SKB

RD&D-PROGRAMME 92

Treatment and final disposal of nuclear waste

Programme for research, development, demonstration and other measures

September 1992

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FOREWORD

The Act on Nuclear Activities (SFS 1984:3) prescribes in Section 12 that a programme shall be prepared for the comprehensive research and development and other measures that are required to safely handle and finally dispose of the radioactive waste from the nuclear power plants. The responsibility lies primarily with the owners of the nuclear power plants. These owners have commissioned SKB to prepare the prescribed programme. According to Section 25 of the Ordinance on Nuclear Activities (SFS 1984:14), this programme shall be submitted to the National Board for Spent Nuclear Fuel in the month of September every third year.

The purpose of this third programme is to fulfil the above obligations.

The programme is presented in one main report and three background reports. The programme is called RD&D-Programme 92, where RD&D stands for Research, Development and Demonstration. The reason for the change of name compared to previous R&D programmes is to underscore the fact that, starting with the work at the Äspö Hard Rock Laboratory and the plans presented in this programme, the emphasis of the programme has been shifted towards demonstrating different parts of the selected disposal system. The main report describes the programme in its entirety, while the background reports provide more detailed presentations of the siting of a deep repository, the Äspö Hard Rock Laboratory and the various R&D projects 1993–1998. To obtain a full picture of the present-day state of knowledge, the relevant background report must also be consulted.

Stockholm, September 1992

SWEDISH NUCLEAR FUEL AND WASTE MANAGEMENT COMPANY

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1 INTRODUCTORY OVERVIEW

1.1 REQUIREMENTS AND GOALS

The goal of radioactive waste management in Sweden is to dispose of all radioactive waste products generated at the Swedish nuclear power plants in a safe manner. Furthermore, all other radioactive waste that arises in Sweden shall be safely disposed of.

The Act on Nuclear Activities /1-1/ requires that the owners of the Swedish nuclear power plants adopt the measures that are needed to achieve this goal. The owners of the Swedish nuclear power plants have commissioned the Swedish Nuclear Fuel and Waste Management Company (SKB) to implement the measures that are needed. The provisions of the Act on Nuclear Activities imply that the industry must actively work to ensure that the necessary measures are adopted.

1.2 THE SITUATION TODAY

A safe handling and final disposal of the waste from nuclear power requires goal-oriented research, development and planning. Furthermore, the required facilities and systems must be built and put into operation. Figure 1-1 provides an overview of the different parts of the Swedish nuclear waste management system. They are described in detail in the annual report of the costs of waste management – PLAN 92 /1-2/.

Essential parts of the waste management system are already in operation. These include the central interim storage facility for spent nuclear fuel, CLAB, the final repository for radioactive operational waste, SFR, and the transportation system. The parts that have not yet been finalized are an encapsulation plant for spent nuclear fuel etc. and a final repository for long-lived waste, particularly spent nuclear fuel.

The existing system has been developed and built up systematically on the basis of proposals put forth by, among others, the Aka Committee /1-3/ in the mid-1970s, and the research and development work initiated with the KBS Project during the latter half of the 1970s.

Proposals and alternative options have since been reviewed and studied by both regulatory authorities and the nuclear power industry in extensive R&D projects during the 1980s. This means that the important issues relating to encapsulation and final disposal of spent nuclear fuel in Swedish bedrock have been thoroughly elucidated.

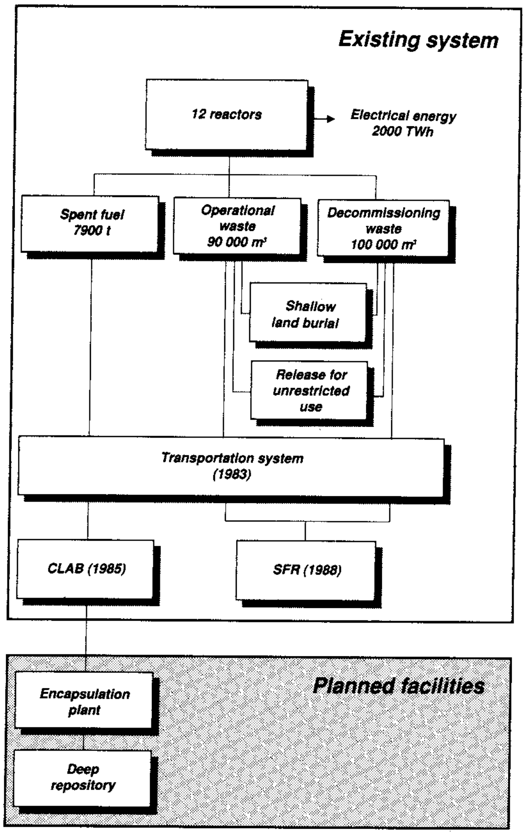


Figure 1-1. The Swedish waste management system.

Similar parallel studies have been and are being carried out in most countries with significant nuclear power programmes. Owing to the stringent requirements introduced in the so-called Stipulation Act in 1977, the work in Sweden got under way with great determination and ample resources. This has given the Swedish activities an internationally recognized position and led to broad international cooperation. The interest from other countries is due not to the fact that conditions for final disposal are better in Sweden, but to the systematic way the work has been done and reported on, and the quality of the facilities that have already been built.

The work that has been carried out during a period of about fifteen years in Sweden, and equivalent work in other countries, has led to broad agreement among the international experts that methods exist for implementing final disposal of high-level waste and spent nuclear fuel and that methods also exist for demonstrating the long-term safety of such disposal. Clear expressions of this agreement include, for example, the approval of the KBS 3 report /1-4/ in Sweden and of similar studies in Finland /1-5/ and Switzerland /1-6/. The "collective opinions" expressed by international expert groups within the OECD/NEA, the IAEA and the EC are also worth mentioning /1-7,8/. An important conclusion in the most recent of these collective opinions is that further efforts should be focused on gathering and evaluation of data from proposed final repository sites.

The Swedish studies (as well as the Finnish, Swiss, Canadian and French studies /1-5,6,9,10/) have shown that the crystalline bedrock generally possesses such properties at depth that it is well-suited to host a final repository. Changes in the environment take place extremely slowly in deep-lying geological formations. This means that the safe conditions we can verify today on the basis of extensive analyses and measurements can be expected to persist for thousands and tens of thousands of years. After having examined safety, technical feasibility and other aspects for a number of different alternatives, work in Sweden has now reached a point where it has to be concentrated to a main line.

The principle of final disposal is that it shall be arranged so that the waste is kept isolated in a safe manner during the time that the waste has a higher radiotoxicity than is otherwise found in nature. Spent nuclear fuel contains large quantities of radioactive materials. Most of these will have decayed after a few hundred years. After a thousand years, all that will remain, besides uranium and its daughter products, is a few long-lived radionuclides, such as plutonium, with a very long decay time. After 100,000 years, the radiotoxicity of the fuel will have declined to a level equivalent to that in uranium ores.

To bring about the desired long-term isolation, a final repository for spent fuel is designed according to the multi-barrier principle. The spent fuel consists primarily of uranium dioxide, a ceramic material that has low solubility in groundwater. The most important long-lived radionuclides – which are formed in conjunction with irradiation in the reactor, e.g. plutonium – are embedded in the ceramic material and are likewise low-soluble in water. The fuel is enclosed in a canister with good mechanical strength and made of a material with a long corrosion life. The canisters are placed in specially arranged chambers in the rock and surrounded with a buffer material. The materials in the engineered barriers have documented long-term stability and the repository only affects the natural conditions in the rock slightly. Recently completed safety assessments – in particular SKB 91 /1-11/ – show that excellent conditions exist for designing the near field in the repository so that the radioactive materials are kept isolated for more than one million years. Moreover, the rock has a great capacity to sorb the radionuclides that dominate the radiotoxicity of the fuel and thereby constitutes an additional barrier.

The SKB 91 safety assessment, which SKB carried out during 1989–1992, shows that the requirements on the properties of the bedrock are limited. "...SKB 91 shows that a repository constructed deep down in Swedish crystalline basement rock with engineered barriers possessing long-term stability fulfils the safety requirements proposed by the authorities with ample margin. The safety of such a repository is only slightly dependent on the ability of the surrounding rock to retard and sorb leaking radioactive materials. The primary function of the rock is to provide stable mechanical and chemical conditions over a long period of time so that the long-term performance of the engineered barriers is not jeopardized" /1-11/. The studies and investigations that have been conducted of the bedrock in Sweden during the past 15-year period show that these properties exist at many places and that there are thus many sites possessing the necessary geological and technical prerequisites for constructing a safe repository.

Present-day knowledge is sufficient for selecting a preferred system design, for designating candidate sites for siting a repository, for characterizing these sites and for adapting the repository to local conditions.

1.3 GENERAL PLAN FOR FURTHER WORK

SKB's previous plan for siting and building a repository for spent fuel entailed that after pre-investigations at three sites and detailed characterization at two during the 1990s, a decision would be taken a few years into the 21st century to build a repository for about 8,000 tonnes of fuel at one of the sites. During the circulation of R&D-Programme 89 for comment and review /1-12/, a proposal from SKN was discussed to the effect that a demonstration-scale repository should first be built, for example 5–10% of the full-scale repository. In its decision concerning R&D-Programme 89 /1-13/, the Government asserted "...that one of the premises for further research and development activities should be that a final repository for nuclear waste and spent nuclear fuel shall be able to be put into operation gradually with checkpoints and opportunities for adjustments. In the next R&D programme under the Act on Nuclear Activities, SKB should explore the possibilities of including a demonstration-scale final repository as a step in the work of designing a final repository."

In the planning of the present RD&D programme, SKB considered this possibility of building and commissioning the repository in stages. The result is that SKB finds that a demonstration phase has considerable advantages. The present programme thereby calls for completion of the research, development and demonstration work by first building the final repository as a deep repository for demonstration deposition of spent nuclear fuel. When the demonstration deposition has been completed, the results will be evaluated before a decision is made whether or not to expand the facility to accommodate all the waste. This plan also makes it possible to consider whether the deposited waste should be retrieved for alternative treatment. The latter option means that it must be possible to retrieve deposited fuel during the period the facility is being operated for demonstration purposes. The siting process is only affected to a limited extent by whether the planning applies to a deep repository for demonstration deposition or to a complete deep repository. The requirements on background information from SKB in the different phases (pre-investigation, detailed investigation, construction of repository) are essentially the same.

The most important reason for SKB's plan to build a repository for demonstration deposition is that this makes it possible to demonstrate the following, without the necessity of making what are sometimes described and perceived as definite decisions:

- the siting process with all its technical, administrative and political decisions,
- the process and the methods for step-by-step investigation and characterization of the deep repository site,
- system design and construction,
- full-scale encapsulation of spent fuel,
- the handling chain of spent fuel from CLAB to deposition in the repository,
- the operation of a deep repository,
- the licensing of handling, encapsulation and deep disposal, including the assessment of long-term safety,
- (retrievability of the waste packages).

Beyond this it is also possible to study the condition of the barriers a given shorter or longer time after deposition. This is, however, something that preferably can and should be investigated with non-radioactive material in the Äspö Hard Rock Laboratory, which is under construction at Simpevarp approximately 20 km north of Oskarshamn.

The long-term safety of the final repository cannot be demonstrated through field tests. Allowability in this respect must always be based on a technical-scientific assessment of the performance of the repository over a long period of time. However, the background information that is gathered in conjunction with the construction of the deep repository for demonstration deposition allows a safety assessment to be performed based on site-specific "full-scale" data.

The reason SKB is planning a demonstration deposition is not doubts as to the feasibility and safety of the deep disposal scheme. The plan should be viewed as an expression of an awareness of and respect for the fact that the solution of the nuclear waste problem arrived at by the R&D work needs to be demonstrated concretely to concerned people in society far beyond the circle of experts for confidence-building purposes. It is SKB's opinion that a demonstration deposition of spent nuclear fuel with full freedom of choice for the future is a good way to enlist broad support for the method of disposing of the nuclear waste.

The planned demonstration deposition also means that the present-day generation is deciding for a span of time that roughly corresponds to its own active time, leaving it up to the next generation to make its own decision with as much background information as possible.

The work up until all nuclear waste in Sweden has been deposited in a closed deep repository is therefore planned to be carried out in two main phases: Demonstration deposition and final disposal. In all the work extends over a period of more than 60 years. The decision to take the step to final disposal will not be taken until after demonstration deposition has been completed, the results evaluated and other alternatives considered. These decisions lie beyond the year 2010. The plans that are discussed in this programme have to do with the activities that are required to site and construct the facilities that are needed for a demonstration deposition. It is SKB's judgement that the deep repository will later be expanded to full scale. However, it is not meaningful to discuss at this point in time the details of how this will be done. The important task for now is to demonstrate a possible method for long-term safe disposal and to provide future engineers and decision-makers with the best possible background information for their decisions.

Figure 1-2 shows a timeschedule for the facilities that are needed to dispose of the long-lived radioactive waste.

The following additional units will, as is shown by Figure 1-2, be needed for a demonstration deposition of spent nuclear fuel:

- Encapsulation plant for spent fuel, including a buffer store for the encapsulated fuel. The buffer store shall be able to be expanded so that it can be used as an interim storage facility if the demonstration deposition is interrupted and the canisters are retrieved.
- Deep repository for encapsulated spent nuclear fuel.
- Transportation system between CLAB and the encapsulation plant for spent fuel and between the latter and the site of the deep repository.

CLAB and the sea transportation system with associated terminal transportation system are in operation today. Figure 1-3 shows a timeschedule for the encapsulation plant for spent fuel 1993-1998 and Figure 1-4 for the deep repository up to the completion of demonstration deposition.

In principle, the interim storage period of 40 years can be retained for further planning even with the time-

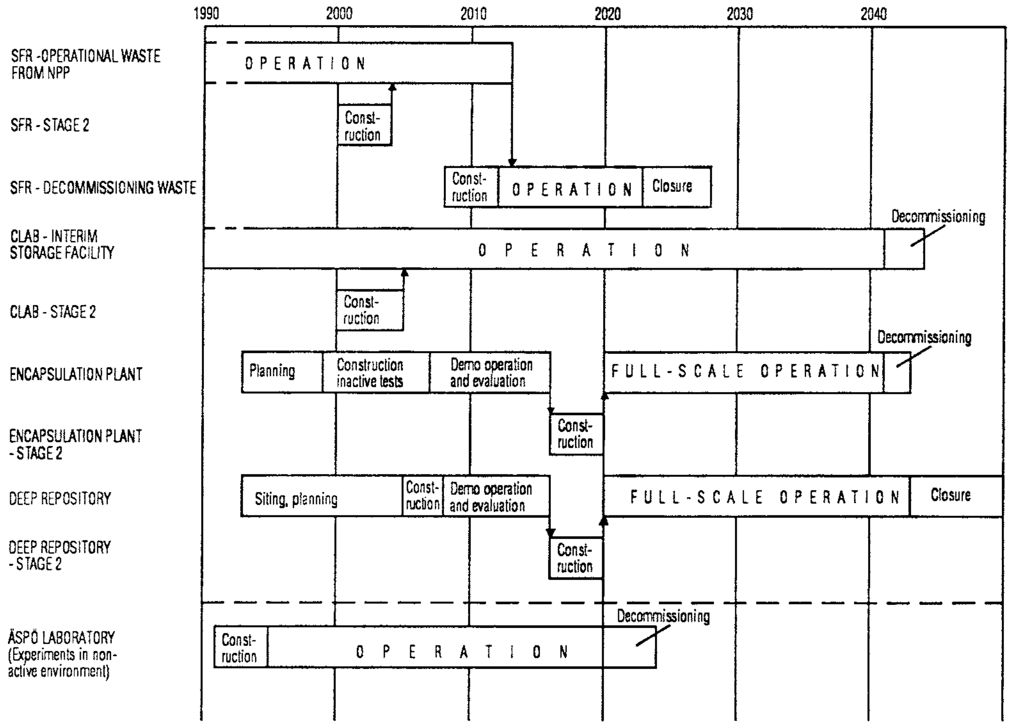


Figure 1-2. Approximate timeschedule – facilities for management of the waste products of nuclear power.

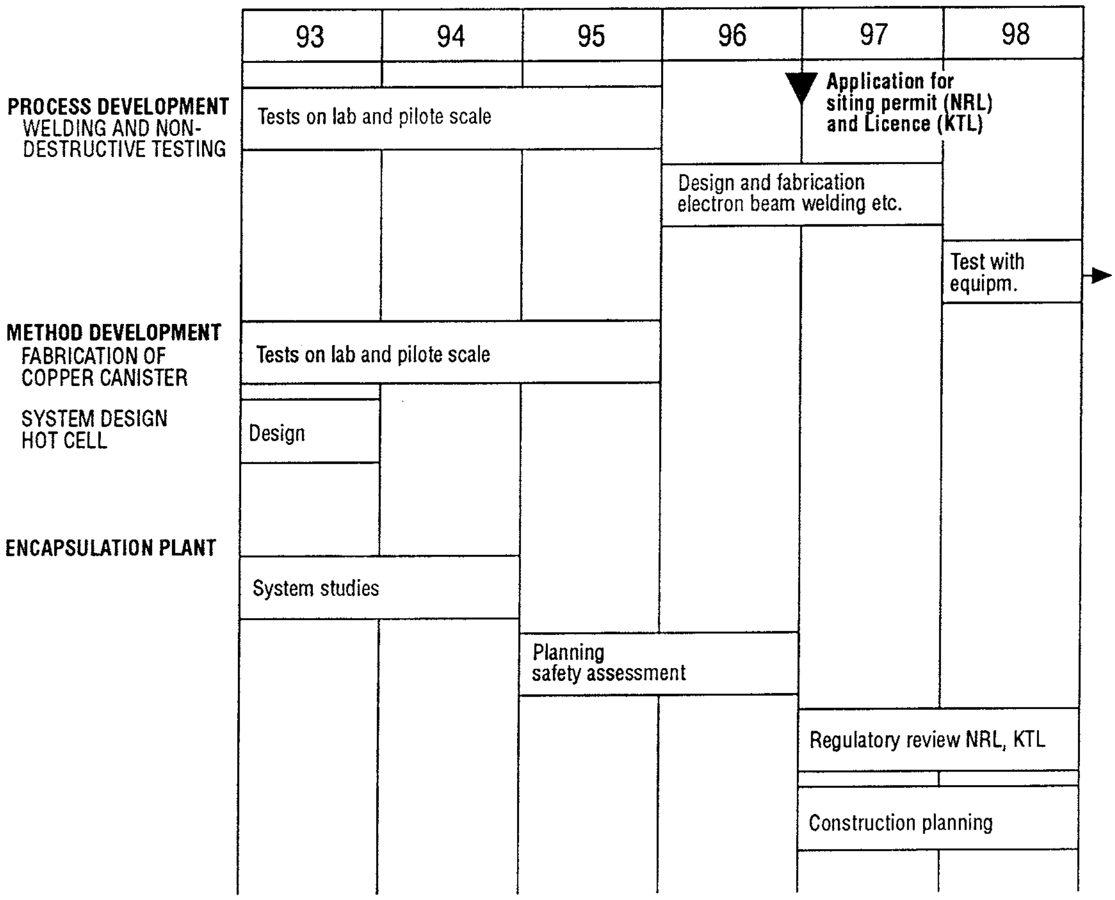


Figure 1-3. Timeschedule for encapsulation plant for spent fuel 1993–1998.
NRL = Act /1-20/ concerning the management of Natural Resources.
KTL = Act on Nuclear Activities

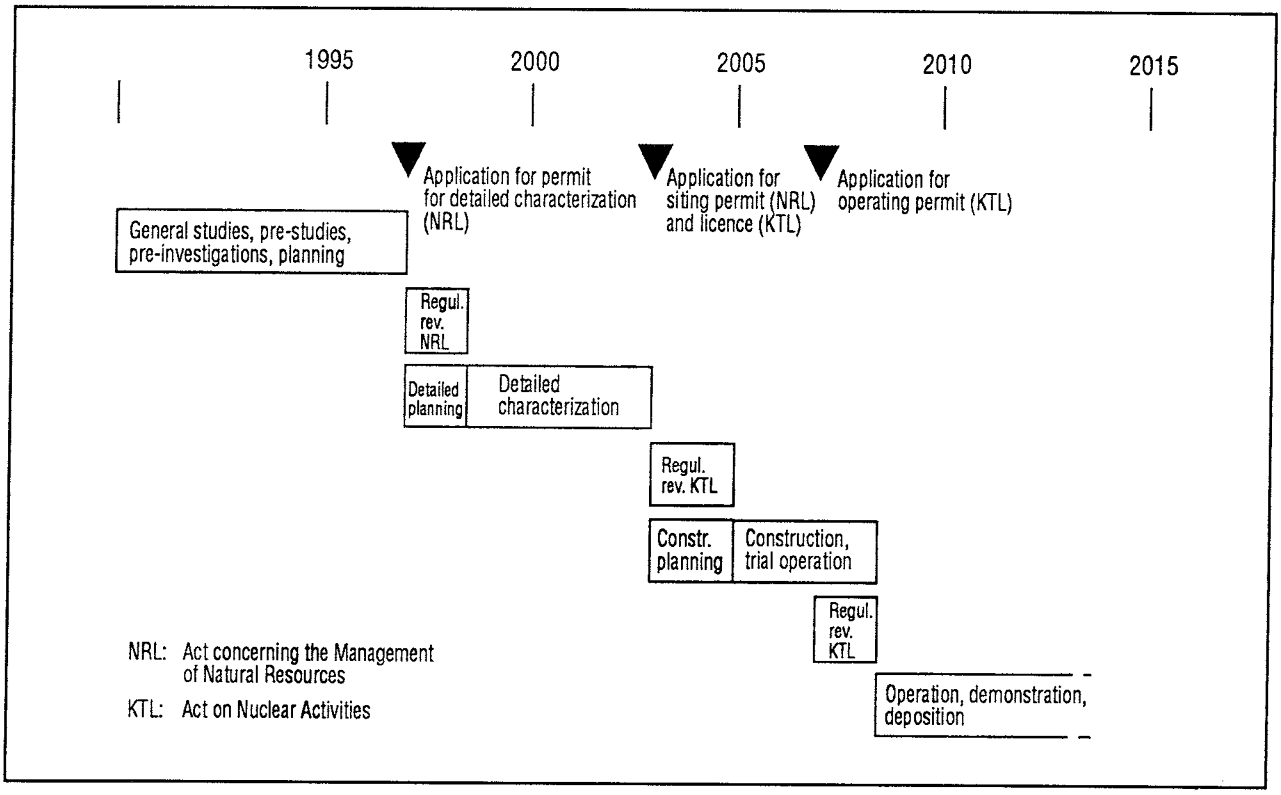


Figure 1-4. *Example of timeschedule for the deep repository up to the completion of demonstration deposition. The timeschedule gives the earliest possible completion dates.*

schedule for demonstration assumed here. SKB believes that the demonstration can be completed within about 20 years. Thus, as is evident from Figure 1-2, it is possible to follow this up with final disposal of the remaining fuel and waste immediately after 2020 if the decision to do so is made in about 20 years.

1.4 SITING

For the encapsulation of spent fuel, SKB plans to expand the central interim storage facility for spent fuel (CLAB) at the Oskarshamn Nuclear Power Station. The spent fuel is already being stored at CLAB, and SKB believes that expansion of CLAB with an encapsulation plant for spent fuel has clear advantages in terms of logistics, resource utilization and environmental impact. If special reasons emerge during the course of the work in favour of encapsulating at the deep repository instead, SKB will of course also consider the question of alternative siting of the encapsulation plant.

Siting and construction of a deep repository is planned to take place in stages during the 1990s and a few years into the 21st century. According to the estimates that can be made now of the time required to take decisions, carry out necessary inquiries and investigations and obtain necessary permits, demonstration deposition could be begun in about 15 years at the earliest.

The selection of candidate sites for the deep repository will be based on the fundamental requirements that must be made on a deep repository site from safety-related, technical, societal and legal viewpoints. It must be demonstrated for the selected site and selected repository system that the safety requirements imposed by the authorities are met. It must be possible to build the repository and carry out deposition in the intended manner. The siting process, the investigations and the construction work shall be carried out so that all relevant legal and planning-related requirements are met. And last, but not least, it shall be possible to carry out the project in harmony with the municipality and the local population.

An important point of departure for the planning of the siting process is the Government's decision regarding R&D-Programme 89 /1-13/. It states the following: *"The Government notes that SKB's choice of sites for a final repository will be reviewed by different authorities in connection with SKB's application for permission to carry out detailed characterization of two such sites under the Act (1987:12) of natural resources etc., the Environment Protection Act (1969:387) and the Planning and Building Act (1987:383)."* Furthermore, the Government emphasized that SKB should, during the course of the siting work, furnish information to concerned national authorities, county administrations and municipalities.

Based on these guidelines, the work of siting and construction of the deep repository is planned to proceed in the following stages, see Figure 1-4:

Stage 1:
General studies. Analysis of siting factors. Possible pre-studies of presumptive candidate sites. Selection of candidate sites. Pre-investigations at a couple of sites, including preliminary design. Technical and socio-economic studies. Evaluation of the results. NRL application for detailed characterization including an environmental impact statement with an initial safety assessment.

Stage 2:
Detailed characterization including excavation of necessary shafts and tunnels to planned repository depth. Evaluation of the results. Safety report. Environmental impact statement. Detailed design. Application for siting permit and licence (NRL, KTL).

Stage 3:
Construction and installation of equipment for handling/deposition. Final safety report. Application for operating permit (KTL).

Stage 4:
Commissioning. Demonstration deposition.

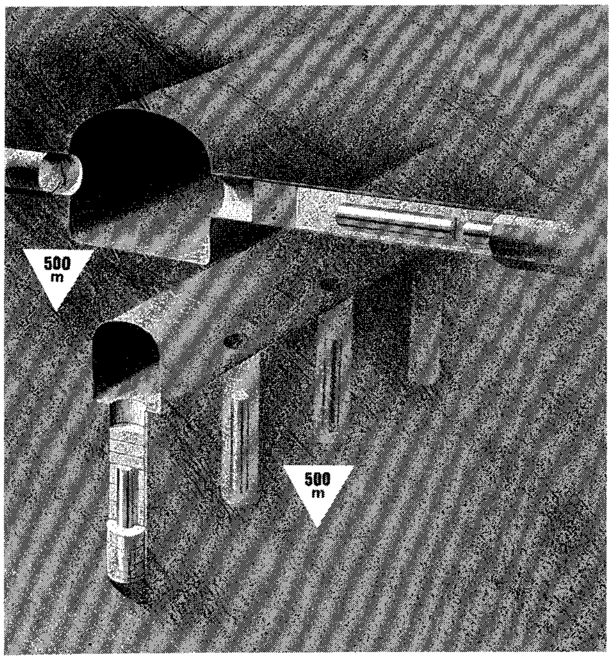
1.5 PLANNED SYSTEM FOR ENCAPSULATION AND DEEP DISPOSAL OF SPENT NUCLEAR FUEL

During the period 1986–1992, SKB has studied different alternative designs of a deep repository for final disposal of spent fuel.

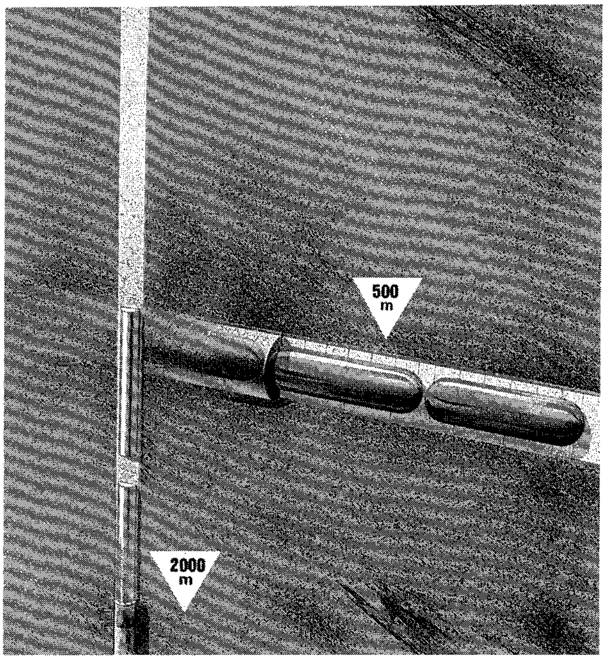
The reference alternative for SKB's work, for example with the PLAN reports /1-2/, has been a canister and repository design according to the description in the KBS-3 report /1-4/, ever since the latter was approved in conjunction with the fuelling permits for Forsmark 3 and Oskarshamn 3 in 1984. During the period 1986-1988, the so-called WP-Cave alternative was studied. As was reported in R&D-Programme 89, the conclusion of this evaluation was that the alternative was judged to be less advantageous than a more "distributed" design, e.g. the one described in KBS-3 /1-14/. SKB has not found reason to modify this conclusion.

Several other alternatives have been studied during the period 1989-1992, namely:

Deep holes (VDH = Very Deep Holes)
Deposition of a large number of canisters at a depth of between 2 and 4 km in 4 km deep boreholes /1-15/.



KBS-3 and Medium-long tunnels



Deep holes and Long tunnels

Figure 1-5. Alternative designs of the deep repository.

Long tunnels (VLH = Very Long Holes)
Horizontal deposition of relatively large canisters in the centre of full-face-bored tunnels with a tunnel length of several km /1-16/.

Medium-long tunnels (MLH = Medium Long Holes)
An intermediate solution between KBS-3 and long tunnels, the canister size is the same as for KBS-3 but deposition takes place horizontally in the centre of full-face-bored tunnels /1-17/.

Figure 1-5 illustrates the different alternatives. Highly-compacted bentonite is used as a buffer material around the canisters in all alternatives, except Deep Holes. Different canister designs have also been evaluated within the framework of the studies conducted of these alternatives. See Figure 1-6. The studies have been conducted within a single project, PASS /1-17/.

The conclusion of the studies is that the continued work on designing a deep repository for demonstration deposition should be concentrated on one alternative. In this way the desired concentration and goal orientation is achieved in the development and planning work.

Of the canister alternatives studied, the composite canister holding 12 BWR assemblies is by far the most advantageous and is chosen as the main alternative for the continued work. This canister consists of a steel container, which provides mechanical protection, surrounded by a copper container, which provides long-lasting corrosion protection. The corrosion life of the canister is estimated to be more than one million years, which is considerably longer than the time the waste has to be isolated /1-11/. The mechanical integrity of the canister has been evaluated by an expert group /1-18/. Since the canister is a vital barrier, some additional development should be conducted for the alternative of a lead-filled copper canister as a reserve alternative to the composite canister.

Of the studied repository designs, "Deep Holes" and "Long Tunnels" are judged to have clear disadvantages compared with KBS-3 and "Medium-Long Tunnels". The comparison between the two latter alternatives has, however, not yielded as clear an outcome. Technically, "Medium-Long Tunnels" is judged to be slightly more complicated in terms of the deposition procedure (emplacement of the canisters in the repository), but has on the other hand a significant potential for a lower cost. As far as long-term safety is concerned, the alternatives are equivalent. The KBS-3 design is kept as the main alternative for further work. In connection with adaptation to local conditions on the selected site, this design can be further optimized, whereby technically

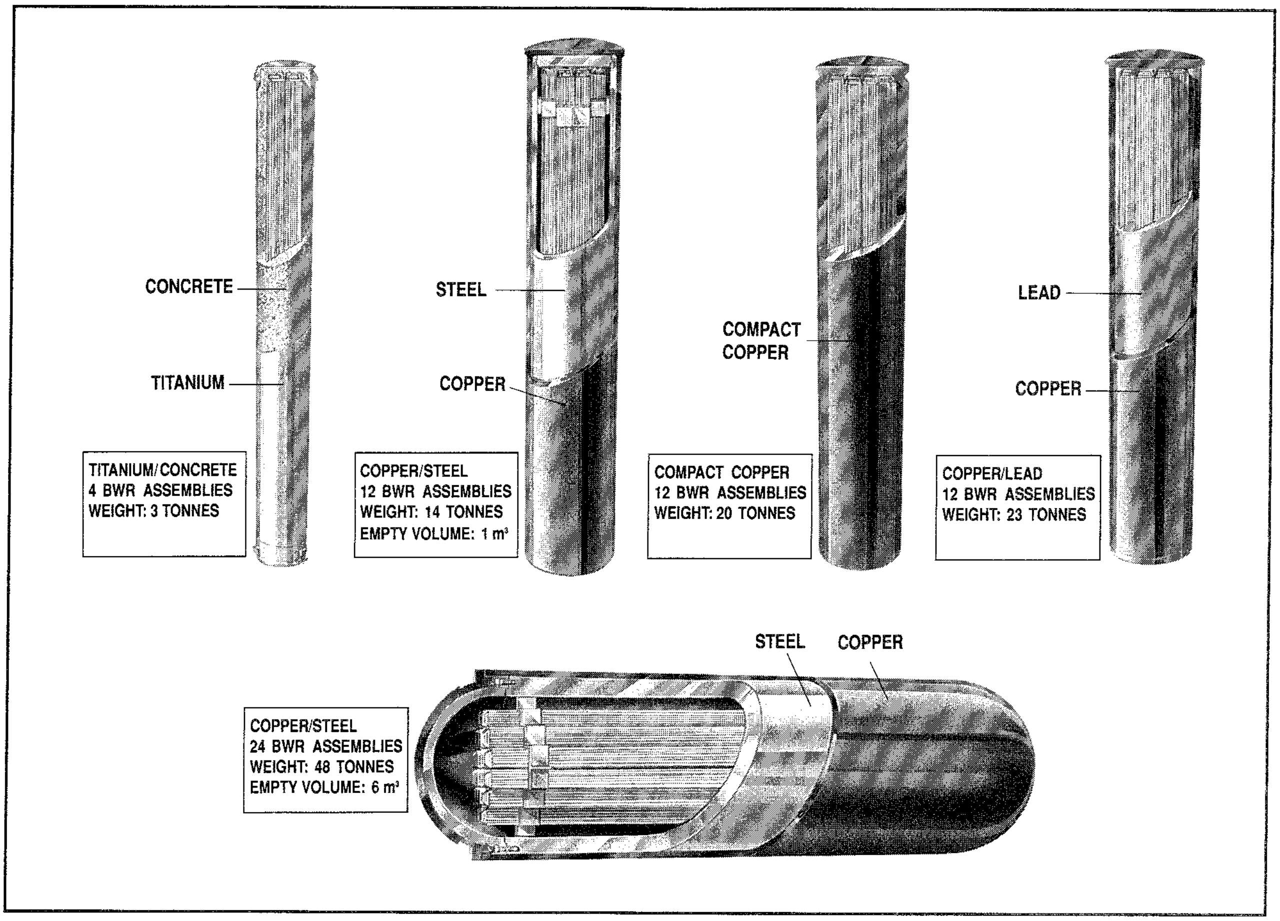


***Figure 1-6.** Studied canister designs for the different alternative designs of the deep repository.*

closely-related variants of the design can be given further consideration.

1.6 OUTLINE OF THE RD&D PROGRAMME

Two main activities are required in the development work in order to carry out a demonstration deposition of spent nuclear fuel in a deep repository: encapsulation and deep disposal. Safety assessments and supportive research and development are also required.

The encapsulation work entails final selection and testing of methods for fabrication, sealing and quality inspection of canisters, as well as construction, licensing, installation, trial operation and operation of a facility for encapsulation. The work of deep disposal entails siting, design, construction, licensing, installation, trial operation and operation of a facility and equipment for demonstration deposition in a deep repository.

The supportive R&D work entails further development of methods, models and data within the areas of spent fuel properties, geoscience, chemistry, materials and biosphere aimed at

- further refining the knowledge base and skills in modelling of processes that are important for the performance of the repository in order to better be able to quantify uncertainties and safety margins,
- following up of international developments in relevant fields.

The research is planned so that a continuity is obtained in the work and an updating of the knowledge base and analysis methods is done in good time before major assessments of performance or safety. Much of the supportive RD&D work will be concentrated to the Äspö Hard Rock Laboratory. Another important support for further development of the safety assessment is further studies of spent fuel properties, and natural analogues.

Besides work that comprises direct support for the main line – deep repository for demonstration deposition – some follow-up of alternative methods and systems for disposing of spent fuel is planned so that knowledge of these will be retained and further refined. In this way a basis will be created for the future evaluation of such systems in comparison with what is being demonstrated in Sweden. In addition, work is planned on other long-lived waste as well as for SFR and for decommissioning of nuclear power plants.

An important part of the RD&D programme is international cooperation, which is extensive and takes place in several different forms.

1.7 OUTLINE OF THE REPORT

Chapters 2-5 present the guidelines we have in Sweden for the management of nuclear waste, describe the properties of the waste and provide a brief overview of existing facilities and systems.

Chapter 6 constitutes an introduction to the main part of the programme in the form of a review of different methods for disposal of long-lived waste.

Chapters 7-9 describe the main focus of the programme with a description of the entire stage-by-stage construction process (chapter 7) and the siting and construction of an encapsulation plant for spent fuel (chapter 8) and a deep repository for demonstration deposition (chapter 9).

Chapter 10, 11 and 12 summarize planned work within safety assessment, supportive R&D and the Äspö Hard Rock Laboratory.

Chapter 13 summarizes the results of the international Stripa Project that was concluded during 1992 /1-19/. R&D for alternative methods and other waste are presented in chapter 14, while chapter 15 deals with the decommissioning of nuclear facilities. Costs and priorities are dealt with in chapter 16, and the report concludes with an account of international cooperation.

Chapters 6-12 comprise the core of the programme and are recommended priority reading, along with chapter 16.

Several background reports to this main report provide a more detailed description of:

- Siting.
- Detailed programme for supportive research and development 1993–1998.
- The Äspö Hard Rock Laboratory.

There is also a broad body of information in SKB Annual Report 1991 /1-21/, SKB's Technical Reports, the SKB 91 safety assessment /1-11/, the reports on the Stripa Project /1-19/ and the report on the PASS Project /1-17/.

2 PREMISES

2.1 GUIDELINES FOR RADIOACTIVE WASTE MANAGEMENT IN SWEDEN

The goal of radioactive waste management in Sweden is to dispose of all radioactive waste products generated at the Swedish nuclear power plants and other nuclear facilities in the country in a safe manner. Furthermore, all other radioactive waste produced in Sweden shall be safely disposed of.

The following general guidelines were presented at an early stage:

- The radioactive waste products shall be disposed of in Sweden.
- The spent nuclear fuel shall be temporarily stored and finally disposed of without reprocessing.
- Technical systems and facilities shall meet high standards of safety and radiation protection and satisfy the requirements of the Swedish authorities.
- The systems for waste management shall be designed so that requirements on the control of fissile material can be satisfied.
- In all essential respects, the waste problem shall be solved by the generation that utilizes electricity production from the nuclear power stations.
- A decision on the design of the final repository for spent nuclear fuel shall not be taken until around the year 2000 so that it can be based on a broad body of knowledge.
- The necessary technical solutions for the disposal of the Swedish waste shall be developed within the country, at the same time as available foreign knowledge shall be gathered.
- The work shall be guided by the regulatory authorities' continuous review and assessment and the directives issued by the.
- The activities shall be conducted openly and with good public insight.

These general guidelines were set forth in SKB's R&D-Programme 86 /2-1/ and did not occasion any special comments on the part of the reviewing bodies at that time. They were reiterated in R&D-Programme 89 /2-2/ and certain parts were then discussed and questioned with respect to the spent nuclear fuel by the National Board for Spent Nuclear Fuel (SKN). The Board stated, for example:

- "In the opinion of the Board it can be questioned whether the interests of future generations are best served by having the waste problem solved in all essential respects by our generation, and by having the decision as to the ultimate design of the final repository taken already around the year 2000. In this case, we in Sweden will probably be the first country to take such a decision with regard to final disposal in crystalline rock. Other countries will probably continue to debate the disposal problems far into the 21st century. New treatment methods or deposition technologies which have more favourable safety characteristics or are more cost-effective than today's will probably be developed for both radiotoxic and chemical wastes. The ultimate decisions on final disposal should not be taken until the strategy and technology for disposal of long-lived hazardous waste has matured to such a point that there is no risk that the method chosen will be found to be the wrong choice within the foreseeable future.
  SKB should therefore explore the possibility of implementing the final disposal scheme in stages with 'checkpoints' and opportunities for remedial actions along the way...".
- "The National Board for Spent Nuclear Fuel believes that a demonstration facility should take the place of the final repository in SKB's current planning..." /2-3/.

The Government takes up SKN's viewpoints in its decision on R&D-Programme 89 and states the following:

- "The Government stresses that no commitment should be made to a given handling or disposal method until all potential safety and radiation protection problems are understood.
  The Government believes that one of the premises for further research and development activities should be that a final repository for nuclear waste and spent nuclear fuel can be put into operation in stages with checkpoints and opportunities for remedial actions along the way. SKB should explore the possibilities of including a demonstration-scale final repository as a stage in the process of designing a final repository..." /2-4/.

The background to these statements by SKN and the Government has been discussed in various contexts by, among others, SKN and KASAM (the Consultative

Committee for Nuclear Waste Management). SKB has carefully considered the arguments put forth in the discussion. Our conclusion is that a broad political and public opinion seems to agree on the following fundamental principles for nuclear waste management in Sweden:

- We already have nuclear waste, and it must be disposed of in a safe manner in our own country.
- Future safety should be based on a disposal method that does not require supervision and/or maintenance, since this would entail that generation after generation, far into the future, would have to retain knowledge of the waste and have the will, capability and resources to perform such supervision and maintenance. We know too little about the society of the future to base long-term safety on this assumption.
- While working concretely and resolutely towards realizing the final disposal of all nuclear fuel, it is advisable to retain as much freedom of choice as possible with a view towards the possibility that alternative and somehow superior or simpler solutions may be found, or the possibility that there may be a re-evaluation of the current attitude towards the re-use (reprocessing) of some of the fissile materials (U, Pu) in the fuel. The Nordic radiation protection authorities have formulated the following principle: *The burden on future generations shall be limited by implementing at an appropriate time a safe disposal option which does not rely on long-term institutional controls or remedial actions as a necessary safety factor* /2-5/. This requirement is also formulated on an international level /2-6/ and has been generally accepted as a fundamental principle by all countries with nuclear power.

As regards the operational waste from the nuclear power plants and some other waste from research etc., there are already facilities and systems in operation which satisfy the requirements that follow from the general guidelines.

The viewpoints that have been proffered with respect to the value of preserving freedom of choice have been taken into account in the present RD&D programme, see further chapter 7.

2.2 APPLICABLE LEGISLATION ETC.

The obligations of the owners of nuclear power reactors with regard to handling and final disposal of radioactive waste are set forth in the Act /2-7/ on Nuclear Activities, in the Ordinance on Nuclear Activities /2-8/ and in certain licences and guidelines issued by the Government.

The provisions and guidelines entail in brief that the owners of nuclear power plants are responsible for:

- adopting the measures that are needed in order to handle and finally dispose of generated nuclear waste in a safe manner and to decommission and dismantle the nuclear power plants and appurtenant facilities,
- the comprehensive research and development activities that are required to carry out these measures, including studies of alternative handling and disposal methods,
- preparing a programme for research and development and other measures every third year starting in 1986, including an account of the research results obtained.

The changes recommended by the committee on "Review of legislation in the nuclear energy field" /2-9/ does not alter these provisions. However, the committee recommends a ban on the erection of facilities for the final disposal of spent nuclear fuel or nuclear waste from nuclear facilities in another country. Furthermore, the committee recommends that the authorities be given broader powers to issue stipulations in conjunction with their review of the programme for research and development and other measures.

2.3 BACKGROUND

Research regarding the handling and final disposal of radioactive waste started on a large scale in Sweden in connection with the establishment of the National Council for Radioactive Waste (PRAV) in 1975. The Council was created on the recommendation of the AKA Committee /2-10/. The research was intensified in conjunction with the enactment of the "Stipulation Act" in 1976/77, when the KBS (Nuclear Power Safety) Project was started by the nuclear utilities. The project was administratively tied to SKB. The project developed two final disposal methods: KBS-1 for vitrified high-level reprocessing waste (1977) /2-11/ and KBS-2 for the handling and final storage of unreprocessed spent nuclear fuel (1978) /2-12/.

The KBS-1 report was submitted in support of applications for fuelling permits for the Ringhals 3 and 4 and Forsmark 1 and 2 reactors. The Government issued fuelling permits in 1979 and 1980.

When the Financing Act /2-13/ entered into force, the National Council for Radioactive Waste was abolished and the National Board for Spent Nuclear Fuel (NAK, later SKN) was created in its stead. The purpose of this Board is to review, regulate and oversee the activities of the nuclear utilities (SKB) within the waste management field. As from 1 July 1992, SKN's duties have been transferred to the Swedish Nuclear Power Inspectorate.

In 1983, SKB presented a new report on the final disposal of spent nuclear fuel. The report was based on the same method as that described in KBS-2, but the new report, KBS-3, was based on a much broader and deeper body of knowledge /2-14/.

The KBS-3 report was presented in support of the applications for fuelling permits for the Forsmark 3 and Oskarshamn 3 reactors. The Government granted these permits under the Act on Nuclear Activities /2-7/ in June 1984. A research programme /2-15/ prepared by SKB in February 1984 was also submitted in support of the permit applications. Since then, operating permits for Barsebäck 2, Ringhals 3 and 4 and Forsmark 1 and 2 have also been based on KBS-3.

In September 1989, SKB presented the second research programme under the Act on Nuclear Activities /2-2/. The results of SKB's research work are reported continuously in SKB's technical reports. Annual summaries are included in the SKB Annual Report /2-16, 2-17/.

2.4 R&D-PROGRAMME 89 - EXPERT REVIEW

After R&D-Programme 89 had been submitted to SKN in September 1989, the programme was circulated for review and comment to a large number of institutions and individuals in Sweden. The review period expired on 1 February 1990.

On the basis of the viewpoints received and its own judgements, SKN compiled a review report and submitted it to the Government in March 1990 /2-3/. A summary of different reviewers' viewpoints on SKB's research and development work has been compiled /2-18/. The Government's decision in regard to R&D-Programme 89 was handed down in December 1990 /2-4/.

Wherever possible, SKB has taken heed of the comments received on R&D-Programme 89 in the present RD&D-Programme.

3 WASTE FROM THE SWEDISH NUCLEAR POWER PROGRAMME

3.1 CLASSIFICATION OF RADIOACTIVE WASTE

Radioactive waste from the Swedish nuclear power programme varies widely in terms of form and activity content, all the way from virtually inactive trash to spent fuel, which has a very high activity content. Different waste forms therefore impose different demands on handling and final disposal.

From the handling viewpoint, it is practical to distinguish between low-level, intermediate-level and high-level waste. Low-level waste can be handled and stored in simple packages, without any special protective measures. Intermediate-level waste must be radiation-shielded for safe handling. High-level waste requires not only radiation shielding, but also cooling for a certain period of time in order to permit safe storage.

From the viewpoint of final disposal, the half-life of the radionuclides contained in the waste is of great importance. A distinction is made between short- and long-lived wastes.

Short-lived waste mainly contains radionuclides with a half-life shorter than 30 years, i.e. it will have decayed to a harmless level within a few hundred years. This waste will be deposited in the final repository for radioactive operational waste, SFR, at Forsmark. Some very low-level and short-lived waste can be dumped on a simple refuse tip (shallow land burial).

Long-lived waste remains radioactive for thousands of years or more and requires a more qualified final disposal.

Table 3-1 shows an example of the classification of wastes from the Swedish nuclear power programme on the basis of activity and life.

Table 3-1. Example of classification of radioactive wastes.

LIFE	RADIOACTIVITY High	Intermediate	Low
Long (thousands of years)	Spent fuel	Certain core components	
Intermediate (a few hundred years)		Ion exchange resins Discarded components Decommissioning waste	Maintenance waste

3.2 WASTE FROM THE NUCLEAR POWER PLANTS

The waste from the nuclear power plants is usually divided into the following groups with regard to its subsequent handling:

- Spent nuclear fuel.
- Operational waste.
- Core components and reactor internals.
- Decommissioning waste.

3.2.1 Spent nuclear fuel

Most of the radioactive materials (approx. 99%) that are formed in a nuclear power plant are present in the spent fuel.

Some of the fuel types used in Swedish power reactors are described in KBS-3 /3-1/. A fuel assembly for a boiling water reactor (BWR) contains about 180 kg of uranium, while an assembly for a pressurized water reactor (BWR) contains about 460 kg of uranium. The design differs somewhat between different manufacturers and between fuel produced at different times. From the viewpoint of final disposal, the differences between different fuel types are generally of little consequence. This also applies to odd fuel assembly types with oxide fuel clad in Zircaloy, for example MOX fuel and Ågesta fuel.

The spent fuel consists mainly of unfissioned uranium, while most of the radioactivity comes from the fission products and transuranics present in the fuel. Examples of composition, activity level and other data for spent fuel are given in /3-2/.

The high level of activity in spent fuel means that it continues to emit heat for a long time after it has been discharged from the reactor. This is of great importance for how the spent fuel will be handled and disposed of. The residual heat in the fuel decreases by a factor of 10 between 1 and 40 years after discharge. It then takes another 1 000 years or so for the residual heat to decrease by yet another factor of 10.

3.2.2 Operational waste

The category "operational waste" includes a number of different types of waste obtained in connection with the operation and maintenance of the reactors. The main constituents are ion exchange resins and filters obtained continuously during operation from cleanup of the reactor water. The operational waste also includes replaced components from the reactor systems as well as protective clothing, plastic, paper, insulating materials etc. that have been used in areas where activity is present and may therefore be contaminated.

The operational waste is low- and intermediate-level and mainly contains radionuclides with half-lives shorter than 30 years. The concentration of long-lived radionuclides is very low. The radioactivity of the operational waste will therefore have declined to a level comparable to the natural background radioactivity in the rock within a few hundred years.

The operational waste is conditioned at the nuclear power plants to give it a packaging and form that is appropriate for its subsequent handling. Different conditioning methods are employed at different nuclear power plants. This is described in greater detail in /3-3/.

Similar waste also comes from the operation of the central interim storage facility for spent nuclear fuel, CLAB, and from Studsvik.

3.2.3 Core components and reactor internals

Components located in or near the core inside the reactor vessel are exposed to a strong neutron flux and thereby develop a high level of induced activity. Some of these components, for example neutron detectors, are successively replaced at intervals of a few years. Others, for example the moderator tank, are used for the entire lifetime of the reactor and only become waste when the reactors are dismantled.

Fuel channels and other structural components in the fuel assemblies are included among the core components here.

The core components and some reactor internals have a very high radiation level when they are discharged from the reactor. This radioactivity is dominated by cobalt-60, which has a half-life of about 5 years, which means that the radiation level declines by a factor of 1 000 in 50 years. Core components and reactor internals also contain some radionuclides with a long half-life, such as nickel-59 (90 000 years) and niobium-94 (20 000 years). The radiotoxicity of these nuclides is lower than that of the transuranics, and requirements on the final disposal of these components are therefore less stringent than for spent fuel.

3.2.4 Decommissioning waste

When a nuclear power plant is shut down for good, parts of the plant are radioactive and must therefore be disposed of in a safe manner. These parts include firstly the reactor vessel and reactor internals, as well as the concrete nearest the reactor vessel, which contain induced activity, and secondly different parts of the reactor systems, which have been radioactively contaminated. However, most of the plant has not come into contact with radioactive materials and the waste can therefore be handled like normal waste from the dismantling of industrial facilities.

The waste obtained from decommissioning consists primarily of components of steel – e.g. tanks, vessels, pipes and valves – from the reactor's process systems. It also includes large quantities of concrete, 90% of which is uncontaminated. The dismantling and demolition work also gives rise to a certain amount of process waste from the water and air purification systems that are in operation during the decommissioning period.

The radioactive decommissioning waste is all low- and intermediate-level. However, the activity level varies considerably between different parts. A large portion of the metal scrap can be released for unrestricted use. The concrete and some other materials can be dumped on an ordinary industrial landfill, possibly adjacent to the reactor facility. However, most of the active decommissioning waste has an activity level that calls for deposition in SFR. As mentioned above, certain highly radioactive reactor internals from the reactor vessel are also obtained from decommissioning and require special treatment.

A large portion of the activity consists of surface contamination, which can be removed by means of various decontamination methods. The quantity of material that can be released for unrestricted use is therefore dependent on how far the decontamination work is carried.

3.3 OTHER RADIOACTIVE WASTE

Besides the nuclear power plants, the main sources of radioactive waste in Sweden are the central interim storage facility for spent nuclear fuel, CLAB, and the coming encapsulation plant for spent fuel, as well as the Studsvik Research Facility. Waste from the use of radioactive materials in industry, medical care and research is also collected at Studsvik.

3.3.1 Waste from CLAB and encapsulation plant

The waste from CLAB is of the same kind as the operational waste from the reactors. It is also conditioned in the same manner. Similar waste will also be obtained from the encapsulation plant for spent fuel.

3.3.2 Waste from Studsvik

Nuclear research has been conducted at Studsvik since the mid-1960s. It has included operation of the R2 research reactor and different types of studies of radioactive products, for example fuel rods. This activity has generated waste, which has been accumulated in storage facilities in Studsvik. Work on conditioning and classifying this waste has been under way since 1986.

The operation of the R2 reactor produces both spent fuel and operational waste. The fuel is normally sent back to the USA and therefore does not have to be disposed of in Sweden. A few years ago, the USA imposed a moratorium on reception of this type of fuel. The R2 fuel is being stored in Sweden until the moratorium is lifted.

Other waste from R2 is similar to the operational waste from the nuclear power plants and is treated in a similar manner.

The waste from the R&D activities is, on the other hand, of a different character. Some of this waste consists of fuel residues and contains considerable quantities of long-lived transuranics. It therefore requires a similar final disposal as the spent fuel. It is encapsulated in steel tubes and transferred to interim storage in CLAB. Fuel from the Ågesta reactor, which was previously stored in Studsvik, has also been transferred to CLAB. Left in Studsvik is fuel from the disused R1 research reactor. This fuel has to be further treated prior to final disposal.

The R&D activities also generate different types of long-lived low- and intermediate-level wastes, which also make high demands on final disposal. This waste is currently being packaged in different types of containers suitable for final disposal and will be temporarily stored at Studsvik until the final repository for long-lived low- and intermediate-level waste is put into operation.

3.3.3 Waste from reprocessing

In the reprocessing of spent nuclear fuel, uranium and plutonium are separated from fission products and other transuranics. This process gives rise to both high-level vitrified waste, which contains most of the radioactivity, and low- and intermediate-level waste solidified in concrete or bitumen.

Most of the waste from reprocessing contains large quantities of transuranics and is therefore long-lived.

Current plans call for the reprocessing of only 140 tonnes of Swedish fuel. This will be done at BNFL in Great Britain, which also takes care of the waste. The previous reprocessing contracts with COGEMA have been re-assigned to other customers and will not be used for Swedish fuel.

Reprocessing waste is therefore no longer included in the Swedish plans for the back end of the nuclear fuel cycle.

3.4 ESTIMATED WASTE QUANTITIES

The total quantity of radioactive waste from the Swedish nuclear power programme has been estimated in PLAN 92 /3-4/. The results are shown in Table 3-2.

Table 3-2. Main types of radioactive waste products.

Product	Main source	Unit	No. of units	Volume in final repository, m^3
Spent fuel		Tonnes U	7 900	9 800
Alpha-contaminated waste	Low- and intermediate-level waste from Studsvik	Drums, moulds	1 600	1 500
Core components	Reactor internals	Moulds	2 400	19 700
Low- and intermediate-level waste	Operational waste from nuclear power plants and treatment plants	Drums, moulds	56 000	91 500
Decommissioning waste	From decommissioning of nuclear power plants and treatment plants	10-20 m^3 containers	5 500	111 500

4 EXISTING SYSTEM FOR MANAGEMENT OF RADIOACTIVE WASTE FROM NUCLEAR POWER PLANTS

4.1 GENERAL

The safe handling and final disposal of the waste from nuclear power requires planning, construction and operation of a number of facilities and systems. Figure 4-1 illustrates schematically the different parts of the planned Swedish waste management system. These parts are described in detail in the annual report of the costs for management and disposal of the radioactive waste products of nuclear power, PLAN 89, which the power utilities have submitted through SKB /4-1/. Only a brief overview is presented here.

The facilities are also planned to accommodate the radioactive waste in Sweden that does not come from electricity-generating reactors, see Chapter 3.

The design of the system is based on the following fundamental principles:

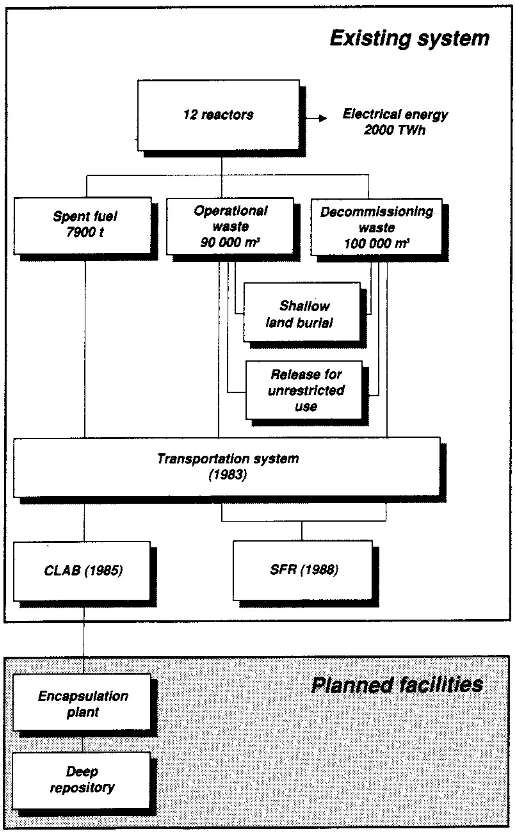


Figure 4-1. *The Swedish waste management system.*

- Short-lived waste will be disposed of as soon as possible after it has been generated.
- Spent fuel will be stored for about 40 years before it is placed in the final repository. This will limit heat generation in the final repository.
- Other long-lived waste will be deposited in connection with the final deposition of spent fuel.

Essential parts of the waste management system are already in operation, namely the central interim storage facility for spent nuclear fuel, CLAB, the final repository for reactor waste, SFR, and the transportation system. Plans call for the expansion of CLAB and SFR at the beginning of the 21st century so that they can accommodate all spent fuel and waste from the Swedish nuclear power programme.

The parts remaining to be built are a encapsulation plant for the encapsulation of spent fuel and a final repository for long-lived waste. Extensive R&D work is being pursued for these parts of the system, the aim of which is to find a suitable design and site.

Management of the radioactive waste products of nuclear power also includes decommissioning of the nuclear power plants and other facilities when they have been taken out of operation and final disposal of the waste from decommissioning, see chapter 15.

4.2 FACILITIES AND SYSTEMS IN OPERATION

4.2.1 Final repository for radioactive operational waste, SFR

The final repository for radioactive operational waste, SFR, is situated at the Forsmark Nuclear Power Station /4-2/. Operational waste from the Swedish nuclear power plants is being emplaced in SFR, along with similar waste from CLAB and Studsvik. The Studsvik waste also contains waste from the use of radioisotopes within research, industry and medicine.

The waste being deposited in SFR is low- and intermediate-level and short-lived, which means that its radioactivity will have declined to the same level as that of the natural background radioactivity in the rock within a few hundred years. Table 4-1 shows the quantities of wastes that are planned to be deposited in SFR. A smaller quantity of operational waste and decommissioning waste from CLAB and the encapsulation plant for spent fuel will be deposited in the deep repository for high-level waste. This explains the difference in the waste quantities between Table 3-2 and Table 4-1.

Table 4-1. Waste to be deposited in SFR.

	Storage volume (m^3)
Operational waste	
Intermediate-level	65 000
Low-level	25 000
	90 000
Decommissioning waste	
Intermeadiate level	12 000
Low-level	88 000
	100 000

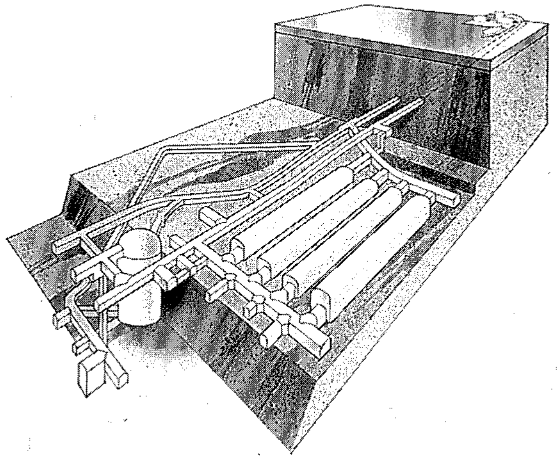


Figure 4-2. Plan of rock caverns and tunnels in the final repository for radioactive operational waste, SFR.

SFR is situated in rock with a rock cover of about 50 m. It consists of different rock caverns, designed with reference to the activity content of the different kinds of waste, see Figure 4-2.

SFR was put into service in 1988. Rock vaults for disposal of decommissioning waste are also planned in connection with SFR. These vaults will be approved and built when the time comes to decommission the nuclear power plants. A new Government licence is required before that.

The application for a siting permit for SFR stated that the facility may later be expanded so that core

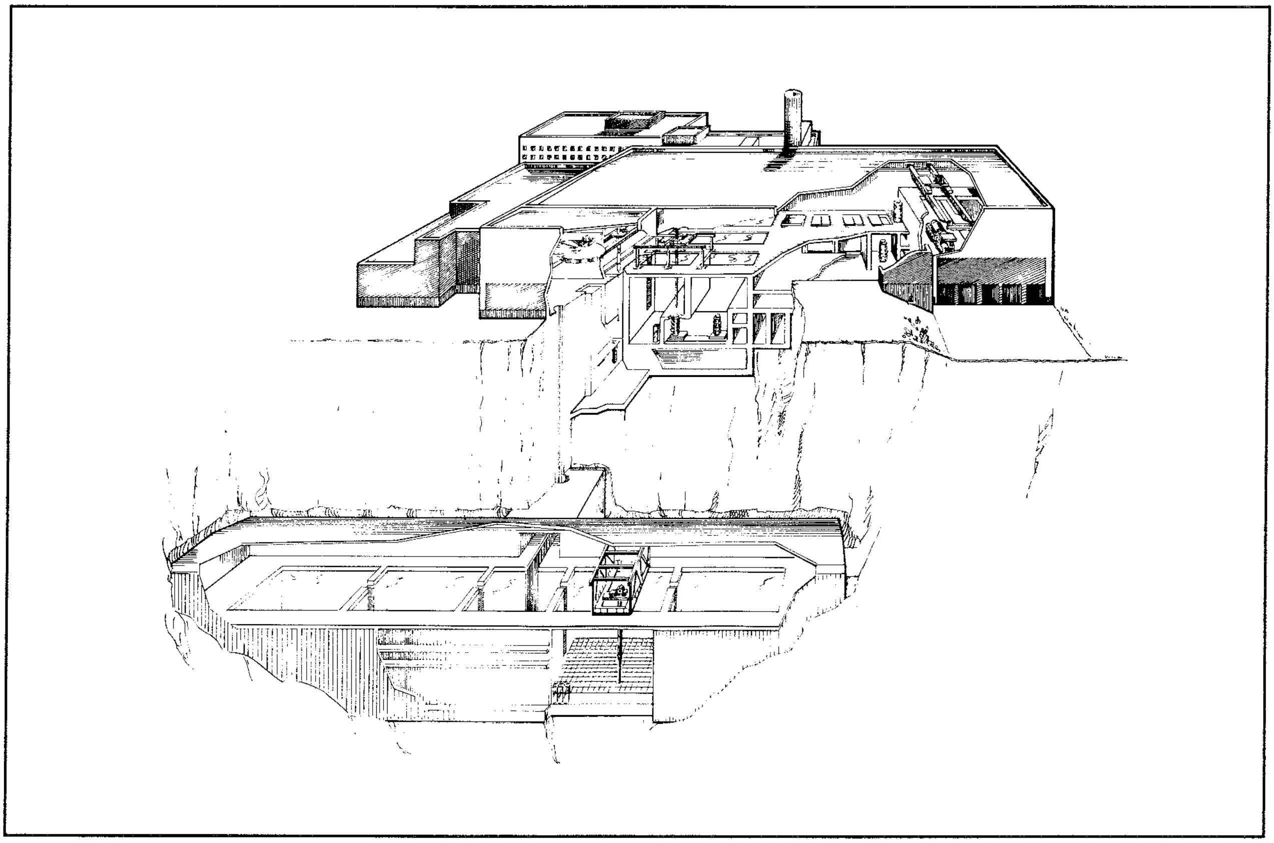


Figure 4-3. Plan of receiving building and rock cavern in the central interim storage facility for spent nuclear fuel, CLAB.

components and reactor internals can also be deposited there. Now it is assumed for practical reasons that this waste will instead be disposed of in connection with the final repository for long-lived waste. However, the option of depositing it in SFR should be kept open.

4.2.2 Central interim storage facility for spent nuclear fuel, CLAB

The fuel will be stored for about 40 years in the central interim storage facility for spent nuclear fuel, CLAB, which is located adjacent to the Oskarshamn Nuclear Power Station. During this interim storage period, the fuel's activity content and residual heat will decline by about 90%. CLAB was taken into operation in 1985, thereby relieving the need for on-site storage capacity at the nuclear power stations /4-3/.

CLAB consists of an above-ground receiving building and an underground storage complex in rock, see Figure 4-3. The fuel is handled and stored under water. The present-day capacity of the facility is about 5 000 tonnes of spent fuel in 4 pools. An expansion is planned for the beginning of the 21st century so that all fuel from the Swedish programme, about 8 000 tonnes, can be stored in CLAB. The facility is prepared for this and the expansion can be carried out at the same time as fuel is being brought into and stored in the pools in the existing rock cavern. Besides spent fuel from the Swedish nuclear power plants, fuel from the Ågesta reactor and fuel residues from studies at Studsvik are also being stored in CLAB.

Core components and reactor internals can also be stored in CLAB.

4.2.3 The transportation system

A transportation system based on sea transports is used for shipments of spent fuel and radioactive waste /4-4/. It consists of a ship, M/S Sigyn, transport containers and terminal equipment, see Figure 4-4. The transport containers meet the stringent requirements on radiation shielding and ability to withstand external stresses that have been issued by the International Atomic Energy Agency (IAEA). Different types of transport containers are used for spent fuel and for low- and intermediate-level waste.

M/S Sigyn has been in use since 1982. Since 1988, the ship has been transporting both fuel from the nuclear power plants to CLAB and operational waste to SFR. If needed, the transportation system can later be augmented with equipment for e.g. rail transport, in preparation for the shipments to the final repository for long-lived waste. The need will depend on where the final repository is located.

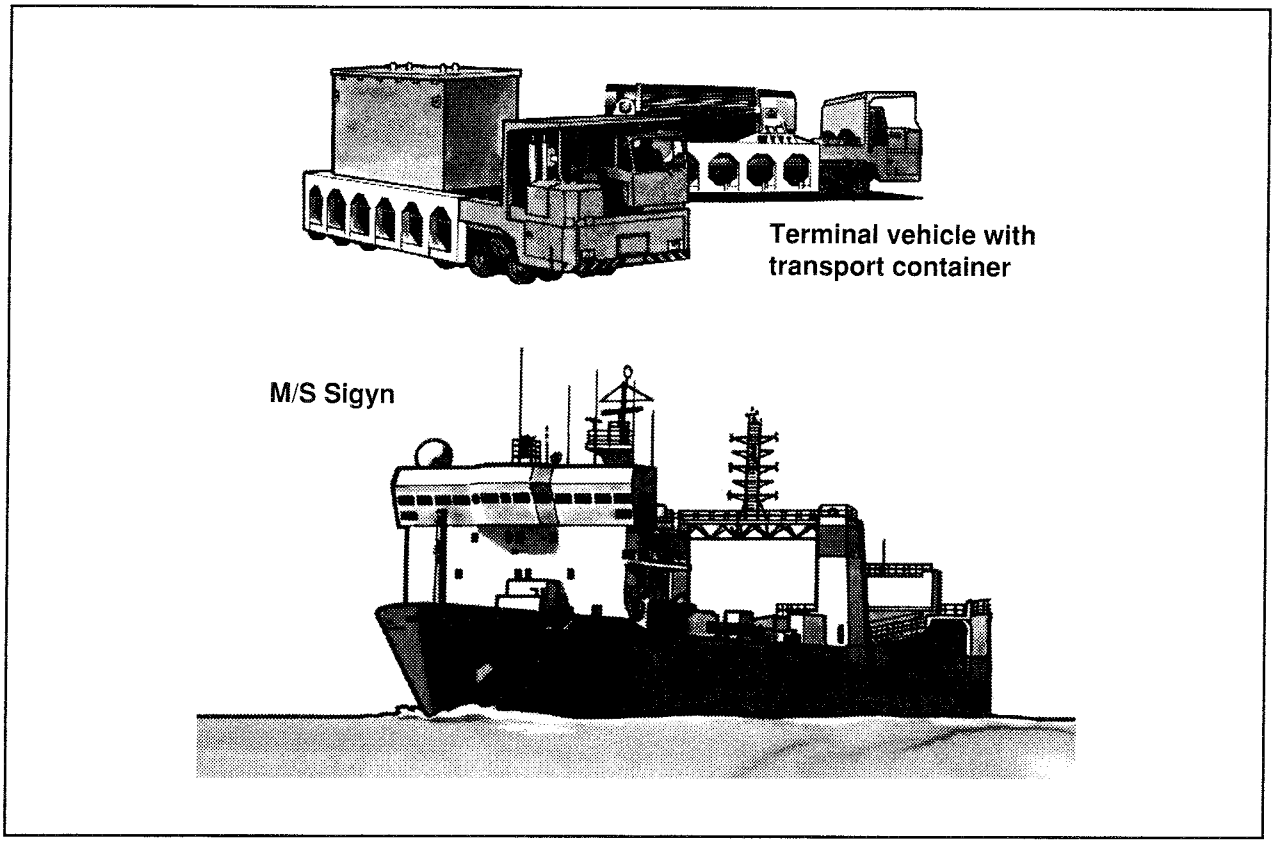


Figure 4-4. The Swedish transportation system for radioactive waste.

5 METHODS FOR TREATMENT OF SPENT NUCLEAR FUEL

5.1 GENERAL

A number of possible principles for treatment and final disposal of radioactive waste are elucidated in SKB PLAN 87 and in SKB's PLAN 82 part 1 /5-1,2/.

The following different treatment methods can thereby be distinguished:

- encapsulation of the fuel for direct final disposal;
- reprocessing of the fuel and recycling of uranium and plutonium;
- partitioning and transmutation.

In the first case, the spent fuel is regarded as waste without any further value, while in the case of reprocessing and partitioning, the remaining energy raw materials in the fuel – mainly uranium and plutonium – are recovered for further use as nuclear fuel.

In all cases, however, high-level and long-lived waste products are obtained which must be collected for final disposal. In the case of partitioning and transmutation, an effort is made to minimize the quantity of long-lived radionuclides in the waste by converting (transmuting) them to more short-lived nuclides. The quantity of long-lived plutonium is reduced considerably in reprocessing as well. In both reprocessing and transmutation, however, the volume of long-lived waste increases due to the generation of secondary waste in the processes.

The main alternative for management of the spent nuclear fuel from the Swedish nuclear power plants has been during the 1980s and still is interim storage for a relatively long period of time, about 40 years, followed by encapsulation and direct final disposal. Other treatment methods are being studied in other countries, however. This chapter provides an overview of the most important of these methods.

5.2 ENCAPSULATION OF FUEL

Through direct encapsulation of the spent nuclear fuel, it is enclosed and isolated in containers without any chemical or mechanical treatment of the fuel pellets themselves. However, different methods can be employed to reduce the volume of the entire fuel assembly that needs to be encapsulated together with the fuel pellets.

Figure 5-1 shows typical BWR and PWR assemblies as they look when they are taken out of the reactors. The BWR assemblies are equipped with boxes, while the PWR assemblies are open.

The boxes are longer than the fuel and the canisters can be made shorter if the boxes are removed prior to encapsulation. On the other hand, a special handling procedure must be devised for the removed boxes.

The assemblies without boxes can be further dismantled so that each fuel rod can be handled individually. The rods can then be arranged more densely in each canister, called rod consolidation. Other metal parts are handled separately. In practice, a rod consolidation can be achieved that doubles the quantity of fuel per unit volume in the canister compared to encapsulation of assemblies without boxes.

A final possible processing step is cutting the rods into lengths. The quantity of fuel per unit volume in the canister is not changed; the operation merely allows the use of shorter canisters. From a safety viewpoint, this step is disadvantageous since the encapsulation process is complicated and the Zircaloy around the fuel pellets is cut open, releasing a small quantity of gas in the fuel-clad gap.

The planned procedure in the Swedish programme is to separate boxes and fuel assemblies and deposit the boxes embedded in concrete moulds in a repository near the deep repository for the spent fuel. The option of keeping the boxes on the fuel is also being studied.

Rod consolidation could be considered if an interim storage method is chosen in CLAB that involves rod consolidation. However, this is not being considered at present. Chopping of the fuel rods has never been considered in view of the safety-related operational problems.

Encapsulation itself can be done in several ways. Several alternatives have been studied within the framework of SKB's PASS project. See chapters 6, 11 and the background report on the PASS project. As will later be explained in chapter 6, an alternative consisting of a steel canister placed in a corrosion-protective copper shell has been chosen for further analysis. The inside void can be filled with sand or a similar material in the cold state. The PLAN reports /5-3/ are based on an alternative canister model, consisting of a copper cylinder which is cast with lead after the fuel assemblies have been placed in it and then sealed. Figures 5-2 and 5-3 show the composite canister and the lead-filled canister, respectively. Note the slightly different outside diameters of the two canister alternatives. The volume of the canisters is equivalent to just under 1.5 m^3/tonne of uranium.

Encapsulation of spent fuel is also being studied in some other countries, e.g. the USA, Canada, Finland, Spain and Germany. Steel is the alternative being given primary consideration in the USA, Spain and Germany,

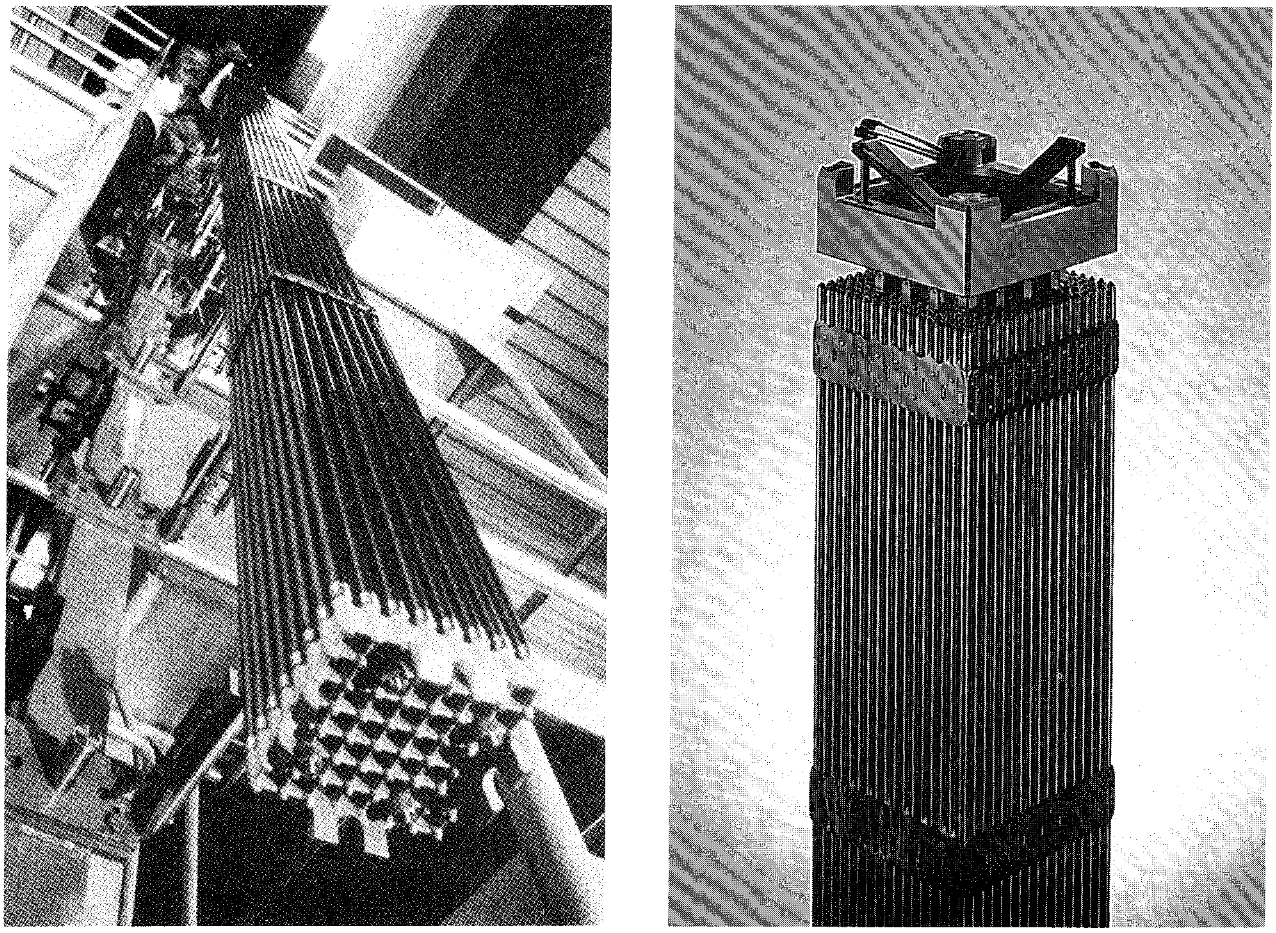


Figure 5-1. Fuel assemblies of BWR type (left) and PWR type (right).

titanium is being studied in Canada, while TVO in Finland has chosen the same canister design as SKB. In Germany, a pilot facility for encapsulation of spent fuel is being built in Gorleben. Prior to encapsulation, the fuel assemblies there will be dismantled in order to obtain a higher degree of filling in the canister. The possibility of cutting the rods to permit shorter canisters is also being studied /5-4/.

5.3 REPROCESSING

When it is removed from the reactor, the spent fuel contains uranium, plutonium, fission products and other transuranics. Uranium and plutonium can, if separated, be re-used for new reactor fuel, while the remaining materials (about 4% of the fuel) constitute waste. Such a separation takes place in connection with reprocessing of the spent fuel. Through re-use of reprocessed uranium and plutonium in light-water reactors, the need for natural uranium can theoretically be reduced by about 30%. It should be pointed out in this context that the residual content of U-235 in high-burnup fuel from Swedish reactors is only about 0.6%. Enrichment of such uranium is not profitable.

Reprocessing is done on a commercial scale today at La Hague and Marcoule in France and at Sellafield in Great Britain. Smaller prototype plants exist in Japan and India. Reprocessing was previously done in the USA and Germany as well, but these plants are now closed.

A reprocessing plant is a very large chemical factory. The UP2 and UP3 reprocessing plants at La Hague have a combined capacity of 1,200 t/y, enough to keep 50 reactors supplied with fuel, and together occupy an area of about 300 ha.

In brief, reprocessing consists of the following steps:

- Reception and interim storage of fuel.
- Chopping of the fuel and dissolution in strong acid.
- Chemical separation of uranium, plutonium and waste in a multi-stage liquid-liquid extraction.
- Purification of the uranium and plutonium streams and conversion to oxide.
- Treatment of different waste streams.

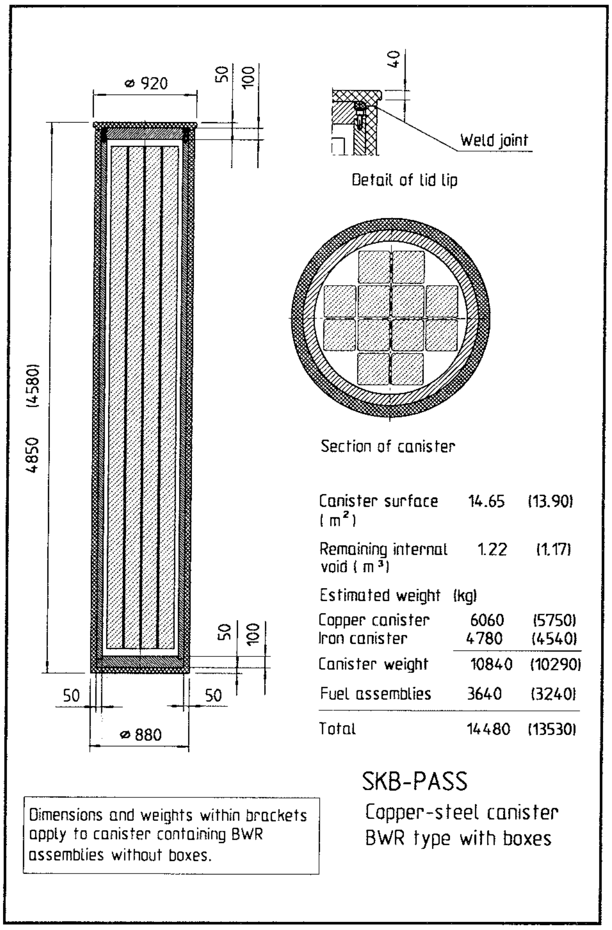


Figure 5-2. Composite canister.

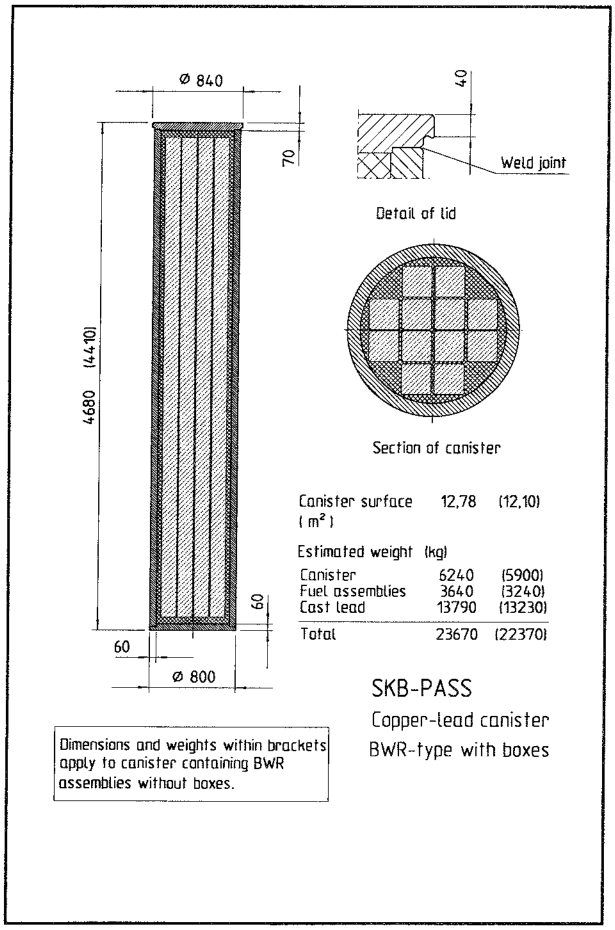


Figure 5-3. Lead-filled canister.

Uranium and plutonium

After reprocessing, uranium and plutonium exist in such a form that they could be re-used in fuel fabrication.

The uranium typically has a residual content of 0.4 - 0.9% U-235. Before it is re-used it therefore needs to be enriched. In view of the fact that reprocessed uranium also contains small quantities of other uranium isotopes (gamma-emitting isotopes and neutron-absorbing isotopes), enrichment normally takes place in a separate part of the enrichment plant. With repeated recycling of uranium, these isotopes are further accumulated, which is why only one recycling is planned for the present.

Plutonium is re-used in mixed oxide or MOX fuel, whereby plutonium oxide is mixed with uranium oxide and sintered to pellets with a homogeneous mix. Fabrication of MOX fuel takes place in completely closed systems in special factories. MOX fuel is only recycled on a commercial scale today in Germany. Tests with MOX fuel have also been performed in other countries and commercial use is planned within a few years. Plutonium also deteriorates on repeated recyclings, which is why only a few recyclings are planned for the present.

Waste from reprocessing

Different types of waste are obtained during reprocessing. The most important are:

- High-level vitrified waste (about 0.1 m^3/t U). This waste contains most of the fission products and transuranics.
- Cladding waste (about 0.7 m^3/t U), which consists of the fuel cladding and the other structural parts of the fuel that remain after dissolution of the fuel. The cladding waste can also contain small quantities of undissolved fuel. The cladding waste is normally embedded in concrete.
- Low- and intermediate-level waste sludge embedded in bitumen or concrete (about 0.5 m^3/t U).
- Different types of solid wastes embedded in concrete (about 5 m^3/t U).

A large portion of this waste contains so much long-lived activity that it requires final disposal at great depth. Such a repository for long-lived low- and intermediate-level waste is planned at Sellafield in England /5-5/. A repository for similar waste from military facilities has been built at Carlsbad in New Mexico, USA /5-6/. Moreover, the high-level vitrified waste generates heat so that it is subject to requirements similar to those made on spent fuel.

Reprocessing in the world

Reprocessing is the predominant strategy in many countries in the world, in particular France, Great Britain, Japan and Russia, where large reprocessing plants have been built or are planned. Germany also previously had plans for its own reprocessing plant, but these were abandoned a few years ago in favour of continued reprocessing abroad. Other countries such as Switzerland, Belgium and the Netherlands are reprocessing their fuel in France and Great Britain, and several eastern European countries have sent fuel to the former Soviet Union for reprocessing.

A waning interest in reprocessing has been evident in recent years, however, due in part to higher costs, low natural uranium prices and difficulties getting large-scale reprocessing going. The increased burnup of the fuel has also reduced the value of the recovered uranium and plutonium. For this reason, direct disposal is also being studied in several countries as an alternative to reprocessing, for example in Germany.

Reprocessing of Swedish fuel

Until the end of the 1970s, the policy was that Swedish fuel should be reprocessed. Thus, contracts were signed with BNFL in Great Britain and COGEMA in France for reprocessing of specific quantities. Of these, only the contract with BNFL has been exercised, whereby 140 tonnes of fuel from the Oskarshamn station has been sent to Sellafield. It is now being stored there pending reprocessing at the end of the 1990s. According to the contract with BNFL, the waste from reprocessing of this fuel will be kept in Great Britain for final disposal. The right to exercise the contract with COGEMA has been re-assigned to other customers, and Sweden has no plans to send any fuel to La Hague. A small quantity was sent at the beginning of the 1980s, but it has been taken over by German power utilities in exchange for used Mox fuel.

Thus, the Swedish power utilities only plan to reprocess the aforementioned 140 tonnes. The remaining fuel, about 7,800 tonnes, will be disposed of directly.

5.4 PARTITION AND TRANSMUTATION

A further development of reprocessing and recycling of uranium and plutonium is what is usually known as partitioning and transmutation (P-T). Partitioning entails a more complete separation (than in today's customary reprocessing) of long-lived radionuclides from the stable or short-lived nuclides in the spent fuel. The long-lived nuclides are then converted – transmuted – to more short-lived or stable nuclides by neutron bombardment. Ideally, this would eliminate the possibility that the long-lived nuclides would pose a long-term radiological hazard.

The neutron source can either be a nuclear reactor or an accelerator-driven spallation source. In order to achieve high efficiency in the transmutation process, very high neutron fluxes are required. This limits the options to fast reactors (i.e. reactors that are driven by fast neutrons in contrast to thermal neutrons, such as in an LWR) or spallation sources.

Studies of partition and transmutation have been conducted for the past twenty years. During the '70s and early '80s, interest was centred on the transmutation of "minor actinides": Np-237 and others. The general conclusion of these studies was that since the long-term hazard posed by these nuclides is small, the incentive to reduce it further by means of transmutation is weak.

In recent years, interest in transmutation has once again increased due to the claim that certain factors that have influenced the previous conclusions have now changed. This, it is felt, warrants a re-evaluation of transmutation. Several such factors are:

- the radiotoxicity of the actinides has been re-evaluated in accordance with the ICRP's new guidelines;
- through the use of new accelerator technology it is possible to create much more powerful spallation sources that were previously thought possible;
- it is believed that not only actinides but also long-lived fission products can be transmuted with the new technology;
- new technical developments of separation (reprocessing) technology and progress within robotics are increasing the possibilities of efficient separation.

A number of countries are conducting major research programmes within the field of partition and transmutation. Japan, France and the CIS (Russia) have presented national programmes. In the USA, several groups at the large government research laboratories are involved in major projects. Other countries are conducting smaller-scale follow-up efforts.

Research on the separation of actinides has been conducted for a couple of decades. Both water-based processes and pyrometallurgical processes are being studied. Knowledge of the former is very extensive, while pyrometallurgy for this application is in an early stage of development. The problem is that an efficient transmutation of the long-lived nuclides requires repeated irradiations with intervening separation. This requires very high efficiency in each separation step in order to approach the ideal result of substantially reducing the long-term risk. A great deal of work remains to be done before industrial-scale processes are available which can achieve this goal. The French, who have great experience of reprocessing and a large research programme in this field, have indicated that a realistically close goal would perhaps be a reduction of these materials by a factor of 10 compared with their concentration in vitrified waste from present-day reprocessing. Eventually, with new technology, a reduction factor of 100 is believed possible.

As already mentioned, research on transmutation is being mainly concentrated on two types of neutron sources: fast reactors and spallation sources. Of these, there is no doubt that the fast reactor has come considerably further in development. For transmutation, it is not the breeding features of this type of reactor that are of interest, but the high neutron flux. However, it should be possible to modify the reactor for transmutation without too much difficulty. The French are discussing the possibility of converting the Super-Phenix breeder reactor to a plutonium and neptunium burner within a few years. The development of spallation sources is still at a very early stage. Estimates indicate that 15–20 years of development work at very high costs will be necessary to build a prototype for transmutation of existing military waste in the USA. This type of waste requires a "simpler" technology than the waste from reprocessed nuclear reactor fuel.

The technical principles of partitioning and transmutation can be regarded as being established, with certain reservations. However, it is difficult with the present-day scenario for nuclear fuel supply to see any safety (or cost) advantages. Application of this technology (if it is developed for use on an industrial scale) will not eliminate the need for final disposal of radioactive waste. Small residual quantities will always remain. This view is shared by waste management organizations in all the nuclear power countries, even where relatively large resources are being devoted to partition and transmutation. At present there is no incentive to invest in this method for the management of spent nuclear fuel from light-water reactors operated with a once-through cycle, i.e. without reprocessing of the fuel. For Sweden's part, a small follow-up research effort linked to ongoing international development efforts is considered sufficient, see section 14.1.

A more detailed account of partitioning and transmutation is provided in /5-7/.

6 METHODS FOR DISPOSAL OF LONG-LIVED RADIOACTIVE WASTE

6.1 OVERVIEW

The definition of long-lived waste given in chapter 3 is waste that should be kept isolated for a much longer period of time than a few hundred years. The waste contains long-lived radionuclides, for example transuranics such as plutonium and other nuclides with half-lives of thousands of years or more. The spent nuclear fuel from nuclear power plants is the most important of these long-lived wastes and also contains most of the radioactive materials that are formed in nuclear power production – more than 99%. The spent nuclear fuel and the most active waste from reprocessing comprise high-level waste – besides being long-lived, it emits so much radiation that the heat it evolves must be cooled away or removed.

Different methods have been developed for disposing of high-level waste. For supervised interim storage, either wet storage – storage under water in pools – or dry storage – storage in air-cooled containers – is used. No technical time limit on how long such supervised storage can be continued has been proved to exist. Spent fuel assemblies have been stored in water since the late 1950s without any change in the integrity of the fuel having been observed. It is probably possible to store the waste in this manner for hundreds of years if the facilities are properly managed. Since substantial quantities of radioactive materials will remain for tens of thousands of years, however, disposal methods are being developed that do not require supervision or maintenance – so-called "final disposal".

The principle of final disposal is that it shall be arranged in such a manner that the waste is kept isolated in a safe manner, without requiring supervision and maintenance. In recent years the wish has also been expressed in Sweden that the final repository should be designed so that the waste can be accessed and remedial actions taken, if this is decided to be expedient in the future. The expressions used in this context are "retrievability" or "reparability".

Final disposal of high-level or other long-lived waste has not yet been carried out anywhere in the world. The methods that have been developed all aim at some type of deep geological disposal, i.e. emplacement sufficiently deep down in the earth to ensure the repository will not be affected by the changes that take place on the surface of the earth during the time the waste has to be kept isolated. Different countries are planning for deep disposal in different types of rock, depending on the special conditions existing in each country.

The different disposal concepts developed in several countries are presented briefly in the following sections. An overview of the alternatives that have been studied in Sweden is also provided. During the past two years, SKB has studied several interesting final disposal alternatives for Swedish conditions in the PASS Project. The conclusions from this project serve as a basis for the plan for future measures that is presented in greater detail in chapter 7 and the following.

6.2 INTERIM STORAGE OF SPENT NUCLEAR FUEL

Up to the mid-1970s, interim storage of spent nuclear fuel in water pools was the generally accepted storage concept. Dry interim storage was introduced in the 1960s and developed in the 1970s in such a manner that dry storage came to be seen as an interesting alternative to wet storage both technically and economically.

A number of large-scale facilities have been built for wet storage during the 1980s with capacities varying from 600 to 10 000 tonnes of spent nuclear fuel, for example in England (Sellafield), France (La Hague) and Sweden (CLAB), see Figures 6-1 and 6-2.

As far as dry storage is concerned, a number of different concepts currently exist: metal containers, concrete silo, dry wells and vaults. Dry storage has been introduced industrially in the USA, Canada, Germany and Switzerland, among other countries.

In 1991, the total world capacity for interim storage of spent nuclear fuel was about 40 700 tonnes. This figure does not include fuel pools in direct connection with the reactors. Of this capacity, about 38 200 tonnes is wet storage and 2 500 tonnes is dry storage. Table 6-1 shows how this total capacity is distributed among different countries.

6.2.1 Wet storage of Zircaloy-clad UO_2 fuel

Zircaloy-clad uranium dioxide fuel has been stored in water pools for decades. Based on this, the following conclusions and observations have been compiled within the framework of the IAEA's research programme BEFAST-II /6-11/:

Figure 6-1. Reception building in central interim storage facility for spent nuclear fuel, CLAB.

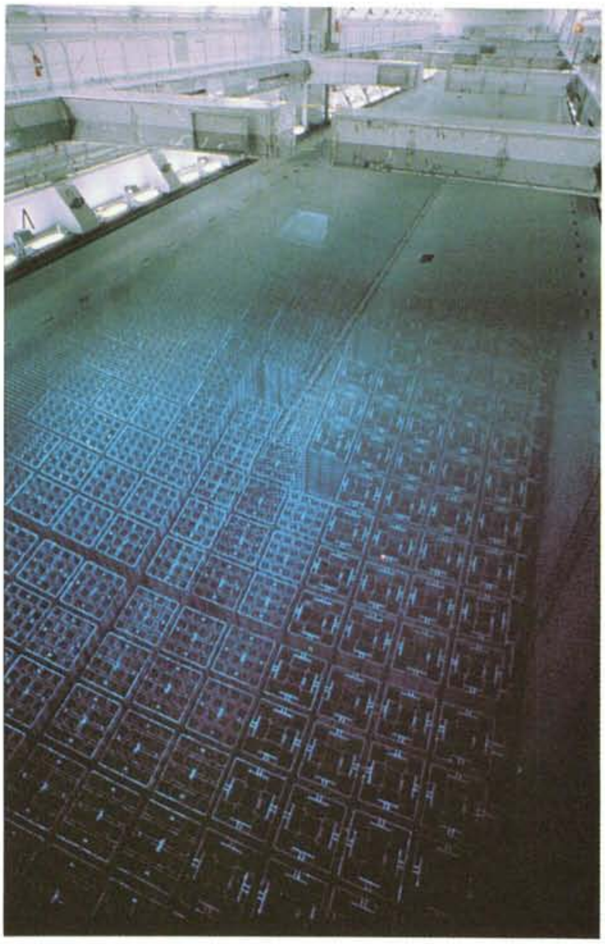


Figure 6-2. Pools for wet storage of spent nuclear fuel in CLAB.

Table 6-1. Capacities for interim storage of spent fuel expressed in tonnes. (Data provided by IAEA.)

	In service	Under construction	Planned
Argentina	365		
Belgium			
Bulgaria	600		
Canada	475	200	12 600
China			500
Czechoslovakia	600		1 800
Finland	1 270		
France	11 000		
Germany	2 150		700
India	523		
Japan	140		3 000
Republic of Korea			3 000
Russia	7 400	1 900	
Spain			5 500
Sweden	3 000	2 000	3 000
United Kingdom	10 350		
Ukraine	1 900		
USA	900		15 000
Total	**40 673**	**4 100**	**46 100**

1. Long-term storage in water of Zircaloy-clad spent nuclear fuel with no loss of integrity of the fuel or the cladding is a fully acceptable and suitable method, even in cases where the cladding has through-wall defects that derive from operation.
2. It is of crucial importance for preventing degradation of the fuel that the water chemistry in the storage pools be kept within specified limits.
3. It is necessary that the specified water temperature be maintained in order to maintain the integrity of the fuel, minimize the concentration of activity in the pool water and preserve the integrity of the pools.
4. In the case of storage for very long periods of time (more than 50 years), the durability of certain pool liner materials, such as non-stabilized stainless steel and epoxies, may require further investigation.
5. New advanced monitoring methods are being introduced as complements to existing inspection methods for fuel in pools and hot cells. Examples of such methods are gamma scanning, neutron interrogation and ultrasonic inspection.

Wet storage remains the main alternative for all countries that have participated in BEFAST II, although most are also following the development of different dry storage methods.

In the case of storage for very long periods of time (more than 50 years), BEFAST II believes that further studies are needed regarding both fuel cladding and pool components.

6.2.2 Dry storage of Zircaloy-clad UO_2 fuel

BEFAST II summarizes and draws the following conclusions from available experience of dry storage of Zircaloy-clad UO_2 fuel:

Dry storage is licensed in Germany and the USA for storage in an inert atmosphere and in Canada for storage in air. The maximum licensed cladding temperature is 380°C in the USA and 410°C in Germany. Different options have been chosen depending on the circumstances: high temperatures require inert gas (e.g. Germany and the USA), while low temperatures permit storage in air.

No degradation of the stored fuel has been observed in any of the applied cases. Krypton emission has been noted in a few cases, which indicates that a few individual rods may be damaged.

Dry storage is being used to an increasing extent as a complement to wet storage of Zircaloy-clad oxide fuel. It appears to be possible to achieve as long storage periods as for wet storage.

6.3 POTENTIAL FINAL DISPOSAL METHODS

Final disposal entails isolation of the waste without any requirements on monitoring or inspection and in a manner that makes it difficult or impossible to get at the waste, at least inadvertently. For spent nuclear fuel, a period of supervised storage comprises an inescapable link in the handling chain. It can be extended over a very long time without any safety-related or technical problems, as long as the present-day level of technology and social stability is preserved.

The following principles for final disposal of spent nuclear fuel and other long-lived waste have figured in the international discussion:

- Final disposal in deep geological formations.
- Final disposal beneath the seabed in deep sea sediments.
- Final disposal in or under major continental ice sheets (e.g. Antarctica).
- Launching into outer space.
- Supervised storage for an indefinite period of time.

The methods for final disposal are in principle independent of whether the fuel has been reprocessed or is disposed of without treatment. Final storage underneath inland ice sheets is hardly a feasible option for Sweden, nor is it attracting much interest in other countries. Launching into space would certainly put the waste beyond reach, providing safety in the launching procedure could be guaranteed. But it is scarcely economically feasible for Sweden. Supervised storage differs fundamentally from interim storage, which was described in section 6.2. It can, of course, also be arranged in other ways than those tried thus far, for example as an initial phase in deep geological disposal. All countries who are studying final disposal have chosen isolation in various deep geological formations as the main alternative.

In connection with the management of spent nuclear fuel, safeguarding of fissionable materials must be taken into consideration in the long run as well. This means that handling and final disposal must be designed in such a manner that secret or covert retrieval is improbable.

The following fundamental requirements on a final repository have been discussed in Sweden in recent years:

"A final repository should be designed so that it renders surveillance and inspection unnecessary for safe function, without rendering future intervention and remedial actions impossible if new knowledge should show that the repository is unsuitable for one reason or another."

SKB believes that this fundamental requirement can be met by a deep repository in rock designed according to the basic principles that have long been studied in the Swedish work.

6.4 DEEP GEOLOGICAL DISPOSAL

6.4.1 Principles

A final, unsupervised disposal must make allowance for a number of fundamental phenomena that impose special demands on the disposal.

The time aspect: The spent nuclear fuel contains more than 99% of all radioactive materials that are formed in a nuclear reactor. Most of these are short-lived and decay during interim storage at the power plant or in a central facility. Some are a little more long-lived and decay in a few hundred years. After this time, heat generation in the waste is virtually negligible. A few nuclides have a very long life – thousands of year or more – and remain in the fuel for a very long time. After about 100 000 years, the radiotoxicity of spent nuclear fuel has reached a level that is comparable with a rich uranium ore. The goal is therefore to isolate the waste for such a long period of time.

Changes on the earth's surface: These occur naturally and can also be caused by man. Changes take place in a relatively short time perspective. One topical issue, for example, is acid rain. Another is man's use of natural resources, for example power station dams and mining activities. Changes of the climate with warmer and colder weather and changed amounts of precipitation take place both in short and long time perspectives. A shift in climate towards a glaciation period is expected to have commenced within 5 000 – 10 000 years. An inland ice sheet can increase the hydrostatic pressure under ground, scrape off present-day loose sand and soil layers on the rock surface, and erode and grind down the rock surface as it moves. The ice cover blocks the possible transport pathway for radionuclides via agriculture to man. An ice age period extends over some 100 000 years before the climate once again enters a warmer phase.

Bedrock movements: The ground surface in large parts of Scandinavia is moving upward at a rate of a few millimetres per year. When the pressure of the inland ice sheet is added on top, the movement will be reversed but still slow. These movements are a far cry from the violent earthquakes and volcanic activity that occur elsewhere on the earth.

The changes that will thus occur or can be expected to occur on or near the surface of the earth are so substantial that a safe isolation of the waste for the necessary period of time cannot be arranged there. If, however, we go a few hundred metres or more down in the rock, changes of any significance take place on an entirely different time scale. The changes that can be extrapolated from known geological data into the future and the external forces that can be expected to occur will not change the properties of the bedrock at a depth of several hundred metres over a very long time (millions of years). Thus, good prospects exist at many places for finding a sufficiently stable environment to isolate the waste for 100 000 years.

As is discussed elsewhere in this report, the bedrock possesses natural properties that contribute to isolation of the waste. These properties vary depending on the type of rock. In some formations, such as salt, there is no mobile groundwater. Moreover, the salt is often overlain by several other sedimentary formations – several natural barriers are present and the rock's isolation is very effective. In other formations, for example granitic crystalline rock, the bedrock is saturated with water and riddled with fractures in which the water can move. In such cases, engineered barriers are used to isolate the waste from the mobile groundwater.

The basic principle for geological final disposal is that the waste shall be surrounded by several partially or completely independent barriers.

The principle of multiple barriers in deep geological repositories is completely dominant in the world today, regardless of whether the repositories are intended for the disposal of high-level waste from reprocessing or of spent nuclear fuel. The differences between different countries stem from differences in the geological medium and in the design of the engineered barriers.

6.4.2 Methods in different countries

No country has yet begun its final disposal of long-lived high-level nuclear waste. Different geological formations are currently being studied in different countries:

- Salt.
- Granite (crystalline rock).
- Clay.
- Tuff (sedimented volcanic ash).
- Other sedimentary rock types.

A common denominator for all of these formations is that they are very old – tens of millions of years or more – and that changes proceed very slowly. Many countries have large deposits of salt, granite or clay. Since salt and clay are plastic materials, the engineered barriers are designed differently in these media, compared with granite. Moreover, salt provides an entirely different chemical environment than groundwater in granite, warranting different choices of canister material. The chamber for the encapsulated waste is made larger, in some cases much larger, than the canister and the void volume is backfilled. In salt and clay, backfilling is planned to be done with excavated material from the site. In granitic rock, the main material being considered for backfill nearest the canister is swelling, hydraulically impermeable bentonite clay. Only in one

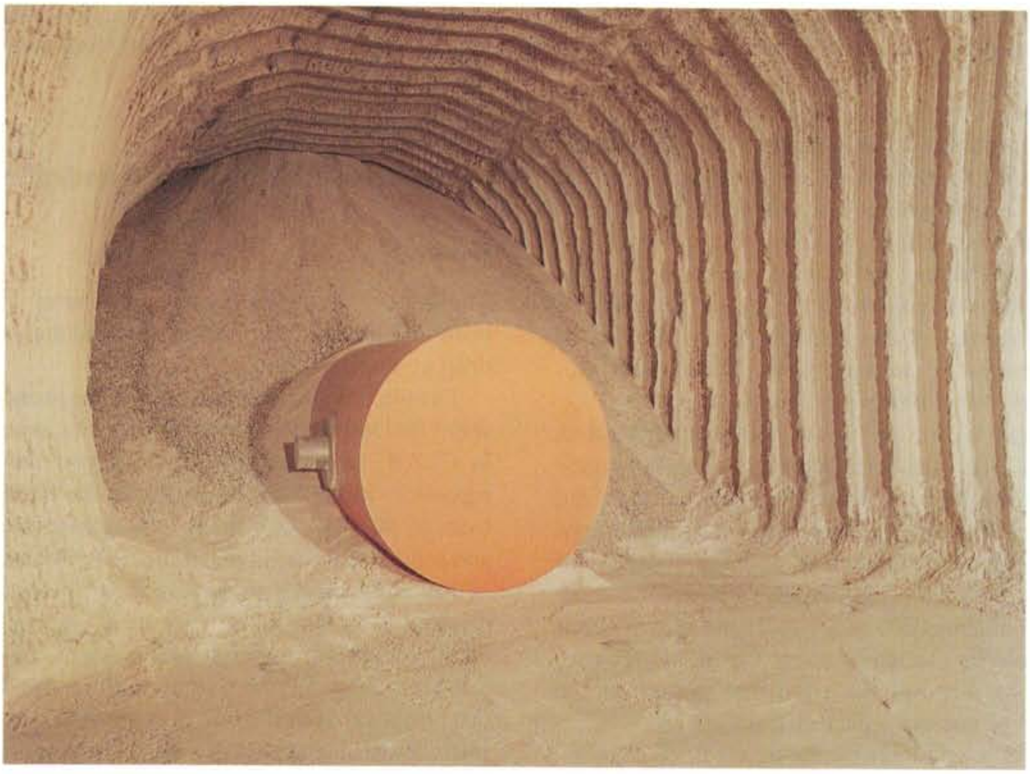


Figure 6-3. Demonstration of horizontal emplacement of a German Custos canister in a salt formation.

case – at Yucca Mountain in the USA – is deep disposal in tuff, above the groundwater table, being considered. It is assumed that backfilling there can be done with crushed rock from the excavation work.

The quantity of waste per package (canister) varies in different concepts. The key factor is the thermal load that the repository environment can withstand (bentonite is deemed to be able to withstand about 100°C without any change in its essential properties) and the weight which the transport units etc. can handle. The German Custos canister (formerly the Pollux canister) is the heaviest that has been proposed. It weighs about 65 tonnes, including the built-in radiation shielding, which is planned to be disposed of together with the canister /6-1/.

The main line in the German concept for final disposal of high-level waste is emplacement in salt formations. A shaft is currently being sunk in a salt dome in Gorleben as part of a detailed characterization programme to determine whether the site is suitable for a final repository, see Figure 6-3. Since both reprocessing with subsequent vitrification of the liquid high-level waste and direct disposal of spent nuclear fuel are included as constituents in the German nuclear waste programme, both of these types of waste disposal are being studied.

Clay strata as a media for the final disposal of high-level waste is the main alternative in the Belgian programme. Investigations in an underground rock laboratory are being conducted in the Boom formation at Mol. Different parts of a hypothetical repository are being studied. Since the Belgians intend to reprocess the spent nuclear fuel, vitrified high-level waste is the waste form that will be deposited /6-2/.

The main alternative being pursued at Yucca Mountain in the USA is to build a final repository in tuff. As mentioned above, the waste is intended to be emplaced in non-water-saturated (dry) rock. The programme is focused primarily on direct disposal of spent nuclear fuel /6-3/.

Crystalline water-saturated rock as a medium for final disposal is the main alternative in several countries, e.g. Switzerland, Canada, Finland and Sweden. The design of the repositories and the choice of materials varies from one country to the next, but the systems always consist of multiple, independent barriers. France and Russia also have final disposal in crystalline rock as one of the most interesting disposal alternatives.

6.4.3 Disposal beneath the seabed

Since 1976 the OECD/NEA has been working with the possibility of disposing of radioactive waste underneath the seabed within a working group called the "Seabed Working Group", with participants from 11 countries. The group is furnishing the regulatory authorities and concerned organizations in the member states with scientific and technical information to enable a judgement to be made in each country of the long-term safety and technical feasibility of the disposal alternative.

The concept of sub-seabed disposal entails a multi-barrier system with a suitable waste form (glass and/or corrosion-resistant canisters). Sediment formations at great depth in the oceans are recommended so that radionuclides will be retained after the waste canisters have disintegrated due, for example, to corrosion. The site should also be selected with a view towards the chemical properties of the sediment layers, the risk of erosion damages to the sediments, seismic and volcanic activity, mineral deposits etc. /6-4/

In practical terms, the waste can be emplaced in "penetrators", i.e. waste canisters in the form of projectiles that are released from the ocean surface and penetrate under their own force deep into the bottom sediments. The penetrators need to have a weight of several tonnes. If an emplacement distance between the canisters of about 200 m is chosen, an area of about 500 km would be needed to cover the global requirement of waste disposal for a ten-year period, at the current level of nuclear power production.

Disposal of waste beneath the ocean floor can also be accomplished by means of deep-sea drilling with existing technology. With this method, the waste could be disposed of in 800 m deep boreholes beneath the seabed in which waste canisters are stacked up to 300 m from the bottom, after which the borehole is sealed.

SKB's work is focused on land-based storage in crystalline bedrock. Certain limited efforts have, however, been concentrated on disposal beneath the bottom of the Baltic Sea, but still in crystalline rock. Such a storage alternative exhibits both advantages and disadvantages compared with the shore-based repository alternative /6-5/.

An important advantage with a repository in rock underneath the sea is that the groundwater is only affected by very small gradients, so that the water flux through the repository will be very low. If the repository is in addition surrounded by groundwater with increasing salinity with depth, the water flux will be even less.

Sedimentary rocks overlying the crystalline basement rock are encountered along the coasts of Sweden, especially in the southeastern parts. If the repository is situated underneath such sandstone sediments with higher hydraulic conductivity than the basement rock, an additional positive barrier effect is obtained due to the fact that the high-permeable sandstone acts as a hydraulic barrier that inhibits vertical groundwater transport.

Other advantages with a repository in the subsea bedrock is that it would most likely reduce landowner conflicts as well as conflicts of interest regarding the groundwater. Well scenarios could possibly be omitted from safety assessments.

The disadvantages include the fact that all geological characterization would be more difficult and more costly to carry out for subsea rock. The costs of pre-investigations, detailed characterization and construction would probably be higher.

6.4.4 System studied in Sweden

General

The general guidelines given in chapter 2 are to be observed in the designing of the Swedish system for deep geological disposal.

The planning to date is based on the premise that the spent fuel will, after interim storage for about 40 years in CLAB, be encapsulated and deposited in a deep repository in the Swedish bedrock. The interim storage time of 40 years is chosen to simplify final disposal and reduce the burden on future generations.

The fundamental prerequisite for the handling and disposal systems is that they meet society's requirements on safety, aimed at protecting humans and nature from harmful effects now and in the future. The final disposal system shall provide safe isolation for a sufficiently long period of time. This isolation shall not be dependent on inspection or improvements; it shall be possible to seal and abandon the deep repository.

However, the final disposal scheme shall not unnecessarily prevent a future generation from accessing the waste in order to modify the final repository or to treat the waste in some other way. At the same time, this freedom of choice must not jeopardize the long-term safety of the selected design.

In the long-range perspective, safety is based on the deep repository's isolating capacity. The technical solutions that have been studied are based on the following fundamental principles:

- Final disposal in Swedish crystalline bedrock.
- The multibarrier principle with mutually independent natural and engineered barriers.
- Natural materials in the engineered barriers.
- Limited temperature, radiation dose and other impact on the rock.

In 1983, SKB presented the KBS-3 report, which described a system for final disposal of the spent nuclear fuel in Swedish bedrock. After extensive circulation of the report for review and commentary by Swedish and foreign experts, the Government declared in 1984 that "it has been found that the method in its entirety can be approved with regard to safety and radiation protection."

KBS-3

KBS-3 constitutes the reference design in the annual cost calculations that are carried out for all activities for handling and disposal of the waste from nuclear

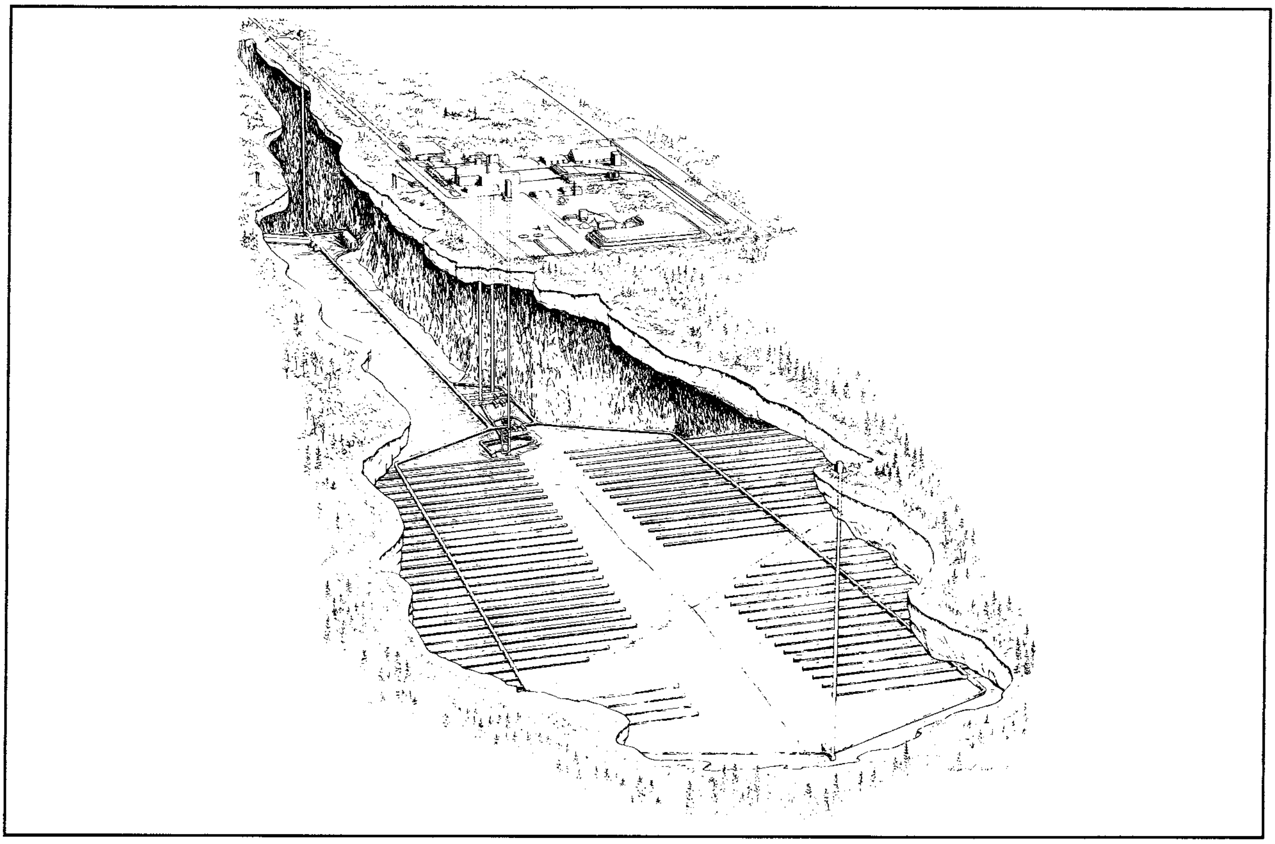


Figure 6-4. KBS-3 design of a deep repository.

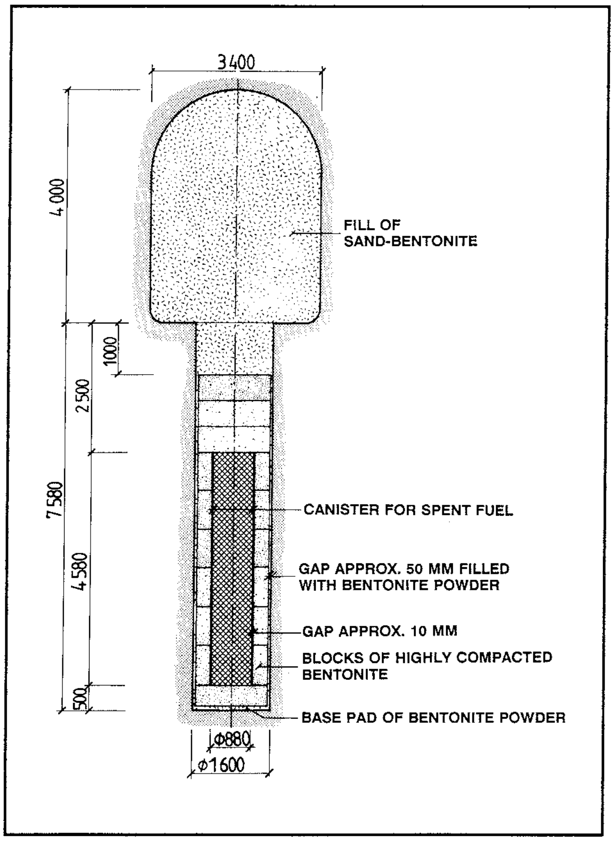


Figure 6-5. KBS-3 design of canister positions.

power and decommissioning of the nuclear facilities. The design of the system in the most recent report /6-6/ is shown in Figures 6-4 and 6-5.

The deep repository consists of a number of parallel tunnels at a depth of about 500 m. They are interconnected by a central tunnel for transportation and communication. The layout of the deposition tunnels is adapted to local conditions in the rock. The rock can be divided into different blocks to avoid major discontinuities in the bedrock.

Vertical holes with room for one canister each are bored from the floor of the tunnels. Copper canisters containing the spent fuel are placed in the boreholes and surrounded by a layer of compacted bentonite clay. After the deposition procedure, the tunnel is backfilled with a mixture of sand and bentonite.

Alternative designs

Since 1984, several different design principles have been studied. They are "WP-Cave," "Deep Boreholes (Very Deep Holes)", "Long Tunnels (Very Long Boreholes)" and "Medium-Long Tunnels (Medium-Long Boreholes)", see Figure 6-6.

The work with "WP-Cave" and "Deep Boreholes" was described in R&D-Programme 89. "WP-Cave" has not been studied further, while Deep Boreholes has been included in the comparison of different alterna-

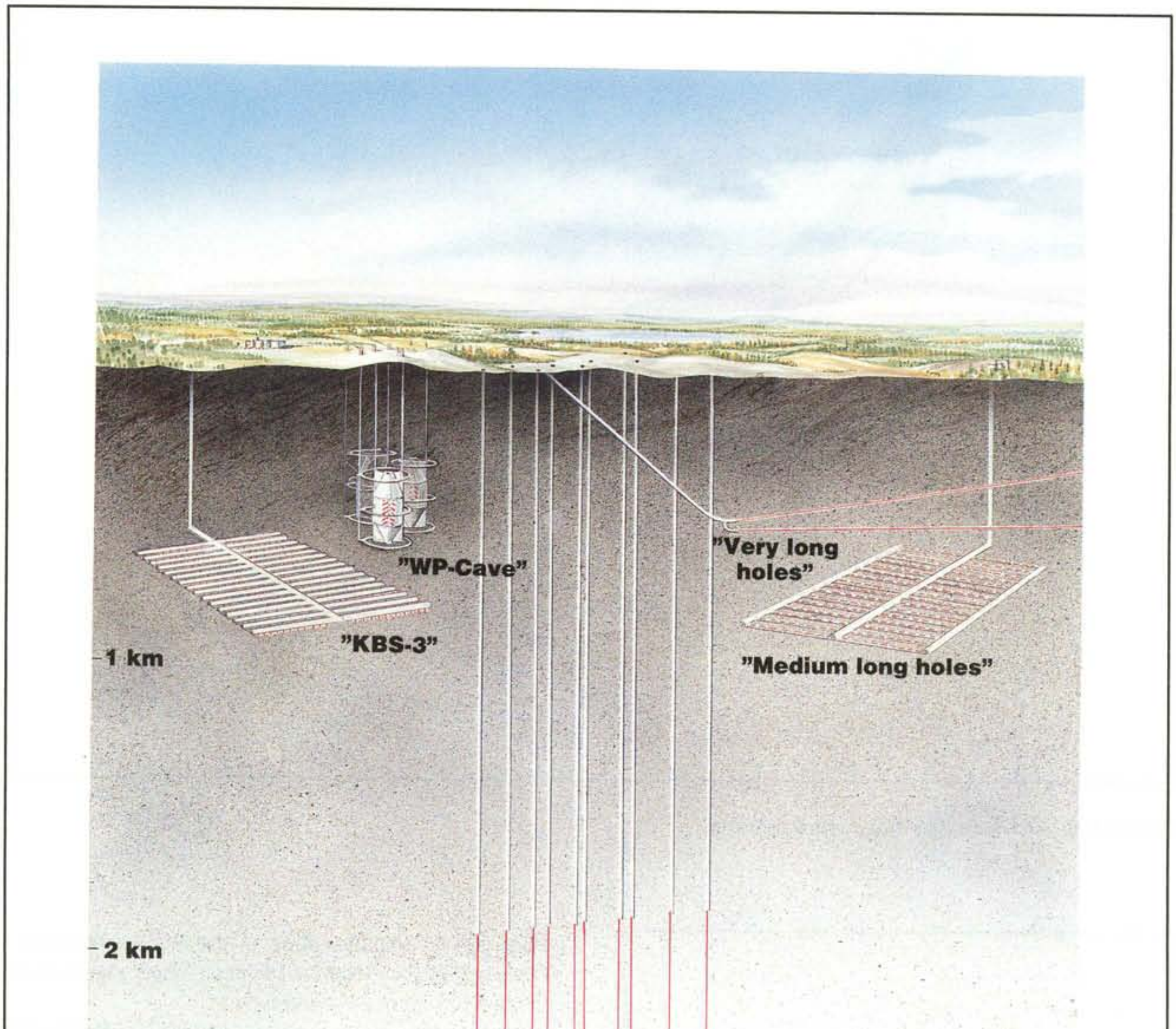
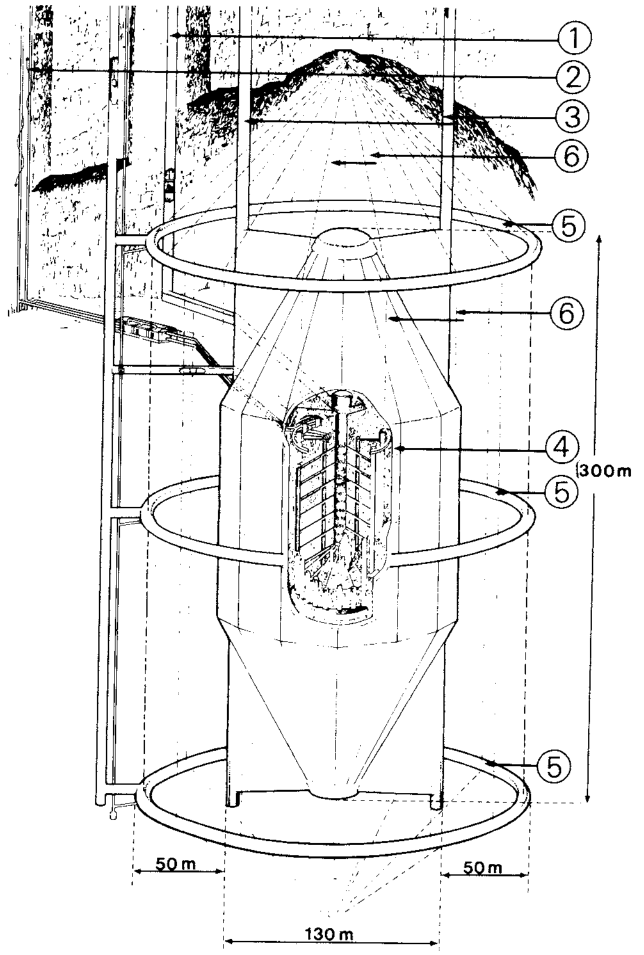


Figure 6-6. Alternative designs of deep repository.



1. *Shaft for waste canisters*
2. *Ventilation shaft*
3. *Shaft for excavation and backfilling of bentonite/sand barrier*
4. *Bentonite/sand barrier with thickness of about 5 m*
5. *Drift in hydraulic cage*
6. *Borehole in hydraulic cage*

Figure 6-7. *Design of WP-Cave.*

tive designs carried out in PASS. A brief description of the four alternative designs is provided below.

WP-Cave

The version of WP-Cave that has mainly been studied /6-7/ is shown in Figure 6-7. The canisters with the spent fuel are placed in the centre of the facility and surrounded by a barrier of rock. Around this a slot is excavated, which is backfilled with a mixture of bentonite and sand. Outside the bentonite/sand barrier, a hydraulic cage is arranged, consisting of horizontal tunnels interconnected by drilled holes.

The analyzed system entails, among other things:

- Canisters of steel.
- A five metre thick bentonite/sand barrier with 10% bentonite in the bottom, 20% in the cylindrical part and 50% in the top part.
- A hydraulic cage that first drains the rock during the construction and deposition phase.
- Size as shown in Figure 6-7.
- 100 years' open cooling prior to closure.

Approximately seven WP-Caves of this size would be needed to accommodate the Swedish waste.

Deep Boreholes – VDH (Very Deep Holes)

The principle is illustrated by Figure 6-8. The canisters with the spent fuel are stacked on top of each other in boreholes between 4 and 2 km depth. The outside diameter of the canisters is determined by the largest hole diameter considered possible to bore to these depths. The canisters are emplaced in a casing that is needed to stabilize the borehole's rock wall and prevent it from caving in. Bentonite clay fills the voids around the canister and the gap between the rock wall and the casing. The canisters are also separated by bentonite plugs /6-8/.

The two uppermost km of the hole are plugged to prevent vertical water transport between the repository and the surface.

The design analyzed by SKB entails:

- Titanium canisters filled with concrete.
- Rod consolidated or non-consolidated packed fuel assemblies in the canisters.
- Non-corroding materials in the casing.
- Boring and casing technique in accordance with experience from deep boring in Gravberg.

Some 30-odd boreholes of this design are needed to hold all spent nuclear fuel from the Swedish programme.

Long Tunnels – VLH (Very Long Holes)

The name comes from the fact that the system was developed with the object of having the deposition area situated far away from the industrial area on the surface. The layout therefore had the appearance shown in Figure 6-9, see /6-9/.

The system is based on the fact that relatively large canisters are emplaced horizontally in rows in bored tunnels. The canisters are surrounded by compacted bentonite blocks. The chosen dimensions are based on an economical optimization of the quantity of fuel per canister to meet the requirements on maximum temperature in the bentonite.

The system analyzed by SKB entails, among other things:

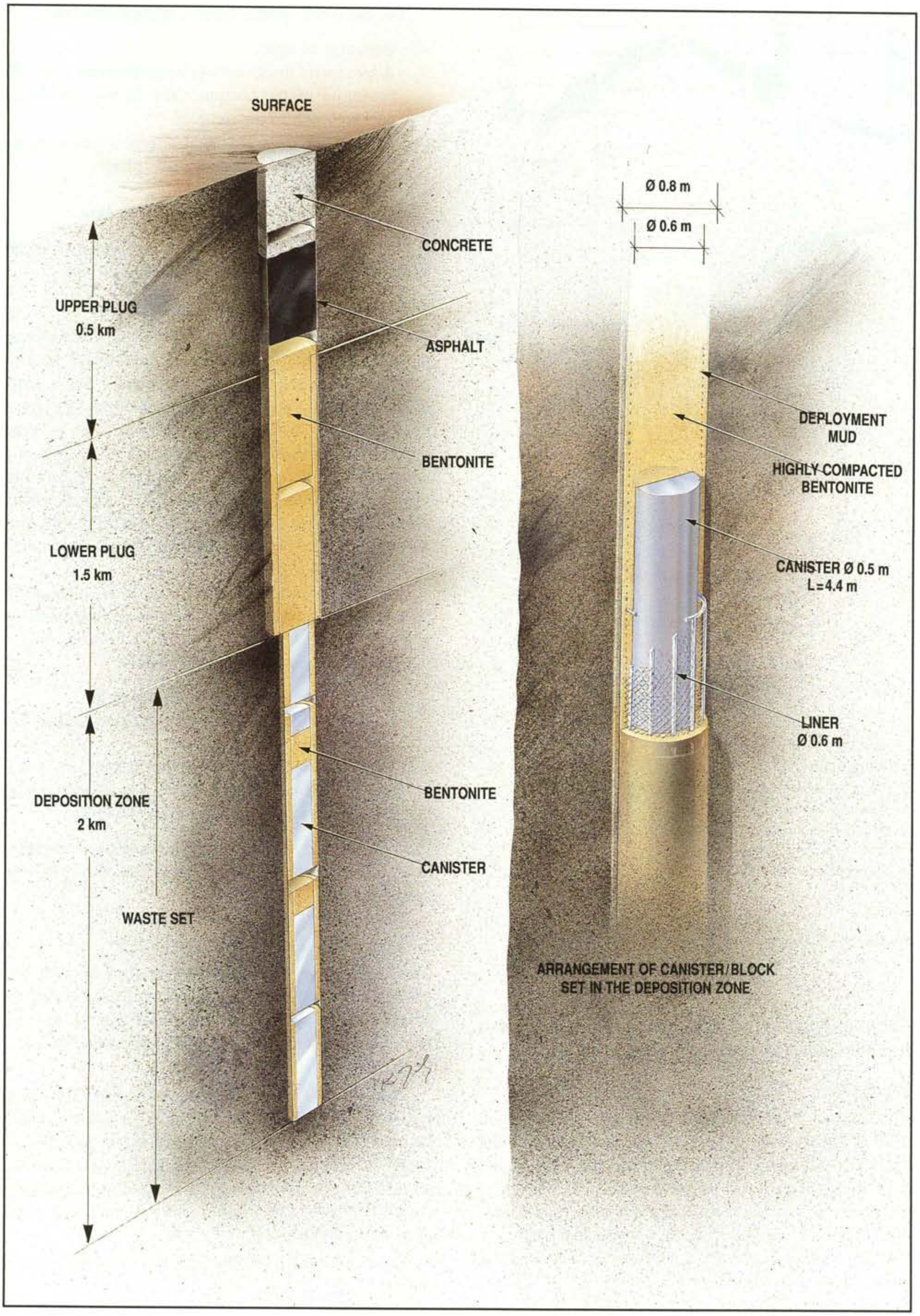


Figure 6-8. *Alternative design "Very Deep Holes", VDH.*

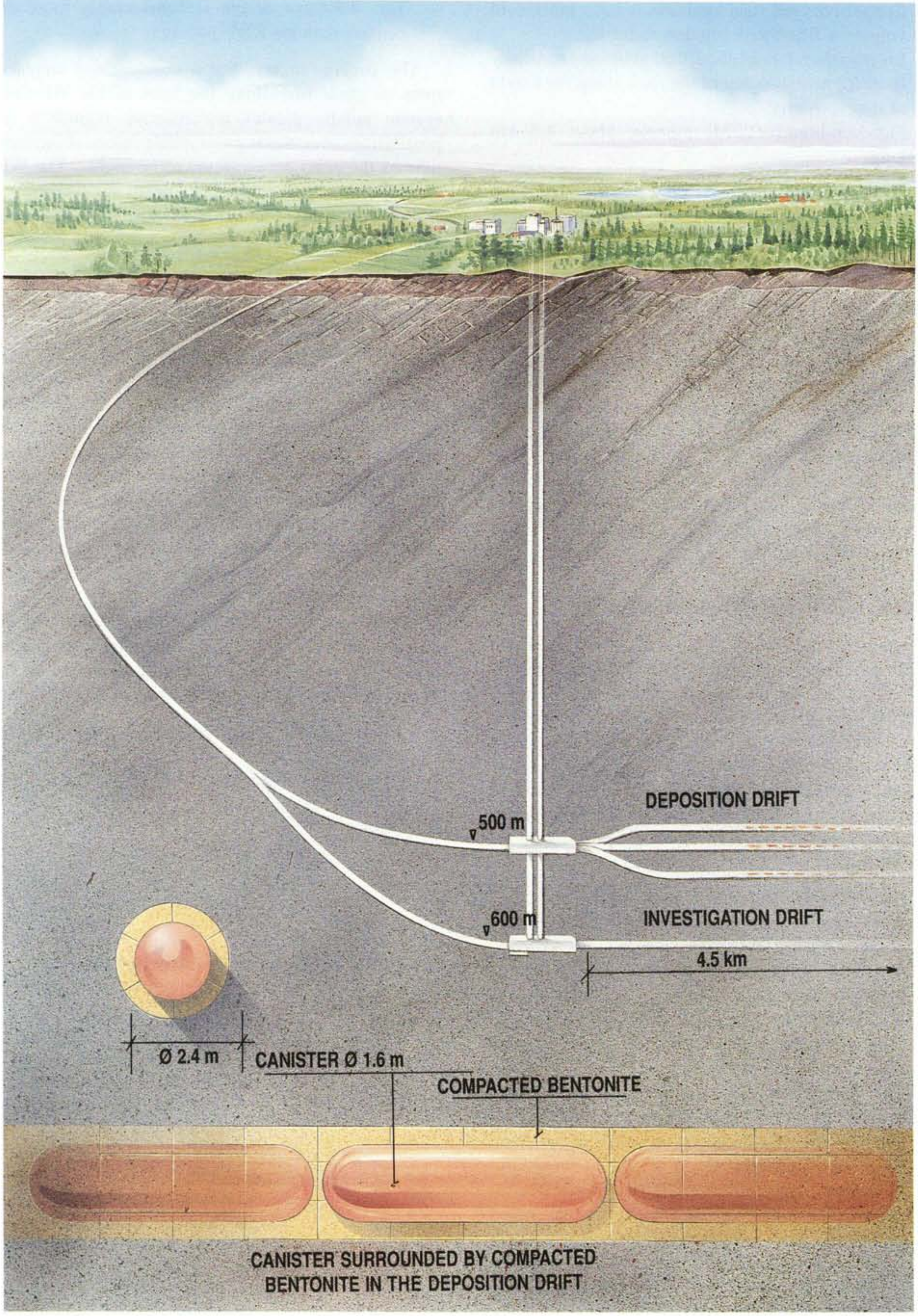


***Figure 6-9**. Alternative design "Long tunnels", or Very Long Holes, VLH.*

- Copper-covered steel canisters, whose inner void volume is filled with suitable material.
- Investigation tunnel about 100 m below repository level for investigation of the rock along the extent of the repository.
- Full-face-bored (TBM) tunnels, about 4.5 km long.

It has, however, also been shown that there are no technical obstacles to making the deposition tunnels shorter and orienting them parallel in the type of rock block specified in KBS-3.

Medium-Long Tunnels – MLH (Medium-Long Holes)

The system features a design with horizontally emplaced canisters of KBS-3 size in rows in bored tunnels, see the schematic drawing in Figure 1-5, chapter 1. The canisters are surrounded by compacted bentonite blocks. The layout of the deposition tunnels conforms to KBS-3. The same conditions apply as for KBS-3 with regard to utilization of rock blocks limited by major discontinuities. The additional side tunnels are required to provide access and permit handling of the deposition equipment.

Comparison of WP-Cave with KBS-3

The comparison between WP-Cave and KBS-3 which SKB carried out in 1988-89 is reported in detail in /6-7/. It was noted that the two systems represent two different principles with regard to the concentration of the spent fuel in the deep repository. WP-Cave is the concentrated system, while KBS-3 is based on a distributed emplacement. This difference was found to entail decisive advantages for the KBS-3 system. The comparison resulted in the following summarizing assessments:

- Both systems can provide adequate safety.
- A utilization of the potential of WP-Cave requires extensive work on model development in areas where the current state of knowledge and available data are incomplete. This is especially true of long-term stability in the groundwater's flow paths in the temperature fields that exist inside and outside the bentonite barrier.
- The higher temperatures in WP-Cave lead to greater uncertainty in the calculated consequences, due to uncertainty in data and dominant processes. Reducing this uncertainty would require many years of international, coordinated research. The situation in this regard is unchanged today from 1989, when the comparison was made.
- Both systems, including the barriers they incorporate, could be built with a commonly practical adaptation of available technology.
- On non-site-specific grounds it was not possible in 1989, nor is it possible today, to say whether it would be easier to find suitable sites for one design or the other.
- The WP-Cave design is considerably more expensive than the KBS-3 design.

The conclusion was that a concentrated emplacement of spent fuel along the lines of the WP-Cave system entails greater uncertainties regarding the possibilities of achieving acceptable safety, at the same time as the costs are expected to be higher. The WP-Cave design was therefore not studied any further.

Conditions today have not changed compared with those on which the above assessment was based in 1989.

Comparison of other alternatives with KBS-3

Since 1989, further development of the VDH, VLH and MLH systems has been pursued (development methods and deposition equipment and technology) in parallel with analysis of the differences between the systems as a basis for a ranking. In addition, alternative canister designs have been considered for each repository system, especially for canisters of the KBS-3 type. A ranking has been done for the canister alternatives as well. The work has been pursued within the framework of a coordinated project, PASS (Project Alternative System Studies) /6-10/.

For the purpose to describe in a systematic way the differences between the identified systems, and the importance each property has been accorded in the overall comparison, the project work was divided into:

- Technology for rock works and deposition (Technology).
- Long-term performance and safety (Long-Term Safety).
- Costs.

Table 6-2. Canister alternatives in PASS.

REPOSITORY SYSTEM	CANISTER VARIANTS
KBS-3	Copper filled with lead Copper/steel – composite canister Copper – hot isostatically pressed Steel filled with lead Steel
VDH	Titanium – self-supporting Titanium – concrete-filled Copper – hot isostatically pressed
VLH	Copper Copper/steel – composite canister Steel
MLH	Same as KBS-3

The different canister alternatives that have been evaluated within the PASS project are shown in Table 6-2.

The alternatives were compared and ranked separately for each area. Then the three rankings were weighed together into one outcome. The principle was used with regard to both canister alternatives and repository systems.

In the first round the canister alternatives were compared. The result was that the composite canister obtained the best ranking for KBS-3, MLH and VLH. On this basis, the composite canister was chosen as the sole alternative in the comparison for repository systems at a depth of about 500 m (KBS-3, VLH, MLH). VDH was combined with the cheapeast alternative – concrete-filled titanium canister.

The comparison of deep repository systems was then carried out with four alternatives, see Table 6-3. The assessment naturally took into account whether the choice of canister in itself entailed restrictions that would not exist with another choice of canister. In no case was there reason to reconsider the original canister choice in the comparison of the repository systems.

Table 6-3. Repository systems compared in PASS with assumed canister alternative.

SYSTEM	CANISTER ALTERNATIVE
KBS-3	Composite canister, Cu/Fe, outside diameter = 0.88 m
Very deep holes, VDH	Titanium/concrete canister, outside diameter = 0.5 m
Very long tunnels (holes), VLH	Composite canister, Cu/Fe, outside diameter = 1.6 m
Medium-long tunnels (holes), MLH	Composite canister, Cu/Fe, outside diameter = 0.88 m

6.5 CONCLUSIONS FOR FUTURE WORK

6.5.1 Canister alternatives

The analysis in PASS /6-10/ resulted in the judgement that the composite canister was the most favourable alternative, followed by the lead-filled canister alternative. The main reasons are that it was considered to be more advantageous in terms of mechanical integrity and the fact that a cold encapsulation entailed a simpler process technology.

A "cold" encapsulation in a pure steel canister has advantages from both the production standpoint (steel is a proven constructional material) and the cost standpoint. However, one decisive disadvantage is the fact that the corrosion life of a steel canister is short.

The canister alternatives have not been compared in detail for very deep holes, since it became clear during the course of the project that VDH would be ranked last among the repository systems studied.

6.5.2 Deep repository systems

After a weighing-together of the three areas "Technology", "Long-Term Safety" and "Cost", the system comparison for the chosen canister alternatives (see Table 6-2) resulted in three different groups:

1. KBS-3 and MLH.
2. VLH.
3. VDH.

VDH came last in all three areas and consequently last in the final ranking.

The comparison in terms of "Long-Term Safety" showed no significant differences between VLH, KBS-3 and MLH. All three systems were judged to be equivalent. Within the area "Technology", VLH came in 3rd place. To this was added a high cost per tonne of fuel for the big VLH canister. On the other hand, there is a potential for a relatively low construction cost for a VLH deep repository. In the weighing-together of the three areas, VLH was ranked after KBS-3 and MLH.

The order between KBS-3 and MLH will then depend on an evaluation of "Technology" against "Costs". The technology for KBS-3 is judged to be more robust, especially with regard to the actual deposition procedure. There is a significant cost difference to the advantage of MLH in the basic calculation. The optimization potentials for the two systems are roughly the same. However, the uncertainty regarding the technology for deposition is judged to be so great today that MLH cannot be chosen as the main alternative. Nor is it clear whether or how this uncertainty can eventually be dispelled. The conclusion is that the KBS-3 design is ranked just ahead of MLH.

A detailed account of the evaluation of the studied alternatives is provided in /6-10/.

6.5.3 Conclusions

The result of PASS is:

- Canisters with room for 12 BWR assemblies, or equivalent thermal load (KBS-3 and MLH), are recommended ahead of canisters with twice the capacity (VLH).
- A copper-covered steel canister (composite canister) is recommended.
- The KBS-3 design is recommended.

As a consequence hereof, the continued R&D work is based on:

1. Retention of the reference system according to the KBS-3 design.

2. Change in reference design of the canister to the composite design.

The conclusion of this is that KBS-3 is retained as the reference alternative with regard to the repository design, but that the copper-covered steel canister – the composite canister – is chosen as the reference canister. In conjunction with adaptation to local conditions on the selected site, the layout of the repository can be further optimized, whereby technologically closely-related variants can be given further consideration. With regard to long-term performance and safety, KBS-3, as well as VLH and MLH, represent systems with good margins.

VDH is not being studied any further as an integral system. On the other hand, certain geoscientific questions raised regarding conditions at great depths should be given further study. In a longer time perspective, this may give reason to reconsider this concept as a radically different deep disposal method.

In the case of deep disposal in granitic crystalline rock, the canister is a vital engineered barrier for long-term safety. Some continued work on a reserve alternative is therefore warranted. The previous reserve canister – lead-filled copper canister – is therefore being studied further for clarification of the details of the process technology involved in lead casting and canister sealing. Other canister alternatives (HIP, steel and steel/lead) will not be studied any further.

6.6 HANDLING AND DISPOSAL OF OTHER LONG-LIVED WASTE THAN SPENT NUCLEAR FUEL

Other long-lived radioactive waste from the Swedish nuclear power programme and from research and development activities is also planned to be disposed of in connection with the deep repository for spent nuclear fuel. The most important types of long-lived waste are:

- waste from the research activities at Studsvik,
- waste from the encapsulation plant for fuel,
- core components and certain "reactor internals".

The research activities at Studsvik have generated considerable quantities of waste. Some of this waste contains so much long-lived nuclides, mainly plutonium, that it may not be disposed of in SFR. It stems above all from studies of fuel and from older research activities concerning plutonium fuel and the properties of plutonium. The waste may also contain fuel fragments. This waste is conditioned in Studsvik. Liquids are solidified with cement, while solid waste is packed in drums that are placed in concrete moulds for later final disposal. In connection with the conditioning treatment, the waste is characterized as accurately as possible. There are also packages with older long-lived waste at Studsvik.

Waste from the encapsulation plant will normally be equivalent to waste from CLAB, and thereby only contain small quantities of long-lived material. In connection with accidents involving fuel failure, however, other long-lived waste can be obtained. Some of this waste will be packed in the fuel canisters, while some will be treated and packed in suitable final disposal containers.

Core components and reactor internals contain long-lived activity of another type. They have been bombarded with a high neutron flux during operation in the reactor core, forming induced activity. This consists mainly of relatively short-lived nuclides such as ^{60}Co, but certain long-lived nickel and niobium isotopes are also created. The ^{60}Co concentration is so high that core components in particular need heavy radiation shielding for a long period of time (more than 50 years). These components are therefore transported to CLAB for interim storage. The transport containers are similar to those used for fuel shipments.

The core components also include fuel boxes for BWR fuel. These can either be handled as parts of the fuel assemblies and be deposited with the fuel in the canisters, or they can be handled separately together with the other core components.

The long-lived waste (and some other operational waste that is generated after SFR has been closed) is planned to be disposed of in close connection with the deep repository for spent nuclear fuel. In PLAN 92 /6-6/ the repositories are shown co-sited, whereby the same transport routes and the same central area under ground are used. To avoid disturbances in the deep repository for spent nuclear fuel caused by the other types of waste, these will be emplaced about 1 km from each other. The tunnel between the different repository sections will be sealed in the same manner as the access tunnels in the fuel repository.

The section of the repository where Studsvik waste and waste from the encapsulation plant will be emplaced consists of a rock vault where the waste is stacked in concrete cells and grouted with concrete. The space between the concrete cells and the rock is filled with a sand-bentonite mixture. All handling is done by remote-controlled overhead crane. The principles of the barriers in this repository section agree with those applied in the silo repository in SFR.

The section of the repository where concrete moulds with core components etc. are to be deposited consists of two 350 m long rock vaults. The concrete moulds are carried in by a remote-controlled overhead crane and the moulds are stacked on top of one another. After stacking, they are grouted with concrete.

After concluded deposition in the repository for other long-lived waste, the tunnels are sealed with sand-bentonite.

Planned research and development on other long-lived waste is described in section 14.2 and in /6-12/.

7 PLAN FOR CONSTRUCTION IN STAGES

7.1 MAIN FUNCTIONS IN THE SYSTEM

The following units already exist or will be needed to manage the spent nuclear fuel and the other long-lived waste:

- Central interim storage facility for spent nuclear fuel (CLAB) for interim storage for 40 years;
- Encapsulation plant for spent nuclear fuel. Intended for encapsulation of fuel, core components and reactor internals; and equipped with a buffer store for encapsulated fuel. The store has to be expanded if deposition is delayed or interrupted;
- Repository for long-lived low- and intermediate-level waste. The repository is also intended for operational waste from CLAB and the encapsulation plant after SFR has been closed;
- Deep repository for core components;
- Transportation system between CLAB and the encapsulation plant and between the latter and the deep repository.

Moreover, the decommissioning waste from CLAB and the encapsulation plant, as well as transport containers, will be emplaced in a separate part of the deep repository.

Of these facilities, CLAB and sea transports, as well as terminal transports, are in operation today. CLAB is situated at the Oskarshamn Nuclear Power Station. Final decisions have not been made as yet on the sites of the encapsulation plant, the deep repository and a possible interim store for encapsulated fuel.

7.2 DEMONSTRATION DEPOSITION

SKB's previous work and time schedule for siting and building a repository for spent nuclear fuel entailed that after pre-investigations at three sites and detailed characterization at two during the 1990s, a decision would be taken a few years into the 21st century to build a repository at one of the sites. During the circulation of R&D-Programme 89 for comment and review, a proposal from SKN was discussed to the effect that a demonstration-scale repository should first be built, for example 5-10% of the full-scale repository /7-1/. In its decision concerning R&D-Programme 89, the Government asserted "... that one of the premises for further research and development activities should be that a final repository for nuclear waste and spent nuclear fuel shall be able to be put into operation gradually with checkpoints and opportunities for adjustments. In the next R&D programme under the Act on Nuclear Activities, SKB should explore the possibilities of including a demonstration-scale final repository as a step in the work of designing a final repository" /7-2/.

In the planning of the present RD&D programme, SKB considered this possibility of building and commissioning the repository in stages. The conclusion is that SKB finds that a demonstration deposition has considerable advantages. The present programme thereby calls for fulfilment of the research, development and demonstration work by first building the final repository as a deep repository for demonstration deposition of spent nuclear fuel. When this has been completed, the results will be evaluated before a decision is made whether or not to expand the facility to a full-scale repository. This plan also makes it possible to consider retrieving deposited waste for alternative treatment. The latter option means that it must be possible to retrieve all deposited fuel during the period the facility is being operated for demonstration purposes. At the same time, the site for the repository must be chosen so that there will be room for all the fuel. The siting process is only affected to a limited extent by whether the planning applies to a deep repository for demonstration deposition or to a complete deep repository. The requirements on background information and permits under different laws in the different phases are essentially the same.

The most important reason to build a repository for demonstration deposition is that this makes it possible to demonstrate the following, without the necessity of making what are sometimes described and perceived as final decisions:

- the siting process with all its technical, administrative and political decisions,
- the process and the methods for step-by-step investigation and characterization of the deep repository site,
- system design and construction,
- full-scale encapsulation of spent nuclear fuel
- the handling chain of spent nuclear fuel from CLAB to deposition in the repository,
- the operation of a deep repository,
- the licensing of handling, encapsulation and deep disposal, including the analysis of long-term safety,
- (retrievability of the waste packages).

Beyond this it is also possible to study the condition of the barriers a given shorter or longer time after deposition. However, the influence of the surrounding environment on the barriers is intended to be investigated primarily with non-radioactive material in the Äspö Hard Rock Laboratory.

The long-term safety of the final repository cannot be demonstrated through field tests. Allowability in this respect must always be based on a technical-scientific assessment of the performance of the repository over a long period of time. However, the background information that is gathered in conjunction with the construction of the deep repository for demonstration deposition means that a safety assessment can be performed based on site-specific "full-scale" data, and that it can be reviewed by all concerned authorities.

The reason SKB is planning a demonstration deposition is not doubt as to the feasibility and safety of the deep disposal scheme. The plan should be viewed as an expression of an awareness of and respect for the fact that the solution of the nuclear waste problem arrived at by the R&D work needs to be demonstrated concretely to concerned people in society far beyond the circle of experts for confidence-building purposes. It is SKB's opinion that a demonstration deposition of spent nuclear fuel with full freedom of choice for the future is a good way to obtain broad support for the method of disposing of the nuclear waste.

The planned demonstration deposition also means that the present-day generation is deciding for a span of time that roughly corresponds to its own active time, leaving it up to the next generation to make its own decision with as much background information as possible. This division of responsibility and freedom of choice between our own and the next generation can be described in the following manner:

1. It is the obligation of the current generation (nuclear power users and waste producers of active age between 1970 and 2010) to:
   - develop a safe deep repository system,
   - site a deep repository whose first stage permits demonstration deposition,
   - build the repository with retrievability, under strict supervision by the authorities and with requirements on long-term safety,
   - set aside money that covers future costs for the entire system.

2. It is the responsibility and free choice of the next generation (those who inherit our waste and are active after 2010) to:

   - evaluate the results of the demonstration deposition,

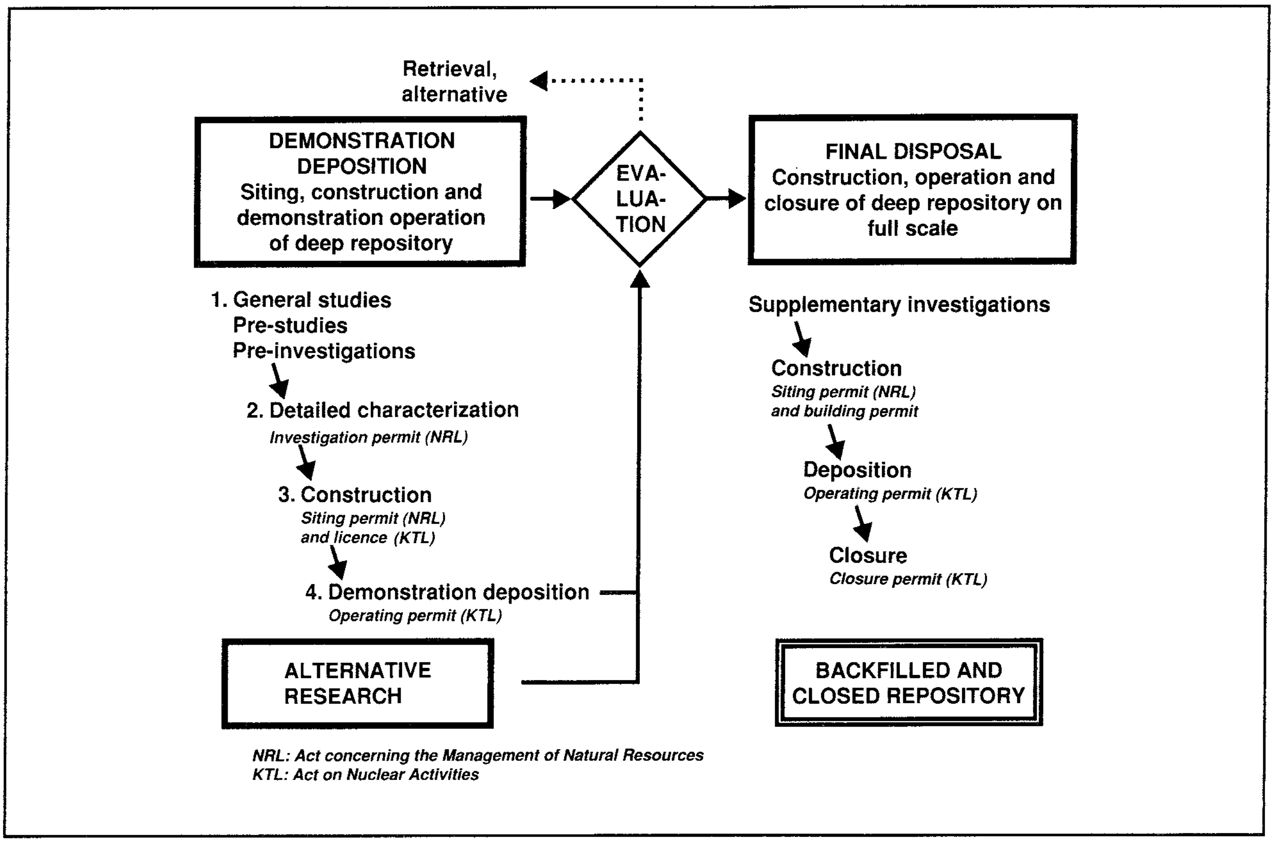


Figure 7-1. *Diagram of process up to closed deep repository.*

- perform a new, independent assessment of the long-term safety of a complete final repository,
- evaluate alternative methods,
- opt for either:
  i construction and closure of a complete final repository, or
  ii retrieval and alternative management.

The work up until all nuclear waste in Sweden has been deposited in a sealed deep repository is therefore planned to be carried out in two main phases: Demonstration deposition and final disposal. In all the work extends over a period of more than 60 years. The decision to take the step to final disposal will not be taken until after demonstration deposition has been completed, the results evaluated and other alternatives considered. These decisions lie beyond the year 2010. The plans that are discussed in this and following chapters have to do with the activities that are required to site and build the facilities that are needed for a demonstration deposition. It is SKB's judgement that the deep repository will later be expanded to full scale. However, it is not meaningful to discuss at this point in time the details of how this will be done. The important task for now is to demonstrate a possible method for long-term safe disposal and to provide future engineers and decision-makers with the best possible background information for their decisions.

In principle, it is possible for the purposes of further planning to keep the interim storage time of 40 years even with the demonstration deposition assumed here. SKB assumes that this can be carried out within about 20 years. It is thus possible to follow through with final disposal of remaining fuel and waste shortly after 2020, if this is so decided in about 20 years.

Figure 7-1 shows a diagram of the entire process.

Figure 1-2 in chapter 1 shows a rough schedule for all facilities for the storage and final disposal of radioactive waste. Construction of a facility for the final disposal of low- and intermediate-level waste will not begin until demonstration deposition of spent fuel has been carried out.

Planned activities for carrying out the work involved with a deep repository for demonstration deposition are presented as follows:

- encapsulation of spent nuclear fuel – section 8.3,
- interim storage and transportation of encapsulated fuel – sections 8.4 and 8.5,
- siting and excavation of deep repository for demonstration deposition – chapter 9 plus a background report,

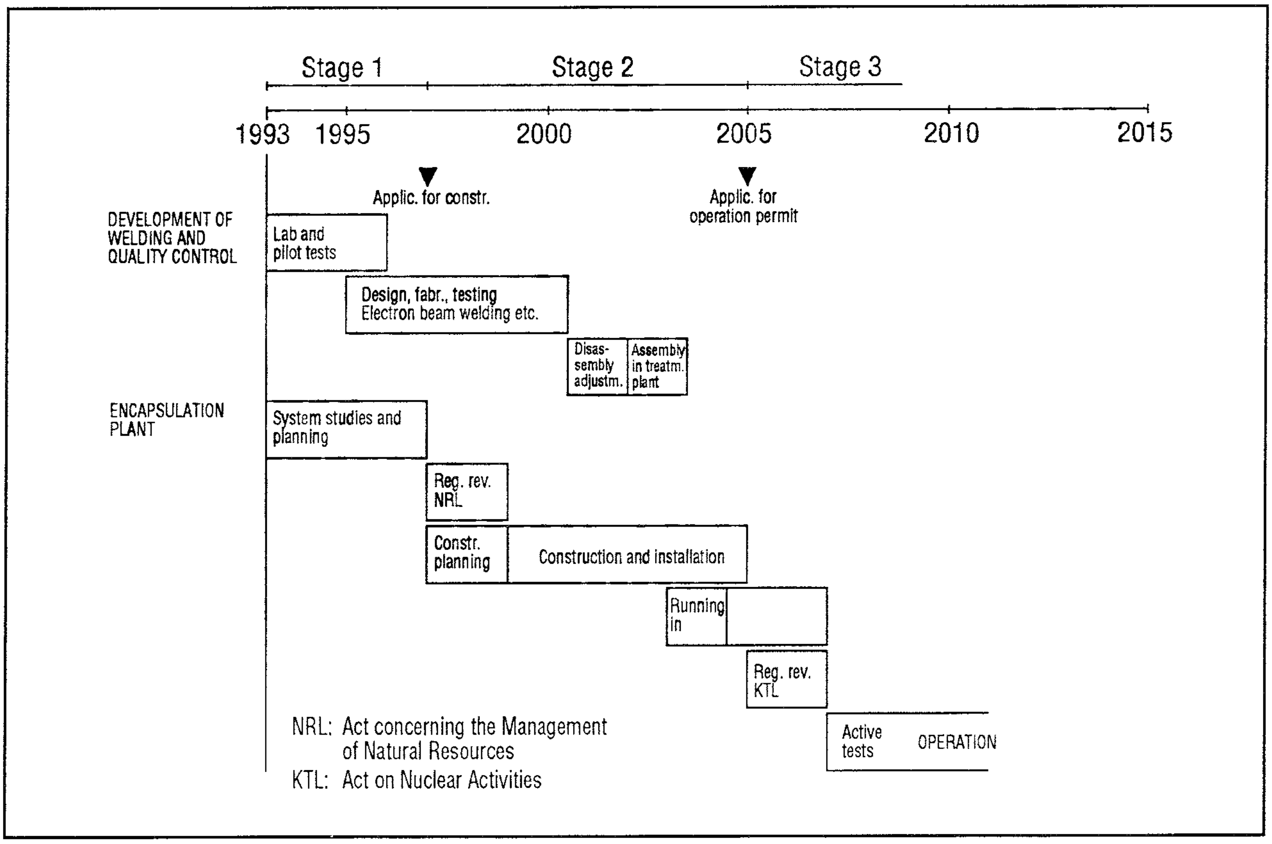


Figure 7-2. Timeschedule for encapsulation station. The timeschedule gives the earliest possible completion dates.

- safety assessments – chapter 10,
- supportive R&D – chapter 11 plus a background report,
- the Äspö Hard Rock Laboratory – chapter 12 plus a background report.

Figure 1-4 in chapter 1 shows a rough schedule for siting and design of the repository for demonstration deposition. Figure 7-2 shows a rough schedule for construction of an encapsulation plant for spent nuclear fuel.

8 ENCAPSULATION OF SPENT NUCLEAR FUEL

8.1 GENERAL

The plan presented for demonstration deposition of spent nuclear fuel entails that a facility for encapsulation of spent nuclear fuel needs to be put into operation in about 2007. SKB therefore plans to initiate detailed system studies and planning of this facility as soon as possible. In view of the fact that the spent fuel is already being stored at CLAB, there are in SKB's judgement clear advantages to building the encapsulation plant as an extension of CLAB. Besides transport and handling reasons, the main advantages of this siting are the availability of competent resources at the Oskarshamn station and at CLAB, plus the available infrastructure.

SKB is therefore planning to expand CLAB with a plant section for encapsulation of spent nuclear fuel. If special reasons emerge to indicate that encapsulation should be done at the deep repository instead, SKB will naturally also take up the question of an alternative siting of the encapsulation plant.

Different designs of canisters have been studied in the PASS project, see section 6.5. A composite canister with room for 12 BWR assemblies or 4 PWR assemblies is proposed as the main alternative. The canister, which has an outer corrosion protection of copper over an inner steel canister, meets requirements on long-term durability, while at the same time permitting simplified handling and filling at room temperature.

The design of the encapsulation plant is based on this canister, but shall at the same time be made flexible so that it is possible to switch to other canister alternatives.

A buffer store for finished canisters is also planned in connection with the encapsulation section. Provision shall be made for expansion of this buffer store in the event deposited canisters are to be retrieved after completed demonstration deposition. Containers for transport of finished canisters from CLAB to the deep repository will also be developed.

The studies of the encapsulation plant with associated questions concerning canister design will be pursued with the objective of submitting applications under the Nuclear Activities Act and the Natural Resources Act for permits to build the facility at the end of 1996. At the same time, preparations will be made for an expansion of the storage capacity at CLAB.

8.2 CANISTER - DESIGN, FABRICATION AND SEALING

Design

The reference alternative (see section 6.5) is a composite canister with an outer corrosion protection of copper over an inner steel structure that is designed to resist the pressures that may be encountered at the deposition depth. Each canister holds up to 12 BWR assemblies with fuel boxes or four PWR assemblies. A filled canister will have a residual void volume of about 1 m. This can be reduced to about 0.35 m by post-filling with a particulate material, e.g. quartz sand, glass beads or lead shot. To eliminate the risk that corrosive species will form inside the canister due to radiolysis, the canister can be filled with dry inert gas. The design of the canister is shown by Figure 5-2 in chapter 5.

Fabrication method

There are several different alternatives for fabrication of the copper canister. The most promising methods are hot isostatic pressing of either a copper tube or a canister, and rolling and forming of a copper plate with a longitudinal weld. The latter method has the advantage of permitting better control of the grain size in the copper. This is of importance for ultrasonic inspection of the weld on sealing.

Welding of copper

At present, electron beam welding is the most interesting alternative. Development work on electron beam welding during the past few years has primarily been concentrated on developing a method for welding without a vacuum. This work has been pursued within an international cooperation project, the EUREKA project. The technology is now available, and the method will be tested during the coming year for welding of thick copper sections.

The application of both methods for electron beam welding – in a vacuum chamber and under atmospheric pressure – is currently being investigated. The latter method offers much greater flexibility, but cannot weld the same metal thicknesses as conventional electron beam welding. The result of ongoing development work is crucial for the final choice of method, which will be made in around two years.

The development of welding technology for a full-sized copper canister will be given high priority during the period. Thereafter the development work will be focused on welding of full-sized prototype canisters.

Fill material

After filling with fuel, the composite canister has a void volume of about 1 m. It may for various reasons prove desirable to reduce the void volume by post-filling with a particulate material, e.g. quartz sand, glass beads or lead shot. By using a neutron-absorbing material, such as boron glass, it may be possible to reduce the requirements in criticality analyses. With the proposed design of the canister as it is now, the fuel's burnup must be taken into account in order to obtain sufficient criticality safety.

Further studies are planned before it is determined whether a solid fill material is to be used and, if so, which material is preferable.

Post-fabrication inspection – NDT

All steps in fabrication, forming and sealing of the canister shall be checked and documented. For the steps prior to packing of the spent fuel and sealing, this can be done using conventional methods, although specific refinements may be required. The methodology and technology for inspection of the sealing weld will have to be developed, however. This may also entail inspection of the microstructure of the copper material and the design of the weld.

This development work is being given high priority and is being carried out in close collaboration with development of the weld technology. Previous studies have shown that ultrasonic testing is a feasible method, but the technology must be optimized for copper and for detection of weld defects that are typical for electron beam welding. An alternative, or complementary, method may be X-ray tomography. The development work will also be aimed at evaluating the potential of this method for inspection of the sealing weld.

Service life

The corrosion resistance of copper in an oxygen-free environment is well known. No processes that could lead to failure of the canister due to corrosion in a shorter time than millions of years have been identified. Once reducing conditions have been re-established in the repository, the canister will be exposed to very little corrosion. The spent fuel will be completely isolated from groundwater in the rock for millions of years. Some corrosion can occur during the period when the canister still has an oxidizing environment. During this period, stress corrosion cracking and pitting are possible. Stress corrosion cracking, however, requires a chemical environment that is not found in deep groundwaters. Nor can a little pitting during the short period when oxygen still remains in the repository lead to any substantial shortening of the service life of the copper canister.

Alternative canister design

The alternative to a steel-supported copper canister is a copper canister which obtains its mechanical stability from nearly 100% filling of the cavity around the fuel with lead, see Figure 5-3 in chapter 5. From a corrosion standpoint, this canister is equivalent to the reference canister. Moreover, sealing and inspection can be done in the same way. However, the technology and methodology of lead filling requires some further development work.

8.3 ENCAPSULATION PLANT

8.3.1 Design

After interim storage in CLAB, the spent fuel will be encapsulated, after which it will be ready for emplacement in the final repository.

The planning work is based on the assumption that encapsulation takes place in an addition to CLAB. The encapsulation plant is thereby designed with the following functions:

Receiving section for storage canisters with spent nuclear fuel from CLAB:

- Encapsulation section for packing of fuel and any fill material, for sealing of the canister and for quality inspection.
- Despatch section for canisters. Despatch takes place in radiation-shielded transport casks.
- Service section, situated next to the encapsulation section, containing storeroom, workshop etc.
- Auxiliary systems, including cooling and ventilation system plus electrical and control equipment.
- Annex with personnel quarters and offices.

In a second stage, an expansion of the encapsulation plant is planned to enable core components and reactor internals to be treated. This will entail equipping the encapsulation plant with a treatment section for packing of core components and reactor internals in concrete moulds, filling of the moulds with concrete, buffer store for moulds and despatch to the deep repository.

The schematic layout of the encapsulation plant is illustrated in Figure 8-1. Encapsulation of the fuel is

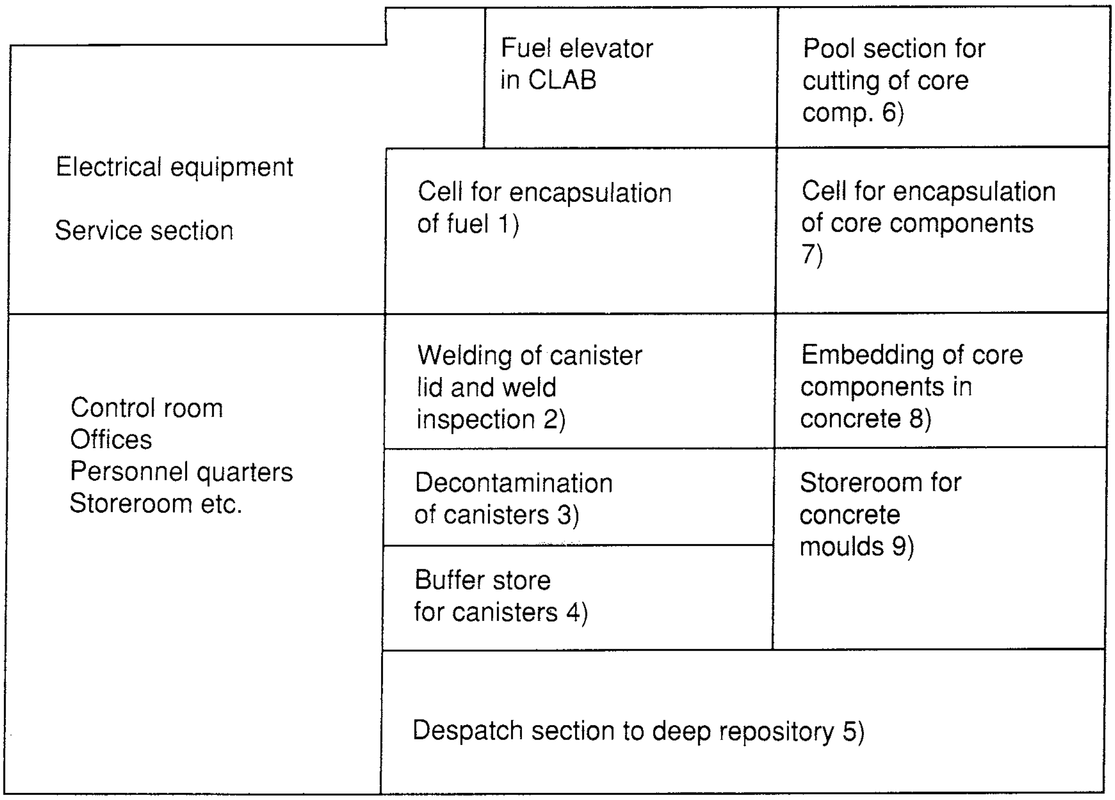


Figure 8-1. *Schematic layout of encapsulation plant.*

done within areas 1-4. Despatch of finished canisters for transport to the deep repository is done in area 5. Encapsulation of core components and reactor internals is planned to be done within areas 6-9.

Encapsulation of fuel is planned in brief to be done in the following manner.

The fuel is transferred from CLAB in storage canisters via the existing fuel elevator. The storage canister with its contents is transferred to the encapsulation cell. Before the fuel is lifted into the cell, the storage canister is drained and the fuel dried. In the cell, the fuel assemblies are placed in fuel racks.

A steel canister with copper jacket, placed in a radiation shield, is connected to the fuel cell and the fuel is transferred to the canister. In connection herewith, safeguard checks of the fuel are made. Then the inner void space is filled with boron glass beads, lead shot, sand or an equivalent material.

The steel lid is attached.

The canister is closed by means of electron beam welding, after which the weld is inspected.

Then the canister is transported to the interim store for washing and contamination check.

Encapsulation of core components and reactor internals is planned to be done as described in PLAN 92. This part of the encapsulation plant will be completed at a later stage after demonstration deposition has been carried out.

8.3.2 Schedule

A rough schedule is presented in Figure 7-2 in chapter 7. The work is divided into stages, each of which includes specified steps with defined goals.

Stage goals

Stage 1 includes system studies and planning of the facility as well as preparation of a preliminary safety report. The goal is to gather the necessary background data for investment decisions and for an application for a permit to build the facility. Such an application shall be considered under both NRL (the Act Concerning the Management of Natural Resources) and KTL (the Act on Nuclear Activities). Completion of the planning work requires a successful conclusion of the development work on electron beam welding and on NDT (Non-Destructive Testing). In addition, it is necessary to determine whether any part of canister fabrication will take place at the encapsulation plant, and if so by means of which method.

All in all, it is estimated that the work in stage 1 will take about 4 years. Regulatory review under NRL and KTL is estimated to take 1½ to 2 years.

Stage 2 starts with regulatory review of the building permit. Construction planning and design is pursued in parallel with the regulatory review procedure. When the permits have been obtained, a step-by-step construction of the facility is begun. The goal is a finished encapsulation plant for the fabrication of active canisters. The stage also includes the concluding development work in an inactive environment. The electron beam welder, the lead casting units (if lead casting is to be done), handling and lifting equipment etc. shall thus be able to be tried out in situ and are built first. Other parts of the facility, which can be built and put into operation without prior testing, are built in a later phase. The stage also includes inactive trial operation of the entire facility. Finally, a final safety report is compiled and an application for a permit for active operation is submitted. The time for the stage up to submission of the application for a permit for active operation is estimated to be 7-10 years (8 years in the schedule in Figure 7-2).

Stage 3 includes regulatory review prior to active operation. Trials and testing in an inactive environment are conducted in parallel with the regulatory review. After a permit has been granted, testing with active material is done, followed by demonstration operation of the facility and evaluation.

8.4 INTERIM STORAGE OF ENCAPSULATED FUEL

It is unlikely that the encapsulation plant can stand finished earlier than the deep repository. Moreover, after demonstration deposition it is possible that the deposited canisters will have to be retrieved. In both cases, it will be necessary to be able to store sealed copper canisters for an interim period pending further handling. A store for this purpose must fill the following functions:

- Cool the canisters.
- Protect the copper canisters against corrosion.
- Provide the necessary physical protection.

Such a store should exist in connection with CLAB, when it is extended with an encapsulation section. This could also then serve as a buffer store for ongoing production, if for example winter conditions prevent shipment of the encapsulated fuel.

Planning and licensing will be done in parallel with the work on the encapsulation plant.

8.5 TRANSPORTATION OF ENCAPSULATED FUEL

In the main alternative for the continued planning, canisters with fuel have to be transported from the encapsulation plant to the deep repository. Transport by sea, road and/or rail are possible options.

A new type of transport cask needs to be developed for these transports, where the approx. 20-tonne canisters can be handled.

The requirements on this transport cask coincide in all essential respects with the requirements on a normal transport cask for spent fuel. It shall satisfy the requirements in the IAEA's transport regulations /8-1/ concerning

- radiation shielding,
- integrity,
- criticality safety,
- fire resistance, and
- mechanical strength.

Fire resistance and mechanical strength shall be so good that the required radiation shielding, criticality safety and integrity shall be retained even after a severe accident. This is characterized in the IAEA's transport regulations in terms of the containers being able to withstand

- drop from a height of 9 metres,
- immersion in water to a depth of 200 metres,
- fire at 800°C for 30 minutes.

To satisfy these requirements, the design of the transport cask should closely resemble the design of today's transport casks for spent fuel. However, since the levels of radiation and heat generation in the fuel are considerably lower during transport to a deep repository than during transport to CLAB, the requirements on radiation shielding and cooling are less stringent. For example, cooling fins should not be needed on the cask. The canister in itself also contributes to the radiation shielding, so the wall thickness of the transport cask will more likely be determined by strength considerations.

The fact that the fuel is encapsulated is an advantage from the standpoints of radiation shielding and integrity, but entails more exacting requirements from the strength standpoint in connection with accidents, owing to the weight of the canister. The effect of an accident on the fuel inside the canister also needs to be assessed.

Transport casks for encapsulated fuel do not exist today; some development work is needed. Among other things, the criticality aspects require further elucidation.

8.6 PLANNED ACTIVITIES FOR 1993–1998

In keeping with the schedule in Figure 7-2, the plan for the period 1993–1998 calls for Stage 1 to be completed and Stage 2 to be commenced. This means that system studies for the encapsulation plant will be conducted and the plant will be planned and designed, and that a preliminary safety report and an environmental impact statement will be prepared. These studies will serve as a basis for an investment decision and an application for a permit to build the plant.

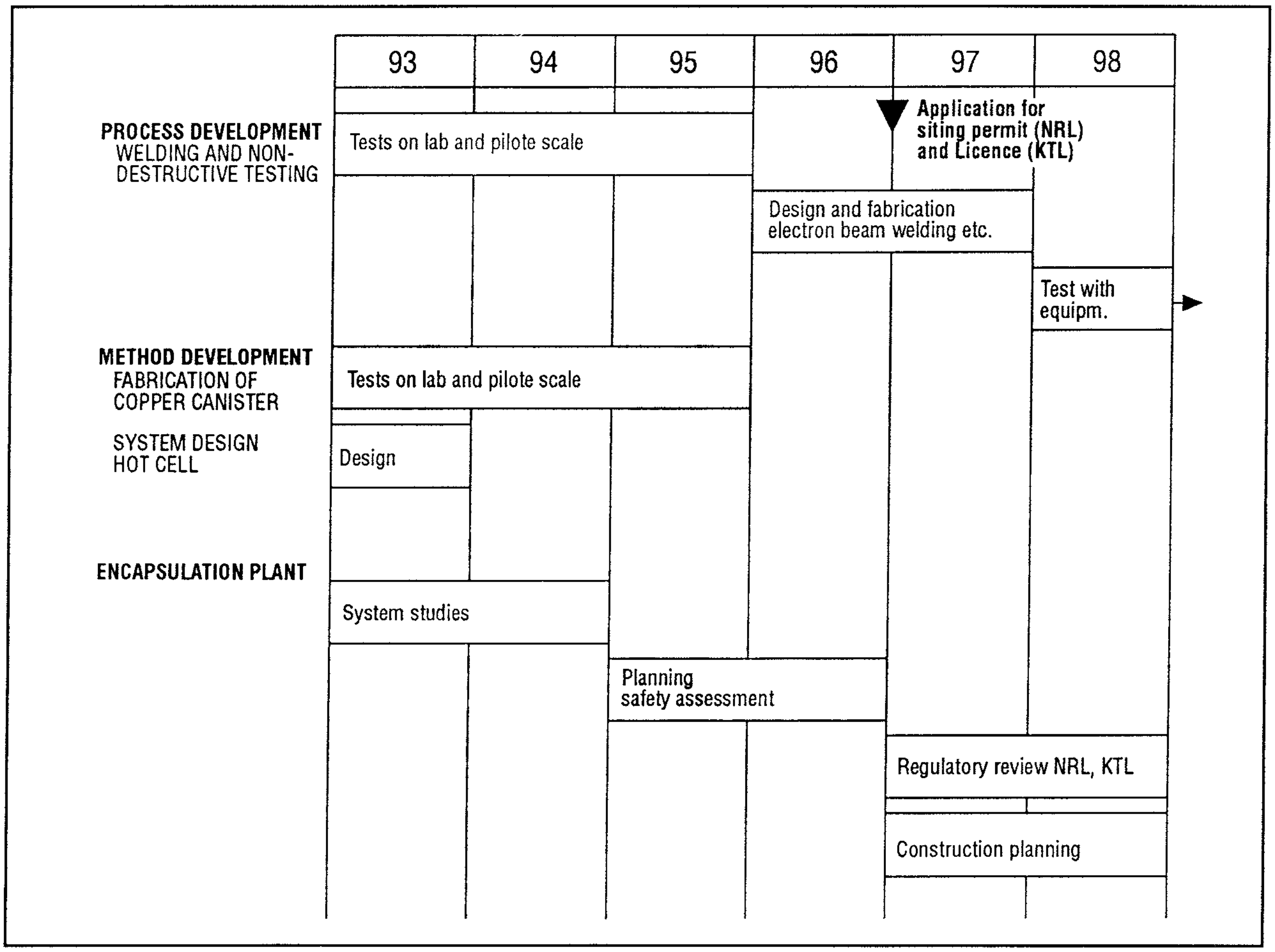


Figure 8-2. *Timeschedule for encapsulation plant 1993 – 1998.*
NRL = Act /1-20/ concerning the management of Natural Resources
KTL = Act on Nuclear Activities

In parallel with the regulatory review, detailed construction planning will be carried out during the latter part of the period.

As a basis for the planning and design of the facility, canister development will be pursued so that a final canister design and fabrication technology are chosen during the period, and so that full-scale fabrication tests are carried out. Furthermore, studies of welding technology for sealing of the canister will be concluded and full-scale tests plus development of an inspection method will be carried out.

A schedule for the work with the encapsulation plant for the period 1993–1998 is presented in Figure 8-2.

In parallel with the work on the encapsulation plant, questions surrounding interim storage of canisters and transportation of filled canisters will be cleared up.

9 SITING AND CONSTRUCTION OF A DEEP REPOSITORY FOR DEMONSTRATION DEPOSITION

This chapter summarizes the premises for and organization of SKB's work with siting and constructing a deep repository for demonstration deposition of spent nuclear fuel.

9.1 BACKGROUND, GOALS AND GENERAL PREMISES

The fundamental goal of "safe final disposal of nuclear waste generated in the operation of the nuclear power plants" can in principle – as was demonstrated in the KBS-3 report – be achieved by building an underground rock facility in which encapsulated fuel is emplaced in a specified manner. The combination of engineered (the low-soluble fuel, the canister, the bentonite buffer) and natural (the rock) barriers effectively isolates the waste for a sufficiently long period of time. In other words, safety rests on a natural-scientific and technological basis. However, in order to achieve the safety striven for, the system of engineered barriers must be achieved in reality, i.e. the necessary facilities must be sited, built, operated and finally closed and sealed. The realization that this is as much a legal and a societal process as it is a technical one must guide the siting work.

A deep repository can be regarded as a medium-sized industrial facility on the surface plus an underground part with relatively small but deep-lying rock caverns. Well-established rules apply to every industrial siting. Just like any other industrial facility, the deep repository will be judged according to these rules. But a deep repository is also a nuclear installation, and the question of how the nuclear waste is to be disposed of is a very controversial one. Because of this, combined with the ambition to carefully study all aspects of repository function and meet stringent safety requirements, the siting process for a deep repository is extra comprehensive and thorough.

Figure 9-1 illustrates schematically the goals, premises and possibilities for the siting work. An analysis of the premises provides an idea of, firstly, what requirements must be met and, secondly, what flexibility exists in the means of achieving the goals as well as under what circumstances there is a risk of being blocked. The premises within the subject areas indi-

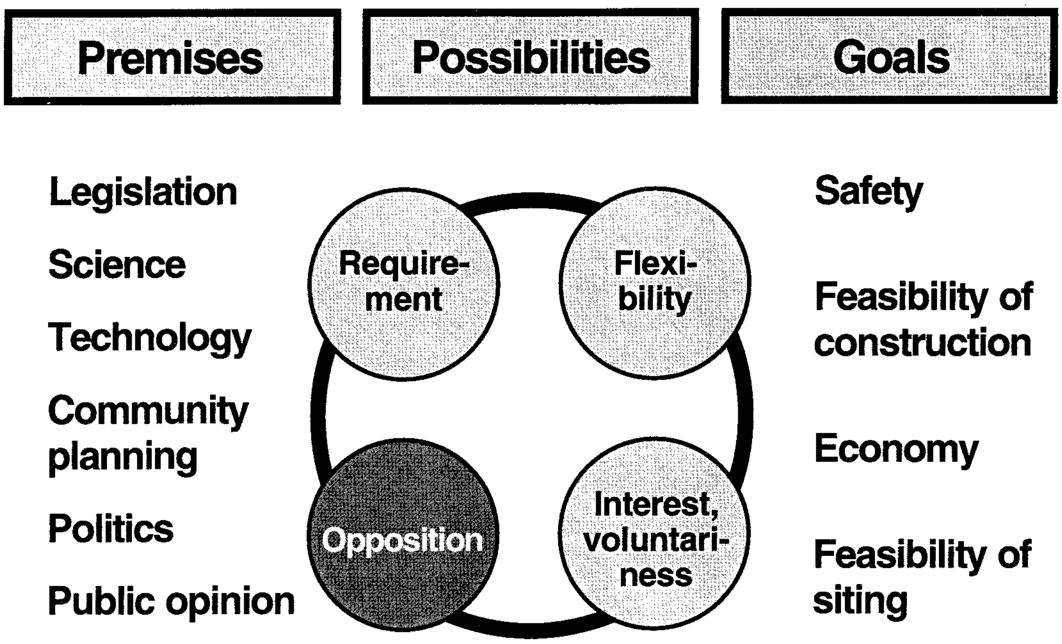


Figure 9-1. *Schematic illustration of goals, premises and possibilities.*

cated in Figure 9-1 are being analyzed in the ongoing siting work. Since siting is a long process, it is possible that some premises will change during the course of the process.

9.1.1 Legal premises

Sweden was early to enact laws governing safety, division of responsibilities and financing forms within the field of nuclear waste management. This has contributed to the fact that there is now a functioning structure and a clear division of roles within the research and development work to guide the siting and construction of the remaining facilities. The premises in this respect must therefore be considered good. SKB has compiled a summary of the laws and ordinances that will or may be applicable at some point in the siting process /9-1/. This summary reveals that:

- The Act Concerning the Management of Natural Resources, NRL, and the Act on Nuclear Activities, KTL, are the two central laws under which the Government must grant permission.
- Certain other laws or regulations will or may be applicable at some point in the process.
- An environmental impact statement (MKB) should be prepared at an early stage of the siting process and then be updated and particularized as the work progresses. The environmental impact statement constitutes a supporting document in the licensing process. It will also serve as a tool for informing all concerned parties and giving them an opportunity to offer viewpoints on what issues are important to take up in connection with the various licensing processes.
- The siting process will require an effective interaction between concerned authorities, the Government, relevant municipalities, county administrations and SKB.

9.1.2 Scientific and technical premises

Research on the handling and final disposal of nuclear waste got under way in earnest in Sweden in the mid-1970s. SKB and others have conducted comprehensive and targeted research and development work. Geology, materials, chemistry and other aspects have been studied in detail. Several integrated safety assessments have been conducted in Sweden. Similar safety assessments have been conducted in other countries and internationally with similar conclusions regarding feasibility and safety.

With the above background and based on its own experience from 15 years of R&D activities, whose current status is described in other sections of this RD&D programme, SKB draws the following conclusions:

- The scientific and technical premises that are necessary for implementing a safe final disposal of the spent nuclear fuel exist in Sweden.
- The Swedish crystalline bedrock offers good geological conditions for a deep repository, and it is therefore possible to find sites that meet the stipulated requirements in most parts of the country.
- The best way to proceed in realizing the deep repository system is to select, investigate and evaluate specific candidate sites.

9.1.3 Societal premises

The siting of facilities in a system for deep geological disposal of spent nuclear fuel must be accomplished in harmony with the premises and plans that exist in society. The investigations and the facilities will require land space, create jobs and have spin-off effects for local industry and services. Furthermore, they can have an impact on the environment visually and in the form of transportation, drilling, rock blasting and construction activities. All of these aspects must be described and discussed with the municipality, concerned citizens and public authorities.

In summary, it is SKB's judgement that

- the societal premises are important to both site selection and detailed emplacement and configuration of the facilities on the selected site;
- it is possible to satisfy social, planning and environmental requirements with good margin;
- the investigation activities and the deep repository facilities can bring substantial benefits to a locality. They represent the application of advanced environmental and geotechnical engineering and will attract great scientific and international interest.

9.1.4 Political and public-opinion premises

The siting and construction of a deep repository for spent nuclear fuel is a sensitive political and public-opinion issue. Experience from both Sweden and other countries show that strong feelings and opinions can be aroused. Opposition to the siting of industrial facilities of all kinds is not unusual in today's democratic society, where many opposing interests, desires and values must be reconciled. On the other hand, there is no reason to overdramatize the potential opposition to a deep repository either.

It is SKB's conclusion that

- it is important to have open information channels to and good relations with the concerned municipalities and the citizens who are directly affected by or feel strongly about the siting. This is a necessary prerequisite for success in carrying out the important environmental protection work entailed by deep disposal of the spent nuclear fuel.

9.2 DESIGN OF THE DEEP REPOSITORY

9.2.1 General layout

A deep repository for the disposal of encapsulated spent fuel consists of facilities both above and below ground. A layout sketch of these facilities is shown in Figure 9-2.

The underground facilities consist primarily of a system of parallel deposition tunnels with associated transport tunnels, service areas and ventilation systems. In practice, the layout will be adapted to the local structure of the rock. In connection with this adaptation, the design described here can be further optimized, whereby technically closely-related variants can be considered.

The waste canisters are deposited in vertical holes bored from the bottom of the deposition tunnels. The design is shown in Figure 9-3.

A deep repository for demonstration deposition constitutes a part of the whole repository. In Figure 9-2, 10% of the repository has been marked. Only these parts will be built for the demonstration deposition.

The deep repository can be reached via a ramp, as in Figure 9-2, and/or a shaft from the industrial area on the surface. The ramp and one or two ventilation shafts will be excavated during detailed characterization and be designed for the operating requirements of the expanded facility.

9.2.2 Technology for deposition

The waste canisters are transported down underground either via the ramp in radiation-shielded containers or in a shielded elevator in a shaft. Both alternatives are still being kept open. On the deposition level, the canister is driven up to the mouth of the deposition tunnel (transferred to a vehicle under ground in the shaft alternative) and is then transferred to a specially designed deposition vehicle. This deposition vehicle drives up to the deposition hole, where it raises the canister to the upright position and lowers it down into the hole. The bottom part of the bentonite buffer has already been emplaced in the hole. When the canister is in position, other bentonite blocks are placed in position by the same deposition vehicle that handled the canister. The handling sequence in the deposition tunnel is illustrated in Figure 9-4.

The proposed equipment consists of industrially proven components and can be designed, built and tested in accordance with accepted engineering practice. The work can be based on the experience obtained from the construction of CLAB and SFR.

The bentonite around the canisters consists of pre-compacted blocks. Such blocks have been manufactured in existing presses for the pilot tests that have been conducted at Stripa, whereby commercial bentonite grades with a low moisture content have been used. If the moisture content in the bentonite is increased, the thermal conductivity of the bentonite blocks increases, which leads to a lower temperature peak in the bentonite at the same thermal load in the canister. A suitable compaction technology and moisture content will be determined by means of buffer mass tests in the Äspö Hard Rock Laboratory.

The deposition tunnels will be backfilled with a mixture of bentonite and sand. The lower half of the tunnel can be compacted using conventional compacting machines. These machines will not fit in the upper part, especially against the roof. The compaction needs to be harder than what is obtained if the mixture is merely sprayed in with, for example, shotcreting equipment, which was tried at Stripa. A compaction method will be developed and tests conducted in the Äspö Hard Rock Laboratory.

Before the deposition equipment is put into use, trials can be conducted in the Äspö Hard Rock Laboratory.

Alternative designs of the deep repository and deposition methods have been studied in the PASS project.

9.2.3 Retrievability

It is necessary to be able to retrieve canisters both during and after deposition.

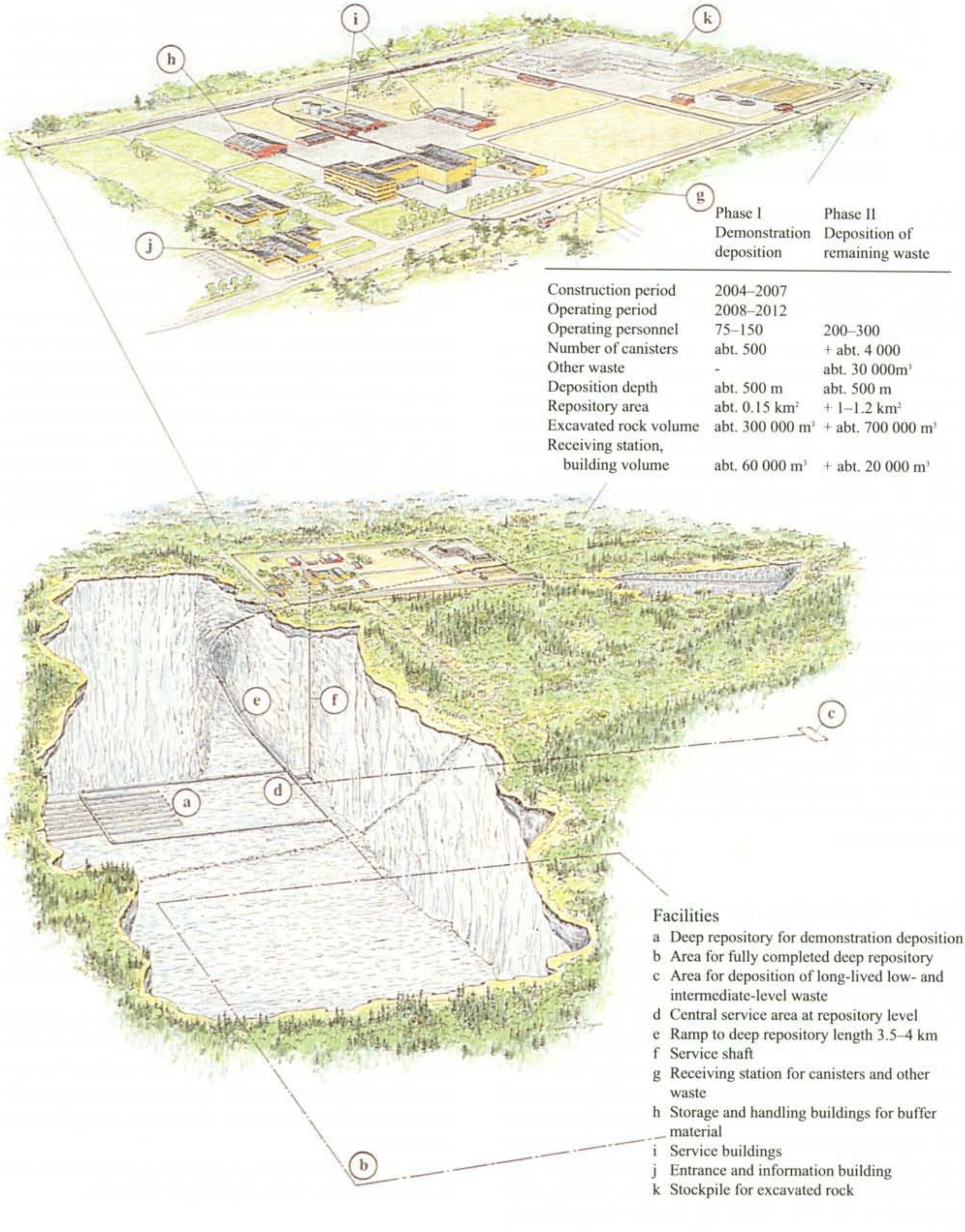
The possibility of accidents during deposition cannot be ruled out. This may require remedial measures before deposition can be continued. This must be possible even when the canister has just been emplaced. The requirement of retrievability is one of several criteria to be considered in the design and testing of the deposition machine.

After demonstration deposition, it shall be possible to interrupt the activities and decide whether or not to retrieve the deposited canisters. Technically, the solution may be to use the same machine under ground for retrieval as for deposition. Otherwise special equipment is required.

This question will be examined in connection with the development of the deposition machines.

9.2.4 Construction methods in rock and buffer, backfilling and sealing

Underground construction is a proven technology in Sweden. Documented experience exists from varying Swedish rock conditions, from the ground surface



	Phase I Demonstration deposition	Phase II Deposition of remaining waste
Construction period	2004–2007	
Operating period	2008–2012	
Operating personnel	75–150	200–300
Number of canisters	abt. 500	+ abt. 4 000
Other waste	-	abt. 30 000m³
Deposition depth	abt. 500 m	abt. 500 m
Repository area	abt. 0.15 km²	+ 1–1.2 km²
Excavated rock volume	abt. 300 000 m³	+ abt. 700 000 m³
Receiving station, building volume	abt. 60 000 m³	+ abt. 20 000 m³

Facilities

a Deep repository for demonstration deposition
b Area for fully completed deep repository
c Area for deposition of long-lived low- and intermediate-level waste
d Central service area at repository level
e Ramp to deep repository length 3.5–4 km
f Service shaft
g Receiving station for canisters and other waste
h Storage and handling buildings for buffer material
i Service buildings
j Entrance and information building
k Stockpile for excavated rock

***Figure 9-2.** Plan of possible design of a deep repository. About 10% of a complete repository as marked in the figure is built for demonstration deposition.*

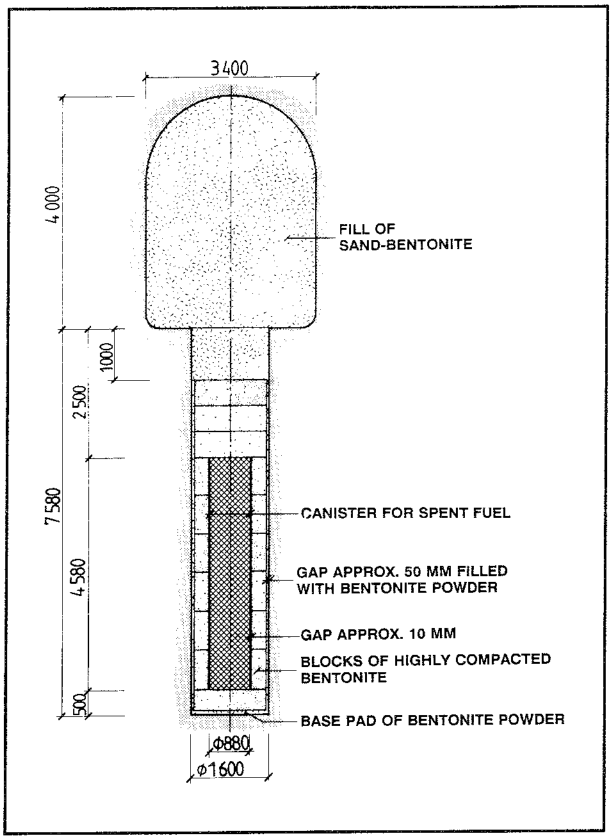


Figure 9-3. *KBS-3 design of canister positions.*

down to a depth of about 1 000 m. Extensive mining operations have been conducted and are being conducted around the depth, about 500 m, for which the deep repository is planned.

It is assumed that known technology (drilling/blasting, tunnel boring, raise boring) will be used where possible. This will be supplemented with further developed methods where required.

An important construction-related question is how well the seepage of water into the repository can be limited, especially in the deposition holes. Models for the properties and behaviour of the grouting material in fractures have been developed within the Stripa project. The modelling work and validation of the models continues. A problematical situation has been identified during tunnelling at the Äspö Hard Rock Laboratory, namely sealing of highly conductive fractures with high water pressure. The goal is to further develop suitable sealing technology for the needs that may be found to exist in conjunction with detailed characterization and construction of the deep repository.

A description of how blasting or tunnel boring will be done within the deposition area will be presented in about 1995, since this can influence the method used for detailed investigation of the area under ground.

Equipment for boring of deposition holes is on the drawing board but has not been tried out. The rock walls are affected much less by full-face boring than

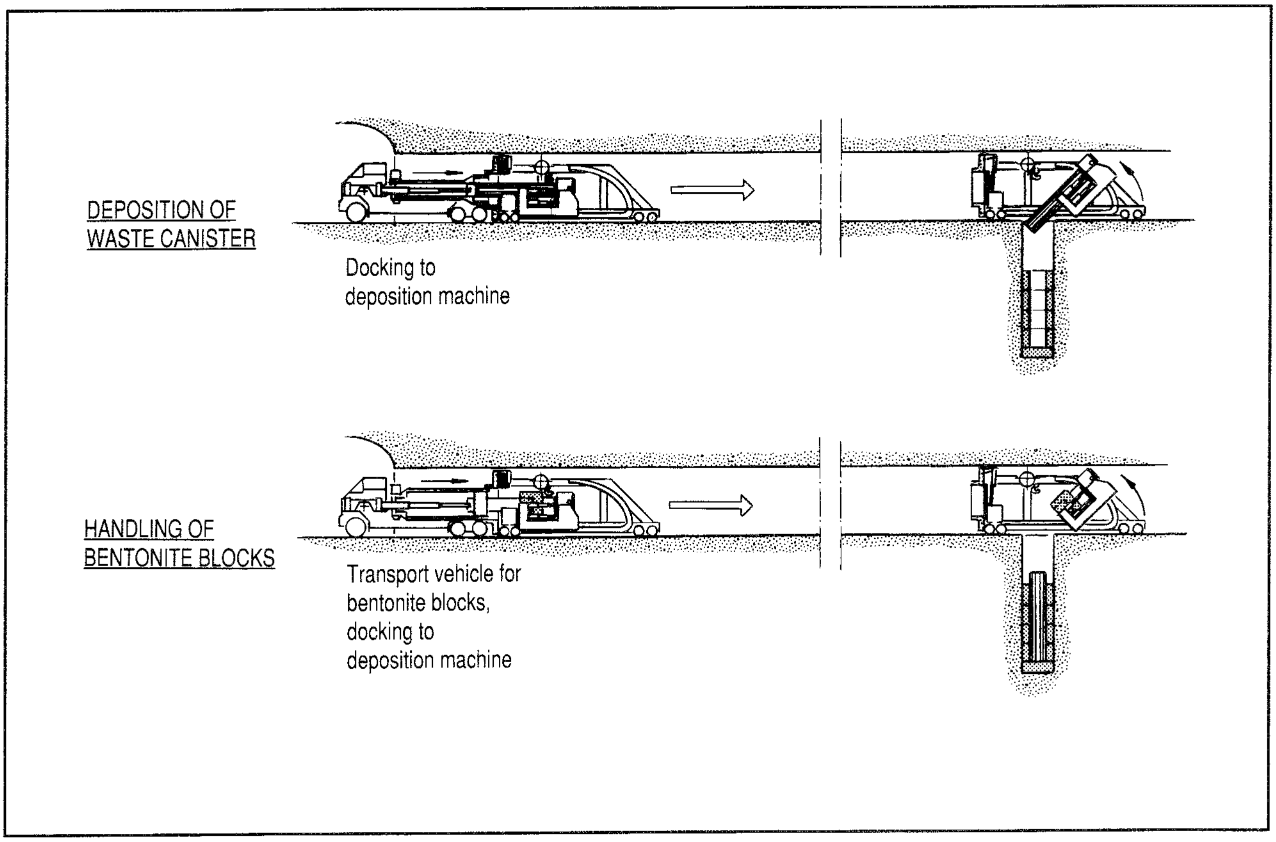


Figure 9-4. *Deposition method.*

by blasting, but the impact is nevertheless significant for an understanding of the properties imparted to the rock around the hole.

Work aimed at improving knowledge of the impact of the construction activities on the rock wall will continue. Tests in a natural environment will be able to be conducted in the Äspö Hard Rock Laboratory.

Backfilling of the space around the canisters (compacted bentonite blocks) and backfilling of the deposition drifts (sand/bentonite mixtures) will be done with material whose properties are well understood today. What remains to be developed is above all methods for compacting bentonite blocks and machines for backfilling with sand/bentonite.

Plugging after concluded deposition does not need to be done until the time comes for the repository to be sealed. Pilot tests have been conducted at Stripa and the work is continuing with, among other things, a study of the conditions pertaining to sealing of the rock surrounding the drift or the shaft where plugging is to be done. Later it will be possible to perform tests in the Äspö Hard Rock Laboratory.

9.2.5 Future research

The geometric configuration of the repository will be preliminarily defined at the start of the siting process. It will then be revised as more and more site-specific information is obtained from pre-investigations and detailed geological site characterization.

Before the detailed characterization phase is begun, technology for passage of water-bearing zones with high water pressure needs to be devised. An account will be submitted together with the application for permission for detailed characterization. The central question is how water-bearing fractures can be sealed and what material is to be used in different situations. Material models will be devised and validated up until 1998. After that, larger-scale tests will be conducted in the Äspö Hard Rock Laboratory.

9.3 SITING PROCESS AND SCHEDULE

An important point of departure for SKB's planning of the siting process is the Government's decision regarding SKB's previous R&D-Programme 89. It states the following: *"The Government notes that SKB's choice of sites for a final repository will be reviewed by different authorities in connection with SKB's application for permission to carry out detailed characterization of two such sites under the Act (1987:12) concerning management of natural resources etc., the Environment Protection Act (1969:387) and the Planning and Building Act (1987:383)."*

Furthermore, the Government emphasized the fact that SKB should, during the course of the siting work, furnish information to concerned national authorities, county administrations and municipalities.

Based on these guidelines and assessments of what background material must be gathered for planning and construction of the facility and for environmental impact and safety assessments, the siting process is planned to proceed in a number of main stages.

9.3.1 Stage 1. General studies, pre-studies and pre-investigations

In this stage, a broad review of the premises for siting of a deep repository will be done to begin with. Important siting factors will be outlined and analyzed.

A truly clear picture of premises and conditions is not obtained until concrete area- and site-specific investigations are carried out. Pre-studies are therefore carried out for those municipalities which, for example through their own initiatives, display an interest in having a closer examination made of the premises for a deep repository. In a pre-study, fundamental facts are gathered and evaluated on e.g. transport matters and the technical, social, societal and geological premises for a deep repository in the municipality. With the aid of a pre-study, both SKB and the municipality can – at an early stage and without committing themselves – obtain a preliminary idea of the premises and decide whether it is worthwhile to examine more closely the feasibility of a deep repository.

In parallel with the general studies and the pre-studies that lead to the selection of candidate sites, the coming work on the candidate sites is planned and prepared. A programme for geoscientific pre-investigations is drawn up. The studies and analyses of the technical premises and of the repository system continue and are reported in a preliminary system and facility description. A preliminary environmental impact statement is prepared and a programme for local participation, information and socio-economic studies is drawn up.

Based on the data obtained from the general studies and the pre-studies, SKB will establish local offices and commence pre-investigations on a couple of sites. These investigations include:

a) Geoscientific surveys and assessments in several stages from the ground surface and in boreholes. The goal is to determine the exact location of a rock volume for a deep repository and to preliminarily verify the suitability of the site.

b) System studies for the surface and sub-surface facilities. Planning and design work. Devising of site-adapted layouts. Analyses of environmental and safety aspects.

c) Technical and socio-economic studies to shed further light on and determine the impact of the siting of a deep repository in the locality on the com-

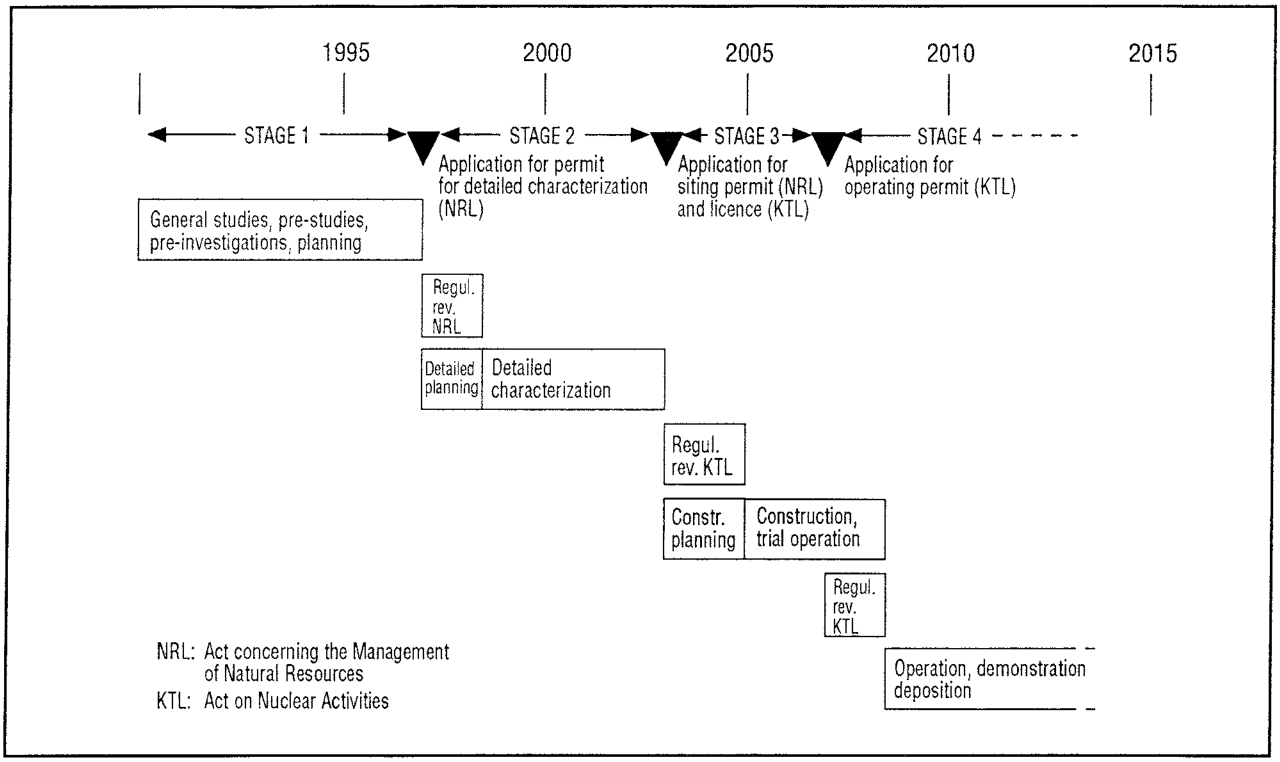


Figure 9-5. *Example of timeschedule for the deep repository up to the completion of demonstration deposition. The timeschedule gives the earliest possible completion dates.*

munity, the environment, the local economy and local industry.

d) Studies of suitable transport modes and routes for encapsulated spent nuclear fuel from CLAB to the deep repository.

When the pre-investigations in the field are begun, suitable forms should have been established for allowing the municipality and the citizens affected by the siting insight into, and an opportunity to follow and offer viewpoints on, the activities. It is assumed that the socio-economic studies will be conducted in co-operation with the municipality and the concerned local citizenry.

Stage 1 is concluded with a compilation of all the necessary background material in an application under the Act Concerning the Management of Natural Resources (NRL) for permission to carry out detailed characterization. An environmental impact statement, a safety assessment and a programme for detailed characterization will be appended to the application.

9.3.2 Stage 2. Detailed characterization

Detailed characterization entails building a tunnel and/or shaft down to that part of the rock where the deep repository is planned to be built. Below the tunnel/shaft and down at repository level, the rock is characterized using methods and programmes developed and tested in the Äspö Hard Rock Laboratory, see chapter 12. The goal of the detailed characterization is to

- finally verify the suitability of the site for a deep repository,
- obtain background data for a detailed layout,
- obtain supporting data for an application for permission to build a deep repository for demonstration deposition.

The technical and socio-economic studies begun in stage 1 are pursued in greater depth during stage 2. The deep repository is planned and designed. SKB's establishment in the area is expanded. Stage 2 is concluded by a compilation of all supporting material for applications for a siting permit under the Act Concerning the Management of Natural Resources (NRL) and a licence under the Act on Nuclear Activities (KTL) and submitting it to the regulatory authorities.

9.3.3 Stage 3. Construction of deep repository for demonstration deposition

This stage includes expansion of the deep repository up to a facility ready to receive encapsulated spent nuclear fuel for demonstration deposition. The rock excavation work can be carried out in a few years owing to the fact that a large part of the access drifts have already been driven as a part of the detailed characterization in stage 2. The facility must have equipment to receive shipments of canisters with fuel, emplace the canisters in deposition positions and emplace bentonite in the deposition tunnels.

Stage 3 leads to a final safety report in support of an application for an operating permit under the Act on Nuclear Activities.

If a permit is obtained, stage 4 commences: Demonstration deposition of encapsulated fuel lasting several years, plus the subsequent evaluation.

9.3.4 Schedule

A rough schedule is shown in Figure 9-5.

The schedule is based on certain assumptions and assessments of the speed with which it is possible to gather the background data needed for the permit applications. The most important prerequisites are:

- Complete pre-investigations are conducted at two sites.
- Full-scale detailed characterization is carried out on one site. Only if the site chosen for detailed characterization should prove to be unsuitable should detailed characterization be started at another site. The law's requirements on investigation of different alternatives in siting applications and environmental impact statements are satisfied by the comprehensive body of material compiled in stage 1.

With these assumptions, and providing the investigations and evaluations can be carried out effectively, the demonstration deposition can be completed within about 20 years.

9.4 FUNDAMENTAL REQUIREMENTS AND IMPORTANT SITING FACTORS IN SITE SELECTION

The selection of candidate sites will be made in accordance with the fundamental requirements that must be made on a deep repository site from safety-related, technical, societal and legal viewpoints. It must be possible by means of a safety assessment to demonstrate for a selected site and selected repository system that the safety requirements stipulated by the regulatory authorities are complied with. It must be possible to build the repository and execute the deposition technically in the intended manner. Siting, investigations and construction shall be carried out in such a manner that all relevant legal and planning requirements are met. And last, but not least, it shall be possible to execute the project in a spirit of harmony with the municipality and the local population.

9.4.1 Safety-related requirements

The fundamental safety principle for the deep repository system being planned by SKB is to enclose and isolate the spent nuclear fuel in tightly sealed canisters that are deposited at a depth of about 500 metres on the selected repository site. The entire operation and system design is aimed at ensuring that this waste isolation will be achieved and endure over a very long period of time so that the radioactive materials decay inside the canister and cannot escape. This means that the rock's most important safety-related function for a final repository is to guarantee stable conditions for the engineered barriers over a long period of time. SKB's geoscientific research and the safety analysis, SKB 91, show that the rock at many places in Sweden is capable of performing this safety-related function.

9.4.2 Technical requirements

The technical requirements that are made on the bedrock on the candidate sites are primarily connected with constructability. It shall be possible to build a repository at a depth of about 500 m on the chosen candidate site without causing excessively great problems with collapse-prone rock volumes or large water seepage.

Another technical requirement, which is also favourable from a safety viewpoint, is that the candidate site shall be easy to interpret, i.e. that the pre-investigations shall give such a result that rock properties of importance for constructability can be determined with good certainty. This is important for being able to design the repository as well as possible with a view towards long-term safety, at the same time as good knowledge of these structures is essential for being able to plan the configuration and location of the tunnel system and deposition holes.

9.4.3 Societal requirements

The main societal factors that must be taken into account in the siting process are plans for land use, transports of spent fuel, public opinion, landowners and infrastructure. These factors are of great importance for the suitability of an area and are in practice decisive once the technical and safety-related requirements have been satisfied.

9.4.4 Summary of important siting factors

Table 9-1 provides a brief summary of the most essential siting factors.

It is important to note that most unfavourable factors can be compensated for by means of different remedial measures. For example, a heterogeneous bedrock or a low degree of surface rock exposure can be

Table 9-1. Siting factors.

Siting factor	Comment
Technical/geoscientific:	
Long-term stable environment	The repository should be situated in parts of the rock that do no comprise weak zones of fractured rock in which significant future fault movements could be released. The rock volumes used for the repository should not contain valuable minerals or the like which could lead to future intrusion that might disturb the safety barriers. The groundwater at the selected site/depth should have long-term-stable chemically reducing conditions.
Safety	The rock shall comprise an extra safety barrier through its capacity to absorb and retain any released nuclides. This capacity is dependent on groundwater conditions (flows, flow paths), groundwater chemistry and retardation mechanisms along the flow paths. These conditions can be allowed for by taking into account factors such as hydraulic gradient, distance between proposed repository site and discharge area, presence of water-bearing fracture zones and vein minerals, and presence of saline groundwater.
Constructability	A site's constructability is determined by the locations and characters of fracture zones, the presence of rock types with a tendency for collapse or water-bearing, size and orientation of rock stresses and mechanical properties of the bedrock.
Predictability	It is an advantage if a site is easy to interpret, i.e. permits a high certainty in predictions of bedrock conditions between investigated parts of a site. Predictability is dependent on degree of exposure and bedrock conditions.
Societal:	
Land use	Consideration shall be given to areas of national interest for nature conservation, restricted areas, culturally protected or archaeological interesting areas, etc. Careful attention shall be given to the counties' nature conservation plans and the municipalities' comprehensive plans. The impact of population density, area-based enterprises (e.g. forestry, animal husbandry, aquaculture, agriculture) etc. shall be studied. Areas with planned industrial land can be of particular interest.
Transportation	It is technically possible to transport the waste in a safe manner to all potential sites in Sweden. Safety, logistics, need for new investments, public opinion and costs will be investigated for potential sites.
Infrastructure	The need for and influence of existing infrastructure and local enterprise will be clarified for potential sites.
Public opinion	Good cooperation with concerned parties is important. The municipality and the local population will receive information and be given an opportunity to follow and offer viewpoints on the work.

compensated for by increased drilling. A lack of, or shortcomings in, the infrastructure can be compensated for by the construction of railways, roads, housing etc. These compensations do lead to higher total costs for the deep repository, however, which must be weighed against the advantages of the site.

A more detailed discussion of the geoscientific, technical and societal premises for the siting of a deep repository is provided in the background report.

9.5 EXPERIENCE FROM PREVIOUS SITE INVESTIGATIONS AND SITING

There has not been any real siting work of a project nature for the deep repository before the start of the siting project that SKB set up in the autumn of 1991. On the other hand, a large body of material has been assembled on the geoscientific issues. The experience gained by SKB in the siting of the central interim storage facility for spent fuel (CLAB), the final repository for radioactive operational waste (SFR) and the Äspö Hard Rock Laboratory is also valuable in the ongoing work of siting the deep repository.

9.5.1 Previous site investigations

During the past 15 years, large resources have been invested in investigating the suitability of different geological environments for the final disposal of spent nuclear fuel. Some ten or so areas in Sweden have been subjected to extensive surveys (several deep boreholes). In addition, more limited investigations have been conducted at a number of other areas. Figure 9-6 shows where these sites are located in Sweden. A summary account of the investigations at these sites is provided in the background report and detailed data are published in a number of technical reports on the investigation results. These are presented in /9-2-12/. Besides the work that has been done within the nuclear waste management programme, a great deal of experience is available in Sweden from other underground rock construction projects.

Most of the investigated areas would probably qualify as sites for a deep repository, but there are differences which make the areas more or less suitable.

One important observation is that suitable or less suitable areas cannot be associated with any particular part of the country or any particular geological environment. Instead it is the local conditions in the area, and in the surrounding region, that determine the suitability of an area.

9.5.2 Siting of SKB's existing facilities

SKB is currently operating three facilities within the existing system for management of and research concerning nuclear waste, namely CLAB, SFR and the Äspö Hard Rock Laboratory.

The first two are nuclear facilities that have been sited and constructed in compliance with legislation applicable at the time, chiefly the Building Act and the Atomic Energy Act. Siting and construction of the Äspö Hard Rock Laboratory has been approved under the Act Concerning the Management of Natural Resources.

The history of the siting and construction of these facilities is presented in the background report.

It was made clear already in R&D-Programme 89 that the actual site of the Äspö Hard Rock Laboratory will not be considered for the siting of the final repository. If, however, suitable geological conditions are found in the vicinity, this may be one of the candidate sites where detailed characterization is carried out prior to final siting of the final repository.

The pre-studies that were done for the siting of CLAB and SFR focused at an early stage on the four nuclear power station locations and the Studsvik nuclear research station. The choice of these sites was based on the fact that all parties found it to be a considerable asset that nuclear activities were already established there. Barsebäck and Ringhals were not judged to be suitable for locating these facilities. The final choice of site was based on site-specific surveys of e.g. rock characteristics, transportation infrastructure, physical national planning, regional planning, the employment situation and municipal services.

The processing of permit applications, particularly with regard to SFR, has given both SKB and the regulatory authorities valuable experience. The concerned municipality had to make a preliminary ruling under the Building Act at an early stage. The technical review was then carried out under the Atomic Energy Act by the Swedish Nuclear Power Inspectorate and under the Radiation Protection Act by the Swedish Radiation Protection Institute. After the technical review had been completed, the matter was returned to the municipality for reconsideration and a final ruling. As a basis for this ruling, the municipality thus had access to the results of the technical review. The Government subsequently granted permits under the Building Act and the Atomic Energy Act.

The processing of the permit applications is described in greater detail in the background report.

According to the committee for review of legislation in the nuclear energy field, it should be possible to process permit applications for the deep repository in a similar manner under the Nuclear Activities Act, the Radiation Protection Act and the Act Concerning the Management of Natural Resources. The committee believes that the permit application processing procedure employed for SFR provided for sufficient coordination and that the regulatory authorities and SKB can be expected to profit from the experience gained.

9.6 PLANNED ACTIVITIES 1993–1998

A rough schedule for the activities during the next six years is shown in Figure 9-7.

The work is divided into general studies, pre-studies, pre-investigations, compilation of documentation for permit application under NRL, and, at the end of the period, planning and commencement of detailed characterization. The same fundamental issues relating to technology, geoscience, society, environment and safety are dealt with in each phase, but with a gradually increasingly site-specific and detailed focus.

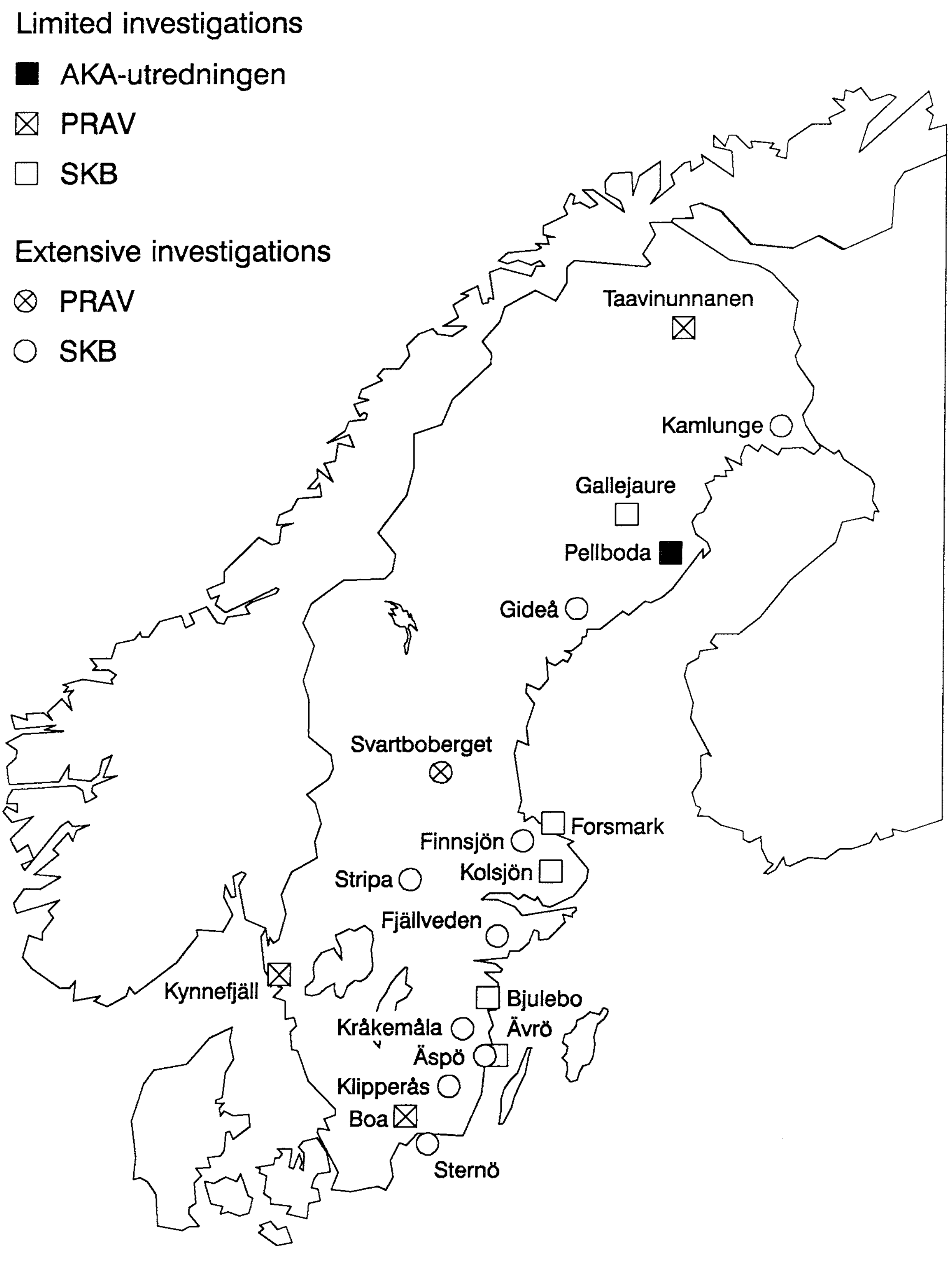


Figure 9-6. *Sites in Sweden where field activities have been conducted within the Swedish waste management programme to obtain knowledge on the properties of the Swedish bedrock and/or to develop and test investigation methods. The coming selection of candidate sites is not based on these sites but is based on the general knowledge that has been gained via the investigations.*

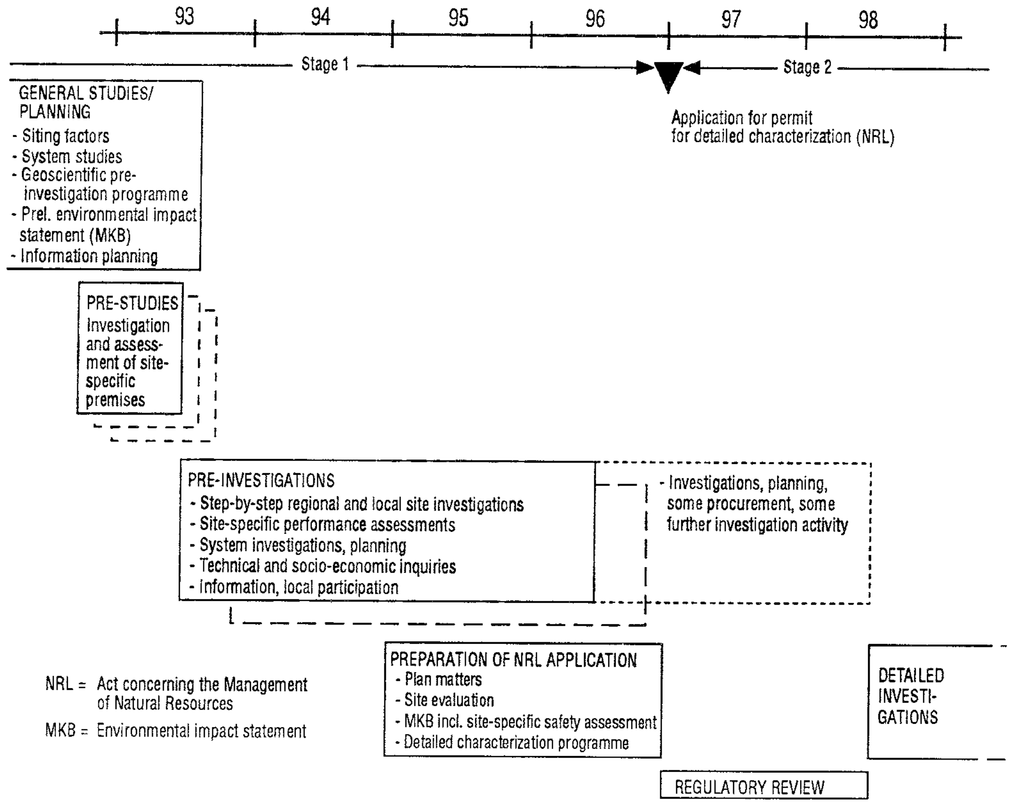


Figure 9-7. Activity plan and timeschedule for the siting work during the period 1993 – 98.

9.6.1 Planning and execution of geoscientific pre-investigations

A "Programme for geoscientific pre-investigations" is in the process of being prepared. The programme will be based on the collective knowledge and experience that exists at SKB within safety and performance assessment, handling technology, rock construction, geoscience, site investigations, measurement technology, data processing etc. It will be based on the premise that the repository is designed according to the KBS-3 concept and further that the repository site is located underneath a land area. Results from projects such as SKB 91, the Äspö Hard Rock Laboratory, the Stripa project etc. will serve as a basis for defining which site-specific parameters are to be determined and how the investigations are to be carried out.

The programme will contain the following points, among others:

- Investigation strategy.
- Technical investigation programme.
- Synopsis report (describes goals and expected results).
- Data handling.
- Organization.
- Administration.
- Work and method descriptions.
- Quality assurance programme.

Of these points, the first three deal with the contents of the pre-investigations, while the other points describe different kinds of tools for executing the programme.

The pre-investigations will be carried out in steps, which in the first place has proven effective for building up an understanding of geohydrological structures, and in the second place facilitates coordination with the safety assessment and the constructability assessment as well as the early planning and design work. Preliminarily, these pre-investigation steps will be:

- Regional characterization.
- Siting investigations.
- Fundamental investigations.
- Supplementary investigations.

Regional characterization

The purpose of the regional characterization is to obtain knowledge of large-scale geological and geohydrological conditions in the region surrounding the potential repository site. The investigations shall, among other things, provide data for boundary conditions to be used in calculations regarding groundwater flow and nuclide transport. The regional characterization shall also provide data for locating and describing discharge area(s) for the groundwater that passes by the repository and data for modelling of the present and future biosphere.

The regional characterization is mainly based on existing bodies of data, maps, aerial photos, aerial geophysics, well data, earlier survey results, etc. Depending on the scope of the data, it may need to be supplemented. The field surveys foreseen are mainly geological mapping and aerial and ground geophysical surveys.

Siting investigations

The purpose of these investigations is to provide a deeper picture of the geological, hydrological and geohydrochemical conditions on a few sites within the candidate area that have been identified by the previous general studies. The data shall, together with other non-geological factors, serve as a basis for proceeding with one site per candidate area.

The field measurements will consists of surface investigations and borehole investigations of limited scope, preferably 100-200 m percussion-drilled boreholes and perhaps one or two deep cored boreholes.

Fundamental investigations

This step is aimed at building up detailed geological and geohydrological models for the candidate site. The main body of basic data is collected during this step, through surface measurements and borehole investigations. The boreholes are positioned and aimed so that the most important fracture zones are intersected and so that good representativeness of directionally dependent parameters such as fracture directions is obtained in the borehole investigations. One or more of the holes should reach a depth of about 1 000 m. Cross-hole measurements will be an important source of data in this step.

This investigation step results in a conceptual model of the rock volume on the candidate site, with subject-specific descriptions of importance for safety and constructability assessments.

Supplementary investigations

This characterization step is aimed at checking and reinforcing the conceptual model by increasing the degree of detail and filling in gaps where the body of data has been limited.

The end result is used for a preliminary assessment of long-term safety and for constructability analysis and design. As mentioned previously, documentation shall also be compiled for an application for a permit for detailed characterization under the Act Concerning the Management of Natural Resources (NRL) and for a preliminary environmental impact statement (MKB).

Quality assurance programme

Quality-assuring a pre-investigation of a rock volume is not like quality-assuring a technical design. It is necessary to devise routines that are suitable for this type of activity so that the quality assurance is perceived as being a part of the project.

To assure quality in results, it is essential that the methodology for the measurement, calculation, analysis etc. is documented. This also applies to technical documentation of instruments. Work and method descriptions as well as technical documentation will be prepared where they do not already exist. Experience from the Äspö Hard Rock Laboratory will be drawn on in the work of quality assurance.

9.6.2 Technical and socio-economic studies

Aside from the purely geoscientific and safety-oriented investigations of a candidate site, a number of technical and societal aspects may have to be explored in order to determine whether a site is suitable and to describe the impact a deep repository may have on the environment and society in a broad sense.

An example is the societal impact on a community resulting from siting a deep repository there. Particularly in the case of a small town, this impact may be considerable. The establishment of what amounts to a local industry requires unusually long-range planning in this case. Demonstration deposition will, unless otherwise decided, be followed by an expansion to a full-scale repository. This means that the activities will be pursued for more than 50 years. The result will be an increase in both economic and social activities, which will create jobs and lead to some influx of personnel to the area. This in turn will have indirect effects on local small businesses and service firms.

The technical, economic and social aspects sketched above can have an impact on the local community and nearby residents. SKB's intention is to explore and clarify these effects as far as possible, in cooperation with the municipality and the concerned parties.

Studies of the above questions are best carried out when the candidate sites have been designated and cooperation has been established with the municipalities. Up until then, SKB plans to conduct studies of a more general nature around e.g. transport aspects, employment effects and environmental impact.

9.6.3 Environmental impact statements and safety assessments

Traditionally, and ever since the nuclear power plants were built, the nuclear power industry has evaluated the safety and possible environmental impact of their facilities and published these evaluations in special reports. Both in Sweden and abroad, preliminary and final safety reports have been prepared in conjunction with the building of nuclear power plants or other nuclear installations, such as SKB's final repository for radioactive operational waste (SFR).

The concept of an environmental impact statement (MKB) is relatively new in Swedish legislation. Regulations requiring environmental impact statements were introduced in environmentally-related legislation in 1991.

According to the Act Concerning the Management of Natural Resources, an environmental impact statement (referred to in the Act as an environmental impact assessment) shall *"facilitate the joint assessment of the impact of a proposed installation, of an activity or various other measures on the environment, on public health and on the management of natural resources"*.

The committee white papers, bills and laws that deal with environmental impact statements in Sweden place great importance on the fact that the environmental impact caused by an industrial installation shall be described in such a manner that it is understood by the general public. This distinguishes it to some extent from the safety assessments mentioned above. An MKB also has a broader scope than a traditional radiological safety assessment, which is more to be regarded as a part (an important part) of the background material on which an MKB is based.

The so-called "zero alternative", i.e. what it means for the environment if the facility is not built, shall also be assessed in an MKB.

SKB plans to prepare a preliminary environmental impact statement at an early stage of the siting work. The purpose is that it should serve as a basis for the discussions with the municipality, local population and regulatory authorities of the facility's environmental impact. In this way, SKB can obtain valuable viewpoints which can then be taken into account in the work with the formal permit applications. Table 9-2 shows and comments on the environmental impact statements and safety assessments that will be prepared. A more detailed discussion is provided in the background report and in chapter 10 on safety assessments.

9.6.4 Planning of measures for information and local participation

The law says that the nuclear power utilities are obliged to take all measures required to safely handle and dispose of the nuclear waste. In order to build the necessary facilities, the Swedish Nuclear Fuel and

Table 9-2. Plan for environmental impact assessments and detailed safety assessments.

Siting process	Environmental impact assessment (MKB)	Detailed safety assessment
Stage 1. General studies, Pre-studies, Pre-investigations	A preliminary MKB document is prepared at an early stage with a general assessment of the environmental impact of the entire process (investigations, construction, operation, closure, long-term effects).	The safety assessment in SKB-91 is supplemented with near-field analysis for the chosen alternative, section 6.5.
	An MKB is prepared in support of an application for a permit to conduct detailed characterization (NRL)	Analyses of bedrock conditions are performed for each site in conjunction with the execution of pre-investigations. A site-specific assessment of long-term safety is carried out and appended to the NRL application.
Stage 2. Detailed characterization	An updated MKB in support of an application for a siting permit (NRL) and a licence (KTL).	A preliminary safety report (PSR) based on the comprehensive data obtained during the detailed characterization and in connection with the planning of the facilities is compiled in support of an application for a licence under KTL.
Stage 3. Construction	Possible updating of existing MKB.	A final safety report (FSR) comprises support for an application for an operating permit under KTL.

Waste Management Company, which in practice bears the responsibility, must gain society's confidence in the methods developed by the scientists. It is therefore important to spread knowledge that enables the citizens to view radioactive waste in a rational perspective, which neither overrates nor underrates what is involved. Open and factual information is a prerequisite in meeting the justified demands of the public on insight and in the democratic decision-making process.

The general goal of SKB's information is to broaden and deepen knowledge in the community concerning:

- The radioactive waste: how much there is today, how much there will be in all and what dangers it poses.
- The system which SKB has built up and which is already taking care of all radioactive waste. (Transportation system, SFR, CLAB.)
- The scope and thrust of the work on future deep repositories which SKB and others are pursuing, the underlying ethical principles and the extensive knowledge that now exists concerning the possibilities of isolating the waste.

The people living near SKB's facilities SFR and CLAB receive information of a local nature. This is done in cooperation with the nuclear power plants near which SKB's existing facilities are located.

When activities are started in a community, SKB will provide information tailored for local needs. Local information contains, besides the more general facts that are presented in the nationwide information, detailed descriptions of what a facility may look like and how it will affect the community.

The information will describe, for example, how the surface part of the facility may be designed and how the need for infrastructure, communications, labour etc. can be met. The steps in the scientific investigations on the site, SKB's proposed joint consultation process and the legal licensing process will also be described. The information is intended to answer the questions nearby residents may have concerning such a facility.

The information programme may take the form of, for instance, local information offices, local newsletters, study groups, seminars and lectures in schools and at places of work, or organized study visits to the study site itself and to SKB's existing facilities. The information programme shall also provide feedback to SKB on which issues, viewpoints and questions are considered important by the local population.

To reinforce local information and local influence, a local body should be set up at an early stage, composed in a manner similar to that of the local safety committees which, according to the proposal in the committee report on changes in the Act on Nuclear Activities, will be involved from the detailed characterization phase. (A local safety committee is appointed by the Government for a term of up to three years. A maximum of ten members are recommended by the municipality in which the facility is situated, while a maximum of three are recommended by nearby municipalities that are affected in one way or another.) SKB will give such a body an opportunity to become informed concerning those issues that arise during the course of the siting process. The body can also take responsibility for compiling facts and information for the public, authorities and institutions at the local level.

10 SAFETY ASSESSMENTS

10.1 GENERAL

The nuclear activity must be carried out in an acceptable manner with respect to safety and radiation protection. In different phases of the development of a deep repository, the feasibility of the plans and proposals is reviewed, along with their technical and economic effectiveness and the radiological safety they can offer. These reviews comprise a basis for the management of the activity.

Safety is judged with the aid of performance and safety assessments. The performance assessments comprise studies of sub-systems and their chemical or physical interaction in their environment. External and internal environmental conditions under which performance or safety is to be evaluated are elucidated in scenario analyses. Scenario analyses and performance assessments comprise parts of the total safety assessment.

An integrated safety assessment is an important tool for clarifying the aggregate safety effect of the different barriers. The result makes it possible to evaluate the need for additional measures, for example with regard to the design of the repository, the execution of the barriers or additional research to reinforce the body of data or refine the calculation models. Safety assessments provide some of the background material for SKB's decisions and choices to implement the scheme, for the regulatory authorities' assessment of the progress in SKB's R&D work and for the regulatory authorities' rulings on permit applications.

SKB's RD&D programme is currently in a phase where the general studies of feasibility and safety have been concluded. The review of different alternatives for the disposal scheme has led to a narrowing of the focus of the work so that from now on it will concern a distributed repository with a tunnel system at a depth of about 500 m where the spent fuel is placed in corrosion-resistant canisters. The geometric configuration of the repository can be adapted to local conditions.

The analysis methods have been refined so that they can provide a basis for evaluation of repository sites and for how the repository should be situated and configured to take effective advantage of the site's natural capacity to protect the waste. The recently completed safety assessment SKB 91 showed that the properties that must exist in the rock on a candidate site to ensure long-term safety do not differ significantly from those that normally exist at many places in the crystalline basement rock.

10.2 GOALS

The assessment of the performance of the repository in various respects and the safety judgements obtained therefrom comprise a basis for the decisions that must be made in order to meet SKB's general schedule. This yields a schedule for when different types of further development and local adaptation of analysis methodology and calculation models have to be completed.

The goals of measures within the field of performance and safety assessment are as follows:

- during the pre-investigation phase of candidate sites, i.e. during the period 1993–1996:
  - to carry out analyses and evaluations of conditions of importance for constructability and the performance of the site as a protective barrier as a basis for continued site characterization and emplacement of the repository;
  - to carry out safety assessments during 1996 for site-adapted final repository facilities on two candidate sites, to a quality standard that is required for permit application under NRL for the detailed geological site characterization;
  - to evaluate during the period the need for further method development in preparation for future performance and safety assessments in accordance with SKB's general schedules, and to begin this method development.
- during the planning and design of the encapsulation plant, and during the siting and design of the deep repository for demonstration deposition, i.e. up to about 1998:
  - to carry out an initial safety review of the encapsulation process in connection with the system studies;
  - to broaden and augment this to a level of detail required for an application for a licence under KTL for the encapsulation plant by 1997;
  - to carry out a safety assessment of transportation, handling and deposition of the spent nuclear fuel and other waste (including possible retrieval and subsequent storage) by 1996, with adaptation to the studies for the encapsulation plant and the pre-investigations on the candidate sites.

After the ensuing 6-year period, safety reports for the deep repository shall comprise a basis for planned applications for a siting permit and an environmental licence in about 2003 and a permit for operation of the encapsulation plant and the deep repository in about 2008.

Planning and design of the repository for other long-lived waste will not begin until after demonstration deposition has commenced, i.e. not until after 2010.

10.3 STATE OF DEVELOPMENT AND NEED FOR MEASURES

10.3.1 Operating safety

Methods developed for safety assessments in the process industry and at nuclear installations are used for the assessments.

For radioactive waste, the safety of the scheme must be demonstrated for both an active handling phase (operating safety, including e.g. conditioning, storage, transportation and deposition of the waste) and a passive post-closure phase after the final repository has been sealed (long-term safety).

Methods and routines for safety assessment of systems in active operation have been developed, and are still being refined, within the nuclear power industry. They are judged to be sufficiently developed and tested for the account required for the licensing of all the handling of radioactive waste that is required according to SKB's system design. They have previously been employed in connection with the licensing of the transportation system, CLAB and SFR.

The technical experience that has been gained from previously safety-reviewed nuclear installations will provide good guidance for most of the handling and the measures that are adopted in the encapsulation plant, in the transportation system and in connection with the demonstration deposition in the deep repository. A number of operations are, however, untried, and development measures have been planned as a support for their design and planning. The development measures concern:

- Electron beam welding of the canister's copper jacket.
- Selection of material for possible post-filling of the canister.
- Fabrication inspection and non-destructive testing.
- Excavation of deposition positions, emplacement of canisters and post-emplacement inspection.
- Backfilling of tunnels.

The thrust and scope of these development measures is described in conjunction with the encapsulation plant (chapter 8) and the deep repository (chapter 9).

The assessment of operating safety in the deep repository will also include the handling procedures required for a possible retrieval and interim storage of the waste deposited during the deposition phase.

The link between operating safety and long-term safety consists of the quality obtained for the engineered barriers or the probability and scope of possible undetected fabrication defects.

10.3.2 Long-term safety

General

Methods for execution of long-term safety assessments for radioactive waste have been developed over a period of nearly two decades and have been applied in numerous major summary reports. The method development has been geared to the needs of the national and international programmes for management of wastes from nuclear energy production.

The assessment methods have been applied and further refined in a number of major integrated safety assessments in Sweden and other countries. Among the ones performed in recent years are KBS-3 /10-2/, Project 90 /10-3/ and SKB 91 /10-1/ in Sweden and Project Gewähr in Switzerland /10-4/ for disposal in crystalline bedrock; the PAGIS project in the EC embracing disposal in salt, clay, shale and granite; and the safety report for the WIPP in salt /10-5/ and for a potential repository in tuff at Yucca Mountain in the USA /10-6/. Moreover, an international study has been made of the feasibility of disposing of high-level radioactive waste in deep sea sediments /10-7/. At present a safety assessment is being compiled for a Canadian final repository for spent fuel, and updatings are being made of previous assessments for spent nuclear fuel in Finland and reprocessed waste in Switzerland.

During 1990, the OECD/NEA's Radioactive Waste Management Committee and the IAEA's International Radioactive Waste Management Advisory Committee examined the methods for assessing the safety of radioactive waste disposal systems and reviewed the experience available from the use of safety assessment methods on different repository types and in different geological environments.

The committees concluded /10-8/

- that safety assessment methods are available today to evaluate adequately the potential long-term radiological impacts of a carefully designed radioactive waste disposal system on humans and the environment, and
- that appropriate use of safety assessment methods, coupled with sufficient information from the proposed disposal sites, can provide the technical basis to decide whether specific disposal systems would offer to society a satisfactory level of safety for both current and future generations.

Within SKB, the assessment methodology has been utilized for the safety assessment in SKB 91 /10-1/. Compared with previous Swedish assessments, the performance of the canister and the transport of radionuclides in fracture zones have received an improved description. Furthermore, consideration has been given to the physical extent of the repository on the repository site and to the uncertainty that derives from spatial variability in the properties of the rock.

The safety assessment SKB 91 shows that a repository excavated deep down in the crystalline basement and with long-term-stable engineered barriers satisfies the safety requirements proposed by the authorities with ample margin. The safety of such a repository is dependent only to a small extent on the capacity of the host rock to retard and sorb leaking radionuclides. An adequate function of the rock is to provide stable mechanical and chemical conditions over a long period of time so that the long-term function of the engineered barriers is not jeopardized.

The safety-related demands that must be made on a site where a final repository is to be built are thus limited and are probably more than satisfied on most sites SKB has investigated in Sweden. The assessments show that the potential for extra safety that presumably exists in the far field can be affected by a number of factors. Examples of such factors are the presence and location of flat-lying structures and their hydraulic conductivity.

SKB 91 offers an example of how performance and safety assessments can be used to shed light on the importance of different geological structures in a potential repository area and to clarify factors that are essential from a safety viewpoint. The methodology can be utilized in a siting process to configure the repository so that the rock's capacity to contribute to the safety of the repository is optimally utilized.

A practical system for quality assurance was tested in conjunction with SKB 91 to organize data and models for the assessment. The emphasis was on documentation of input data and traceability. The test yielded experience of value for future assessments.

Since the end of the 1980s, SKI and SSI have, together with other Nordic nuclear safety authorities, worked to prepare a joint document governing principles and acceptance criteria for the final disposal of high-level radioactive waste. In SKB 91, these Nordic proposals have been utilized to relate the results of the assessments to acceptance criteria. SKB assumes that this work will continue.

Scenarios

Future changes in the repository system's external and internal environment may affect its long-term safety. In order to analyze possible changes in a systematic manner, a methodology has been developed where the repository system's fundamental features, future "sudden" events and processes occurring in the system [Features, Events and Processes (FEPs)] are identified /10-9/. Ties and links between different processes and events are documented both in text and graphically. From this overview of possible changes, an assessment is made of which scenarios are of importance in order to shed light on the system's safety. The consequences are calculated with numerical models.

An important question in scenario analysis for a repository is how it can be demonstrated that no phenomena or environmental factors of importance for the safety of the repository have been overlooked. To obtain the best possible comprehensiveness in the assessment, the background material for the choice of scenarios must be continuously updated. Broad international collaboration within the field of scenario analysis will be of great value in staying abreast of new approaches and ideas in order to arrive at a consensus on which scenarios are relevant and to evaluate the methodology that has been used in developing them.

Analysis models

A series of calculation models were utilized in SKB 91 to quantify the performance of the repository after it has been sealed. These calculation models and others are available for future analyses but may, depending on the nature of the near field and the features of the site, need to be modified or replaced. The available models can, often with only minor modifications, be utilized for all repositories with spatially distributed deposition of fuel in long-lived canisters.

Models and databases that can calculate the radionuclide inventory in an acceptable manner are already available today.

Once a canister and a repository design have been selected, an optimization of the modelling of these parts will be done as the methods for manufacture and inspection of the engineered barriers are defined in greater detail. The fuel studies will progressively reinforce the input data for the calculation of fuel dissolution.

Some refinement of models for groundwater movements is foreseen in order to permit effective utilization of the different types of input data that are expected to become available from the candidate sites at different points in time, for example interference tests, fracture mapping in tunnel walls, etc.

Refining of models for the transport of radionuclides in the bedrock will be pursued with the development and testing of models based on the channel network and discrete fracture concepts.

During the siting phase, it is not deemed necessary to depart from the simplification entailed by a biosphere model which is constant in time as posited in SKB 91, with the possible exception of a time-bound change in sea level. A local optimization of the modelling of the radionuclides' dispersal and availability in the biosphere will be done, however.

Experience from SKB 91 will be utilized to further simplify data and modelling in connection with safety assessments.

10.4 PROGRAMME FOR THE PERIOD 1993–1998

The goals during the period 1993-1998 entail in practice the following:

- Supplementary performance studies will be done of the near-field's (possibly also the far-field's) barriers with respect to
  - the prioritized near-field design, and
  - the progressive detailing of manufacturing, deposition and inspection methods that is obtained in connection with the planning of the encapsulation plant.
- The further refinement of analysis models and analysis methodology is planned so that the following studies associated with geological pre-investigations, see section 9.6, can be conducted on the first candidate site:
  - A general regional hydrogeological modelling in support of the regional characterization of the repository area.
  - Modelling, within the framework of the siting investigation phase, of, among other things, pathlines for groundwater as a basis for emplacement of a hypothetical repository on the candidate site.
  - During the pre-investigation's phases for fundamental and supplementary investigations, continuous evaluation will be made of the candidate site's hydrogeological conditions with the aid of the analysis models.
  - A safety assessment for a hypothetical repository on the candidate site will be prepared as part of the basis for the evaluation of the candidate site.
- A parallel effort with the same modelling tools will be undertaken for a second candidate site with a delay of about six months.
- Processing of the above material to a performance and safety assessment during 1996, which will be submitted in support of an application for a permit for detailed geological investigations on one of the sites. The safety assessment also comprises part of the background material for the updating of the environmental impact statement which is due by the same time.
- Safety evaluation and compilation of safety reports for encapsulation, transportation and deposition of the waste are done in connection with the planning and design of the encapsulation plant and the deep repository. The quality of the engineered barriers and the consequences of possible accidents constitute links to the assessment of long-term radiological safety.
- The further refinement of models and analysis methods in preparation for the detailed characterization phase that is done on the basis of experience from the Äspö Hard Rock Laboratory and the pre-investigation phase will be planned so that it can be concluded during 1998.
- Continued follow-up of and participation in the international work within the field.

11 SUPPORTIVE RESEARCH AND DEVELOPMENT - SUMMARY

11.1 GENERAL

In this RD&D-Programme, SKB has chosen a design for the canister and the deep repository that will be given priority in the continued work. Furthermore, SKB presents a plan for siting of the repository and a model for commissioning of the repository in stages via early construction of a smaller part of the repository as a demonstration facility.

To carry out the siting of the deep repository for demonstration deposition that is described in chapter 9 and the safety assessments that are described in chapter 10, a certain amount of supportive research and development is required. A detailed programme for this is described in the Background Report, Detailed R&D-Programme 1993–1998, with exhaustive references. A summary of these R&D plans is presented without references in this chapter. Regarding method development for the safety assessment, however, see chapter 10, and regarding the Äspö Hard Rock Laboratory see chapter 12.

The supportive research and development that is planned aims at

- improving the knowledge base and skills in modelling processes of importance for the performance of the repository in order to better be able to quantify remaining uncertainties and safety margins occasioned thereby,
- furnishing data as a basis for detailed design of the systems included in the repository and the encapsulation plant,
- furnishing data as a basis for optimizing handling technology and fabrication procedures to existing conditions under ground and in a radiation environment,
- following up international developments in relevant fields of science and technology.

The research and development is planned so that a continuity is obtained in the work and an updating of the knowledge base and analysis methods is done in good time before major assessments of performance or safety. For the coming 10-year period, this means that summaries shall be ready

- for the safety report that is to be appended to the application under NRL for permission for detailed site characterization,
- for the application for a building permit for the encapsulation plant under KTL,
- around 2001 for the siting application and the environmental licence application for that part of the deep repository that is to be used for demonstration deposition.

Method selection, design and technology development are noted in this chapter but are mainly dealt with in conjunction with the planning and design of the encapsulation plant and the deep repository.

11.2 PROPERTIES OF SPENT NUCLEAR FUEL

11.2.1 State of knowledge and development needs

Spent fuel as a waste form is a given premise in all of the alternatives being studied. Studies of the stability and durability of spent fuel in groundwater are therefore an important part of the RD&D-Programme. The purpose of the experimental investigations that have been conducted since 1982 is to clarify the mechanisms behind the release of radionuclides from the fuel under mildly oxidizing and, above all, under reducing conditions. This knowledge will then be used to model fuel under repository conditions in both the long and the short time perspective.

Actinides

The results from the last ten years' experimental programmes show that uranium under oxidizing conditions rapidly reaches a concentration of about $1 \cdot 10^{-5}$ M in carbonate-containing groundwater. Equivalent plutonium concentrations lie around $1 \cdot 10^{-9}$ M. These values indicate solubility control for both uranium and plutonium with separate solubility-limiting uranium and plutonium phases. In deionized water the uranium concentration is very low and often below the analytical detection limit, i.e. below 10^{-7} M. The experimental set-up for these experiments is very simple, as shown in Figure 11-1. The situation for plutonium is different. The concentrations rise to slightly over 10^{-8} M, i.e. more than 10 times the plutonium quantity that is present in the dissolved uranium. Thus, under oxidizing conditions, the leaching of radionuclides is not limited by the uranium solubility.

The same low uranium concentrations are measured under anaerobic conditions as in deionized water, i.e.

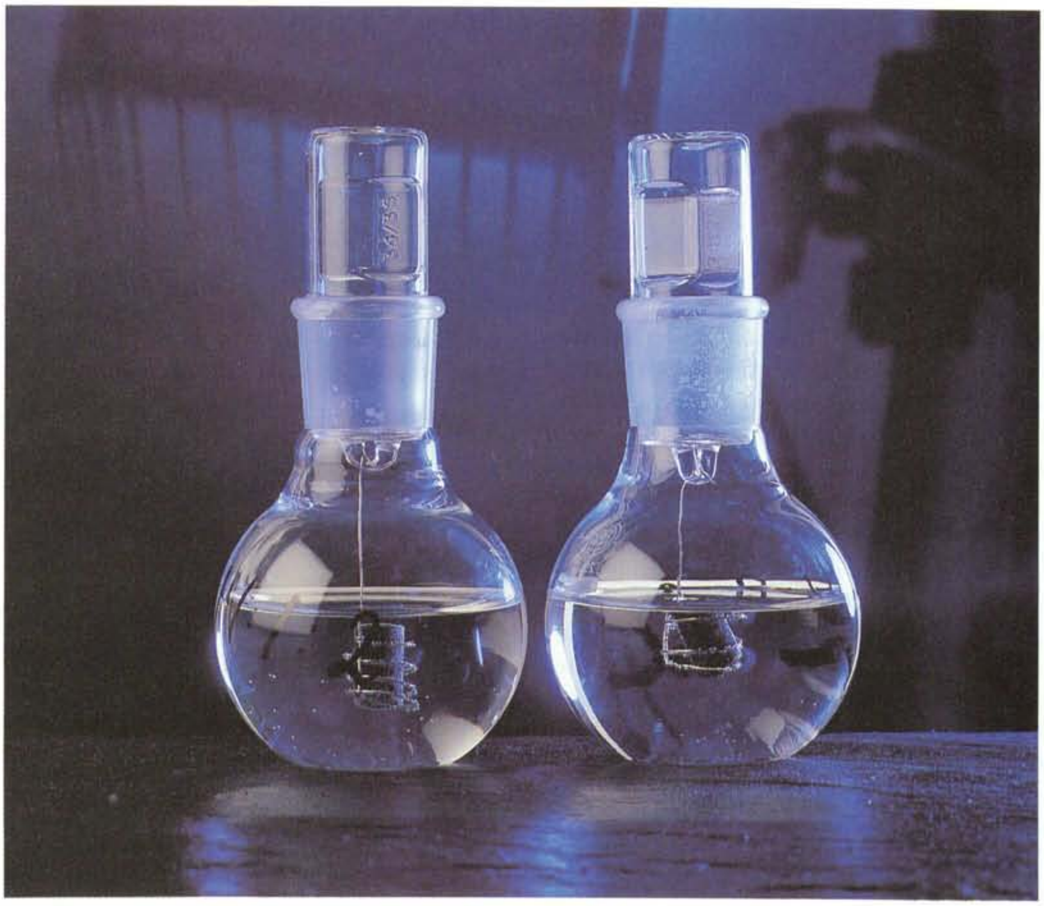


Figure 11-1. Experimental set-up for leaching of spent fuel under oxidizing conditions.

below 10^{-7} M. No plutonium can be detected in these solutions, however. It is not yet entirely clear whether this is because the plutonium dissolution is limited by the uranium dissolution, or whether any dissolved plutonium has been sorbed or precipitated on the walls of the leaching vessel. Improved means of analysis for plutonium, and above all for neptunium and uranium, can provide further information on the mechanisms behind fuel corrosion under both oxidizing and anaerobic conditions.

Fission products

The behaviour of the fission products can be illustrated with caesium, strontium and technetium, each of which exhibits a different pattern. Caesium is segregated during reactor irradiation to a given fraction, typically about 1%, in the grain boundaries and in the gap between the fuel and the Zircaloy cladding. This caesium will be easily accessible for leaching when the uranium dioxide comes into contact with water.

A very rapid initial release of caesium can be observed in the leaching experiments as well. After this pulse, the release rate declines rapidly under oxidizing conditions, falling to 10^{-6} of the inventory per day after about 1000 days. Current data indicate that the leach rate stabilizes at this level.

Strontium is regarded as a fission product which is in solid solution in the uranium dioxide and should therefore be released through matrix dissolution. When the fuel comes into contact with water, strontium is released to begin with at a constant rate, but after about 14 days the release rate here begins to decline as well. After 1000 days it has reached down to 10^{-7} of the inventory per day. Unlike caesium, strontium leaching has not yet shown any tendency to stabilize.

The leaching of caesium and strontium is affected very little by the redox conditions during the experiment. It is not entirely unexpected that the oxidation and dissolution of the UO_2 matrix does not appreciably affect the leaching-out of caesium, which is enriched in the grain boundaries and cracks in the fuel. In the case of strontium, on the other hand, this observation is more difficult to explain. Either there is a fraction here as well that has been enriched in such zones in the fuel that it can be dissolved without the UO_2 matrix having been appreciably affected. Or the strontium release may be an indicator of an ongoing oxidation of the fuel matrix, with accompanying leaching-out of non-solubility-limited elements. Under reducing and anaerobic conditions, the strontium leaching would then represent the influence of radiolytically produced oxidants. Additional work is required to clarify the processes that control the leaching of caesium and strontium.

Technetium exhibits a behaviour that differs from that of both strontium and caesium. Under oxidizing conditions the leach rate is constant and independent of the contact time, with a spread between 10^{-5} and 10^{-6} d^{-1}. After long contact times, this is a higher fraction than for both caesium and strontium and indicates that the mechanisms for technetium leaching are not the same as for these two elements. It is known that technetium forms separate phases in the fuel together with molybdenum, ruthenium, rhodium and palladium. The technetium leaching is probably controlled solely by oxidation of these inclusions. Under reducing conditions, the technetium concentration is drastically lower and is comparable to the solubility of TcO_2.

Model development

A model of fuel corrosion was used in SKB 91 that assumed oxidative dissolution caused by radiolytically produced oxidants resulting from the alpha radiolysis of water. The strontium release under oxidizing conditions was used as a very conservative upper limit for the oxidation rate, and thereby the release of radionuclides. With these premises, the time required for complete conversion of the fuel was found to be several million years.

There is, however, no clear evidence that strontium leaching is really a good measure of fuel oxidation. If strontium segregates in the fuel, the oxidation can be much lower than has been assumed. Nor has it been clarified what effect radiolytically produced oxidants have on the UO_2 matrix. It may be much less than has been assumed in SKB 91. Certain data indicate that it is so low that a model for oxidative dissolution is not applicable.

Natural analogues

A model must be based on a satisfactory insight into those mechanisms and processes that control the fuel dissolution. Such knowledge can be obtained through experimental investigations. Studies of conversion processes in natural uraninite deposits supplement the experiments and provide an opportunity for qualitatively describing the development of the weathering products over long periods of time and relating this to the geochemical environment.

11.2.2 Goals for the period 1993–1998

Present-day knowledge and data can be utilized to determine an upper limit for the release of the radionuclides from spent nuclear fuel. The fuel studies are continuing with the goals

- to progressively refine the models for the safety assessment in 1996,
- to develop a realistic model for radionuclide release from the fuel by the end of the 1990s in time for the application for an environmental licence for the deep repository for demonstration deposition.

11.2.3 Future research

During the next few years, the emphasis in the fuel studies will be on:

- Studies of corrosion of high-level fuel under different redox conditions, temperature and chemical environments, combined with investigations of the fuel's microstructure and fission product distribution before and after exposure to water.
- Clarifying the influence of radiolysis on the redox conditions at the fuel surface and the influence of radiolysis on fuel corrosion.
- Increased use of natural analogues to understand the long-term processes connected with fuel corrosion, i.e. the formation of weathering products on uraninite under mildly oxidizing and reducing conditions. As a part of this work, thermodynamic data will also be collected for uraninite's potential weathering products under repository-relevant conditions.
- Intensified work on modelling of fuel dissolution under repository-relevant conditions.

The studies of spent fuel are being conducted in close contact with other countries, especially the USA and Canada. Within model development and within the radiolysis studies, direct cooperation is taking place between SKB and AECL in Canada in the form of joint projects.

11.3 CANISTER AND CANISTER MATERIALS

11.3.1 General

A number of different canister materials have been studied since 1976, both ceramic and metallic. Several of these have proved to have very good corrosion resistance, for example copper, Al_2O_3, glass-ceramic and titanium. For the ceramic materials, delayed failure was early identified as a possible and difficult-to-foresee failure mechanism. Delayed failure is caused by slow crack growth from initial defects.

Several attempts to estimate the rate of crack growth in a ceramic canister have been made, above all in Sweden and Switzerland. Tested Al_2O_3 materials proved to have estimated lives that vary by many orders of magnitude. The differences are probably due to differences in grain size, impurities, sintering procedures etc. The measurements were made on small test specimens, but it is well known that bodies with larger volumes and surface areas have shorter lives. Moreover, in order to be able to judge the service life of a full-sized canister, it is necessary to know the distribution of stresses in the canister walls, which requires full-scale tests. In view of the very great uncertainties in obtained experimental results, determining a minimum service life for a ceramic canister with certainty would appear to be tricky.

The risk of delayed failure exists for titanium as well due to hydrogen embrittlement, but crevice corrosion is also possible in chloride-containing water. Titanium comprises the main alternative in Canada, where a great deal of effort has been devoted to determining the risks of delayed failure. The probability has been shown to be very low, even though it cannot be neglected entirely. Compared to copper, however, titanium does not appear to offer any clear advantages as a canister material in Sweden.

Copper is a material that lends the canister a very long life from the corrosion viewpoint, at the same time as it has satisfactory mechanical properties. The main alternative for a canister for the final disposal of spent nuclear fuel is a composite canister consisting of an outer corrosion protection of copper and an inner steel container, which lends mechanical stability to the canister. An alternative version is a lead-filled copper canister. Both of these canisters have the same corrosion resistance and therefore the same expected service life, but the lead-filled canister may have a slight advantage if the copper jacket is penetrated, owing to the extra barrier comprised by the lead fill. This is of secondary importance in view of the long life of the copper canister, but may be of some importance in the event of early canister damage. When this advantage is weighed against the disadvantages of the lead handling during encapsulation, however, the composite canister emerges as a better alternative.

11.3.2 State of knowledge and development needs

Material property

The creep properties of the material are of great importance for the final choice of copper grade. Pure oxygen-free copper has proved to have reduce creep ductility at elevated temperatures. Similar phenomena have not been observed for micro-alloyed oxygen-free copper. The question of creep deformation and creep failure are essential for both of the above canister alternatives and will require further research.

Corrosion - copper, steel

The corrosion properties of copper have been relatively well explored and will require only minor research efforts during the coming years. Some questions concerning local corrosion on the copper canister under mildly oxidizing conditions may need to be explored further, but since oxidizing conditions are only expected to prevail in the repository for a short time after deposition and sealing, these studies will not be given high priority. In addition to these activities, the premises for stress corrosion cracking on copper will require further elucidation.

There is also a certain risk of corrosion on the inside of the steel canister. With the limited quantity of corrodants that could conceivably be enclosed inside the canister, this does not constitute a problem if the risk of stress corrosion cracking can be neglected. Species that can induce stress corrosion cracking could, however, be formed by radiolysis of humid air inside the canister. This risk is eliminated in an atmosphere of inert gas.

The consequences of corrosion on the internal steel parts after penetration of the outer copper jacket have not yet been completely explored. This applies above all to pressure build-up caused by the growth of corrosion products and the consequences of hydrogen gas production in the repository. Furthermore, the risks of radiolytically induced stress corrosion cracking on the steel canister must be further studied.

Alternative materials - lead

As mentioned above, a lead-filled copper canister is an alternative to the composite canister. From the viewpoint of corrosion, this canister is naturally equivalent to the composite canister up until penetration of the copper jacket.

Remaining uncertainties mainly have to do with the fabrication technology. The most important questions concern the technology of filling a full-sized canister with lead and control of the solidification process for the lead. This will be studied by modelling the lead casting process and by means of practical tests, on a model scale to begin with.

Beyond this, the problems surrounding sealing by electron beam welding must also be explored. Of particular importance are the temperature conditions around the weld zone and in the upper part of the lead fill, where there is a risk of remelting of the lead.

11.3.3 Goals for the period 1993–1998 and onward

Copper has been chosen as an outer corrosion-protecting material in the canisters containing the spent nuclear fuel. The goals of the further studies of canister materials are:

- to select a copper grade on the basis of creep properties by 1996,
- to further refine knowledge and data regarding corrosion of copper and steel.

11.3.4 Future research

The future research activities will be focused on:

- completing the studies concerning;
  - the premises for stress corrosion cracking of copper,
  - corrosion and radiolytically induced stress corrosion cracking on the inside of the steel canister,
  - lead casting on a model scale,
  - local corrosion in an oxygen environment.
- selecting a suitable copper grade by 1996 on the basis of creep deformation and creep failure as well as weldability;
- supporting development of fabrication technology, welding technology and methods for non-destructive testing according to chapter 8.

11.4 BUFFER AND BACKFILL

11.4.1 General

The canister is surrounded by a clay buffer, which shall above all limit groundwater transport in the immediate vicinity of the canister. This limits the transport to the canister of the small quantities of corrodants that may be present in the groundwater, as well as the transport away from the canister of any nuclides that may dissolve. The buffer shall also create a suitable mechanical and chemical environment around the canister. The main candidate is a bentonite clay called MX-80 (Wyoming bentonite). Other qualities have also been studied and been found to have similar good properties.

11.4.2 Technology description and state of knowledge

During the actual deposition process, water uptake in bentonite is a disruptive phenomenon. A low inflow of water permits interruptions to be made in the deposition process and corrective action to be taken in case of unexpected events. A low inflow of water can be achieved by a suitable emplacement of the deposition positions and by grouting and sealing of the rock where needed. Sealing methods developed and tested within the Stripa project can be utilized for this purpose.

Sealing of access drifts and deposition tunnels is planned to be done by backfilling of the entire chamber with a mixture of bentonite and sand. The reference materials are MX-80 and quartz sand in ratios of 10/90 (lower part) and 20/80 (upper part). Methods have been developed and tested within the Buffer Mass Test in the Stripa project. Further tests are planned for compaction of the upper part of the drift.

The properties of the bentonite are of crucial importance for the function of the buffer. The buffer material around the canisters will be emplaced in the form of compacted blocks. Studies of this material have thus far focused on the bentonite's

- degradation processes,
- physical properties,
- hydraulic transport properties and
- interaction between buffer/backfill and rock.

Different models have been devised and their reliability tested by means of experiments. Certain both fundamental and modelling questions remain on the programme.

Additives to bentonite have been tried to improve the bentonite clay as a diffusion barrier. The value of such additives has been found to be dubious.

11.4.3 Goals for 1993–1998

The goal during the period is to present in 1995

- a summary of the most essential properties of various bentonite buffers,
- the methods that are to be used to determine these properties, and
- models for calculation of degradation processes, influence of other substances (cement, salt, groundwater etc.), homogenization after water saturation and mechanical protection in connection with various movements in the bedrock.

In addition, the technology for manufacturing compacted bentonite blocks, the deposition technology and the plugging technology shall be fully developed by the year 2000 for use in demonstration deposition.

11.4.4 Future research

During 1993–1994, a study being conducted together with NAGRA of the influence of cement on bentonite clay will be completed. The study focuses above all on the chemical degradation of the bentonite.

During 1995–1998, further studies will be conducted of:

- Transport processes. Ion transports and the changes caused by them will be described with the aid of a general microstructural model.
- Mechanical processes. Calculations will be made using the finite element program ABAQUS. The homogenization process in compacted bentonite remains to be modelled.

Furthermore, the planning and design of the deep repository will be supported by

- studies of manufacturing technology and suitable block size for the highly compacted bentonite around the canisters,
- studies of grouting of fine fractures with bentonite or cement to limit water seepage during the construction and deposition phases,
- certain technology studies of backfilling of drifts and shafts with sand/bentonite mixtures, espe-

cially for backfilling in the roof regions of the drifts.

11.5 GEOSCIENCE

11.5.1 General

Geologically seen, Sweden is situated within the Baltic Shield, which is dominated by very old crystalline rock types. Granites and gneisses dominate, and these rock types are usually older than 900 million years.

The bedrock has a number of central properties that are exploited to obtain the desired repository performance and safety. These are:

- Mechanical protection.
- Chemically stable environment.
- Slow and stable groundwater flux.

These properties can be more or less coupled to each other through physical or chemical processes.

SKB's geoscientific R&D work shall satisfy the need for knowledge and data pertaining to construction-related questions, repository and barrier performance and the long-term safety assessments. The programme covers knowledge-building within geology, geophysics, rock mechanics and geohydrology. The programme also includes method and instrument development, as well as further development and testing of numerical calculation models.

The work that has been done since the 1970s, aimed at final disposal of spent nuclear fuel deep down in the bedrock, has shown that good scientific and technical conditions exist at many places for carrying out such a disposal. During the period 1993–1998, the geoscientific work will mainly concern continued knowledge-building for applications within the programme for siting of the demonstration repository. Furthermore, the conclusions from the SKB 91 safety assessment will be followed up.

For the siting process, there is reason to further systematize essential geoscientific criteria. Based on experience from previous study site investigations, from Stripa and from Äspö, programmes (including method selection) for pre-investigations and detailed characterization will be developed. The constructability of a selected site for the repository shall be able to be judged with good reliability.

As far as geological material for safety assessments is concerned, conceptual models are planned to be further refined and adapted to the numerical calculation tools during the coming research period. Scaling problems and volume representativeness of different properties in the bedrock will constitute central questions in the model work. Similarly, it is essential to view the performance and long-term safety of the repository in its regional hydrological context. A further aspect that needs to be investigated is what methodology should be applied to judge movement-proneness of fractures and fracture zones around a repository.

11.5.2 State of knowledge

A general description is given below of the activities relating to

- groundwater movements,
- stability of the rock,
- calculation models, and
- method and instrument development.

Groundwater movements – conceptual modelling

Regional flow and transport conditions are being dealt with within the framework of the general geoscientific activities. Site-specific questions are being investigated at the Äspö Hard Rock Laboratory, see chapter 12. Nuclide transport and redox-related flow problems are described and commented on in section 11.6.

During the past decade, both theoretical and experimental studies have contributed towards increasing our understanding of the heterogeneous flow process in the rock fractures. As computers have become more powerful, geostatistics, including the processing of large quantities of data, has progressed. An endeavour to increase the site-specific deterministic element in the calculation work, where this is possible, can be foreseen for the near future.

Stability of the bedrock

Site-specific rock mechanics questions at a repository are mainly being dealt with at the Äspö Hard Rock Laboratory, see chapter 12. SKB's other general geoscientific activities include more fundamental strength-related studies and tectonic assessments.

During the many millions of years the Baltic Shield has been in existence, it is highly likely that Sweden's crystalline basement has been subjected to stresses in all directions. Plate tectonics, sediment covers and successive glaciations have loaded the Proterozoic bedrock in its brittle state so that to the extent new movements occur, they occur mainly in existing fault structures or fractures. In the event of such a reactivation, it is only the relative movement between separate rock blocks that is of interest. In a 100 000-year perspective, the relative movements that are expected between rock blocks at the 500 m level have very small displacement sums. A study in southeastern Sweden shows that in 450 million years, only a small fraction (10%) of the fractures have moved, and then with a maximum displacement sum of only 5 cm.

In northern Sweden there are neotectonic structures with greater movement sums, such as the Lansjärv Fault. The cause of these relative recent movements is probably a combination of rapid deglaciation and compression from the nearby Mid-Atlantic Ridge. There are indications of postglacial earthquakes. The movements in the Lansjärv area have been interpreted as successive reactivations in existing fractures and faults. It is uncertain over how long a period of time the reactivation has occurred.

SKB and Teollisuuden Voima OY (TVO) in Finland have jointly surveyed the international state of knowledge regarding the occurrence of ice ages. Based on a ice-age scenario developed by them, it is now possible to carry out approximate regional model calculations with an emphasis on rock mechanics and groundwater hydraulics.

Different methods for dating the most recent movements in fracture zones have been developed during the 1980s. An increased usage of these methods can be foreseen in the future.

Geohydrological and rock-mechanical calculation models

Analytical and numerical calculation models for the behaviour of groundwater in crystalline bedrock and for rock-mechanical conditions are constantly being refined within SKB. More site-specific model development is being done at the Äspö Hard Rock Laboratory, and the models for the different barriers are being coupled together within the framework of SKB's safety assessment activities. The use of stochastic modelling has increased over the past decade when it comes to groundwater hydraulics. Current models take very little account of the volume representativeness of input data. It is expected that the results of more large-scale pumping tests – interference tests – will have greater importance for the model structuring in the future. Today the results of the interference tests are coupled to discrete water-bearing zones, but there is reason to try to fit these pump results into stochastic model simulations.

There is interest in simulating how future glaciation periods affect the regional flow situation in a repository area. Specific problems for this modelling are boundary conditions in the form of, for example, groundwater recharge and sea levels. Furthermore, the conductivity field is affected by ice loads, freezing phenomena and increased pore water pressures.

Rock-mechanics modelling normally has a deterministic approach. Simplified rheological assumptions are included and little or no attention is paid to spatial variability in the properties of the rock. Furthermore, the scale dependence of the properties is seldom dealt with in calculations for e.g. tunnels or rock caverns. However, the models are also used for tectonic problems and on a detailed scale.

The impact of a nuclear waste repository on the rock mass during construction and operation involves thermal, hydrological and mechanical processes. These processes mutually affect each other to a greater or lesser extent. In recent years SKB has participated in the development of so-called "coupled models", which take these integrated processes into account.

Methods and instruments

Ever since 1977, SKB has been actively developing measurement methods and instruments for field surveys. The current state of knowledge within the area is based on many years of experience from surveys on study sites and within the Äspö project.

A relatively detailed review of the investigation technology was done in SKB's R&D-Programme 89. The application of pre-investigation methods used in the Äspö project has been described in collected form in a technical report. Drilling technology and borehole design have been developed by SKB so that contamination of groundwater is minimized and new hydrogeological measurement technology can be used. Examples are different types of controlled test pumpings in connection with flow logging to establish water-bearing levels and interference tests to determine the geometric extent of water-bearing fracture zones. Tracer tests have been used to determine groundwater flow and transport pathways, see Figure 11-2.

Borehole radar has been further refined and equipped with directional antennas so that the extent of fracture zones can be determined from a single hole.

Horizontal or subhorizontal fracture zones are of great importance for the flow of groundwater around a deep repository. It is therefore essential to be able to detect and locate these zones at an early stage. The measurement technology within this area still has limitations, which is why the potentially suitable methods of seismic reflection measurement and Vertical Seismic Profiling (VSP) should be studied further.

11.5.3 Goals for the period 1993–1998

SKB's geoscientific RD&D activities are aimed at satisfying the need for knowledge and data for long-term safety assessments, siting, repository performance assessment and for construction-related questions to a level and at the pace that are required for SKB's siting of a demonstration repository.

The primary goal of the work is:

- to further refine knowledge of hydrogeological and rock-mechanical conditions in order to permit better quantification of uncertainties and margins in the rock's capability to isolate the waste, as a basis for the siting process.

Important subsidiary goals are thereby:

- to further refine models for calculation of groundwater in fractured rock, for water flows in conjunction with glaciation and deglaciation, for coupled phenomena such as temperature, rock stresses and hydraulic conductivity and for rock

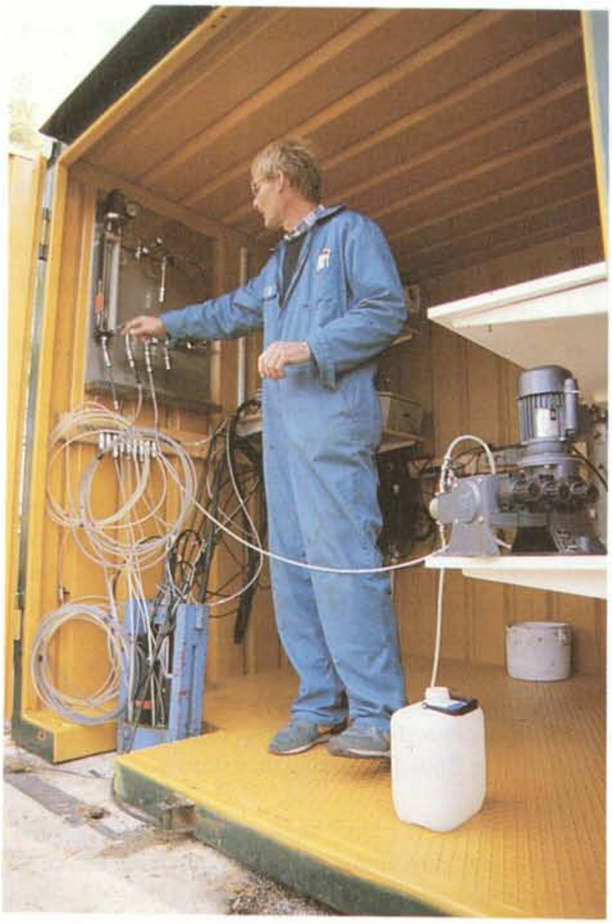


***Figure 11-2.** Different types of tracer tests are performed in or between boreholes down to a depth of 1 000 m. The picture shows injection of tracer in such a test.*

mechanics at a pace that is adjusted to the model need for assessments of the performance and long-term safety of the candidate sites.

- to see to it that suitable measurement methods and equipment are available for high-quality collection of such measurement data as are required to characterize the rock volume or volumes that will be investigated in preparation for the construction of SKB's deep repository for spent nuclear fuel.

11.5.4 Future research

Groundwater movements

The following activities are planned during the period 1993-1998 for the study of groundwater movements.

Methods for describing the geometry of individual fractures and their hydraulic properties will be further refined. This work will be pursued in the form of laboratory experiments and as a complement to the work being done at Äspö. Similarly, interpretation methods for analysis and determination of hydraulic properties in the field will be further developed. Special interest will thereby be devoted to scale problems and volume representativeness in connection with pumping tests.

The conceptual basis for numerical modelling will be examined with an emphasis on, for example, the dependence of the flow pattern on fracture mineralizations, rock stresses and former permafrost depths. Furthermore, the risks of brief pressure changes in the groundwater storage at repository level due to earthquakes will be investigated.

The hydrogeological premises (boundary conditions) for groundwater flow and transport at a repository in a regional perspective will also continue to be explored. Consideration will thereby be given to today's climatic situation as well as conditions during expected future glaciations and deglaciations.

Stability of the bedrock

The following research is planned within the field of bedrock stability.

A summary will be compiled during the coming research period of the principal load directions that have impacted the Baltic Shield during its historical brittle-tectonic period. The directions of dykes, previous sediment indications, erosion traces, fracture mineralizations etc. will be investigated in a regional perspective. Furthermore, the recent plate-tectonic processes and glaciations will provide background material for such a summary. The hypothesis is that the Baltic Shield has most probably been subjected to all conceivable stress directions, which would in turn mean that new fractures will not be formed in the upper part of the crust during the current tectonic regime.

A number of recent studies shed light on the, by international standards, very stable seismic conditions that prevail in Sweden. There is nevertheless a need to compile different Nordic studies into a joint document, where the risks of earthquakes are discussed and evaluated. A joint effort by TVO and SKB is planned for this compilation (cf. the joint ice-age scenario).

A continued knowledge-building process is planned for the different methods that are used for datings of the most recent movements in fracture zones. Besides different isotope techniques, ESR (Electron Spin Resonance) and paleomagnetic methods will be employed. The most suitable methods will then be used for detailed characterization within the siting programme.

Geohydrological and rock-mechanical calculation models

The following work is planned as far as mathematical models are concerned.

A study will be made of how the volume representativeness and dimensionality of hydraulic data can be incorporated in a model structure. Site-specific stochastic groundwater modelling with indicator simulation shall also be able to integrate and take into account general geological and geophysical information in the conductivity distribution.

A regional flow model is being developed for conditions during glaciation and deglaciation. The model is based on the scenario for future climatic conditions that has been devised at SKB.

Some work in a regional perspective will be devoted to convection models in fractured rock.

SKB will continue its involvement in the further refinement of coupled thermo-hydro-mechanical models. This will take place in part within the framework of the DECOVALEX programme initiated by SKI.

The measurement technology within rock mechanics is currently under vigorous development. A larger and better body of data can therefore be expected for future rock mechanics modellings. SKB intends to follow the development of more stochastic approaches that is taking place alongside of the sensitivity analyses within deterministic modelling that are used today.

Methods and instruments

Method and instrument development will cover the following principle areas:

- Studies and possibly testing of seismic reflection measurement and VSP in order to facilitate identification and characterization of horizontal structures.
- Further refinement of image processing in connection with borehole TV to improve the analysis of fracture orientations in boreholes.
- Evaluation of measurement methods and instruments for pre-investigations, on the basis of experience from the Äspö Hard Rock Laboratory.
- Further refinement of methodology for underground documentation and detailed characterization.
- Completion of ongoing development of a special chemical probe, CHEMLAB, for in-situ migration tests in boreholes.

11.6 CHEMISTRY/RADIONUCLIDE MIGRATION

11.6.1 General

The chemical environment is determined to a high degree by the chemical nature of the groundwater. This in turn is affected by reactions between the groundwater and the bedrock.

The prevailing chemical conditions affect:

- the durability of canisters and other engineered barriers,
- the dissolution rate of the spent nuclear fuel,
- the transport of dissolved radionuclides.

Should radionuclides in the waste be dissolved by penetrating groundwater, it is largely the chemical properties of the radionuclides that determine whether they will be transported with the water or be retained in the rock and decay.

Retention of radionuclides takes place through precipitation, sorption and diffusion. To determine the importance of radionuclide migration, such processes must be modelled quantitatively.

Retention of radionuclides in rock and backfill material can be prevented by the reaction of strong complexing agents with the nuclides, reducing their sorption tendency. The retention processes can also be short-circuited by adherence of the radionuclides to mobile colloidal particles in the groundwater. Microbes can also theoretically act in this manner. Microbes can also bring about geochemical changes.

11.6.2 State of knowledge

Groundwater chemistry

The groundwater in the shallower rock strata is fresh, whereas it is saline at depth. At coastal sites the boundary between saline and fresh water is near the ground surface. The saline water may be of marine origin, but a very long residence time can also make the water saline. This gives the water time to dissolve salt out of the rock. The fresh water is of meteoric origin. The history of the saline water has been studied in depth within the framework of the Äspö project.

Knowledge of the chemical nature of the groundwater has increased substantially in the past 10 years. In the early 1980s there was no knowledge of the distribution between saline and fresh water in the bedrock. Saline water of unknown origin appeared sporadically.

The quality of analyses performed improved dramatically when a mobile field laboratory for chemical analysis was designed and commissioned in 1984. The analysis unit was combined with borehole equipment for measuring the most sensitive parameters, pH and Eh, down in the sealed-off sampling section. The borehole equipment was subsequently supplemented with a gas sampler. The results obtained with the mobile field lab were different in essential respects from those obtained previously. The concentrations of bivalent iron turned out to be as high as the total iron concentration, i.e. there is no trivalent iron present in the deep groundwaters. The measured Eh values confirm that this is the case.

Changed drilling water marking facilitated the analysis of contamination by drilling water. The results of the analyses showed that the disturbance caused by drilling was far greater than had been believed. This led to modification of the drilling method and purge pumping.

All of the improvements described above led to better interpretation of the results. Since disturbances could be eliminated, it was possible to evaluate the water's residence time in the rock, based on chemical composition and isotope data. These interpretations show that there is no clear-cut residence time, but that the water is mixed continuously and that both very old and very young water is always present, although in different proportions.

Redox conditions are of great importance for the safety of the repository. Under prevailing reducing conditions, the copper canister is thermodynamically stable. Eh measurements and determinations of redox kinetics and redox capacity have been carried out for the purpose of understanding and confirming the redox conditions in a deep repository.

The Eh measurements that were made for KBS-3 indicated oxygen-free and usually slightly reducing conditions, which were assumed to be governed by the minerals in the rock. Continued measurements with improved measurement methods show that the iron system always determines the Eh value. The iron concentrations are in turn determined by the minerals in the rock.

The redox buffering capacity is of just as great importance as the Eh value itself. In KBS-3 it was assumed that this capacity consisted of the amount of bivalent iron minerals in the rock, 1–10% of the weight. However, it was not clear to what extent this buffer was available for reducing oxidized radionuclides. A calculation was therefore performed in KBS-3 where the released nuclides were allowed to migrate in oxidized form up to the biosphere. More recent measurements show that the available buffering capacity is less than the total quantity of iron (II) minerals in the rock, but much greater than the quantity on the mineral surfaces in the fracture walls. This is quite adequate for reducing radionuclides.

Fracture-filling mineral chemistry

The interaction between the minerals in the rock and the groundwater affects the chemical composition of the water. The mineralogy reveals the chemical conditions that preceded those existing today. Isotope data are also utilized in this work. Some elements are also used as analogues of radionuclides in the waste, for example naturally occurring strontium in fracture-filling minerals and groundwater as an analogy to radioactive strontium. In this way it is possible to see how a released radionuclide would behave in a rock fracture.

In KBS-3 a connection was assumed to exist between minerals and fracture infilling on the one hand and the chemistry of the groundwater on the other, even though no processes could be confirmed with certainty. Further work has led to a more nuanced picture, which shows that most mineral phases were formed under conditions that do not prevail today, usually hydrothermal ones. Knowledge concerning which minerals have interacted with today's groundwater has made it possible to utilize these few mineral phases for equilibrium modellings. The equilibrium modellings are aimed at verifying the stability of the observed chemical composition of the groundwater.

Radionuclide chemistry

The chemical form and properties of the radionuclides are determined by the environment in the deep repository. Uranium, thorium, neptunium, plutonium, americium, curium, protactinium, radium, technetium, nickel, niobium and tin are the elements with complex chemistry that are important from a safety viewpoint. Speciation, solubility, co-precipitation and kinetics are of essential importance. SKB now uses the computer code EQ3NR for equilibrium calculations with radionuclides.

The thermodynamic database used is largely the one that belongs to EQ3NR, but with SKB's own improvements. As a result of SKB's own measurements and the ones performed in international cooperation, the quality of the data has been improved for uranium, plutonium and technetium, among other elements. The measurements now extend not only to carbonate and hydroxide complexes, but also to phosphate. The reduction kinetics for uranium and technetium have also been examined. For example, it has now been shown that pertechnetate is reduced in the geochemical environment in the rock, a fact which was not known with certainty in KBS-3. There is better data in support of co-precipitation, but it is not yet good enough to enable this phenomenon to be used as a barrier property in a safety assessment.

The radiolysis experiments performed since KBS-3 confirmed for the most part the assumptions made there.

Organic complexes, colloids and microbes could conceivably bind radionuclides and transport them with the groundwater flow without retardation. The state of knowledge in this area has been substantially improved since the beginning of the 1980s. The concentrations and nature of colloids and humic substances in the groundwater have been investigated on a large number of sites both inside and outside Sweden. Laboratory experiments with radionuclides and colloids have been conducted and the humic substances' complexes with metal ions have been studied. It has been shown that radionuclides are sorbed by colloids and bacteria in the groundwater, but that these processes are not of safety-related importance for radionuclide migration.

Not until after KBS-3 was it shown that microbes are present in deep groundwaters. They include sulphate-reducing bacteria and methane bacteria. The bacteria occur in the water and on mineral surfaces in water-bearing fractures and may be of importance for geochemical conditions. Chemical changes caused by bacteria can, for example, affect canister corrosion and fuel dissolution. These investigations are therefore being given priority.

Sorption and diffusion are processes which, like solubility and precipitation, affect nuclide migration. Sorption and diffusion processes in rock are being studied in particular, but also in the materials, such as concrete and bentonite, that are being used for construction and backfill. A large number of diffusion experiments with radionuclides and bentonite have been concluded and evaluated since KBS-3. Better values have been obtained for slowly migrating radionuclides, e.g. plutonium. These have been used in new safety assessments and yield a better barrier effect for the buffer.

Additives to bentonite intended to improve the clay as a diffusion barrier have been tried. The value of these additives has been found to be dubious.

Concrete is an important construction material that is moreover a waste form for low- and intermediate-level waste. Concrete and its properties need to be studied further with a specific emphasis on long-lived waste such as core components etc.

Experimental studies have been commenced of the fundamental mechanisms of radionuclide sorption on mineral surfaces. The surface complexation model is being tried first. Sorption on mineral surfaces in rock, bentonite and concrete is an important barrier property. Experiments with complexing agents show that sorption is a relatively robust property. Strong complexing agents are of greater safety-related importance for e.g. solubility than for sorption. In other words, a complexing agent can greatly increase solubility without changing the sorption properties to a corresponding degree.

Validation of the processes in transport models and radionuclide migration

The transport of the radionuclides in the geosphere is dependent on the groundwater movements in the rock and the tendency of the nuclides to migrate with the groundwater. These two phenomena are completely separate from each other, but both contribute to nuclide migration.

The tendency of the nuclides to follow the groundwater flow is crucial; the velocity of the groundwater flow only affects those nuclides that adhere weakly in the rock. With non-sorbing tracers, travel times can be determined for validation of models for groundwater flow. Such tracer tests have been conducted at a number of different sites and evaluated within the framework of international projects.

Tests with diffusion of tracers in Stripa were conducted for several years and showed the existence of an interconnected pore system of microfissures etc. which is accessible for the in-diffusion of dissolved radionuclides. This mechanism, matrix diffusion, is an important retardation mechanism for non-sorbing radionuclides.

Experiments in Stripa and Finnsjön indicate that the water flow is limited to a number of flow paths in the rock. Rapid flow paths, "channels", can allow radionuclides to travel a long way before decaying. Channelling also limits the contact area, the "flow-wetted surface area", which dissolved radionuclides have with the rock. This affects the scope of both sorption and in-diffusion into the micropore system. A low value of the flow-wetted surface area was used in more recent assessments, such as SKB 91.

Tracer tests with technetium in Finnsjön confirm that reduction accompanied by high sorption occurs.

The technology for investigating hydraulically permeable zones in the rock with tracer tests has been improved. A large number of non-sorbing tracers in the form of dyes and short-lived radionuclides have been tested with success. With tracer tests it is possible, for instance, to see that the subhorizontal zone in Finnsjön, zone 2, is interconnected.

Natural analogues

The safety assessment shall be valid for hundreds of thousands of years. It is not possible to conduct experiments on that time scale, not even so-called "accelerated tests". Slow processes of importance for long-term safety could escape detection. One way to circumvent this problem is to rely on physical laws, such as thermodynamic restrictions. Another way is to make observations of natural phenomena where the conditions are similar and the duration of the process is comparable.

The latter are called "studies of natural analogues for deep disposal of radioactive waste" and include studies of both archeological finds and geological formations. According to a definition by the IAEA, natural analogues are "experiments in nature not controlled by man". References have been made to natural analogues in all of SKB's safety reports.

Since the mid-1980s, SKB has been involved in a number of international projects for the study of natural analogues in the form of mineral deposits. Large efforts have been made in Poços de Caldas, Cigar Lake and Oklo. Figure 11-3 shows one of the exposed natural reactors in Oklo, Gabon. The method of investigating geology, mineralogy, geochemistry, hydrogeology etc. in boreholes and then modelling the results does not differ much from what is customary in connection with geoscientific site investigations.

11.6.3 Goals for R&D in the field of chemistry

The two overall goals for the chemistry programme are to

Figure 11-3. The picture was taken at Oklo in Gabon and shows the site of a former natural reactor.

- clarify the chemical state in a deep repository in Swedish crystalline bedrock,
- determine the chemical and transport properties of the radionuclides in the prevailing chemical environment.

The chemical reactions take place in contact with the groundwater. A subsidiary goal is therefore to determine the normal chemical composition of the groundwater. Another subsidiary goal is to determine which variations in the composition of the water could be caused by natural processes or by the influence of the repository and its components.

To quantify the importance of radionuclide migration, data and models must be built up that describe the processes quantitatively. The work is planned so that revised data and models will be available for safety assessments in 1996 and 2003.

11.6.4 Future research

Groundwater and fracture mineral chemistry

A classification system for groundwater and fracture mineral data is being developed in cooperation with TVO.

The distribution of trace metals between groundwater and fracture-filling minerals is being investigated as an analogue for radionuclide retention.

Investigations of the chemical interaction between rock and groundwater and the influence of mixing processes are being given priority. Equilibrium modelling and sampling in nearly stagnant groundwaters from tunnels in Äspö are included in this.

The importance of acidification, redox reactions and microbial processes is being explored.

Radionuclide chemistry

Thermodynamic data for solubility and speciation of actinides in deep groundwaters will be determined. Co-precipitation and its importance for nuclide migration are being studied.

The groundwater's content of organic complexing agents, colloids and microbes is being investigated on Äspö and within the framework of international analogue studies. Laboratory experiments are currently in progress.

Sorption effects are being described with K_d data in transport modelling. Development of surface complexation models is in progress and may eventually replace the K_d concept.

Validation of the processes in transport models and nuclide migration

Groundwater flows are being validated with tracer tests with non-sorbing tracers. Further experiments of this kind will be carried out at Äspö. The natural tracers already present in the groundwater are being used for large-scale studies of groundwater flow.

Weakly sorbing tracers are needed for migration tests. Such tracers will be developed and tested in the lab for later use in field tests at Äspö.

A special piece of equipment, CHEMLAB, is being developed in cooperation with CEA in Cadarache for testing assumptions and models concerning the retention of radionuclides in the near field. The essential part of CHEMLAB is a probe that permits controlled experiments in situ on a small scale.

11.7 BIOSPHERE STUDIES

11.7.1 General

The biosphere studies cover the transport of radionuclides from groundwater in rock to impact on man or other organisms. The goal of these studies is to be able to describe the consequences of a postulated release of radioactive nuclides from a final repository.

The following processes and calculation steps are significant for the safety assessment:

- Transport from groundwater in rock to a local ecological system via various local recipients such as sediment, soil, water etc.
- Transport, dilution, accumulation and deposition in local, regional and global ecological systems.
- Transport to man through e.g. production and distribution of food.
- Uptake in the human body depending on dietary habits and uptake fractions.
- Calculation of individual doses and collective doses and comparison with natural conditions.

11.7.2 State of knowledge

Changes in the biosphere

The largest uncertainty factor in the biosphere has to do with the natural evolution of the ecosystems during the time the repository's function is of interest. In a long perspective, the picture is dominated by climatic changes and glaciation. The biosphere will then undergo a very radical change and can reappear in a very large number of forms. The question, however, is how meaningful it is to make dose estimates at this stage. Geological studies are being conducted and can yield a qualitative picture. The site-specific studies in particular need to be supplemented with an assessment of how man may come to exploit the ecosystems within a given area, for example what the land will be used for during the next 1 000 years.

Wells

Rock-drilled wells are a potential transport pathway from deep groundwater directly to environments in the vicinity of man. The well's interaction with the groundwater is handled with hydraulic models. The location of wells and their use are, however, linked to man's exploitation of the biosphere. The question of how wells are to be handled in the safety assessment is a part of the larger question of how man, intentionally or unintentionally, may affect the function of the repository. This is being discussed internationally and among concerned authorities to establish a common ground. Some further work is planned in surveying the occurrence and properties of the wells and how the water is used.

Transport through sediments and soil

The use of the K_d concept in dealing with the transport of radionuclides through soil is subject to certain objections, since the transport passes several different strata which are, moreover, not homogeneous. A study is under way to find an alternative way to model transport through sediments and soil. Site-specific quantitative descriptions of what different types of recipients exist and may possibly exist during the next 1 000 years will be of great use in enabling the number of different modelling cases to be limited.

Models and data

Dispersal calculations were carried in KBS-3 out using traditional box modelling based on available data on present-day ecological systems. Dose calculation is carried out with similar models employing radiophysical data and summarized in internationally accepted recommendations issued by the ICRP. SKB 91 employed a standardized, time-constant biosphere to weight the effects of the different nuclides and translate releases into impacts on the environment.

The uncertainty of the model predictions was not dealt with in KBS-3. In SKB 91 these uncertainties were calculated on the basis of assumed variations in input data, but were not used in the broader assessment. We need an integrated treatment of conceptual uncertainties, uncertainties in input data and natural variations, which BIOMOVS II will hopefully contribute towards.

Validation of models will take place continuously, for example internationally within VAMP and a continuation of BIOMOVS, and with the aid of data from the Chernobyl fallout in Gideå and Finnsjön. Some of the parameters used in the box models are poorly investigated for Nordic conditions. Proposals for a number of international standard biospheres are being prepared within BIOMOVS to facilitate comparisons.

Site-specific studies

For cases where the releases will not begin until after many tens of thousands of years, when ice ages will probably have occurred, the location of the repository cannot affect the dispersal of possible releases in the biosphere.

In the case of scenarios where radionuclides are assumed to be released within a few thousand years, for example in the case of an initial canister defect, the special conditions on the site will be important for dispersal in local ecosystems, since present-day premises for land use etc. can for the most part be expected to persist during this period of time.

It is therefore appropriate to continue the site-specific model studies and also to take into account the local impact of the final repository. Such investigations are planned to be conducted on the candidate sites that are proposed for siting of the final repository. The intention is that it shall be possible to use less generalized models as well.

Acceptance criteria

Widespread international agreement exists on the principles of radiological protection. However, practice in quantifying safety and expressing the safety requirements differs in different countries.

The choice of radiological acceptance criteria is of great importance for how biosphere analyses will be done. The Swedish authorities are expected to define the acceptance requirements for a deep repository during the next few years. This work and the work of the international bodies will be monitored closely.

11.7.3 Goals for the period 1993–1998

The R&D work will be concentrated on being able to make an estimate of what consequences different release scenarios from a deep repository have in a time perspective of up to about 10 000 years. Subsidiary goals in this process are:

- Try to quantify the uncertainties that stem from the fact that the biosphere is constantly changing.
- Improve the body of data on which the dispersal models rest.
- Validate the models through studies of analogous dispersal processes.

11.7.4 Future research

SKB will actively participate in ongoing international discussions of how allowability issues are to be handled when it comes to the impact of human actions on the future performance of the repository, for example through wells.

Ongoing studies of radionuclide sorption in soils will be completed in time for the 1996 safety assessment.

The work of examining the reliability of the models for radionuclide dispersal in the biosphere in BIOMOVS will continue.

The general properties of wells (location, capacity, depth etc.), correlated to how the water is used, should be described for today's situation and how it could possibly change in the future.

11.8 QUALITY ASSURANCE

Correctly applied and matched to the needs at hand, quality assurance is an excellent means of obtaining assurance that a job being done achieves the right, documented quality, and that information is actually documented and not wasted.

SKB early identified the need for quality assurance in parts of its activities. Formal quality assurance procedures have been introduced in those activities that are mainly concerned with the operation of existing systems and facilities. Documentation has been prepared for CLAB and SFR, as well as for the transportation system. A number of technical audits have also been carried out, among others one by the Nuclear Power Inspectorate. The procedures are currently being re-evaluated and will be compiled in a manual for all the activities being conducted within SKB's department for Systems and Facilities.

Quality assurance will be an important management instrument in different phases of the realization of an encapsulation plant and of a deep repository for demonstration deposition of spent nuclear fuel.

Quality assurance procedures are under preparation for the Äspö Hard Rock Laboratory. The application of these procedures will then serve as a foundation for the quality assurance of the investigation programmes in the siting project. Quality-assurance-like procedures are already being applied to the collection of data for SKB's database GEOTAB.

A system for quality assurance of the work in a safety study was tested in connection with the execution of SKB 91. The system focused on documentation and assurance of the numerical data that were used in the project. The primary intention was to gain experience for the safety studies that will be conducted in the future. An evaluation of the used system has been initiated and a modified system for quality assurance will be developed for future safety studies.

Quality assurance procedures have also been applied in the production of the software for the computer program PROPER with its models for radionuclide transport. An evaluation of the quality assurance need in conjunction with SKB's future production and maintenance of computer programs intended for direct use in safety assessments has been commenced.

12 ÄSPÖ HARD ROCK LABORATORY

12.1 BACKGROUND

The scientific investigations within SKB's research programme are necessary steps along the way of designing a deep repository and selecting and investigating a suitable site.

A balanced appraisal of the facts, requirements and assessments during the preparation of R&D-Programme 86 /12-1/ led to a proposal for the construction of an underground rock laboratory. This proposal was presented in the aforementioned research programme and was very positively received by the reviewing bodies.

In the autumn of 1986, SKB initiated the field work for the siting of an underground rock laboratory in the Simpevarp area in the municipality of Oskarshamn. At the end of 1988, SKB arrived at a decision in principle to site the facility on southern Äspö about 2 km north of the Oskarshamn Nuclear Power Station. After regulatory review, SKB ordered the blasting of the accessed tunnel to the Äspö Hard Rock Laboratory commenced in the autumn of 1990. In conjunction with the tunnelling work, which has now (September 1992) reached a depth of more than 200 m, a large number of investigations have been carried out.

The motives for the Äspö Hard Rock Laboratory and the goals of the project were presented in greater detail in R&D-Programme 89 /12-2/.

12.2 GOALS

Against the background of the motives presented, SKB decided to build the Äspö Hard Rock Laboratory, the purpose being to:

- create an opportunity for research, development and demonstration in a realistic and undisturbed rock environment down to the depth planned for the future deep repository.

The Äspö Hard Rock Laboratory shall constitute an important complement to the other work being conducted within SKB's RD&D-programme. The standards of quality in the research are very high. An overall ambition is that the Äspö Hard Rock Laboratory should become an internationally leading centre for research, development and demonstration regarding the construction of deep repositories for high-level waste.

The main goals for the Äspö Hard Rock Laboratory are (as stipulated in R&D-Programme 89) to further refine and/or test three different kinds of skills in preparation for the construction of a deep repository, more specifically the ability to:

- Test the quality and appropriateness of different methods for characterizing the bedrock with respect to conditions of importance for a deep repository.
- Refine and demonstrate methods for adapting a deep repository to the local properties of the rock in connection with design, planning and construction.
- Collect material and data of importance for the safety of the deep repository and for confidence in the quality of the safety assessments.

The last goal is general for SKB's entire RD&D-Programme.

To meet the overall schedule for SKB's RD&D work, the following stage goals have been set up for the activities at the Äspö Hard Rock Laboratory.

Prior to the siting of a deep repository for demonstration deposition in the mid-1990s, the activities of the Äspö Hard Rock Laboratory shall serve to:

1 Verify pre-investigation methods
- Demonstrate that investigations on the ground surface and in boreholes provide sufficient data on essential safety-related properties of the rock at repository level, and

2 Finalize detailed characterization methodology
- Refine and verify the methods and the technology needed for characterization of the rock in the detailed site characterization.

As a basis for a good optimization of the deep repository system and for a safety assessment prior to the siting application, which is planned to be submitted a couple of years after 2000, it is necessary to:

3 Test models for groundwater flow and radionuclide migration
- Refine and test on a large scale at repository depth methods and models for describing groundwater flow and radionuclide migration.

Prior to construction of the deep repository a few years after 2000, the following shall be done at planned repository depth and under representative conditions:

4 Demonstrate construction and handling methods
- Provide access to rock where methods and technology for guaranteeing high quality in the design, construction and operation of a deep repository can be refined and tested, and

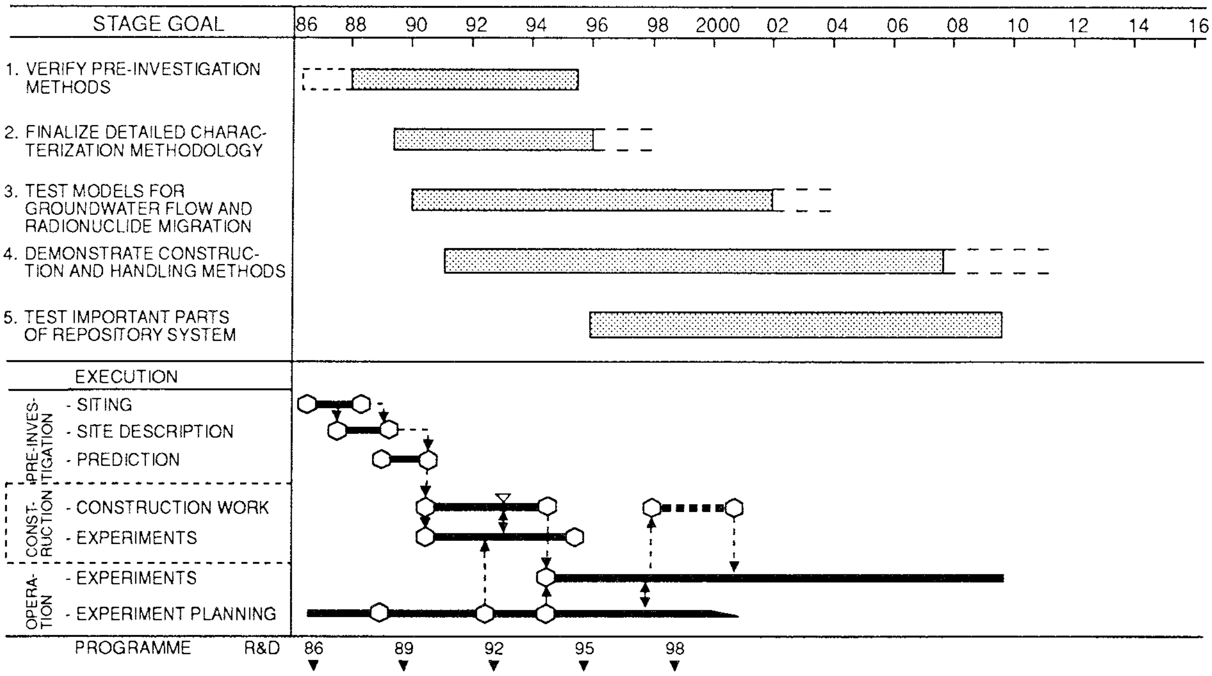


__Figure 12-1.__ Master timeschedule for the Äspö Hard Rock Laboratory, September 1992.

5 Test important parts of the repository system

– Test, investigate and demonstrate on a full scale different components that are of importance for the long-term safety of a deep repository system.

These tests shall be able to be carried out with sufficient scope as regards time and scale to provide the necessary support material for regulatory approval of the start of construction. Certain tests may therefore have to be started during the 90s.

12.3 RESULTS - GENERAL

The work in the Äspö Hard Rock Laboratory has been divided into three phases: the pre-investigation phase, the construction phase and the operating phase (see the schedule in Figure 12-1).

In the **pre-investigation phase** (1986–1990), a site was chosen for the Äspö Hard Rock Laboratory. The natural conditions in the bedrock were described. The project's construction and operating phases were planned.

During the **construction phase** (1990–1994), a number of investigations and experiments are being conducted in parallel with the civil engineering activities. The tunnel is being excavated to full depth, about 460 m.

The **operating phase** will commence in 1995. This RD&D-programme defines the thrust of the investigations and tests that will be carried out during the operating phase. The final programme for the operating phase will be adjusted on the basis of the results from other projects and experience gained from the construction phase.

12.3.1 Tallying against goals from R&D-Programme 86, R&D-Programme 89

The background to the construction of an underground rock laboratory was presented in R&D-Programme 86. The goals for the activities during the period 1987–1992 were also presented.

- Collect the geoscientific data required to determine whether it is possible to locate the Hard Rock Laboratory around Simpevarp and thereby meet the need for detailed characterization for validation;
- Collect the data required for the preliminary geometric layout of the Hard Rock Laboratory;
- Establish programmes for shaft sinking/tunnelling and measurements;
- Make a prediction of the geohydrological and geochemical changes that occur in conjunction with construction of the Hard Rock Laboratory.

All of these goals have been fulfilled /12-3, 4, 5, 6, 7/.

A more detailed description of the project was presented in R&D-Programme 89. The motives for the project were further elaborated. The goals of the project were defined in the form of main goals and stage goals. The goals are adapted to the results that are needed to construct the Swedish deep geological repository for spent nuclear fuel. A detailed programme for the pre-investigation phase and the construction phase, as well as a preliminary programme for the operating phase, were presented. The goals set up for the pre-investigation phase have been fulfilled. The goals set up for the construction phase are deemed to be in the process of being fulfilled.

In conjunction with the review of R&D-Programme 89, SKN requested supplementary reports regarding the Äspö Hard Rock Laboratory before commencement of the building works. Such supplementary reports have been submitted.

12.3.2 Siting, permits

The motives for siting the Äspö Hard Rock Laboratory on the southern part of Äspö were presented in R&D-Programme 89. In conjunction with the regulatory review, the Government announced that the facility was to be reviewed under the Act Concerning the Management of Natural Resources. To reduce the impact of the facility on its surroundings, SKB made layout changes so that the entrance to the facility was moved from Äspö to Simpevarp, see Figure 12-2.

Important permits and agreements pertaining to the Äspö Hard Rock Laboratory are now:

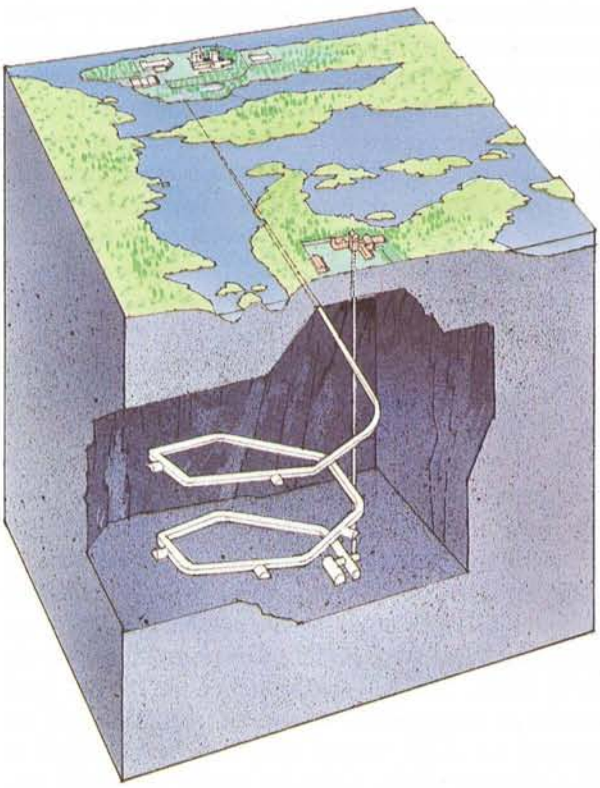


Figure 12-2. *Facility layout of the Äspö Hard Rock Laboratory.*

Act Concerning the Management of Natural Resources

A permit under the Act Concerning the Management of Natural Resources was obtained from the Government on 19 April 1990. An important point in the permit was that SKB should draw up a plan for any necessary handling of rock waste in consultation with the County Administration in the Municipality of Oskarshamn. For this purpose, a working group was formed with representatives of the County Administration, the Municipality, SKB and OKG which worked during the period 1990–1991.

Another stipulation for the permit is that "The construction and civil engineering measures that are required for the undertaking shall have been completed by not later than 31 December 1994".

Water rights ruling

A water rights ruling was handed down by the Water Rights Court in the Växjö District Court on 11 September 1990. An important stipulation is that a monitoring programme for the level of the groundwater table and water analysis shall be carried out. Private wells shall be monitored through 1995. The project's "own" boreholes on Ävrö, Äspö, Laxemar and Hålö/Bockholmen shall be monitored through 2004.

Building permits

Two sets of building permits have been obtained from the Municipality of Oskarshamn: One on 20 June 1990 (regarding the establishment works on the Simpevarp Peninsula) and one on 17 October 1990 (Äspö Village and the rock facility).

Land-use agreements

Several agreements exist with OKG regarding the use of land for the site office, access tunnel, roads, temporary rock waste dumps, harbour etc.

12.3.3 International participation

One important result is that the Äspö Hard Rock Laboratory has attracted international participation, see further the special section on international participation, section 12.8.

12.4 RESULTS - VERIFICATION OF PRE-INVESTIGATION METHODS

Quite a few results have already been achieved in the project. These results are reported in relation to the stage goals presented in section 12.2.

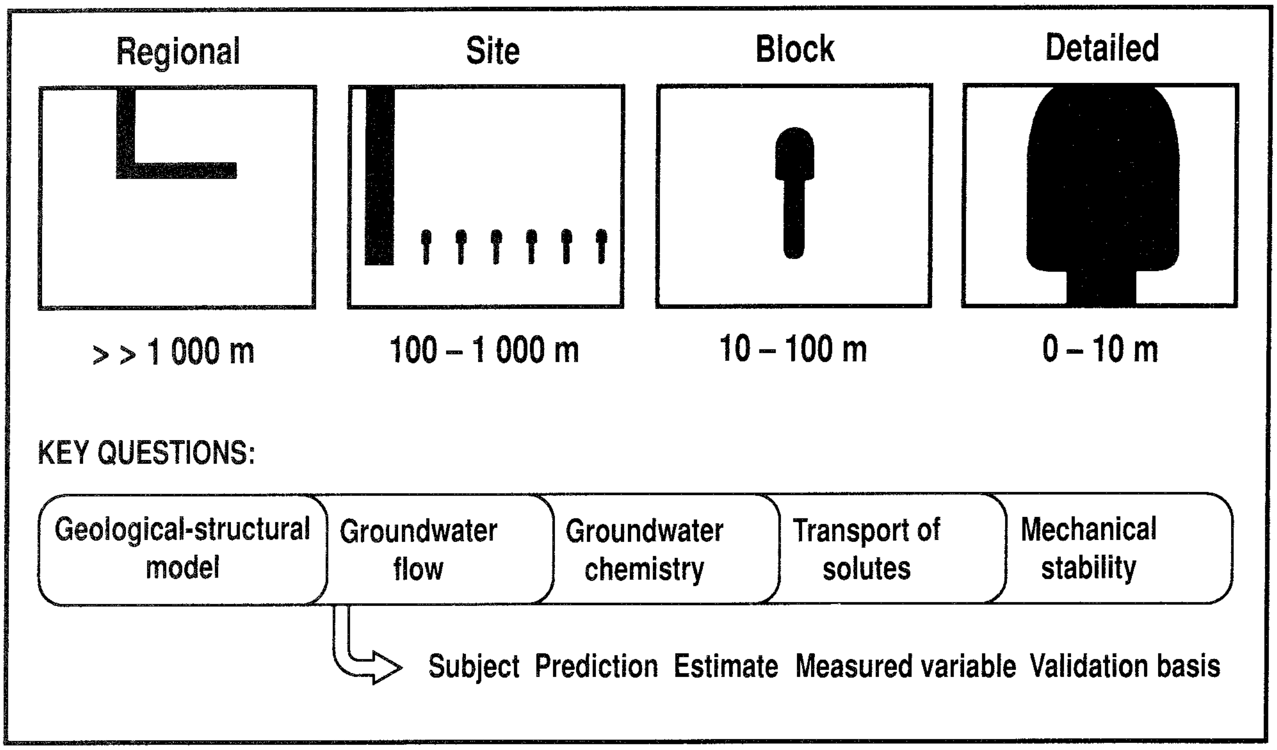


Figure 12-3. Overview of geometric scales and key questions.

Pre-investigations of the bedrock and their interpretation are of great importance for designing the deep repository, planning future investigations and assessing the suitability of the site for a deep repository. For this reason, SKB has defined validation of the pre-investigations as one of the stage goals for the project.

The concept of validation has been defined by a number of international organizations and further refined within the Äspö Hard Rock Laboratory.

The IAEA's definition /12-8/ is that "validation is a process carried out by comparison of model predictions with independent field observations and experimental measurements. A model cannot be considered validated until sufficient testing has been performed to ensure an acceptable level of predictive accuracy." An additional comment was made within the HYDROCOIN project /12-9/ to the effect that an acceptable prediction accuracy is a matter of judgement and varies with the specific problem being tested in the model.

The validation process for the Äspö Hard Rock Laboratory has been further refined within the project and by the project's Scientific Advisory Committee /12-10/.

- a systematic comparison of prediction and outcome,
- a careful scrutiny of the underlying structures and processes,
- a (subjective) judgement of whether the prediction is good enough.

The above validation process is ultimately a tool for evaluating the accuracy of the pre-investigations and assessing the capabilities and limitations of different investigation methods. This knowledge then provides guidance in establishing a balanced pre-investigation for the future candidate sites for the deep repository.

A large number of predictions concerning the bedrock that were judged to be relevant for a deep repository were defined in /12-10/. Another criterion was to only make predictions that can later be checked. On the basis of the discussions that were carried on within the project during the period 1987–1989, it was judged fruitful to structure the so-called "conceptual" models to obtain clarity.

The predictions were structured according to different geometric scales and different key issues, see Figure 12-3.

Rock investigations embrace a number of investigation methods. Each method gives rise to its data quantity with associated interpretation. By co-evaluating the methods, the interpretations can be sharpened considerably. Within the Äspö Hard Rock Laboratory, such major co-evaluations have been performed on three occasions during the pre-investigations /12-12, 12-3, 12-5/. The last interpretation /12-5/ has then served as the basis for systematic predictions of the bedrock on Äspö. These have included location of major fracture zones, hydraulic impact around the tunnel, changes in groundwater chemistry etc., see further /12-14/. These predictions are now being checked and reported on during the construction phase in different stages. At this point the quality of the predictions has been checked for one sub-stage /12-13/. This report covers the first 700 metres of the access tunnel, based on relatively limited investigations. The outcome has been judged by the Scientific Advisory Committee to be relatively good in view of the limited investigations. The results from tunnelling between 700 m and 1475 m are currently being evaluated.

Considerable development work has been carried out in connection with the execution of Verification of Pre-investigations. This includes areas such as conceptual understanding, reporting of investigation results, measurement technology, instruments, numerical models, see further /12-14/.

12.5 RESULTS - DETAILED CHARACTERIZATION METHODOLOGY

SKB is planning to carry out the site investigations for the candidate sites in stages. After initial pre-investigations, more detailed investigations will be conducted. This "detailed characterization" involves excavating a tunnel/shaft to the planned repository depth. Supplementary data are collected during the excavation process so that a solid body of background data exists for planning and designing the deep repository and as a basis for assessments of the performance and safety of the repository on the investigated site.

The construction of the Äspö Hard Rock Laboratory has already yielded valuable experience regarding how detailed characterization should be carried out. A fundamental element is that expectation models exist for what the bedrock looks like as the rock excavation work progresses. The predictions that are presented in /12-6/ are the point of departure for the Äspö Hard Rock Laboratory's rock description.

As the Äspö Hard Rock Laboratory is built, a systematic comparison is being made with the models established during the pre-investigations, see Figure 12-4. The comparison, supplemented with additional investigations, will gradually provide a more complete picture of the rock at Äspö.

Data collection is primarily taking place via:

- monitoring of groundwater pressure and salinity in about 140 points in surrounding boreholes,
- daily documentation in the tunnel in conjunction with each blast,
- investigation holes every 20 m along the tunnel,
- special investigations to answer specific questions.

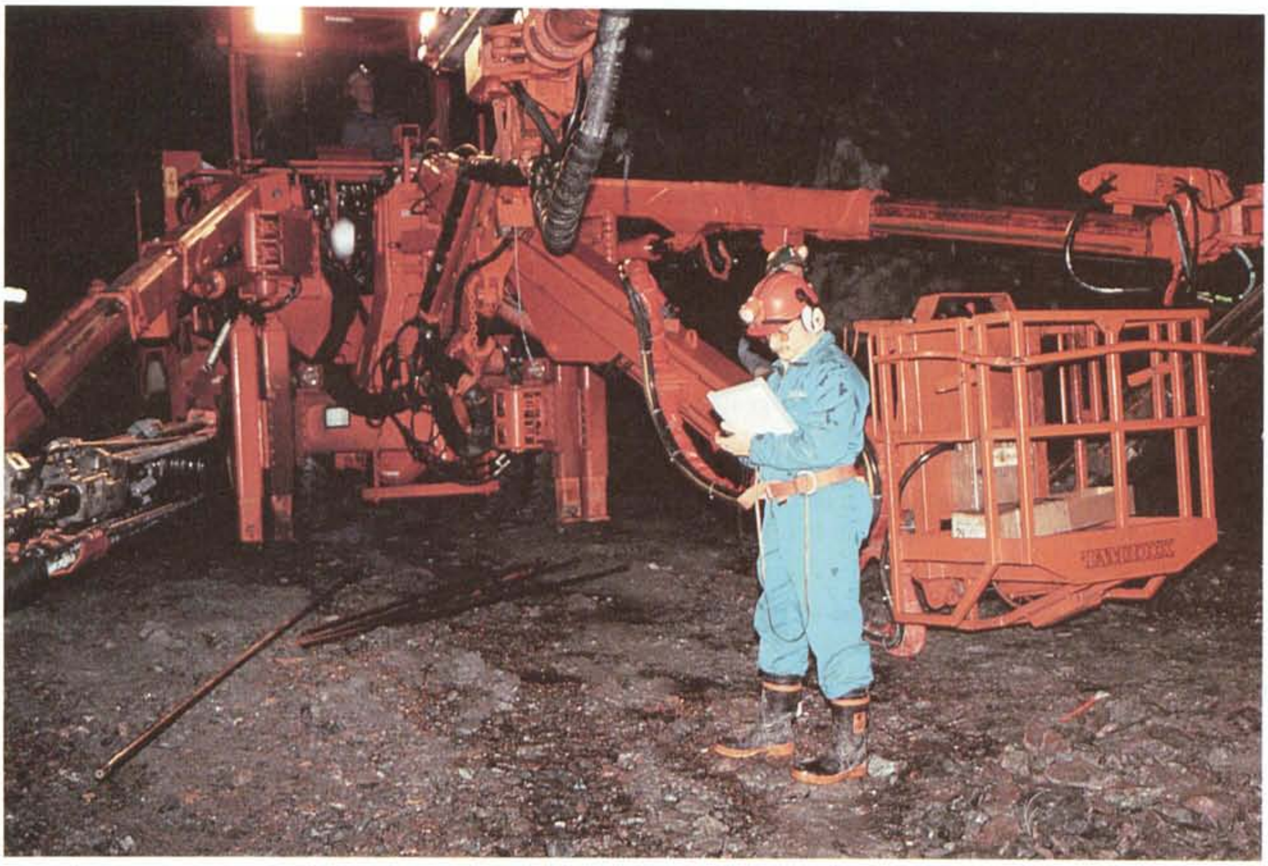


Figure 12-4. Each "round" is documented in accordance with /12-10/. The results are then compared with the predictions of the bedrock reported in /12-6/.

Considerable development work has been done, see further /12-14/.

Daily documentation is done systematically in compliance with a manual /12-7/. The minimum called for by the manual is such documentation in the tunnel as is needed to verify all the prediction models that have been set up during the pre-investigation phase. One important development effort has been to design the data presentation so that it is easy to obtain overviews. Examples of overviews are shown in /12-14/.

Data are for the most part digitized and stored in databases at the Äspö Hard Rock Laboratory.

The excavation work for the Äspö Hard Rock Laboratory is done using a normal drilling/blasting procedure. The routines that have been developed for coordination between construction and investigations are working well. There is also constant feedback during the course of the excavation work between the investigation results and the design of the facility (design-as-you-go). Geodata obtained from the running investigations are used to check the locations of hoist and ventilation shafts, connections at the hoist's stopoff levels, location of the tunnel, and so on. This feedback between investigations and planning/design is an important aspect of the construction of a deep repository.

Two large special investigations have been carried out during the construction phase. The first is a study of the effect of the blasting damage on the remaining rock, while the other is a special study of how the passage of a major fracture zone (called NE-1) can be accomplished in a controlled manner.

When SKB submitted its supplementary reports on the Äspö Hard Rock Laboratory, see section 12.3.1, SKN requested that the damaged zone created by the blasting process be studied without delay. The "disturbed zone", see section 12.4, is of interest for further studies, and the work that was done was restricted to work on blasting damage in remaining rock. Three different blasting procedures were tried in the tunnel.

Important conclusions of the investigations are that

- it is possible to measure the blasting damage,
- the precision of the drilling and local geological conditions can be of just as great importance as the charging of the contour holes,
- in practical blasting, a damage zone of 0.3–0.5 m is created in walls and roof and 1–1.5 m in the tunnel floor.

The results of the experiment have so far been reported in five Progress Reports /12-15 – 19/, two articles /12-20, 12-21/ and two lectures /12-22, 12-23/.

The second special study that has been carried out in conjunction with the ongoing construction of the Äspö Hard Rock Laboratory has concerned the development of methodology for passage of fracture zones.

The pre-investigations showed that the tunnel passes a large fracture zone at the south end of Äspö at a depth of about 180 m. Extensive preparations were made prior to the passage of the zone so that the passage could take place under controlled conditions with fulfilment of a number of subsidiary goals. The subsidiary goals include pinpointing of the location of the fracture zone, characterization of the zone and controlled pre-grouting – mainly with regard to the spread of the grouting compound. Passage of the zone NE-1 took place in the spring of 1992. The properties and location of the zone agree by and large with the predictions set up during the pre-investigations. The large water flows and pressures encountered in the zone required extensive sealing and rock support measures in order to permit safe passage of the zone. The work is under evaluation, but one of the conclusions from this zone passage is that the technique for passing large water-bearing zones at great depth with high water pressures needs to be further developed.

Considerable method development has taken place in conjunction with the preparations for passage of the NE-1 zone, see further /12-14/.

Certain other special investigations in the Äspö Hard Rock Laboratory deal with dating of fracture-filling minerals. Several separate structural-geological analyses have been carried out to examine the connection between structural geology and water-bearing fractures.

Work has begun on the introduction of a formal quality assurance plan.

12.6 RESULTS - TEST OF MODELS FOR GROUNDWATER FLOW AND RADIONUCLIDE MIGRATION

Several fundamental tests of models were performed during the pre-investigations. Important development has been concerned with, among other things, an understanding of the fresh/saline water interface, the scale dependence of hydraulic conductivity, numerical models, etc. See also /12-14/.

The pre-investigation phase concluded with a large combined pumping/tracer test employing short-lived radioactive isotopes. The test was predicted in the conceptual model that was set up before the start of construction /12-5/. Evaluation of the test shows that the results conform to the existing conceptual model of Äspö /12-5/, but that there is room for improvements. The test also permitted an estimate to be made of the rock's flow porosity and dispersion properties.

It should be mentioned that the predictions that were set up during the pre-investigations /12-6/ included only to a small extent predictions of solute transport. Models for solute transport will mainly be tested during the operating phase of the project.

Supplementary data collection will take place during the construction phase to provide further knowledge concerning the disturbed zone, see section 12.7.4

Figure 12-5. After every fourth "round", two 20 m long investigation holes are drilled in front of the tunnel. The hydraulic injection test is performed to measure water permeability (transmissivity).

and Figure 12-5. A part of this work has been the completed blasting damage test.

A redox test is currently being conducted at a depth of about 70 m (section 510 in the tunnel) to study whether, and if so to what degree, oxygenated surface water can alter the capabilities of the rock to maintain a reducing chemical environment. Analyses of water chemistry data, isotope analyses and mineral-chemical analyses /12-25/ show, among other things, that oxygen breakthrough took place very soon after the tunnel had passed a small fracture zone.

Experimental equipment is being developed in preparation for qualified experiments with radionuclides. An example of such equipment is the CHEMLAB probe that is being developed in cooperation with CEA, France.

Development of models for groundwater flow and radionuclide migration is an important goal in the project. The international cooperation confers considerable extra value on the project here. This cooperation is being pursued above all in a Task Force, which is a working group composed of specialists from the participating organizations.

12.7 TECHNICAL PROGRAMME FOR THE ACTIVITIES 1993–1998

12.7.1 General

The ongoing construction phase of the Äspö Hard Rock Laboratory is being utilized to verify pre-investigations and establish detailed characterization methodology – stage goals 1 and 2.

In parallel with these activities, planning is being pursued for the operating phase of the Äspö Hard Rock Laboratory, aimed at testing of models for groundwater flow and demonstration of construction and handling methods, as well as tests of important parts of the repository system – stage goals 3, 4 and 5.

The primary aim of data collection during the first stage of the construction phase to a depth of about 300 m has been to verify the pre-investigations. The work of verifying the pre-investigations has required a relatively rigid model for data collection-reporting-evaluation and a "fixed" configuration of the tunnel route.

A large number of predictions regarding the bedrock were presented in /12-6/. These forecasts are being followed up and evaluated in parallel with the collection of data in connection with the tunnelling work. The evaluation has two aims. The first is to evaluate the validity of the predictions, the second to evaluate the validity of the methods employed.

In view of the results and experience obtained thus far during the construction phase, it is deemed possible to largely complete the work with the data obtained to a depth of about 300 m.

The current forms for the work are, however, less well-suited to fulfilling the second main goal of the project: to refine and demonstrate methods for adapting a deep repository to the local properties of the rock in connection with design, planning and construction.

For this reason the second stage of the construction phase to a depth of about 460 m will be given a slightly different direction. This does not, however, entail any lowering of the level of ambition with regard to either documentation, follow-up or evaluation.

In conjunction with tunnelling to a depth of about 460 m, supplementary preparations will also be made for the experiments that will be carried out during the operating phase.

In the account submitted by SKB in R&D-Programme 89, a pause in the tunnelling of about 6 months was planned between stage 1 and stage 2.

The ongoing work is, however, being conducted more flexibly than originally planned. Instead of one big report/evaluation, several smaller ones are being done instead.

SKB's desire to refine and demonstrate methods for design-as-you-go means that the work of investigation must be done more hand-in-hand with the work of design/planning. Design-as-you-go requires a more flexible reporting of results than the reporting of the predictions from the pre-investigation phase. Some of the data collection and methodology evaluation that had been planned to be done during the pause between stage 1 and stage 2 is being scheduled earlier. The pause between step 1 and step 2 can thereby be omitted. The technical programme for the activities 1993-1998 is written in relation to the stage goals that apply for the project. Goals, main activities etc. are discussed under the respective subheading.

12.7.2 Verifying pre-investigations

The present-day structure of data collection – comparison with prediction – evaluation – scrutiny in the Scientific Advisory Committee will be retained for data down to a depth of about 300 m. This data quantity is deemed to be sufficient for testing much of the conceptual model for Äspö /12-5/ and the corresponding predictions /12-6/. It is also deemed to be adequate for making the evaluation of the methods for the pre-investigations which is the ultimate purpose of Verification of Pre-investigations. SKB shares the opinions presented during the course of the work by SKN and SKI that method evaluation is an essential task for the project. Method evaluation should answer questions regarding, among other things, the resolution, interpretability, appropriateness and usefulness of methods.

During the tunnelling work from about 300 m down to about 460 m, base documentation of the bedrock is being carried out as before. After completed tunnelling, a "final" conceptual model of the bedrock on Äspö will be presented. This model will be compared with the models set up during the pre-investigations on Äspö /12-6/ and the conceptual model set up on the basis of data obtained in connection with tunnelling down to a depth of about 330 m, see further the following section and /12-14/.

12.7.3 Finalizing detailed characterization methodology

The purposes of detailed characterization for a deep repository for spent nuclear fuel are to /12-2/:

- confirm that a suitable repository volume is available,
- provide sufficient data for the safety assessment that is needed to obtain permission for construction of a final repository,
- yield information for site adaptation of the access to the repository,
- provide preliminary data to guide the construction of the repository,
- provide data to permit the repository system to be optimized with respect to engineered barriers and geometric configuration,
- characterize the area between the repository and the discharge area to the necessary extent so that a very good body of supporting data can be presented along with the siting application.

The work done thus far on Äspö exhibits certain analogies to the construction of a deep repository (site selection, site adaptation of the facility, preliminary data to guide the construction etc.).

The geometric configuration of the deep repository will influence the construction and safety of the repository. The rock is a highly heterogeneous material. Certain parts of the rock are severely crushed (fracture zones). The philosophy for repository construction is to modify the geometric configuration of the repository as the work progresses so as to avoid the poor-quality volumes of rock.

The construction of the Äspö Hard Rock Laboratory from a depth of 300 m to a depth of 460 m will primarily be utilized to test methodology for **progressive characterization and progressive design.** The design-as-you-go philosophy has a long tradition within underground construction. This tradition involves a) investigation of the rock b) design c) excavation and documentation d) evaluation with corrective measures regarding investigation, design and documentation.

The activities are being carried out within the framework of a special programme: Rock Volume Descriptions. Besides Rock Volume Descriptions, activities are being carried out within the areas Passage of Zones and Constructability Analysis.

Rock volume description

Detailed characterization during excavation from 300 to 460 m is being done with three aims. The first two are siting of the facility and siting of experimental areas. As pure methodology development, methods will also be tested for locating suitable near fields to disposal canisters.

Siting of the facility

Southern Äspö is delimited by several thick fracture zones. Figure A54 in /12-6/ shows that there is a possibility that the major fracture zone NE-1 will intersect the facility. In view of, among other things, the construction-related problems involved, this zone should be avoided.

Supplementary investigations will be commenced in the autumn of 1992 to pinpoint this zone more precisely. On the basis of the investigation results, the tunnel between 300 and 460 m will be routed through the better (for construction purposes) rock. The investigations are primarily being focused on geological and rock-mechanical conditions. The work is being supported with visual aids (CAD). Results are being reported in Technical Notes. Completion of the investigation and construction work will be followed by evaluation and framing of recommendations.

Siting of experimental volumes

The experiments that are to be carried out during the operating phase will be preliminarily sited during 1993. Supplementary investigations will be conducted during the construction period to finally site the experiments. The investigations of the rock volume for the experiments are being conducted with the philosophy that has been used to verify the pre-investigations with respect to e.g. scale and key issues.

Siting of suitable near fields

A suitable rock environment for deposition of canisters is that the near field is mechanically stable, that the chemical environment is reducing, and that the water flux is low.

Pre-investigations for a repository aim at determining a suitable repository volume. The subsequent detailed characterization is supposed to confirm that a suitable repository volume is available. Unsuitable canister positions will gradually be rejected during the construction of the repository. A methodology for this will be tested at the Äspö Hard Rock Laboratory.

An initial phase includes estimating the number of canister positions in a given rock volume on Äspö that meet requirements on mechanical stability, chemically reducing environment and low water flux.

Qualified structural-geological analyses have been conducted on Äspö. These have been supplemented with dating of fracture-filling minerals, rock stress measurements etc. The data will now be used to test a methodology for locating non-movement-prone rock volumes.

The number of possible canister positions is largely determined by the requirement on low water flux. Water flux (and thereby the number of canister positions) is dependent on the conductivity of the low-conductive rock. The number of canister positions is also dependent on the choice of block size and the "respect distance" to major fracture zones.

Passage of fracture zones

In conjunction with the ongoing tunnelling work, tests have already been conducted of a methodology for pinpointing fracture zones, for characterizing the zone in connection with zone passage and for conducting controlled grouting. The results of this work will be reported during 1993. The report material will include recommendations for supplementary development of construction technology, in particular sealing technology.

Constructability analysis

Experience from design/planning and execution of the facility will be compiled and reported.

A methodology for "constructability" analysis will be tested. Investigation data will be used in a systematic fashion to describe how the construction work is to be planned and executed technically. The analysis will also lead to systematic assessments of the amount of labour (rock support work) and time required.

12.7.4 Test of models for groundwater flow and radionuclide migration

The siting application for a deep repository must include an account of long-term safety. This requires knowledge of groundwater flow and radionuclide migration. The Äspö Hard Rock Laboratory provides an opportunity for practical application and development of models and theories.

Based on evaluation of previously executed work within the Stripa project, the Äspö Hard Rock Laboratory, the SKB 91 safety assessment etc., plans are being made for an experimental programme as described below. This programme will be intimately linked to the development and application of numerical models.

Like the rest of SKB's activities, the planning will be continuously modified in response to experience feedback. The viewpoints of the international participants in the project will also be taken into consideration.

Tests on a detailed scale

Background

Most solutes are transported more slowly than the average flow velocity of the groundwater. This is due to a number of different processes that give rise to a retardation of the solutes in relation to the flowing groundwater. Important processes are dispersion and retention. Retention, or retardation, is caused by the following mechanisms:

- Radionuclides sorb on mineral surfaces passed by the groundwater.
- Radionuclides diffuse out from water-bearing fractures into the stagnant water in the interconnected system of microfissures in the rock and are sorbed on the mineral surfaces there.

Non-sorbing substances are retarded due to the fact that they remain in the stagnant water in the micropores and are thereby withheld from the transport in the flowing water in water-bearing fractures. Radionuclide retention according to this pattern is considerable and is often referred to as "matrix diffusion".

The purpose of this experiment is to obtain data that quantitatively shed light on the effect of the different processes in different fractures and flow regimes.

Fundamental to the description is a good notion of how the groundwater flow is distributed in a fracture plane. The flow distribution determines the variations in flow velocity and thereby the dispersion. It also influences the surface area that comes into contact with the running water and thereby the surface area that is available for sorption. This surface area is often called the "flow-wetted surface area" and is an important parameter in the equations used to calculate radionuclide transport.

Obtaining data on the flow-wetted surface area is a difficult task. A direct measurement of the flow-wetted surface area may require injection of sorbing tracers or some type of gel in a fracture, which is then carefully excavated. An alternative is to measure the increase in travel time caused by the sorption mechanisms by comparing the travel time for sorbing and non-sorbing radionuclides. Several experiments are being conducted to obtain statistics on the variation in the transport parameters.

This test is being carried out on what is called the "detailed scale", i.e. about 5 m. The mechanisms that are coupled to the flow-wetted surface area are also dealt with under the heading "Redox reactions".

Goals

The goals of the detailed-scale tests are to determine the parameters and the processes that govern the transport of sorbing nuclides in individual fractures and to obtain knowledge on the variability of the transport parameters for fractures of different character.

Possible experimental design

A possible experimental design is described in /12-14/. This will be discussed more extensively in the Task Force being formed, see further the section on numerical models.

Tests on a block scale

Background

According to the present concept, a future deep geological repository will be configured in such a manner as to avoid major water-bearing zones. Based on the knowledge of the occurrence of water-bearing zones that has been gained thus far, individual water-bearing fractures or minor zones can be expected to intersect the tunnels in a deep repository at regular intervals. Containers of spent nuclear fuel are intended to be placed in the "good" rock at a suitable distance from minor water-bearing zones. This experiment is aimed at characterizing such minor water-bearing zones and their connection to the fracture system in the surrounding "good" rock. The experiment will thus yield valuable data on possible radionuclide transport from the deposition hole to permeable zones in the surrounding rock.

The experiment will be conducted on a geometric scale called the block scale, about 50 m. This scale can be considered representative of the near field surrounding the spent fuel containers.

Goals

The goals of the block-scale experiments are to characterize the groundwater flow and transport in a minor fracture zone or a water-bearing fracture of large extent and its hydraulic connection with surrounding fracture networks and to prepare the experimental area for the experiment "Radionuclide retention".

Possible experimental design

A possible experimental design is described in /12-14/. This will be discussed more extensively in the Task Force being formed, see further the section on numerical models.

Tests on a site scale, regional zones

Draft proposals for possible tests on a site scale and in regional zones are described in greater detail in /12-14/. These experiments are not planned to be commenced before the turn of the century.

Radionuclide retention

Background

Laboratory investigations for validating models and checking the data used to describe radionuclide dissolution in groundwater, the influence of radiolysis, fuel corrosion, sorption on mineral surfaces, diffusion in the rock matrix and diffusion in backfill material have been conducted over a ten-year period. It is, however, very difficult to simulate accurately in a laboratory such conditions as the following:

- Naturally reducing conditions.
- Natural concentration of colloidal particles,
- Natural content of microbes,
- Natural concentration of dissolved gases,
- Undisturbed rock, i.e. rock with micropore systems and even major fractures that are not pressure-relieved by sampling.

All of these conditions are important for the rock as a barrier, i.e. they have a great influence on the dissolution or retention of radionuclides if radioactive waste is exposed to groundwater.

Goals

The goals of the investigations are to:

- test dissolution and migration of radionuclides in situ,
- validate models and check constants used to describe the dissolution of radionuclides in groundwater, influence of radiolysis, fuel corrosion, sorption on mineral surfaces, diffusion in the rock matrix, diffusion in backfill material, transport out of a defective canister and transport in an individual rock fracture,
- specially test the influence of naturally reducing conditions on solubility and sorption of radionuclides,
- test the ability of the groundwater to take up and transport radionuclides with natural colloids, humic substances and fulvic acids,
- investigate the influence of bacteria on chemical conditions and radionuclide migration, and
- investigate the chemical influence of grouting and backfill materials such as bentonite and cement.

Experimental design

The development of a special instrument – the CHEMLAB probe – has commenced.

The experiments that are to be performed in the probe and out in the fracture zone are being prepared by laboratory tests. The intention is to utilize the zone that has been characterized within the framework of the experiment Tests on a Block Scale, see above. A draft proposal for an experimental programme is presented in /12-14/.

Redox reactions

Background

The SKB investigations show that the groundwater is reducing wherever it is sampled in the bedrock. (The shallow samplings derive from a depth of 40–50 m, the deepest from more than 1000 m. In all, 82 levels in 25 different cored holes have been sampled since 1982. All of these have been quality-classified.) Its buffering capacity is low, however, especially in water with a high pH and associated low solubility for bivalent iron. On the other hand there is a high buffering capacity in the different reducing minerals that occur.

Even if the rock's redox buffering capacity is more than enough on the whole, it is conceivable that the redox conditions could change in a channel that leads water into the repository while it is open. After sealing of the repository, this channel constitutes a potential transport pathway for radionuclides up to the biosphere. In such a situation, it is possible that materials that have been oxidized by radiolysis could migrate up to the biosphere without reduction and sorption of the radionuclides. The consequences of this are the same as of the scenario in KBS-3 where oxidizing conditions have been assumed in the atmosphere, i.e. the individual dose is 100 times higher than under otherwise identical but reducing conditions.

The purpose of the redox experiment is to investigate parameters that are of importance for the propagation of the redox front in the rock. Two such parameters are channelling and "flow-wetted surface area". Besides oxygen reduction, it is important to know how redox-sensitive nuclides are affected by different redox reactions, such as uranium reduction, colloid formation and co-precipitation.

Investigations of groundwater from boreholes show that bacteria are present in all groundwater that is sampled. This may possibly be due to contamination during drilling and other events preceding sampling.

Another purpose of the work within the framework of Redox Reactions is to answer the following questions:

- Are bacteria present in undisturbed rock?
- If so, which species dominate?
- How large a fraction of the bacteria are on fracture surfaces?
- What do they live on?
- Which are active?
- Do they produce complexing agents?

Goals

The goal of the redox experiment is to clarify how quickly oxygen reduction takes place in a water-bearing fracture zone and the effects of the penetration of a fracture system that has previously been reducing by an oxidizing front. The effects on water and mineral composition will be studied. The mechanisms behind the processes will be examined so that the results can be applied generally to nuclide transport calculations. The dependence of sorption mechanisms on the flow-wetted surface area will in particular be explored.

Another goal is to determine the presence of bacteria and their influence on groundwater chemistry.

Ongoing investigations

The redox experiment was begun in 1991. The experiment is being conducted in a side tunnel at the 510 m section in the access tunnel to the Äspö Hard Rock Laboratory. Changes in the character of the water and the fracture minerals are being studied via three short cored holes that penetrate a water-bearing zone. The experimental design is described in greater detail in /12-14/.

Planned investigations

The block-scale redox test described in /12-14/ is qualitatively complete. However, it will be difficult to determine the redox buffering capacity in the flow paths since the minerals, the ratio between mineral surface area and volume and the flow velocities cannot be determined explicitly. A similar test is therefore planned on a smaller scale, where all parameters can be checked. Besides studying oxygen reduction alone, it is also possible to study on a small scale how other redox-sensitive nuclides – chiefly uranium, but also actinide analogues – are affected by the redox front.

The availability of iron and sulphide-containing minerals is of importance for whether fracture surfaces can react with redox-sensitive actinides and uranium. Besides pure redox reactions, sorption and co-precipitation mechanisms can also contribute to fixing these nuclides. The sorption reactions are probably irreversible. Redox reactions and sorption will – if possible – be studied in the fracture or fracture zone previously characterized for the block-scale redox test. The results will also be used for the purpose of defining the "flow-wetted surface area".

Possible experimental design

Uranium(VI) is injected continuously via boreholes in a zone/fracture until breakthrough is observed in another borehole. A draft proposal for an experimental programme is presented in /12-14/.

Disturbed zone around drifts

Background

The making of holes in the rock – whether they be in the form of drifts, shafts, deposition holes or boreholes – entails a disturbance of the rock surrounding the hole in relation to the state that existed before the hole was made. The impact on the surrounding rock depends on such factors as the hole-making method, the size of the hole, stress conditions, the structure of the rock type and the presence of fractures. The term "disturbed zone" is often used in this context. This refers to the zone around the hole where the properties of the rock have been altered in some respect due to the existence of the hole or as a consequence of the work (e.g. blasting) that has been carried out to create the hole.

The properties and extent of the disturbed zone must be taken into account in the design of the repository, in the interpretation of tunnel data and in the assessment of long-term safety. To determine the extent to which the disturbed zone affects the long-term safety of a deep repository, it is necessary to understand the processes that affect the properties and extent of the disturbed zone.

The disturbed zone also affects the water uptake in the bentonite surrounding the canisters.

The properties and behaviour of the disturbed zone have been studied in situ at a number of underground research laboratories during the past few decades. The research work that has the greatest relevance for the Swedish programme is the work that has been done in the Strip Mine since 1977, the Underground Research Laboratory (URL) in Canada, the Grimsel Test site (GTS) in Switzerland and at several laboratories in the USA /12-11/. In 1991 a test was conducted in the Äspö Hard Rock Laboratory's access tunnel where the extent and character of blasting damages were studied as a function of different blasting plans /12-22/.

The experiments that have been carried out to date have identified a number of mechanisms that are evidently of importance for the properties of the disturbed zone. The mechanisms that have been judged to be potentially important are:

- the initial stress load that is obtained at blasting and passage of the tdrift front,
- new fractures created by blasting,
- stress redistribution and rock movements caused by the cavity created by excavation of the drift,
- two-phase flow caused by ventilation (drying), degassing of gases dissolved in the groundwater and/or intrusion of blasting gases,
- chemical reactions and mineralogical changes in the tunnel's near field (can be caused by mechanical impact, blasting gases, oxygen intrusion, mixing of groundwaters with different chemistry, or bacterial activity),
- impact on the rock caused by buffer material (e.g. swelling pressure, intrusion of bentonite in fractures),
- thermal impact on the rock due to the heat from the waste canister,
- creep effects in the rock caused by stress relief and its impact on the long-term mechanical stability and the hydraulic properties of deposition tunnels.

In the experiments performed within the framework of the so-called Site Characterization and Validation project at Stripa, degassing of the groundwater was identified as a potentially important process in the disturbed zone. The gases dissolved in the groundwater are released and form bubbles at low pressures. In this way, a zone may be created around the drift that is not completely water-saturated. The unsaturated zone can then grow due to drying-out of the rock through ventilation. Unsaturated conditions entail that two-phase flow of gas and water can occur in the vicinity of a drained borehole or a drift, which leads to a reduction of hydraulic conductivity. This process is expected to reduce the inflow into drifts and deposition sites during the construction of a deep repository, but when the water pressure returns to normal when the repository is backfilled, two-phase flow will not be of any significance for the long-term performance of the repository. However, the process affects the observations that are made during the construction of the deep repository.

Goals

The goals of the investigations of the disturbed zone are to quantify the parameters that control processes in the disturbed zone, to describe the relative importance of the processes for the performance of the repository system, and to develop and validate quantitative models for essential processes in the disturbed zone.

A goal that lies close in time is to build up a fundamental understanding and quantitative description of degassing of groundwater and its effect on the hydraulic properties of the rock, as well as any hysteresis effects in conjunction with the restoration of water-saturated conditions.

Possible experimental design

The disturbed zone constitutes a large problem complex where a number of processes interact in a complex fashion. To be able to describe the different processes and experimentally verify quantitative relationships, it is important to separate different processes to as great an extent as possible and study each one independently. The research on the disturbed zone has therefore been divided into a number of sub-projects.

First the influence of degassing and two-phase flow will be studied via measurements in several holes drilled especially for the purpose. Degassing and two-phase flow are processes that are dependent on the water pressure and are in principle independent of the existence of a drift. These processes can be studied to advantage in boreholes, where the influence of changes in rock stresses can be minimized. The results from the borehole experiments can then be used to estimate the influence of two-phase flow around a drift.

In a later phase a more comprehensive experiment will be conducted where the hydraulic and mechanical properties of the rock mass are studied in conjunction with the drilling of a simulated deposition drift. This project is being designed in detail against the background of the experience gained from the first project.

A draft proposal for an experimental programme is presented in /12-14/. This draft will be discussed more extensively in the Task Force being formed, see further the section on numerical models.

Numerical models

Background

Numerical modelling of, for example, groundwater flow has been an integral part of the Äspö Hard Rock Laboratory from the very beginning. Originally, the modelling was begun with a simple generic modelling of different alternative designs of the laboratory, taking into account the saline groundwaters that exist. Gradually the models were refined into a comprehensive prediction model of the access tunnel and its effects on groundwater levels and water flux. The access tunnel is currently under construction and the model is being tallied against data obtained during the construction phase.

Groundwater modelling will continue to be an important part of the project. Transport and flow modelling is currently under way for the purpose of refining the existing models and including solute transport in a more comprehensive fashion. The development of models for radionuclide transport is also continuing with the aim of improving the ability of the models to take account of the fact that the transport takes place in a complex network of water-bearing channels. This work is directly linked to the planned tests.

The planned experiments will be integrated as much as possible with the development of conceptual and numerical models.

Goals

The modelling ties in with the Äspö Hard Rock Laboratory's stage goal to refine and test on a large scale methods and models for describing groundwater flow and radionuclide migration in rock.

The goals for modelling are:

- to understand and conceptualize groundwater flow and transport of dissolved and sorbing nuclides in fractured rock,
- to predict these processes with numerical models in the different experiments,
- to verify and validate used models,
- to transfer knowledge between organizations participating in the project,
- to compare and evaluate models in order to assess their suitability as a basis for licensing of a deep geological repository.

Experimental design

Experiments are preceded by predictions or at least scoping calculations.

A principal task is to successfully model the planned experiments.

For the purpose of coordination between experimentalists and model developers, a Task Force (working group) is being formed with participants from the participating international organizations.

The group will follow a number of modelling projects linked to the experiments. The choice of project is determined by the theoretical significance and the degree of difficulty of the tasks, as well as the importance of the project for the organizations participating in the project.

The combined pumping test and radioactive tracer test (LPT-2) has been suggested to be an introductory study for the participating modelling groups.

12.7.5 Demonstrating construction and handling methods

The Äspö Hard Rock Laboratory provides an opportunity to refine and test technology for guaranteeing high quality in the design, construction and operation of a deep repository.

It is foreseen that the technology to be used for the Demonstration Project (see chapter 6) will be tested at the Äspö Hard Rock Laboratory.

It has been shown in connection with the ongoing construction of the Äspö Hard Rock Laboratory that the traditional construction technology is not sufficiently robust as regards passage of water-bearing zones with high water pressures. Technology development in this area is of value.

It is important for safety that a deep repository be configured and adapted with respect to the actual geological conditions on a site. It is foreseen that a methodology for a progressive selection of the location of repository tunnels and canister positions needs to be tested and demonstrated. The detailed planning of this activity can take place when the repository concept has been finalized. Some minor construction activity is foreseen in conjunction with this work.

12.7.6 Testing important parts of the repository system

The Äspö Hard Rock Laboratory provides an opportunity to test, investigate and demonstrate on full scale different components that are of importance for the long-term safety of a deep repository. Such testing will be done of, for example, the technology to be used for demonstration deposition in the deep repository, see chapter 9.

Detailed planning will be done when the repository concept has been finalized.

12.7.7 Construction works

The Äspö Hard Rock Laboratory is being designed to meet the needs of research, development and demonstration.

The underground part is designed as a tunnel from Simpevarp to the southern part of Äspö. Two spiral turns of tunnel are being built to a depth of about 460 m on Äspö. Niches and side tunnels will be built as needed. The underground part is being connected to the ground surface via an elevator shaft and two ventilation shafts.

Äspö Research Village is being built on southern Äspö, see Figure 12-6.

12.7.8 Quality assurance

The work of establishing a formal quality assurance plan for the Äspö Hard Rock Laboratory to guarantee the desired quality will be completed during 1993. Achieving traceability of data and models is of particular importance.

12.8 INTERNATIONAL PARTICIPATION

SKB has led the STRIPA project together with several international organizations. International cooperation promotes quality, resource utilization and acceptance for both SKB and the participating organizations. Agreements on participation in the Äspö Hard Rock Laboratory have been signed during 1992 with TVO of Finland, ANDRA of France and NIREX of the UK. Negotiations have largely been concluded with DOE of the USA.

Together with previous agreements with PNC and CRIEPI of Japan and AECL of Canada, this means that seven foreign organizations will be associated with the project through different agreements. Through these agreements, SKB gains access to the foremost experts in the concerned countries within the research fields vital for the Äspö Hard Rock Laboratory. This lays the foundation for a high international level of quality in the work.

In practice, this cooperation takes place in the form of personnel from the organizations being present on the site (PNC and CRIEPI), testing of instruments (ANDRA), and refining and further testing of models for groundwater flow and solute transport (PNC, CRIEPI, US/DOE, ANDRA, TVO, NIREX).

The scientific exchange also takes place through participation in the Scientific Advisory Committee and within the framework of the Task Force.

Several of the participating organizations have planned additional investigations and experiments. The investigations are included in the agreements concluded with the respective organizations and will be conducted in addition to the programmes described here. SKB regards this additional work as adding significant extra value to the project.

Figure 12-6. Bird's-eye view of Äspö Researche Village.

12.9 EXECUTION, ORGANIZATION, INFORMATION

Like SKB's other R&D projects, the R&D work at the Äspö Hard Rock Laboratory is being executed through contracts to universities, colleges, research institutes, consultants, industrial companies and other Swedish and foreign researchers. This makes it possible to achieve a high standard of quality and competence, since the most qualified experts can be chosen for different investigations and experiments. Different alternative ways or models can be tried for some issues.

The direction and contents of the programme are established by a Programme committee within SKB's Research and Development unit. With the RD&D-Programme as a basis, annual planning reports are published which describe in some detail the work during the coming year. Two reference groups have been appointed to give advice and viewpoints on programmes and results: the Scientific Advisory Committee and the Construction Advisory Committee. The international cooperation is coordinated by a Technical Coordinating Board.

The project is headed by a project manager within SKB's Research and Development unit. A project group is in charge of executing the work. Persons in charge of geology, geohydrology, geochemistry, instruments and construction issue recommendations for overall programmes, draw up object plans, analyze and evaluate results etc. The site office does the work on the site. Different sub-projects are defined as the need arises to achieve good coordination.

The construction works are being carried out by contractors. The contract for the rock excavation works was signed with Siab on 14 June 1990 and covers a finished tunnel to full depth and shafts to the surface for hoist and ventilation.

A personnel hoist for communications between Äspö Research Village and the underground facility will be delivered by ABB Drives AB.

The ventilation system will be delivered by Svenska Fläkt AB.

Most of the remaining procurement relates to the construction works for Äspö Research Village.

Information on the project is disseminated in a number of ways. The public and nearby residents are informed on the site. A special Visitors' Niche has been set up in the access tunnel where SKB's activities and the Äspö Hard Rock Laboratory are presented, see Figure 12-7. In addition, general information is available at the exhibition hall in Simpevarp village and on a "nature trail path" out on Äspö. Both are open to the public.

Figure 12-7. *Visitors' Niche.*

13 The STRIPA project

13.1 INTRODUCTION

The international Stripa project was initiated in 1980 as an autonomous project with the OECD/Nuclear Energy Agency (NEA). The project had two purposes: (1) to develop technology for investigating deeply situated crystalline bedrock that is potentially suitable for the geological final disposal of high-level waste, and (2) to study special engineering methods for enhancing the safety of a geological repository over long periods of time. During the thirteen years and three phases of the project, the activities have been concentrated to the Stripa Mine in central Sweden, where in-situ experiments and tests have been conducted. The last investigations were concluded and the mine was abandoned on 30 June 1991.

Finland, France, Canada, Japan, Spain, Sweden, Switzerland, the United Kingdom and the United States have all participated in different phases of the Stripa project. The organizational structure in Phase 3 (see Figure 13-1) shows the two most important functions throughout the course of the project: (1) scientific review and decisions on programmes and (2) execution of the research work. The overall management of the project has been in the hands of the Joint Technical Committee (JTC), consisting of representatives from each of the participating countries. Responsibility for the day-to-day project work rested with SKB. The Technical Subgroup (TSG), which consisted of representatives from the participating countries, was responsible for running evaluation and follow-up of the investigations and for recommendations to the JTC regarding the continued research work. During Phases 1 and 2, two Technical Subgroups worked independently with: (1) groundwater flow and nuclide transport, and (2) engineered barriers and rock mechanics. Owing to the more integrated character of the research activities that were pursued during Phase 3, an integrated Technical Subgroup was established with assistance from two working groups. The two working groups were responsible for the fields: modelling of groundwater flow in the rock's fracture system, and methods for rock sealing.

The following summary description of the Stripa project is very brief. The final reporting and analysis of the project is in its most intensive phase as this is being written. This work will be concluded in December

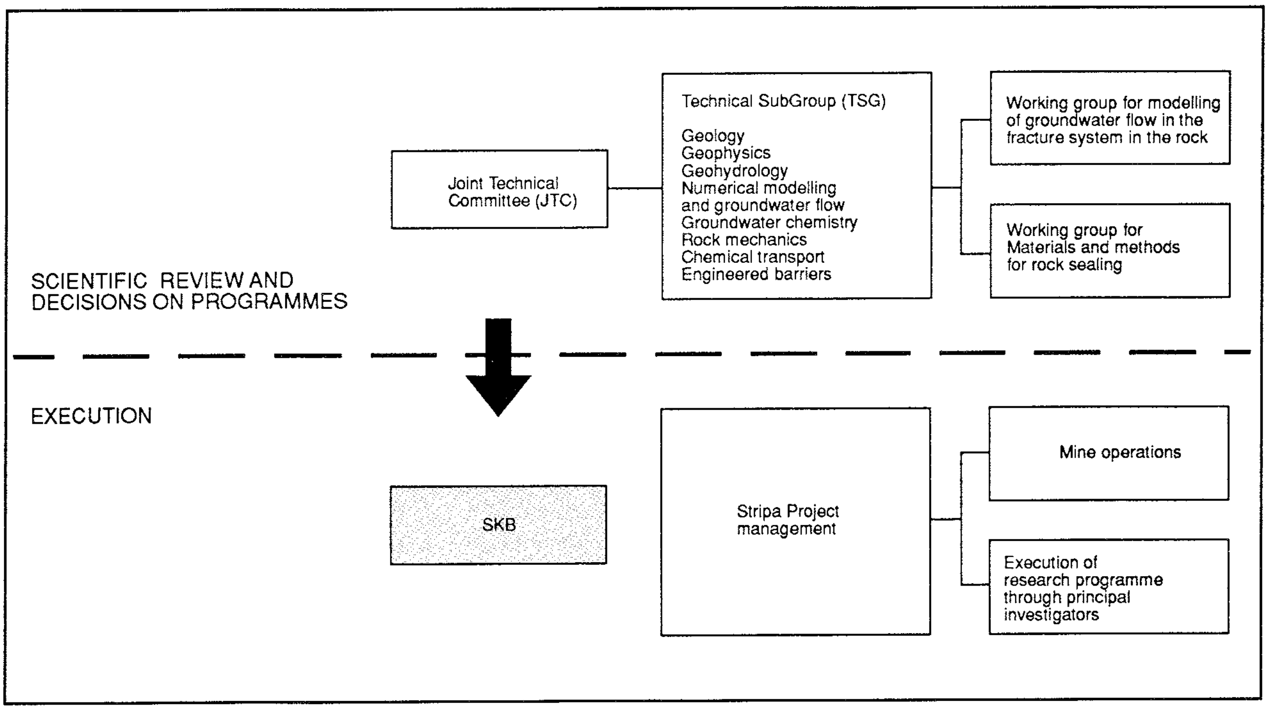


Figure 13-1. Organization of the Stripa Project.

1992. A complete summarizing report in Swedish will be submitted in January 1993.

13.2 STRIPA MINE

The Stripa Mine is situated in a tradition-rich mining district around 250 km west of Stockholm. Mining operations began there in the mid-1400s, and a total of 16.5 million tonnes of iron ore have been mined during the different mining eras between 1448 and 1976. The Stripa Mine's iron ore consists of quartz-banded haematite surrounded by a large volume of granite of medium-grained grey to light red type. The age of the granite has been estimated at 1.7 billion years. So far all in-situ experiments have been carried out in the granite.

The granite volume in which the Strip Mine is situated is traversed by a number of regional fracture zones spaced at intervals of around one kilometre. It is known that numerous fracture systems are present in the rock mass between these regional zones, but the majority of the fractures are clogged mainly with chlorite, but also with calcite. Drifts and shafts in the mine influence the groundwater's regional flow paths, which is some cases are up to 10 km in length. Over the course of time some of the groundwater flow towards nearby Lake Råsvalen has been affected to such an extent that the flow has ceased. The rock volume that has been used for in-situ tests is situated in a local system of groundwater flow where the mine serves as a recharge area. The downward-directed flow of "younger" low-salinity groundwater is mixed with the upward-directed flow of "older" more saline groundwater. As a result, the granite in the vicinity of the mine will contain a blend of the two water types. However, the mine has been considered to be very dry, since only 500 l of water/min have been pumped up.

Since 1977, when the earliest experiments were commenced within the Swedish-American Cooperative (SAC) programme, approximately 700 m of new drift have been driven in the granite from the two existing drifts at the 360-metre and 410-metre levels. The main area where the in-situ tests were conducted is situated approximately 800 m from the main shaft.

13.3 PHASE 1 - 1980–1985

Phase 1 of the Stripa project lasted from 1980 to 1985 /13-1/. Participating countries were Canada, Finland, France, Japan, Sweden, Switzerland and the United States. The goals were:

- to refine methods for measurement of the hydraulic conductivity of a fractured rock mass in horizontal and inclined boreholes,
- to refine methods and routines for determining the chemical composition of the groundwater in the rock, including the origin of the groundwater and changes over time,
- to increase understanding of the properties of solutes in the groundwater when the water flows through a fractured rock mass,
- to evaluate the properties of the clay material in a fractured water-saturated rock mass under simulated disposal conditions.

The hydrogeological investigations were conducted in vertical and horizontal boreholes at the 360-metre level in the Stripa Mine. Since the groundwater from the orebody drains into the mine, the investigations were concentrated on measuring the increase in the water pressure in closed boreholes and the decrease of the pressure in open boreholes. A straddle-packer system was used to avoid large gradients along the borehole and to distinguish the intact rock from fracture zones.

Combinations of single-hole measurements and cross-hole measurements (interference tests) resulted in a determination of the hydraulic properties of the intact versus the fractured rock. Furthermore, the measurements showed that a large-scale anisotropy exists in the hydraulic conductivity with straight flow paths that can extend over large distances and with high-conductive water pathways between boreholes. The geochemical studies showed that dating of groundwater is not meaningful when it is based on measurement of single radioactive isotopes. The results indicated rather that different isotopes have different residence times due to different origins and that several different processes affect the isotopes in the circulating groundwater.

The flow experiments studied the natural groundwater's transport of dissolved tracers in open fractures. The results indicated that tracer transport is controlled mainly by the fact that the water flows in channels in the open fractures. Furthermore, in certain irregularly situated points in the fracture, water from different channels is mixed. The channels are also believed to contain areas with stagnant water through which the solutes are transported by diffusion.

The tests with clay as a buffer material included an experiment where six electric heaters, each enclosed in highly compacted bentonite, were placed in six boreholes in the bottom of a drift. The boreholes had a diameter of 0.76 m and were 3 m deep. Two of the boreholes were situated near the end of the drift and where the entire drift was later backfilled with a mixture of bentonite and sand. The test lasted three years, during which time continuous recordings were made of the bentonite's swelling pressure and temperature. Aside from a few occasions when the bentonite reached a temperature of 125°C for a relatively brief duration, the maximum temperatures in the highly compacted bentonite lay around 75°C. The results show that the groundwater from the surrounding rock penetrates the bentonite very uniformly, even when the boreholes are intersected by only a limited number of water-bearing fractures. Similarly, the pressure in the bentonite increased uniformly and the maximum pressure rise could be predicted well. Nor did that part of the bentonite that had been heated to 125$C exhibit any mineralogical changes, aside from a slight tendency to precipitate silica compounds.

13.4 PHASE 2 - 1983–1988

Phase 2 of the Stripa project lasted from 1983 to 1988 /13-2/. Participating countries were Canada, Finland, France, Japan, Spain, Sweden, Switzerland, the United Kingdom and the United States. The goals were:

- to refine technology and methods for cross-hole measurements including borehole radar, seismics and hydrology for identification and characterization of fracture zones,
- to refine technology and methods for evaluation of the migration behaviour of non-sorbing radionuclides that are transported with the groundwater through a rock mass of granite,
- to study further hydrological and geochemical properties of the Stripa granite to improve present-day methods for interpretation and presentation of data,
- to evaluate the sealing capacity of highly compacted bentonite against groundwater flow in the sealing of boreholes and of blasting damages in shafts and tunnels.

The basic idea of the programme for cross-hole measurements was to develop technology that only requires a few boreholes. The technology that was developed and tested included methods for high-efficiency borehole radar and borehole seismics as well as hydraulic testing methods. The borehole radar proved to be able to be used with a distance of up to 200–300 m between the boreholes and with a resolution of 1–3 m. Interpretation programs were developed for determining the exact position of fracture zones by studying the reflection from measurements in single holes or with data from cross-hole measurements. Data from borehole seismics can be interpreted with tomography, whereby a picture of the seismic velocity variation in the rock between two boreholes can be obtained.

The cross-hole programme showed that the distribution of major structures in a large rock volume is best described by a combination of borehole radar in single holes and cross-hole and/or seismic data that has been analyzed with the aid of inverse tomography. When the conceptual geological model has been established in this manner, the hydraulic properties of the fracture zones are determined with the aid of hydraulic cross-hole measurements where water pressure and flow rates are varied sinusoidally. The final result is a hydrogeological model of the rock mass.

A 3-dimensional migration experiment was conducted in a 100 m long drift. Large plastic sheets were fastened to the roof and the upper part of the drift walls. The water inflow to the plastic sheets was measured during a period of more than two years while the hydraulic head in the rock was recorded in three 70 m long boreholes that had been drilled vertically upward in the drift roof. In addition, nine different tracers were injected in nine different fracture zones in the vertical boreholes during a period of more than one and a half years. Based on the breakthrough curves for six of the nine tracers and the distribution of the water flow to the tunnel, the conclusion was drawn that the groundwater flow had a very uneven distribution in the rock mass and that a non-negligible flow of water takes place in channels that have only little contact with the main channels. The porosity of the fractured granite within an area 10 m from the tunnel was estimated to be twice as large as the porosity in the undisturbed rock mass.

The results of the hydrogeochemical investigations showed that water inclusions in the Stripa granite are probably hundreds of millions of years older than the mobile groundwater and that these may have contributed to the formation of the saline groundwaters that have been observed. It was further possible on the basis of radioisotope data to define a series of vertical zones in which young surface water and water from the pre-Cambrian bedrock has penetrated and which has then remained unaffected for many thousands of years.

Bentonite has proved to be extremely effective in sealing against circulating groundwater in boreholes, shafts and tunnels. Boreholes with a diameter of 56 mm and 76 mm were plugged with cylinders of bentonite placed in 2 m long perforated casings. These plugs exhibited no leakage up to a head of the magnitude required to induce hydraulic splitting of the rock in the test area.

The test with sealing of shafts and tunnels was designed so that it was possible to compare the sealing efficacy of bentonite in relation to conditions in a reference test in the same area. The results showed that water leakage was reduced by a factor of about 30 when the tunnel was sealed with bentonite. The hydraulic conductivity in a bentonite plug was considerably lower than the hydraulic conductivity in the surrounding rock. Owing to the swelling pressure of the bentonite, a very tight and integrated clay/rock interface is created, preventing water flow. Bentonite can be used to advantage in rock containing Ca-rich groundwater, since the initial microstructure is preserved on ion exchange.

13.5 PHASE 3 - 1986–1992

Phase 3 of the Stripa project lasted from 1986 to 1992. Participating countries were Canada, Finland, Japan, Sweden, Switzerland, Great Britain and the United States. The goals were:

- to predict groundwater flow and nuclide transport within a previously undisturbed rock volume in the Stripa Mine and to compare these predictions with data collected using existing and improved instruments and methods for site investigation,
- to select and verify the suitability of material for sealing of fractures and fracture zones for a long period of time in crystalline rock, and to demonstrate the effectiveness of representative sealing methods and the practical application of developed injection technology.

In support of the first goal, a Site Characterization and Validation (SCV) programme was carried out starting in 1986 /13-3/. The site selected for the execution of the SCV programme was a block in a previously undisturbed granite volume situated approximately 100 m north of the old mine drifts between the 360 m and 410 m levels. The granite volume was about 125 m on a side and 50 m in height. The SCV programme was divided into five stages so that field data could be compared with predictions on repeated occasions. In Stage 1, called Preliminary Characterization of the Rock Volume, a few holes were drilled in the adjacent granite volume. A preliminary database was established with the aid of data from geological, geophysical, geochemical and hydraulic investigations in the boreholes. These investigations included use of cross-hole seismics, borehole radar and single-hole hydraulic tests. In Stage 2, called Preliminary Predictions, a geohydrological model of the SCV block was developed on the basis of the preliminary database established during Stage 1. Preliminary predictions of the geometry of the largest fracture systems and their physical properties were made, along with predictions of the groundwater flow within the SCV block. In Stage 3, called Detailed Characterization and Preliminary Evaluation, additional holes were drilled towards the interior of the SCV block to obtain additional data. These data were used to evaluate the precision of the preliminary prediction made during Stage 2 regarding the geohydrological conditions, the groundwater flow and the qualitative aspects of the flow distribution within the SCV block.

In Stage 4, called Detailed Predictions, forecasts were made of the geometry of the fracture systems on the basis of data obtained in Stage 3. Moreover, a combination of deterministic and statistical mathematical models was used to predict groundwater flow and tracer migration within the SCV block. In Stage 5, called Detailed Evaluation, a drift was excavated into the interior of the SCV block for validation of the models/predictions. The water flow in the drift was measured and tracer-doped water was injected into the rock mass at different distances from the validation drift and collected when it entered the drift. With the information obtained by means of visual inspection of the drift, the water flow and the experiments with tracer-doped water, an evaluation could be made of the validity of the predictions made on the basis of investigation data and mathematical models.

In the first stage of the sealing investigations in Phase 3, a comparison was made between the properties of different sealing materials. This provided a basis for the choice of Na-bentonite-based clay and Portland-based cement for further laboratory and field tests. At the same time, the pilot-scale tests carried out in the Stripa Mine showed that fractures smaller than 100 microns can be sealed effectively with these grouting agents with the use of dynamic injection technology.

In the second stage, full-scale sealing tests were performed in the Stripa Mine. These included (1) sealing of rock around two simulated holes for deposition of waste containers (0.76 m in diameter and 3.5 m deep). Grouting was done using a multipacker system: (2) sealing of a disturbed zone in rock around a blasted-out drift, and (3) sealing of a natural water-bearing fracture zone that traverses a drift. In general, it can be said that the sealing work undoubtedly reduced the hydraulic conductivity of the rock mass and diverted the groundwater flow. However, tests also showed that some practical difficulties existed regarding the technique for application of the sealant.

The long-term stability of sealing materials has to do with the ability of the materials to retain their sealing properties over long periods of time, perhaps hundreds of thousands of years, despite the fact that a change in the direction and/or magnitude of the groundwater flow takes place. The results of laboratory and model tests showed that (1) the chemical stability of Na-bentonite-based clay is largely dependent on the temperature, especially at temperatures above 100°C; (2) the viscosity-reducing additives in the Portland-based cement are strongly associated with the hydrated cement phases and are only leached out in small quantities from the cured cement. Furthermore, the tests show that variations in pore size and pore distribution in the cured cement phase, owing to changes in the cement/water ratio, do not constitute a significant factor in the leaching processes; and (3) factors that influence the long-term stability of seals with cement under expected repository conditions include e.g. an initial hydraulic conductivity of the cement on the order of 10^{-10} – 10^{-12} m/s, a cement with a minimal ettringite content, and application in a place distinguished by a low hydraulic gradient together with a solution-saturated groundwater. The results from the sealing tests in the Stripa Mine are summarized in /13-4/.

13.6 SUMMARY

The most important gains in the Stripa project can thus be summarized as follows:

- Important progress has been made in the development and use of technology for characterization of rock masses, particularly when it comes to borehole-based methods within the fields of radar, seismics and hydrology for detection and description of fracture zones, and geochemical methods for determination of groundwater circulation and groundwater origin,
- There is good agreement between field measurements and prediction of the groundwater's flow and its transport of solutes, where the prediction has been based on the use of modern sophisticated numerical models applied to limited quantities of input data,
- Progress has been made in obtaining a fundamental understanding of the flow of nuclides in fractured rock, especially as far as channelled flow is concerned,
- Methods have been demonstrated for effective sealing against inflow of groundwater to, and flow in, boreholes, drifts, shafts and fractured rock. Through a combination of laboratory tests and modelling, a greater understanding has been achieved of the properties that influence the life of bentonite- and cement-based sealing materials under expected repository conditions.

Regarded as a whole, these scientific gains provide a basis for future systematic and technical solution of the many questions that exist around geological repositories for radioactive waste. This technology has considerable applicability to the waste management programmes in Sweden and many other countries, from the use of tools and technology for site investigation to clarifying concepts for characterization of different geological units and prediction of groundwater movements within them.

14 R&D REGARDING ALTERNATIVE METHODS AND OTHER RADIOACTIVE WASTE

14.1 ALTERNATIVE METHODS

At the same time as the research and development work on direct disposal of spent nuclear fuel is being pursued by the construction of demonstration facilities for encapsulation and deep disposal, good reasons exist to allocate some resources to the follow-up of alternative methods. This is also in line with the requirements of the Act on Nuclear Activities on a comprehensive programme. Internationally, R&D work is being conducted on both alternative treatment methods for the spent nuclear fuel and on alternative final disposal methods for long-lived waste. Through SKB's well-developed international cooperation network (see chapter 17), we are assured of relatively broad insight into the major programmes being conducted in other countries. For certain specific lines of development that can only be expected to lead to applications in the longer term, however, a limited Swedish effort is warranted. In this way Sweden can build up domestic competence in the field and become a sufficiently interesting cooperation partner to be allowed insight into the broader programmes being conducted in other, larger countries.

14.1.1 Alternative treatment methods

As reported in chapter 5, the option of partitioning and transmutation of long-lived nuclides in the high-level waste has attracted renewed interest in recent years. Relatively large programmes are under way or are planned in France, Japan and the USA. These programmes are being followed in Sweden, and certain with them have been established by Swedish researchers.

Certain questions are of particular importance in judging the development potential for these treatment methods, for example the degree of separation of long-lived nuclides, the technically feasible efficiency of the transmutation, material problems and the reliability and safety of the processes. SKB plans to support certain limited research in this direction at Swedish institutions of higher education. The work may be in the nature of doctoral theses and the like.

14.1.2 Alternative final disposal methods

The main thrust of the work on final disposal in all countries is disposal in rock at a depth of 300 to 1000 metres. Alternatives also attracting interest are seabed disposal and disposal in very deep boreholes. Both of these methods make retrievability and future corrective actions more difficult. Nevertheless, analyses conducted to date indicate a considerable potential for safe final disposal where the geological barrier alone provides sufficient isolation of the radioactive materials. Neither method is yet ripe for application or demonstration. Seabed disposal is mainly intended to be done at very great depths beneath the oceans. This requires international collaboration, and at present the tendency is more to prohibit such disposal. Disposal in the crystalline bedrock beneath the Baltic Sea has also been discussed in Sweden, see section 6.4.3. Deep-sea disposal is hardly a feasible option for Sweden.

Disposal in very deep boreholes requires further technical development, but above all greater knowledge (data) on the properties of the rock at depths of several km. Considerable research is being done in this field without any direct connection with nuclear waste disposal. The results can be of interest not only for deep-hole disposal but also for a general understanding of the geology of the rock on a regional scale. A limited follow-up by Sweden of the ongoing research in the field is therefore warranted. The work can be done in the form of, for example, doctoral theses at a college or university. The contacts that have been established in this field with Russian researchers are also worth maintaining, since a considerable portion of the scientific work to date has been done in Russia.

14.2 OTHER LONG-LIVED WASTE

14.2.1 General

Beyond spent fuel there are three other types of long-lived radioactive waste in Sweden. They are:

- waste from the research activities in Studsvik,
- waste from the encapsulation plant for spent fuel, and
- core components and reactor internals.

The waste from Studsvik (which also includes a small quantity of waste from medical applications and other research) contains such large quantities of long-lived radionuclides, especially plutonium, that it may not be emplaced in SFR.

The waste from the encapsulation plant is normally not long-lived. In the event of fuel damages, however, waste contaminated with long-lived fuel residues can be obtained.

Induced activity in the material, primarily Co-60 but also certain long-lived nickel and niobium isotopes, has been created by neutron bombardment of core components and reactor internals.

These waste types are planned to be disposed of in connection with the deep repository for spent nuclear fuel, at roughly the same depth, but with a certain respect distance (about 1 km), in order to prevent them from affecting each other. The waste in this repository will be surrounded by and, in some cases, bentonite. The experience and principles from SFR will thereby be utilized.

14.2.2 Waste characterization

The long-lived waste in Studsvik derives from about 30 years of research activities. The waste has previously been handled and stored without today's requirements on documentation. For the past few years the waste has been conditioned in containers suitable for final storage. In connection therewith it is essential that a detailed characterization of the waste be performed.

As a basis for determining which components in the waste and which properties are essential for final disposal, a review is being made of completed safety assessments for final repositories for long-lived low- and intermediate-level waste. The results of this review will serve as a basis for the requirements on documentation. It can be preliminarily concluded that these requirements are for the most part identical to the requirements on documentation for SFR.

Core components and reactor internals are relatively well-known in terms of form and activity content, and no new studies are planned.

14.2.3 Design of final repository

General recommendations on how the final repository for other long-lived waste is to be designed are given in PLAN 92 /14-1/. For the waste from the research activities, the same barriers as in the silo in SFR are employed, i.e. the waste packages with concrete-embedded waste are placed in larger concrete troughs and grouted with a porous concrete. Sand/bentonite is packed between the concrete trough and the wall.

Concrete embedding is also used for the core components to provide radiation protection and a suitable chemical environment. In this case bentonite filling has not been considered necessary.

According to current plans, the final repository for long-lived waste will be built in connection with the second stage of the deep repository. For this reason, no detailed design work on the final repository is planned during the period 1993-98. Waste of a similar type exists in larger quantities in those countries that reprocess their spent nuclear fuel and final disposal is being studied there, e.g. in the United Kingdom, France, Germany and Switzerland. This work will be followed.

14.2.4 Background for safety assessment

The final repository sections for other long-lived waste will be characterized partly by the strongly basic concrete environment and partly by the heterogeneous composition of the waste. In many respects, the same issues that have been of interest in the safety assessment for SFR will also be of interest for these final repositories.

Information on a number of different topics will be needed for the safety assessment. These include the composition of the waste, the design of the repository, the properties of the concrete and how they change with time, solubility and sorption properties, formation of organic complexes, diffusion in concrete and bentonite etc.

Extensive research within these areas is planned and is being carried out in those countries that have large quantities of long-lived low- and intermediate-level waste. For the most part, SKB's work during the period 1993-98 will be concentrated on following these activities. Certain work will be done on the decomposition of cellulose and the risk of complexation, see section 14.3.

14.3 FUTURE RESEARCH REGARDING FINAL REPOSITORY FOR RADIOACTIVE OPERATIONAL WASTE, SFR

Operating permits for SFR-1, stage 1, have been issued by both SKI and SSI. These permits entail that the facility may be utilized to its full extent. The restriction that applied to the silo repository was rescinded when the deepened safety assessment was approved in May 1992 /14-2/.

The operating permits are associated with certain stipulations. For example, a renewed safety assessment shall be carried out every ten years, as long as the facility is in operation. Furthermore, a renewed safety account shall be submitted in support of an application for a permit for sealing.

For the purpose of gathering as much site-specific knowledge as possible for coming safety assessments and sealing of the facility, a monitoring programme is being carried out that includes recording of groundwater pressure, rock deformations, water inflow and chemical groundwater composition as well as recurrent inspections of the rock. For the silo repository, observations are being made of the movements of the rock silo and of water uptake in the bentonite fill (swelling pressure). For the upper part of the slot fill, wetting is also being studied for the purpose of seeing how much of the bentonite needs to be replaced in connection with sealing of the silo top.

Greater knowledge of the long-term properties of certain waste materials and additives is desirable. In the case of organic material, present-day knowledge of its long-term properties is limited. This is particularly true of decomposition products of cellulose and their possible complexation with radioactive materials (such as Pu). A separate study supported by tests of cellulose decomposition is planned. For the time being, in compliance with stipulations in the operating permits, the quantity of organic material will be monitored and limited to the various repository chambers.

Within other areas of importance for the coming safety assessments – such as development of numerical models, knowledge of groundwater movements in the rock, sealing technology and documentation of the facility for the future – the research/development that is being conducted for the high-level waste should also be able to be applied to SFR.

15 DECOMMISSIONING OF NUCLEAR FACILITIES

15.1 BACKGROUND

When a nuclear power plant is taken out of service, parts of it are contaminated with radioactivity. This means that decommissioning must be carried out in a controlled manner with due consideration given to the need for radiation protection measures beyond conventional industrial safety. Furthermore, certain parts of the decommissioning waste must be managed and disposed of as radioactive waste. This also applies to other nuclear facilities, such as CLAB and the encapsulation plant, when they are taken out of service.

A number of small research reactors and a few small nuclear power plants have already been decommissioned in various countries. At present, several other medium-sized nuclear power plants are being decommissioned, for example in Japan, the USA, Germany, Belgium and the UK. Other reactors that have been taken out of service have been put in order so that they can stand 30-50 years before the actual dismantlement is carried out. No full-sized plants have as yet been shut down and decommissioned.

Experience of decommissioning in Sweden is limited to decommissioning of the R1 research reactor in Stockholm and several smaller facilities at Studsvik. Considerable experience of a similar kind has, however, been obtained from the steam generator replacement at Ringhals-2 and from other repair and rebuilding jobs at the nuclear power plants.

The completed decommissionings and a number of studies show that the methods for decommissioning nuclear power plants are available today. A report from the OECD/NEA /15-1/ states that the next step is to "industrialize" the decommissioning methods that have been tested and demonstrated on a pilot scale, in other words to scale them up to routine industrial application. No fundamental problems are foreseen. The biggest obstacle in the decommissioning work currently seems to be the fact that final repositories for the nuclear waste have not yet been built in most countries, or that the existing repositories are not prepared to accommodate the decommissioning waste.

Most of the equipment that is needed for decommissioning already exists today and is used routinely in maintenance and rebuilding work at the Swedish nuclear power plants. It is only for dismantlement of the reactor vessel and its reactor internals, and for demolition of the concrete shield nearest the reactor vessel, that methods are needed that have not yet been used in Sweden. Experience from the use of such methods is being obtained from ongoing decommissioning projects in other countries. The Swedish nuclear power industry has good insight into these project through a cooperative programme organized under the auspices of the OECD/NEA, where SKB is in charge of the secretariat and programme coordination.

15.2 GOALS AND GENERAL PLAN

The goal of the decommissioning work after a nuclear power plant has been taken out of service is that the site shall be restored and reclaimed after some time so that it can be used without any radiological restrictions. This shall be accomplished in such a manner that neither the personnel engaged in the decommissioning and dismantling work nor the general public are exposed to unnecessary irradiation. Decommissioning will proceed in several stages. The IAEA has defined three stages in the decommissioning work /15-2/, which are defined by the physical status of the plant.

In **stage 1**, fuel and fluids have been removed from the reactor and the control systems disconnected. Access to the plant is restricted and the plant is kept under surveillance and inspected periodically.

In **stage 2**, most of the components containing radioactivity have been concentrated to a limited volume, which is sealed. Less surveillance is required that in stage 1, but continued periodic inspection is desirable.

In **stage 3**, all radioactive materials (above the free release limit) have been removed and the area has been released for unrestricted use. Stage 3 is sometimes called "green field". As an alternative to free release immediately after the conclusion of decommissioning, some waste can be left in a shallow ground repository on the site, requiring surveillance for approximately 50 years.

It is not necessary that the decommissioning proceed sequentially through the three stages. Stage 2 is applied primarily if dismantling is intended to be deferred beyond the time the plant is retired from service. A postponement of 30 to 100 years is the usual figure given. If the dismantling work is intended to commence within a few years of shutdown, it is natural to proceed directly via stage 1 to stage 3.

The schedule that will be used for the Swedish nuclear power plants has not yet been decided on. A number of different factors will influence this decision. The most important are what other kind of activity is planned on the site, and the availability of personnel familiar with the plant. Radiation protection aspects and, not least, general political aspects may also influence the decision.

The procedure for decommissioning the Swedish nuclear power plants has been described in a report from SKB, "Technology and Costs for Decommissioning the Swedish Nuclear Power Plants" /15-3/. This report shows that a decommissioning can be commenced approximately one year after the last reactor has been shut down at a nuclear power station. Putting the plant in mothballs for 30-50 years before the actual dismantling work is commenced is also shown to be a feasible alternative. Immediate dismantling is recommended mainly due to the availability of personnel familiar with the plant. This means that dismantling is recommended to begin immediately after all reactors on the site have been shut down. A deferral of dismantling results in a lower radiation level, permitting certain simplifications of the dismantling work.

When the time comes to carry out the dismantling work, a common nationwide planning approach will prove to be most efficient. This offers advantages in the form of a more rational utilization of special equipment and specially trained personnel, as well as good opportunities for experience feedback.

Thus, the premise for the planning of future decommissioning and of the need for R&D is that decommissioning will not be commenced until 2010 at the earliest. Depending on what future use is planned for the nuclear power station site – for example if it is to be used for alternative power production – there may also be reasons for starting the actual dismantling work at a later date.

The overriding goals for SKB's efforts within the field of decommissioning are:

- to ensure that knowledge and technology for decommissioning is developed in good time before the detailed planning of the decommissioning work is to commence,
- to ensure that the waste from decommissioning can be managed, transported and disposed of, and
- to provide data, by means of cost estimates, as a basis for determining the need for allocating funds to a reserve for the decommissioning work.

The most important means of achieving these goals are:

- follow-up of international developments,
- follow-up of experience from maintenance and rebuilding work at the nuclear power plants, and
- certain special studies and tests.

15.3 ONGOING WORK

15.3.1 Sweden

Most of the technology required for the future decommissioning of the nuclear power plants is, as mentioned above, already available and is used routinely in connection with the maintenance, repair and rebuilding work at the nuclear power plants. Special equipment need only be developed for dismantling of the reactor tank and demolition of heavy concrete structures. A great deal of work is being done within these areas abroad, and it is very important that this work be followed up. Independent efforts in Sweden are not warranted at this time.

A follow-up of repair and rebuilding work takes place at each nuclear power plant. In the case of certain major projects, it is a good idea to conduct separate studies to determine what experience can be of use in future decommissioning. One example is the steam generator replacements at Ringhals.

In earlier studies the option of removing, transporting and disposing of the reactor vessel intact has been taken up as an interesting area of development. A preparatory study has been made of what this would entail at the different nuclear power stations, in terms of transport and at SFR. This study shows that a potential exists to reduce dose load and costs. A more detailed study is in progress.

The most recent calculation of the costs of decommissioning was carried out in 1986 /15-3/. A new technology and cost study is under way and is scheduled to be completed in 1993. The Swedish costs are low by international standards /15-4/, which can be explained by the efficient system that has been developed for transportation and final disposal of nuclear waste in Sweden, which enables large components to be managed without the need for extensive segmenting.

15.3.2 Other countries

The most important work within the decommissioning field is being done in conjunction with actual decommissioning projects for reactors and other nuclear facilities that have been taken out of service. So far some 20-some reactors have been decommissioned to stage 3, i.e. they have been dismantled and the radioactive components removed. In addition, a large number of plants have been taken out of service and decommissioned to stage 1 or 2. Most decommissioning projects have concerned experimental reactors or small power reactors. Only in recent years have some medium-large reactors (< 250 MWe) also been taken out of service.

In parallel with actual decommissioning projects, some work is also being done on the development of dismantling methods. Usually it is connected with a given decommissioning project, however. The work is being done to a large extent on a national basis, but there is also some international cooperation, primarily within the OECD/NEA and the EC.

The OECD/NEA's Cooperative Programme on decommissioning

A special programme has been organized within the OECD/NEA for an exchange of information and ex-

Table 15-1. OECD/NEA cooperative programme within the decommissioning field. List of projects.

Facility	Type	Planned final stage
Eurochemic, Belgium	Reprocessing plant	Stage 3
BR-3, Belgium	PWR, 41 MWt	Stage 3
Gentilly-1, Canada	Heavy water reactor, 250 MWe	Stage 2
NPD, Canada	Heavy water reactor, CANDU, 25 MWe	Stage 1
Rapsodie, France	Sodium-cooled fast reactor, 20 MWt	Stage 2
G2, France	Gas-cooled reactor, 45 MWe	Stage 2
AT1, France	Reprocessing plant for fast reactor fuel	Stage 3
Niederaichbach, West Germany	Gas-cooled heavy water moderated reactor, 106 MWe	Stage 3
Lingen, Germany	BWR, 256 MWe	Stage 1
MZFR, Germany	Heavy water reactor, 50 MWe	Stage 3
Garigliano, Italy	BWR, 160 MWe	Stage 1
Japan Power Demonstration Reactor (JPDR), Japan	BWR, 13 MWe	Stage 3
Windscale Advanced Gas Cooled Reactor, Great Britain	AGR, 33 MWe	Stage 3
BNFL Coprecipitation Plant, Great Britain	MOX fuel fabrication	Stage 3
Shippingport, USA	PWR, 72 MWe	Stage 3
West Valley Demonstration Project, USA	Reprocessing plant for LWR fuel	Stage 3
EBWR, USA	BWR, 100 MWt	Stage 3
Tunney's Pasture	Isotope handling	Stage 3
BNFL B204 Primary Separation Plant, Great Britain	Reprocessing facility	Stage 2

perience between ongoing decommissioning projects. Most major decommissioning projects in the world are included in this programme. At present the programme includes a total of 19 projects in 8 countries. A summary of the projects included in the programme is given in Table 15-1. Twelve of the projects aim at complete decommissioning to stage 3.

Within the cooperative programme there is an exchange of experience from day-to-day activities as well as more extensive discussions and information exchange on specific technical questions. Examples of questions that have been discussed are melting of metallic waste, monitoring methods for low-level waste, removal of asbestos and methodology for cost estimates and cost accounting.

Experience from the first five years within the OECD/NEA programme has been compiled in a report /15-1/. Besides descriptions of individual projects and the work being done within them, a thorough analysis of status and development needs for different decommissioning areas is presented. The areas discussed are:

- activity inventory assessment,
- decontamination methods,
- cutting techniques,
- remote operation,

- radioactive waste management, and
- health and safety.

The development need in each area is identified. In most cases it is a question of translating experience from tested methods to application on an industrial scale, and gaining experience from that. **No area where fundamental development efforts are needed has been identified.**

The projects within the programme that are of special interest to Sweden are Shippingport (USA), JPDR (Japan) and Niederaichbach (Germany). The decommissioning of Shippingport has been completed. A major feature of the decommissioning was the intact removal of the reactor vessel and its shipment by barge to the final repository site.

Within the JPDR project, which involves the decommissioning of a boiling water reactor, extensive testing and development of different dismantling methods has been conducted. Of particular interest is the completed segmenting of the reactor vessel. Segmenting of the reactor vessel is of special interest in Niederaichbach as well.

SKB is heading the programme coordination function for the OECD/NEA programme, and thereby has an opportunity to follow the technical aspects of the different projects closely.

The EC's research programme

The EC has had a cooperative research programme within the field of decommissioning since 1979. At present the EC's third five-year programme is under way.

So far the studies have primarily concerned different dismantling techniques as well as questions pertaining to activity content and waste management /15-5/. The following research areas have been covered:

- long-term integrity of buildings and systems,
- decontamination,
- dismantling techniques,
- treatment of specific waste materials: steel, concrete and graphite,
- large waste containers,
- estimation of waste quantities.

Furthermore, work is under way to formulate guidelines for decommissioning.

A comprehensive account of the results achieved is given in /15-6/.

In the most recent five-year programme, the emphasis has shifted towards application and testing of different dismantling techniques under actual conditions. Thus, four decommissioning projects are included in the programme: the reactors Windscale AGR (UK), Gundremmingen A (Germany) and BR-3 (Belgium), and the reprocessing plant AT-1 (France).

IAEA

Work is under way within the IAEA aimed at summarizing the state of knowledge within the different technical areas and formulating recommendations and advice for future applications for licences for decommissioning.

The IAEA also has a coordinated R&D programme within the decommissioning field. SKB has participated in this programme with a study of the handling of the intact reactor vessel.

Other development

In addition to the international cooperative project mentioned above, development work is being pursued within the field of decommissioning in several other countries as well. Of special interest are the French programme being run by CEA and the projects now being started in the eastern European states, as well as in eastern Germany.

15.4 RESEARCH PROGRAMME 1993–1998

The schedule for carrying out the necessary R&D work on decommissioning is closely linked to the schedule for the decommissioning of the Swedish nuclear power plants. As indicated above, the first decommissioning will not be commenced until a few years after 2010, at the earliest.

A few years before the planned start of decommissioning, a project group will be organized to plan the decommissioning work in detail. The necessary background knowledge regarding dismantling methods, classification of waste, transportation systems etc. must be available by this time.

Most of the methods that are needed are already available and are being used in Sweden. They will be modified to suit the needs of this work in connection with the planning of the decommissioning. Some of the equipment will require further development. Since a great deal of development work is currently being done abroad, there is no need to start any large Swedish development efforts during the coming six-year period.

A new study of technology and costs for decommissioning the Swedish nuclear power plants is in progress and will be completed in 1993. It takes into account the most recent experiences from decommissioning and maintenance projects.

Ongoing and previous studies of decommissioning of Swedish nuclear power plants have indicated some areas where early measures are warranted. The most important are:

- study of the feasibility of disposing of an intact reactor vessel (see above),

- technology for segmenting of reactor internals,
- technology for demolition of the biological shield,
- management of contaminated asbestos insulation,
- methods and equipment for activity measurement of the waste for unrestricted release or simpler disposal,
- decontamination for unrestricted release,
- volume reduction of the waste by means of compaction or melting.

During the next few years, most of the work will be concentrated on the feasibility of disposing of an intact reactor vessel, as well as on methods for unrestricted release of the waste, since this can have great economic importance.

For other areas, efforts will be concentrated on follow-up of activities abroad and of experience from the operation of the nuclear power plants. Towards the end of the period, it may be decided to conduct more systematic work in the other areas. In that context, an evaluation should also be made of the possibility of conducting tests in the retired Ågesta reactor. At present, Studsvik plans to test, in international cooperation, decontamination and melting of Ågesta's steam generators.

The follow-up activities are intended to be pursued as before through the programme coordination function within the OECD/NEA programme and through participation in the IAEA work, etc.

Decommissioning produces a large quantity of lightly contaminated material, which could be released for unrestricted use, after decontamination if necessary. Some experience exists from unrestricted release at the nuclear power plants. However, the low unrestricted release limits make measurement and classification very laborious. Before decommissioning is commenced, it is essential that rules and methods for unrestricted release be developed so that this can be done routinely. The capability to measure low activity levels is thereby of great importance.

Prior to decommissioning of the nuclear power plants, the final repository for decommissioning waste, SFR 3, must also be finished. Since the waste from decommissioning is equivalent to some waste from the operating period, experience from SFR 1 can serve as a basis for the design of SFR 3. This is described in /15-7/. The time from preliminary planning and design to a finished facility has been estimated to be about seven years, which means that this work will not be commenced until a few years into the next century.

For the decommissioning work to be carried out in an efficient manner, it is essential that certain administrative questions be resolved, for example what type of licence is needed and what type of reporting to the regulatory authorities is required for this purpose. This work lies within the sphere of responsibility of the regulatory authorities.

16 PRIORITIES AND COSTS

Carrying out a demonstration deposition of spent nuclear fuel in a deep geological repository requires two main activities in the development work – encapsulation and deep disposal. Beyond this, safety assessments and supportive research and development are required.

The work with encapsulation entails selection and testing of a method for fabrication, sealing and quality inspection of canisters plus planning, design, construction, licensing, installation, trial operation and operation of an encapsulation plant. The work involved with deep disposal includes siting, planning, design, construction, licensing, installation, trial operation and operation of a facility and equipment for demonstration deposition in a deep repository.

The supportive R&D work entails further development of methods, models and data. Their purpose is:

- further refining the knowledge base and skills in modelling of processes that are important for the performance of the repository in order to better be able to quantify uncertainties and safety margins,
- following up international developments in relevant fields.

The research is planned so that a continuity is obtained in the work and an updating of the knowledge base and analysis methods is done in good time before major assessments of performance or safety. Much of the supportive RD&D work will be concentrated to the Äspö Hard Rock Laboratory. Another important support for further development of the safety assessment is further studies of spent fuel properties and natural analogues.

Besides work that comprises direct support for the main line – a deep repository for demonstration deposition – some follow-up of alternative methods and systems is planned so that knowledge of these will be retained and further refined. In this way a basis will be created for the future evaluation of such systems in comparison with what is being demonstrated in Sweden. In addition, work is planned on other long-lived waste as well as on SFR and decommissioning of nuclear power plants.

An important part of the RD&D-programme is international cooperation, which is extensive and takes place in several different forms.

Table 16-1 contains a preliminary cost estimate for the facilities that are described in chapters 8 and 9.

The activity "Encapsulation" includes final design and final selection of fabrication technology for a composite canister of copper and steel. Particular attention is being given to welding technology for sealing of the canister and technology for non-destructive testing of a finished canister. Certain studies of lead casting for the reserve alternative lead-filled copper canister are being conducted in parallel. Encapsulation also includes design, planning and safety assessment of an encapsulation plant up to supporting material for an application for siting permission under the Act Concerning the Management of Natural Resources and a licence under the Act on Nuclear Activities. These applications are planned to be submitted at the beginning of 1997. While the applications are being considered, the work of detailed design and planning of the plant and of full-scale testing of welding equipment will continue.

Table 16-1. Preliminary cost estimate for Encapsulation Plant and Deep Repository 1993–1998. (SEK million in August 1992 prices).

Activity	1993	1994	1995	1996	1997	1998
Encapsulation station	20	28	30	38	34	35
Deep repository						
- basic activities and planning	25	20	20	20	20	20
- site activities	35	60	70	70	30	70
Total deep repository	*60*	*80*	*90*	*90*	*50*	*90*

The activity "Deep Repository" includes, in the first place, certain basic activities such as overall planning, quality assurance, design of plant and equipment for the deep repository, procurement of instruments, certain safety assessments and environmental studies, production of information material etc. and, in the second place, site-related activities, mainly geoscientific field studies. It has been assumed in the cost estimate that pre-studies and pre-investigations will be carried out for two sites starting in 1993 and with the aim of submitting the application for a permit for detailed characterization at the end of 1997. It is assumed that detailed characterization will begin in 1998.

Table 16-2 contains a cost estimate for the RD&D-programme that is presented in chapters 10-12, 14 and 15.

For the Äspö Hard Rock Laboratory, the ongoing construction and research work will be completed by 1994. The estimate assumes that the international co-

Table 16-2. Preliminary cost estimate for RD&D-Programme 92. (SEK million in August 1992 prices).

Activity	1993	1994	1995	1996	1997	1998
Äspö	113	84	55	54	56	52
Supportive R&D	51	46	43	40	37	37
Alternative/ other waste	8	8	8	8	11	12
Decommissioning	3	2	2	2	2	2
Grand total	**175**	**140**	**108**	**104**	**106**	**103**

operation continues to develop as it has thus far and that a considerable body of demonstration and experimentation will be carried out from 1994 onward. Of the stipulated costs for Äspö, about SEK 40 million/year are costs for investigations and experiments, while the rest are construction costs and fixed operating and maintenance costs.

Within the area "Supportive R&D", priority is being given to studies of spent nuclear fuel and its properties in the repository environment, studies of natural analogues, continued development of models for performance and safety assessment and certain further studies within the field of chemistry. Ongoing work on canister and buffer materials is expected to be largely completed by the mid-90s. Table 16-3 below shows the approximate percentage breakdown of the costs for supportive R&D during the six-year period 1993–1998.

Table 16-3. Approximate breakdown of costs for supportive R&D 1993–1998.

Area	Share of total cost
Fuel	35%
Other materials	7%
Geoscience	15%
Chemistry	20%
Natural analogues	8%
Safety assessment	10%
Biosphere	5%

Within the area "R&D for alternative methods and other waste", priority is being given to studies of long-lived low- and intermediate-level waste. Work on partitioning and transmutation is estimated to take two-three months per year, while follow-up of the work on deep-hole drilling will have a limited volume.

17 INTERNATIONAL COOPERATION

Development within the field of nuclear waste management takes place to a great extent in international cooperation and interaction. All countries with a major nuclear power programme have made plans for the management of different forms of radioactive waste and have initiated the research and development that is considered necessary. International activities are therefore being conducted today on a very large scale in the form of experiments, model development, site investigations, data compilations etc. within the field of nuclear waste management, of which the Swedish efforts naturally constitute only a small part. The extent to which Sweden is able to derive direct benefit from the work being done in other countries is primarily dependent on the following three factors:

- technical and geological similarities in repository design and site,
- choice of treatment method for spent nuclear fuel,
- schedules for the execution of research programmes, large-scale tests and demonstration projects as well as the construction/operation of final repositories.

The benefit that Sweden can derive from other countries' research can lie on several different planes:

- contributions to method and model development,
- a broadened and strengthened body of data,
- exploration of other alternatives for repository and barrier design, material selection etc.,
- contributions towards bolstering public confidence in the system through e.g. demonstration trials and large-scale tests,
- achievement of an international consensus regarding safety assessment.

An important part of SKB's programme is therefore to follow and profit from the research and development that is being conducted in other countries in a methodical and systematic manner. This task is made easier by the great interest shown internationally in the work being done in Sweden. This chapter contains a summary of some of the foreign programmes along with an overview of the different international cooperative projects in which SKB is directly involved.

17.1 FOREIGN R&D OF INTEREST FOR SKB'S PROGRAMME

USA

The schedules in the USA are governed to a high degree by the Nuclear Waste Policy Act, which was passed in 1982. The Act has undergone far-reaching changes, the most recent in 1987. The Act states that the federal government is responsible for the final disposal of high-level waste and spent nuclear fuel. US DOE, the United States Department of Energy, is responsible for constructing a final repository which, under the Act, must be ready for active use by no later than 31 January 1998. The site for the American repository has in principle already been selected through the most recent amendment to the Act, and resources are now being focussed on Yucca Mountain in Nevada. The repository is planned to be built here at a depth of 400 m in water-unsaturated rock, providing about 200 m of unsaturated tuff as the primary geological barrier above the present-day groundwater table. The methodology for characterizing Yucca Mountain, model development, studies of waste forms and safety assessment are areas where experience exchange is of value for Sweden, even though the host medium is a different one. Direct contacts have been established with a number of specialists heading different projects in the American waste programme. US DOE has participated for 15 years in the Stripa project and has announced that they are interested in cooperating in the Äspö Hard Rock Laboratory.

Canada

AECL (Atomic Energy of Canada Ltd) is the federal organization in charge of Canada's nuclear power programme. AECL is also responsible for research and development having to do with conditioning and disposal of nuclear fuel waste. The provincially owned power utility Ontario Hydro is responsible for interim storage and transportation of spent nuclear fuel. The division of responsibilities between the federal government and the provincial governments when it comes to the final repository has not yet been defined. Canada's programme for final disposal consists of three phases:

- concept assessment,
- site selection,
- demonstration of disposal vault.

In a 10-year programme, research has been carried out to establish a scientific basis for geological disposal and for technical criteria for site selection and repository design. An "Environmental Impact Statement" is in the process of being prepared and will subsequently be reviewed in a well-defined "Environmental Assessment and Review Process" (EARP). The review will be extensive and will include public hearings. Site investigations and site selection are expected to take place during the 1990s. Once a site has been selected, a

20-year demonstration period is planned, concluding with expansion of the demonstration facility to a disposal vault, which will be put into operation after the year 2010.

The bedrock in Canada closely resembles the Scandinavian bedrock, so many of the geological investigations in Canada are of interest to the Swedish programme. Of particular interest is the URL (Underground Research Laboratory) project, where a shaft is being sunk to a depth of about 450 m in the bedrock. AECL has participated in the Stripa project since 1980. SKB has an agreement with AECL regarding information exchange between URL and the Äspö Hard Rock Laboratory, see section 17.3. Canada is also advanced in chemistry and when it comes to studies of spent fuel.

In the area of natural analogues, SKB has been cooperating with AECL in the Cigar Lake project since 1989. Cigar Lake is a very rich uranium deposit at a depth of about 430 m, embedded in an illitic clay. Conditions in the uranium deposit resemble those foreseen in a final repository. See section 11.6.2 and /17-1/.

Finland

Under Finnish law, responsibility for nuclear waste management in Finland rests with the nuclear power producers. In 1978, the two power utilities IVO and TVO formed a joint commission, YJT, to coordinate the necessary research and development.

For spent nuclear fuel, the policy is to attempt to secure agreements whereby the spent nuclear fuel can be shipped abroad for final disposal. Such an agreement exists with Russia for the Loviisa reactors. Other nuclear fuel is to be first stored for an interim period and then finally disposed of in Finland. A facility for interim storage has been built at Olkiluoto. A site for a final repository will be selected around the year 2000, and final disposal is expected to start around the year 2020.

A list of around 101 sites of interest for a final repository, selected in connection with an inventory, was presented at the beginning of 1986. Preliminary investigations were carried out on 5 of these sites during the period 1988-1992. These will be followed by detailed characterization on 2-3 sites up to the year 2000, when the final site will be selected. Further investigations will be conducted on this site up to the time for a licence application around the year 2010.

Due to close similarities between the Swedish and Finnish bedrocks, an exchange of information is particularly worthwhile.

For low- and intermediate-level waste, TVO has constructed a final repository (VLJ) at Olkiluoto. The repository has been constructed at a depth of 70-100 m and was commissioned in the spring of 1992. IVO will construct a similar repository at Loviisa at the end of this decade.

TVO has participated in Stripa since 1980 and has also decided to participate in the Äspö Hard Rock Laboratory.

France

Responsibility for final disposal of nuclear waste in France lies with "Agence Nationale pour la gestation de Déchets Radioactifs" (ANDRA). For high-level waste, the question of site selection has been preceded by a reconnaissance of several hundred sites where granite, salt, shale and clay have been investigated. Site selection is very much a political question and recommendations for criteria were put forth in 1987 in the Gougel Report. Strong opposition to the site selection that had been made arose during 1990, and the French Government therefore ordered a detailed review of the situation. This work led to a new law governing research on radioactive waste, which was passed on 30 December 1991 by the National Assembly and the Senate.

The law states the following:

- disposal of foreign waste in France is prohibited,
- two underground laboratories shall be built to investigate alternative methods for deep disposal of vitrified high-level waste,
- an independent evaluation commission shall be set up which annually submits a status report on the progress of research within transmutation, deep geological disposal and extended supervised storage on the surface,
- methods for the immobilization of high-level waste shall continue to be improved in order to enhance the safety of temporary storage facilities,
- within 15 years the Government must undertake an overall evaluation of the results of the research and, if appropriate, present a bill to construct a repository for high-level waste.

Low- and intermediate-level waste has been disposed of since 1969 at La Manche. This repository is now full. A new repository has been constructed at Aube about 200 kilometres east of Paris. The Aube repository can accommodate 1 million m^3 of short-lived waste, which is equivalent to about 30 years' nuclear power production.

ANDRA has participated in the Äspö Hard Rock Laboratory since 1992.

SKB also cooperates with Commissariat a l'Energie Atomique, CEA, within the fields of radionuclide chemistry and buffer/backfill.

Germany

Germany intends to dispose of its high-level waste in a salt formation at Gorleben. An extensive investigation programme is being conducted at Gorleben, including shaft sinking to repository depth. A final repository is expected to be able to be brought into service some time around the turn of the century.

The geological studies in salt are of little interest to Sweden. However, along with Sweden, Germany is the country that has studied the direct disposal alternative most systematically. The results of these studies were reported in the spring of 1985 in a large study, PAE, Projekt Andere Entsorgungstechniken. The PAE project is being carried further, aimed at a full-scale demonstration of certain features, for example canister fabrication and handling of encapsulated fuel. Most of the spent nuclear fuel from the German programme will be reprocessed. Direct disposal may be used for certain odd fuel types. SKB is following the work on direct disposal in Germany through an exchange of information with the PAE project.

Low- and intermediate-level waste will be disposed of in a former iron ore mine, Konrad. The licensing work for this final repository started in 1990. The commissioning date has not yet been determined, but is estimated to be some time in 1993.

Switzerland

The Atomic Energy Act in Switzerland requires the nuclear power utilities to present a plan for safe final disposal of radioactive waste. The Swiss federal government and the Swiss nuclear power utilities have jointly formed NAGRA (Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle) to manage the radioactive waste.

In 1985, NAGRA published its study Projekt Gewähr, an equivalent of the KBS-3 report. The report recommends that high-level waste be disposed of at great depth in crystalline rock. An experimental station in rock, equivalent to the Swedish Stripa Mine, has been constructed at Grimsel in the Alps.

The Swiss Government approved Projekt Gewähr in June 1988. However, they requested that supplementary data on a concept for the storage of waste in sedimentary rock be submitted to them by 1990, and that an evaluation report on all investigations in crystalline rock be submitted to them by the end of 1990. Site selection in Switzerland is planned to take place around the year 2000, and a final repository is intended to be taken into service around the year 2020.

For a repository for low- and intermediate-level waste, NAGRA conducted investigations on three sites in the early 1980s. Due to local opposition and political impasses, these sites have been ruled out. In June 1987, an application was submitted for permission to investigate a new site, Wellenberg. The Wellenberg area was thoroughly investigated during the period 1988-92. NAGRA expects to be able to submit a comparative report to the regulatory authorities on all 4 investigated sites in 1993. They hope to be ready to submit an application for permission to site a repository for low- and intermediate-level waste in 1994, and at the same time start the licensing procedure and the work on a pilot tunnel for detailed underground characterization.

NAGRA has participated in Stripa since 1980.

SKB has close contacts with NAGRA and the Swiss programme. Direct cooperation and coordination of the work takes place within the fields canister materials and natural analogues.

UK

A programme of geological investigations for final repositories in granite formations was initiated during the 1970s in the UK. In December 1981, further activities within this programme were postponed for at least 50 years on the grounds that it had been demonstrated that disposal was possible in principle and that high-level waste can be temporarily stored for such a period of time without any problems. Accordingly, no further decisions on final storage of HLW are awaited within the next few decades. R&D in the UK is therefore now being devoted solely to technology for vitrification of HLW and to storage and model studies. The UK is also participating actively in the Stripa project, the Poços de Caldas project and NEA's seabed disposal studies.

For long-lived low- and intermediate-level waste, the planning calls for a deep repository to be built beneath the sea. The repository may be situated at a depth of about 700-1 000 metres. Preparatory investigations have been carried out on two sites, Sellafield and Dounreay. After several years of public hearings and circulation for review and comment, the choice has fallen on Sellafield. It is estimated that the final repository will be ready for use round about 2005.

UK NIREX Limited, which is owned jointly by the power utilities, the United Kingdom Atomic Energy Authority and the British state, is in charge of the waste programme for LLW and ILW in the UK. NIREX has expressed an interest in participating in the Äspö Hard Rock Laboratory.

EC

The EC is conducting a comprehensive and well-coordinated programme within the field of nuclear waste management. The work is being pursued in 5-year programmes, and the current programme period extends from 1990 to 1995. The plan for this period includes the following points:

1. System studies and harmonization of member states' waste management policies.
2. Conditioning of radioactive waste.
3. Characterization of waste forms, encapsulation methods and canister materials.
4. Research on development of underground repositories.
5. Safety assessments.

The work thus far has been done in the field at underground facilities at Asse in Germany, Mol in Belgium and at the French experimental facility at Fanay-Augères.

Sweden is participating through SKB in several EC projects, for example the COCO Club (Colloids and Complexes), CHEMVAL and NAWG (Natural Analogue Working Group).

Russia

In 1988 a cooperation agreement on waste management was concluded with the then SCUAE (State Committee on the Utilization of Atomic Energy). A first seminar was held in the spring of 1989 and a second in the spring of 1991. The Russian programme appears to be oriented towards the disposal of reprocessed waste in granite or salt formations. There are no concrete cooperative projects at the present time.

Japan

Responsibility for the management of radioactive waste in Japan is divided between two government bodies: the Science and Technology Agency, STA, which is a part of the Government offices, and the Ministry of International Trade and Industry, MITI. STA conducts research and development in science and technology, while MITI is active within the industrial side of waste management.

The Power Reactor and Nuclear Fuel Development Co., PNC, and the Japan Atomic Energy Research Institute, JAERI, work under STA. PNC has been charged with the task of presenting a repository concept for high-level radioactive waste and performing a safety assessment for it, while JAERI works with geochemical questions and related safety assessment work.

MITI presides over the Central Research Institute of the Electric Power Industry, CRIEPI, which works together with the power utilities with safety assessment, instrument development and cost estimates.

Japan Nuclear Fuel Ltd., JNFL (formerly JNFI), has overall responsibility for management of the low- and intermediate-level waste. JNFL is currently planning a repository at Rohkashomura in the northern part of the country.

In July 1989, SKB signed a cooperation agreement with JNFL concerning an exchange of information on the management of LLW and ILW.

PNC has participated in Stripa since 1980.

Both PNC and CRIEPI cooperate with SKB in the Äspö Hard Rock Laboratory and have each had a researcher stationed in Simpevarp during 1991.

International organizations

International cooperation within the field of nuclear waste management is coordinated by the UN's International Atomic Energy Agency, IAEA, and the OECD's Nuclear Energy Agency, NEA. These organizations are natural forums for information exchange on radioactive waste, see 17.9 and 17.10.

17.2 SKB's COOPERATION AGREEMENTS WITH FOREIGN ORGANIZATIONS

SKB strives to keep track of relevant results from development work in other countries in a systematic fashion. To this end, SKB has signed formal bilateral agreements with the following organizations in other countries:

- USA – DOE (Department of Energy),
- Canada – AECL (Atomic Energy of Canada) and Ontario Hydro,
- Switzerland – NAGRA (Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle),
- France – CEA (Commissariat a l'Energie Atomique), ANDRA, DCC and IPSN,
- EG – EURATOM,
- Finland – TVO and IVO,
- Russia – former SCUAE (current name of organization unclear),
- Japan – JNFL (Japan Nuclear Fuel Ltd.).

Information exchange without formal agreements also exists with:

- Germany,
- Belgium,
- UK,
- other Nordic countries.

The formal agreements are similar in their construction and cover information exchange and cooperation within handling, treatment, storage and final disposal of radioactive waste. Exchange of up-to-date information (reports), as well as results and methods from research and development, are main points in the agreements. Arranging joint seminars and short visits of specialists to other signatories' facilities are other examples of what is included within the framework of the agreements. General reviews of the signatories' waste programmes and activity planning within the framework of the agreements are held at roughly one-year intervals.

As far as extended exchanges of personnel or extensive direct project cooperation are concerned, special agreements are generally concluded within the framework of the general agreement.

These agreements give specialists within the field of nuclear waste management greater opportunities for contacts and collaboration.

17.3 FUEL LEACHING - SPENT FUEL WORKSHOP

Studies of corrosion of high-level fuel are only being conducted by a few laboratories in the world. The experimental work is both costly and time-consuming, since it has to be carried out in a hot cell. It is therefore important that opportunities be provided for an informal exchange of results and experience. The Spent Fuel Workshops that were started in 1981 at SKB's initiative have served as such a forum.

A total of 11 workshops have been held since the start. From originally having participants only from Sweden, Canada and the USA, the group has been expanded. Participants have also been invited from countries that are interested in direct disposal of spent nuclear fuel, but are not yet conducting any experimental studies of such high-level material.

17.4 INTRAVAL

INTRAVAL was an international project aimed at validating calculation models for radionuclide transport in the geosphere. The project was a follow-up of the previous projects HYDROCOIN and INTRACOIN. All of these projects were initiated by SKI, which also appointed the secretariat that coordinates the work within INTRAVAL.

A total of 14 test cases were included in the project. The results of selected laboratory tests, field tests and studies of natural analogues were evaluated. In many of the cases, it was possible for different model groups to perform predictive modelling before the measurement results became available.

Five of the fourteen test cases were SKB-linked:

- laboratory tests of migration in overcored fractures/KTH and tracer tests at Finnsjön within the fracture zone project/SGAB,
- Stripa 3D migration/KTH,
- the Poços de Caldas project,
- colloid transport/BGS,
- redox front/KTH.

The final report on INTRAVAL was published in 1991. A new phase, INTRAVAL II, was started in 1990. This phase will primarily be devoted to validation of models based on field measurements and on natural analogues. There are fewer test cases than in phase I, and they include validation exercises within the areas scale dependencies, heterogeneity and coupled processes. The project is scheduled to be concluded in the spring of 1994.

17.5 DECOVALEX

Interest in developing coupled models has increased in recent years. The purpose is to be able to describe conditions in the near field of a repository in particular with greater realism. Within the framework of the DECOVALEX project (international cooperative project for the DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation), development and verification of coupled thermo-hydro-mechanical models is being conducted. SKI initiated the project and is also the organization in charge of its execution. Nine countries are participating in the project.

17.6 COOPERATION WITH TVO, FINLAND

SKB has a regular exchange of experience within the nuclear waste management field with TVO in Finland. Several joint research projects have been carried out during the past three-year period. The results of the joint work on ice ages and their possible impact on a final repository for high-level waste were reported in 1991, for example. A joint study of available redox capacity in crystalline rock was also conducted during 1991.

An intensive cooperative programme concerned with alternative repository designs, canister fabrication and material selection has been carried out over the past two years. Models of canisters on a scale of 1:4 have been jointly developed for the purpose of demonstration. An intensive exchange of information is also taking place within the field of strategy and technology for geoscientific site investigations.

17.7 COOPERATION WITH CEA, FRANCE

17.7.1 Clay

A cooperation with CEA regarding clay studies was initiated in 1985. The cooperation has included coordination of research projects and information exchange regarding the relationship between the microstructure, mineralogy etc. of smectite clays and the influence of temperature and irradiation. Hydrothermal tests and irradiation have been carried out during year-long experiments in the laboratory. The irradiation tests have been conducted on SKB's reference clay Mx80 and French smectite clay in a simulated canister environment in the laboratory at Saclay. Tests were conducted at Stripa up until 1991 with highly compacted French smectite clay in a simulated repository environment at about 170°C. Studies of rheological properties have been conducted in the laboratory in Sweden. The cooperation has provided good opportunities for comparisons between the two countries' reference clays for buffer materials, methods for measurement of properties, swelling pressure, hydraulic conductivity, thermal conductivity etc., and technical methods for deposition.

17.7.2 Natural analogues

SKB is involved in the Oklo project in Gabon, where remains of natural reactors have been found. The project is an EC project, but is under the leadership of CEA.

17.8 COOPERATION WITH EURATOM, EC

17.8.1 COCO

The working group COCO (Colloids and Complexes) was formed by the EC to explore the importance of colloids and organic complexes for the migration of radionuclides. An important part of the cooperation is comparative experiments with different methods used at different laboratories. SKB is funding the participation of a Swedish specialist within the field.

17.8.2 CHEMVAL

CHEMVAL is a EC project for verification and validation of chemical equilibrium programmes and coupled models for geochemistry transport.

Stage 1 involved verifying different equilibrium programmes against each other, while Stage 2 involved validating the programmes against natural groundwaters. Stage 3 is verification of coupled models, and Stage 4 entails their validation. The project is also developing its own thermodynamic database and performing sensitivity studies on equilibrium programmes. The entire project was concluded in 1990.

A new phase of the project, CHEMVAL 2, started in 1991 with participants from the EC countries, Sweden, Finland and Switzerland. The project is expected to extend until 1994 and will encompass work within the fields temperature effects, ionic strength effects, organic complexes, sorption, co-precipitation and coupled geochemical transport.

17.8.3 Natural Analogue Working Group, NAWG

NAWG was formed in 1985. Its purpose is to bring together modellers who work with performance assessment of repositories for radioactive waste and scientists within geology etc. so that the best possible exchange of experience can be obtained from studies of natural analogues. Furthermore, the group serves as a forum for presentation and discussion of ongoing national and international studies of natural analogues. The group also provides advice and recommendations within its field of research. SKB contributes with financing of experts in NAWG.

17.9 COOPERATION WITHIN OECD NUCLEAR ENERGY AGENCY

17.9.1 RWMC

One of the OECD/NEA's principal areas of cooperation is radioactive waste management in the member states. The question is dealt with by the Radioactive Waste Management Committee (RWMC), where SKB is represented through Per-Eric Ahlström. Some work is carried out in joint international projects, and work groups are formed to facilitate information exchange or prepare material as a basis for joint decisions or coordination.

Seminars and workshops are arranged within important areas to document and discuss the state of the art and the future direction of the work.

The groups and projects within the area of radioactive waste management where SKB is participating with personnel or funding are listed below.

PAAG (Performance Assessment Advisory Group) acts in an advisory capacity to RWMC in matters pertaining to cooperation on goals, means and methods for performance and safety assessments of final disposal systems.

Member from SKB: Tönis Papp.

SEDE (Site evaluation and design of Experiments for Radioactive Waste Disposal) is advisory to RWMC in matters pertaining to geoscientific site investigations and experimental activities in the member states.

Member from SKB: Bengt Stillborg.

The Cooperative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects serves as a forum for information exchange and cooperation with regard to different decommissioning projects all over the world.

Member from SKB: Hans Forsström. SKB is also funding a programme coordinator, Shankar Menon, Studsvik Energiteknik AB.

The Stripa project is being concluded in 1992, see chapter 13. The project manager and head of project administration is Bengt Stillborg.

Members from SKB: P-E Ahlström (chairman of Joint Technical Committee), Hans Carlsson, SGAB (member of Joint Technical Committee) and Bengt Stillborg (project manager).

SKB is also participating in a couple of working groups under **PAAG**.

The Working Group on the Assessment of Future Human Actions at Radioactive Waste Disposal Sites is working with questions concerning different types of human intrusion in final repositories for radioactive waste. The group was founded in 1990.
Member from SKB: Torsten Eng

PSAG (Probabilistic Safety Assessment Group) is a cooperative group between those who develop and use mathematical models for probabilistic assessments of repository systems. The main emphasis is on coordinating the development, and comparing the quality, of the models.
Member from SKB: Nils Kjellbert.

17.9.2 TDB

The TDB (Thermochemical Data Base) project is under the direction of the OECD/NEA. The goal is to develop a chemical thermodynamic database for a number of elements that are of importance for the safety assessment of the final disposal of radioactive waste. The development of the database entails not only collecting and storing published data, but also a critical review. The review is carried out by a group of international experts selected for each element. The work on uranium resulted in a book published in 1992 /17-2/. At present, neptunium, plutonium, americium and technetium are being reviewed.

The TDB project is a particularly valuable initiative to develop a well-documented, reviewed and internationally accepted database. SKB is supporting the activity and Swedish experts are participating in the review work. For SKB, as well as for other participants, it will of course be necessary to have an operational database available before TDB for different calculation tasks. However, the results obtained from TDB will be entered into the database as they become available. A good example of this is the Uranium Database at SKB.

17.10 COOPERATION WITHIN IAEA

Cooperation is also being pursued within the International Atomic Energy Agency, IAEA, with regard to the management of radioactive waste.

The cooperation is being pursued in different ways, such as the publication of reports constituting:

- proceedings from international symposia,
- guidelines and standards within established areas of activity,
- status reports and methodology descriptions within important areas undergoing rapid development.

The IAEA recently appointed an expert advisory group for its waste management programme (the International Waste Management Advisory Committee, INWAG) and arranges opportunities for information exchange within different special areas through Joint Research Programmes. The IAEA publishes an annual catalogue of ongoing research projects within the waste management field in member states. SKB often supports the participation of Swedish experts in the compilation and/or review of IAEA reports.

The IAEA has also taken a new initiative through the RADWASS programme, aimed at producing international safety standards and guidelines. SKB is participating in this work.

SKB is also participating in an IAEA/EC programme concerning the validation of biosphere models. This programme is called VAMP.

17.11 OTHER COOPERATION

SKB is participating in BIOMOVS, which is an international study for comparing models that describe how radioactive materials disperse in ground and water.

The project is being run jointly by the following organizations:

- The Atomic Energy Control Board of Canada.
- AECL Research, Canada.
- Centro de Investigaciones Energeticas Medioambientales y Technologicas, Spain.
- Empresa Nacional de Residuos Radiactivos, Spain.
- The Swedish Radiation Protection Institute.

17.12 RESEARCH DURING THE PERIOD 1993—1998

SKB will actively follow foreign programmes within the field of nuclear waste management during the period. Through the formal agreements that are already established with organizations from the countries of greatest interest from the Swedish point of view, information is obtained continuously on changes, schedules etc. SKB's participation in international conferences and symposia will be active during the period, both to gain a sounding board for what is happening within the Swedish programme and to keep up with what is happening internationally.

Participation in international projects and working groups will also be an essential part of SKB's activities during the period. The international participation in the Äspö Hard Rock Laboratory will be intensive and serve as a source of information exchange within interesting subject areas. Coordination of research work within bilateral and multilateral projects can produce positive effects in the form of lower costs and wide dissemination and review of the research results.

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