waste canister in such a repository can be surrounded by many metres of rock of extremely low permeability. As is shown in section II:3.4 concerning the volume and velocity of the groundwater, this means that the quantities of water which can come into contact with the canisters are very small, and that the flow time to the surface is very long. As is examined in greater detail elsewhere (II:6 and 7), this, along with the retardation of the waste substances in the rock, means that most of the waste elements will decay before they reach the biosphere. This means that even if there were no encapsulation and the groundwater came into direct contact with the waste, the chemical and mineralogical composition of the bedrock would ensure that long-lived elements such as uranium, plutonium and other actinides would remain virtually insoluble in the water.

The geological history of the Precambrian bedrock, as well as local fracture studies, show that the bedrock conditions and the local permeability of the rock will not change appreciably over the next few million years, and that the probability of fracture movements in the rock which can cause canister damage is extremely small. On the basis of the results, it is maintained that the geological prerequisites for satisfying the Swedish Stipulation Law have been demonstrated to exist.

It must be emphasized that no crush zones have been encountered in the rock in the deep borehole at Karlshamn. Nor have any sig-

nificant crush zones been found in rock caverns here, which have a total combined volume of more than 1 million m^3.

This shows that such zones are widely spaced and probably few in number within the area. Even if crush zones should occur, they would not in themselves pose a threat to an absolutely safe storage of high-level waste, as long as no waste is emplaced directly in such zones. The decisive condition is instead that each individual waste canister can locally be surrounded by a rock volume of low permeability.

Naturally, knowledge of the location and extent of local crush zones is of vital importance in determining the final configuration of a rock repository. For this reason, the rock repository proposed by KBS has been designed as a modular system which can be adapted to local rock conditions. In this way, the question of the location and extent of any crush zones is reduced to an economic question on which the Stipulation Law has no bearing. However, the aforementioned low costs of rock reinforcement encountered in the Karlshamn area indicate favourable conditions in this respect as well. The results of the regional study of the Blekinge coastal gneiss show, moreover, that many other areas should also possess the same favourable character.

The boreholes at Kråkemåla, as well as the Finnsjö 1 and 2 boreholes (near Forsmark) show that the bedrock in these two areas consists of blocks of low permeability which are, however, bounded by water-bearing zones.

The latter comprise between 5 and 20 per cent of the deeper bedrock, and their spacing varies in general from 10 up to a 100 metres or so. Moreover, some indications of high permeability in isolated measuring sections may have been caused by leakage around the packer seals, which would mean that the length of the impervious sections is actually greater than is indicated by the measurements.

Since the boreholes are not located in the same plane, the results show that there are also large volumes of rock at Finnsjö Lake and Kråkemåla with low permeability. Both of these sites therefore offer good opportunities for surrounding the individual waste canisters with several metres of such rock.

The length distribution of the low-permeability sections in the deeper parts of these boreholes is fairly similar in these two areas. This shows that the two areas are roughly equivalent in this respect, and that each of the boreholes is sufficient in itself to provide a representative picture of these conditions at this stage.

The permeability of the water-bearing zones at Kråkemåla is, however, sometimes clearly higher than at Finnsjö. Owing to this factor, along with the pronounced fracture system and elevated natural radioactivity at Kråkemåla, this area must be regarded as less suitable than the Finnsjö area for a rock repository.

The boreholes show that the water-bearing zones in the bedrock consist predominantly of multiple fracture zones, whose width often extends over several measuring sections. The most marked fracture structures in the Finnsjö area, namely the Gåvastbo

fault and the fracture zone which is intersected by the Finnsjö 2 borehole and which passes Finnsjö 1, are of this type. They are also characterized by fairly moderate permeability (between 10^{-6} and 10^{-5} m/s).

The minerals in the fracture zones show that these zones were initiated at a very early stage when these parts of the bedrock were located at greater depth and not at their present shallow location.

Sampling of the groundwater in the boreholes, as well as the number of age determinations and other special analyses carried out on the groundwater samples, have been expanded considerably compared to the situation at the time of the previous KBS report. The results are presented in greater detail in II:3.4.

The boreholes have permitted a determination of the actual elevation of the groundwater table above sea level at points in the study areas. As expected, they show that the groundwater table is considerably flatter than the surface of the land, whose slope has been used in all calculations of groundwater volumes and flow times as a measure of the hydraulic gradient. The hydraulic gradient determined in this manner is only about half of that used in the calculations. This means that the stated groundwater flows should be reduced by half and the flow times doubled.

In calculating the flow times, it has generally been assumed that the permeable zones of the bedrock possess complete continuity with each other and that the water can move constantly in the direction of the theoretical potential field. The borehole results and fracture studies of outcrops show that the latter is not the case. The boreholes do not definitively demonstrate the three-dimensional continuity of the deeper fracture zones, and do not even show that such a continuity actually exists. If the continuity of the deep water-bearing zones is incomplete, then the true flow times would be longer, but no allowance for this has been made in the safety analysis.

As was emphasized above, however, the question of the location and extent of water-bearing zones is not of decisive importance for the safety of the final repository, as long as each individual waste canister can be surrounded by a rock volume of low permeability. In this context, it may also be recalled that KBS' geology programme is not aimed at selecting the final site for a future rock repository, but rather at demonstrating how and where an absolutely safe final storage of high-level waste can be effected. The selected areas therefore comprise examples of sites where a final repository could conceivably be located. Before a final decision is made concerning the location and detailed design of the final repository, extensive studies and tests over many years within possible areas are required.

6.3 SUMMARY OF RESULTS FROM GEOLOGY PROGRAMME

- A survey of the history and evolution of the bedrock in Sweden shows that the bedrock in the Baltic shield has constituted a markedly stable unit in the geology of Europe for more than 600 million years. During the past 25 million years, Europe north of the Alps, as well as adjacent areas

of the north Atlantic, have been developing towards increasing stability. There is therefore virtually no possibility of such widespread rock movements occuring that could bring about such deep weathering or erosion that the integrity or safety of a rock repository located at a depth of several hundred metres in the bedrock could be jeopardized.

- Local fracture movements in the bedrock cannot be excluded, but a closer analysis shows that they will not lead to any appreciable changes in the permeability of the bedrock or to any canister damages when the canisters are emplaced in sound rock.

- Over the past two million years, Sweden has been subjected to between ten and twenty different glaciations. These, and the changes in the level of the land associated with the deglaciation stages, have not lead to any hydraulically coherent fracturing of the bedrock at depth.

  A future ice age cannot differ radically from previous ice ages. Consequently, a future ice age cannot in these respects affect the safety of a deep rock repository.

- Previous land level changes, with accompanying deep groundwater circulation and heavily climate-dependent weathering, have had only a local effect on the bedrock. This confirms that not even extreme changes in ground levels and climate can have any decisive impact on a suitably located rock repository in the Precambrian bedrock.

- Experiences abroad have shown that even severe earthquakes have very limited effects on tunnels and rock caverns. Only weak earth tremours occur in Sweden. Their effect on a deeplying rock repository would be completely negligible. Southeastern Sweden is, moreover, an area with an extremely low earthquake frequency.

- The KBS study area at Karlshamn is made up of a gneiss which is regionally characterized by weak fissuring and low groundwater content. A total of more than one million m^3 of rock caverns are located in the area and its immediate vicinity, which is roughly equal to the calculated volume of a final repository. Data is available for 700 000 m^3 of these rock caverns, which exhibit remarkably low water infiltration and good rock stability. Many of the rock caverns are heated to temperatures which correspond to the temperatures which have been calculated for a final repository for high-level waste without any adverse effects. A borehole in the area reveals consistently good bedrock conditions down to a depth of 500 m and an extremely low water permeability has been measured for the entire borehole between 23 and 500 m. No water-bearing fracture zones have been encountered in the borehole. It can be safely assumed that the bedrock here consists for the most part of rock with extremely low water permeability, which ensures that each waste canister can be surrounded by large volumes of good rock.

- The study areas at Finnsjö Lake, west of Forsmark, and Kråkemåla, north of Oskarshamn, also exhibit large volumes of good rock. This guarantees that each waste canister there

can be surrounded by several metres of rock with low permeability. There are, however, water-bearing fracture zones in both areas, which must be carefully considered in the design of a rock repository.

- The results of the work done since the previous report have confirmed the ranking assigned to the study areas in the previous report, namely Karlshamn, Finnsjö and Kråkemåla, in that order.

- The permeability of good rock has been determined to be equal to or less than $5 \cdot 10^{-11}$ m/s at Stripa and $2 \cdot 10^{-12}$ m/s at Karlshamn. Similar values probably apply to the rock in the other study areas as well, although technical limitations have prevented the measurement of values below $4 \cdot 10^{-10}$ m/s.

  Water-bearing fracture zones within the study areas generally have permeabilities of around 10^{-7} m/s. Higher values were found, however, and values up to 10^{-3} m/s have been measured in isolated zones at Kråkemåla.

- The groundwater flow in good rock in the study areas has been calculated to be 0.2 $1/m^2$ and year, based on the permeability value $K = 10^{-9}$ m/s. If the permeability values for good rock at Stripa and Karlshamn are applied, ten to one hundred times lower water flows are obtained in the impervious rock sections around the waste canisters.

- The groundwater flow pattern is characterized by local flow cells, with downward flow at groundwater divides and upward flow under more pronounced valleys. In between, the groundwater flow is predominantly horizontal. Since both topography and bedrock permeability are largely determined by the structure of the bedrock, the flow pattern of the groundwater in the study areas will persist for a very long time.

- The flow time of the groundwater has been calculated by means of a three-dimensional model over an area of 30 km^2 around the study area at Finnsjö Lake for a water permeability which decreases from 10^{-6} m/s at the surface to $10^{-7.5}$ m/s at a depth of 500 m. Porosity is set at 0.001. A number of 50-metre-wide crush zones with a hundred times higher permeability have been included in the model. The calculations show that a rock repository at a depth of 500 m and with a surface area of 1 km^2 can be located in this area in such a manner that the flow time of the groundwater from the peripheral parts of the repository to the surface of the ground is more than three thousand years. These calculations apply primarily to the water which flows in the fracture zones in the rock. Supplementary calculations show that a few metres of good rock around each waste canister increases the flow time by several thousand years. The flow time for the study area at Karlshamn for groundwater from a depth of 500 m to the surface is probably hundreds of thousands of years.

- On the basis of eleven age determinations carried out on groundwater by means of the carbon-14 method, the flow time

of the groundwater from a repository at a depth of 500 m has been estimated to be around 3 000 years or more.

- Studies of uranium ores and laboratory experiments show that the dispersal of uranium and other actinides by the groundwater is prevented by the reducing conditions which prevail at the depths in question.

- Measurements of the redox potential and oxygen content of Swedish groundwater from great depths, as well as mineralogical and mineral-chemical observations, show that reducing conditions generally prevail in Swedish bedrock. The groundwater at great depths therefore lacks the ability to dissolve and disperse uranium (and other actinides) to any appreciable extent, even if it should come into direct contact with spent fuel in a rock repository.

- Extreme climatic changes give rise to only local changes in the redox potential of groundwater in crystalline rock and cannot hereby affect the safety of a suitably located rock repository.

- The construction and drainage pumping of a rock repository produces only local disturbances of the redox conditions in the immediate vicinity of the rock caverns. Just outside of the affected area, the natural equilibria are restored. The groundwater can therefore cause only a very limited and local dispersal of uranium and other actinides. When a rock repository has been backfilled, such local disturbances will gradually disappear.

7 BUFFER MATERIAL

7.1 GENERAL

Two types of buffer material are used in the final repository: first, a mixture of sand and bentonite similar to that used in the final repository for vitrified waste from reprocessing; and second, highly-compacted pure bentonite.

As was noted under 5.4 above, the sand/bentonite mixture is used to backfill tunnels and shafts. 0.5% ferrophosphate (vivianite) is added to the tunnel fill as a so-called "oxygen-getter" (see chapter 8). For a description of the properties and function of the sand/bentonite mixture, see section 6.3, Volume III of the KBS report on vitrified waste from reprocessing and chapter II:4. The following description deals exclusively with the buffer material of highly-compacted pure bentonite which surrounds the canister in the storage hole and which is used in certain places to seal tunnels and shafts.

The reason why pure bentonite is used in the storage hole instead of a mixture of sand/bentonite as in the vitrified waste alternative is that the demands on the service life of the canister are considerably higher for direct disposal than for vitrified waste from reprocessing. By using pure bentonite of high density, we obtain a buffer material which possesses very low water permeability and the other properties which a good buffer material should have, namely:

- good bearing capacity, so that the canister is held in its position in the storage hole,
- good thermal conductivity, so that the heat generated by the fuel in the canister is transmitted to the rock without the canister being heated to an excessively high temperature,
- good ion exchange capacity, so that the migration of radioactive elements which may leak out from the canister is retarded,
- long-term stability, so that the material retains its properties over the very long period of time during which the function of the final repository is to be maintained.

Another condition is that the buffer material shall not contain components which can decisively reduce the resistance of the copper canister to corrosion.

Besides the above-mentioned properties, bentonite of high density is characterized by the fact that it gives rise to a very high

swelling pressure when the bentonite absorbs water, if swelling is restrained. This provides a guarantee that water-bearing fissures cannot open up in the buffer material. Owing to this swelling pressure, the bentonite also penetrates into and seals fissures which may exist (or which may open up at a later time) in the walls of the storage holes.

Because many of the properties of the buffer material which are fundamental to its function are dependent upon a sufficiently high density being obtained in the storage hole, the bentonite is applied in the hole in the form of highly-compacted blocks of very high density. Under the influence of penetrating groundwater, the bentonite swells and fills out the spaces and cavities resulting from the application procedure. The swelling is restrained by the surrounding rock and the fill in the overlying tunnel so that a very high density is maintained. (The bentonite absorbs water so slowly and groundwater seaps into the storage holes at such a low rate that high swelling pressures cannot arise during the period of time for which the tunnel is kept open.)

Even though many of the most important properties of the bentonite in relation to its function as a buffer material improve with increasing density, there is an upper limit which is dependent upon the fact that exceedingly high swelling pressures can give rise to undesirable stress concentrations in the surrounding rock.

For a more detailed account of the properties and function of the buffer material, see chapter 4, Volume II.

7.2 PROPERTIES

Bentonite is a naturally occuring clay which is characterized by the fact that it swells upon absorbing water. As a reference material, KBS has chosen a bentonite of the type Volclay MX-80, which is mined in Wyoming and South Dakota in the United States. (Other types of natural or synthetic bentonite can probably also be used.) MX-80 is a so-called sodium bentonite whose main constituent (90%) is montmorillonite. It has a large number of uses, for example within foundry and oil-drilling technology. The annual quantities required in the final repository comprise a very small portion of the current annual production and adequate reserves exist.

The bentonite is applied in the storage hole both in the form of blocks (fig. 7-1), which are compacted under 100 MPa of pressure, and in the form of bentonite powder, which is used to fill the spaces between the blocks and the rock and between the blocks and the canister. The bulk density of water-saturated bentonite is about 2.30 t/m^3 for the blocks and about 1.75 t/m^3 for the powder in the spaces. The mean density, when all bentonite in the storage hole has been water-saturated and swelling has ceased, is 2.1 t/m^3. It is determined by the increase in volume which results from the fact that the joints between the blocks are filled up, the bentonite in the spaces (see fig. 4-2) is compressed and

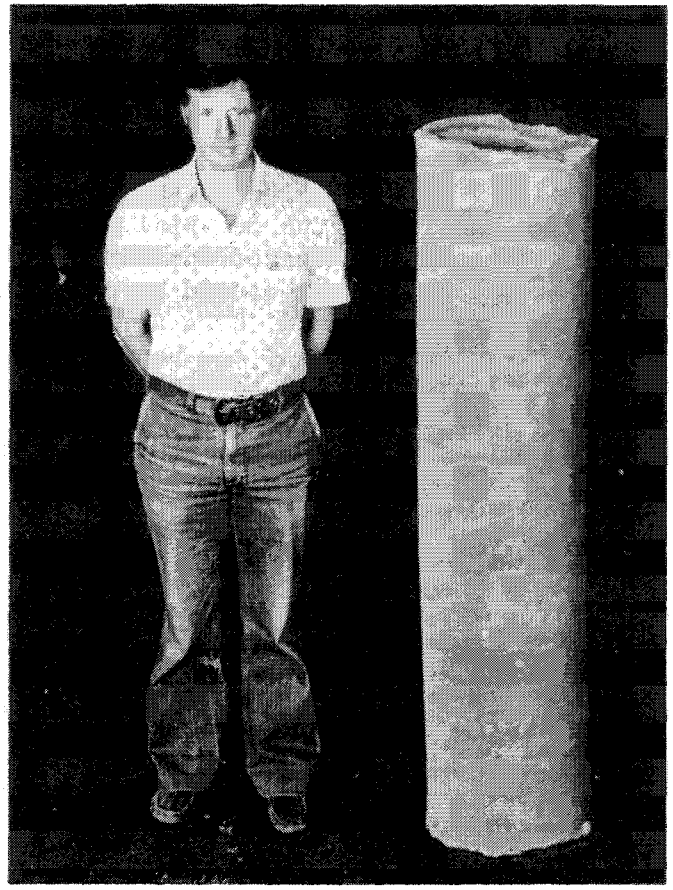


Figure 7-1. Bentonite can be compressed by means of isostatic compaction into large blocks. The photograph shows a test block.

bentonite from the storage hole displaces and compresses the sand/bentonite fill in the overlying tunnel to some extent.

Stability

Natural deposits of bentonite which have been exposed to the temperatures, groundwater conditions and pressures which prevail in the final repository show that the bentonite can be counted on to remain stable for the period of time during which the buffer material must retain those of its properties which are vital to the function of the final repository.

Permeability

The permeability of the bentonite declines with increasing density. At a density of 2.1 t/m^3, the permeability coefficient of the bentonite is about 2×10^{-14} m/s. This means that the material is virtually impenetrable by water and that diffusion is a controlling mechanism for the transport of ions through the buffer material.

Bearing capacity

The blocks of highly-compacted bentonite which are placed in the storage hole possess very high bearing capacity. In appearance

and to the touch, the material resembles soap-stone. Its shear strength is comparable to that of sedimentary rocks. Its water content, i.e. the ratio between the weight of the water and the weight of the solid material, is approximately 10% in the blocks.

Bearing capacity is a function of the material's density. The final average density of the material in the storage hole when it is fully water-saturated (water content about 20%) is, however, so high that the subsidence caused by gravity, even after one million years, is negligible. Even at a bulk density of 1.4 - 1.5 t/m^3, when the consistency of the bentonite is comparable to that of solid modelling clay, its bearing capacity is sufficient to prevent the canister from sinking through the underlying bentonite, even over a very long period of time. This means that, even assuming improbably high material losses through fissures and the like, the bentonite still possesses sufficient bearing capacity.

Swelling pressure

Bentonite, which absorbs water under restrained swelling exerts a swelling pressure which is a function of the density of the material. The ratio between swelling pressure and density has been determined by tests for MX-80 and verified by means of theoretical calculations. At a bulk density of 2.1 t/m^3, the swelling pressure is on the order of 5 MPa.

Its high swelling capacity in connection with water absorption gives bentonite a "self-sealing" capacity in that it is forced into fissures in the walls of the storage holes and prevents the opening of water-bearing fissures or cavities in the buffer material.

Thermal conductivity

The thermal conductivity of blocks of highly-compacted bentonite with a bulk density of 2.0 t/m^3 and a water content of about 10% has been determined to be about 0.75 W/m^oC. The highest temperature at the surface of the canister has thereby been calculated to be 77^oC.

Effect on the resistance of the canister to corrosion

Bentonite can contain sulphides and organic material which can directly or indirectly contribute towards a greater corrosion attack on the canister - see chapter 8. The concentration of these materials should not exceed 200 mg/kg bentonite. If this level is exceeded, the bentonite should be purified by heating in air to 425^oC for 15 hours. Tests have shown that such a heat treatment does not affect the swelling properties of the bentonite.

Ion exchange capacity

Owing to its high content of the clay mineral montmorillonite, bentonite possesses a considerable capacity to retard certain nuclides through ion exchange.

7.3 FUNCTION

Following deposition and during the period of time during which the storage tunnels are kept open and drained, it is probable that no water will be able to penetrate into the storage holes. But even if this should occur, the bentonite powder which fills the spaces between the highly-compacted blocks and the rock will absorb most of the water, since the permeability of this layer is higher than that of the blocks. But it is sufficiently low to greatly restrict further penetration of water after the layer has become water-saturated.

When the final repository has been sealed, the original groundwater conditions are gradually restored. Water then seeps slowly into the hole and is absorbed by the highly-compacted bentonite. As the bentonite absorbs more and more water, its permeability decreases, so that water absorption takes place at an increasingly slow rate. Considering the very low groundwater flow rate and the low permeability of the buffer material and the surrounding impervious (grouted) rock, it will take a very long time (probably hundreds of years) before all of the bentonite becomes water-saturated.

The swelling pressure forces the bentonite into fissures in the walls of the storage holes, sealing them. This includes fissures which may arise following deposition. In the tiny fissures (width < 1 mm) which may occur adjacent to a storage hole, bentonite penetration will cease after a few decimetres. The amount of bentonite which can penetrate into such fissures is so small that it will not affect the density of the buffer material and thereby its function.

The swelling pressure created by the highly compacted bentonite will compress the sand/bentonite fill in the tunnel above. As a result, the bentonite will be able to swell a few decimetres upwards. The specified final density of 2.1 t/m^3 takes this volume increase into consideration, however. This density may be slightly lower in the top part of the storage hole and higher in the bottom part, but this will not have any appreciable effect on the function of the buffer material. The bentonite cannot penetrate into the pores in the sand/bentonite fill in the same manner as it penetrates into fissures, since the pores are too small.

8 CANISTER MATERIAL

8.1 GENERAL

The primary function of the canister is to constitute a long-term durable barrier against the escape and dispersal of radioactive elements from the fuel. Such a dispersal can only take place via the groundwater. As long as the canister has not been penetrated, it therefore comprises an absolute barrier, in contrast to the other barriers in the final repository, which are based on very slow processes of dissolution and transport.

A secondary function of the canister is to afford radiation shielding to reduce radiolysis of the groundwater to a level which is low from the viewpoint of corrosion. This radiation shielding also facilitates handling when the canister is transported down to the final repository and when it is placed in the storage hole.

Penetration of the canister can be caused either by corrosion attack or mechanical stresses leading to failure.

8.2 CHOICE OF MATERIAL AND CANISTER DESIGN

In the final repository, the waste canisters will be subjected to the action of groundwater. In view of the high demands on the service life of the canister, it is therefore necessary that the canister material have a very low reaction rate with water. The supply of small quantities of substances dissolved in the groundwater which can cause corrosion can be limited by surrounding the canister with a buffer material - see chapter 7.

The best guarantee for a low reaction rate with water is thermodynamic stability, which means that no detectable reaction at all takes place, not even over an unlimited period of contact. If such a material is chosen, data on reaction kinetics are unnecessary.

Copper is a material which is practical to use and thermodynamically stable in the presence of pure water. Other metallic materials with this property are silver, gold and the platinum metals, but these are out of the question for economic reasons.

Pure metallic copper, so-called Oxygen Free High Conductivity Copper (SIS 5011), has therefore been chosen as the canister material. The design of the canister is illustrated in fig. 8-1.

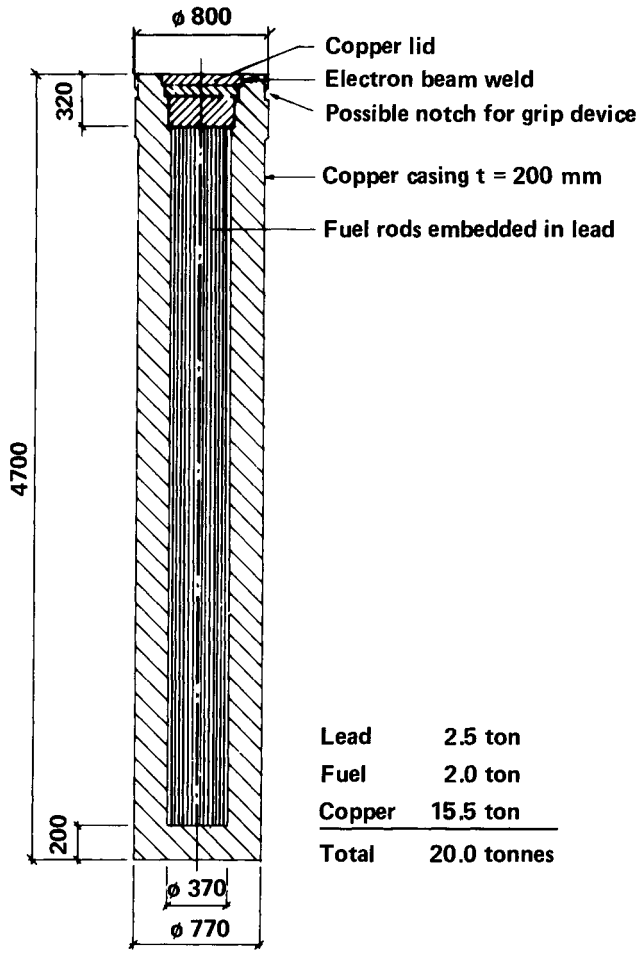


Figure 8-1. Longitudinal section of copper canister with fuel rods embedded in lead.

Its minimum wall thickness is 200 mm, its outer diameter is 770 mm (up to 800 mm to provide a grip for lifting). Its length is 4 700 mm and its weight is about 16 tonnes. The three lids are joined to the canister by means of electron beam welding. The fuel in the canister is embedded in lead, preventing deformation of the canister due to external overpressure.

Copper also possesses the strength properties required for the canister to be able to withstand the mechanical stresses to which it will be subjected.

8.3 CANISTER SERVICE LIFE

8.3.1 Mechanical strength

In the final repository, a canister is subjected to mechanical stresses as a result of both the external overpressure caused by the hydrostatic pressure of the groundwater plus the swelling pressure of the buffer material and the internal overpressure caused by the formation of helium in the fuel.

The external pressure does not give rise to stresses which can lead to failure, not even if the swelling pressure should be unevenly distributed due to uneven water absorption in the bentonite. The lead around the fuel rods will keep the canister from being deformed by the external pressure. The internal pressure

develops so slowly that critical deformation cannot occur within a period of one million years.

Any movements in the surrounding rock could also give rise to stresses on the canister. The elasticity of the buffer material and the ductility of the canister would, however, prevent failure of the canister in the event of the small rock displacements which are possible (see chapter 6).

Thus, mechanical stresses cannot be expected to limit the service life of the canister to less than one million years.

8.3.2 Corrosion

Since copper is thermodynamically stable in pure water, corrosion reactions can only occur with certain substances which are dissolved in the groundwater.

Corrosion reactions with most of the substances which can be expected to occur in the groundwater around the final repository are impossible for thermodynamic reasons. Only free oxygen and dissolved suphides can cause corrosion attack.

The buffer material around the canister, however, possesses extremely low permeability and low diffusivity for substances dissolved in water. These properties retard the passage of corrosive substances in the groundwater through the buffer material. Calculations based on very conservative assumptions show that corrosion caused by such substances cannot exceed fully 50 kg per canister after one million years.

The atmospheric oxygen which is entrapped in the buffer material when the repository is sealed must also be taken into consideration. Most of this oxygen is present in the sand/bentonite fill in the tunnel above the storage hole. This oxygen is, however, removed by the addition of 0.5% iron (II) phosphate (ferrophosphate, known in mineral form as vivianite) as a deoxidizer (so-called "oxygen-getter") to this fill. The amount of oxygen present in the storage hole could only cause the corrosion of at most a few kilogrammes of copper. Corrosion by oxygen formed through the radiolytic breakdown of water resulting from gamma radiation from the waste is negligible, owing to the good radiation shielding provided by the thick canister wall.

Total corrosion will not exceed about 60 kg per canister after one million years. This corresponds to a average corrosion depth of about 0.5 mm. It is possible that the top part of the canister could be attacked more severely owing to the fact that corrosive substances from the tunnel will react predominantly with this part. If it is assumed that such attacks are concentrated to the top 10% of the canister surface, the mean corrosion depth there will be about 2.4 mm.

From this it can be concluded that corrosion penetration of the canister within one million years can only take place if the corrosion attack is concentrated to a very small part of the surface in the form of a spot attack, known as pitting.

However, pure copper is a material with little pitting tendency.

Even if pitting is initiated, the growth of the pit will cease almost completely in time if the wall is sufficiently thick. The continued attack then takes the form of a widening of existing pits and initiation of new pits. With extremely thick walls - as in this case, 200 mm - the attack, after a very long period of time, assumes the character of an eroded surface zone with local variations in depth.

A conservative value for the maximum corrosion depth is obtained by disregarding this evening-out of the attack with time and instead using the maximum ratio between maximum and average corrosion depth (known as the pitting factor) which has been observed (about 25). The maximum attack depth will then be 60 mm after one million years, i.e. 30% of the wall thickness.

8.3.3 Summary evaluation of canister life

From the above, it can be concluded that neither mechanical stresses nor corrosion attack can be expected to give rise to canister penetration within one million years, i.e. the minimum expected canister life. This estimate of canister life is based on extremely unfavourable assumptions concerning groundwater chemistry and conservative assumptions concerning rock fissure content and groundwater flow. Reasonable assumptions of higher values for the latter factors do not have an appreciable effect on the estimate of canister life.

A group of specialists, mainly from the field of corrosion and materials, has, under the auspices of the Swedish Corrosion Research Institute and on commission from KBS, examined the durability of the copper canister from the viewpoint of corrosion. The results of this examination are presented in KBS Technical Report No. 90. The investigation arrives at the conclusion that it is realistic to expect a service life of hundreds of thousands of years.

Eight of the group's nine members stand behind this statement, while one has made a reservation.

The objections of the dissenting member (which are presented in the above-mentioned KBS report) are based on, among other things, the opinion that a future ice age could cause considerable fracturing of the rock, which would lead to a dramatic increase of the groundwater flow and thereby of the corrosion rate on the copper canister. This postulated effect of an ice age on the Swedish bedrock at a depth of 500 m does not agree with the geological interpretation of the pattern of fractures which can be observed today - see chapter 6. In the hypothetical case that a glaciation would result in extensive fracturing of the rock around the canisters, the buffer material, with its great potential swelling capacity, will seal new or widened fissures and the reported conclusion concerning the service life of the canister will not be affected.

8.4 STATUS REPORT FOR CERAMIC CANISTER

Ceramic materials are also being considered for the canister material. According to a method proposed by ASEA, the spent fuel is

enclosed in canisters fabricated by the hot isostatic compaction of aluminium oxide powder at 1 350°C and 100 MPa in a so-called QUINTUS press. A summary of the report on the current status of the development work on this encapsulation method is summarized below. The method is further described in Appendix 1 to the technical volume of this report.

Aluminium oxide occurs in nature in mineral form as corundum and saphire. Deposits on riverbeds and shore gravel show that this mineral possesses very high chemical and mechanical stability over long periods of time, even on a geological time-scale. Corundum millions of years old has, for example, been found in water-rich environments.

The proposed canister is 3 metres long, 0.5 metres in diameter and weighs approximately 2 tonnes. Full-sized canisters have been fabricated in ASEA's high-pressure laboratory at Robertsfors, see fig. 8-2.

The canister is fabricated in two parts, container and lid. The length of the canister, which is determined by currently available manufacturing equipment, does not permit enclosure of full-length fuel rods. A method for rolling the fuel rods together has therefore been developed. For handling reasons, the fuel rolls are stacked in a steel container which is then placed in the canister. A canister holds 144 BWR rods or 174 PWR rods. A total of some 25 000 canisters is required.

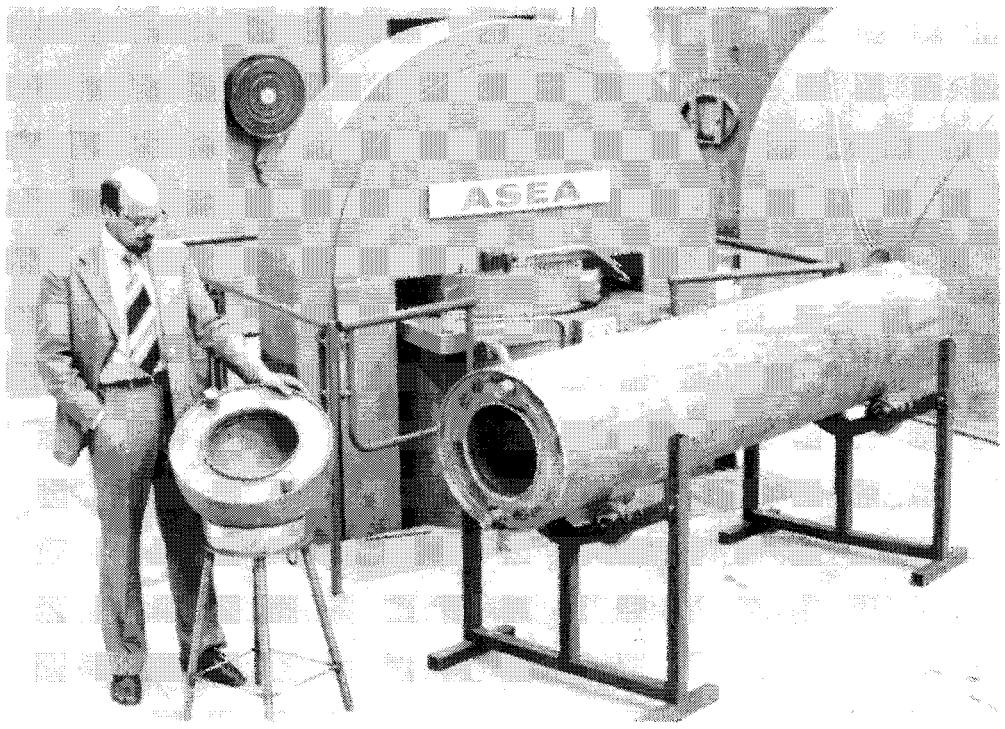


Figure 8-2. Full-sized aluminium oxide canister for direct disposal of spent nuclear fuel. In the background, the QUINTUS press for hot isostatic compaction.

The aluminium oxide container and the lid are enclosed by a thin metal casing and joined together by an additional hot isostatic pressing to form a fully homogeneous body. The canister can then be deposited in the final repository.

Hot isostatic compaction is an industrial method which is used all over the world for the production of, among other things, tools steel and sintered carbide articles.

In the final repository, the canister will be subjected to the action of the surrounding groundwater. However, this takes place at a very slow speed. Tests are currently under way to establish the durability of the hot-isostatically compacted aluminium oxide in the types of groundwater in question. These tests are being conducted at elevated temperature, 100 - 350°C, in order to obtain measurable effects. In the groundwater in question, however, deposition on the surface of the canister is more likely than erosion of the surface.

Tests performed to date have shown that a canister of aluminium oxide with 100 mm thick walls can withstand the action of the groundwater for at least hundreds of thousands of years with an ample safety margin. The durability of the canister is only slightly affected by the surrounding geological environment.

Both theoretical and experimental studies concerning the strength and thermal properties of large ceramic bodies have also been carried out. The results show that the canisters can resist the forces to which they can be subjected in the final repository with adequate margin.

The development work carried out thus far has thus demonstrated that the proposed encapsulation method is a realistic alternative for the final storage of spent fuel.

9 LEACHING AND MATERIAL TRANSPORT

9.1 GENERAL

The canisters with the spent fuel are surrounded by compacted bentonite in the repository, which is situated in impervious rock at great depth. The radionuclides must pass a series of barriers on their way to the biosphere:

- the copper canister, the lead and the zircaloy must be penetrated. This can take place through fracture or through corrosion,
- the uranium dioxide pellets must be dissolved and release the nuclides,
- the nuclides must diffuse out through the clay,
- the nuclides must be transported with the groundwater to the biospehere.

The degradation of the copper canister has been discussed in chapter 8.

The dissolution rate for the ceramic uranium pellets is low. This is discussed in connection with an examination of both Swedish and foreign experiments in section 9.2.

Section 9.3 deals with transport of the radioactive elements. The clay is so dense and impervious that water cannot flow through it. Transport is effected solely by means of diffusion.

Dissolution and leaching of radioactive elements in the fuel's metal components is dealt with in section 9.4.

Chapter 10 treats transport with the groundwater from the repository to the biosphere.

9.2 LEACHING OF FUEL

Only a few experimental studies have been carried out which are directed at the leaching problems connected with the disposal of spent nuclear fuel.

In 1975, however, tests were begun at Battelle-Northwest in the United States to study the leaching rate of some nuclides in irradiated uranium dioxide fuel. KBS has sponsored leaching studies at Studsvik on spent nuclear fuel from a Swedish nuclear reactor.

The American studies were carried out at 25°C on pellet fragments from light water fuel irradiated to very high average burnup (54 450 MWd/MTU). Of the nuclides which were studied, cesium was leached out most rapidly, curium most slowly in the following order:

Cesium > antimon > strontium / yttrium > plutonium > curium

After 150 days, approximately 1.5% of the cesium content of the fuel had been liberated. The fraction of the plutonium which was dissolved in the groundwater after 150 days was 30 millionths.

The leaching trials at Studsvik were carried out at 60°C on 20 mm long pieces of fuel rods from reactor 1 at the Oskarshamn station. A relatively high leaching rate, expressed as the fraction of the total quantity of an element which is leached out per day, was measured in connection with leaching in synthetic groundwater. Towards the end of the leaching period, values of approximately one millionth per day were measured for uranium, strontium-90 and cesium-137 and approximately one ten millionth per day for the total alpha activity.

The experimental leaching studies were carried out under conditions which deviate in many respects from the conditions which can be expected to prevail in the event of the penetration of groundwater into unreprocessed spent nuclear fuel.

The main difference is that the water does not flow freely around the canister in the repository. Since the nuclides diffuse very slowly through the wet clay which surrounds the canister, the water which is closest to the canister will be nearly saturated with dissolution products. This greatly reduces the further dissolution.

The solubility of the fuel pellets is determined to a large extent by the valence state of the uranium. Quadravalent uranium is poorly soluble, while hexavalent uranium can form easily soluble complexes with carbonate ions.

The uranium in the fuel pellets exists in the quadravalent form. The solubility of the pellets is therefore determined by the rate at which oxidizing substances can be transported through the clay barrier up to the fuel.

Another mechanism which can affect the oxidation rate is the formation of oxidizing substances through radiolysis. With an intact canister, radiolysis outside the radiation-shielding metals in the canister is negligible. But if these metals are penetrated and water comes into contact with the fuel, a radiolytic breakdown of the water is caused by radiation from the fuel. If this occurs, oxidizing breakdown products from the radiolysis can convert the uranium into the hexavalent state. The intensive gamma and beta radiation diminishes relatively quickly, while the alpha radiation persists for a long period of time.

The importance of radiolysis following penetration of the copper canister is difficult to assess quantitatively, since it depends on the degree to which water is in direct contact with the uranium dioxide, the formation of radiolysis products and the degree

to which the oxidizing substances are available to oxidize the uranium.

It is nevertheless possible to calculate the upper limit of the leaching rate by assuming that all of the uranium exists in the hexavalent state and that its transport away from the fuel is controlled only by the rate at which the elements can be transported away.

With pessimistically chosen values for the water flow around the clay barrier and the transport capacity of the groundwater, the minimum time obtained for dissolution of the uranium in a single canister is slightly more than one million years.

Dissolution and leaching of the fuel is dealt with in greater detail in chapter II:6.

9.3 TRANSPORT THROUGH BUFFER MATERIAL

The copper canister is placed in a tightly-packed clay consisting primarily of bentonite. This clay swells strongly when it absorbs water, becoming so dense and impervious that water can hardly flow through it at all. Gaseous oxygen and other oxidants in the water are transported through the dense clay much more quickly by diffusion than by the virtually non-existent flow. The same applies to the outward transport of corrosion products and any radioactive substances which escape. The diffusion rate in the compacted clay has been measured and found to be about 100 times lower than in water. Due to the limited water flow, the slow rate of diffusion in the clay and the fact that there are very few and small fissures in the rock, no more than one litre of water per year will exchange oxidants and other substances with each canister.

Positively charged ions will migrate much more slowly through the clay than uncharged molecules and negatively charged ions. The positively charged ions include most of the actinides, such as uranium, plutonium, neptunium and radium. Their migration rate decreases owing to the fact that the clay can absorb large quantities of positive ions by the exchange of these ions for sodium in the clay. Many important nuclides, including cesium-137, strontium-90 and americium-241, are retarded so much (700 years, 900 years and 11 000 years, respectively) that they decay down to one millionth or less of their original radioactivity while they are diffusing through the clay. The others will also be greatly delayed following an eventual canister penetration before they escape. Plutonium, for example, will be retarded at least 8 400 years.

Corrosion of the canister is caused by substances which come from the clay and from the water flowing in the rock outside of the clay. The corrosive substances in the clay in the hole in which the canister is located are transported relatively quickly to the canister wall. This transport takes from several decades to several centuries. The importance of these substances has been reported in chapter 8.

Following a canister penetration, the contents of the fuel rods are dissolved and transported out through the clay. Some sub-

stances which collect at the surface of the pellets are dissolved quickly. These include iodine and cesium. Iodine will diffuse out without any delay, since it does not undergo ion-exchange in the clay. Poorly soluble substances, such as oxides of uranium, plutonium and neptunium, are transported slowly, since the water in which they are transported cannot contain a higher concentration than the solubility allows. Calculations and measurements in uranium mines show that uranium levels under conditions similar to those in Swedish rock are very low. The level of uranium in reducing (oxygen-poor) water is less than 0.1 mg/l. Calculated solubilities for uranium, plutonium and neptunium exhibit similar concentrations. In oxidizing (oxygen-containing) water, the level can be considerably higher. The highest possible value is about 1 070 mg/l (0.1%). This is based on the assumption that all carbonate in the water contributes towards dissolving the uranium. It is therefore probably greatly exagerated. Since no better data have been available, this value has been used anyway. With this value for solubility, it takes 1.4 - 1.8 million years to dissolve and transport away the quantity of uranium (1.1 - 1.4 tonnes) in one canister.

Since conditions in the rock are reducing, it takes a considerably longer time in reality, up to 10 000 times longer. This applies to the other important elements plutonium and neptunium as well.

The above analysis is based on the assumption that the clay which surrounds the canister remains in the hole. It is of less importance whether it has altered its permeability, as long as it does not allow water to pass through completely freely. The clay does not seept out into fissures at a rate which enables it to expand appreciably in a foreseeable period of time. Nor is the clay carried away by groundwater with the salt content which is found in Swedish groundwater. It forms a very stable gel from whose surface very few clay particles detach.

The clay's content of sodium ions will change as they are exchanged for other ions, such as calcium ions from the water. The clay's ion-exchanging properties and its retarding capacity will thereby be altered. But it takes about 3.8 million years for all of the clay to be altered, after which time the radioactive elements for which the retarding action of the clay is important have long since decayed.

Calculations have been carried out for a number of cases where the fissure content of the rock nearest the canister and the water flow rate in the "undisturbed" rock have been varied. In the least favourable case, the transport rate for the substances has thereby increased by a factor of 4 compared to the values given above.

9.4 METAL COMPONENTS OF FUEL

The fuel's metal components - stainless steel, zircaloy and inconel - contain, among other things, radioactive nickel-59. The metal components are compacted and placed in concrete boxes. The cavities are filled with cement and the boxes are stacked in a tunnel at great depth. The tunnel is then filled with concrete.

Groundwater will eventually penetrate into the metal and corrode it. Nickel is a relatively noble (electro-positive) metal which will not necessarily be oxidized under the prevailing conditions. The calculations and evaluations have, however, assumed that all nickel is oxidized. Nickel forms poorly soluble compounds with hydroxide ions. The concrete contains very large quantities of hydroxides, which contribute towards maintaining the pH between 10 and 13 for a very long period of time. During this period of time, the solubility for nickel is about 0.1 mg/l. It has, however, been assumed that the organic substances present in the groundwater can contribute towards increasing the solubility of nickel to 30 mg/l. Under these conditions, a maximum of 0.4 millionths per year of all the nickel in the repository can leak out. And this only under the assumption that the concrete has cracked so that it does not present any obstacle to the flow of water to the metal components.

10 DISPERSAL MECHANISMS

10.1 GENERAL

Escaping radioactive substances are transported with and in the water. Over short distances, approximately 1 metre, and under the conditions which prevail around the final repository, transport via diffusion can be faster than transport via water flow, whereas water flow is the predominant mechanism for transport over long distances. The water flows in fissures. Most substances migrate at a much slower rate than the water, due to the fact that they react with the surface of the fissures and with the fissure-filler material. The water velocity, the distance to the biosphere and the retention factor therefore determine the time which it takes for an element to reach the biosphere. The quantity transported out is determined by the flow of water and its content of the various elements.

A more thorough analysis of the dispersal mechanism which can transport radioactive elements from the rock repository to human beings is provided in the technical portion of the report, chapter II:7.

10.2 DISPERSAL IN THE GEOSPHERE

Measurements in boreholes show that the rock at such great depths as 500 m is very impervious over large areas, even though there may be local fracture zones. On the basis of these measurements and extensive calculations, it can be concluded that the water flow rate in the rock is less than 0.2 litres per year and m^2. The flow rate is not appreciably affected by local areas of high fissure content caused by tunnel construction work or due to the heat evolved by the fuel.

The water in the rock moves in fissures. These fissures have varying water-bearing capacity, depending primarily on their width, but also on any filler material which they may contain, such as the weathering products of the rock. The large fissures are of dominant importance for water transport and thereby the transport of the various elements. Owing to the difference in velocity between the flow in large and small fissures, the transit time is also different in different fissures. Measurements using tracers at Studsvik have shown that the first traces emerge after only about 20% of the mean residence time. Theoretical calculations give similar values. This is of importance for the radioactive elements which would decay during a period of time corresponding

to the mean residence time. The portion of these elements which emerge after 1/5th of the mean residence time can still possess considerable radioactivity.

Each canister exchanges elements with approximately 1 litre of water per year. This amount of water may contain a concentration of elements equal to that inside of the clay barrier. The concentration there is limited by the solubility of the different elements and by the leach rate of the fuel.

Transport to the biosphere is affected by how the element is adsorbed, ion-exchanged or in any other way reacts with the rock. Extensive laboratory measurements of crushed rock and large rock surfaces show that those elements which occur as positive ions in the water are greatly retarded. The table below presents values for the retention factors of some elements in impervious rock. These values are based on measured results obtained after a short contact time between water and rock (column 1). With long contact times, the values for most elements are increased by a factor of 2-10 (column 2). Even greater increases have been noted for some elements. Measurements of the retardation of strontium in connection with field tests in rock fissures at Studsvik show good agreement with laboratory measurements.

Element	Retention factor short contact time	Retention factor long contact time
Uranium	1 900	23 000
Radium	1 200	48 000
Neptunium	1 900	23 000
Plutonium	2 800	5 700
Strontium	120	1 500

A retention factor of 1 000, for example, means that it takes 1 000 times longer for the element to migrate a given distance than for water. In less permeable rock, the retention factor increases, in more permeable rock it decreases. The retention factor probably does not apply to that portion of the element which is strongly bound to the organic substances which are complexing agents and the small, solid colloid particles which follow the water. Measurements of the amount of organic complexing agents and estimates of the transport capacity of the colloids show that a very small amount can be transported with these substances. A total of no more than approximately 120 mg of metal per year and canister could be transported. Only a small portion of this quantity will consist of the more radioactive elements, since these compete for complexing agents and colloids with other metals in the water and uranium, which comprises 95% of the fuel.

The clay, which contains large quantities of small particles and could therefore form colloidal solutions, forms very stable gels in the groundwater owing to its content of bivalent ions, mainly of calcium and magnesium. Extremely few clay particles are therefore present in the water.

Some hydrophilic (water-loving) elements, of which iodine is the most important, will not be retarded in the rock. They reach the biosphere as quickly as the water. On the way, they are heavily diluted with water from large areas.

During their transit time - the transit time of the water times the retention factor - many of the radioactive elements decay. Some elements, however, produce radioactive daughter nuclides which can migrate at a different rate than the mother nuclide. Many decay chains have 2 or 3 daughter nuclides, but longer chains also exist. Calculations have been carried out for all decay chains which have an activity which could contribute to radiation in the biosphere. The results of these calculations are reported in chapter 11.

10.3 DISPERSAL IN THE BIOSPHERE

The radioactive elements from the repository reach the biosphere via the groundwater. The inflow takes place via some receiving body of water. Three different types of primary recipients have been studied:

- valley with deep-drilled well close to the final repository,
- lake close to the final repository,
- Baltic Sea.

The paths of dispersal are illustrated schematically in figure 10-1.

On its way towards the surface, the groundwater-borne elements from the repository meet additional water at various depths and is gradually diluted in a relatively large volume of water which, for the case of a well in the valley, has been calculated to be 500 000 m^3. The radioactive elements which reach the lake are diluted in 25 million m^3. Calculations are based on the local conditions at Finnsjö Lake near Forsmark. These conditions are representative for a large number of sites for the lake case. The local situation for dilution on the way to the well, however, is judged to be relatively unfavourable.

The transport of the radioactive elements through the biosphere is described by a mathematical model which takes the processes described below into account.

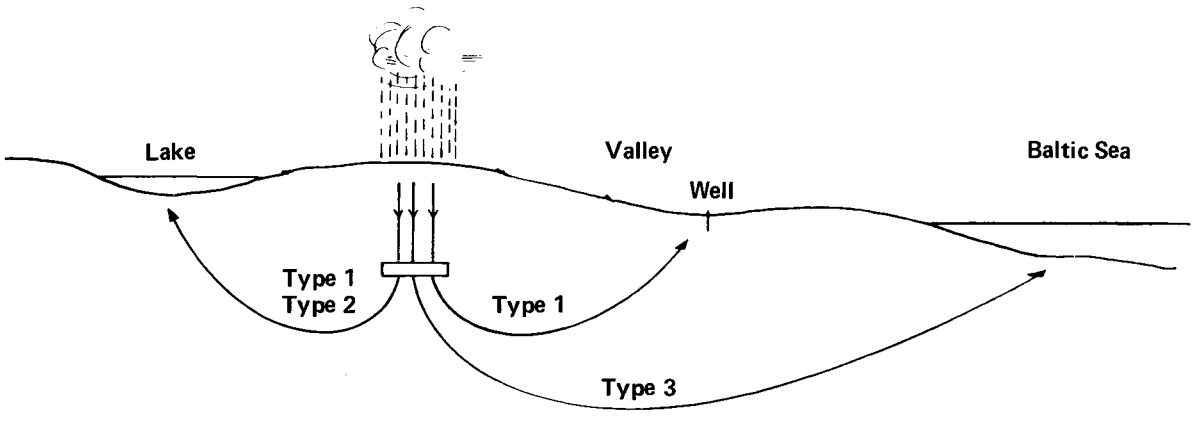


Figure 10-1. The three main paths of transport of radioactive substances to the biosphere.

The radioactive elements are carried out into ever-widening ecological cycles through a region which is 900 km^2 if the primary recipient is located inland and a 30 km long and 2 km wide coastal zone if the groundwater-borne outflow first reaches the Baltic Sea. From the region, the radioactive elements are carried out into the Baltic Sea by runoff from the lake systems or water exchange with the coastal zone. The Baltic Sea exchanges waterborne material with the North Sea. In this manner, the radioactive elements enter global cycles. Transport through the atmosphere is also taken into consideration by the model. But this path of dispersal is only of any significance compared with dispersal through the water for a few elements which evaporate readily from the surface of the ocean, such as carbon and iodine.

Simultaneously with this horizontal dispersal, an exchange takes place vertically through the ecosystems. Radioactive elements which reach the lake, for example, can be exchanged between water and sediment by means of sedimentation and resuspension. The assumption that lake water is used for irrigation on a scale which is typical for the current consumption of surface water in central Sweden leads to the entry of radioactive elements into the soils. From there, they are returned by surface runoff or transport via the groundwater at various velocities, depending on the element, to the lake systems. Upward transport to the atmosphere through resuspension must also be taken into consideration. The airborne material which originates either from resuspension from the land areas or evaporation, aerosol formation etc. from the sea surfaces is exchanged after precipitation between the global soil and groundwater areas and is dispersed to a varying extent to the seas and the atmosphere.

During the cycles in the biosphere, new radioactive elements are generated when certain elements decay. Of particular interest in reference to the long-range exposure is the decay chain uranium-234 - thorium-230 - radium-226. Owing to the fact that the mother nuclides uranium and thorium migrate relatively slowly through the surface soils, many years of input of these radioactive elements into agricultural areas is accumulated before equilibrium is achieved between the annual input and the amount which is carried away each year with the surface water or the groundwater. As a result of this accumulation, the radium-226 which is formed by disintegration and which moves relatively easily through the soils can reach such concentrations in the groundwater that it makes a considerable contribution to the total dose. Drinking water and milk are important paths of exposure in this connection.

Thus, man ingests radioactive elements primarily through food, drinking water and air. The concentrations of the radioactive elements in air and in drinking water and their uptake in plants are of great importance. In calculating the amount of radioactive elements which have been assimilated by the body, the manner in which the different radioactive elements are metabolized in and excreted from the body must also be taken into account. All of these factors are included in the mathematical model used for the dose calculations.

But radiation doses are not determined only by radioactive elements in the body. Radioactive elements in the environment can also irradiate man. The model of the biosphere therefore calcu-

lates future doses to individuals and the population both from radioactive elements which are absorbed in important food substances and drinking water via various paths and elements which, by means of decay in soil, water or sediment, give rise to external exposure of human beings from the ground and from lakes or oceans, shores and fishing tackle.

Figure 10-2 illustrates some of the paths along which radioactive substances in our environment can reach man. Paths of migration and exposure are discussed in section II:7.4. As a general rule, internal exposure through uptake in the tissues of the body makes the predominant contribution to the radiation doses. In those cases where the primary recipient is located inland, drinking water is an important and often dominant path of exposure as regards doses to the most exposed individuals. Meat, milk and fish are also often among the most important paths of exposure. If the outflow takes place directly into the Baltic Sea, however, fish are always the dominant path of exposure. In most cases fish consumption is also responsible for most of the collective exposure, where the collective dose contribution from the regional population is relatively small.

10.4 PREMISES OF THE BIOSPHERE MODEL IN A LONG-RANGE PERSPECTIVE

In the long range, essential changes can take place in the premises on which the study of dispersal in the biosphere and paths of exposure has been based.

For example, an increased utilization of the food resources of the sea can cause a shift towards a diet of a more marine character. Overexploitation of traditional fish populations will probably result in attempts to make use of other sources of nutrition from the sea in the future. Algae, squid and krill constitute potential sources of protein. Among these possible additions to or substitutions in our current diet, algae in particular affects the intake of radioactive elements with food due to the fact that it concentrates certain elements from the water.

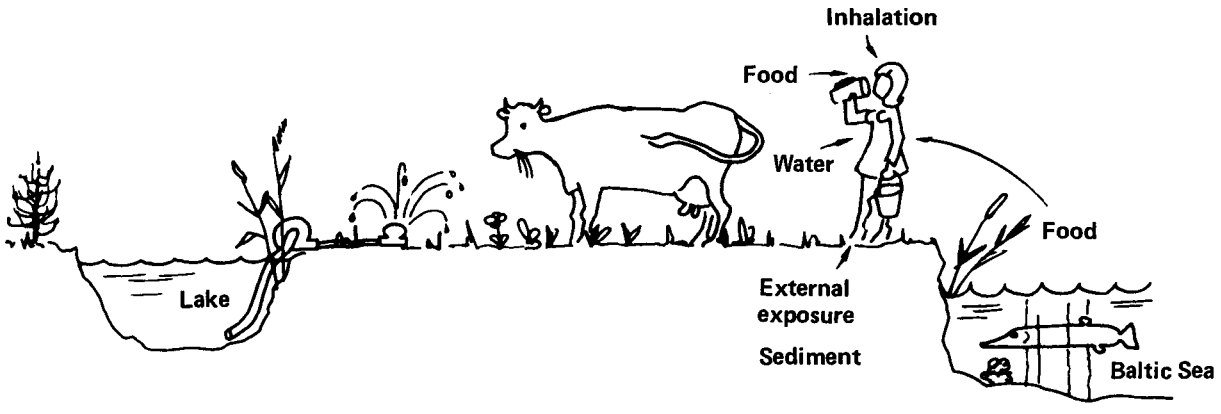


Figure 10-2. Paths of human exposure in the local ecosystem.

Over the time spans covered by the forecast, the local ecosystem can undergo changes which significantly affect the exposure picture. The design of the model makes it possible to analyze the consequences of important changes, for example drying-up of the lake which comprises the primary recipient for leached material from the repository. Dry-up of large portions of the Baltic Sea can also be taken into account. In both cases, the change can give rise to new exposure, pathways by the use of the sediments in agriculture.

Some elements are concentrated to relatively high levels in the sediment of the lake or the Baltic Sea. In the case of the radioactive elements which provide the predominant dose to the critical group or the collective dose to the population, such drying-up and alteration of the paths of exposure does not give rise to any increase in the annual doses. This is due to the fact that uptake in agricultural products grown on the sediment does not contribute as much as the elimination of the fish as a path of exposure takes away.

The consequences of such changes in paths of dispersal and exposure are discussed in section II:7.4.

11 SAFETY ANALYSIS

11.1 GENERAL

There are special laws and regulations in Sweden which govern nuclear power activities and which provide a basis for the safety and protection of personnel, nearby residents and the environment.

The Swedish Nuclear Power Inspectorate and the National Institute of Radiation Protection are responsible for the enforcement of laws and regulations in this field. These organizations are the inspection authorities designated by the Atomic Energy Act and the Radiation Protection Act. The Nuclear Power Inspectorate is mainly responsible for examining the safety of facilities and the design of various safety systems. The National Institute of Radiation Protection deals with matters pertaining to radiation protection in the working environment and in the external environment.

The general criteria for safety and radiation protection which comprise the basis for the inspection work and licensing activities of the inspection authorities require that facilities, processes, safety systems and activities be designed in such a manner that:

- the radiation exposure of personnel, nearby residents and the regional and global population is low,
- the risk level resulting from accidents is low,
- every effort is made to minimize the radiation exposure within the bounds of reasonable cost.

The special rules and regulations issued by the authorities in different countries on the basis of these criteria are summarized in the KBS report on vitrified high-level reprocessing waste.

The explication of the Swedish Stipulation Law (which specifies the criteria which must be fulfilled before new nuclear power plants can be commissioned) states that the final repository must be able to meet radiation protection requirements which aim at preventing radiation injuries. The final repository shall be designed in such a manner that the waste or the spent fuel can be isolated for as long a period of time as is required for its radioactivity to have decreased to a harmless level. The evaluation of the safety of the final repository shall take into considera-

tion the risk that the waste or the spent fuel might escape to the biosphere by means of natural processes, accidents or acts of war.

This chapter provides a short summary of the safety analysis of the activities which are special for the final storage of spent fuel without reprocessing. A more exhaustive account is provided in chapter II:8.

The handling and final disposal of spent unreprocessed nuclear fuel involves processes which are identical or similar to those used in the previously reported alternative with reprocessing and vitrification.

This applies especially to

- the transport of spent fuel
- the handling and storage of spent fuel

The safety aspects of these operations were discussed in the KBS report on vitrified waste.

An additional aspect is safety in connection with extended fuel storage (40 years) in the central storage facility prior to encapsulation and final storage. A comprehensive analysis has been carried out of the fuel's integrity, handleability and radioactivity release in connection with extended storage in water pools. A summary of these results is presented in section 11.2.2.

The encapsulation of the fuel rods in a copper cylinder has been evaluated in detail from the viewpoint of safety. The results are summarized in section 11.2.3.

The final storage of spent unreprocessed nuclear fuel involves the long-term isolation of larger quantities of plutonium and other long-lived actinides than the final storage of vitrified waste from reprocessing. Greater weight has therefore been laid on the requirement that the barriers presented by the canister and the buffer material shall provide long-term protection against the dispersal of the radioactive substances. The properties and function of these barriers have been described in previous chapters. Section 11.3 presents a summary of the safety analysis.

11.2 SAFETY IN HANDLING, TRANSPORTATION, STORAGE AND ENCAPSULATION

11.2.1 Handling and transportation of spent nuclear fuel

For a summary safety analysis of the handling and transportation of spent nuclear fuel, see the KBS report entitled "Handling of spent nuclear fuel and final storage of vitrified high-level reprocessing waste".

11.2.2 Long-term storage of spent fuel in water pools

The proposed handling chain involves a wet storage of the spent

nuclear fuel for about 40 years following discharge from the reactor and prior to the start of final storage.

Good experiences from the storage of fuel in water pools are available from many years of practice. No effects on the cladding or the fuel have been observed in connection with such storage.

These experiences show that general corrosion after 100 years of storage in neutral pure water is scarcely more than 1 μm. Even with the assumption of other reasonable water chemistry, general corrosion will not occur to an extent which could jeopardize safety.

Pitting, galvanic corrosion, crevice corrosion or galvanically induced hydration do not entail any problems for this type of storage either.

Delayed fracture due to hydrogen embrittlement has not been observed on zircaloy-clad fuel in connection with storage. Theoretical analysis of the process indicates low crack growth which does not lead to failure.

During reactor operation, isolated fuel damages may occur as a result of stress corrosion cracking. This is caused by high stresses during operation and the simultaneous effects of certain elements (mainly iodine) which have been released from the fuel.

Since the stress in the cladding material is considerably lower during storage than during reactor operation and since the release of fission products ceases after discharge of the fuel from the reactor, the conditions necessary for stress corrosion cracking do not exist in connection with storage in the water pools.

Experimental studies of the inside of the cladding after many years of reactor operation and storage have shown the following:

- The build-up of oxide on the inner surface was scarcely measurable (approx. 1 μm) after 8 years of irradiation in the Halden reactor.
- Metallographic examination of fuel rods after 10 years of storage in England revealed nothing which could indicate current or incipient degradation.

Small quantities of radioactive elements can be released from damaged fuel rods during storage. But no further degradation of the fuel or the cladding has been found in connection with the storage of such rods.

In connection with the long-term storage of damaged fuel, Cs-134 and Cs-137 are of primary importance. In view of the extensive, good experience available from the handling and storage of damaged fuel in the reactor's storage pools and the low leakage which remains after an initial period, it can be safely assumed that the release of radioactivity during long-term storage will not constitute a problem.

If individual rods should be punctured during storage, this will lead to an insignificant release of krypton-85, but experience shows that this would not entail any radiation protection problems - not even in connection with fuel handling.

11.2.3 Encapsulation of spent fuel in copper canisters

The encapsulation station is described in section 5.3. Figure 5-6 shows the various handling stages in encapsulation.

The facility is designed with a special emphasis on

- minimizing radiation doses for the personnel,
- preventing damages to the fuel resulting from accidents and improper handling,
- minimizing the dispersal of any released radioactivity to the environment.

The integrity and handleability of the spent fuel is not expected to decrease over the 40-year period of pool storage.

The handleability of high-burnup fuel is good, and extensive experience and well-developed routines exist in Sweden for such handling. Experience shows that exposures to personnel employed with fuel handling are small. These experiences are based on the handling of fuel immediately after reactor shutdown. The fuel which is handled in the encapsulation station has decayed for 40 years, which means that handling in the encapsulation station will take place under more favourable conditions from the viewpoint of radiation.

The radiation doses are minimized primarily due to the fact that the fuel is handled by remote control, either under water or in radiation-shielded cells. Handling in water takes place with at least 2.5 m of water coverage over the fuel, which provides fully adequate shielding of both gamma radiation and neutrons.

The various handling operations and equipment in the radiation-shielded encapsulation cells are remote-controlled from the control room. All activities are monitored via radiation-shielding windows from this room. The ventilation air from the encapsulation cells can be passed through filters if measurements show that it contains radioactive material. All radioactive areas are kept under lower pressure than control areas, other plant premises and the surrounding environment. All equipment in the encapsulation cells can be given service and maintenance by lifting the equipment out of the cells or taking it to a separate service cell. After a cell has been emptied of fuel, and decontaminated if necessary, the necessary maintenance can be carried out in the cell as well.

The cell walls against the control rooms are thick enough so that no matter where the unshielded fuel is located in the cells, the dose rate in the control room will be very low. With conservative assumptions, the calculated dose rate is < 0.1 mrem/h.

During the lead casting operation in the encapsulation process, the fuel is heated to about $400^{\circ}C$. As a result, the internal gas pressure in the rods will increase. Thanks to the low fission gas pressure ($\sim$0.3 MPa) in the BWR rods, the effects of heating on the integrity of the cladding are negligible. The internal pressure of the PWR fuel can normally be as much as 5 MPa, which, after heating, can increase up to 12 MPa. This leads to an annular stress of about 110 N/mm^2. But the PWR cladding is designed to withstand stresses of 300 N/mm^2. Oxidation of the rods or the

canister is impossible, since heating and cooling takes place in a protective gas atmosphere.

Fuel handling and storage are designed so as to provide a good margin to criticality in all fuel configurations. K_{eff} is less than 0.95, even with unirradiated fuel in water in the most unfavourable configuration from the viewpoint of criticality, namely with fuel rods packed tightly in the copper racks.

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

The encapsulation station is designed in such a manner that the probability of accidents is very low. Those accidents which are nevertheless conceivable are limited to incidents which entail very little release of radioactive elements.

The following accidents have been studied:

- Fuel is dropped during handling
- Cladding is sawn through
- Mechanical fuel damages in connection with encapsulation
- Rack with fuel rods is dropped in casting cell
- Copper canister is damaged while being welded
- Copper canister is dropped
- Loss of vacuum and protective gas during heating and cooling in casting bell

External events, such as sabotage and acts of war, have also been considered.

The releases to the environment as a result of accidents are summarized in the following table. The lower limit applies to BWR fuel, the upper to PWR fuel. At the time being, it is assumed that 20-30% of the fuel consists of PWR fuel.

Release Ci/year	Kr-85 Ci	I-129 µCi	H-3 Ci	Aerosols (Pu) µCi
Annual average	10-200	2-25	0.1	-
Accident occurring several times a year	2-100	1-10	0.1	
Rare accident[1)]	100-2000	40-1200	4-10	4-40

1) less than once every 10 years

The radiation dose at a distance of 1 km from the station has been calculated to be less than 0.4 µrems/year under normal operating conditions and 40 µrems in the event of a rare accident, i.e. well below the design goal (10 000 µrems/year) for nuclear power stations.

It is espected that the average annual dose for personnel can be kept lower than 200 mrems/year, and no problems are foreseen in keeping the individual doses below the current limit value (5 000 mrems/year).

11.3 SAFETY OF THE FINAL STORAGE OF SPENT NUCLEAR FUEL

11.3.1 Grounds for the safety evaluation of final storage

The Stipulation Law directs that a reactor owner must demonstrate "how and where an absolutely safe final storage of spent, unreprocessed nuclear fuel can be effected". The special explication of the bill states that "the primary consideration here is whether the storage scheme can meet requirements for satisfactory radiation protection". Furthermore, "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 decay to a harmless level".

These very general criteria agree with the principles which are applied within other areas of nuclear power technology as well.

No specific safety criteria for final storage of radioactive waste have been established. However, work is being pursued within this area in many countries and in international cooperation. Pending the results of this work, existing rules which are more or less generally accepted for existing nuclear power installations must be applied.

The following rules should be considered in connection with final storage:

- The ICRP rule that no individual, either now or in the future, shall be exposed to radiation doses which exceed the limits recommended by the ICRP. The present limit for individuals among the general public is 500 mrems per year from all activities which can give rise to irradiation, with the exception of the medical use of ionizing radiation. If a given radiation source can be expected to give rise to exposures over a series of years, the radiation dose from this source shall not exceed 100 mrems per year on the average, figured as a weighted whole-body dose.

- The design goal for new nuclear power plants in Sweden, a maximum of 10 mrems per year (weighted whole-body dose) to nearby residents.

- The maximum permissible radiation dose to nearby residents in Sweden in connection with the operation of nuclear power plants, namely 50 mrems per year (weighted whole-body dose).

- The recommended maximum permissible global weighted collective dose commitment applied in the Nordic countries of 1 manrem per year and MW installed electrical output (MWe), which applies to the entire nuclear fuel cycle. The collective dose commitment shall be calculated over a period of 500 years and distributed between 0.5 manrem/MWe-year for the operation of nuclear power plants and 0.5 manrem/MWe-

year for the rest of the nuclear fuel cycle. The choice of the level of 1 manrem per year and MWe is based on the goal of a maximum of 10 mrems/year and individual with an assumed average global nuclear power production of 10 kW per person. (Note: In Sweden, the total installed electrical output is currently about 25 000 MWe, i.e. about 3 kW per person. Of this, some 0.5 kW per person is produced by nuclear power. 13 nuclear power units would increase this output to about 1.2 kW per person.)

In evaluating the final storage scheme, the fact that radioactive elements occur in nature and that ionizing radiation from these elements is a part of the natural environment of human beings should also be taken into consideration. Natural background radiation varies in Sweden between 70 and 140 mrems per year and individual.

11.3.2 Barriers

In order to meet the requirement for a long-term isolation of the radioactive substances and ensure a safe final storage, the waste is surrounded by a number of barriers.

These barriers are as follows:

- binding of the radioactive waste to a solid, poorly soluble substance,
- enclosure in canisters of a highly durable material,
- packing of the canisters in an impervious buffer material,
- chemical barriers against the dispersal of the radioactive substances,
- final storage in stable bedrock with low groundwater flow.

Each of these barriers provides protection against the release and dispersal of radioactive substances. But each barrier possesses different protective properties and functions which both reinforce and complement each other.

The binding of the radioactive elements to the uranium dioxide fuel is dealt with in chapter 9. The encapsulation of fuel rods is described in section 5.3 and the properties of the canister are described in chapter 8. The encapsulation of metallic waste and other types of waste is described in section 5.3 and the durability of the encapsulation is analyzed in section 9.4. The properties and function of the buffer material are described in chapter 7, and the properties of the geological barrier in chapter 6. The retardation of radioactive elements by sorption effects is described in section 10.2.

The design of the final repository as far as spent fuel rods are concerned is illustrated by figures 5-9 and 5-11, and as far as other waste is concerned by figure 5-12.

11.3.3 Probable sequence of events

The calculations and analyses of the life of the copper canister carried out in chapter 8 show that the canister can be expected to have a service life of more than one million years. During

this period of time, most of the radioactive elements in the spent fuel decay. During the period of time after one million years, the radiotoxicity of the fuel is dominated by decay products of uranium, primarily radium-226. This means that, with the stated service life of the copper canister, the consequences of the final storage of the spent fuel will not be worse than the consequences of the storage of an equivalent amount of uranium-dioxide which has not been irradiated in any reactor. An exception from this rule is the release of iodine-129, which, even after one million years, can give rise to a certain dose in the biosphere if it is dispersed via groundwater. But this dose does not exceed 0.4 mrem per year in the well case. The dispersal of iodine-129 proceeds much faster than the dispersal of uranium, thorium and radium. The calculated doses from iodine-129 are, however, considerably lower than those from radium-226.

If the copper canister is assumed to have a shorter life than one million years, dispersal of radioactive elements could start earlier. However, the chemical environment in the final repository is such that dissolution of the fuel can be expected to proceed extremely slowly. Locally, radiolysis caused mainly by alpha radiation can occur, but only in an area in the immediate vicinity of the canisters. Otherwise, chemical conditions in the rock are such that the solubility of actinides is extremely low. See chapter 6. Furthermore, the buffer mass is very impervious, which means that material transport is controlled by diffusion. Even with cautious assumptions, the calculated time required for the dissolution of all of the uranium in a canister is millions of years. Under such circumstances, the consequences will still be equivalent to those obtained from the storage of unirradiated uranium. Iodine-229 can still give rise to certain early radiation doses which could not be obtained from unirradiated uranium.

Thus, the copper canister and the buffer mass constitute consecutive barriers to the dispersal of the radioactive substances in spent nuclear fuel for virtually an unlimited period of time to come. At the same time, they are to a certain extent redundant barriers, since the buffer mass prevents rapid dispersal even if the copper canister should be penetrated.

11.3.4 Consequences of slow dispersal of radioactive substances

In evaluating the safety of the final repository, it is necessary to study the consequences of various courses of events which are less favourable than the most probable. Such courses of events include penetration of the canister at some given point in time and a relatively more rapid dissolution of the waste substances than is most probable.

As is reported in chapter 8, a group of corrosion experts have concluded that it is realistic to expect a service life of hundreds of thousands of years for the copper canister, with the exception of one expert, who is of the opinion that a service life of more than 1 000 to 10 000 years cannot be ensured.

The basic assumption in the consequence calculations is that the first canisters start to be penetrated after 100 000 years and that the process of canister penetration then continues at a constant rate for 400 000 years. Leaching-out of the fuel in a da-

maged canister is assumed to take 500 000 years. This assumption is justified in chapter 9.

With the assumptions of the reference case concerning canister penetration and leaching time, two different dispersal scenarios have been analyzed. The first, which is referred to as the main case, assumes a water flow time from the repository to the biosphere of 3 000 years combined with a best estimate hypothesis concerning retention factors. The second case is referred to as the pessimistic case and assumes a water transit time of 400 years and more conservatively calculated retention factors. The difference between the two sets of retention factors is described in section II:7.3.

The results of the aforementioned calculation cases and the influence of different variations in the basic assumptions and of different phenomena which can affect these are discussed below.

Main case

Figure 11-1 shows the calculated individual doses in a socalled "critical group" which is assumed to live in the vicinity of the repository and to obtain its water from a nearby deep-drilled well. It should be noted that the figure has logarithmic scales on both axes.

During the time period from 100 000 to one million years, when breakdown of the canisters and dissolution of the fuel is under

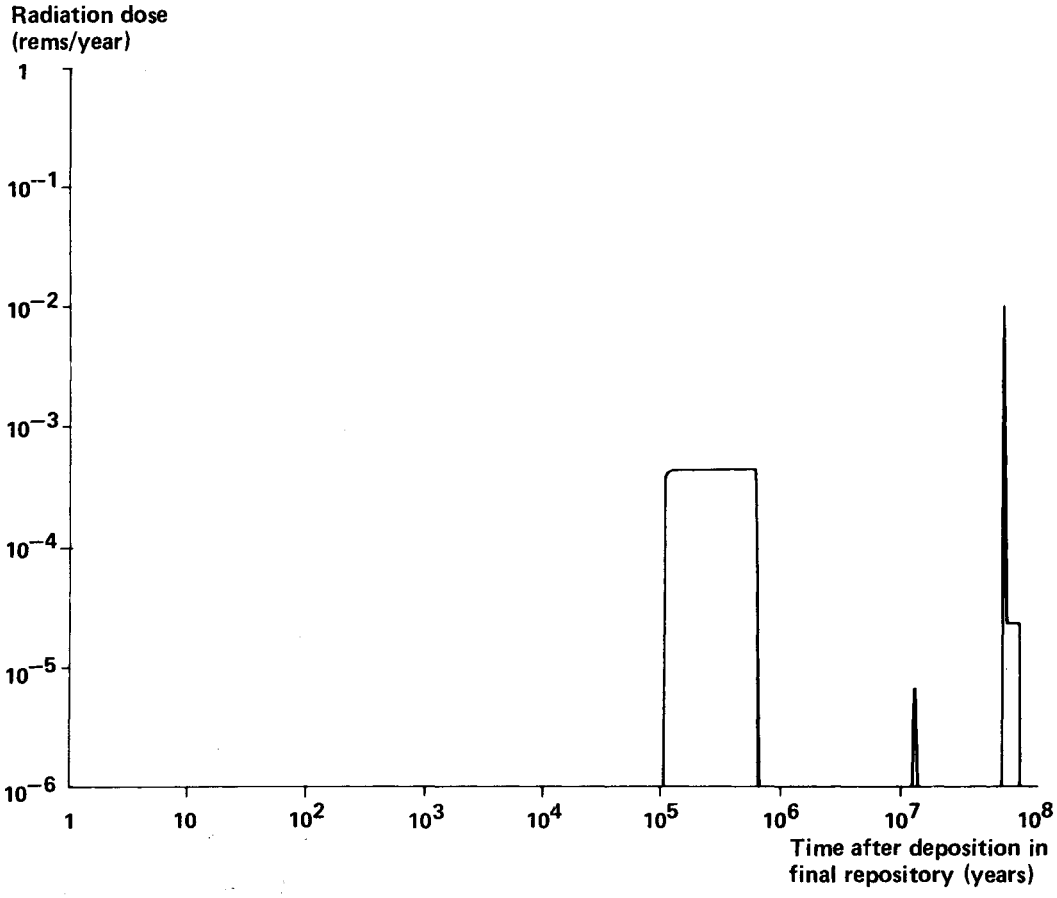


Figure 11-1. Calculated individual doses for critical group (nearby residents) for the main case with a well as the primary recipient.

way, the radiation dose is calculated to be less than 0.5 mrem/year. The dose stems primarily from iodine-129, which is not retarded by the rock. After more than 10 million years, cesium-135 is calculated to give a radiation dose of around 0.007 mrem/year. The largest doses for the main case are not expected to occur until after 70 million years, with a total dose of around 10 mrems/year.

The main contributions come from radium-226, protactinium-231, thorium-230 and the uranium isotopes.

See also the tables in section II:8.6.

If the primary recipient is assumed to be a lake, the radiation doses are only half of those in the well case and only 1/330th if the primary recipient is assumed to be the Baltic coastal zone. The predominant paths of exposure are dealt with in section II:8.5.

Pessimistic case

This case differs from the main case with respect to the water transit time (400 years) and lower retention factors.

Figure 11-2 shows individual doses to the critical group (nearby residents) with a well next to the repository as the source of water. In this case, the doses are concentrated to the period

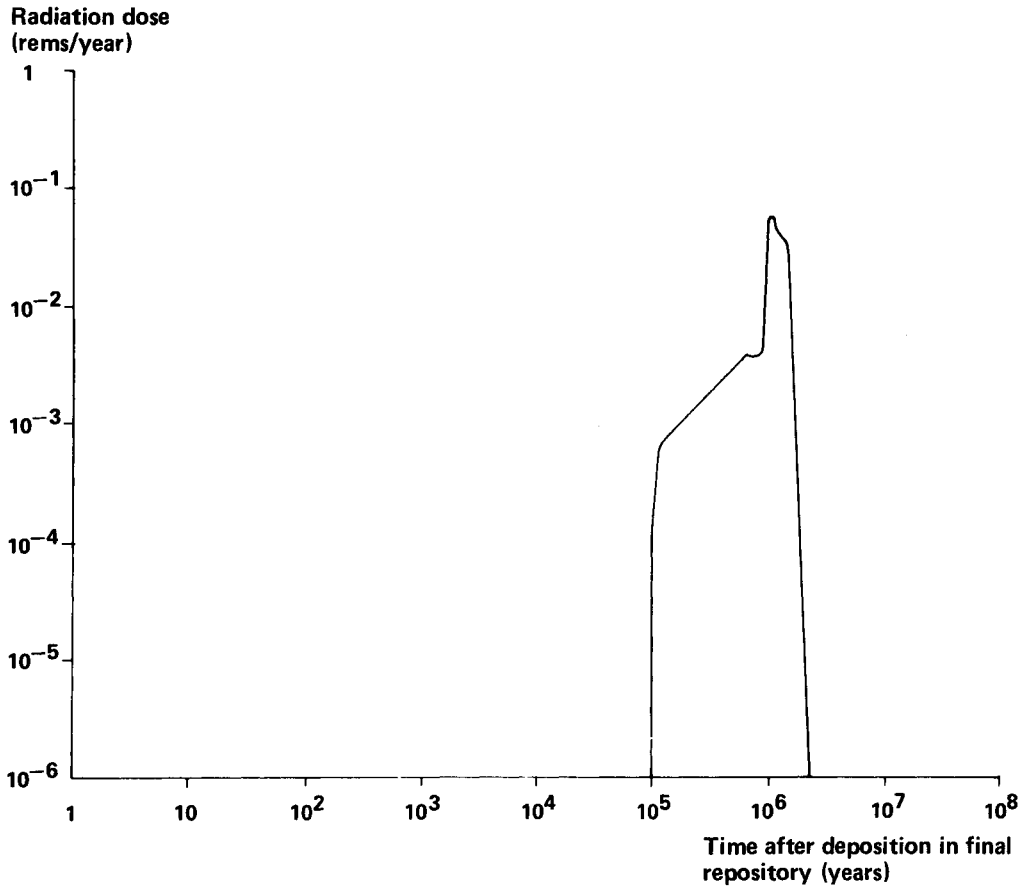


Figure 11-2. Calculated individual doses for critical group (nearby residents) for the pessimistic case with a well as the primary recipient.

from 100 000 years to around 3 million years following deposition.

In this case, the maximum radiation dose is calculated to occur after about one million years and to amount to about 70 mrems/year. Radium-226 makes the dominant contribution to the radiation dose in this pessimistic case as well.

Initial canister damage

In view of the inspection and quality control measures which will be adopted in connection with canister fabrication, the probability of initial canister damage is judged to be very low. But in order to illuminate the consequences of such an event, a case with initial canister damage to one container has been investigated. Individual doses for the well case occur after 3 000 years and are calculated to be 0.003 mrem/year; they derive from iodine-129. Otherwise, the consequences are largely simultaneous with those for the above-mentioned cases and depend upon the assumed dispersal scenario.

Extremely early commenced canister decomposition

One of the corrosion experts was of the opinion that the possibility that the canister's service life might be only a few thousand years cannot be excluded. As has already been pointed out in chapter 8, this is not consistent with the environment and the conditions which prevail in the final repository. If it is nevertheless assumed that the first canister penetration occurs after 3 000 years, radiation doses from iodine-129 can occur after only about 6 000 years.

These doses will be on the order of 0.5 mrem per year in the case with a well as the primary recipient. If it is assumed that decomposition of the canister proceeds at a uniform rate over 400 000 years, as in the main case, the same maximum radiation doses are obtained as for the main case. If the frequency of canister failure is higher during a given period of time, the dose from easily soluble components will increase. If it is conservatively assumed that the frequency is 10 times higher and that the entire iodine content of the fuel is available for leakage, the radiation dose from iodine-129 will increase 10 times during this period. Only a small portion (around 1%) of the iodine is directly accessible for leaching, however. A breakdown of the fuel structure which could expose a larger portion of the iodine is not considered to be a reasonable possibility.

Delayed canister penetration

If it is assumed that canister degradation begins after 500 000 years or later, roughly the same maximum radiation doses are obtained for both the main case's dispersal scenario and for the pessimistic dispersal scenario. The only difference is a shift in time.

Influence of organic complexing agents

The concentration of complex-forming substances in the groundwater is limited. They can therefore only transport a small portion of the radioactive substances. With pessimistic assumptions that the largest possible portion of heavy metals is complex-bound to organic substances and that these substances are transported at the velocity of the groundwater, certain radiation dose increments are obtained. These occur at the same time as the dose from iodine-129 and are calculated together with this to be about 3 mrems/year in the case with a well as the primary recipient. The dominant nuclide is plutonium-239.

Fissure width dispersion

The calculations of the inflow to the recipient have been carried out with the assumption of a constant fissure width. A dispersion effect is obtained if the cracks exhibit a certain size distribution. This causes a small portion of the inflow to take place at an earlier time than the average and the concentration maximum to decrease. An earlier inflow can lead to considerably higher concentrations of relatively short-lived nuclides which would otherwise decay. The overall radiological consequence of an earlier inflow is, however, small, since the doses are dominated by long-lived elements.

Thorium-229 serves as an example, where the inflow increases from $1.3 \cdot 10^{-11}$ Ci/year to $2 \cdot 10^{-5}$ Ci/year. However, the dose increases by a maximum of only about 0.02 mrem per year. The dose increment for cesium-135 is of the same order of magnitude, while it is lower for other nuclides.

Another and more important consequence of fissure width dispersion is a rather radical alteration of the distribution of the doses in time. The peak which is shown in figure 11-1 at 70 million years is widened and lowered considerably in analogy with what is shown in figure 11-3a. In the case of a pessimistic dispersal scenario, dispersion has much less of an effect, in analogy with figure 11-3b.

Collective doses

The collective radiation doses have been calculated for the different cases. See section II:8.5. They vary only to a relatively small extent with the premises. A maximum of 17 manrems per year is obtained for the main case and a maximum of 105 manrems per year for the pessimistic dispersal case.

The collective dose commitment for the most unfavourable 500-year period is 8 500 manrems and 53 000 manrems, respectively, which corresponds to 0.03 and 0.18 manrem per MWe and year.

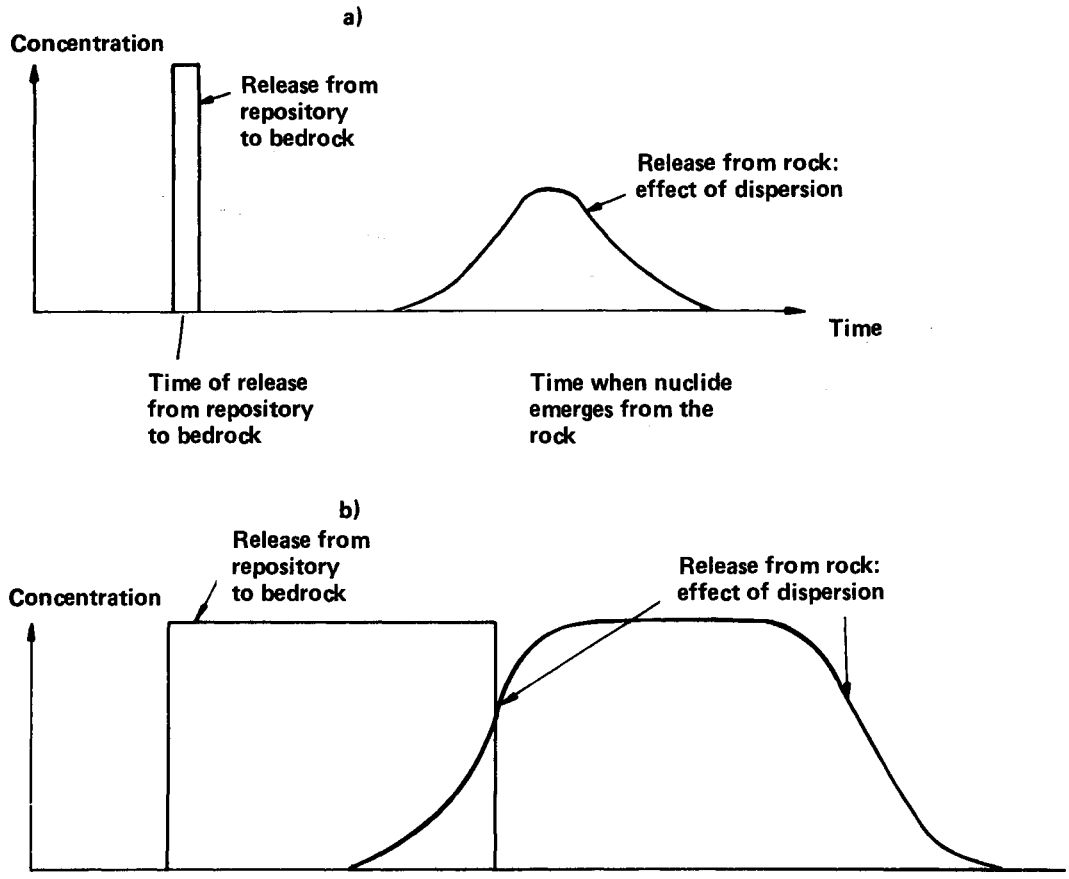


Figure 11-3. Effects of dispersion on pulse release and prolonged release without taking into account the lowering resulting from decay and dilution.

In the main case, the calculated collective dose commitments for different time periods are as follows:

Time period years	Collective dose commitment manrems
0 - 100 000	0
0 - 200 000	0,25 millions
0 - 1 000 000	11 millions
0 - ∞	2 000 millions

11.3.5 Consequences of slow dispersal from final repository for structural components

Certain metal components from the fuel assemblies which have become radioactive due to neutron irradiation are stored in concrete moulds in a separate repository at a depth of 300 m. The short-lived nuclides (cobalt-60 and nickel-63, with half-lives of 5.3 and 92 years, respectively) decay completely in the repository.

In the case of nickel-59, with a half-life of 80 000 years, the inflow to the recipient is a maximum of 0.06 Ci/year, which is

equivalent to an individual dose in the well case of less than 0.1 mrem/year. In the case of carbon-14, with a half-life of 5 730 years, radiation doses of around 1 mrem/year are obtained with unfavourable assumptions concerning dissolution time and retardation in the rock.

11.3.6 Heath effects

Even for the most unfavourable cases with pessimistically chosen data in the calculations, the health hazards resulting from the escape of radioactive substances from the repository are extremely small, if any. This applies both to nearby residents and to the rest of the population for all future time.

Assuming the usual relationship between radiation dose and health effect, the calculated collective doses for the main case and the pessimistic dispersal case are equivalent to 2 and 10 cases of cancer, respectively, among the population of the entire earth for the most unfavourable 500-year period.

The two nuclides iodine-129 and radium-226 dominate the collective dose. These nuclides are concentrated in the body in the thyroid gland and the skeleton, respectively. The chromosomes in the reproductive organs therefore receive a relatively low dose from these nuclides. On the basis of the collective gonad dose, the number of genetic defects is calculated to be no more that 0.1 case during a period of 500 years for the entire world population.

11.3.7 Comparisons with limit values and natural radioactivity

The ICRP recommended dose limit for a critical group which could be subjected to radiation doses over a series of years has recently been established at 100 mrems/year. Even with pessimistic assumptions concerning dispersal conditions, the calculated radiation doses from the final repository are below the limit recommended by ICRP.

The calculated maximum radiation dose (10 mrems/year) in the main case is on a level with the design goal specified by the National Institute of Radiation Protection for Swedish nuclear power plants. The goal is based on the principle that discharges of radioactive elements shall be limited as far as is reasonably achievable in view of the economic and social consequences of every effort to limit radioactive discharges.

It can also be of interest to compare calculated elevated levels of radioactive elements in the primary water recipients directly with those which occur naturally in water. This comparison has special relevance in view of the fact that the dose-dominant nuclide is radium-226, which also occurs in nature.

Radioactive element	Concentrations in natural water in Sweden (pCi/l)		Calculated maximum increase of level in primary recipients near final repository in main case (pCi/l)	
	Drinking water	Sea water[a]	Well	Lake
Radium-226	0.1-40	0.3	0.3	0.5
Uranium-238	0.1-1500[b]	0.3	6	0.1
Potassium-40[c]	ca 20	330	-	-
Cesium-135[c]	-	-	0.3	0.006

a) With 3.5% salt content.

b) Includes U-235 and applies to natural water (not necessarily drinking water).

c) Potassium-40 and cesium-135 are biologically comparable but have somewhat different dose factors (24 000 and 7 300 rems/Ci, respectively).

11.3.8 Influence of bedrock movements

In the preceding section, long-term safety has been dealt with in relation to the slow degradation processes which the waste canisters in the final repository could undergo.

In this and the following section, other processes which could conceivably affect the safety of the final repository are discussed.

In connection with the previous report on vitrified waste, a number of different studies were carried out in order to shed light on the occurrence of earthquakes and bedrock movements and how these could affect a final repository at a depth of 500 metres in the Swedish bedrock.

Rock movements could conceivably damage the canisters as well as alter hydrological conditions. The properties of the copper and the buffer material, however, allow a movement of several cm in the storage holes without the integrity of the canister being affected.

Some idea of the frequency and magnitude of rock movements over long periods of time can be obtained from studies of the fracture pattern in exposed rock outcrops. In order to examplify the frequency of movement in superficial layers of good rock, observations have been made in the Karlshamn area. The greatest observed displacement was 2 cm. On the basis of the frequency distribution of the magnitude of the displacements and under the assumption that the observed displacements have taken place in their entirety in a single deformation step, the probability of a displacement of 3 cm has been estimated. When applied to a final reposi-

tory, the result is that one or a few canisters would be affected over a period of 30 million years.

As is discussed in greater detail in section II:3.2.2, fracture movements during the next few millions of years could only lead to insignificant changes in the permeability of the rock.

The consequences of hypothetically assumed large rock movements which damage ten or so canisters in the repository would be less than those reported for the case of slow canister decomposition.

11.3.9 Risk of criticality in final repository

The possibility that criticality, i.e. a self-sustaining chain reaction, could occur with the fissionable plutonium-239 and uranium-235 which is present in the repository is virtually nil. This matter is dealt with in greater detail in KBS Technical Report 108. A brief summary is provided here.

Criticality with plutonium-239

In principle, sufficient plutonium-239 is present in a single canister from the start in order for criticality to be reached. But first some process involving penetrating water must separate the plutonium from the uranium in the fuel and collect it in concentrated form in a manner suitable for criticality. This is not feasible for a number of reasons.

Since the expected service life of the copper canisters is considerably longer than the half-life of plutonium-239, there will not be enough plutonium present for criticality if and when the copper canisters are penetrated. The case is therefore of interest only for a single container which could have initial canister damage.

Criticality inside a damage canister requires that the uranium present there be selectively dissolved and transported out of the canister before the plutonium-239 has had time to decay sufficiently.

Criticality outside of a damaged canister requires that uranium and plutonium be dissolved together and that plutonium be selectively precipitated in the bentonite buffer outside of the canister. Selective leaching of plutonium is not possible for chemical reasons.

Neither internal nor external criticality with plutonium is a reasonable possibility, however, since it requires a more rapid dissolution (on the order of 50 000 years) of uranium or of uranium and plutonium than what is calculated to be possible (1.8 million years) under existing conditions with regard to water flow and carbonate levels. Furthermore, the plutonium must be distributed in a manner which is appropriate for criticality, which is very improbable.

Criticality also requires that large portions of the bentonite material around the canister be lost. This is considered to be

impossible on the basis of existing knowledge of bentonite and rock properties.

Internal transport resistance in the canister will further impede the transport of material out of the canister.

In summary, the probability of plutonium criticality is extremely low. Furthermore, the consequences of a postulated criticality would be insignificant. The course of events is characterized by a slow heat generation and an increase of the temperature to a theoretical maximum of the boiling point of water (around 265°C at the pressure prevailing at a depth of 500 m). If the temperature tends to rise higher, water boils off and the reaction stops. The effects on nearby canisters will not be such that their integrity will be jeopardized. The amount of long-lived radiotoxic nuclides which are formed in connection with criticality is small in relation to the amount which is already present in the failed canister.

Criticality with uranium-235

Criticality with uranium-235 is not possible inside the canisters owing to neutron physics considerations. Such criticality is only possible in tunnel systems and storage holes outside of the canisters. Owing to the long half-life of uranium-235, the risk of criticality in this case does not apply only to an isolated initially damaged canister. Calculations show that the minimum critical mass in the tunnels is around 4 400 kg, which means that all of the uranium from at least four canisters must be accumulated within the critical geometry. The expected reducing environment, with bivalent iron in the bentonite, keeps uranium in the quadravalent state, which counteracts migration and local precipitation at any distance from the canisters. The risk of criticality can be completely eliminated by the addition of a few percent magnetite to the bentonite, thereby increasing neutron absorption to a sufficient level.

Even though the risk of criticality is thus extremely low or can be eliminated completely, the consequences of a hypothetical case have been calculated. A sudden, heavy release of energy is out of the question. Criticality can only be built up by a slow accumulation of uranium. The thermal power output from a hypothetical critical mass involving all of the uranium deposited in one of the tunnels would be lower than 130 kW. The formation of fission products would be equivalent to that of 900 tonnes of deposited fuel, which would not increase the radiation doses to any great extent in relation to the main case without criticality.

In summary, it can be concluded that a number of extreme or improbable prerequisites must be postulated in order to achieve a critical configuration. The probability of this occurring is judged to be negligible. The consequences of hypothetical cases of criticality from both plutonium-239 and uranium-235 have, moreover, been calculated to be insignificant.

11.3.10 Meteorite impacts

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 very long periods of time show that the probability of meteorite impact which would jeopardize the safety of a final repository is between 10^{-11} and 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.

11.3.11 Acts of war and sabotage

In the long time perspective which is relevant for 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 finally 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. The geological barrier would thus not be penetrated, but might 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, conceivable. 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 would also be limited compared to the other consequences of such acts of war.

Safeguards against sabotage acceptable to the authorities will be provided during intermediate storage, encapsulation and deposition in the final repository. 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 likely targets for sabotage in terrorist actions, the facilities described here are less attractive to potential saboteurs and are most closely comparable to other industrial plants where environmentally hazardous material is handled.

11.3.12 Future disturbance by man

It is conceivable that the knowledge of where the final repository is located may be lost in the distant future and that man at that time may, for some reason, perform drilling or rock work which results in contact with the waste. The final repository is situated in one of our most common types of rock which does not contain any valuable minerals which could conceivably be considered for profitable extraction. The depth and low water content of the impervious rock selected for this purpose make it highly improbable that deep wells will be drilled for water near the repository in the future. No reason can be seen for seeking out such great depths for the construction of rock storage caverns or the like. Furthermore, the loss of the knowledge of the location of the final repository presupposes that our current civilization

is destroyed as a result of some catastrophic event such as a global war of extermination or a new ice age. If the country were 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 prevent harm being done.

11.3.13 A future ice age

A new glaciation of the country is not expected to affect the integrity of the final repository. See chapter II:3.

11.4 SUMMARY SAFETY EVALUATION OF FINAL STORAGE OF SPENT UNREPROCESSED NUCLEAR FUEL

The spent nuclear fuel is isolated by means of encapsulation in copper canisters which are packed in highly-compacted bentonite in sound rock at a depth of 500 m. Rock tunnels and shafts are sealed with a mixture of 80-90% quartz sand and 10-20% bentonite. Metal scrap etc. from the spent fuel is embedded in concrete and stored in special rock tunnels at a depth of 300 m in sound rock. The tunnels are back-filled with concrete. The safety analysis of such a final storage method shows the following:

1. The groundwater chemistry in the Swedish bedrock at the depths for the final repositories (300-500 m) is reducing. The groundwater is virtually oxygen-free. The buffer material - highly-compacted bentonite - possesses extremely low water permeability, so material transport through the buffer mass is controlled by diffusion.

   The avaiable quantity of oxidants which can attack copper is low and the supply rate of oxidants is extremely slow. The copper canister can therefore be expected to have a virtually unlimited life and thereby prevent the dispersal of radioactive substances.

2. The probable consequence of final storage of the spent nuclear fuel in the described manner is that it will not lead to any dispersal of radioactive substances to the biosphere for a very long period of time - more than one million years. This means that the long-term consequences of the final storage of spent unreprocessed nuclear fuel are equivalent to the long-term consequences of the storage of unirradiated uranium dioxide in the same manner.

3. On the basis of a statement by a group of specialists, it is judged realistic to anticipate a minimum service life of hundreds of thousands of years for a copper canister with a wall thickness of 200 mm.

4. Over a time span of 100 000 years, the following important radioactive nuclides decay virtually completely: americium-241, plutonium-238 and -241, strontium-90, cesium-137 and carbon-14. Americium-243, plutonium-239 and -240 also decay to an appreciable extent.

5. If the copper canisters should be penetrated after a few hundred thousand years, some leaching of radioactive substances to the groundwater could take place. This would proceed extremely slowly and would probably take millions of years, even under pessimistic assumptions. After approximately one million years, the toxicity of the waste will be dominated by radium-226, which is a daughter product of uranium-238 with a half-life of 4 510 million years. Uranium-238 and its decay products will therefore persist, even over periods of time which are long by geological standards.

6. For a case with conservatively chosen data, but representing probable assumptions concerning water transit time and nuclide retardation (the main case in fig. 11-4), a maximum calculated radiation dose to individuals of approx. 10 mrems per year is obtained after 70 million years. The corresponding global collective dose is approx. 17 manrems/year.

7. For a case with very pessimistically chosen data for all important parameters (pessimistic case in fig. 11-4), a maximum calculated radiation dose to individuals of approx. 70 mrems per year is obtained after 1 million years. Corresponding global collective dose is approx. 105 manrems/year.

8. The health hazards are extremely small, if any, for the pessimistic case as well.

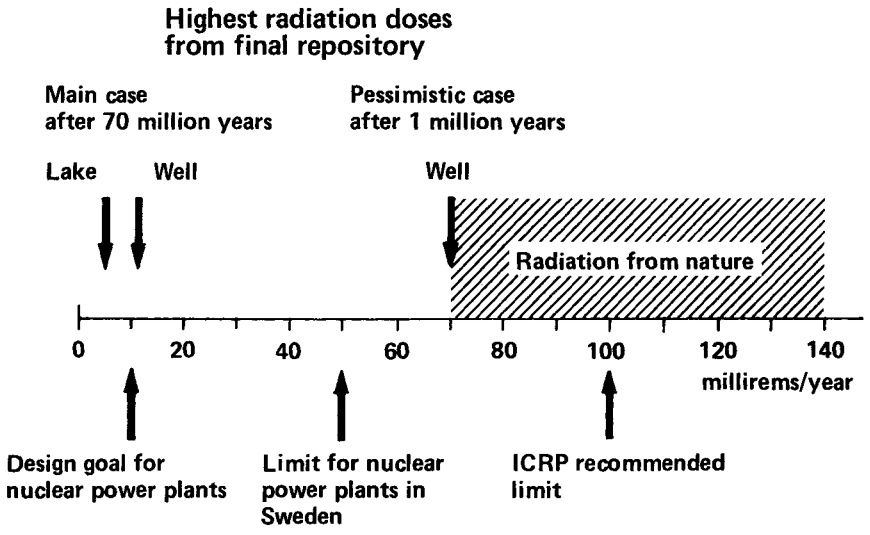


Figure 11-4. The maximum radiation doses from the final storage of spent fuel to which an individual can be exposed compared with some recommended limit values. ICRP is the International Commission on Radiological Protection. Radiation from nature varies from one place to another in Sweden and lies within the hatched area.

9. The calculated increase of the level of radioactive elements in the recipients to which waste products could conceivably spread is, even in unfavourable cases, comparable to natural levels of such elements.

10. The final storage of other waste obtained in the handling of spent unreprocessed nuclear fuel does not give rise to any dose increment of importance for the overall safety evaluation (maximum 1 mrem/year).

11. A number of extreme or highly improbable or unreasonable circumstances must be postulated in order for criticality in or adjacent to the repository to be reached. The probability of this happening is judged to be extremely small. If it is nevertheless assumed that criticality occurs, the process will proceed slowly and the consequences will be insignificant compared with those reported for the final storage scheme as a whole.

12. Even in those cases where a number of unfavourable assumptions have been made, the calculated changes in the background radiation are 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 recommended by the International Commission on Radiological Protection (ICRP). The proposed method for the final storage of spent fuel is therefore deemed to be absolutely safe.

11.5 TIME PERSPECTIVE

The time spans which are dealt with in the safety analysis are of such an order of magnitude that they can scarcely be conceived of in our normal frames of reference. A somewhat better overview of the situation may be obtained by considering the anticipated future sequence of events divided into different phases.

It is hereby assumed that the earth will remain the abode of some form of human life throughout this sequence of events. Otherwise, the discussion of the effects of the final repository on ecological systems on the earth is without interest.

Several thousand years in the future

During this phase, the copper canisters will remain completely intact, with the possible exception of a few isolated canisters which may have been defective from the start. The quantity of radioactivity which can leak out from these defective canisters and reach the biosphere gives rise to negligible doses.

For several thousands of years to come, the final repository can therefore be regarded as being "absolutely safe", regardless of how this concept is defined.

From several thousands of years to several hundreds of thousands of years in the future

Calculations show that there is no chance of any release of radioactivity to the biosphere during this period of time, even if pessimistic assumptions are made concerning the condition and function of the various barriers. During this phase, it is not impossible that the canisters could begin to be penetrated and groundwater could come into contact with the spent fuel. However, the chemical conditions which prevail in the buffer mass and in the rock comprise an effective barrier to any dispersal of radioactivity. From a geological and geochemical viewpoint, conditions during this period of time are not difficult to foresee; nor could major climatic changes affect a final repository at a depth of 500 metres in the Swedish bedrock.

It is therefore asserted that the final repository will fulfill the requirement of being "absolutely safe" during this time period as well.

From several hundreds of thousands of years onward

The safety analysis shows that, with pessimistically chosen assumptions, some release of radioactivity can occur to the biosphere after about 1 million years. The dose increment which would result from this release has been calculated to be of the same order of magnitude or lower than the limits which are applied today and lower than the doses which stem from natural radiation. However, in this time perspective of millions of years, it is not considered to be meaningful or reasonable to discuss in any greater depth the impact of a final repository in relation to present-day standards. Nor has any attempt at such an evaluation been made within other areas of human endeavour which could have long-term environmental impact - not even for time perspective which corresponds to the first phase dealt with above.

In view of the long-term geological stability exhibited by the bedrock in those parts of the country which could be considered for the location of a final repository, the possible impact of a final repository would be very limited. The impact of radiation on ecological systems and their evolution would be dominated by natural radiation, in comparison with which the impact of the final repository would be small locally and negligible globally.

APPENDIX 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
AB Exportspråk
Forsgren Produktion AB
The Mining Industries' Work Study and Consultancy Programme
Gränges Mining Mineral Process Laboratory
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 Grufveaktiebolag
The Swedish Academy of Engineering Sciences
Teleplan AB
AB Vattenbyggnadsbyrån
WP-system AB

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Personnel from the Swedish State Power Board, Oskarshamnsverkets Kraftgrupp AB (OKG), the Swedish Nuclear Fuel Supply Company (SKBF), the Swedish Central Power Supply Board (CDL) and the National Council for Radioactive Waste Management (PRAV) have also contributed their services to KBS. The project has organized reference and working groups within the following areas:

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