Final Storage of Spent Nuclear Fuel – KBS-3

I General
II Geology
III Barriers
IV Safety

SKBF/KBS

Final Storage of Spent Nuclear Fuel – KBS-3

I General

SKBF/KBS
Swedish Nuclear Fuel Supply Co/Division KBS
MAILING ADDRESS: SKBF/KBS, Box 5864, S-102 48 Stockholm, Swede
Telephone: 08-67 95 40

PREFACE

This report - KBS-3 - describes how the spent nuclear fuel from the Swedish reactors can be disposed of in a manner that satisfies high requirements on safety in both the short and long term. The account is based on a body of data that has been assembled in Sweden and other countries over the past few years. A large number of research scientists, consultants and industrial firms have participated in the research and investigation work that has been conducted within the Swedish Nuclear Fuel Supply Company, Division KBS. Some of these have also assisted in compiling the report. The Swedish Nuclear Fuel Supply Company is, however, responsible for the contents of the report.

Stockholm in May 1983
SWEDISH NUCLEAR FUEL SUPPLY COMPANY

Erik Svenke

Lars B. Nilsson

TABLE OF CONTENTS

1 INTRODUCTION

1.1 BACKGROUND

In April of 1977, the Swedish Parliament passed the law concerning special permission to charge nuclear reactors with fuel etc. - known as the "Stipulation Act". The Act sets forth requirements on special permission from the Government for charging a reactor with nuclear fuel for the first time so that a self-sustaining nuclear chain reaction can take place. Permission may be granted only providing that the reactor owner

1 has produced an agreement which adequately satisfies the requirements 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 reprocessing can be effected, or

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

When the Stipulation Act was passed, the Swedish nuclear power utilities organized the KBS project for the purpose of investigating and reporting how the requirements set forth in the Act could be met. Subsequently, the KBS project was incorporated as a permanent department (Department for Handling and Final Storage of Nuclear Power Waste) in the Swedish Nuclear Fuel Supply Co. (SKBF), which is jointly owned by the Swedish nuclear power utilities. KBS's function within SKBF now includes all coordinated research and development activities for the nuclear power utilities with respect to the handling and final storage of the radioactive waste from nuclear power. The KBS project has presented two main reports concerning the alternatives specified in the Stipulation Act:

- Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste (KBS-1), November 1977, and

- Handling and Final Storage of Unreprocessed Spent Nuclear Fuel (KBS-2), September 1978.

The first report formed the basis for fuelling applications for the Ringhals 3 and 4 and Forsmark 1 and 2 reactors. After the submission of certain supplementary information, the Government granted permission in 1979 and 1980 to charge these reactors with nuclear fuel. The second report has not been used for any fuelling application.

Since KBS-1 and KBS-2 were published, the following events with a bearing on waste management have taken place.

- In accordance with a resolution passed by the Parliament in the Spring of 1980, the Swedish nuclear power programme shall include no more than 12 reactors. The last reactor in Sweden shall be closed down by no later than 2010 /1-1/.

- The so-called Financing Act (1981:669) entered into force as of 81-07-01. This Act regulates the division of responsibility between the nuclear power utilities and the government authorities with respect to the planning, implementation and financing of the measures required for a safe disposal of the radioactive waste.

- In March 1983, the Nuclear Legislation Committee submitted its report on legislation in the field of nuclear energy to the Government /1-2/. On the basis of this report, the Government is expected to place a Bill before the Parliament in the autumn of 1983 for a new law governing nuclear activities.

Swedish legislation requires of the nuclear power utilities that they: a) demonstrate as instructed by the Stipulation Act that a safe handling and final storage of the high-level waste or spent nuclear fuel can be effected, and b) annually submit plans in accordance with the provisions of the Financing Act on how they intend to realize the necessary measures. This report intends to fulfil the former requirement. The latter requirement is fulfilled through the submission of plans to the National Board for Spent Nuclear Fuel (NAK). Such a plan was submitted for the first time in June of 1982 /1-3/.

The background material on which the previous report concerning the final storage of spent nuclear fuel, KBS-2, was based is now 4-5 years old. Considerable development of both models and methods of analysis, as well as a broadening of the data base, has taken place both in Sweden and abroad. This development warrants a new report, independent of the previous report, even though the general principles and some of the background material are the same.

1.2 GENERAL PRINCIPLES

On the basis of current legislation and prevailing public opinion in Sweden, the following basic principles for the final storage of radioactive waste can be established:

- A very high level of safety is required, in both the short and the long term.

- Burdens on future generations shall be avoided.

- It shall be possible to carry out the necessary measures with the highest possible degree of national independence.

From the principle that the highest possible degree of national independence is desirable, it follows that the final storage must be able to be arranged in Sweden and with a technology that is available in Sweden.

Different conceptual methods for the disposal of radioactive wastes have been proposed internationally. Table 1-1 shows some of these methods, with comments on their feasibility.

Table 1-1. Principles for the final storage of high-level waste or spent nuclear fuel.

Methods		Comments
1	Supervised storage	Does not constitute a final solution
2	Launching into space	Technology not available in Sweden today
3	Separation and transmutation	Technology not available in Sweden today
4	Deep geologic proposal	Possible in Sweden's stable bedrock
5	Disposal in deep-sea sediment	No sites available within Swedish territory
6	Injection into isolated deep geologic formations (aquifers)	Suitable areas not known in Sweden
7	Disposal under inland ice sheets or under permafrost	No areas of sufficient extent available in Sweden

In Sweden, like in many other countries with similar natural conditions, efforts have been concentrated on studies of deep geologic disposal in crystalline rock (method 4). Salt and clay strata as well as sedimentary rocks are regarded as interesting geologic media in certain countries, where they occur to a sufficient extent.

1.3 OUTLINE OF THE REPORT

This report describes how a system for safe final disposal of Swedish spent nuclear fuel can be designed on the basis of the state of knowledge as of the turn of the year 1982/83. The extensive research work that is underway or planned in Sweden and many other countries will progressively increase our understanding of the various mechanisms that are of importance for the long-range safety of a final repository. This will enable increasingly reliable evaluations to be made of the long-term function of the repository and will lead to modifications of the design of the system. The purpose here is therefore not to specify how and where a final

repository will be constructed, but only how and where it can be constructed in such a manner as to achieve the required safety. No attempt has been made at economic optimization of the system. Account has, however, been taken of the fact that the described solution must lie within the bounds of what is economically feasible. Plans for the realization of the various measures as well as estimated costs have been reported in /1-3/.

The report deals with all handling, transport, storage and treatment that is necessary to create a safe final storage of spent nuclear fuel from the Swedish reactor programme. Those links in the chain that are already under realization, i.e. the transport system and the central storage facility for spent fuel (CLAB), have been described in detail in connection with current or completed review by government authorities. These links are therefore described only briefly, and the greatest emphasis is laid on those links in the handling chain that directly concern the final storage itself and the long-range safety assessments.

Strictly formally, this report could, regarded as supporting documentation for applications for permission to fuel the Forsmark 3 and Oskarshamn 3 reactors, be restricted to dealing with the quantity of spent fuel that will come from these two reactors. But since no final decision has yet been made as to how the spent nuclear fuel from the Swedish nuclear power programme is to be finally disposed of, such an approach is considered unappropriate. The account has therefore been made to embrace the total system planned for the disposal of all Swedish radioactive waste from nuclear power. Hence, the capacities of the various facilities have been based on the entire quantity of spent nuclear fuel arising from the Swedish nuclear power programme. The type, size and location of the reactors is shown in figure 1-1. The quantity of fuel for which reprocessing contracts have been signed constitutes only a small portion of the total quantity of fuel and has therefore not been taken into account here.

Swedish reactors

Reactor	Type	Power	Commercial operation
Oskarshamn 1	BWR	440 MW	1972
Oskarshamn 2	BWR	570 MW	1974
Oskarshamn 3	BWR	1060 MW	(1986) ?
Barsebäck 1	BWR	570 MW	1975
Barsebäck 2	BWR	570 MW	1977
Ringhals 1	BWR	750 MW	1976
Ringhals 2	PWR	800 MW	1975
Ringhals 3	PWR	915 MW	1981
Ringhals 4	PWR	915 MW	1984
Forsmark 1	BWR	900 MW	1980
Forsmark 2	BWR	900 MW	1981
Forsmark 3	BWR	1060 MW	(1985) ?
Central storage facility for spent nuclear fuel (CLAB)			(1985)

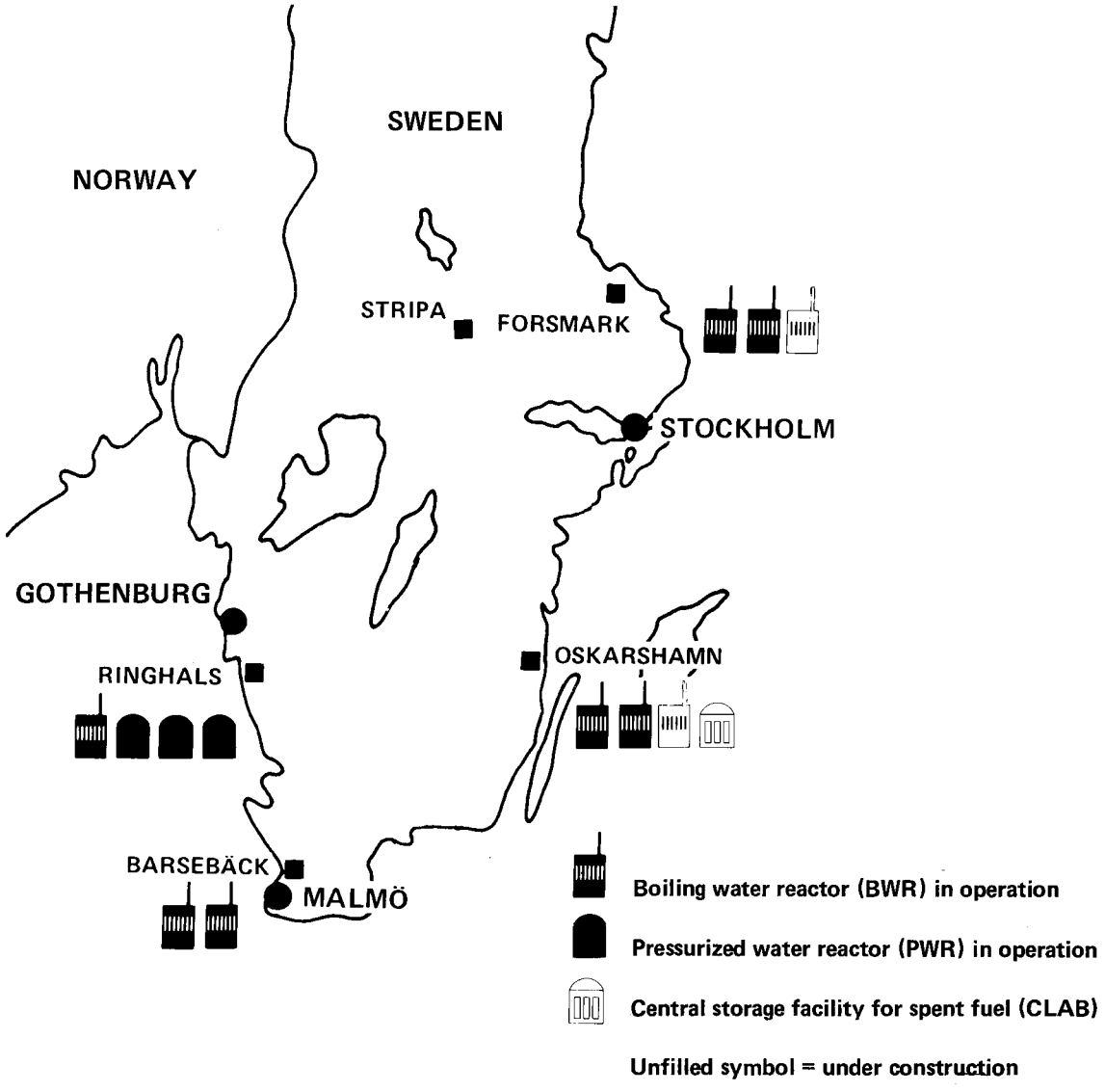


Figure 1-1. The Swedish nuclear power programme and the geographical location of the facilities.

The report has been divided into volumes as follows.

Volume	Chapter	Contents
	Summary	
I General	1-2	Introduction and review of the handling sequence
	3	Description of the spent fuel
	4	Description of facilities and the handling procedure in them
II Geology	5-8	Geological investigation methods, groundwater movements and water chemistry plus stability of the bedrock
III Barriers	9-13	Chemical conditions in and around the storage holes
	14-16	Transport of nuclides from repository via geosphere and biosphere to man plus dosimetry
IV Safety	17	Principles for the safety evaluation
	18	Site-specific conditions in investigated areas
	19-21	Safety assessment of handling and final storage encompassing normal processes, sequences, accidents and extreme events
	22	General safety conclusions

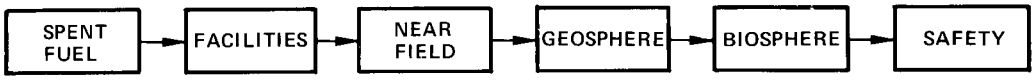
The first 17 chapters of the report deal with general questions not directly connected to any specific site. Chapter 18 deals with the site-specific conditions that constitute the basis for the choice of input parameters in the safety calculations for the storage phase. The safety analyses are reported in chapters 19 to 21, and chapter 22 provides a summary assessment of the safety with which a final storage of spent nuclear fuel can be effected in Swedish bedrock.

1.4 BACKGROUND MATERIAL

References to background material are made in the form of two numbers between oblique strokes, the first of which refers to the chapter and the second to the reference given in the list at the end of each volume (example: /6-4/ means reference 4 in chapter 6).

Most of the background material prepared within SKBF/KBS has been published in the form of technical reports, designated KBS TR Year-No. (example: KBS TR 80-26). In the case of reports within the international Stripa Project, there are two series of reports designated SAC-No. and Stripa TR Year No. All of these reports are available from SKBF, the library at Studsvik, Sweden and the INIS in Vienna, Austria.

2 HANDLING SEQUENCE AND FUNCTION OF THE FINAL REPOSITORY



This chapter provides a brief description of the handling stages in the form of transports, storage and treatment that are required for a safe final storage of spent nuclear fuel. Furthermore, a general description is provided of the final repository and its long-term function.

2.1 GENERAL

2.1.1 Premises

The fundamental principle for the system presented here is that the long-term safety of the repository shall not be dependent upon supervision and corrective measures after the final repository has been sealed.

The system consists of facilities for transport, temporary storage, treatment and final storage (disposal). The treatment involves enclosing the fuel in highly durable canisters prior to deposition.

In all facilities except for the final repository, elements and operations are employed for which experience exists to a greater or lesser extent from industrial applications. The account is concentrated on subsystems where little or no experience is available. In order to meet the high demands on safety, the multiple barrier principle is applied in the final repository, which means that the safety of the repository is not completely dependent on the function of one single barrier. The different barriers shall complement each other via mechanisms that are independent of each other to as great an extent as possible. Moreover, the function of each individual barrier has been evaluated on the basis of unfavourably chosen (pessimistic) assumptions.

2.1.2 The spent fuel

During the operation of the nuclear power stations, the nuclear fuel is altered in such a manner that new atomic nuclei are created, partly through the decay of uranium and other heavy elements and partly through neutron absorption in the fuel and the surrounding material. Many of these new atoms have a surplus of energy, which is emitted in the form of radiation. Most of this energy surplus is emitted very rapidly, within the first few hours after nuclear fission in the reactor has ceased. Of the remainder, a large portion decays during the first months and years. The radioactivity that remains after about 10 years consists for the most part of elements that require several hundred to a thousand years to decay. There are, however, atoms with such long decay periods that they can be equated with very long-lived naturally occurring radioactive elements. Such elements require analysis of the function of the repository over time spans that have not previously been considered in human technological evaluation, even though this would have been equally motivated in many cases.

2.1.3 Function of the final repository

A main characteristic of the potential toxicity of the spent fuel is the initially high toxicity level and the very long period of time with persistent but limited toxicity. The purpose of the repository is to protect human beings from unacceptable radiological impact. This can be achieved in two ways. One is to contain the radioactive substances for a sufficiently long period of time to allow the process of decay to reduce activity to acceptable levels. The other is that the radioactive substances are diluted, i.e. released and dispersed so slowly that the maximum concentrations that can reach man are acceptably low. Both of these means are employed in the final repository system described here. During the first phase, safety is based on the containment principle. However, since complete containment cannot be maintained for an unlimited period of time, safety after a very long time (hundreds of thousands to millions of years) will be dependent upon the rate of release from the containment, the rate of dispersal in the geosphere and the resultant dilution of the remaining radioactive substances (see figure 2-1).

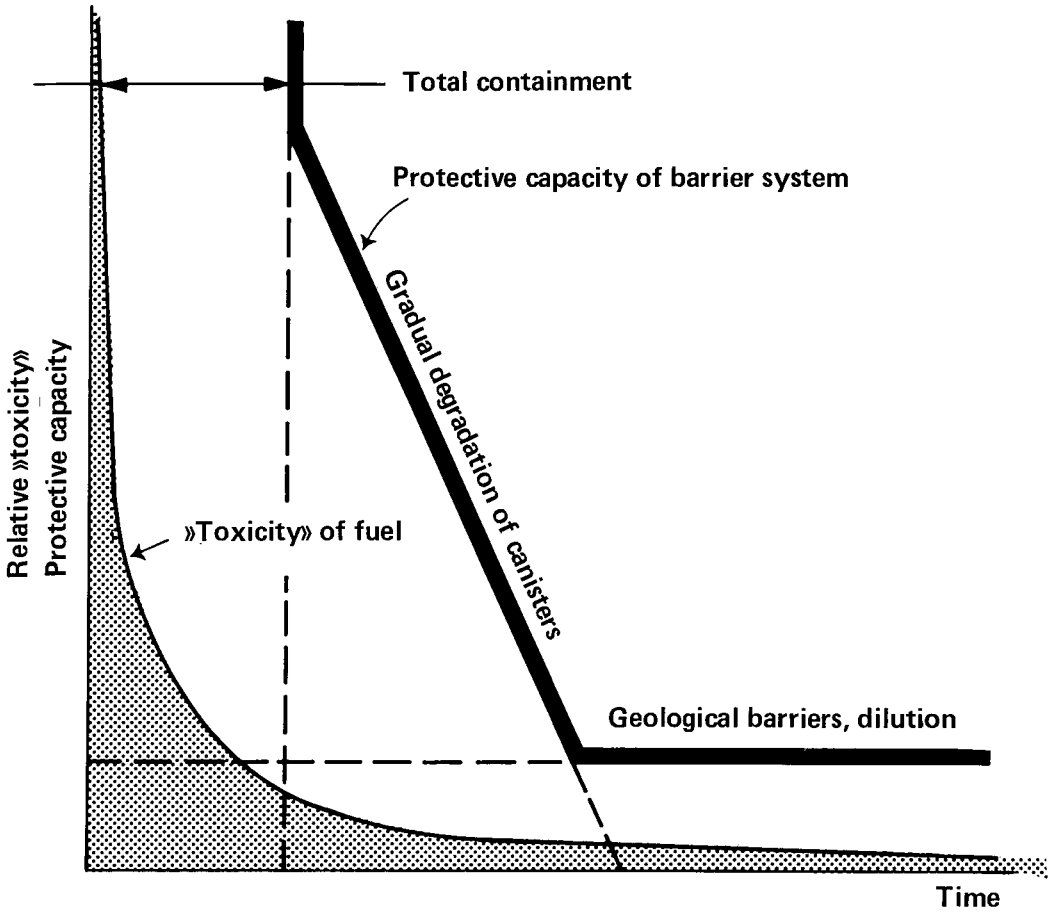


Figure 2-1. Schematic function of the barrier system.

The containment is created by encapsulating the fuel. It can be shown from experiments and theoretical considerations that a long-range total containment can be achieved under the conditions prevailing in Swedish bedrock. The dilution function, which must function during much longer periods of time, can be demonstrated not only by means of experimental and theoretical considerations, but also by comparisons with geological phenomena in nature.

The basic design of the final repository is the same as that previously described in KBS-1 and KBS-2 and entails enclosing the spent fuel in impervious and durable canisters, which are deposited in boreholes drilled in the bottom of rock tunnels at great depth. The canisters are surrounded in the deposition holes with a buffer material, and finally, the entire tunnel system is backfilled. This basic design of a final repository is being considered in many other countries with similar natural geological conditions, e.g. Finland, France, Japan, Canada, Switzerland and the USA /2-1/. In

the Swedish studies, however, the long-term barrier function of the canister has been accorded greater importance than in other countries.

2.2 HANDLING SEQUENCE

The handling sequence for spent nuclear fuel is illustrated by figure 2-2 (the numbers in the text refer to the figure).

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.

2 The fuel is then transported to the central storage facility for spent fuel, CLAB. During transport, the fuel is enclosed in a special cask (2a), which meets the requirements specified in international regulations. The sea transport takes place with a ship specially built for the purpose.

The CLAB, as well as every nuclear power station, has a harbour suitable for these shipments. Terminal handling takes place with the aid of special trailers.

3 The fuel is stored in the CLAB for a period of 40 years. Storage takes place in water-filled pools situated in rock caverns. The facility is built in an initial stage for a storage capacity of 3 000 tonnes of fuel. In order to increase capacity beyond this limit, new rock vaults may be excavated in later stages.

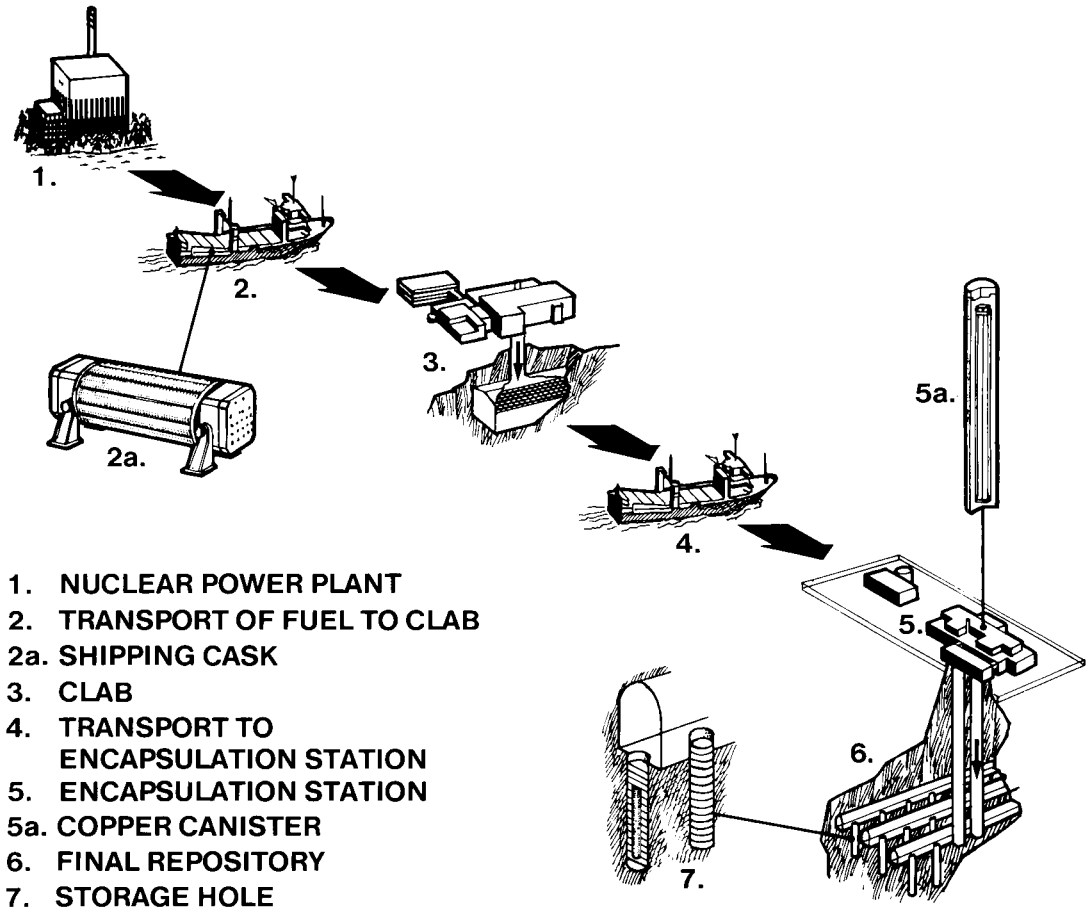


Figure 2-2. Handling sequence for spent nuclear fuel.

4 After 40 years of storage, the fuel is transported to an encapsulation station. This transport can take place in the same manner as the transfer from the power station to the CLAB. The shipping casks can be used for both sea and land transport.

5 The encapsulation station is situated above ground in connection with the final repository. In the encapsulation station, the spent fuel is enclosed in copper canisters (5a). The amount of fuel in each canister is limited to about 1.4 tonnes so that the temperature at the canister surface will be well below 100°C. The space between the fuel rods is filled with lead or copper. A total of 4 400 copper canisters are required for the Swedish programme.

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 parallel tunnels. Deposition holes 1.5 m in diameter and 7.5 m deep are drilled in the floors of the tunnels. One waste canister is stored in each hole. The repository can be made in one or more storeys. In the case of a single-storey repository, the tunnels are spaced at 25 m, and in the case of a 2-storey repository, 33 m.

7 In the deposition holes, the copper canisters are surrounded by blocks of highly compacted bentonite. Bentonite is a clay that swells when it absorbs water.

After all canisters have been deposited, the facility can be sealed by filling all tunnels and shafts with a mixture of sand and bentonite.

2.3 FUNCTION OF THE FINAL REPOSITORY

2.3.1 Penetration of groundwater into the repository

During the construction and deposition phase, the excavated rock vaults are drainage-pumped. This lowers the water table over the repository. When the repository is sealed, drainage pumping is stopped. The groundwater then starts to intrude into the repository at the same time as the water table slowly returns to its natural level.

When water comes into contact with the buffer material, which consists of highly compacted bentonite, it starts to swell and fill up all cavities in the deposition holes. The hydraulic conductivity of the buffer material will then be at least as low as that of the surrounding rock mass. No water flow of significance can take place in the impervious buffer material. Transport of the substances dissolved in the water is dominated completely by diffusion.

When water penetrates into the repository, the bentonite in the backfill in tunnels and shafts also swells. Here as well, permeability is reduced to levels below that of the surrounding rock.

2.3.2 Canister corrosion

Copper does not corrode in pure water. However, the groundwater may contain dissolved substances that can attack the copper canister. When the groundwater has penetrated through the buffer, such substances can diffuse through the water-filled pores and reach the canister surface. An extremely slow decomposition of the canister material may then begin.

The rate at which the copper canister corrodes is determined by the canister environment, the flow of water in the rock, the groundwater's content of corrosive substances and their transport through the clay buffer, and the rate of transport of corrosion products away from the canister surface. The location and depth of the repository are chosen with the intention of limiting the rate of water turnover and avoiding an unsuitable groundwater composition. The buffer material constitutes a transport barrier between the groundwater and the copper canister and affects the chemical environment in the repository.

2.3.3 Fuel leaching

If a canister is penetrated, the groundwater can come into contact with the spent fuel. The fuel then starts to dissolve in the water and radioactive substances can escape to the surroundings. This process of dissolution proceeds very slowly, mainly due to the fact that the substances contained in the fuel are poorly soluble and that the supply of water is very limited. The dissolution rate is also affected by the chemical composition of the groundwater and the chemical environment created in the deposition hole by the combination of buffer, canister, fuel and groundwater. Temperature and radiation are also of importance.

2.3.4 Transport of radioactivity in the geosphere

The geological barrier has two essential properties: in the first place, low hydraulic conductivity; in the second place, the ability

to bind most substances dissolved in the water. Many substances will occur in poorly soluble chemical form. Low hydraulic conductivity will mean slow transport to the ground surface. The sorption properties of the fracture faces and the diffusion of the dissolved substances into microfissures in the rock will further delay the transport of the radioactive substances. Due to these factors, together with the long previous isolation of the substances in the canisters, the radioactive substances will decay and be transformed to stable and harmless nuclides to a great extent before they reach the biosphere.

The frequency of fractures in the bedrock and the hydraulic conductivity of the rock and the fractures, along with the topography of the area, determine the groundwater's flow paths and velocity.

2.3.5 Dispersal in the biosphere

The deep groundwater reaches the biosphere in so-called "outflow areas". Outflow can take place in a well or a fresh water lake with connecting water courses and finally reach a saltwater system. Each recipient has its characteristics with regard to dilution and further dispersal in the biosphere.

Dispersal in the biosphere can be calculated with the aid of models that take into account how different substances are transferred from e.g. water to fish and from fish to man. Some dispersal pathways are simple and easy to analyze, for example the transfer of a substance from a well to man via drinking water. Other dispersal pathways are long and complicated with many steps, where each step is controlled by specific transfer factors between the different parts of the biosphere.

A fundamental uncertainty in evaluating dispersal via food chains follows from the long period of time that will pass before the radioactive substances reach the biosphere. During this time, evolution may have altered both the human species and the ecology. This fundamental uncertainty must be taken into account in the total safety evaluation.

2.4 SITING ASPECTS

Only for one of the facilities included in the handling system described here, namely the CLAB, has a siting decision been made. Construction is currently underway at Oskarshamn.

A final repository can only be built at a site where the presence of a sufficiently large rock formation with the geological, hydrological and geochemical properties required for a safe final repository has been demonstrated. Siting is only secondarily determined by factors of an economic or social nature. The transport system permits siting both on the coast and inland.

Today's plans assume that a final decision will be reached on the site of the final repository towards the end of the 1990s.

It would seem advantageous that the treatment station where the spent fuel is to be encapsulated be co-sited with the final repository. This has the advantage of geographically concentrated activities permitting coordination gains and rational transports. Coordination in time also speaks for such a solution.

2.5 REPOSITORY DEPTH

In determining the depth of a final repository, it is necessary to take into consideration a number of factors that are more or less dependent on site-specific conditions.

Firstly, the repository must be located deep enough to make sure that

- the function of the repository is not jeopardized by long-term erosion, including glaciations, and events on the surface of the ground (explosions, underground construction, drilling of wells)

- the hydraulic conductivity of the rock is sufficiently low

- reducing conditions prevail

On the other hand, the repository must not be located so deep that

- high rock stresses jeopardize the stability of the rock vaults during the construction and deposition phase

- the natural temperature is so high that working conditions cannot be managed satisfactorily or that pre-established requirements on maximum temperature during the storage phase ($100^{o}C$) cannot be fulfilled

- that investigations from the surface cannot be carried out.

The results of the site investigations, along with general geological knowledge, indicate that the requirements stated in the first group above will be met if the repository is located at a level deeper than 400-500 m. The requirements in the second group should not pose any problem as long as the repository is located above the 1000 m level.

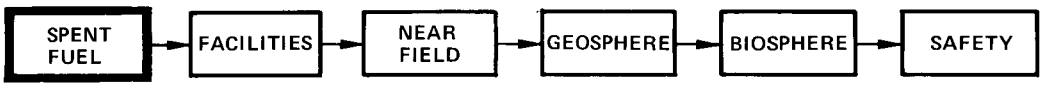
In a later phase, when the time comes to finalize the detailed design of a final repository at a specific site, special studies will be made to determine the appropriate repository depth in the light of site-specific conditions, at the same time as which an attempt will be made to optimize the total barrier system.

2.6 FLEXIBILITY AND DEVELOPMENT POTENTIAL

The present-day state of knowledge on which this account is based is sufficient to demonstrate that a safe final storage of spent nuclear fuel can be effected. Moreover, continued technological and scientific progress will always offer a potential for better solutions than those that are judged at a given point in time as being the best available.

The handling chain proposed here, with 40 years' supervized intermediate storage of the fuel prior to final storage, offers opportunities for a couple of decades of continued and more detailed studies of methods and materials against the background of both an improved state of knowledge and more detailed safety regulations that are expected to be forecoming both nationally and in international cooperation.

3 SPENT NUCLEAR FUEL



This chapter provides a description of the spent nuclear fuel and presents the fuel data on which the described handling chain and the safety analysis are based.

3.1 PREMISES

3.1.1 Quantity of spent nuclear fuel

The total quantity of spent nuclear fuel from the Swedish nuclear power programme adopted by the Swedish Parliament will be dependent on how long the individual reactors remain in operation. This cannot be specified at the present time. However, no reactor is to be operated after the year 2010.

The calculations in this report are based on a total quantity of spent fuel equivalent to 6 000 tonnes of uranium, 4 550 tonnes from BWRs and 1 450 tonnes from PWRs. An operating period of 25 years for each reactor has thereby been assumed, as in the report - Plan 82 - submitted to the National Board for Spent Nuclear Fuel /3-1/. If all reactors are kept in operation up to and including the year 2010, the total quantity of spent fuel will be slightly over 7 000 tonnes. The burnups given in Table 3-1 have been used in the calculation. As a result of the conservative rounding-offs that have been done at various stages in the calculations and the margins of safety that have been included, all conclusions of any importance will be unchanged if the assumed quantity of fuel should increase by about 20%.

3.1.2 Other premises

The following assumptions have been made as a basis for calculations of fuel quantities and activity content in this report.

- As reference fuels for the handling sections, fuel intended for Forsmark 3 and Oskarshamn 3 (ASEA-ATOM) has been used for BWRs and fuel intended for Ringhals 3 (Westinghouse) for PWRs.

- The spent fuel is encapsulated after a cooling period of 40 years from the time the fuel is taken out of the reactor and then deposited directly after encapsulation.

- The fuel bundles are encapsulated without prior disassembly. Boxes from BWR fuel and boron glass rod bundles from PWRs are separated and disposed of separately.

- In the calculation of nuclide content for the safety analysis, it is assumed that all fuel derives from PWRs with a burnup of 38 000 MWd/tU. This leads to an overestimate of the total quantity of radioactive substances in the repository by more than 20%.

3.2 SPENT NUCLEAR FUEL, 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 core in a light water reactor contains a large number of fuel assemblies. The number of fuel assemblies and the total quantity of uranium vary with reactor size and type. For the reactors in use in Sweden, the amount of uranium in the core is between 70 and 126 tonnes. Boiling water reactors (BWRs) have a lower power density and a larger quantity of fuel for a given size than pressurized water reactors (PWRs).

The fuel in the core is partly exchanged in connection with the annual reactor shutdowns. Each fuel assembly is irradiated for between three and seven years before it reaches full burnup. This varies at the present time from 25 000 to 35 000 megawatt-days per tonne of uranium (MWd/tU) for BWRs and 30 000 to 40 000 MWd/tU for PWRs. As can be seen in table 3-1, a higher burnup is planned in the future.

Table 3-1. Planned average burnup in Swedish reactors (MWd/tU)

Year of discharge	BWR	PWR
- 1985	30 000	34 000
1986 - 1990	32 000	38 000
1991 -	36 000	40 000

As the fuel is used, radioactive elements are created in the fuel assemblies through the following physical process:

- nuclear fission of uranium and generated plutonium produces fission products, for example cesium -137 and strontium -90

- neutron capture in uranium produces isotopes of elements that are heavier than uranium (transuranium elements). Successive neutron captures produce heavier and heavier elements, for example neptunium (Np), plutonium (Pu), americium (Am) and curium (Cm)

- neutron capture in cladding tubes and the other metal parts of the fuel produces activation products, for example cobalt -60 and nickel -59.

Table 3-2 gives examples of the composition of spent fuel.

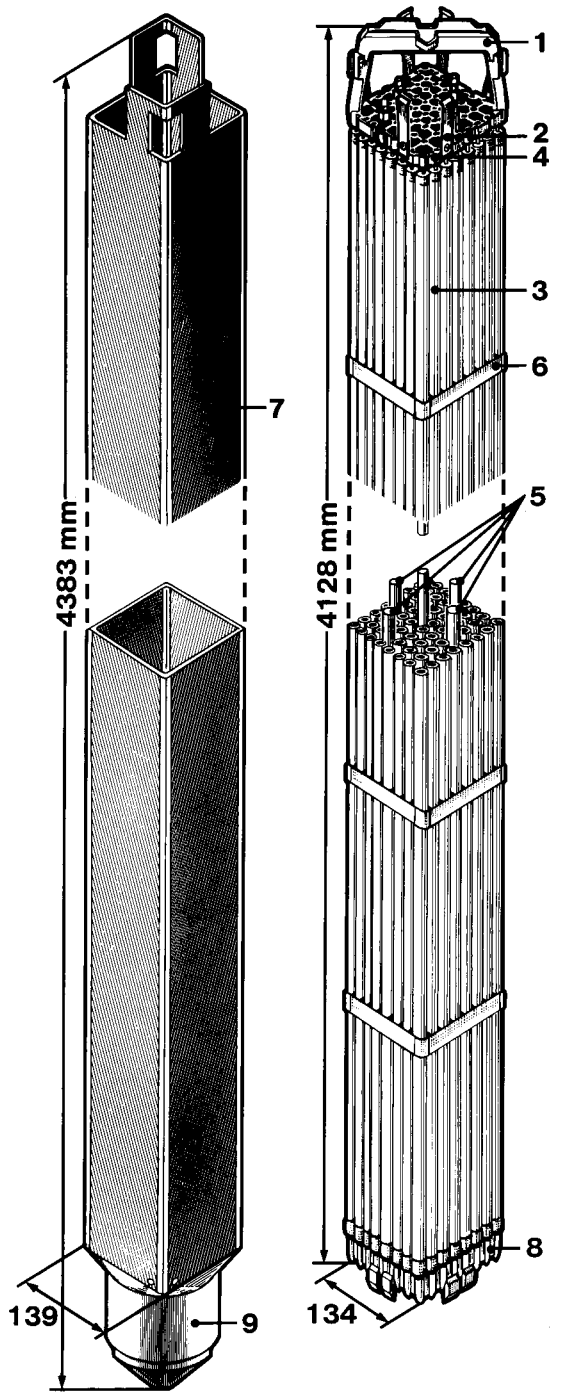
Table 3-2. Examples of composition of fuel (percentage by weight) on discharge from the reactor

	BWR, 28 000 MWd/tU	PWR, 38 000 MWd/tU
Fission products	2.9	3.9
Uranium (of which 0.6%-0.8% fissionable)	96.2	95.1
Plutonium (approx. 75% fissionable)	0.8	0.9
Other heavy nuclides	0.06	0.09

3.3 DESCRIPTION OF THE REFERENCE FUEL

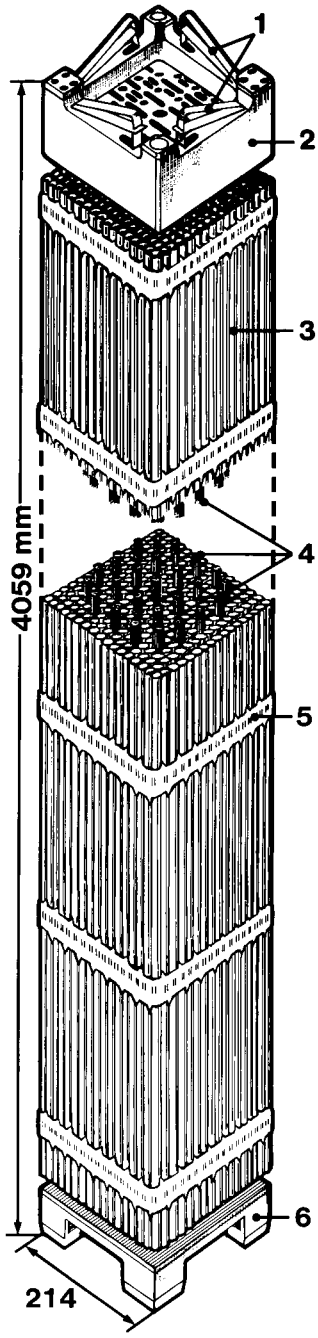
The construction of the fuel assemblies differs depending on which type of reactor they are intended for /3-2/.

Figure 3-1 shows a fuel assembly for the Forsmark 3 and Oskarshamn 3 reactors (ASEA-ATOM). It consists of a fuel bundle in a fuel box. The fuel bundle contains 64 rods (3), one of which is a spacer-capture rod, which does not contain uranium. The rods are held in position by a top tie plate (4), a bottom tie plate (8) and six spacer grids (6). Four of the fuel rods are tie-rods (5) and their ends go through the top and bottom tie plates and are locked with nuts. The ends of the other rods are stuck into holes in the tie plates. The fuel bundle is lifted by means of a handle (1) attached to the top tie plate. The bundle is enclosed by a box (7) with lifting lugs at the top for handling and a transition piece (9) at the bottom on which the assembly rests in the reactor. The spring clips (2) hold the fuel assembly in place in the reactor core. A BWR assembly weighs about 300 kg. Other data for the fuel are given in table 3-3.



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

Figure 3-1. Fuel assembly for a BWR reactor (ASEA-ATOM).



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

Figure 3-2. Fuel assembly for a PWR reactor (Westinghouse Ringhals 3 and 4).

Table 3-3. Data for unirradiated fuel assemblies

Material quantities (kg/assembly)	BWR (F3/03)	PWR (R3)
Fuel bundle		
Uranium	178	464
Oxygen in UO_2	24	62
Spring clips	2.2	5
Cladding	48	105
Top tie plate	1.6	8
Bottom tie plate	1.1	6
Guide tubes for control rods		10
Spacer grids	0.9	6
Spacer capture rod	0.8	-
Total	256	666
Fuel box		
Fuel channel	35	
Transition piece	6	
Total	41	
Other data		
Number of fuel rods per assembly	63	264
Rod diameter (mm)	12.25/11.75	9.50
Rod length (mm)	3 998	3 852
Length of fuel bundle (mm)	4 128	4 059
Width of fuel bundle (mm)	134	214
Width of top tie plate (mm)	151	
Length of fuel box (mm)	4 383	
Width of fuel box (mm)	139	

Figure 3-2 shows a fuel assembly for Ringhals 3 (Westinghouse). It contains 289 positions with 264 fuel rods (3), 24 control rod guide tubes (4) and one guide tube for neutron flux measurement. The guide tubes (thimbles) are welded to a top tie plate (top nozzle) (2) and bolted by means of weld-locked bolts in a bottom tie plate (bottom nozzle) (6). They are also held fast by eight spacer grids (grid assemblies) (5). The fuel rods are attached to the spacer grids by spring clips and are not connected to the bottom or top tie plate. The fuel bundle is handled by means of a grip in the top tie plate. Spring clips on the top tie plate (1) restrain the fuel in the reactor core. No box is used for PWR fuel. One PWR assembly weighs about 670 kg. Other data are given in table 3-3.

The structural components in the BWR and PWR assemblies are made of stainless steel, inconel, incoloy or zircaloy. Some fuel assemblies in the first core in a PWR reactor contain neutron-absorbing boron glass rods. The boron glass rods are bound together in bundles of 16, 20 or 24 rods in each. The bundles accompany the assembly when it is discharged. Data for these rods are given in table 3-4.

Table 3-4. Data for boron glass rod bundles, Ringhals 3

Number of bundles	68
Number of pins/bundle	16/20/24
Weight/bundle (kg)	16/19/23
Length of bundle (mm)	3 850
Width of bundle (mm)	168

The initial fuel for Forsmark 3/Oskarshamn 3 has been described above. Other BWR fuel differs in details. For example, BWR fuel can have up to four capture rods. The future may also see the use of fuel of the SVEA type /3-3/ or fuel with more fuel rods.

The reference fuel for PWRs is representative for fuel in Ringhals 3 and 4. Fuel with a smaller number of rods (204 fuel rods and 21 guide tubes) is used in Ringhals 2. The weight of the uranium is nearly identical, however.

The above-described differences in the design of the fuel are of no appreciable importance for the handling procedures and the investigations that are reported in this report.

3.4 RADIOACTIVITY AND RESIDUAL HEAT IN SPENT NUCLEAR FUEL

In the encapsulation station, handling of the fuel bundles on the one hand and of the fuel boxes and boron glass rod bundles on the other hand are separate. The source strengths of these components are therefore given in separate sections.

3.4.1 Fuel bundles

The fuel bundles contain the uranium fuel as well as the metal components of the fuel such as cladding, top and bottom tie plates, spacer grids etc. The composition of the uranium fuel is shown in table 3-2. It contains fission products, uranium, plutonium and other transuranium elements, formed by successive neutron capture, as well as the decay products of these nuclides. The metal components of the fuel contain activation products.

The data given in the following for the activity content of the fuel bundle have been calculated by means of the computer program ORIGEN 2 /3-4/. Data for PWR fuel with a burnup of 38 000 MWd/tU (reference fuel) have been used in these computations. The calculations are described in greater detail in /3-5/ and /3-2/, where data for BWR fuel and for other burnups are also reported.

Table 3-5 gives the activity of the most important heavy nuclides and their daughter products at different times after discharge. The decay scheme for these nuclides is shown in figure 3-3. Table 3-6 gives the activity in fission products at different times and table 3-7 the corresponding data for the activation products. Figure 3-4 shows how the total activity in the fuel bundle declines with time.

Power density during reactor operation lies between 20 and 40 MW/tU. After the reactor is shut down and the spent fuel is discharged, it still generates some heat, known as residual heat (decay power), which derives from the disintegration of generated radionuclides. This residual heat is, however, small in comparison with the reactor's operating power and it declines with time. One minute after shutdown, the heat output of the fuel is down to 5% of the operating output and then continues to drop rapidly. In table 3-8 and figure 3-5, the residual heat in the reference fuel is given at different decay times longer than one year /3-5/.

Strontium-90 and cesium-137 dominate the residual heat in the beginning. The heavy nuclides predominate after just over a hundred years. Residual heat output in the spent fuel declines by roughly two orders of margnitude between one and a thousand years after discharge.

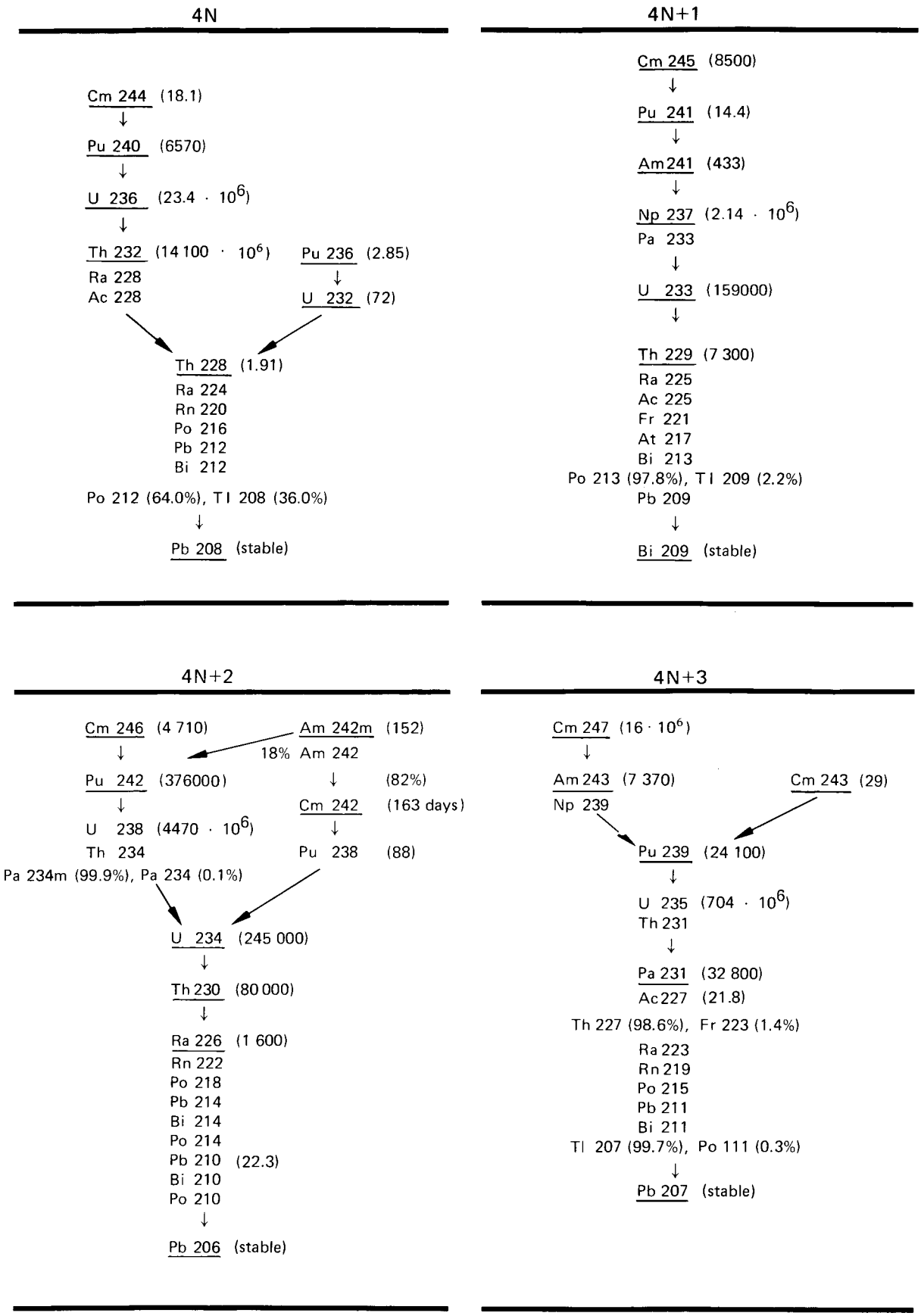


Figure 3-3. The decay chains of the heavy nuclides. Half-lives are given within parentheses in years. Half-lives shorter than 20 years have generally not been included. N is an integer. Nuclides in the 4N chain have atomic weights that are integral multiples of 4. All heavy nuclides are included in these decay chains. There is no connection between them.

Table 3-5. Radioactivity of heavy nuclides (GBq/tU) in spent fuel (PWR, 38 000 MWd/tU)

Nuclides	Half-life (years)	Activity (GBq/tU) after 40 yrs	100 yrs	1 000 yrs	10 000 yrs	100 000 yrs	10^6 yrs	10^7 yrs
^{243}Am	7 370	1 200	1 200	1 100	480	0,1		
^{239}Np	*	1 200	1 200	1 100	480	0,1		
^{241}Am	433	180 000	190 000	44 000	5,9	0,004		
^{242}Pu	376 000	110	110	110	100	89	18	
^{241}Pu	14,4	930 000	52 000	12	5,9	0,004		
^{240}Pu	6 570	14 000	14 000	13 000	4 800	0,35		
^{239}Pu	24 100	11 000	11 000	11 000	8 100	630		
^{237}Np	$2,14 \times 10^6$	16	19	48	56	56	41	2,3
^{233}Pa	*	16	19	48	56	56	41	2,3
^{238}U	$4\ 470 \times 10^6$	12	12	12	12	12	12	12
^{234}Th	*	12	12	12	12	12	12	12
^{234m}Pa	*	12	12	12	12	12	12	12
^{236}U	$23,4 \times 10^6$	10	10	10	13	14	13	10
^{235}U	704×10^6	0,52	0,52	0,52	0,59	0,89	0,89	0,89
^{231}Th	*	0,52	0,52	0,52	0,59	0,89	0,89	0,89
^{234}U	245 000	52	67	89	85	70	16	12
^{233}U	159 000	0,003	0,007	0,15	2,3	20	44	2,3
^{232}U	72	0,78	0,44	-	-	-	-	-
^{231}Pa	32 800	0,001	0,002	0,011	0,11	0,70	0,89	0,89
^{227}Ac	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{227}Th	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{223}Ra	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{219}Rn	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{215}Po	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{211}Pb	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{211}Bi	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{207}Tl	*	0,0007	0,001	0,011	0,11	0,70	0,89	0,89
^{230}Th	80 000	0,018	0,048	0,74	7,4	44	18	12
^{229}Th	7 300	-	-	0,006	0,81	18	44	2,3
^{225}Ra	*	-	-	0,006	0,81	18	44	2,3
^{225}Ac	*	-	-	0,006	0,81	18	44	2,3
^{221}Fr	*	-	-	0,006	0,81	18	44	2,3
^{217}At	*	-	-	0,006	0,81	18	44	2,3
^{213}Bi	*	-	-	0,006	0,81	18	44	2,3
^{213}Po	*	-	-	0,006	0,81	18	44	2,3
^{209}Pb	*	-	-	0,006	0,81	18	44	2,3
^{226}Ra	1 600	-	0,001	0,13	5,9	44	18	12
^{222}Rn	*	-	0,001	0,13	5,9	44	18	12
^{218}Po	*	-	0,001	0,13	5,9	44	18	12
^{214}Pb	*	-	0,001	0,13	5,9	44	18	12
^{214}Bi	*	-	0,001	0,13	5,9	44	18	12
^{214}Po	*	-	0,001	0,13	5,9	44	18	12
^{210}Pb	*	-	0,001	0,13	5,9	44	18	12
^{210}Bi	*	-	0,001	0,13	5,9	44	18	12
^{210}Po	*	-	0,001	0,13	5,9	44	18	12
Total		$1,3 \times 10^6$	330 000	71 000	15 000	1 600	760	210

* Daughter nuclide in equilibrium with parent nuclide.

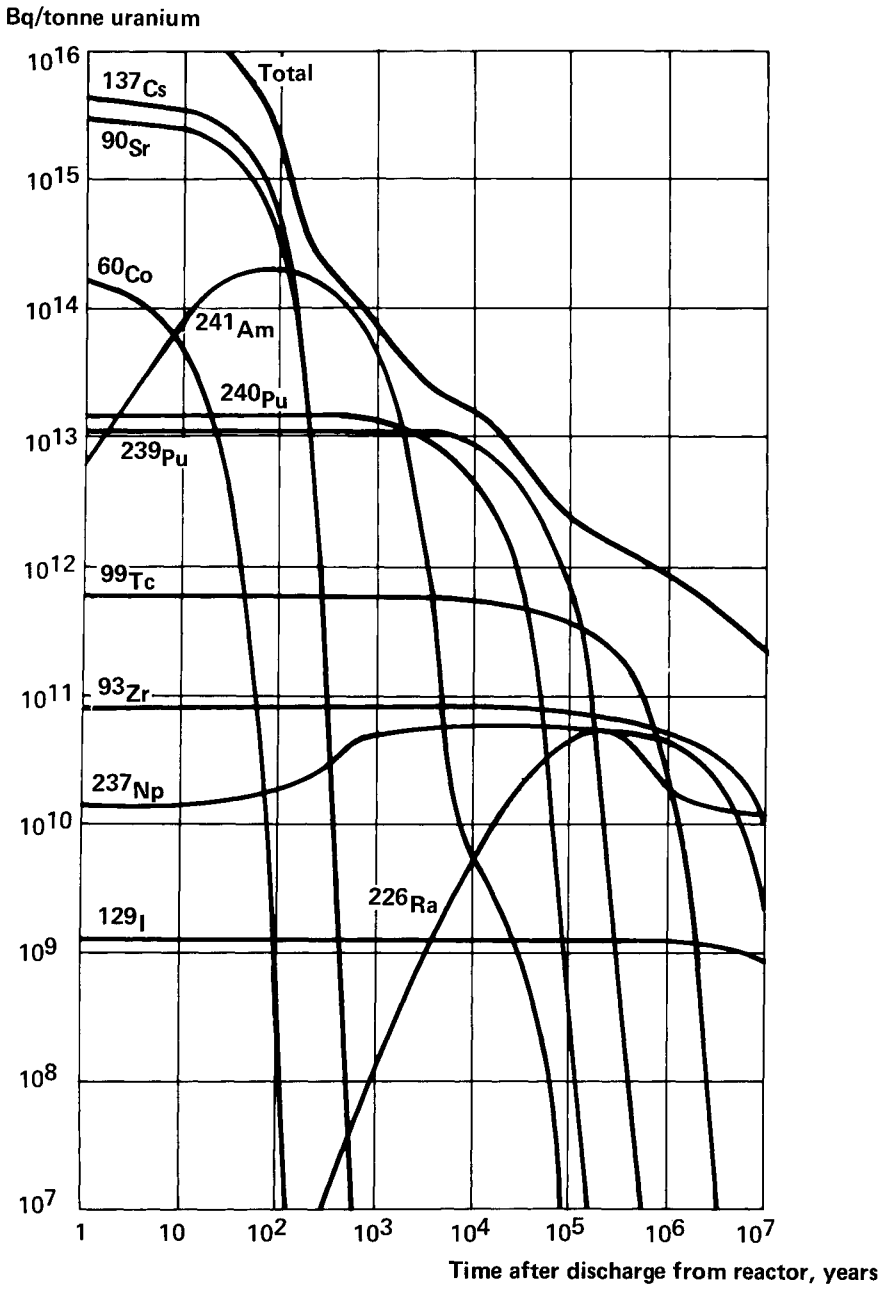


Figure 3-4. Radioactivity in spent fuel. The graph shows the radioactive elements in a PWR fuel bundle with a burnup of 38 000 MWd/tU, power density 38.5 MW/tU and enrichment 3.2 % uranium -235.

Table 3-6. Radioactivity of fission products (GBq/tU) in spent fuel (PWR, 38 000 MWd/tU)

Nuclide	Half-life (years)	Activity (GBq/tU) after 40 yrs	100 yrs	1 000 yrs	10 000 yrs	100 000 yrs	10^6 yrs	10^7 yrs
^{3}H	12,3	2 500	85					
^{14}C	5 730	0,005	0,005	0,004	0,002			
^{79}Se	65 000	17	17	17	16	5,9		
^{85}Kr	10,7	28 000	590					
^{90}Sr	28,8	$1,1 \times 10^6$	270 000					
^{90}Y	*	$1,1 \times 10^6$	270 000					
^{93}Zr	$1,5 \times 10^6$	74	74	74	74	70	48	0,81
^{93m}Nb	*	63	70	70	70	67	44	0,78
^{99}Tc	214 000	560	560	560	520	410	21	
^{107}Pd	$6,5 \times 10^6$	5,2	5,2	5,2	5,2	5,2	4,8	1,9
^{126}Sn	100 000	34	34	34	31	17	0,033	
^{126m}Sb	*	34	34	34	31	17	0,033	
^{126}Sb	*	4,8	4,8	4,8	4,4	2,4	0,004	
^{129}I	16×10^6	1,3	1,3	1,3	1,3	1,3	1,3	0,85
^{135}Cs	3×10^6	14	14	14	14	14	11	0,7
^{137}Cs	30,2	$1,7 \times 10^6$	440 000					
^{137m}Ba	*	$1,6 \times 10^6$	410 000					
^{151}Sm	90	10 000	6 700	6,3				
Total		$5,7 \times 10^6$	$1,4 \times 10^6$	810	780	600	130	5,0

* Daughter nuclide in equilibrium with parent nuclide.

Table 3-7. Induced activity in metal components of fuel, broken down by nuclides, PWR (fuel boxes and boron glass rods not included)

Nuclides	Half-life (years)	Activity (GBq/tU) after 10 yrs	100 yrs	1 000 yrs	10 000 yrs	100 000 yrs	10^6 yrs	10^7 yrs
^{14}C	5 730	12	12	11	3,6	-	-	-
^{60}Co	5,3	910	0,3	-	-	-	-	-
^{59}Ni	75 000	140	140	140	130	60	0,02	-
^{63}Ni	100	17 000	11 000	12	-	-	-	-
^{93}Zr	$1,5 \times 10^6$	3,7	3,7	3,7	3,7	3,5	2,3	0,04
^{93m}Nb		1,5	3,7	3,7	3,7	3,5	2,3	0,04
^{94}Nb	20 000	12	12	11	8,2	0,36	-	-
Total		18 000	11 000	180	150	70	5	0,1

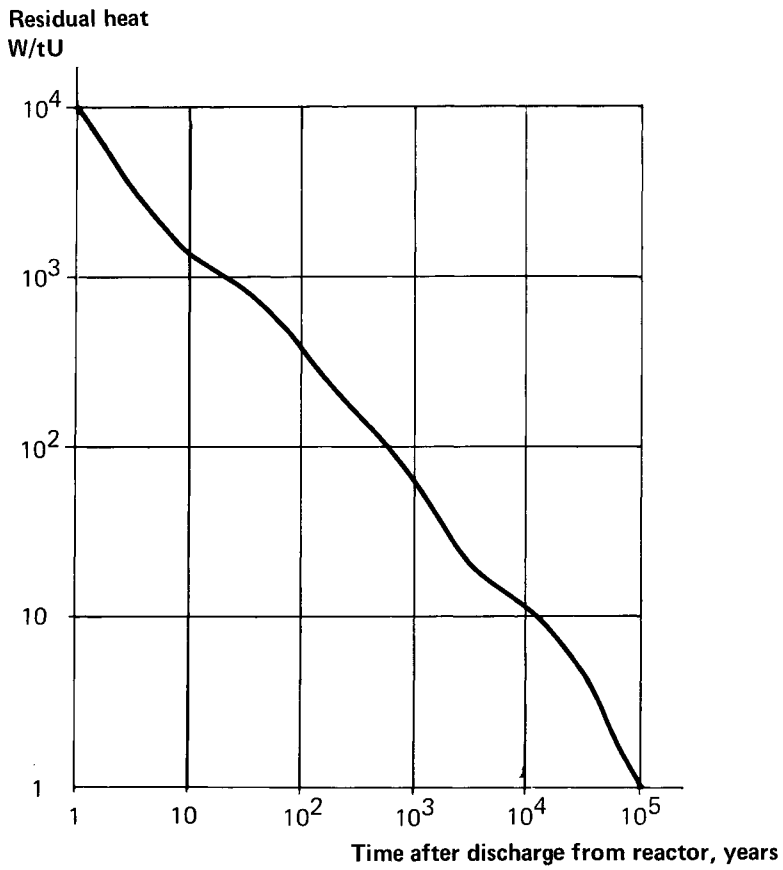


Figure 3-5. Residual heat in spent fuel from PWR. Burnup is 38 000 MWd/tU, power density 38.5 MW/tU and enrichment 3.2 % uranium -235.

Table 3-8. Residual heat in spent fuel:
PWR 38 000 MWd/tU, 38.5 MW/tU, 3.2% U-235
BWR 33 000 MWd/tU, 22.0 MW/tU, 2.8% U-235.

	Power (W per tonne uranium)					
	PWR			BWR		
Time (years after discharge)	FP	HN	Total	FP	HN	Total
1	9 600	330	9 900	6 800	280	7 100
10	1 000	320	1 300	860	280	1 100
40	440	310	740	370	270	650
100	110	250	360	90	230	320
1 000	0.023	61	61	0.020	57	57
10 000	0.022	12	12	0.019	11	11

FP = fission products
HN = heavy nuclides

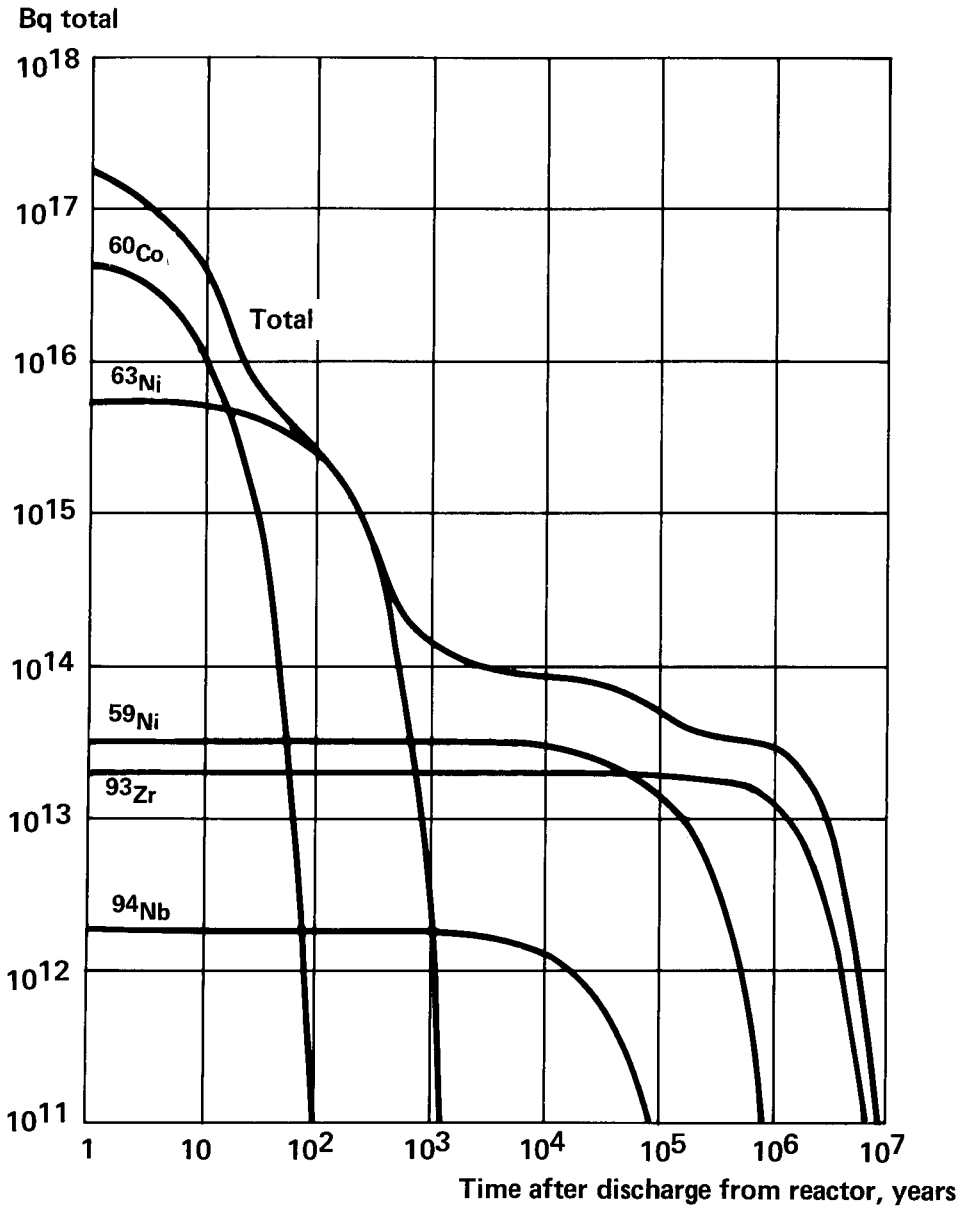


Figure 3-6. Radioactivity in fuel boxes and boron glass rod bundles. Total in repository.

Table 3-9. Induced activity in fuel boxes and boron glass rod bundles (total in repository).

Nuclides	Half-life (years)	Activity (TBq) after 40 yrs	100 yrs	1 000 yrs	10 000 yrs	100 000 yrs	10^6 yrs	10^7 yrs
^{14}C	5 730	59	59	53	18	0,3	-	-
^{60}Co	5,3	210	0,07	-	-	-	-	-
^{59}Ni	75 000	33	33	32	30	14	0,006	-
^{63}Ni	100	3 800	2 400	2,8	-	-	-	-
^{93}Zr	1,5 x 10^6	20	20	20	20	19	12	0,02
^{93m}Nb		8,9	20	20	20	19	12	0,02
^{94}Nb	20 000	1,9	1,9	1,8	1,3	0,06	-	-
Total		4 100	2 600	130	90	52	30	0,04

1 TBq (tera bequerel) = 1 000 GBq (gigabequerel)

3.4.2 Fuel boxes and boron glass absorber rods

A total of about 25 000 fuel boxes and 204 boron glass rod bundles are obtained from the Swedish nuclear power programme. In calculating the number of fuel boxes, it has been assumed that they are used only once. In reality, some fuel boxes will be used several times, making the number fewer. Here, however, the higher number is assumed.

The fuel boxes and the boron glass rod bundles contain induced activity only. Total activity in these components at different points in time is shown in table 3-9 and figure 3-6. Heat generation in this waste is low, on the order of 1 W/t after 40 years.

3.5 VARIATIONS AND UNCERTAINTIES IN DATA

The data given in section 3.4 relate to a special case, PWR fuel (Ringhals 3) with a burnup of 38 000 MWd/tU. This case has been chosen as a reference point for the safety analysis. The assumption that all fuel is of this type and has this burnup entails an overestimate of the total activity content of the final repository. Various factors that influence the estimated activity content of the fuel and residual heat are discussed in this section.

3.5.1 Fuel type

As has been mentioned, different types of fuel assemblies are used in PWRs and BWRs. The operating conditions in the fuel also differ, and as a rule, PWR fuel attains a higher burnup. It therefore also has a higher initial enrichment. For fuel that has reached the same burnup, these differences only have a marginal effect on the composition of long-lived radioactive products in the fuel. The difference is less than 10%. The differences are greater during the years immediately following discharge from the reactor, owing to the difference in power density, but after only 10 years, they have evened out. This also applies to the residual heat.

Through improvements in fuel sizing on the basis of reactor physics considerations, it will be possible to achieve higher burnups in

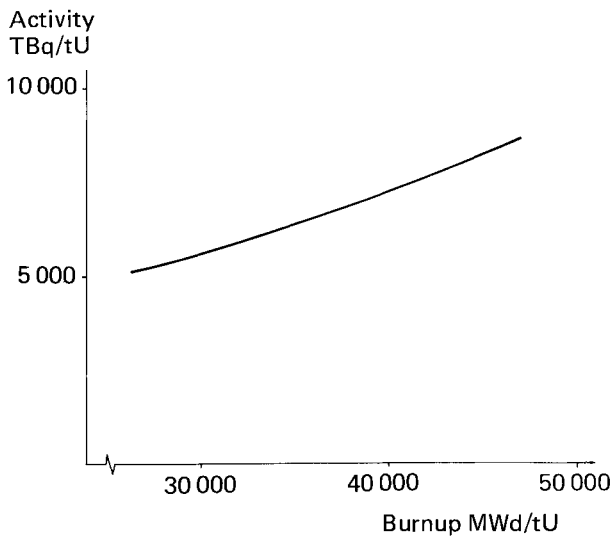


Figure 3-7. Total activity in spent fuel 40 years after discharge from the reactor.

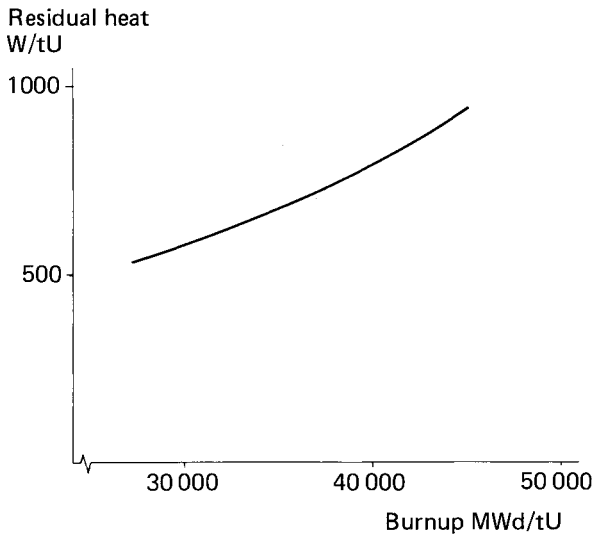


Figure 3-8. Residual heat in spent fuel 40 years after discharge from the reactor.

the fuel without increasing enrichment in the future. Due to more efficient utilization of the fuel, the residual content of uranium -235 will then decrease.

3.5.2 Burnup

Burnup directly affects the activity content of the spent fuel and thereby also the residual heat, as is evident from figures 3-7 and 3-8. This is of importance for the choice of time when the fuel is to be deposited in the final repository and for how much fuel can be placed in one canister.

Section 3.2 states that the average burnup for BWR fuel is currently about 30 000 MWd/tU and for PWR fuel about 34 000 MWd/tU, and that this burnup is planned to be increased according to table 3-1. These values apply for normal reload fuel. The fuel in the reactors' first cores reaches a lower burnup, which is also the case for some of the fuel in the final core. This means that in all, for the Swedish nuclear power programme, spent fuel will be obtained with very different burnups, between 10 000 and 50 000 MWd/tU. Most of the fuel assemblies will lie within the range 30 000-40 000 MWd/tU, and the average value for the total quantity, about 6 000 tonnes of

fuel, will be about 30 000 MWd/tU. At this burnup, the residual heat flux is about 580 W/tU 40 years after discharge from the reactor.

3.5.3 Recycling of plutonium and uranium

The data presented in this chapter have been calculated for pure uranium fuel. Current plans call for some Swedish fuel to be reprocessed. The plutonium and uranium obtained in connection with reprocessing is intended to be reused in Swedish reactors. This has been described in /3-6/. The type of fuel that is then used, MOX fuel, will contain a higher concentration of transuranium elements and have a higher residual heat flux. How large the increase will be is dependent on whether the fuel is reused in a BWR or a PWR. In /3-6/, it is concluded that these changes have only a marginal effect on the measures required for a safe handling and final disposal of the spent fuel. MOX fuel is not further dealt with here.

3.5.4 Accuracy of the calculation programs

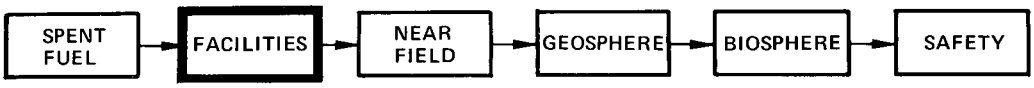
The activity content of the spent fuel has been calculated by means of the ORIGEN 2 computer program. The ORIGEN program contains a simplified model for neutron flux calculation. The programme was originally developed for calculating fission products generated in the fuel, but has since been modified to permit the calculation of the build-up of heavy nuclides as well.

The concentration of fission products has been overestimated for the important nuclides in earlier versions of the ORIGEN program. The new data library which is now being used gives lower values that agree better with measured values.

In order to estimate the accuracy of the calculation of heavy nuclides, nuclide concentrations calculated with ORIGEN 2 are compared in /3-5/ with results from calculations using the cell code program CASMO /3-7/. For certain nuclides, the discrepancies

are rather great (≃ 30%). However, these discrepancies do not affect the nuclides that are of dominant safety-related importance. CASMO has in turn been compared with measurements, whereby small discrepancies (<10%) have been found /3-7/.

4 FACILITIES AND EQUIPMENT



This chapter describes the facilities required for the treatment and final disposal of spent fuel. The description mainly concerns the function and sizing of the facilities. Characteristics of and quality requirements on the copper canister and the buffer and backfill material are dealt with in chapters 9 and 10. Radiological safety during the operating phase is dealt with in chapter 19.

4.1 GENERAL

The facilities that are required for final storage and associated activities are illustrated schematically in figure 4-1.

The system that is required for transports between the facilities is under development and has been approved in its essential respects by the government authorities for the transport of spent nuclear fuel. The central storage facility for spent nuclear fuel, CLAB, is under construction and is subject to the detailed scrutiny of the authorities. Both of these facilities are therefore dealt with only briefly for the purposes of orientation.

The encapsulation plant and the final repository are dealt with in somewhat greater detail, since they involve activities that have not been pursued before or been subjected to government scrutiny. Handling procedures within these facilities consist in large part of operations that are patterned on similar operations within the engineering industry or the nuclear power industry. The following description is primarily aimed at demonstrating that the fuel can be handled in such a manner that current radiological safety requirements for the environment and personnel are fulfilled.

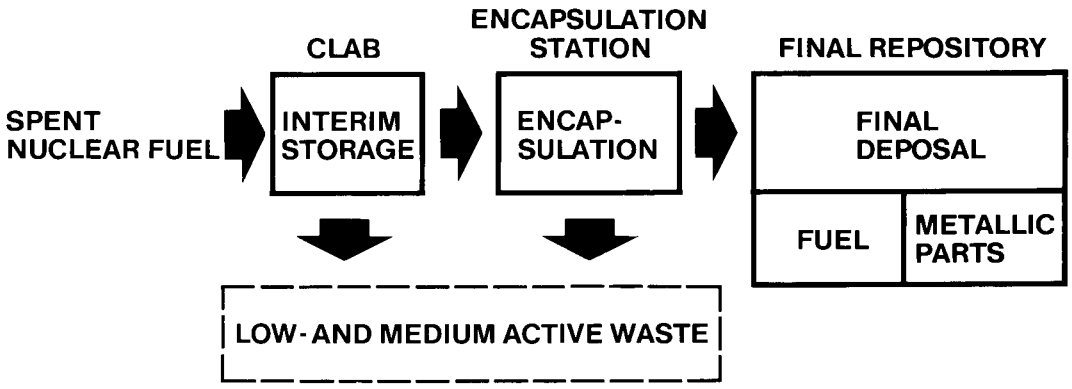


Figure 4-1. Facilities on the final disposal system for spent nuclear fuel.

Radioactive operating waste is generated in connection with water purification and operating maintenance during the operation of both the CLAB and the encapsulation station. This waste is of the same character as the waste that comes from the nuclear power stations (filters, ion exchange resins and solid waste). The handling of these low- and medium-level wastes is not dealt with here. They will be disposed of in the same manner as the operating waste from the nuclear power stations.

The described facilities have been designed and sized on the assumption that all Swedish spent nuclear fuel is to be disposed of by direct deposition. The account differs in this respect from what is postulated in SKBF's report to the National Board for Spent Nuclear Fuel - Plan 82 - where it is assumed that 850 tonnes of fuel will be reprocessed under existing contracts and the remainder disposed of directly.

Reprocessing of a small portion of the total quantity of spent fuel will not affect the fundamental design of the facilities described here, and their capacities (sizing) will only be effected marginally.

4.2 TRANSPORT SYSTEM

A transport system for radioactive waste products has been developed under the auspices of SKBF. The specially designed transport casks are of central importance for safety in connection with the

Figure 4-2. Transport vehicle with transport cask for spent nuclear fuel and M/S Sigyn in the background.

shipment of the spent fuel. Transport casks must meet the requirements of the IAEA's recommendations, which means that they have to pass certain strength tests (dropping onto an unyielding surface), fire tests and submersion in water without losing their integrity.

The transport casks currently used for the Swedish shipments are designated TN 17/Mk 2, see figure 4-2. They weigh about 76 metric tonnes empty and can hold a fuel quantity of about 3 tonnes of uranium, either 17 BWR assemblies or 6 PWR assemblies. They are used for dry transport. The interior of the cask is filled with nitrogen gas and kept under a partial vacuum (about 0.5 bar). The casks have been approved for use by the Swedish Nuclear Power Inspectorate, SKI /4-1/.

For the purpose of transporting spent fuel from the nuclear power plants to the CLAB and from the nuclear power plants to the French reprocessing plant in La Hague, a Swedish transport system was taken into service at the turn of the year 1982/83. It consists of, besides the transport casks, a specially built ship and a terminal transport vehicle.

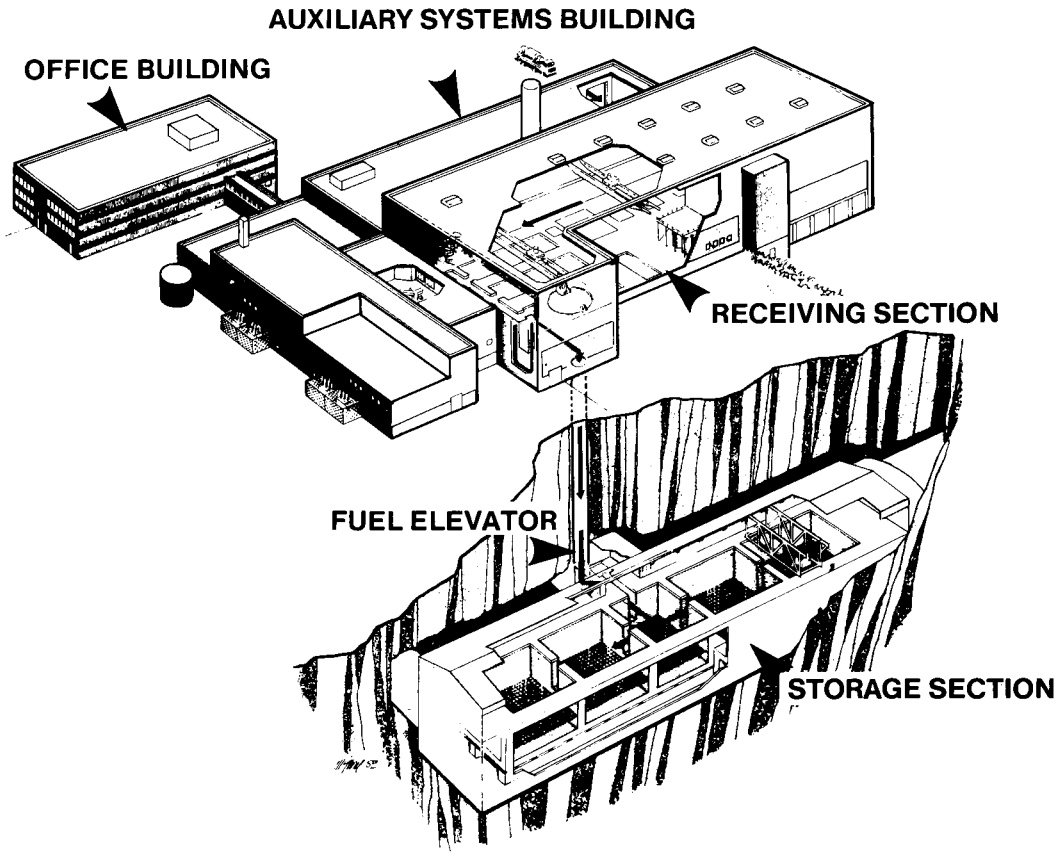


Figure 4-3. Perspective drawing of the central storage facility for spent fuel (CLAB).

The ship is 90 m long and 18 m wide. The transport casks can either be driven on board via an aft ramp into the cargo hold or lifted on board by a crane through the cargo hatches. The casks in their transport frames are placed in fixed positions and lashed to the cargo deck. The ship can take a maximum of 10 transport casks.

For safe operation, the ship has a double hull and double bottom . It has twin engines, located in the side spaces on either side of the hold, twin screws and rudders, plus a bow thruster. The safety systems on board include extensive navigation and communication systems, equipment for measurement of radiation with alarms to the bridge, fire alarm and fire extinguishing systems and equipment to facilitate location and salvage if the ship should sink.

4.3 CENTRAL STORAGE FACILITY FOR SPENT FUEL, CLAB

The central storage facility for spent fuel, CLAB, is situated on the Simpevarp peninsula adjacent to the Oskarshamn Nuclear Power

Station. The construction work began in May of 1980. The plant is scheduled to be ready to receive spent nuclear fuel in January of 1985. The facility is scheduled to attain full receiving capacity during the second half of 1985.

The licence-holder and owner of the CLAB is SKBF. Under contract, OKG is responsible for the construction and later also the operation of the CLAB.

The CLAB shall receive spent fuel from 12 Swedish nuclear power reactors for intermediate storage for about 40 years pending final storage of the fuel. The facility that is currently under construction has a storage capacity of 3 000 tonnes of fuel (uranium). The site is prepared for expansion with additional storage capacity.

The facility - see fig. 4-3 and /4-2/ - consists above ground of a receiving building with pools, an auxiliary systems building and an electrical building with control room, and underground of a rock cavern with storage pools. Two shafts connect the storage section with the surface section, one for vertical transport of storage racks with fuel assemblies, and one for personnel transport, water, electricity and ventilation. A rock tunnel also leads from the storage section to the surface.

The spent fuel arrives at the CLAB in transport casks of the type described in section 4.2. The transport vehicle with the cask is driven into an air lock in the receiving building. The cask is then placed in a cooling cell and connected to a water system for cooling and decontamination. From there, the cask is moved to one of the two receiving pools, where the lid is removed and replaced with a connection fitting that seals against a subfloor in the pool. When a tight connection has been obtained, the inner lid is removed. With this design, the water inside the cask never comes into contact with the outside of the cask, only with the part of the pool where the fuel assemblies are lifted out of the cask. When unloaded, the fuel is placed in portable fuel racks that hold 16 BWR or 5 PWR assemblies. The racks with fuel are then moved to their positions in one of the four storage pools underground. This vertical transfer takes place in a water-filled elevator cage, to a central connection pool in the storage section.

The storage pools contain about 3 000 m^3 of water each. The water is cleaned and cooled continuously. Each pool holds 750 tonnes of fuel. The racks are moved to their storage positions by a handling crane.

4.4 ENCAPSULATION STATION

4.4.1 General

Before final disposal, the fuel will be enclosed in a copper canister. This taken place in the encapsulation station /4-3/.

The spent fuel arrives at the encapsulation station in transport casks as described in section 4.2.

The facility has a capacity for the manufacture of five canisters per week, each containing fuel equivalent to approximately 1.4 tonnes of uranium. This provides good margin to the rate of deposition that corresponds to the annual flow of fuel from 12 reactors (approx. 200 tonnes or 140 canisters per year). Some of the BWR fuel is provided with boxes (fuel channels) and some of the PWR fuel has boron glass rods. The boxes and boron glass rods are separated from the fuel assemblies and cast in concrete moulds.

4.4.2 The encapsulation process

Two alternative methods have been studied for fabrication of the copper canisters. In the one method, the fuel assemblies are placed in a prefabricated copper canister and the cavities filled with molten lead, after which a tight-fitting lid is welded on, see figure 4-4. In the other method, the cavities are filled with copper powder and a lid is placed on, after which the package is treated in a furnace for hot isostatic pressing (HIP), where the entire package is pressed together into a solid body, see figs. 4-5. Both methods are feasible, and there is not enough evidence existing today to indicate which is preferable. Both methods are therefore described below.

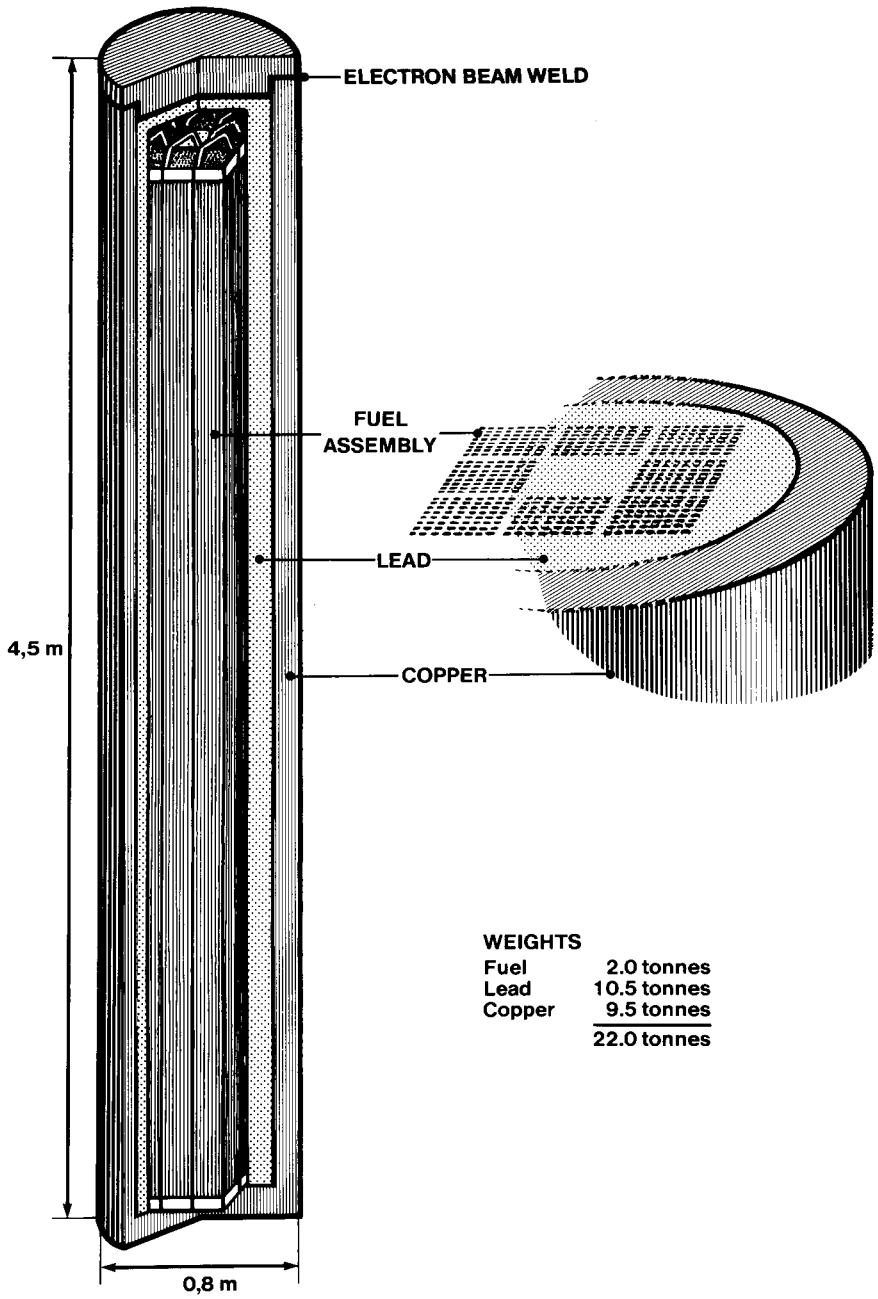


Figure 4-4. Welded copper canister.

In both alternatives, the canisters are fabricated to external dimensions of 0.8 m in diameter and 4.5 m in length and hold 8 BWR fuel bundles in the welded and lead-filled canister alternative and 9 in the pressed canister alternative, or in both alternatives 4 BWR fuel bundles + 2 PWR bundles. The reason for the difference with only BWR fuel is that a free central space in the lead-filled canister may be required for lead filling.

In order to make sure that the temperature will not rise too high in any part of the repository (cf. section 4.5.4), some control of the supply of fuel to the canisters is required so that a mixture

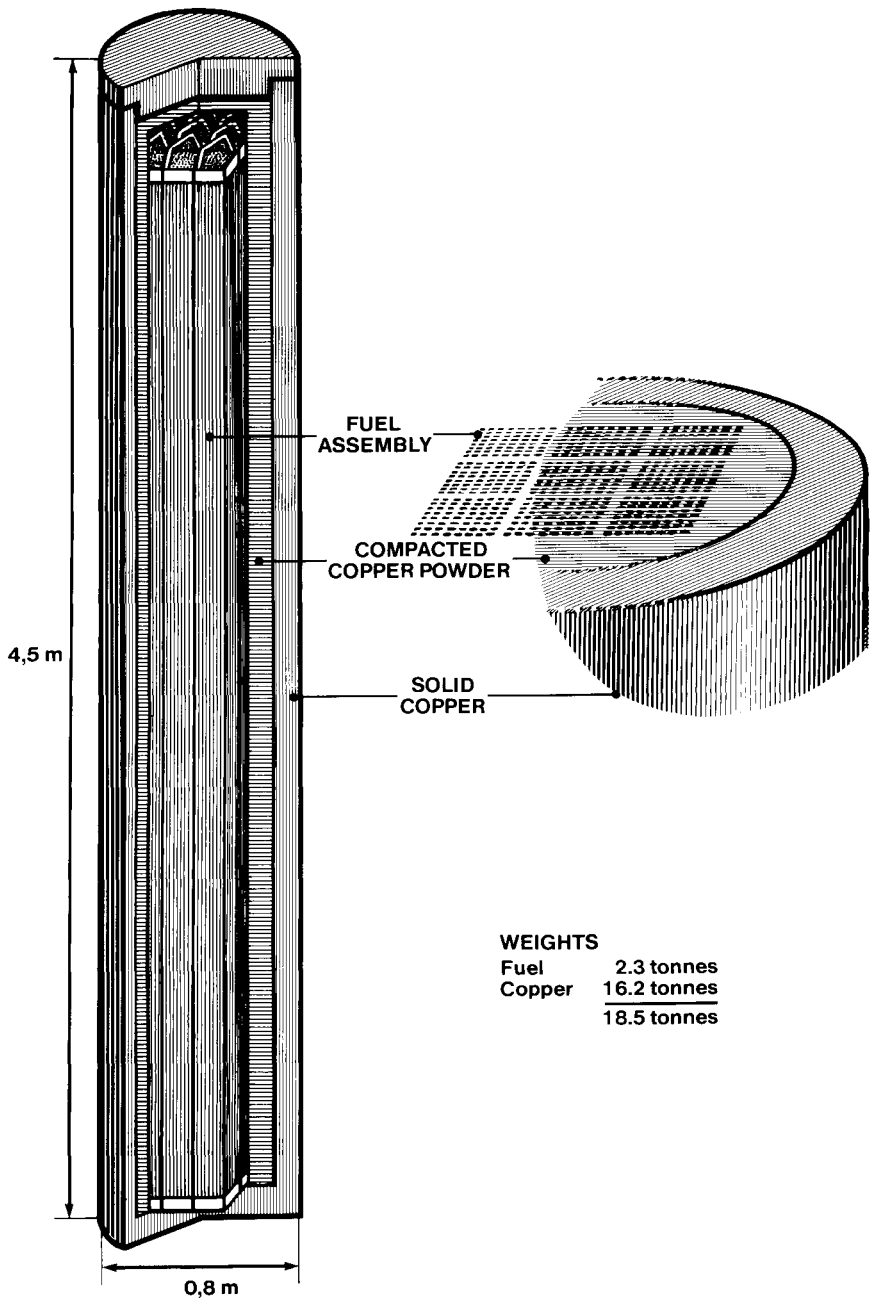


Figure 4-5. Copper canister produced by hot isostatic pressing.

of fuel with different degrees of burnup and/or ages is obtained in the individual canisters. With the average burnup that is expected, a canister can be filled with fuel equivalent to up to 1.4 tonnes of uranium, which is equivalent to 8 BWR assemblies or 2 BWR + 2 PWR assemblies. In the design shown in figures 4-4 and 4-5, the copper canister has a wall thickness of 10 cm. The thickness required from the viewpoint of corrosion is discussed in chapter 10.

Mechanically damaged fuel is placed in special copper containers, which are encapsulated together with the rest of the fuel. Experience has shown that such damages are of very limited extent.

Canister with welded lid

The handling sequence in the encapsulation station is illustrated in fig. 4-6.

In the encapsulation station's receiving section, the fuel transport cask is placed in an unloading position. The fuel assemblies are lifted out of the cask and placed in a rack for transfer to the canister or to a buffer store in a storage pool. The receiving section has basically the same design as in the CLAB. In connection with receiving, the BWR assemblies' boxes and the PWR assemblies' boron glass rods are removed. When a rack has been filled with fuel, it is transferred to the encapsulation section. Up to this point, the fuel has been handled and stored under water with sufficient water coverage to provide adequate radiation shielding. From now on, the fuel is handled in air in cells via remote control, whereby the personnel are protected against radiation by the concrete walls of the cells.

The fuel assemblies are allowed to dry in air and placed in a prefabricated copper canister standing in a transport wagon. The canister is then transferred to a casting cell, where it is filled with molten lead. The technique for lead filling has been verified through practical trials /4-3/. The filled canister is then transferred via a cooling cell to a welding cell. In the welding cell, the canister is provided with a lid, which is welded on airtightly with an electron beam weld. Extensive tests have been carried out to verify the welding method /4-3/. The weld is checked by means of ultrasound. The finished canister is then transferred to a wagon for transport down to the final repository.

Canister produced by hot isostatic pressing

The handling sequence up to the point where the prefabricated canister with fuel is transported into the cell where the canister is to be sealed is the same as in the welded canister alternative. In the cell, the canister is filled with copper powder and fitted with an airtight-welded cover plate with evacuation tube for treatment of the copper powder with hydrogen gas in order to reduce its oxygen content. After this treatment, the canister is evacuated and

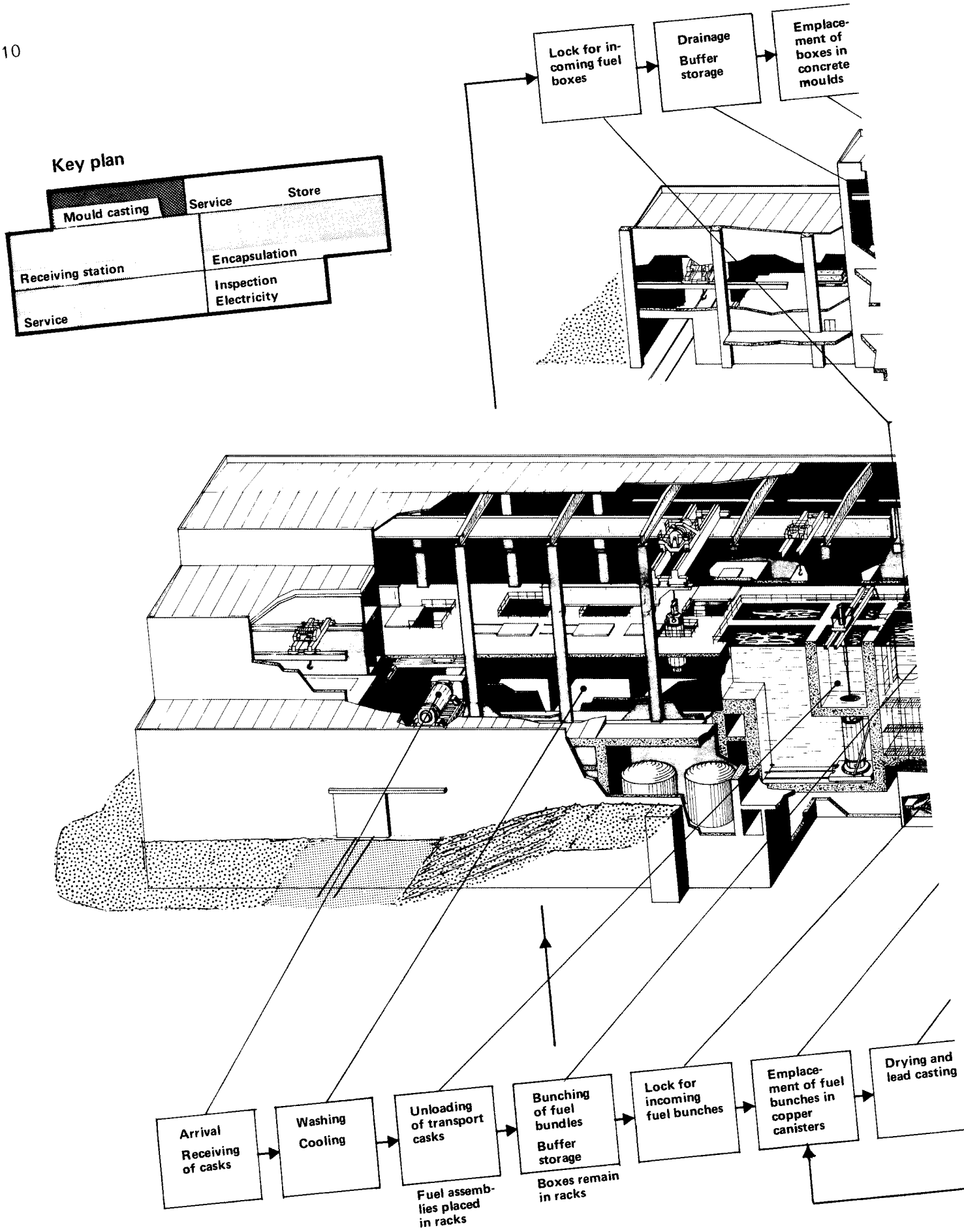
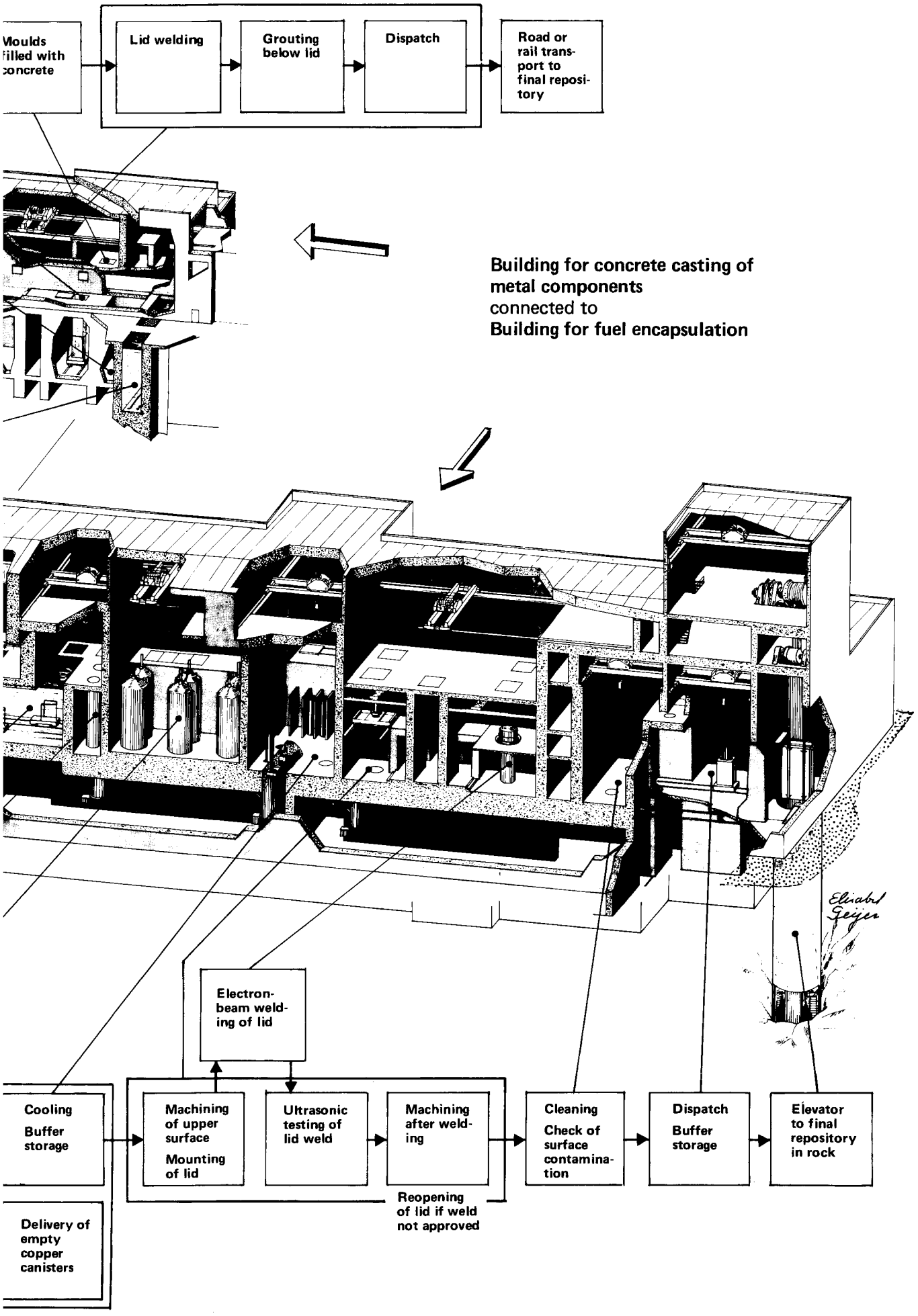


Figure 4-6. Encapsulation station with process scheme.


Moulds filled with concrete
Lid welding
Grouting below lid
Dispatch
Road or rail transport to final repository
Building for concrete casting of metal components
connected to
Building for fuel encapsulation
Elisabet Geijer
Electron-beam welding of lid
Cooling
Buffer storage
Machining of upper surface
Mounting of lid
Ultrasonic testing of lid weld
Machining after welding
Reopening of lid if weld not approved
Cleaning
Check of surface contamination
Dispatch
Buffer storage
Elevator to final repository in rock
Delivery of empty copper canisters

the evacuation tube is sealed and cut. The copper lid is then mounted and another airtight-welded cover plate with evacuation tube is applied. After hydrogen treatment of the joint surface between the canister and the lid and evacuation of the canister, the evacuation tube is sealed and cut. The canister is then transferred to the cell for hot isostatic pressing, where it is pressed isostatically with argon gas at a temperature of 500°C in a pressure of 150 MPa. The copper powder is thereby compacted to solid copper and the lid is joined tightly to the canister. The hot isostatic pressing method is based on extensive experience from the pressing of other materials and from full-scale tests with copper enclosure of simulated fuel /4-3/. After pressing, the joint is tested ultrasonically. The finished canister is then transferred to a wagon for transport to the final repository.

Treatment of certain fuel components

Fuel boxes are placed in the receiving section in racks with 7 x 7 positions for transport to a cell for embedding in concrete moulds. In the cell, see fig. 4-6, the boxes are placed in a concrete mould, see fig. 4-7. The mould is filled with concrete and a concrete lid with edge protectors is welded on. Boron glass rod bundles are placed in certain moulds, where the inner box positions (3 x 3) are left free of boxes. The concrete mould is then transported to a special final repository /4-3/.

4.4.3 Material needs

The total number of welded canisters will be about 4 400, of which about 2 800 will contain only BWR fuel and about 1 600 will contain both BWR and PWR fuel. The total quantity of copper in the repository, assuming a wall thickness of 10 cm on the canisters, will then be about 42 000 tonnes and the total quantity of lead about 47 000 tonnes. The copper need increases to 72 000 tonnes for isostatically pressed canisters.

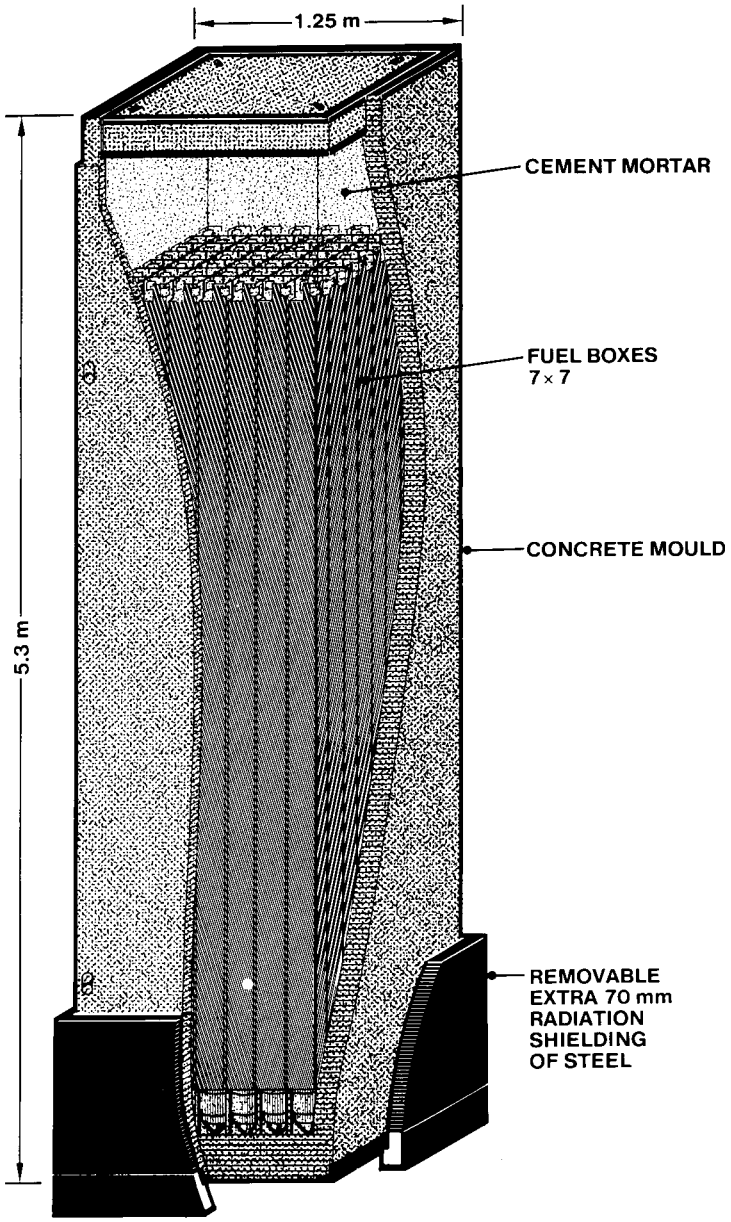


Figure 4-7. Concrete mould with fuel boxes.

The copper and lead needs for different wall thicknesses and methods of fabrication are:

Wall thickness (cm)	Copper need (tonnes) HIP	Welded	Lead need (tonnes) Welded
1	36 000	6 600	47 000
6	55 000	25 000	47 000
10	72 000	42 000	47 000

Encapsulation will be conducted at an even pace for about 30 years. At a canister wall thickness of 10 cm, the annual copper need will be less than 3% of Sweden's copper production in 1982 /4-4/.

4.5 FINAL REPOSITORY FOR SPENT FUEL

4.5.1 General

The encapsulated fuel is intended to be finally disposed of by deposition in a final repository at a depth of about 500 m in a selected rock formation. The final repository will be designed in such a manner that each individual canister will be able to retain its containment function over a very long period of time. The location and geometry of the final repository will be adapted to the geological and geotechnical characteristics of the bedrock. Of particular importance is a low rate of groundwater turnover in the near zone around the canisters.

4.5.2 Deposition method

The fuel canisters are deposited in vertical holes drilled in storage tunnels, see fig. 4-8. In the deposition holes, the canisters are surrounded by buffer material, see fig. 4-9. When deposition has been concluded within a certain part of the repository or in the entire repository, tunnels and shafts are backfilled, see fig. 4-10. The buffer material in the deposition holes and the backfill material in tunnels and shafts are described in greater detail in chapter 9.

4.5.3 Construction method

The final repository is designed as a system of horizontal parallel storage tunnels. Communication tunnels and shafts connect the final repository with the ground surface. Depending on local conditions, above all the location of water-bearing zones, the repository can be made in a single storey or divided into two storeys with a level

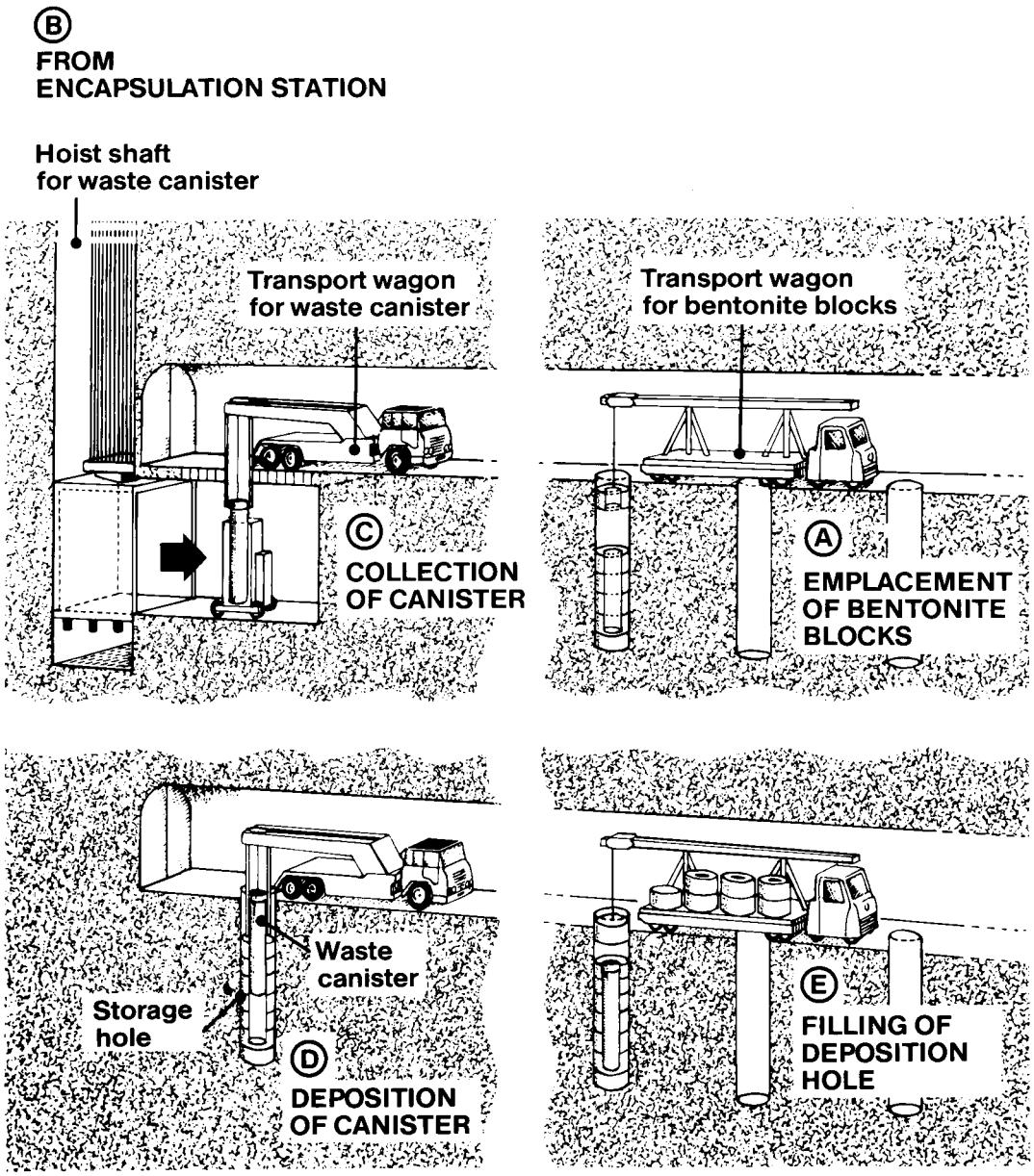


Figure 4-8. Handling scheme for canister deposition in final repository. The handling sequence is A–E.

difference of about 100 m, see fig. 4-11. The technology that has been used for civil engineering works and mines down to a depth of about 1 km in the Swedish bedrock will be employed for excavation of the final repository and drilling of the deposition holes /4-5/. Either careful blasting or full-face driving may be used for the storage tunnels.

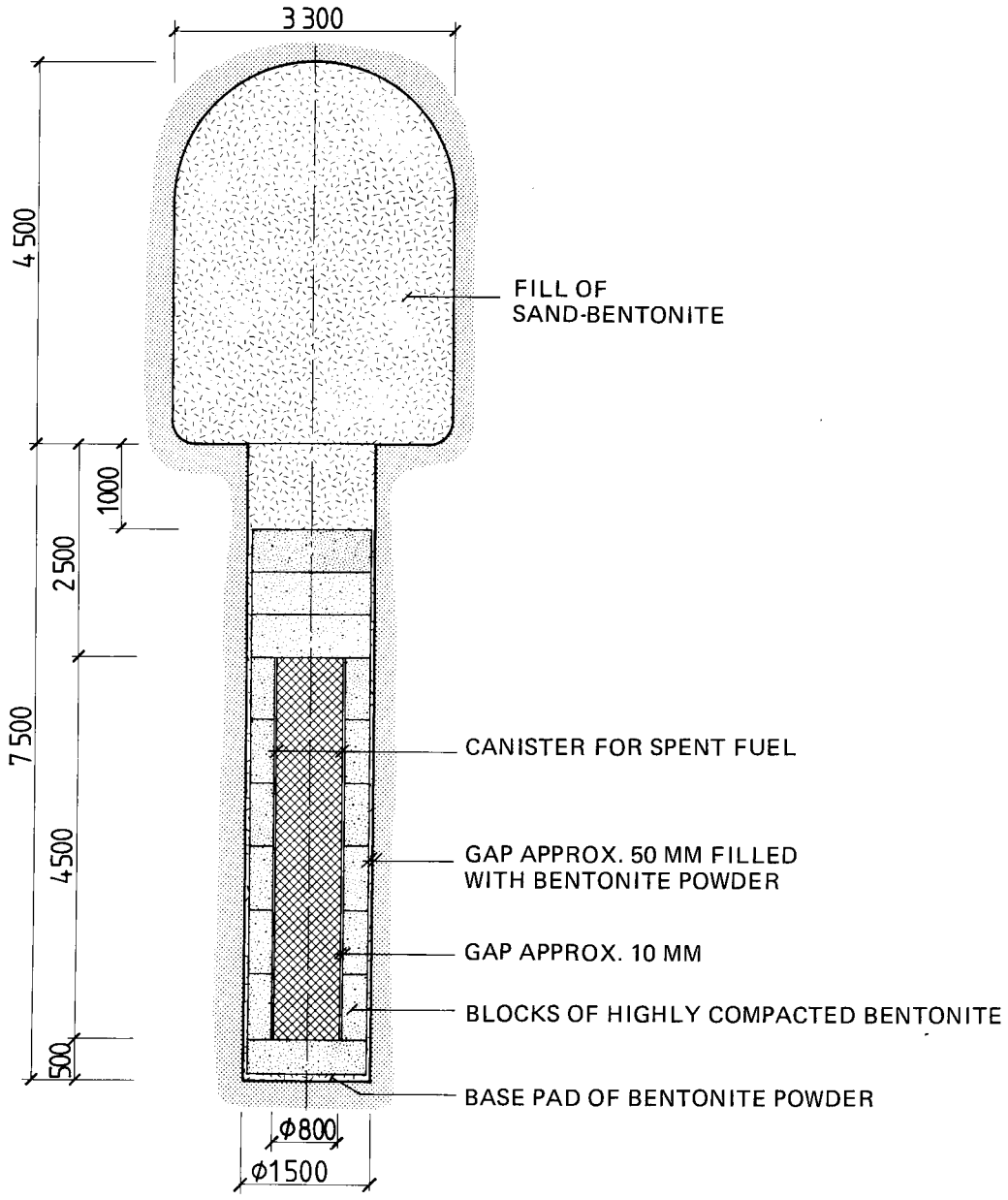


Figure 4-9. Deposition hole with canister and buffer material and with backfill in storage tunnel.

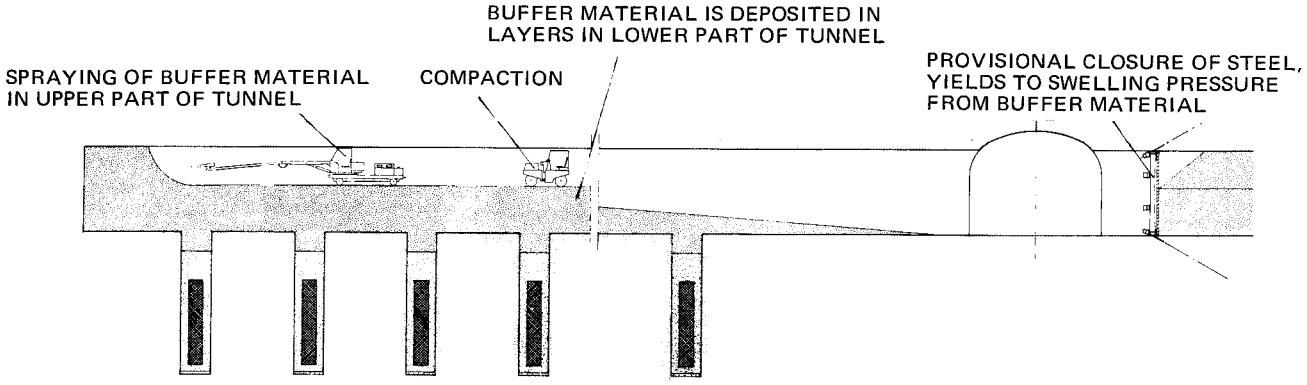


Figure 4-10. When the final repository is sealed, the tunnels are filled with a mixture of quartz sand and bentonite. The lower part of the fill is deposited by tractors and vibrorolled. The upper part of the tunnel is filled by spraying.

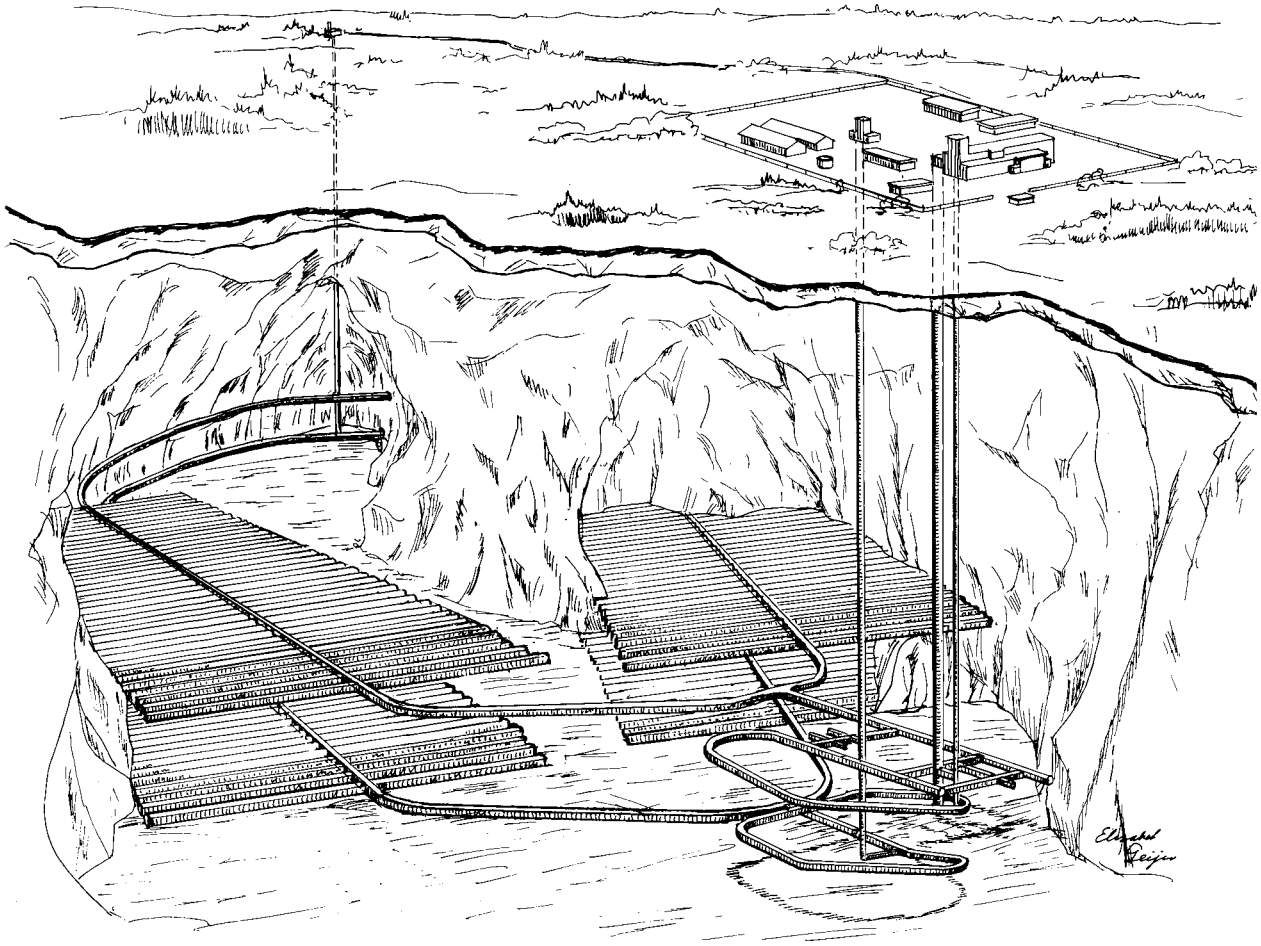


Figure 4-11. Tunnels and shafts in the final repository for spent fuel.

4.5.4 Temperature conditions

The spent fuel in a canister gives off heat, which raises the temperature in the surrounding buffer material and rock in the final repository. A welded canister containing 1.4 tonnes of BWR fuel gives off about 0.8 kW at the time of deposition. Variation of the temperature with time is determined by the heat-conducting properties of the surrounding material, the geometry of the deposition hole and the final repository and the decline of the residual heat flux. With a canister spacing of 6 m and 25 m between the storage tunnels, the maximum temperature in a single-storey repository will be 80°C at the most for the hottest canister, assuming an initial temperature in the rock of 15°C, see fig. 4-12. For a two-storey repository with 100 m of rock between the storeys, the tunnel spacing must be increased to 33 m in order that the maximum temperature will not exceed 80°C if the initial temperature in the

lower storey is 20°C, see fig. 4-13. In this case, two temperature maxima occur. If more fuel or fuel with a higher burnup is placed in the canisters, the higher residual heat must be compensated for by further increasing the spacing between the tunnels or the deposition holes /4-6/.

The temperatures given are calculated assuminq that the air-dry buffer material has a thermal conductivity of 0.75 W/(m · K) and that the rock has a thermal conductivity of 3 W/(m · K). The heat-conducting properties of the rock and the natural starting temperature will vary from place to place depending on geology and climate. However, these variations are not greater than that a final repository can easily be modified to keep the maximum temperature to about 80°C. The need to limit the temperature in the buffer material around the canister is discussed in chapter 9.

4.5.5. Design with respect to geological and geotechnical characteristics

Experience from rock excavation work and knowledge of the rock stress conditions in the Swedish bedrock show that no particular difficulty should be involved in constructing tunnels of the limited dimensions involved here as far as the stability of the rock is concerned /4-7/.

The heating of the rock mass brought about by the residual heat in the fuel will alter the state of stress in the rock around the final repository. However, the natural stress situation that normally exists in Swedish bedrock contains adequate security against rupture within the temperature interval involved here /4-8/.

As construction work on the final repository proceeds, increasingly detailed information will be gathered concerning the rock's geological and geotechnical characteristics by the drilling of test holes and geophysical measurements. The method for geophysical cross-hole measurement is currently under development within the Stripa project. The results of these investigations, which are primarily aimed at mapping fracture and crush zones, will form the basis for the detailed design of the final repository. The repository will be designed so that the deposition holes will be located at a

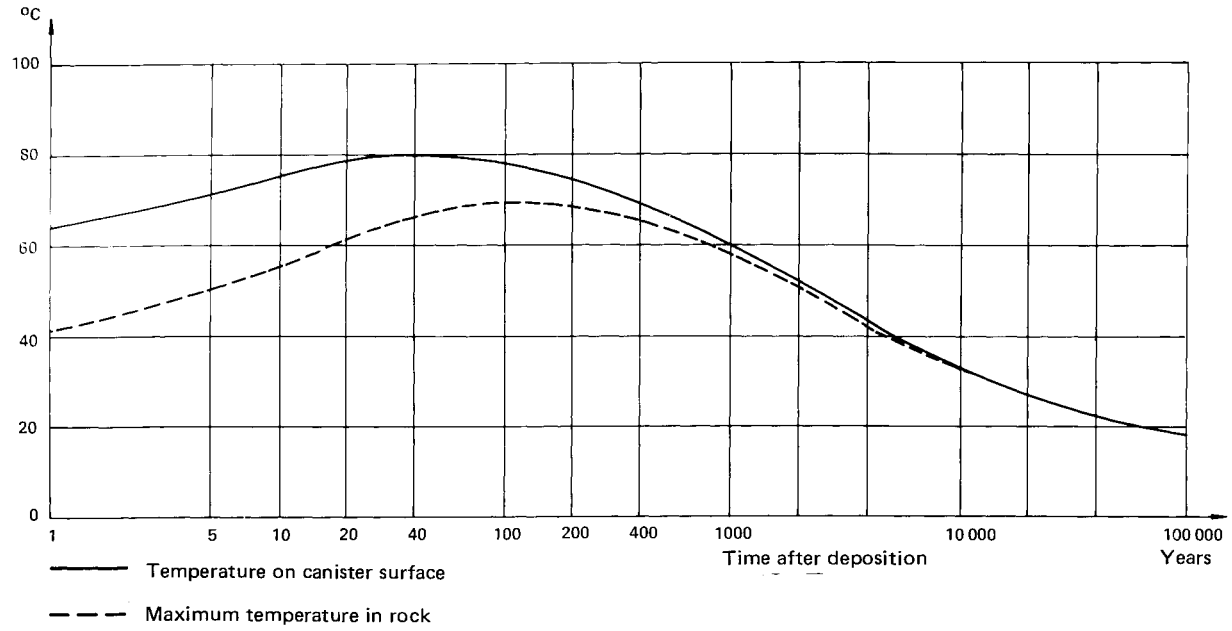


Figure 4-12. The temperature in the centre of a single-storey final repository.

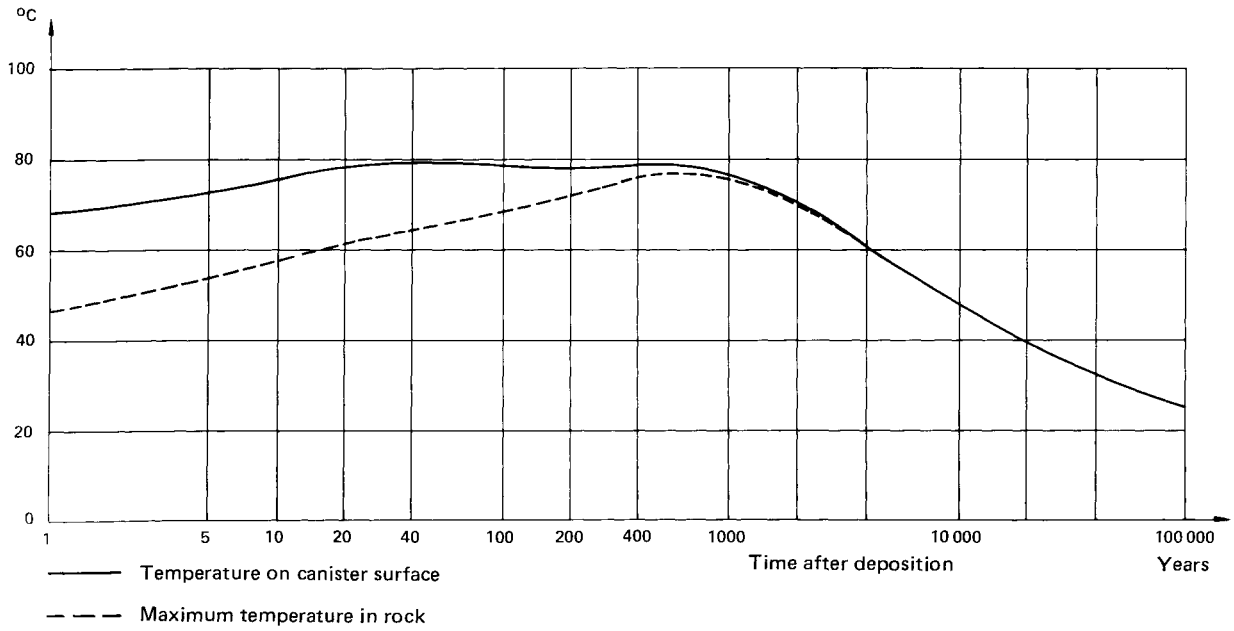


Figure 4-13. The temperature in the centre of a two-storey final repository.

distance of at least 100 m from the nearest known zones with appreciably elevated water flow or where the rock is so crushed and weakened that the possibility of future rock movements cannot be excluded. When the tunnels are blasted out, a core borehole will be drilled in the position for each deposition hole. Conductivity in the core boreholes will be measured by means of water pressure tests and observation of water seepage. On the basis of these measurements and the geological documentation, a decision will be made as to whether the site is to be used for the deposition of a waste canister or not. An example of how a final repository can be adapted to geological conditions is shown in fig. 4-14.

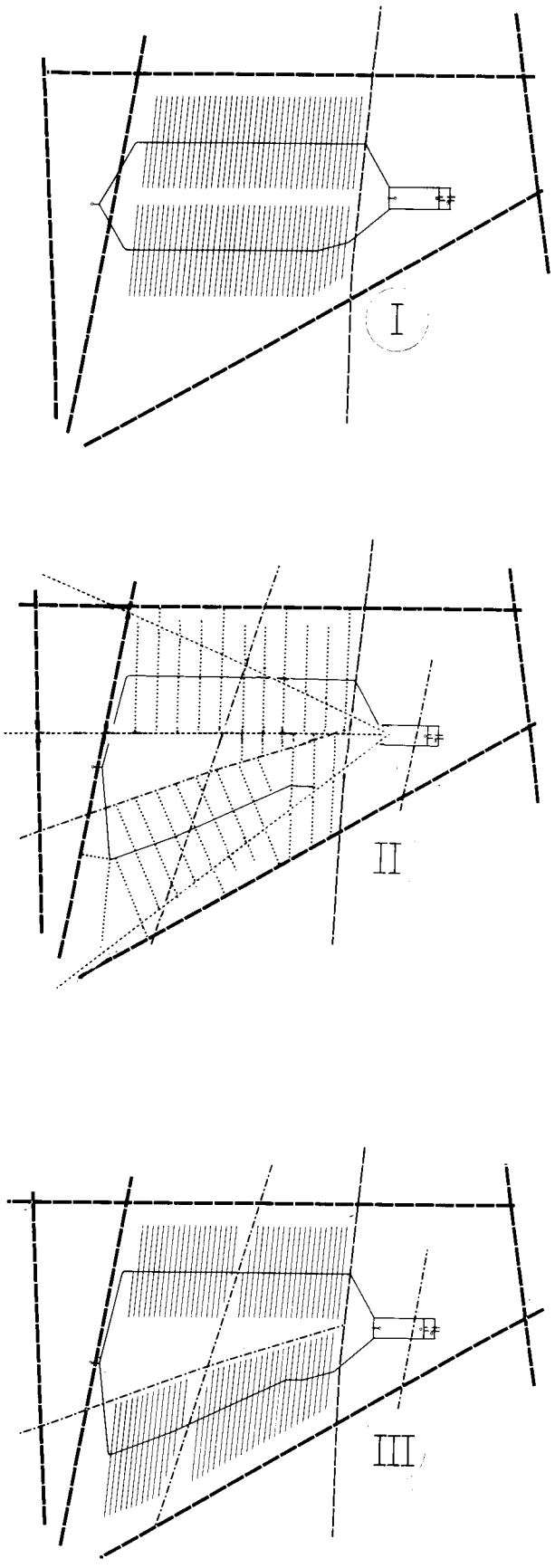


Figure 4-14. Example of adaptation of final repository to fracture geometry of rock.

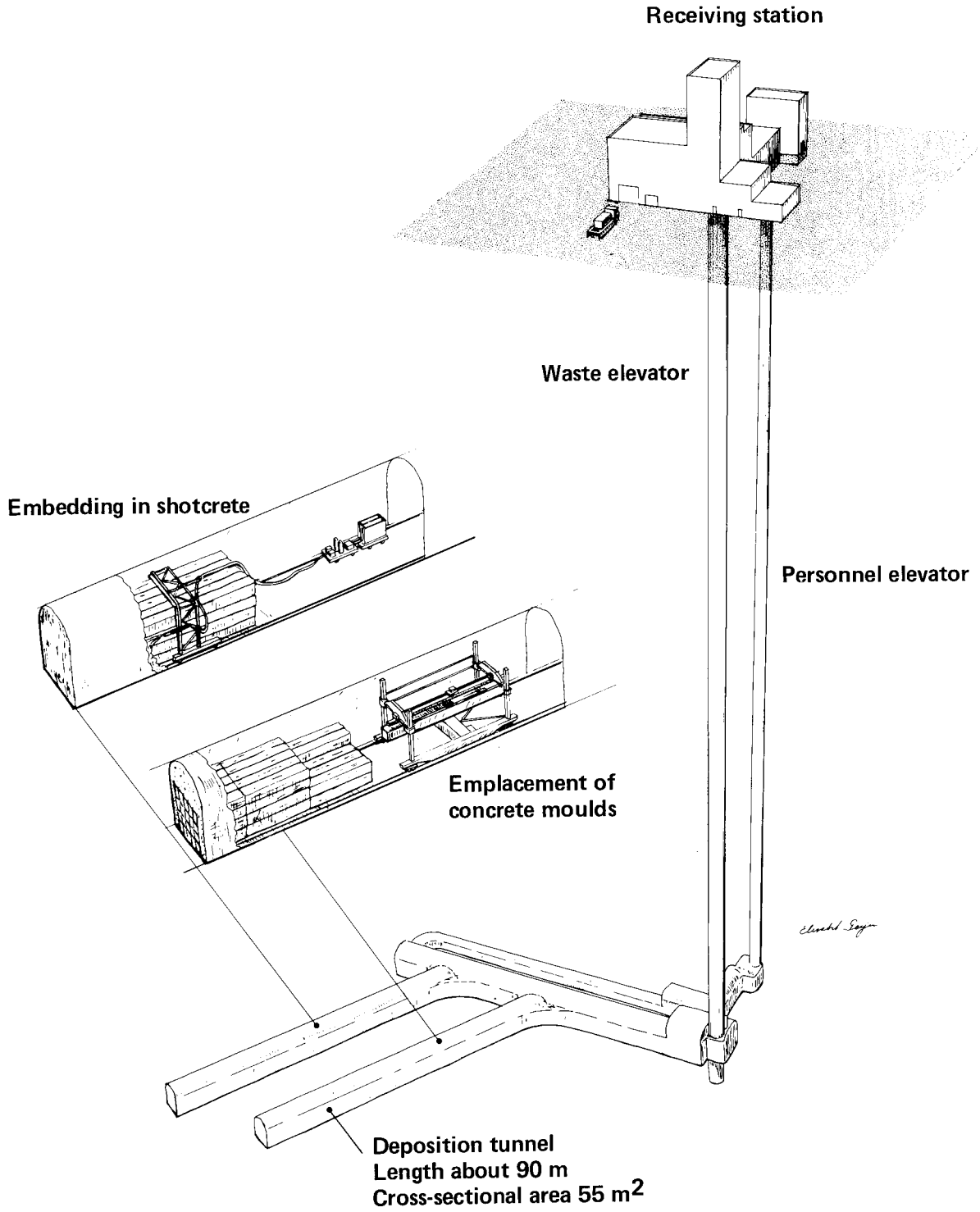


Figure 4-15. Final repository for fuel boxes etc.

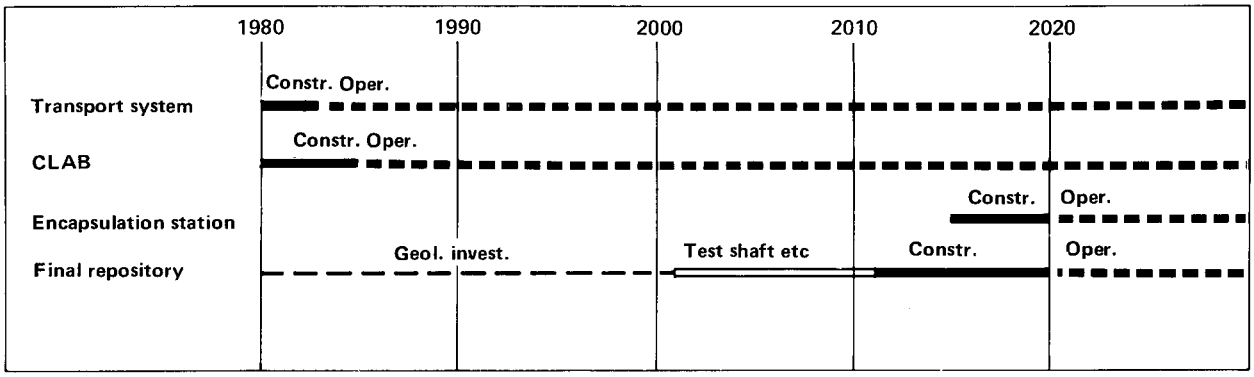


Figure 4-16. General timetable for facilities.

Storage tunnels with a total length of about 30 km will hold 4 400 canisters. The tunnel system requires a total area of 0.7 km^2 in a single-storey repository. For reasons of heat dissipation, a two-storey repository requires a greater distance between the tunnels and the area will be 2 x 0.5 km^2 if all storage positions can be used.

The total excavated volume of a final repository is about 600 000 m^3 of solid rock. Approximately 250 000 tonnes of bentonite will be required for the backfill.

4.5.6 Final repository for fuel boxes etc.

Concrete moulds with fuel boxes and boron glass rod bundles are transported via an elevator shaft and tunnel to the storage site at a depth of about 300 m, where they are stacked and backfilled with shotcrete, see fig. 4-15. The tunnels have a cross-sectional area of about 55 m^2, and a tunnel length of about 160 m is required for all moulds /4-3/.

Since large quantities of concrete can cause the pH of the groundwater to increase downstream of the mould repository (cf. chap. 7), the repository is intended to be located about a kilometer or so away from the repository for spent fuel. Its location will be chosen so that the groundwater flow does not affect the spent fuel repository.

4.6 TIMETABLE

The timetables for the facilities described above have been determined by the need for additional storage capacity for spent fuel in Sweden, by transport needs stemming from existing reprocessing contracts and by the 40-year intermediate storage period prior to deposition that has been chosen to limit radioactivity and residual heat in the final repository.

A general timetable is shown in fig. 4-16.

REFERENCES

CHAPTER 1

1-1 Swedish Parliament Publications Prop 1979/80:170; NU 1979/80:70; rskr 1979/80:410

1-2 Lagstiftning på kärnenergiområdet - Förslag till ny lag om kärnteknisk verksamhet,
(Legislation in the field of nuclear energy - Proposal for new law governing nuclear activities,) SOU 1983:12.

1-3 Plan för kärnkraftens radioaktiva restprodukter, Plan 82 del 1 och 2, Svensk Kärnbränsleförsjörjning AB, juni 1982.
(Radioactive waste management plan, Plan 82 parts 1 and 2, Swedish Nuclear Fuel Supply Co., June 1982.)

CHAPTER 2

2-1 GELIN R
Final Disposal of High-Level Waste and Spent Nuclear Fuel-Foreign Activities
Studsvik Energiteknik AB
KBS TR 83-42, May 1983

CHAPTER 3

3-1 Radioactive Waste Management Plan, Plan 82 parts 1 and 2
KBS TR 82-09, June 1982

3-2 LÖNNERBERG B, LARKER H, AGESKOG L
Encapsulation and Handling of Spent Nuclear Fuel for Final Disposal.
1. Welded Copper Canisters
2. Pressed Copper Canisters (HIPOW)
3. BWR Channels in Concrete
ASEA-ATOM, ASEA, VBB
KBS TR 83-20, May 1983

3-3 NYLUND O, FREDIN B, JOHANSSON A
SVEA - A New BWR Fuel Assembly Concept for Improved Fuel Utilization, ANS Topical Meeting on the Technical Bases for Nuclear Fuel Cycle Policy, 1981

3-4 CROFF A G
ORIGEN 2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, ORNL-5621

3-5 EDLUND O
Calculation of Activity Content and Related Properties in PWR and BWR Fuel Using ORIGEN 2
Studsvik Energiteknik AB
KBS TR 83-12, 1983-03-07

3-6 FORSSTRÖM H
Plutoniumanvändning i svenska reaktorer
("Plutonium re-use in Swedish reactors)
SKBF
The back end of the nuclear fuel cycle, September 1982

3-7 EDENIUS M, HELLSTRAND E, JOHANSSON E
CASMO, Benchmark Report, Studsvik /RF-78/6293

CHAPTER 4

4-1 Certificate S/40/B(U)F 1982-07-06

4-2 Centralt Lager för Använt Bränsle (CLAB). Preliminär Säkerhetsrapport (PSR).
("Central storage facility for spent fuel (CLAB). Preliminary safety report (PSR)")
SKBF October 1978

4-3 LÖNNERBERG B, LARKER H, AGESKOG L
Encapsulation and Handling of Spent Nuclear Fuel for Final Disposal
1. Welded Copper Canisters
2. Pressed Copper Canisters (HIPOW)
3. BWR Channels in Concrete

ASEA-ATOM, ASEA, VBB
KBS TR 83-20, May 1983

4-4 World Metal Statistics, April 1983 (Volume 36, Number 4) World Bureau of Metal Statistics

4-5 NORD G
Drilling holes in Rock for Final Storage of Spent Nuclear Fuel Swedish
Detonic Research Foundation
KBS TR 80-12, September 1980

4-6 TARANDI T
Calculated Temperature Field in and around a Repository for Spent Nuclear Fuel
VBB
KBS TR 83-22, April 1983

4-7 BERGMAN S G A
Spänningsmätningar i Skandinavisk berggrund - förutsättningar, resultat och tolkning
("Stress measurements in Scandinavian bedrock - premises, results and interpretation")
KBS TR 64, Nov 1977

4-8 STEPHANSSON O, LEIJON B
Bergmekanisk bedömning av temperaturbelastning vid slutförvaring av radioaktivt avfall i berg.
("Rock mechanical evaluation of temperature load in connection with final storage of radioactive waste in rock.")
University of Luleå
KBS TR 79-03, 1979-01-10