RD&D-PROGRAMME 95

Treatment and final disposal of nuclear waste

Programme for encapsulation, deep geological disposal, and research, development and demonstration

September 1995

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FOREWORD

The Act on Nuclear Activities (SFS 1992:1536) prescribes in Section 12 that a programme shall be prepared for the comprehensive research and development and other measures that are required to safely manage and 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 in the month of September every third year.

The purpose of this fourth programme is to fulfil the obligations described above. The programme takes up where the programme described in RD&D-Programme 92, and in the supplement to it submitted in 1994, leaves off. The emphasis is on implementation of the projects that are required in order to commence deposition of encapsulated fuel in accordance with the plans presented in the aforementioned programme. The programme also covers the supportive research and development activities that are required for the aforesaid projects, as well as follow-up of and research on alternative methods.

This report describes the programme in its entirety. Several important studies have also been completed during the preparation of this RD&D-programme. Thus, for example, the final results of a feasibility study for Storuman Municipality have been published. Additional reports scheduled to be published in the autumn of 1995 include a feasibility study for Malå Municipality, a nationwide survey of preconditions and background for the siting work – General Study 95 – and a template for safety reports – SR 95 – which describes methods for and the outline of forthcoming descriptions of the long-term safety of a deep repository. These reports contain material of great interest for SKB's continued programme.

Stockholm, September 1995

SWEDISH NUCLEAR FUEL AND WASTE MANAGEMENT COMPANY

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RD&D-PROGRAMME 95 – BRIEF SUMMARY

In RD&D-Programme 92, SKB presented a partially new strategy for its activities. The new strategy entailed a focusing and concentration on the implementation of deep disposal of a limited quantity (about 800 tonnes) of encapsulated spent nuclear fuel during the coming 20-year period. Following this initial deposition, the results of the work will be evaluated, and only then will a decision be taken as to how and when regular deposition of the main body of the fuel and other long-lived nuclear waste will take place.

The planning in RD&D-Programme 92 was based on the assessment that available knowledge is sufficient in order to:

- select a prioritized system design for management of the spent nuclear fuel,
- designate candidate sites for the deep repository,
- characterize these sites,
- carry out the necessary safety assessments, and
- adapt the configuration of the repository to local conditions.

Events since the presentation of RD&D-Programme 92 have confirmed and strengthened this assessment. After comprehensive review and commentary, the programme strategy was accepted in all essential respects by the Swedish regulatory authorities and Government.

At the end of 1992, SKB focused and intensified its work on the planning, design and siting of a plant for encapsulation of spent nuclear fuel and of a deep repository. The necessary development work is being coordinated with the planning and design work in the manner described in the Government-requested supplement to RD&D-Programme 92. The same applies to the research and development that is needed to carry out safety assessments and provide a basis for future safety reports.

It has been proposed that the encapsulation plant be situated at the central interim storage facility for spent nuclear fuel, CLAB, at the Oskarshamn Nuclear Power Station. Siting of the deep repository will take place in stages, and the work has been commenced with feasibility studies. These feasibility studies, which are planned for five to ten municipalities, are taking more time than was predicted in 1992. After the feasibility studies, geoscientific site investigations of two sites are planned. After this, one site will be selected for detailed characterization with shaft/tunnels to repository depth. A summarizing account of the nationwide general studies has been requested by the Government and the regulatory authorities. Such an account will be published in a separate report in the autumn of 1995.

The goal is to commence deposition of encapsulated fuel in 2008. However, the time schedule must be flexible to allow enough time for the siting of the deep repository to be completed and for related decisions to be taken. SKB's ambition is to carry out siting and construction of the required facilities in consensus with the concerned municipalities and local populations. The work of carrying out an environmental impact assessment (EIA) in an open and broad process occupies an central role in this context. The Government has stipulated that the county administrative boards in the concerned counties shall have a coordinating function in this EIA process. In its decision regarding SKB's supplement to RD&D-Programme 92, the Government has also clarified certain important questions in the licensing process, for example the link between the encapsulation plant and the deep repository, as well as the fact that the commencement of detailed characterization for the deep repository also implies the start of construction of the deep repository and therefore requires permission under both the Act Concerning the Management of Natural Resources (NRL) and the Act on Nuclear Activities (KTL).

The time schedule is also dependent on the fact that certain knowledge must be available before the next step is taken. Among other things, various choices and applications must be based on comprehensive assessments of the long-term safety of the deep repository. These will be based on data available at the time, whose accuracy will gradually be improved. This means that the time schedule will also be affected by the pace at which the required continued development work can be carried out. SKB believes that the uncertainties surrounding the time schedule can be overcome and that there is a good chance the target date will be reached.

Important development work is planned within the following areas:

- canister fabrication and canister sealing – a testing plant for sealing and non-destructive testing is being considered,
- design of canister insert,
- design of handling equipment for deposition of encapsulated fuel,
- material and methods for backfilling of deposition tunnels and other rock caverns,
- scrutiny of uncertainties and validity of the methods to be used in safety assessments,
- continued development of methodology for definition of scenarios to be described in safety reports.

R&D work aimed at refining knowledge and data for the performance of safety assessments is continuing within such fields as geoscience, chemistry, natural analogues and biosphere, as well as with regard to properties of spent nuclear fuel and buffer materials.

The Äspö Hard Rock Laboratory (HRL) is a central resource for continued development and research on barrier functions, measurement methods and work methods. A comprehensive programme with verifying tests in accordance with the plans presented in RD&D-Programme 92 has been initiated and will continue during the coming years.

Broad international cooperation constitutes an important component of SKB's work. Nine foreign organizations from eight countries are participating in the work at the Äspö HRL under bilateral agreements. An extensive international exchange of information is taking place within other areas as well. Through this cooperation, Sweden is obtaining direct access to world-leading experts in many fields. This contributes to upholding the high quality of the R&D work.

SKB will continue to follow the development of alternative methods for handling, treatment and final disposal of spent nuclear fuel, for example by supporting Swedish research within a certain key areas.

In addition to the technical and safety-related aspects, it is important to continue to develop the forms for communication of knowledge and facts on nuclear waste management in society. SKB will devote considerable efforts to the implementation of the EIA process in conjunction with siting and construction of both the encapsulation plant and the deep repository. This will require broad and objective information presented in a pedagogical fashion and received with an open mind.

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SUMMARIZING OVERVIEW

1 INTRODUCTION

The goal of SKB's radioactive waste management activities is the safe disposal of all radioactive waste products arising at the nuclear power plants and other nuclear installations in Sweden. Furthermore, SKB must safely dispose of all other radioactive waste that arises in Sweden.

Radioactive waste from the Swedish nuclear power programme varies in form and activity content, from virtually inactive trash to spent nuclear fuel, which has a very high activity content. Different waste forms must therefore be managed and disposed of in different ways.

Research on the management and disposal of radioactive waste began on a large scale in Sweden in 1975. The work thus initiated has led over the subsequent 20 years to the development and construction of a system which today manages all radioactive waste generated by nuclear power activities in Sweden, as well as by medical care, research and industry.

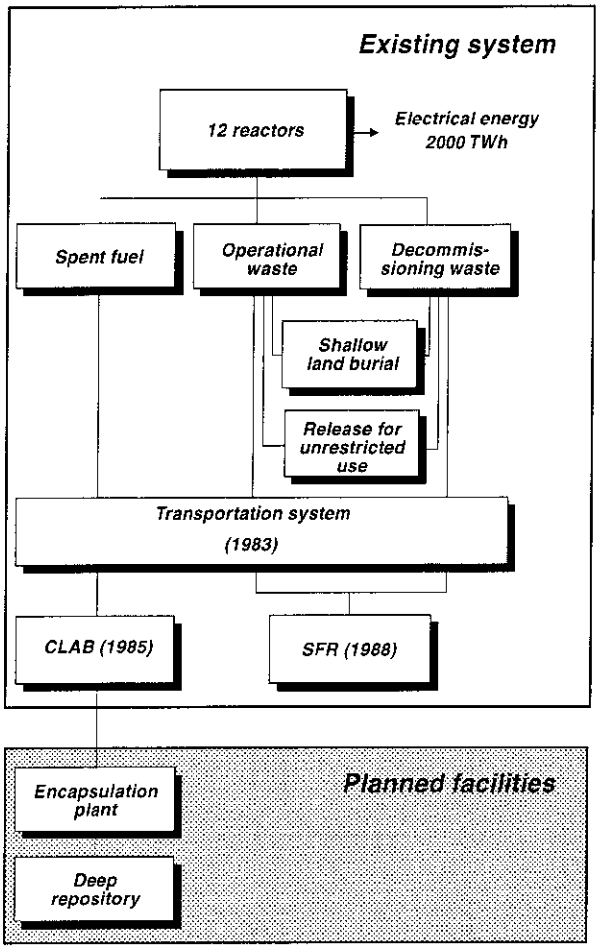


Figure 1. The Swedish waste management system.

Figure 1 provides an overview of the different parts of the Swedish waste management system. They are described in detail in the annual report of the costs of waste management (PLAN 95 is the most recent edition) submitted by SKB.

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

- Short-lived waste is disposed of as soon as possible after arising.
- Spent nuclear fuel is stored for 30–40 years before being emplaced in a deep repository. This limits heat generation in the deep repository.
- Other long-lived waste is disposed of in connection with the deep repository for spent nuclear 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 radioactive operational waste, SFR, and the transportation system. Construction of additional storage space in CLAB and SFR is planned just after the turn of the century in order to be able to take care of all spent fuel and waste from the Swedish programme.

What remains to be built of the required system is an encapsulation plant for spent nuclear fuel and a deep repository for long-lived waste, plus the necessary modifications of the transportation system to enable it to ship encapsulated nuclear fuel and other waste to the deep repository. Finally, SFR needs to be expanded to receive decommissioning waste.

2 GOAL

In RD&D-Programme 92, SKB presented its – and thereby the Swedish nuclear power industry's – plans for execution of the deep disposal of the long-lived radioactive waste, including the spent nuclear fuel that arises in conjunction with the operation of the Swedish nuclear power plants. These plans entail the following, in brief:

- The goal is, in compliance with all environmental and safety requirements, to commence deposition in a deep repository of a small portion (about 800 tonnes) of the spent nuclear fuel in the year 2008.

- The preferred alternative is encapsulation in copper canisters. Deep geological disposal is intended to be carried out in accordance with the KBS-3 concept, or a closely-related optimized version of this concept, at a depth of about 500 metres in crystalline rock.

- An encapsulation plant will be built as an addition to CLAB.
- A deep repository will be sited at a suitable location in Sweden where both rigorous safety requirements can be fulfilled and the necessary activities can be carried out with the consent and consensus of the concerned municipality and the concerned population.
- The safety and radiation protection aspects will be thoroughly penetrated and reported before decisions are made on essential binding measures.

This is the same target deadline as that given in RD&D-Programme 92. However, as a consequence of developments since that time, the time schedule for permit applications has been somewhat delayed.

The time schedule for the work with the deep repository and the encapsulation plant is not completely under SKB's control, however. It is also dependent to a high degree on the acceptance which SKB's activities receive in concerned municipalities and on the pace at which different decisions can be taken at both the local and national levels. The schedule is also affected by the local technical and geological conditions that exist on the sites that will be investigated. SKB believes that the uncertainties surrounding the time schedule can be overcome and that there is a good chance the target date will be reached.

The planning in RD&D-Programme 92 was based on the assessment that available knowledge is sufficient in order to select a prioritized system design for management of the spent nuclear fuel, designate candidate sites for the deep repository, characterize these sites, adapt the configuration of the repository to local conditions and carry out the necessary safety assessments. Events since the presentation of RD&D-Programme 92 have confirmed and strengthened this assessment. After comprehensive review and commentary, the programme strategy was accepted in all essential respects by the Swedish regulatory authorities and Government.

At the same time as work proceeds on this main task, SKB will follow and, to a limited extent, fund R&D on alternative lines of development.

3 STEP-BY-STEP DEVELOPMENT AND CONSTRUCTION

In recent years, Swedish environmental legislation has introduced requirements on environmental impact assessment – EIA – of the facility or the system for which a permit or licence is being applied. The background documents for the law state that an EIA shall also shed light on the consequences of alternative methods/designs, including a so-called "zero alternative" for the case that the facility or system for which a permit or licence is being applied for is not realized. In the debate concerning the Swedish nuclear waste programme, criticism is sometimes raised that "no method has yet been chosen", or "SKB is rushing the siting of the deep repository, no method has been developed yet", and so on. But a historical retrospect shows that a gradual narrowing-down of the studied methods for treatment and disposal of the high-level waste has taken place. Furthermore, choices have been made between different fundamental categories of methods. This process of narrowing-down and selecting alternatives has been presented and discussed in the programmes reported at regular intervals and has been based on a broad review and commentary procedure and extensive public scrutiny. Through this process, the principal strategy of the Swedish nuclear waste programme has been developed in consensus between industry, regulatory authorities and the Government. In conjunction with upcoming siting applications for the planned facilities, alternatives will be described and considered for the method, the design and the siting for which the application applies.

The gradual focusing of the programme that has taken place since the Aka Committee report in the 1970s has entailed that the work has, since the mid-1980s, been oriented towards the final goal of deep geological disposal in the Swedish bedrock of the spent fuel without prior reprocessing. Different alternative designs of a deep repository have been studied. The results of these studies have been reported in the programmes published by SKB in 1989 and 1992. The conclusion drawn in RD&D-Programme 92 is that *Deep geological disposal is intended to be carried out in accordance with the KBS-3 concept or closely-related optimized version of this concept, at a depth of about 500 metres in crystalline rock.* This strategy for the first construction stage (see below) has been accepted in all essential respects by the Swedish regulatory authorities and Government.

At present, all spent nuclear fuel is stored in CLAB or at the nuclear power plants. Experience from long-term storage of zircaloy-encapsulated fuel dates back to the late 1950s. This experience shows that storage is possible for at least 50 years, and can probably be extended to at least 100 years if necessary. In so far as storage time and storage capacity are concerned, it is therefore possible to prolong storage at CLAB for several more decades. This will be a natural "zero alternative" in the future EIA work for the deep repository and the encapsulation plant. Supplementary safety assessments (time horizon hundreds of years) are planned for this alternative.

Construction of the deep repository is planned to take place in two stages with a thorough evaluation of the first stage and a new, updated safety account and review before the second stage commences. The first stage is intended to include about 800 tonnes of spent nuclear fuel (about 10% of the expected quantity) and can be completed in about 20 years at the earliest. It will be possible in the subsequent evaluation to take into account all development during these 20 years. If deemed desirable, the already deposited fuel can be retrieved for alternative treatment.

Decisions on siting, construction and operation of the encapsulation plant and deep repository will be taken in steps after licensing review based on a gradually improving knowledge base, where new knowledge can be taken into account at each step. An important component is the EIA process, which will lead to environmental impact assessments (EIAs) for the facilities. This process will involve both SKB and central authorities on the one hand, and the concerned municipalities, county administrative boards and local stakeholders on the other hand. The EIA process has already begun for the encapsulation plant, while for the deep repository it is being prepared for in the feasibility studies and will be carried out to its full extent in conjunction with the site investigations. In reality, a great deal of the work was already conducted in the spirit of the EIA process through the comprehensive public scrutiny of SKB's plans and programmes long before this was made a formal requirement.

4 ENCAPSULATION OF FUEL

The programme for canisters and encapsulation mainly consists of development and fabrication of canisters and planning, design and construction of an encapsulation plant.

The canisters must be designed and fabricated so that they remain intact during a very long time under the conditions that will prevail in the deep repository. In other words, they must not be penetrated by corrosion in the groundwater present in the rock, or be squeezed apart by the mechanical stresses to which they are subjected in the deep repository.

To achieve this, the canister is planned to consist of an insert of e.g. steel, which provides mechanical strength, and an outer shell of copper, which provides corrosion protection. The work of formulating the design requirements for the canister is under way. Requirements during the fabrication of the canister and during handling in the encapsulation plant, transportation and in the deep repository are also being taken into account.

As described in RD&D-Programme 92, encapsulation is planned to take place adjacent to CLAB. The fuel will be received in the encapsulation plant from the storage pools in CLAB. After the fuel assemblies have been checked and dried, they will be placed in the canister. Before the lid on the inner steel container is put in place, the air in the canister will be replaced with inert gas. Then the copper canister will be sealed by attachment of a copper lid by means of electron beam welding. Stringent requirements will be put on the leaktightness of this weld and the ability to test this leaktightness.

In designing the encapsulation plant, great emphasis will be placed on radiation protection for the personnel and the surrounding environment. This means, among other things, that the actual encapsulation procedure will be performed by remote control in heavily radiation-shielded compartments known as hot cells. A large portion of the handling of canisters will also be done by remote control. Experience from CLAB and SFR, as well as from various foreign facilities with similar handling of fuel, will be drawn upon.

4.1 CANISTER DEVELOPMENT

The most important questions surrounding the development of the canister concern material selection, fabrication method, and sealing and testing method. The design of the insert is also of importance for the practical handling of the canisters and to avoid the risk of criticality should the canister become filled with water.

4.1.1 Material selection

Copper has been chosen as a canister material because it does not corrode in oxygen-free water. The life of the copper canister from a corrosion viewpoint is therefore determined by the quantity of dissolved corrosive substances, mainly sulphides, that come into contact with the canister surface. The greatest corrosion depth on a copper canister is conservatively estimated to be about 5 mm in 100,000 years in the groundwater that is expected to be present at great depth in Swedish bedrock. The state of knowledge within the field of corrosion of pure copper is deemed to be good. However, in the event an alloyed copper material should be used in the future, supplementary corrosion studies will be required.

Beside corrosion properties, the creep properties, grain size and weldability of copper are also of importance. These properties are largely dependent on the fabrication method and the choice of copper grade. Verifying investigations of these properties are planned as a basis for the choice of method and grade.

4.1.2 Fabrication

Tests have been conducted of different fabrication methods for the copper canister (and the steel insert) over the past few years. The tests have gradually increased in scale, and the first full-scale canisters were produced in the summer of 1995. Two methods have been tested: roll pressing and extrusion. In the former case, two tube halves are rolled and bent, after which the two halves are welded together with an electron beam weld. In the latter case, a full-length tube is fabricated directly by extrusion. After machining of the copper tube, a bottom is welded on by electron beam welding in both cases before the steel insert is put inside. Further fabrication trials are planned both with the already tested methods and possibly with another method, for instance powder metallurgical fabrication with hot isostatic pressing (HIP) or electrodeposition. The purpose is to show that reproducible and testable canisters can be manufactured in serial production.

4.1.3 Sealing

When the canister has been filled with fuel it will be sealed and the sealing weld checked. Electron beam welding at reduced vacuum is planned to be used for sealing. A trial series of lid welding on full scale has been carried out with good results. Before a final choice of welding method is made, however, modifications of lid and weld design are necessary. Furthermore, development work is needed on the welding equipment to achieve the necessary capacity and reliability.

Development work on non-destructive testing of the weld is proceeding in parallel. The requirements on this equipment are determined by what defects must be able to be detected. Work is currently being done to define acceptable defect sizes. The methods being tested for non-destructive testing are ultrasonic (the pulse-echo method) and radiographic testing. Different types of scanners are being developed to improve the resolution of these methods. Results obtained to date indicate that these methods will be applicable, but that further development work remains to be done.

4.1.4 Full-scale tests

Sealing and inspection of the seal are crucial in order to obtain a leaktight canister. So far tests have been conducted under laboratory conditions. SKB is now considering fabricating a trial series of full-sized canisters in a separate pilot plant. This would then be constructed over the next few years so that the results of the tests can be taken into account in the engineering work for the encapsulation plant.

4.1.5 Insert in copper canister

Different designs are being considered for the insert. Tests have been made with an insert in the form of a steel tube. In order to obtain sufficient margin to criticality (in the event the canister becomes water-filled) in this version, the void around the fuel assemblies must be filled with a particulate material, e.g. glass beads. An alternative design of the insert is currently being studied, where the insert takes the form of a cast container with channels for the fuel assemblies. The insert can be made of cast steel, cast iron or bronze. Trial fabrication of a cast insert is planned. With this design, the criticality problem can be managed without extra filling.

4.2 DESIGN OF THE ENCAPSULATION PLANT

The goal of the ongoing work is that the encapsulation plant should be finished and ready to deliver encapsulated fuel to the deep repository in 2008. The work of designing the encapsulation plant is being pursued step-by-step in accordance with established routines for the construction of power plants and similar industrial facilities. A feasibility study has been carried out and the results reported in an initial step.

The work is currently in the preliminary design phase. The results of this phase will serve as a basis for SKB's decision to apply for a permit to build the facility. The work is being conducted in such a manner that it will be possible to submit the permit application during 1997. In conjunction with the application, SKB must also give an account of how serial fabrication of canisters will take place, and how the canister will meet the requirements made for long-term safety. The date of the application will therefore be dependent on how fast facility design, canister development and the safety assessment for canisters in the deep repository proceed.

An environmental impact assessment (EIA) must be compiled in support of the application. Discussions are held with the county administrative board, municipality and safety authorities about what is to be included in the EIA and how consultation in conjunction with the EIA work is to take place.

5 DEEP REPOSITORY

The programme for siting and construction of the deep repository was presented in RD&D-Programme 92 and its supplement. The time schedule has been modified since site investigations have not yet begun. The task during the period 1996–2001 is to compile supporting material for an application for permission to site and build the deep repository on a specific location. The most important near-term goal is to commence site investigations.

The work with the deep repository encompasses the following:

- background material and plans for design, construction, operation and closure of the deep repository,
- siting studies (general studies, feasibility studies, site investigations) as a basis for future choice of site for the deep repository,

- background material and plans for investigation and evaluation of candidate sites.

5.1 DESIGN OF THE DEEP REPOSITORY

A deep repository is an industrial establishment with facilities both below and above ground. The work with the design of the deep repository aims at achieving good function with respect to safety, environment and technology. Questions being investigated are concerned with the advantages and disadvantages of different repository concepts, construction methods, technology for deposition, closure and retrieval, safeguards, industrial safety and effects on society and the environment.

SKB has begun the planning work by preparing general plant descriptions. These provide examples of some possible ways to design the deep repository with its buildings, land areas, rock caverns, tunnels and shafts. They also contain requirements on and principles for the various functions of the deep repository. A possible schematic design of the deep repository is shown in Figure 2. To a large extent, construction and operation of the deep repository can be based on experience and proven technology from nuclear installations and underground rock facilities. Special attention is devoted to, for example, the impact of the rock construction work on the surrounding rock, methods for manufacture and application of the bentonite buffer, and technology for backfilling and sealing the repository. Studies of crushed rock as an aggregate material instead of sand, which was the principal alternative in KBS-3, indicates that crushed rock, possibly mixed with 10–20% bentonite, can also provide the desired function for backfilling of deposition tunnels etc.

A special study has been made of the advantages and disadvantages of different repository depths. The study considers depths down to 2000 metres. The conclusion is that the advantages that can be achieved with a deeper location do not compensate for the growing difficulties encountered in building the repository and in investigating and characterizing the rock.

The continued work of designing the repository includes adapting the layout to the conditions on specific sites as site investigations get under way. Extensive development of technology is needed to arrive at purpose-suited handling equipment for transferring canisters from transport casks in the deposition tunnels and emplacing them in the deposition holes.

Much of the technology for handling of buffer material and canisters will be tested in the Äspö HRL. The effects of the deep repository on society and the environment will be explored in conjunction with the siting studies. Studies performed to date show that it should be possible to adapt the design and operation of the deep repository so that the effects on the environment will be small.

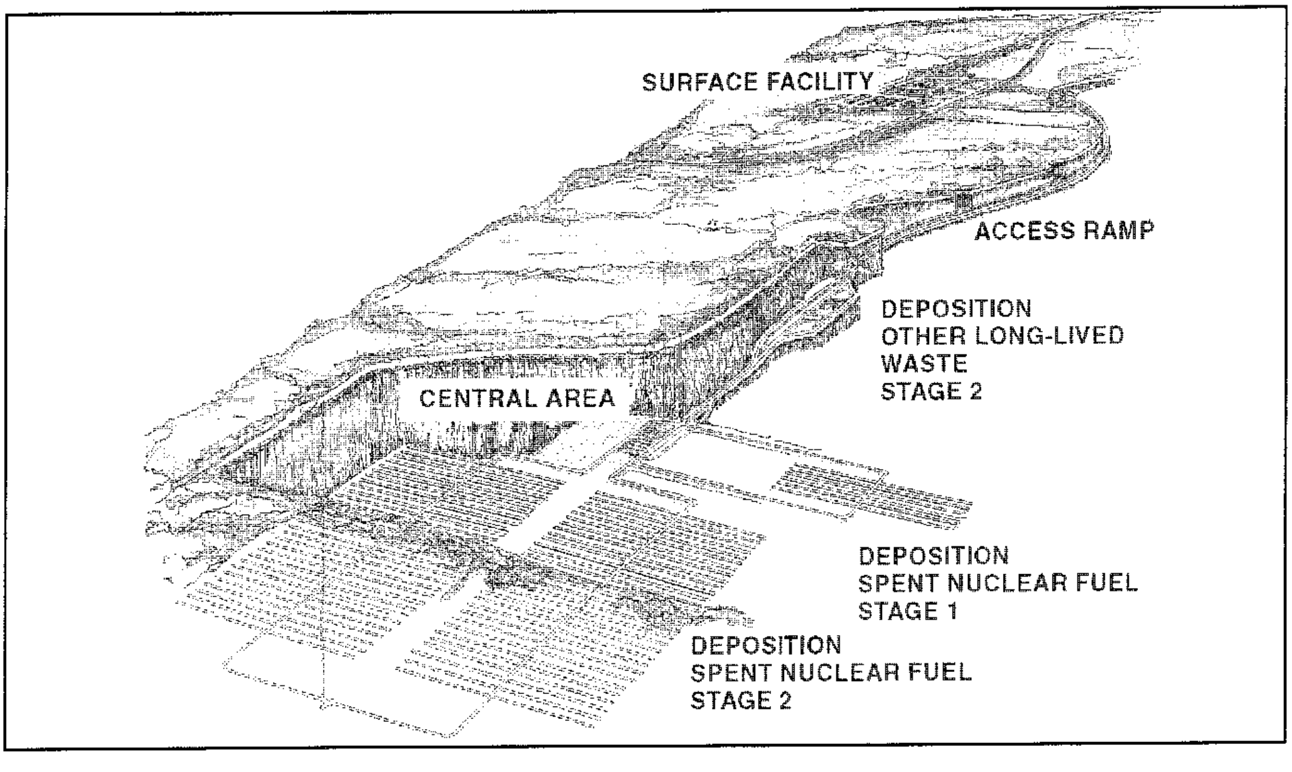


Figure 2. *Schematic drawing of the deep repository.*

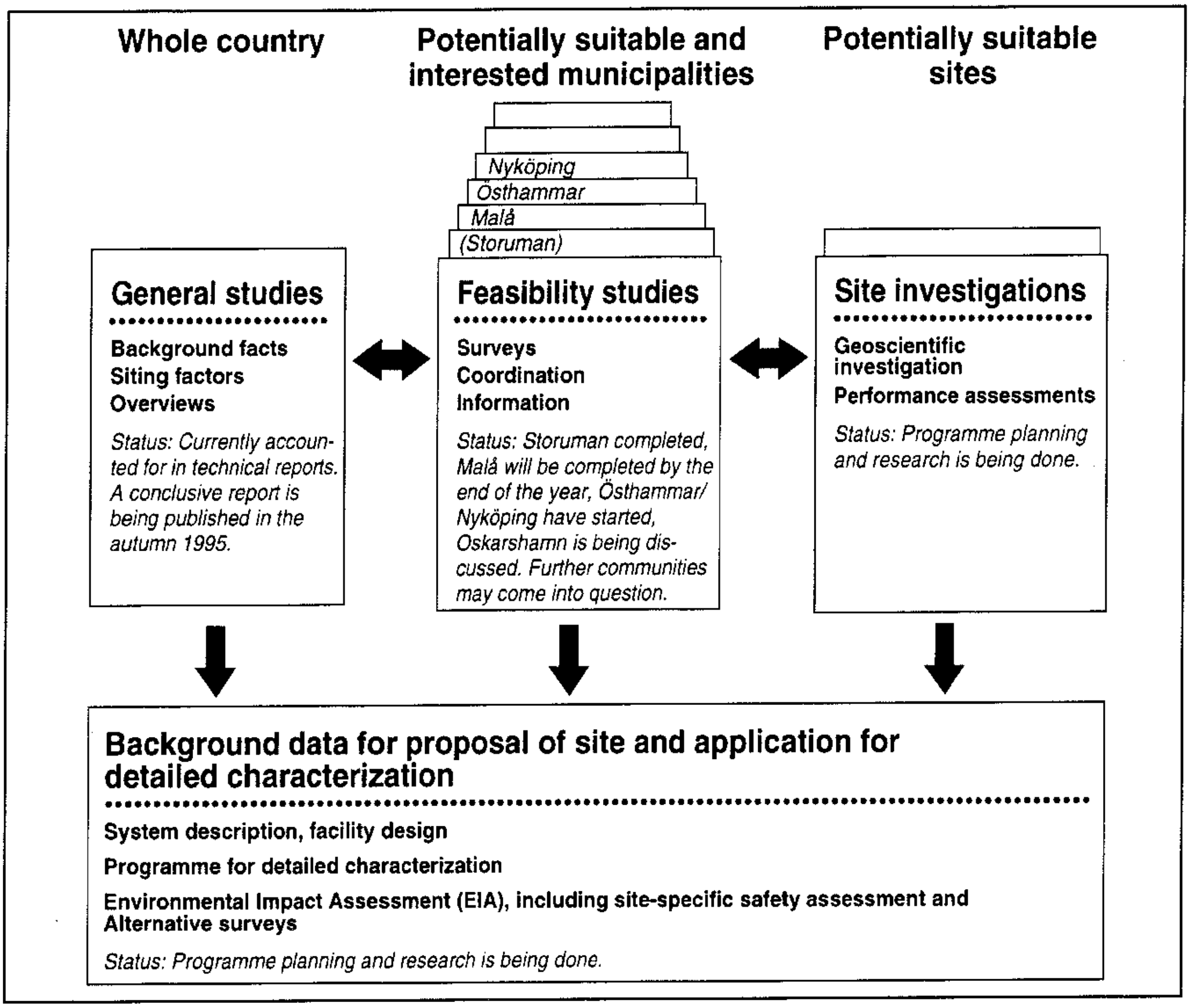


Figure 3. *Main components in the siting work, plus completed and ongoing activities.*

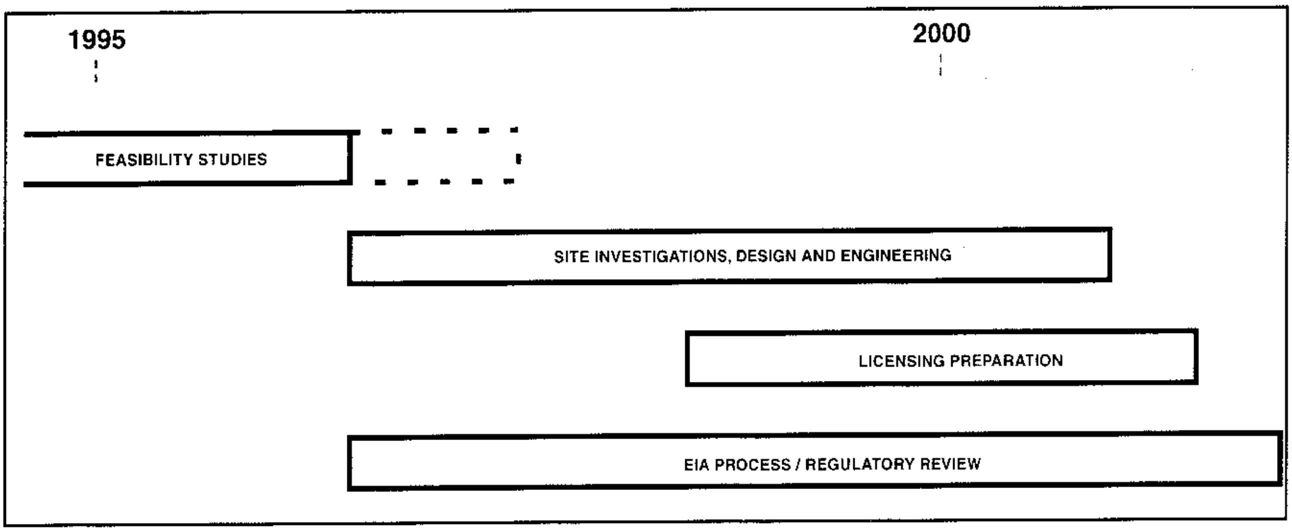


Figure 4. *Schedule for phase 1, siting.*

5.2 SITING STUDIES

SKB intends to obtain background data for the siting of the deep repository by means of

- **feasibility studies** (in 5 to 10 municipalities), aimed at identifying interesting areas for site investigations and illustrating the possible consequences of a deep repository siting in the municipality and the region,
- **site investigations** (on two sites), aimed at providing background data for designing a deep repository with respect to the properties of the site and carrying out an environmental impact assessment with an assessment of long-term safety. Site investigations entail measurements in deep boreholes, among other things.

This work is intended to lead to a proposal to carry out a detailed characterization and construct a deep repository on a site.

Guidelines for the siting studies are the criteria for safety, technology, land and environment, and societal aspects which SKB set forth in RD&D-Programme 92 and its supplement.

In discussions about siting of the deep repository, it is often said that "SKB must find the best site", by which is understood to mean the "best" from a safety viewpoint. However, there is no best site, objectively speaking. The site must meet very stringent safety requirements. A holistic assessment of long-term safety requires access to site-specific data on bedrock conditions. Such data can only be obtained by means of comprehensive investigations on sites selected on the basis of incomplete data. Thus, when it has been demonstrated that safety requirements have been met, it is not meaningful to talk about "even better sites". This has also been pointed out by the Swedish Nuclear Power Inspectorate in its review comments on RD&D-Programme 92.

The main components of the work, and how far the work has progressed, are illustrated schematically in Figure 3.

A time schedule for the siting studies is shown in Figure 4. It is based on the assumption that the feasibility studies can be carried out efficiently from a technical point of view, and that the site investigations can commence in 1997. The timing of the start of the site investigations is critical for the timing of the entire siting programme.

The siting work is supported by general studies, which provide a general background on the fundamental conditions all over the country or parts thereof.

Investigations within the framework of the general studies have been carried out continuously since the mid-1970s. A collective account is published in a separate report in conjunction with this RD&D-Programme 95. A large number of geoscientific and societal factors that are or may be of importance in connection with the siting of the deep repository are described in the report.

The study shows that the fundamental prerequisites with regard to land and rock conditions can be found at many places in most parts of the country. To identify interesting areas for site investigations, a thorough analysis is required of local factors that cannot be judged on a country scale but can best be carried out in feasibility studies on a municipal level. A subsequent holistic assessment of, above all, long-term safety requires access to site-specific data on the bedrock.

The general studies do not provide any direct guidance for identifying areas of interest for locating a deep repository. Their scale is too coarse for that. However, the general studies do provide sufficient information to identify parts of the country which are not of interest for more detailed siting studies. These areas are mainly, for geological reasons, the Scandinavian mountain range, Skåne and Gotland. Areas situated in unusual rock types or where valuable ore deposits are likely to be found are not of interest either. The same applies in areas protected by law, such as national parks, nature conservation areas, and so forth. The general studies and underlying material are also used by SKB in contacts with municipalities to make preliminary assessments of advantages and disadvantages for siting of a deep repository and thereby of the area's interest for feasibility studies by SKB.

5.2.1 Feasibility studies

In the feasibility studies, the prospects for and consequences of a deep repository siting within specific municipalities are examined. The studies are based principally on existing material. SKB needs no formal permits to conduct a feasibility study. However, in practice the feasibility studies are conducted in consensus between SKB and the municipality in question. A feasibility study takes approximately one year to complete and results in a comprehensive body of material which is supposed to provide a good picture of the general prospects for siting of a deep repository in the municipality. Areas that may be of interest for site investigations with reference to their bedrock and soil conditions are identified. Technical and infrastructure aspects with regard to construction and transportation are explored. Possible consequences (positive and negative) for the environment, local economy, business community and society are illuminated.

A feasibility study should thus furnish a broad body of facts for both the municipality and SKB. Both parties can then decide for themselves whether they are interested in starting a site investigation. The same facts are made available to all stakeholders, who thereby have an opportunity to have a say and state their views long before any decisions need to be taken on the siting of the deep repository.

SKB has conducted a feasibility study in Storuman Municipality. The final report was published in February 1995. The feasibility study shows that good prospects may exist for a deep repository in the municipality. The municipality held a (local) referendum on 17 September 1995 regarding the question: "Should SKB be allowed to continue searching for a final disposal site in Storuman Municipality?" The result was 28% Yes, 71% No and 1% Blank votes. Voter turnout was about 73%. This means that SKB will cease its work in Storuman.

Another feasibility study has been under way in Malå since the winter of 1994. A summarizing status report was published in May 1995. The final report is expected to be ready in the autumn of 1995.

In a special survey of a general nature, SKB has examined the prospects for feasibility studies of municipalities with nuclear activities (Varberg, Kävlinge, Oskarshamn, Nyköping, Östhammar). For all municipalities except Kävlinge, the survey showed that feasibility studies are of interest for SKB's part. Such studies have now been started in consultation with the municipalities of Nyköping and Östhammar and are being discussed with the municipalities of Varberg and Oskarshamn. Feasibility studies may be undertaken for several additional municipalities as well.

The feasibility studies have led to extensive discussions, particularly in the concerned municipalities, but also regionally and on the national plane. SKB as well as other concerned parties (authorities, policy-makers, the general public) have gained important and valuable experience for the continued work. Difficulties and possibilities in the siting work have been clarified. The work with feasibility studies has taken longer than SKB planned in RD&D-Programme 92, in part because the issue arouses a great deal of discussion right from the start.

The siting of the deep repository is a key issue in Sweden's nuclear waste programme. With the feasibility studies, the concrete work of siting has got under way in various parts of the country. Further persistent and open-minded efforts are needed to arrive at a good solution that meets the requirements of both environment and safety on the one hand and local consensus and understanding on the other.

5.2.2 Site investigations

SKB plans to conduct site investigations within prioritized areas in two of the municipalities where feasibility studies have been conducted. A large body of data will be gathered on each site by means of the following activities:

- Geoscientific investigations in the field.
- Facility design and planning.
- Performance and safety assessments.
- Analysis of societal aspects and land and environment aspects.
- Preparation of an environmental impact assessment in an EIA process.
- Compilation of material for licensing process.

Preparations and planning work are currently under way in all these areas. In part the work is building further on the data gathered in connection with the feasibility studies. The largest efforts during the site investigation phase will be the geoscientific investigations and the performance and safety assessments based on them.

5.3 GEOSCIENTIFIC FIELD INVESTIGATIONS

Even though Swedish bedrock in general offers good conditions for a deep repository, it is the local conditions in the bedrock that will determine the suitability of a site. Before site investigations start, an investigation programme will be presented in a special report. The site investigations are intended to provide a broad geoscientific understanding of the site and its regional environs. They also aim at gathering geoscientific data for a site-adapted layout of the repository and an analysis of the long-term safety of the repository. The most important factors that need to be clarified in the field work are enumerated and discussed in the supplement to RD&D-Programme 92 as well as in the present programme.

The site investigations will be conducted in several stages. The purpose of the initial stage is to verify the feasibility study's assessment of the area's suitability and to identify more precisely in which part of the area the investigations are to be concentrated. Then fracture zones, rock type boundaries and other geological conditions are surveyed in depth with an increasing degree of detail. Measurements and analyses of groundwater chemistry, groundwater movements and rock stresses comprise an important element.

Facility design and safety assessments are carried out in interaction with the investigations. Provided that each stage of the investigations indicates suitable conditions, the programme is carried to completion, providing a comprehensive body of support material for a permit application.

Methods and technology for geoscientific investigations in the field have been included in SKB's programmes for a long time. The development of these methods and technology began with the study site investigations during the 1970s and 1980s at some 10 locations in the country. Prior to designing the investigation programme, the findings of these investigations were re-examined. Furthermore, important method development took place in the international Stripa project. The work for the Äspö Hard Rock Laboratory can be said to have constituted a "dress rehearsal" of the investigation methodology (see section 8). This work has entailed comprehensive testing and coordinated evaluation on a real site and under realistic conditions before the methods are applied to the deep repository project.

6 SAFETY ASSESSMENT

6.1 SAFETY FUNCTIONS

The safety of the repository is dependent on the toxicity and accessibility of the waste. The assessment of the repository's safety is influenced by time in that the quantity of toxic radionuclides declines, and in that the quantification of the repository's safety functions decreases in precision with time. Potential transport pathways for radionuclides to man can change with time, but such changes will take place at different rates for the different parts in the barrier system. Experience shows that essential changes in the biosphere occur on the time scale 100–1000 years. The geological environment deep down in the Fennoscandian Shield, however, exhibits stable conditions on a time scale of millions of years.

Consequently, the feasibility of quantifying the safety of the repository (or the risk from the repository) is dependent on the period of time in which one is interested. The Swedish National Radiation Protection Institute has discussed the influence of the time horizon on radiation protection and finds that:

- Particularly great attention should be given to describing protection for the period up to closure of the repository and during the first thousand years thereafter, with a special focus on nearby residents.
- The individual dose up to the next ice age, i.e. up to about 10,000 years, should be reported as a best estimate with an estimated margin of error. Environmental protection should be described for the same period of time.
- For the period from the next ice age onward, qualitative assessments should be made of what might happen with the repository, including scenarios taking into account the risk of increased releases.

To achieve the desired safety during the construction of a deep repository, during the operating phase and during the long-term containment phase, requirements are put on the function of the repository and its components. The composite function of all the repository's components must together provide adequate safety.

In order to achieve long-term safety, the disposal system is designed to isolate the spent nuclear fuel from the biosphere. This isolation is achieved by encapsulating the spent nuclear fuel in impervious canisters which are deposited deep in the crystalline bedrock on a selected repository site. If this isolation should be broken, the repository has the function of retaining the radionuclides and retarding their transport. Furthermore, transport pathways and dilution conditions in the biosphere can be controlled so that any radionuclides that escape will only reach man in very small quantities.

The materials used in the repository have been selected with a view to the possibility of verifying their long-term stability and safety performance in the repository with experience from nature. For the same reason, the thermal and chemical disturbance which the repository is allowed to cause in its surroundings is limited. The safety philosophy for the deep repository is based on the multibarrier principle, i.e. safety must not be dependent on the satisfactory performance of a single barrier.

The safety functions can be divided into three levels:

Level 1 – Isolation

Isolation enables the radionuclides to decay without coming into contact with man and his environment.

Level 2 – Retardation

If the isolation is broken, the quantity of radionuclides that can reach the biosphere is limited by:

- very slow dissolution of the spent fuel,
- sorption and very slow transport of radionuclides in the near field,
- sorption and slow transport of radionuclides in the bedrock.

Level 3 – Recipient conditions

The transport pathways along which any released radionuclides can reach man are controlled to a great extent by the conditions where the deep groundwater first reaches the biosphere (dilution, water use, land use and other exploitation of natural resources). A favourable recipient means that the radiation dose to man and the environment is limited. The recipient and the transport pathways are, however, influenced by natural changes in the biosphere.

The safety functions at levels 1 and 2 are the most important and the next-most important. They are achieved by means of requirements on the properties and performance of both engineered and natural barriers and on the design of the deep repository. Within the frames otherwise defined, good safety function at level 3 is also striven for by a suitable placement and configuration of the deep repository.

6.2 SAFETY ASSESSMENTS

A number of major decisions and permits are required in order to site, design and build the facilities and systems

Table 1. Forthcoming safety assessments.

Safety assessment	Bases for decision on permit for: Encapsulation plant	Deep repository
Encapsulation plant SR-I	Siting Construction	
Deep repository SR-D		Siting Construction, incl. detailed geoscientific investigations and some construction
Initial operation (Stage 1)	Start encapsulation – spent fuel Initial operation – Stage 1	Initial operation – Stage 1 – deposition, gradual extension of deposition tunnels
Regular operation (Stage 2)	Supplementary extension Regular operation – Stage 2	Regular operation – Stage 2 – deposition, gradual extension of deposition tunnels
Decommissioning	Decommissioning	Possible supervised storage, closure

required to dispose of radioactive waste, see Table 1. The bases for these decisions include accounts of both radiological safety during operation of the facilities or systems, as well as of safety during the long-term passive deep disposal period that begins when the repository has been sealed.

For this safety accounting, assessments must be performed of the safety of the systems and facilities and of the potential of the different candidate sites to serve as good repository sites. Systematic methods have been developed to carry out the assessments. The radiological safety of the personnel and the environment during the operating phase will be evaluated and reported on in accordance with the practice that has evolved at the nuclear power stations. These assessment methods and reporting principles have also been applied to the transportation system, CLAB and SFR.

The evaluation of the long-term safety of a deep repository basically follows the same principles as those established for operating safety. Since a deep repository has to function in close interaction with the natural geological environment, however, it has been necessary to develop specific methods to carry out these assessments. The assessment of the safety of the deep repository must project conditions over very long spans of time. The borderline between normal and abnormal conditions can then be difficult to identify.

Safety assessments of entire disposal systems or the assessment of the performance of individual barriers or sub-systems follow an established work sequence:

- Definition of the purpose of the assessment.
- Definition of given assumptions for the assessment, i.e. types and quantities of radioactive waste, the disposal system and its dimensions, and the location and surroundings of the repository.
- Definition of the scope and delimitations of the assessment, and of the safety goals.
- Clarification of both the probable and the less probable or improbable conditions for which the system/facility is to be assessed (scenarios).
- Clarification of the time-dependent processes which are essential for the intended performance of the system/facility in different scenarios.
- Definition of calculation models for quantifying the performance of the repository and the couplings between the models, where possible.
- Quantification of the performance of the repository and essential changes in performance.
- Qualitative assessment of important but non-quantifiable processes or events that can impact on the performance of the repository.
- Discussion of the uncertainties in qualitative and quantitative sub-assessments and evaluation of their validity with respect to the purpose of the overall assessment.

The development in Sweden of a methodology for assessments of long-term safety has been under way since the end of the 1970s and is still being pursued in close international cooperation. The general method development has in the main been concerned with the scenario concept and handling of uncertainty/validity. Specific development of models and data to quantify the processes essential for safety was a dominant activity during the '80s.

A summary of the state of knowledge as a result of a review conducted by experts within the OECD Nuclear Energy Agency, the International Atomic Energy Agency and the European Commission is given in an International Collective Opinion. There it was observed

- that safety assessment methods are available today to evaluate adequately the potential long-term radiological impacts of a carefully design 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.

It was also noted that the collection and evaluation of data from proposed repository sites are tasks on which further progress is needed, and that the methods for safety assessments can and will be refined as a consequence of ongoing activities. SKB shares this opinion.

The principal method development work during the '90s within SKB has been concentrated on developing practical methods for determining scenarios, carrying out model calculations and examining the validity and relevance of the results. In view of the fact that safety reports are to be made on several occasions, discussions have also been held with the authorities on what a safety report should contain and how it can be organized. This work will continue and will be brought into line with the regulations currently being formulated by the Swedish Nuclear Power Inspectorate and the National Radiation Protection Institute.

A fundamental methodology for carrying out safety assessments is presented in the description of the state of knowledge. The detailed account of methods and of the numerical tools which SKB has at its disposal, as well as the account of quality and the confidence which SKB has in these models, has been gathered in a separate report designated SR 95. This has been organized in a way that could serve as a suitable model (template) for the upcoming safety reports. It is being finalized in conjunction with this programme.

7 SUPPORTIVE RESEARCH AND DEVELOPMENT

The supportive R&D activities are aimed at:

- building up a good understanding of the phenomena and processes that can be of importance for long-term safety in the deep geological disposal of radioactive waste,
- refining models for essential processes in the repository,
- building up the necessary databases and comparison material in preparation for the planned site investigations so that the performance and safety of the repository can be evaluated.

In RD&D-Programme 92, SKB concluded that an adequate understanding and capability for quantification of the safety of the deep repository existed in order for SKB to begin siting and construction of the facilities required for deep disposal. The development work that has been carried out since then has gradually taken on a clearer focus on

- methodology for practical site investigation,
- methodology for construction and verification of barriers,
- review of other waste forms than spent nuclear fuel, and
- evaluation of the dependence of safety on extreme events such as ice ages, major earthquakes, intrusion, etc.

These conditions are reflected both in the contents of the account and in the continued programme.

Even though the scope of non-project-bound R&D activities continues to diminish, this does not mean it can be dispensed with. Integrated experiments, practical tests and site-specific bedrock investigations are expected to continue to identify questions and problems that require further deepening of knowledge and databases. Specific needs for deepened understanding can also arise from the need to predict the probable future evolution of the repository without pessimistic simplifications, and the need to better quantify the safety margins in present-day designs.

SKB is anxious to maintain a high level of competence within areas essential to safety assessment and execution of the programme's key areas.

7.1 SPENT NUCLEAR FUEL

SKB's fuel studies have been going on for more than 15 years now. The methods for characterizing the spent fuel and measuring how radioactive materials in the fuel are released in contact with groundwater have undergone considerable development and a large database has been collected. The emphasis of the investigations has gradually shifted from idealized experiments under "pure" conditions to greater realism under the conditions expected in a deep repository.

The findings from the fuel studies have been continuously applied to define the source term in different consequence calculations. The models for how radionuclides are released are still highly simplified, with a large overestimation of the release rate for most radionuclides. SKB's continued efforts will be directed at trying to develop a more realistic release model, where allowance is made for both thermodynamic and kinetic conditions.

7.2 BUFFER AND BACKFILL

Knowledge of bentonite as a buffer material and for possible admixture in the backfill material has been accumulated by means of studies in laboratories and in the field. The state of knowledge is now deemed to be adequate to justify the use of bentonite in the repository. A summary account of data and properties is currently in preparation.

Continued studies will be concentrated on improving knowledge concerning the behaviour of bentonite during wetting, validation and refinement of calculation models, and manufacturing and application technology. Studies concerning the interaction between bentonite and cement are under way, as is scrutiny of the bentonite's influence on living conditions for microbes.

The performance of the bentonite during the wetting phase will be tested in large-scale experiments.

7.3 BEDROCK

With the advent of the Äspö HRL, the studies of the suitability of the Swedish bedrock for deep geologic disposal have been focused on site-specific investigations. The processes that are important for the safety of the deep repository have been identified. Methods have been developed and tested under field conditions for characterizing the rock and measuring the parameters that are needed to quantify these processes.

Models have now been developed to a level that makes it possible to define and minimize various safety-related effects on a deep repository with the aid of descriptions and calculations. Continuing work is aimed at comparing the consequences of different conceptual assumptions. The ambition is also to be able to validate the models used wherever possible. This will continue on the basis of site-specific data accumulation both in Sweden and abroad.

The supportive geoscientific R&D is now focused on broadening and deepening knowledge concerning long-range or general questions, such as:

- examination of how future glaciations may develop and what their effect will be on a deep repository,
- examination of tectonic conditions in a geological time perspective as a point of departure for assessments of the long-term mechanical stability of the deep repository,
- how are the results of site characterization and modelling dependent on the investigation scale,
- studies of the natural variations of the groundwater chemistry around a deep repository in time and space,
- learning to understand how the chemical composition of the groundwater reflects the results of the water's long-term movements in fractures in the bedrock,
- development of databases for conditions in the bedrock at levels deeper than 1,000 m.

Other R&D efforts will be focused on the questions that may come up in connection with the forthcoming site investigations.

7.4 CHEMISTRY

The processes that are important for the performance of the deep repository and for the release, transport and retardation of the radionuclides have been identified. Thermodynamic databases have been developed to provide support for assumptions concerning chemical speciation, solubility, etc. Laboratory measurements have yielded databases for the diffusion of different substances in buffer and rock matrix, as well as for the sorption of these substances in near-field materials or in the rock.

Numerical models have been developed to quantify release, transport and retardation. In part, these models have been structured to be conservatively simplified, which means that if a process cannot be quantified in detail, the calculations should not underestimate the negative consequences of the process for the repository.

The ongoing work is largely focused on further examining various phenomena that could negatively affect the barriers or processes that are important for the intended function of the repository. Examples are the effects of micro-organisms on the canister, groundwater chemistry and transport, formation of colloids and complexes etc. that could hasten the transport of the radioactive substances in the rock. Other efforts are examining the foundations of the sorption concept to enable the transport models to be validated as far as possible.

7.5 BIOSPHERE

Through studies of the transport of radionuclides and other contaminants in the biosphere, a considerable database and modelling capability has been built up within the field. An inevitable uncertainty in predictions over long periods of time arises due to the fact that changes in the biosphere take place at a far greater pace than the changes in the bedrock and in the vicinity of the deep repository.

This means that dose calculations only have a relatively "short-lived" (perhaps 1000-year) relevance. This limits the usefulness of the concept of "dose" as an index of the repository's impact on the human environment. Dose calculations are, however, often used over longer time spans as a means of weighing together conceivable effects of releases of many different radionuclides.

Efforts within the biosphere area are currently being focused on establishing and testing methodology for establishing the properties in site-specific biospheres that are essential for the dose calculations. To start with, a survey of areas that could be recipients for groundwater coming from repository depth is planned. Other efforts are being focused on limiting the uncertainties by trying to delimit the possible changes that will occur with time on the basis of site-specific or regional conditions. Extensive international cooperation is under way to compare different ways to model transport in the biosphere, and to forecast the future evolution of the biosphere.

7.6 NATURAL ANALOGUES

The processes that influence safety in a deep repository are known and can be tested in laboratories or studied through examples in nature. The latter provide an opportunity to study slow processes that occur over very long time spans.

Through observations and measurements of selected natural systems (natural analogues), it has been possible over the past ten years to study and test most of the processes that are expected to influence the performance of the barriers and thereby the safety of a final repository. Examples of such processes are mineral alterations, coprecipitation, corrosion, dissolution processes, matrix diffusion, etc. plus the presence of organic matter, microbes and colloids. Furthermore, the analogues provide an opportunity to study the interaction of several processes with an interdisciplinary approach that is difficult to duplicate in a laboratory.

The studies of analogues to the function of the deep repository will continue in order to test the validity of models and to underpin long-range forecasts.

7.7 OTHER WASTE

Other categories of waste besides spent nuclear fuel will also be disposed of in the deep repository. In order to avoid complicating the chemical conditions in particular, these wastes will be stored in separate caverns. An inventory of quantities, activity content and chemical form has been performed as a prestudy of how the waste is to be packaged and disposed of. The work is now continuing in a second phase with analyses of essential functions and possible scenarios in preparation for safety assessments, decisions on repository layout and design.

8 ÄSPÖ HARD ROCK LABORATORY

The Äspö HRL constitutes an important part of SKB's work to design a deep repository and to develop and test methods for investigating a suitable site. In the autumn of 1986, SKB initiated field work for the siting of the underground laboratory in the Simpevarp area in the municipality of Oskarshamn. Construction was started in the autumn of 1990. The underground portion is a tunnel running from the Simpevarp Peninsula to the southern part of Äspö Island. On Äspö Island, the 3,600 m long tunnel runs in two turns down to a depth of 450 m. The last 400 metres were excavated with a tunnel boring machine (TBM) with a diameter of 5 metres. The first part of the tunnel was excavated by drill-and-blast. The construction work was completed during 1995.

During the pre-investigation phase, 1986–1990, extensive investigations were carried out of the natural conditions in the bedrock both from the ground surface and from an extensive set of boreholes. Predictions were made with respect to the geological, geohydrological, geochemical etc. conditions that would be observed during the construction phase. During the construction phase, 1990–1995, extensive investigations and experiments were carried out in parallel with the tunnelling work. The operating phase commenced in 1995. This programme defines the thrust of the investigations and tests that are planned to be carried out during the operating phase.

The activities at the Äspö HRL have also attracted great international interest. Participation agreements exist with nine foreign organizations.

8.1 GOALS

One of the fundamental motives for SKB's decision to build the Äspö HRL was to create an opportunity for research, development and demonstration in a realistic and undisturbed rock environment down to the depth planned for a future deep repository.

In the planning and design of the activities at the Äspö HRL during the operating phase, priority is being given to projects which aim at:

- increasing scientific understanding of the deep repository's safety margins,
- developing and testing technology which reduces costs and simplifies the final repository concept without sacrificing high quality and safety,
- demonstrating the technology that will be used for the deposition of spent nuclear fuel and other long-lived waste.

8.2 RESULTS

At the start of the work on Äspö, the following stage goals were defined:

- Demonstrate that investigations on the ground surface and in boreholes provide sufficient data on essential safety-related properties of the rock at repository level.

- Refine and verify the methods that are needed for characterization of the rock in the detailed characterization of a site.

The first of these stage goals has been fulfilled now that the construction phase has been completed and the final reports on the investigations and research done in connection with construction are being completed during 1995 and early part of 1996. The results on the whole show that the methods that are available for investigating rock are well-suited to gathering the knowledge and the data on the bedrock at a specific site that are needed to construct a deep repository and demonstrate that it fulfils the safety requirements.

The second stage goal has been fulfilled in part with the work that has been conducted in parallel with the construction. Additional studies that will be concluded during the next few years are being pursued within the framework of this stage goal, including:

- Detailed investigations of the disturbed zone around blasted and drilled tunnels (the ZEDEX project).

- Development of an interactive computer system (Rock Visualization System) for interpretation and presentation of measurement results as well as design of the repository.

- Field and laboratory experiments aimed at deepening knowledge on the capacity of the rock to retain or retard the transport of radionuclides in fractured rock (Tracer Retention Understanding Experiments).

- Field and laboratory experiments aimed at studying how and at what rate the oxygen present in the repository at closure is consumed by reactions with the rock (REX – Redox experiments on detailed scale).

- Experiments in a specially developed borehole laboratory (CHEMLAB) which permits chemical experiments to be conducted under repository-like conditions with respect to groundwater composition and pressure. The experiments are being conducted to verify the models and check the constants that are used to describe the dissolution of radionuclides in groundwater, the effects of radiolysis, fuel corrosion, sorption on mineral surfaces, diffusion in the rock matrix, diffusion in backfill material, transport out of a damaged canister and transport in a single rock fracture.

- Coordinated evaluation of hydrogeological groundwater flow models and hydrogeochemical mixing models.

- Field and laboratory experiments to investigation to what extent degassing of groundwater at low pressures affects measurements of hydraulic characteristics in tunnels and boreholes situated under ground.

A considerable portion of the work on evaluation of models is being carried out in the internationally composed Äspö Task Force on Groundwater Flow and Transport of Solutes.

The Äspö HRL also provides an opportunity to test, investigate and demonstrate on a full scale various components in the deep disposal system that are of importance for long-term safety. It is also important to show that high quality can be obtained in the design, construction and operation of a deep repository.

The following ongoing and planned projects are concerned with these questions:

- Design, build and test the function of a prototype repository in the Äspö HRL. A full-scale prototype of the deep repository will be built to simulate all stages in the deposition sequence in a realistic environment. It will also provide an opportunity to observe a simulated repository during several years before the time comes to deposit the first canister in the deep repository.

- Test the bentonite buffer's performance in the deep repository environment over a long period of time

8.3 PROGRAMME FOR 1996–2001

Several projects at Äspö are aimed at evaluating the usefulness and reliability of different models that describe the rock's barrier function and at developing and testing methods for determination of parameters included in the models. Studies are also being conducted of the disturbance entailed by construction and operation of a repository in the rock in order to ensure that the disturbance does not have a negative influence on the long-term safety of the repository.

The following ongoing and planned projects are concerned with these questions:

- Detailed characterization of fractures in order to develop models for and data on fracture properties that can be handled in radionuclide transport calculations.

(up to 20 years) in order to test models and confirm results from laboratory experiments.

- Test the hydraulic and mechanical properties of different tunnel-back-filling materials and develop and test technology for compaction and retrieval of tunnel back-fill.

- Refine statistic methodology for estimating the portion of the deposition tunnels' length that can be used for emplacement of canisters based on geological, rock-mechanics, hydrogeological and other information from different investigation phases.

- Test injection methodology to verify know-how and technology for grouting/reinforcement of large transmissive discontinuities and strongly water-conducting discontinuities of moderate thickness and extent.

9 ALTERNATIVE METHODS

Deep geological disposal of long-lived radioactive waste is generally accepted among international experts as a good method that can be implemented in a way that satisfies fundamental ethical and environmental requirements. At the same time as the research and development work on direct disposal of spent nuclear fuel is being pursued by construction of facilities for encapsulation and an initial stage of deep disposal, however, good reasons exist to allocate some resources to the follow-up of alternative methods to SKB's main line. 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 extensive international cooperation network, broad insight is ensured in the major programmes that are in progress in other countries. For certain specific lines of development with possible applications in the longer term, however, a limited domestic effort is warranted. In this way Sweden can build up domestic competence in the field which will allow Sweden to gain insight into the broader programmes being pursued in other countries. In the Government decision on RD&D-Programme 92, the Government stipulated that SKB should, in the present programme, give an account of its assessment of the alternative methods that are being considered.

The greatest interest in the scientific and general debate concerning alternative methods to deep geological disposal has been devoted in recent years to transmutation of the long-lived radionuclides in the waste. This method requires reprocessing of the spent nuclear fuel and separation of the long-lived nuclides from the rest of the waste (partitioning). With the aid of nuclear reactions, the long-lived nuclides are then converted (transmuted) to short-lived or non-radioactive elements. The technology exists today for transmutation of a considerable portion of the plutonium that is formed in nuclear fuel. For certain other long-lived elements, transmutation could be made feasible (to a limited extent) through moderate development. However, development of proposed, more advanced methods for transmutation to an industrial scale requires, in certain respects, a technical breakthrough with a long period of work and large resources. The mere fact that transmutation requires reprocessing means that the costs of waste treatment based on transmutation of long-lived nuclides at today's cost level is much higher than the costs of direct disposal of unreprocessed nuclear fuel.

It is difficult to find any economic or short-term safety-related justification for transmutation when compared with present-day industrially established systems for management of spent nuclear fuel (direct disposal and reprocessing/vitrification plus final disposal). Rather, it is found that the radiation doses to the personnel in conjunction with handling and treatment will be much greater for transmutation. Even if the development of transmutation is successful, there will nonetheless be a need for deep disposal of such long-lived waste as inevitably arises from the relatively complicated treatment process.

Other alternatives to geological disposal are not the subject of any direct interest in Sweden or internationally, having been more or less dismissed. Supervised storage is the only exception for the time being. Supervised storage can be carried out in many ways, and is currently being practised in CLAB for the Swedish spent nuclear fuel. However, supervised storage does not fulfil the long-range goal expressed in the law's requirement on final disposal in a safe manner.

Among various alternative methods for geological disposal considered in Sweden, some interest has been expressed in disposal in very deep boreholes. SKB is conducting a limited follow-up programme of this alternative. The current judgement is that extensive and prolonged research would be required to establish a deep repository based on very deep boreholes.

10 DECOMMISSIONING OF NUCLEAR FACILITIES

When a nuclear power plant or other nuclear installation is taken out of service, parts of it are contaminated with radioactivity. Decommissioning and dismantling must therefore be carried out in a controlled manner and the radioactive material must be managed and disposed of as radioactive waste. Decommissioning projects are currently under way in various parts of the world. In most cases, they involve the decommissioning of small research and prototype reactors. In recent years, dismantlement work has also begun on large nuclear reactors, for example in the USA and Germany.

Internationally, extensive cooperation exists within the decommissioning field for the purpose of exchanging experience. SKB is actively involved in such a cooperative programme within the OECD/NEA. At present, this includes 29 plants in ten countries. A few years ago, a thorough analysis of status and development needs within the decommissioning field was done within the programme. No area was identified which requires fundamental development work. A need does, however, exist to translate tested methods to an industrial scale. This is now being done in conjunction with different projects, and the work is being carried out by industrial companies. For this reason, the EU, among others, has cut back its research programme within the field.

No development work directly concerned with decommissioning has been done in Sweden. Extensive experience does, however, exist from various repair and maintenance jobs that is directly applicable to decommissioning, e.g. the thorough decontamination and rebuilding of Oskarshamn 1, and the steam generator replacements in Ringhals 2 and 3. Theoretical studies of the technology and costs for decommissioning the Swedish nuclear power plants have been conducted on several occasions, most recently in 1994. These studies are intended to provide a basis for fee calculations.

No major R&D within the field of decommissioning is planned for the coming six-year period. The work will be concentrated on follow-up of activities abroad and of major Swedish repair jobs. Further, tests will be conducted of handling and possible disposal of large components obtained from previous rebuilding jobs. At the end of the period, it may be time to begin design work on the final repository for decommissioning waste.

1 INTRODUCTION, BACKGROUND

1.1 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 installations in the country in a safe manner. Furthermore, all other radioactive waste that arises in Sweden must be safely disposed of.

The following general guidelines were presented in SKB's R&D-Programme 86 /1-1/:

- 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 stringent standards of safety and radiation protection and satisfy the requirements of the Swedish authorities.
- The systems for waste management shall be designed so that the requirements on fissile material safeguards can be satisfied.
- The waste problem shall be solved in all essential respects by the generation that utilizes the electricity generated by the nuclear power plants.
- A final 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 shall be worked out inside the country, at the same time as all available foreign knowledge on the subject shall be gathered.
- The regulatory authorities' scrutiny and directives regarding the nuclear power utilities' handling of the waste question shall guide the conduct of the work.
- The activities shall be conducted openly and with good public insight.

These general guidelines did not occasion any special comments on the part of the reviewing bodies. They were reiterated in R&D-Programme 89 /1-2/ and certain parts were then discussed and questioned with respect to the spent nuclear fuel by the National Board for Spent Nuclear Fuel. The Board's viewpoints also occasioned certain comments in the Government's decision on R&D-Programme 89. This led to further deliberation on the part of SKB, and the following conclusion was presented in RD&D-Programme 92 /1-3/:

A broad political and public opinion seems to agree on the following fundamental principles for nuclear waste management in Sweden:

- Sweden already has nuclear waste, and it must be disposed of in a safe manner within the 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* /1-4/. The same requirement is also formulated on an international level /1-5/ 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 put forth with respect to the value of preserving freedom of choice have been heeded in the current RD&D-Programme.

1.2 APPLICABLE LEGISLATION ETC.

The obligations of the owners of nuclear power reactors with regard to management and final disposal of radioactive waste are set forth in the Act /1-6/ on Nuclear Activities, in the Ordinance on Nuclear Activities /1-7/ and in the Act /1-11/ on the financing of future expenses for spent nuclear fuel etc., as well as in certain permits 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 manage 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 management 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,
- covering all costs for management and final disposal of the nuclear waste.

1.3 HISTORY

Research regarding the management 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 /1-8/. The research was intensified in conjunction with the enactment of the "Stipulation Act" in 1976/77, when the KBS (Nuclear Fuel Safety) Project was started by the nuclear utilities. The project, which was administrated by SKB, developed two final disposal methods: KBS-1 for vitrified high-level reprocessing waste (1977) /1-9/ and KBS-2 for the final disposal of unreprocessed spent nuclear fuel (1978) /1-10/.

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 /1-11/ entered into force, PRAV was abolished and the National Board for Spent Nuclear Fuel (NAK, later SKN) was created in its stead. The purpose of this Board was 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 were 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 /1-12/.

The KBS-3 report was submitted 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 /1-6/ in June 1984. A research programme /1-13/ 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 changed to be based on KBS-3.

In September 1992, SKB presented the third research programme under the Act on Nuclear Activities /1-3/. The results of SKB's research work are reported continuously in SKB's technical reports. Annual summaries are included in the SKB Annual Report /1-14, 15, 16/. A more concise annual account is also presented in SKB's annual Activities report, which is distributed widely.

1.4 RD&D-PROGRAMME 92 – EXPERT REVIEW

After RD&D-Programme 92 had been submitted to SKI in September 1992, the programme was circulated for review and comment to a large number of institutions in Sweden. On the basis of the viewpoints received and its own judgements, SKI compiled a review report and submitted it to the Government in March 1993 /1-17/. In June 1993, KASAM submitted a report with its own commentary on the RD&D-Programme to the Government /1-18/. The Government's decision in regard to RD&D-Programme 92 was handed down in December 1993 /1-19/. In it, the Government stipulated that a supplementary account should be prepared with regard to certain points. SKB submitted such a supplement in August 1994 /1-20/. This supplementary report was circulated for review and comment in the same way as the main report /1-21/ and a Government decision was handed down in May 1995 /1-22/.

Wherever possible, SKB has heeded the comments received on RD&D-Programme 92, including the supplementary report, in the present RD&D-Programme 95.

1.5 WASTE FROM THE SWEDISH NUCLEAR POWER PROGRAMME

Radioactive waste from the Swedish nuclear power programme varies widely in terms of form and activity content, 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 a handling viewpoint, it is practical to distinguish between low-level waste (LLW), intermediate-level waste (ILW) and high-level waste (HLW). LLW can be handled and stored in simple packages, without any special protective measures. ILW must be radiation-shielded for safe handling. HLW requires not

Table 1-1. Waste quantities in the Swedish nuclear waste programme.

Product	Principal origin	Unit	Number of units	Volume in final repository m^3
Spent fuel		canisters	4,500	13,500
Alfa-contaminated waste	LLW and ILW from Studsvik	drums and moulds	2,800	1,700
Core components	Reactor internals	moulds	1,400	9,600
LLW and ILW	Operational waste from nuclear power plants and treatment plants	drums and moulds	55,900	91,000
Decommissioning waste	From decommissioning of nuclear power plants and treatment plants	mainly 20 m^3 ISO containers	8,500	156,400
Total quantity, approx.			**73,100**	**272,200**

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 waste.

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 deposited 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.

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.

The different waste types were described in detail in RD&D-Programme 92 /1-3/, since which time no essential changes have occurred with regard to the different waste types that must be managed and disposed of in Sweden. Current waste quantities for different categories of waste are given in detail in the report PLAN 95 /1-23/ and summarized in Table 1-1.

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

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 1-1 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 95, which the power utilities have submitted through SKB /1-23/.

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

- Short-lived waste will be disposed of as soon as possible after it has arisen.
- Spent fuel will be stored for 30–40 years before it is emplaced in the final repository. This will limit heat generation in the final repository.

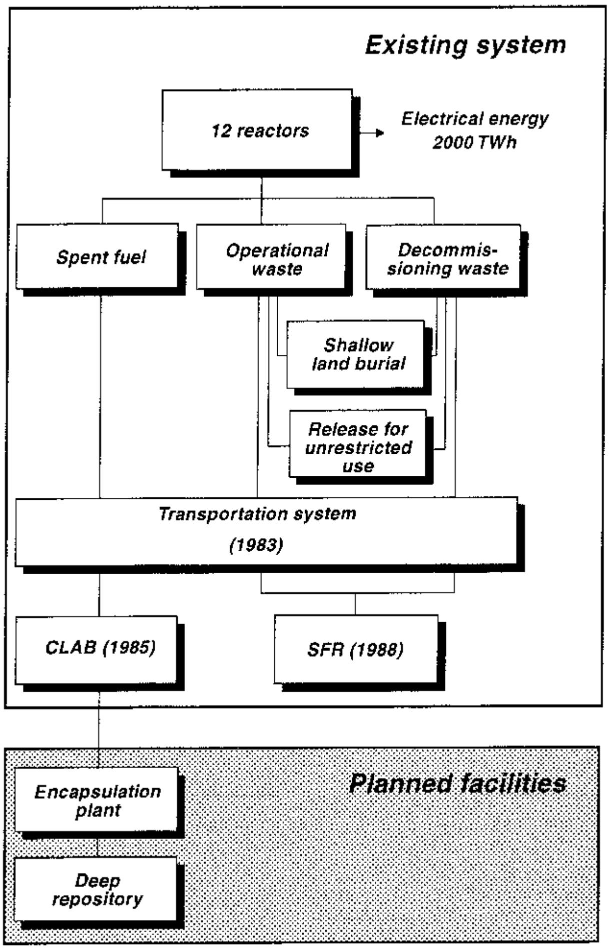


Figure 1-1. The Swedish waste management system.

- Other long-lived waste will be deposited of 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.

SFR has been in operation since 1988. It currently has capacity to accommodate 60,000 m^3 of radioactive operational waste and can be expanded for additional needs. Expansion is also planned to accommodate short-lived low- and intermediate-level decommissioning waste from nuclear facilities.

CLAB has been in operation since 1985. It currently has capacity to store 5,000 tonnes (uranium weight) of spent nuclear fuel, and plans call for this capacity to be expanded to about 8,000 tonnes at the beginning of the next decade.

The transportation system is based on sea transport and consists of a ship, M/S Sigyn, a number of transport containers and terminal vehicles. The system has been in operation since 1982 and has gradually been expanded and augmented to its present-day function and capacity.

The existing facilities and systems are operated with very high availability and good reliability, have a very good working environment and insignificant environmental impact.

What remains to be built of the necessary system is an encapsulation plant for spent fuel, a deep repository for long-lived waste and necessary additions to the transportation system to permit shipment of encapsulated nuclear fuel and other waste to the deep repository. Extensive work is under way for these parts of the system for the purpose of determining a suitable design and site. The plans for this work are described in detail in this report.

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 service and final disposal of the waste from decommissioning, see chapter 14.

2 GOAL OF THE PROGRAMME

2.1 GOAL

In RD&D-Programme 92 /2-1/, SKB gave an account of its, and thereby the concerned nuclear power industry's, planning for deep disposal of the long-lived radioactive waste, including the spent nuclear fuel, that arises in connection with the operation of the Swedish nuclear power plants. In brief, this planning entails the following:

> **The goal** is to begin deposition in a deep repository of a small portion (about 800 tonnes) of the spent nuclear fuel in 2008, in compliance with all environmental and safety standards.
>
> For this an encapsulation plant and a deep repository are required. Furthermore, additions to the existing transportation system are needed to ship the encapsulated fuel from the encapsulation plant to the deep repository. The basic scheme is encapsulation in copper canisters that meet the safety requirements imposed by the deep repository. Deep disposal is intended to be implemented in accordance with the KBS 3 concept or a closely-related optimized version at a depth of about 500 m in crystalline bedrock.
>
> The encapsulation plant will be built as an addition to CLAB. The deep repository will be located on a suitable site in Sweden that enables both the stringent safety requirements to be met and the necessary work to be carried out in consensus with the concerned municipality and the local populace.
>
> The safety and radiation protection aspects will be thoroughly penetrated and accounted for before decisions on essential binding measures are taken.

At the same time as work proceeds on this main task, SKB will follow and, to a limited extent, fund R&D on alternative lines of development mainly conducted in other countries, however. This strategy of SKB's programme, which was described thoroughly in RD&D-Programme 92, is in harmony with the provisions of the Act on Nuclear Activities /2-2/ regarding a comprehensive programme.

The main strategy described in RD&D-Programme 92 has been accepted in all essential respects by the regulatory authorities and the Government.

SKI /2-3/ states that:

...RD&D-Programme 92 meets the basic requirements set forth in Section 12 of the Act on Nuclear Activities on a programme for research and development with regard to objectives, breadth and depth.

SKI can accept that the continued RD&D measures will mainly be concentrated on a method of type KBS-3. ...A KBS-3-like repository should also be able to be designed so that it can offer a reasonable balance between abandonability, retrievability and inaccessibility for the fissile material.

SKI believes that it is a good action strategy to expand the deep repository in stages...

KASAM /2-4/ states that:

KASAM recommends that SKB focus its RD&D activities during the period 1993-1998 on a demonstration-scale disposal with option to retrieve as the first stage in the ultimate management of the spent nuclear fuel.

that this stage encompass 5-10% of the entire estimated fuel quantity from the Swedish reactor programme, and

that KBS-3 is a reasonable choice for the demonstration disposal.

The Government states in its decision of 16 December 1993 /2-5/ regarding SKB's RD&D-Programme 92 that:

The Government finds, in concurrence with SKI, that RD&D-Programme 92 meets the provisions of Section 12 of the Act on Nuclear Activities.

The Government observes that ... the work of depositing spent nuclear fuel and nuclear waste in a deep repository is planned to be carried out in two phases, namely demonstration deposition and final disposal.

The Government finds, in concurrence with SKI and KASAM, that the change of the programme has considerable advantages, even if the long-term properties of the final repository cannot be demonstrated. The Government would particularly like to emphasize that even if the KBS-3 method is a reasonable choice for demonstration deposition, SKB should not commit itself to any specific *handling and disposal method before a coherent and thorough assessment of associated safety and radiation protection aspects has been submitted.*

The continued planning of the measures required to implement the chosen main line requires a number of consequential decisions pertaining to environmental impact assessments, siting, safety accounting, investment, permits under various laws, etc. These decisions must be supported by various kinds of data, which will be obtained through the ongoing work. They may of course also lead to re-evaluation and changes of the chosen strategy.

This RD&D-programme describes the current plans in detail for the coming six-year period and in general for measures necessary in the future.

2.2 GENERAL DESIGN OF THE DEEP REPOSITORY

The following description of the deep repository follows the principles set forth in KBS 3 /2-6/ and RD&D-Programme 92 /2-1/ as well as SKB's PLAN-report /2-7/. A more detailed description of how the facilities may be designed is provided in "Brief preliminary plant description," /2-8/.

The repository is situated at a depth of about 500 m, depending on conditions at the selected site. From tunnels at this depth, deposition holes are bored in which copper canisters with spent nuclear fuel are emplaced and surrounded with bentonite clay – see Figure 2-1. The tunnels can be backfilled with a mixture of bentonite and quartz sand or other suitable material. Closely-related optimized variations of the general scheme described here may be considered.

The deep repository is built in two stages. In the first stage, approximately 400 canisters of spent nuclear fuel (about 800 tonnes uranium weight) are deposited. This initial operating period is planned to start in 2008 at the earliest and last for about 5 years, after which the experience gained will be evaluated. The option then also exists to retrieve the canisters, if this should be deemed necessary for any reason.

If the result of the evaluation is that continued deposition is suitable and acceptable – which is our expectation – the entire repository is built (stage 2) and the activities continue until all waste has been deposited, which is estimated to occur in around 2040. The total quantity of spent nuclear fuel then deposited is estimated to be about 8,000 tonnes, which is the quantity generated by the present-day Swedish nuclear power programme up to the year 2010.

Deposition of "other radioactive waste" is planned in a special part of the repository during stage 2. This waste resembles that deposited today in the Final Repository for Radioactive Operational Waste, SFR, in Forsmark, except that it contains more long-lived radionuclides than the SFR waste. The total quantity of such waste is estimated at 25,000 m^3.

The waste is deposited in three separate repository areas: area for canisters deposited during the initial operating period (stage 1), area for canisters deposited during

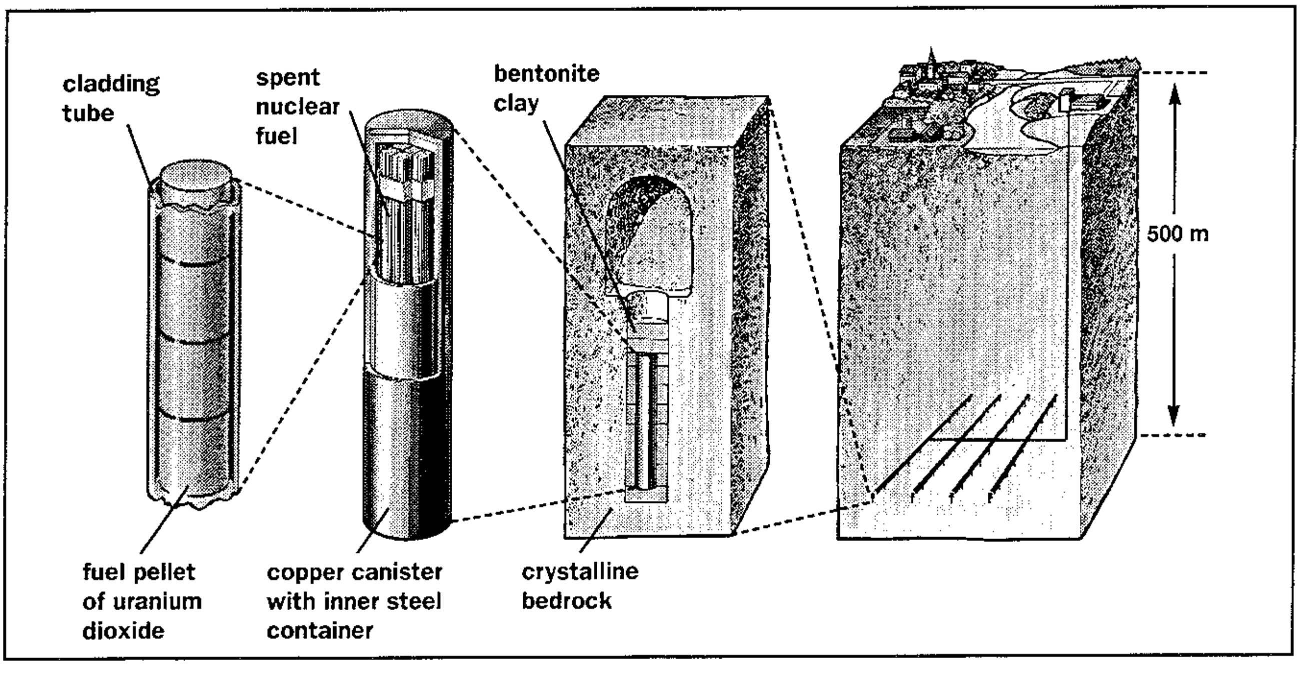


Figure 2-1. *Deep repository in accordance with the KBS-3 concept.*

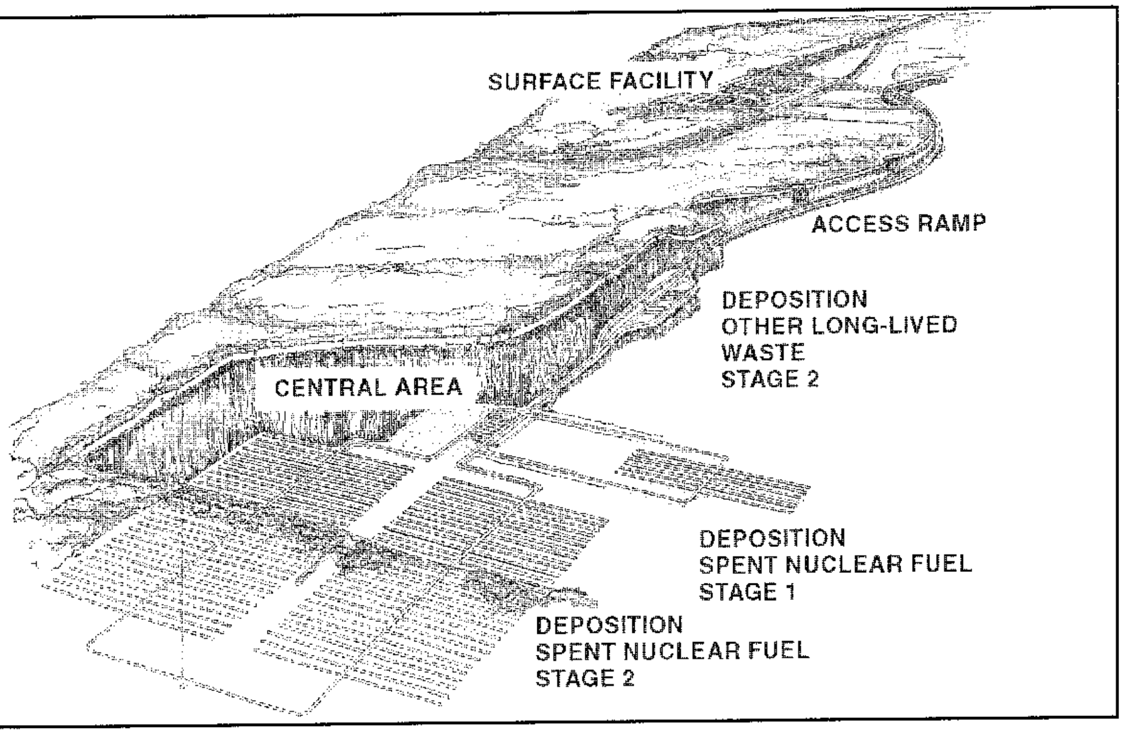


Figure 2-2. *Schematic drawing of the deep repository.*

regular operation (stage 2) and area for other waste (stage 2). Altogether, these repository areas occupy an area of about 1 km^2. Figure 2-2 shows a schematic drawing of the deep repository.

If, on the other hand, the evaluation leads to a decision to terminate deposition after the initial operation (stage 1), the option is then open to retrieve the deposited canisters and place them in interim storage.

3 STEP-BY-STEP DEVELOPMENT AND CONSTRUCTION

In recent years, Swedish environmental legislation has introduced requirements on environmental impact assessment – EIA – of the facility or the system for which a permit or licence is being applied for. The background materials for the law state that an EIA shall also shed light on the consequences of alternative methods/designs, including a so-called "zero alternative" for the case that the facility or system for which a permit or licence is being applied for is not realized. In the debate concerning the Swedish nuclear waste programme, criticism is sometimes raised that "no method has yet been chosen", or "SKB is rushing the siting of the deep repository, since no method has been developed yet", or suchlike. But a historical retrospect shows that a gradual narrowing-down of the studied methods for treatment and disposal of the high-level waste has taken place. Furthermore, choices have been made between different fundamental categories of methods. This process of narrowing-down and selecting alternatives has been presented and discussed in the programmes reported at regular intervals and has been based on a broad review and commentary procedure and extensive public scrutiny. Through this process, the principal strategy of the Swedish nuclear waste programme has been developed in consensus between industry, public authorities and the Government. In conjunction with upcoming siting applications for the planned facilities, alternatives will be described and considered for the method, the design and the siting for which the application applies.

This chapter contains a brief account of the most important steps in method choices already made. Further, the implications of a zero alternative are discussed, along with planned stages for step-by-step construction of the deep repository. Finally, EIA and the EIA process are discussed.

3.1 METHOD SELECTION

3.1.1 Deep disposal in crystalline rock

The development of the Swedish system for management of spent nuclear fuel began in the 1970s with the Aka Committee report, which was published in the spring of 1976 /3-1/. The most important proposals of the Committee were to build an interim storage facility for spent nuclear fuel and a final repository in rock for low- and intermediate-level waste. According to the proposal, the facilities should be located at one of the nuclear power stations or Studsvik, and the waste shipments should be based on sea transport. The proposals led to the construction of CLAB and SFR and the build-up of a sea transportation system during the 1980s.

The Aka Committee also proposed that

- studies should be initiated of the possibility of building a Swedish reprocessing plant,
- studies should be commenced of direct final disposal of the spent nuclear fuel without preceding reprocessing,
- final disposal of radioactive waste should take place in crystalline rock,
- geological studies of sites suitable for final disposal should be initiated.

The Aka Committee report also helped bring the nuclear waste issue into the focus of political debate. The non-socialist Government that was formed after the 1976 election proposed to the Riksdag that a Stipulation Act should be passed to lay down conditions for permits to fuel new nuclear power reactors, and such a law was passed by the Riksdag in 1977 /3-2/. Under the Stipulation Act, responsibility for pursuing development in the area of nuclear waste management was shifted to the owners of the nuclear power plants. They in turn started the KBS project, which was conducted within SKB.

The Stipulation Act required that it be demonstrated "how and where an absolutely safe final disposal of the high-level waste obtained from reprocessing can take place, or" ... "how and where an absolutely safe final disposal of spent, unreprocessed nuclear fuel can take place".

The Aka Committee report and the Stipulation Act together entailed a focusing of the work on deep disposal of the high-level waste in Swedish crystalline rock. This line was accepted by industry, the regulatory authorities and the Government, which subsequently scrutinized the various reports from the KBS project. An initial step in the choice of method was thereby taken de facto.

After the feasibility of safe disposal in accordance with the KBS concept was accepted, alternative concepts for treatment and final disposal of the spent nuclear fuel were once again discussed, for example in the 1986 research programme, pages 17–18 in part I /3-3/. There it is observed that "Accordingly, the research programme has been oriented towards the final goal that final disposal of the spent nuclear fuel **shall be achieved deep down in the Swedish bedrock**." Having found that this programme "...meets the requirements set forth in Section 12 of the Act on Nuclear Activities..." and that "...the work should in the main be pursued in accordance with the strategy .... laid down in the programme" /3-19/, the authorities have also accepted this point of departure. This has also been the case with subsequent programmes that have been scrutinized and essentially accepted by the regulatory authorities and the Government.

On the international plane as well, deep geological disposal of the high-level waste is the alternative that is recommended and preferred by industry and government authorities to an increasing degree in nuclear power countries. This is evident not least from the "Collective Opinions" issued by the OECD/NEA's Radioactive Waste Management Committee /3-17, 18/.

3.1.2 Direct disposal without reprocessing

Within the framework of the KBS project, two alternative ways were studied for treatment of the spent nuclear fuel: reprocessing and direct disposal without reprocessing. The first report from the project, KBS-1, dealt with final disposal of vitrified waste from reprocessing /3-5/. Together with a contract for reprocessing of spent fuel with the French company Cogema, this report comprised the supporting material for the applications for permits to fuel the Ringhals 3 and 4 and Forsmark 1 and 2 reactors. Permits to fuel these reactors were obtained after the referendum on nuclear power that was held in the spring of 1980.

The Riksdag decision on the use of nuclear power following the aforementioned referendum limited the number of reactors to the 12 units that were then in operation or under construction. Further, it was decided that these units were to be decommissioned not later than 2010. In addition, a strong cutback in the construction of nuclear power plants all over the world took place in the 1980s, contributing to a sharp reduction in the price of uranium. These factors taken together meant that there was no longer any technical or economical justification for reprocessing of the spent nuclear fuel. Furthermore, political opposition to the reprocessing and reuse of plutonium grew ever stronger, the argument being that plutonium might be diverted to military purposes. The Swedish nuclear power industry therefore decided at an early stage that applications for fuelling permits for the two last reactors in the programme approved by the Riksdag would be based on final disposal of the fuel without reprocessing. The KBS-3 report was prepared and presented in 1983, and with this as a basis the fuelling permits for Forsmark 3 and Oskarshamn 3 were granted in June 1984.

After these fuelling permits had been obtained, other reactors that had obtained fuelling permission under the Stipulation Act also switched to having their permits based on the KBS-3 report /3-6/. This made it possible for the nuclear power utilities and SKB to transfer the right to reprocessing under the contracts with Cogema to other Cogema customers and thereby phase out any further reprocessing of fuel from the Swedish nuclear power plants. As a result, the work was concentrated on direct final disposal without prior reprocessing. A second method choice had thereby been made and accepted by industry, the regulatory authorities and the Government. This has also been noted in previously submitted R&D programmes accepted by the Government.

3.1.3 Alternatives for deep geological disposal

Deep disposal of spent nuclear fuel in Swedish crystalline rock can be designed and executed in a number of different ways. The Act on Nuclear Activities /3-4/ requires that the research programme that is carried out be comprehensive in the sense that different alternatives be studied. Based on the fundamental principles for deep disposal (see further in chapter 4), many different options are available. A number of such options have been studied during the 1980s and 1990s, some very thoroughly. Among the options studied are WP-Cave, Very Deep Holes, Very Long Holes and Medium-Long Holes /3-7, 3-8, 3-9, 3-10/. RD&D-Programme 92 /3-11/ provided a summary account of these studies and drew the conclusion that "the design according to KBS-3 is retained as the main alternative for the continued work. 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". The basic design has already been described in section 2.2. The strategy has been accepted in all essential respects by the regulatory authorities and the Government.

The principles and conceptual solutions described in the KBS reports have won great respect and recognition internationally. Compared with other alternatives, the KBS-3 design has proved to be very good for the type of bedrock that exists in Sweden and many other countries. Responsible organizations in countries with similar bedrock, e.g. Finland and Canada, have settled on concepts that are very similar to KBS-3.

3.2 ZERO ALTERNATIVE

At present all spent nuclear fuel is stored in CLAB. This storage is planned to last 30–40 years before the fuel is deposited in a deep repository. Experience from long-term storage of zircaloy-clad fuel dates back to the end of the 1950s. This experience shows that storage is possible for at least 50 years and can probably be prolonged to at least 100 years if necessary /3-12, 20/. In other words, as far as storage time and storage capacity are concerned it is possible to continue storing the spent fuel in CLAB for several more decades. This then becomes a natural "zero alternative" in the future EIA work for the deep repository and the encapsulation plant. Supplementary safety assessments (time horizon hundreds of years) are planned for this alternative.

Interim storage of spent nuclear fuel can also take the form of "dry storage" nowadays. Technology for this has been developed and tested in Germany and the United States, among other places /3-13,14/. Dry storage requires less maintenance and supervision than wet storage and can therefore be considered as a secondary "zero alternative" to deep disposal in the unlikely event that a decision is not taken to establish a deep repository within, say, 50 years. Supplementary safety assessment may have to be done for this case as well.

3.3 OVERVIEW OF MEASURES FOR STEP-BY-STEP CONSTRUCTION

A diagram of the major steps in the construction of an encapsulation plant and a deep repository and in the deposition of encapsulated fuel in the deep repository was presented in the supplement to RD&D-Programme 92 /3-15/. This diagram is reproduced in slightly modified and simplified form in Figure 3-1.

The measures that concern the deep repository encompass a time span of about sixty years or longer from the start of feasibility studies up to completed closure of a fully constructed repository. The following stages are distinguished for the **deep repository:**

Siting, which aims at gathering the material needed for decisions on siting (Act Concerning the Management of Natural Resources) and construction (Act on Nuclear Activities) of the repository. The work proceeds in two stages, where the first stage encompasses feasibility studies and general studies and the second stage site investigations. The latter are primarily carried out on two sites, which are chosen on the basis of the feasibility and general studies.

Detailed characterization of one site. This requires a permit under the Act Concerning the Management of Natural Resources (NRL) /3-16/. The work entails some rock construction works, which means in reality that **Construction stage 1** is also begun, which pertains to a facility for deposition of about 400 canisters of spent nuclear fuel from the Swedish programme. The Government's decision on the supplement to RD&D-Programme 92 thus states and clarifies that detailed characterization entails commencement of construction of a nuclear facility, which also requires a permit under the Act on Nuclear Activities

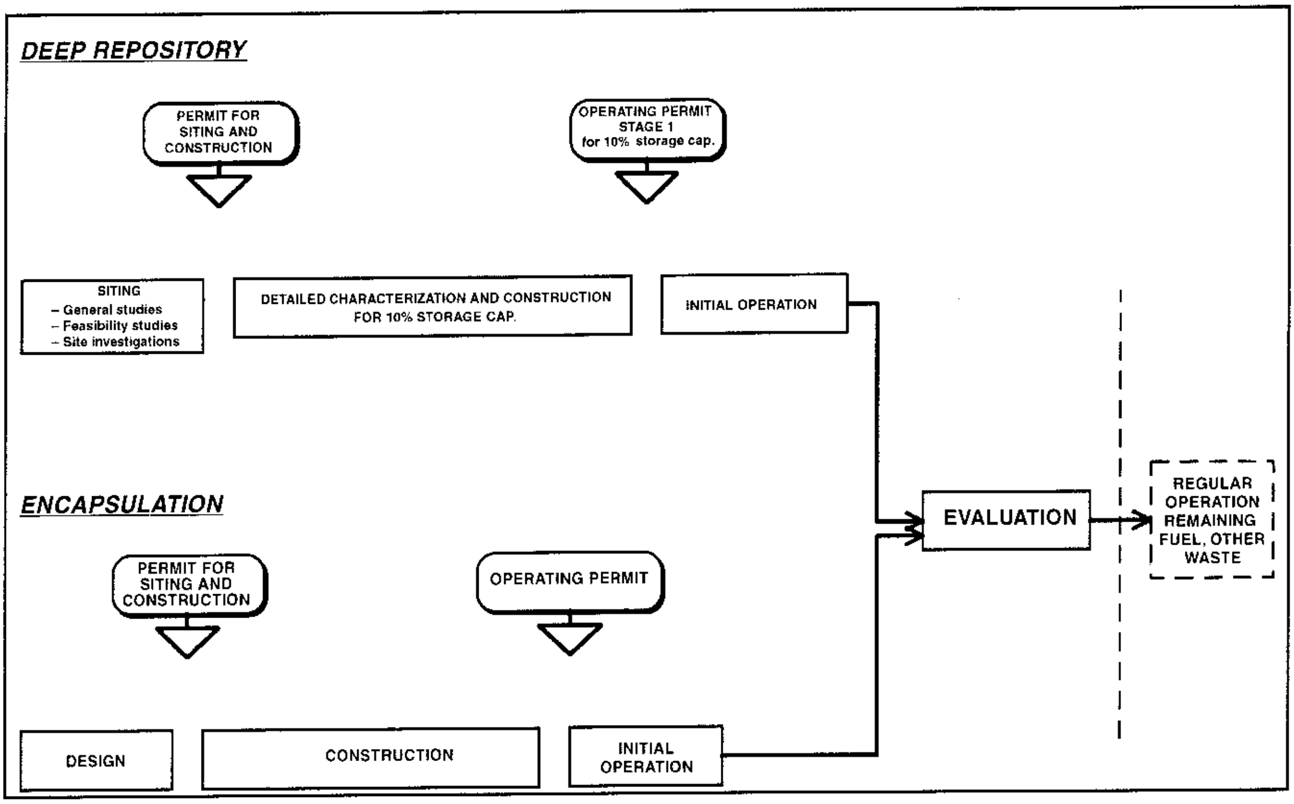


Figure 3-1. Diagram of the major steps in the programme, along with the requisite permits. The upper half of the figure concerns siting, construction and operation of a deep repository, while the lower half concerns the equivalent steps for an encapsulation plant. For each facility, the diagram shows main activities and important permits.

(KTL) /3-21/. The surface facility, common chambers under ground and the first deposition chambers under ground are built during this stage. At the same time, the requisite equipment for deposition of encapsulated fuel and for associated activities is manufactured, delivered and tested.

Initial operation, which comprises deposition of encapsulated spent nuclear fuel, about 10% of the quantity produced by the Swedish nuclear power plants up to 2010, i.e. about 800 tonnes (uranium weight) or about 400 canisters. The deposition chambers are excavated gradually at the necessary pace.

Evaluation. If this turns out favourably for continued deposition of the remaining quantity of spent fuel and other long-lived waste, application is made for the necessary permits for further construction and regular operation.

Construction and regular operation, which comprises the deposition chambers for all remaining spent nuclear fuel and all other long-lived waste. The work includes rock works, construction works and delivery and trial operation of equipment for the parts of the deep repository not included in stage 1. To the extent warranted by experience from initial operation and evaluation, the design of the facility and equipment is modified. The remaining waste is deposited during regular operation. This stage is the longest stage of the entire chain.

Supervised storage in the deep repository for as long as desired. **Closure** of the deep repository. This activity is not indicated in the diagram.

The following stages are distinguished for the **encapsulation plant**:

Siting and Design of the plant, including a final decision on canister design and execution of the necessary development work. The work is carried out in several stages with a gradually increasing degree of detailing up to applications for permits under NRL and KTL, which are planned to be submitted simultaneously.

Construction stage 1, which comprises detailed design and construction of the plant for encapsulation of spent nuclear fuel plus inactive trial operation of the same.

Initial operation, which comprises active trial operation plus encapsulation of about 10% of the spent nuclear fuel, about 400 canisters.

Evaluation. Coincides with evaluation of the initial stage of deep disposal as described above.

Construction and regular operation, which includes construction of a section for encapsulation of other long-lived waste (mainly core components) and encapsulation of remaining spent nuclear fuel and other long-lived waste.

Possible **decommissioning** of the plant and deposition of arising decommissioning waste – not indicated in the diagram.

3.4 ENVIRONMENTAL IMPACT ASSESSMENT AND EIA PROCESS

Permits under the Act Concerning the Management of Natural Resources (NRL) and the Act on Nuclear Facilities (KTL) are required for the siting and construction of an encapsulation plant or a deep repository. These permits are issued by the Government. In addition, approval is required in different stages under various statutes such as the Environment Protection Act, the Radiation Protection Act and possibly the Water Act, as well as a building permit. Under all the above laws, it is nowadays either necessary for a permit application to include an environmental impact assessment (NRL, Environment Protection Act, Water Act), or possible for a competent authority to prescribe that an environmental impact assessment must be performed (KTL, Radiation Protection Act). The authorities may also decide what the EIA must contain, but it is above all in the process by means of which impact assessments take shape that a meaningful EIA develops.

The purpose of an environmental impact assessment is to permit a coherent assessment of the impact of the facility on the environment and on human health and safety, as well as on the management of the country's natural resources. The fundamental scope and contents of an environmental impact assessment are illustrated in Figure 3-2. The EIA is thus a tool that enables an activity to be evaluated from the environmental viewpoint before a permit for the activity is issued, in other words the EIA is a part of the supporting material on which the licensing authority bases its decision on whether to issue a permit and the terms of the permit. Further, the EIA is supposed to enable concerned parties and the public to gain insight into the siting process and give them an opportunity to offer viewpoints. A description of how SKB plans the EIA process was given in a KASAM seminar /3-22/.

Formally, it would be possible for different requirements on EIAs to be stipulated in different laws. However, this would be contrary to the intentions behind the EIA legislation (basis for a coherent assessment). Instead, it is an important function of the EIA process itself

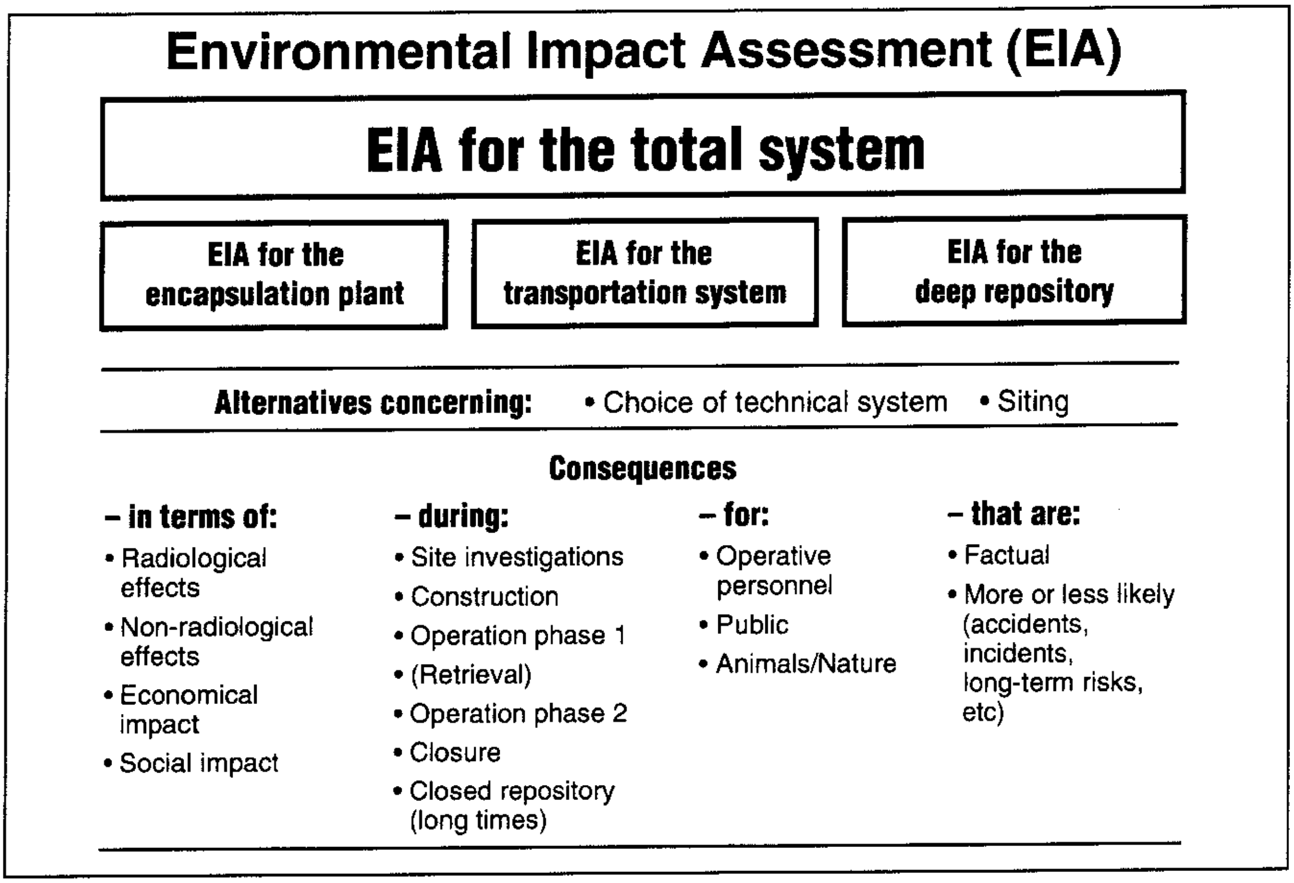


Figure 3-2. *Illustration of structure of environmental impact assessment.*

to create a coherent basis for a decision. According to the ordinance on environmental impact assessments, an EIA shall contain a reasoned account of alternative sitings and designs, plus data on the consequences of failing to adopt the applied-for measures (the "zero alternative").

The legislation does not lay down any further guidelines for how the EIA process is to be carried out. This means that the concerned parties ("the actors") enjoy wide freedom to jointly design the process themselves. SKB has therefore not issued its own detailed description of how the EIA process ought to be carried out. Instead, SKB has, at an early stage, entered into a local dialogue with concerned authorities at the start of the planning and investigation work for the encapsulation plant and the deep repository.

Traditionally, and ever since the introduction of nuclear power, the nuclear power industry has evaluated the safety and environmental impact of its plants and published these evaluations in special reports. In conjunction with the construction of nuclear power plants or other nuclear facilities, for example SKB's final repository for radioactive operational waste (SFR), they have prepared preliminary and final safety reports (PSR and SFR, respectively). These closely resemble the equivalent reports that are prepared for similar facilities abroad. Non-radiological environmental aspects have also been dealt with in the supporting material for decisions on siting and construction. Contacts have been made at an early stage with concerned authorities, municipalities and nearby residents. This has led to valuable experience which will be of use in the siting of the deep repository.

However, the future EIA process will entail a broader and deeper dialogue between all stakeholders. Important actors will above all be those who may have the facility in their midst (the municipality and nearby residents), the enterprise who will build and operate the facility (SKB), as well as supervisory authorities and the county administrative board. It is gratifying that the Government has now agreed to give concerned municipalities resources so that they can participate in a qualified and credible manner in the EIA process.

The formal EIA work has been commenced for the encapsulation plant. A joint consultation group for EIA matters has been formed under the leadership of the county administrative board in Kalmar County. Other participants are Oskarshamn Municipality, SKI (the Swedish Nuclear Power Inspectorate), SSI (the National Radiation Protection Institute) and SKB. A review of alternatives and a general assessment of various environmental impacts has been carried out as an initial step. In parallel, Oskarshamn Municipality has initiated a comprehensive information and joint consultation programme.

Feasibility studies are currently being conducted for the deep repository in several municipalities. Even though these studies are not conducted within a formally defined EIA process, the work nevertheless contains many of the elements that are important in such a process. They are, for example, conducted with openness and great insight and participation from many different interested parties. The set of issues that are examined is of great breadth, and the various stakeholders have a say in which issues are to be taken up.

If the siting studies in a municipality proceed after the feasibility study with site investigations, the work of preparing the environmental impact assessment which the site investigations are supposed to lead to must be begun early. SKB therefore maintains that the forms for the EIA process should be agreed upon among the concerned parties. Since a designated site exists, nearby residents, landowners and competing land-use interests can be identified and allowed insight and influence in the process. Experience from the feasibility study will be of great value to the relevant municipalities. Both the municipal leaders and various interest groups have acquired considerable knowledge in the matter through the feasibility study and are therefore able to safeguard their interests and contribute constructively to a stable and credible process.

The exact design of the EIA process must be agreed upon through discussion between the municipality, SKB and the regulatory authorities. The model and the experience that exist from the EIA work for the encapsulation plant can serve as an important background to the discussion of how the EIA process for the deep repository should best be designed. The Government has also stated that concerned county administrative boards should assume coordinating responsibility. The transportation question should also be explored in conjunction with the EIA work for the deep repository. The shipments may also involve other municipalities than those included in the studies of the encapsulation plant and the deep repository. These "transit municipalities" must be given an opportunity to participate in the EIA work on an equal footing as far as the environmental impact assessment of the shipments is concerned.

An important question in the EIA work both for the encapsulation plant and for the deep repository is the relationship between the two. Thus, for example, the prospects for co-siting must be penetrated in the EIA. Then when a decision is eventually made on the siting of the two facilities, the assessments for one of the facilities will logically be influenced by how far the work has come with the other. A decision will not be made to build one of the facilities unless the work with the other has progressed far enough.

4 DEEP GEOLOGICAL DISPOSAL – PRINCIPLES AND REQUIREMENTS

This chapter discusses the knowledge that is required and certain fundamental principles and requirements for deep geological disposal of spent nuclear fuel in crystalline rock. The results that have been achieved and the state of knowledge on which this programme is based are described more specifically in chapters 5, 6 and 8.

4.1 WHAT KNOWLEDGE IS REQUIRED?

The Swedish nuclear fuel must be finally disposed of in a safe manner. Fulfilling this goal in practice requires knowledge and expertise within a number of fields. It is necessary to know how to

- Handle, condition, transport and dispose of the waste.
- Site and build the necessary facilities.
- Analyze and describe safety aspects and environmental consequences.

For central issues, such as safety assessment, a fundamental understanding of physical and chemical processes is vital. This knowledge must lie at a high international level, even in important details. In large parts, the work today can build on established knowledge and proven experience. In certain respects, new technology and new knowledge must be developed.

4.1.1 Handling, conditioning, transport and disposal of the waste

Handling and transport of nuclear waste and spent nuclear fuel are based on known technology. Practical experience has existed for several decades in both Sweden and other countries. Encapsulation of spent nuclear fuel in the manner planned by SKB has not, however, been carried out in practice. Design, fabrication and sealing of canisters is therefore a key area in the programme. The comprehensive development work in this area is described in greater detail in chapters 6 and 7.

For natural reasons, transport of encapsulated spent nuclear fuel has not been carried out in practice. However, encapsulated fuel is better protected and has a much lower radiation level and lower heat output than the spent fuel that is transported today between the nuclear power plants and CLAB. Nuclear fuel has been transported by road and/or rail for many years in a number of countries. Taken together, this means that the method for transport to the deep repository of the canisters with fuel can be based in all essential respects on established knowledge. What is needed is adaptation of the technical design. See further chapters 8 and 9.

Handling of the canisters at the deep repository includes a number of operations:

- Reception and transport of cask with canisters.
- Transfer of canisters to handling machine in deposition tunnel.
- Emplacement of canisters at deposition positions.

Of these operations, handling of canisters in deposition tunnels and emplacement of them in deposition positions is the new and important one. Qualified design and engineering work needs to be carried out in order to develop appropriate handling equipment. Other handling of waste in the deep repository can be done by adaptation of already established technology. Chapters 8 and 9 provide a more detailed account of the state of knowledge and programme for deep repository technology.

Buffer and backfill materials are also needed in the deep repository. It must be possible to

- Manufacture and apply the bentonite buffer in the deposition holes.
- Backfill the deposition tunnels.
- Seal the entire repository.

Development of the methods to be used for these purposes comprises an important part of the nuclear waste programme. The work has been in progress for a long time. Experiments and tests were conducted within the Stripa Project during the 1980s and further work will be carried out in the Äspö Hard Rock Laboratory. Results, state of knowledge and programme are described in greater detail in chapters 5 and 8–12.

4.1.2 Siting and construction of the necessary facilities

Siting, particularly of a deep repository, is a multifaceted and controversial activity. It involves both technology and safety aspects on the one hand and community planning, politics and public-opinion aspects on the other. The concrete experience and knowledge that exists relates to previous establishments of controversial facilities of a similar nature in Sweden and abroad. SKB and the owners of the nuclear power plants have long experience from different types of siting, including the establishment of CLAB, SFR and the Äspö Hard Rock Laboratory. However, no such previous experience is fully applicable to the deep repository.

Following the publication of RD&D-Programme 92 /4-1/, a concrete siting process was initiated. Background material (general studies) has been compiled and siting investigations (feasibility studies) have been carried out /4-2/ or commenced. The siting process and siting criteria have, in accordance with the requirements imposed by the Government /4-3/, been described more thoroughly in the supplement to RD&D-Programme 92 /4-4/.

The feasibility studies have led to extensive discussions, especially in the concerned municipalities, but also regionally and on the national plane. SKB as well as other concerned parties (authorities, policymakers, the general public) have gained important and valuable experience for the continued work. Difficulties and possibilities in the siting work have been clarified. The work with feasibility studies has taken longer than SKB planned in RD&D-Programme 92, in part because the issue arouses a great deal of discussion right from the start.

The development work has gradually led to the creation of a solid platform for continuing to gather the information that is needed as a basis for siting of the deep repository. The process and the criteria have been clarified by SKB and accepted by the regulatory authorities and the Government. In its decision /4-5/, the Government has also clarified the licensing procedure. Furthermore, the Government has decided to allocate funds to concerned municipalities for participation in the process. The concrete siting work has got under way in the form of feasibility studies at various locations in the country.

The siting of the deep repository is a key issue in the nuclear waste programme. It should be possible to arrive at a good solution through continued thorough, persistent and open-minded siting work. Simultaneously with this RD&D-Programme, SKB is publishing a consolidated account of the general studies /4-6/. State of knowledge and plans concerning siting are described in chapters 8 and 9. To house the encapsulation plant, SKB is planning an addition to CLAB, since this offers several obvious advantages. During the ongoing EIA work, alternative sitings will also be analyzed, such as adjacent to the deep repository, see chapter 7.

Extensive knowledge and experience can be drawn on when it comes to constructing the necessary facilities (encapsulation plant and deep repository). Experience from the construction of nuclear power plants and CLAB can be utilized in designing and building the encapsulation plant and the surface facilities at the deep repository. Furthermore, substantial know-how exists in Sweden from the construction of underground rock facilities for mining, power production, oil storage and defence. SKB has direct experience from the rock facilities at CLAB, SFR and the Äspö HRL. All of this provides a broad and solid knowledge base. Special development initiatives are needed on some important points, mainly methods and technology for geoscientific site investigations and detailed characterization of a selected site. This has been included as a large and important part of SKB's programme for a long time. A "dress rehearsal" of the methodology can be said to have been held through the works for the Äspö HRL. These entail testing and evaluation on a real site and under realistic conditions before the methods are applied in the deep repository project. An account of the state of knowledge regarding rock investigations and construction of the facilities is given in chapters 6 and 8. The programme for the Äspö HRL is described in chapter 12.

4.1.3 Safety assessments and environmental impact assessments

In order to be allowed to execute the planned final disposal scheme, SKB must thoroughly analyze and account for safety aspects and environmental consequences. Traditionally, and ever since the introduction of nuclear power, the nuclear power industry has evaluated the safety and environmental impact of its plants. Previously, in conjunction with the construction of CLAB and SFR, SKB has prepared preliminary and final safety reports. Non-radiological environmental aspects have also been dealt with in the supporting material for decisions on siting and construction of these facilities. A large and valuable body of experience therefore exists which can be drawn on in the work with safety assessments and environmental impact assessments. In the preparation of the environmental impact assessments, extensive consultation is supposed to take place between applicants, local stakeholders and central authorities. See also section 3.4.

To develop methods and gather scientific data as a basis for an assessment of the long-term safety of a deep repository, SKB has long been conducting a comprehensive R&D programme. Important areas in this programme are:

- characterization of spent fuel and studies of the durability of the fuel in groundwater,
- durability of canisters in a deep repository,
- performance of bentonite buffer,
- chemical composition of the groundwater and groundwater movements in the rock,
- chemical conditions and reactions in a deep repository,
- stability of the rock,
- radionuclide transport in the deep repository, bedrock and biosphere,
- dose calculation.

The principles for radiation protection and safety are described in section 4.2 below. The repository's safety functions and functional requirements are described summarily in 4.3.

The state of knowledge within the parts of the programme that pertain to the background data for assessment of long-term safety is described in chapter 5. A special report (SR 95 /4-11/) describes a template for how future safety reports will be structured and the methods and calculation tools that SKB has at its disposal today for carrying out assessments of long-term safety.

A programme for continued R&D within this area is presented in chapters 10, 11 and 12.

4.2 PRINCIPLES FOR RADIATION PROTECTION AND SAFETY

4.2.1 General

Radioactive waste must be handled in keeping with established principles for protection against ionizing radiation /4-7, 8/.

- The activity must be justified, protection must be optimized and the individual must be protected by dose limits.
- The radiation protection considers human health and nature with regard to conditions for biological diversity and utilization of natural resources.
- The radiation protection must be independent of whether the doses arise today or in the future, or if they are emitted inside or outside national boundaries.
- The radiation protection in management and long-term disposal of radioactive waste must be equivalent to that in other radiological activities, e.g. other portions of the nuclear fuel cycle.

In view of the long period of time which must be taken into account when planning a deep repository, specific guidelines have been proposed for a final repository /4-9/:

- The repository shall not be dependent for its long-term safety on monitoring and maintenance by future generations. This is not to say, however, that the repository cannot be monitored for a period after disposal of the waste or closure of the repository.
- The repository shall not be designed so that it unnecessarily impairs future attempts to change the repository or to retrieve the waste.
- The long-term safety of the repository shall be based on passive multiple barriers so that the degradation of one barrier does not substantially impair the overall performance of the disposal system.
- During a reasonably predictable period of time, the radiation doses to individuals caused by expected releases shall be lower than 0.1 mSv/y, after which the radionuclide flow from the repository shall be limited to a level corresponding to naturally occurring flows.
- Probabilities and consequence of unexpected extreme events shall be judged in comparison with the risk of injury in the critical group at the above individual dose limit.

The safety of the repository is dependent on the toxicity and accessibility of the waste. The assessment of the repository's safety is influenced by time in that the quantity of toxic radionuclides declines, and in that the uncertainty in the quantification of the repository's safety functions increases with time. The concept "reasonably predictable period of time" refers primarily to the time-dependent uncertainty in the performance of the repository. Potential transport pathways for radionuclides to man can change with time, but such changes will take place at different rates for the different parts in the barrier system. Experience shows that essential changes in the biosphere occur on the time scale 100–1000 years. The geological environment deep down in the Fennoscandian Shield, however, exhibits stable conditions on a time horizon of millions of years.

Consequently, the feasibility of quantifying the safety of the repository (or the risk from the repository) is dependent on the period of time in which one is interested. SSI has discussed the influence of the time horizon on radiation protection /4-10/ and finds that:

- Particularly great attention should be given to describing protection for the period up to closure of the repository and the first thousand years thereafter, with a special focus on nearby residents.
- The individual dose up to the next ice age, i.e. up to about 10,000 years, should be reported as a best estimate with an estimated margin of error. Environmental protection should be described for the same period of time.
- For the period from the next ice age onward, qualitative assessments should be made of what might happen with the repository, including deliberations regarding the risk of increased releases.

SKB intends to utilize these guidelines in forthcoming accounts of radiation protection and safety for different scenarios for a deep repository on different time scales.

4.2.2 Safety functions of the repository

To achieve the desired safety during the construction of a deep repository, during the operating phase and during

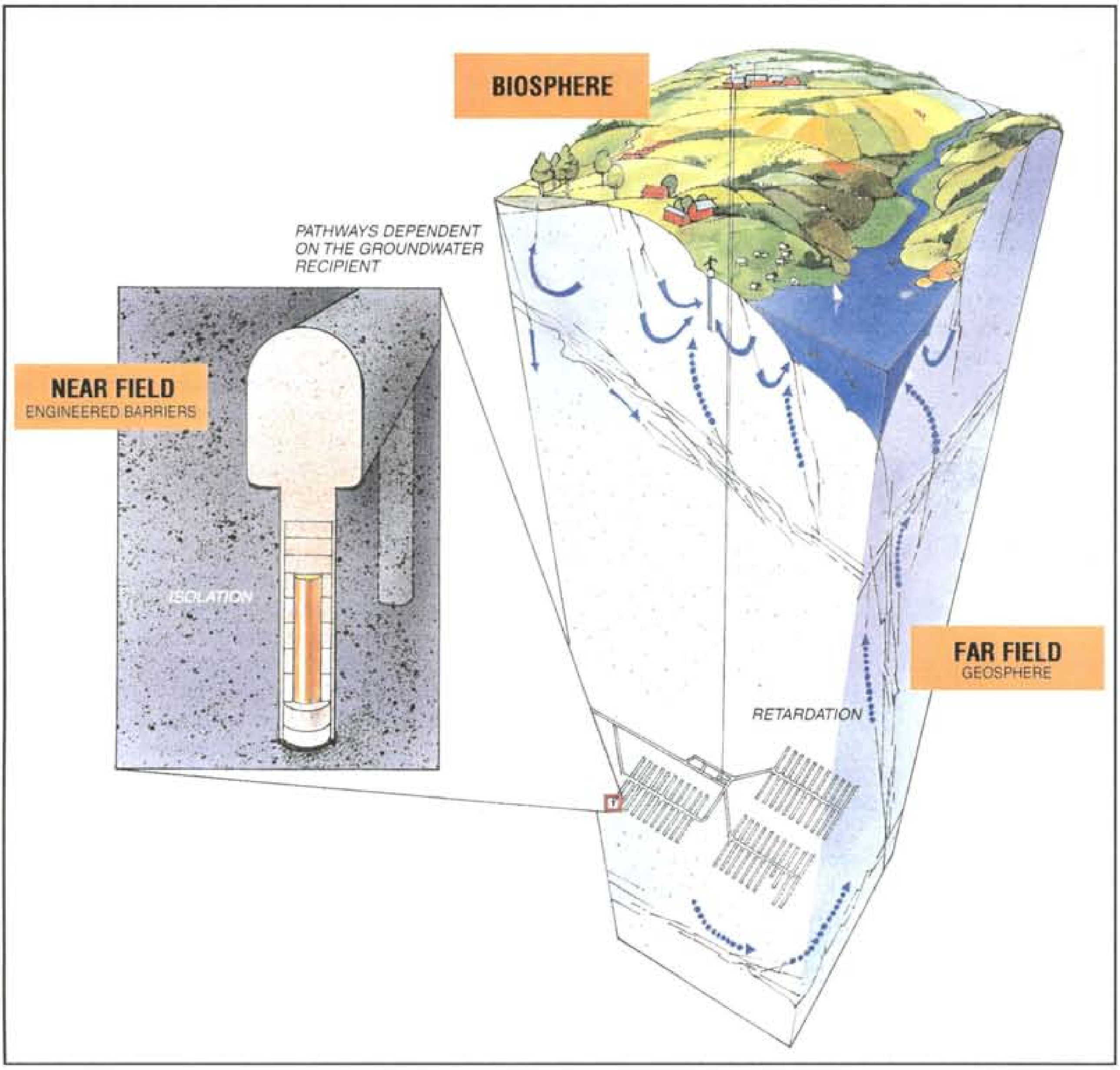


Figure 4-1. *Parts of the deep disposal system and their most important safety functions.*

the long-term containment phase, requirements are made on the function of the repository and its components. The integrated function of all the repository's components must together provide adequate safety in the activities.

In order to achieve long-term safety, the disposal system is designed to **isolate** the spent nuclear fuel from the biosphere. This isolation is achieved by encapsulating the spent nuclear fuel in impervious canisters which are deposited deep in the crystalline bedrock on a selected repository site. In addition, the repository has the function of **retaining** the radionuclides and **retarding** their transport if this isolation should be broken. Furthermore, by proper site selection and suitable adaptation of the repository to the actual site, transport pathways and dilution conditions in the biosphere can be influenced so that any radionuclides that escape will only reach man in very small quantities. Figure 4-1 shows the different parts of the disposal system and their primary safety functions.

The materials used in the repository have been selected with a view to the possibility of verifying their long-term stability and safety performance in the repository with experience from nature. For the same reason, the thermal and chemical disturbance which the repository is allowed to cause in its surroundings is limited. The safety philosophy for the deep repository is based on the multi-barrier principle, i.e. safety must not be solely dependent on the satisfactory performance of a single barrier. The safety functions are affected by site selection, layout, and by the design and sizing of the engineered barriers. The functions can be divided up into three levels:

Level 1 – Isolation

As long as the waste is isolated, the radionuclides can decay without coming into contact with man and his environment.

Level 2 – Retardation

If the isolation is broken, the quantity of radionuclides that can reach the biosphere is limited by:

- very slow dissolution of the spent fuel,
- sorption and very slow transport of radionuclides in the near field,
- sorption and slow transport of radionuclides in the bedrock.

Level 3 – Recipient conditions

The transport pathways along which any released radionuclides can reach man are controlled to a great extent by the conditions where the deep groundwater first reaches the biosphere (dilution, water use, land use and other exploitation of natural resources). A favourable recipient means that the radiation dose to man and the environment is limited. The recipient and the transport pathways are, however, influenced by natural changes in the biosphere.

The safety functions at levels 1 and 2 are the most important and next-most important. They are achieved by means of requirements on the properties and performance of both engineered and natural barriers and on the design of the deep repository. Within the frames otherwise defined, good safety function at level 3 is also striven for by a suitable placement and configuration of the deep repository.

4.2.3 Safety functions of the barriers

The guidelines for final disposal of spent nuclear fuel discussed in Sweden /4-9, 10/ define safety goals for the deep repository as a whole. This provides an opportunity to design the function of different barriers, within the framework of the multibarrier principle, so that the necessary safety can be achieved in a reliable and effective manner. The functional requirements are also affected by the goal that radioactive waste management should be conducted in a balanced manner with respect to operational and long-term safety.

The conditions in different barriers that contribute towards defining the safety functions are described below.

Isolation

At the depths being considered for a deep repository according to KBS-3, mass transport normally takes place only with groundwater. Special transport pathways, caused by e.g. human intrusion, must be examined separately. Isolation is achieved by encapsulating the spent fuel in leaktight canisters. To give the canisters a stable and favourable environment, they are emplaced in deposition positions in a tunnel system at a depth of about 400–700 m in the rock and surrounded with a plastic clay material with low permeability to groundwater.

The ability of the canister to isolate the spent fuel is influenced in practice by

- the design and quality of the engineered barriers,
- site-specific conditions,
- repository depth and the design of the repository and its near-field.

To ensure adequate isolation of sufficient duration, acceptance levels or requirements must be stipulated for, above all

- the canister materials and dimensions as well as inspection methods to verify canister quality,
- the chemical and physical properties of the buffer material,
- the methods for conditioning and application of the buffer and inspection methods to verify its application,
- the geometry of the deposition holes and the hole-making methods,
- the mechanical, chemical and hydraulic conditions in the immediate surroundings of the deposition hole,
- the geohydrochemical conditions in the repository area,
- the stability of the host rock,
- the radiation and temperature levels in the repository.

The isolation can be broken by an undetected defect from fabrication, by internal or external mechanical stresses, or by internal or external corrosion. The goal is that the isolation should be able to resist the cumulative effects of corrosion, buffer swelling and hydrostatic head during the approximately 100,000 years required until the potential hazardousness of the spent fuel has reached the level of natural uranium. The repository should also be able to retain its isolation in the face of rock movements in the repository caused by anticipated stress redistributions. The number of possible canister failures resulting from changes and loads expected to occur in connection with a future ice age should be limited.

Retardation

If the isolation should be broken due to the fact that a canister is damaged, fuel will come into contact with water, whereby radionuclides can be dissolved. Radionuclides that have been released from the fuel matrix can be transported through the defective canister via the buffer material to the mobile groundwater in the rock, and from there further through the bedrock to recipients in the biosphere. The mass flow is determined by the transport and retardation processes that act in the near field and surrounding bedrock.

The dissolution process is affected by

- the spent fuel and its properties, i.e.
  - the inventory and properties of the radionuclides
  - the properties of the fuel matrix,
  - the temperature and radiation field in and around the canister,
- the engineered barriers, i.e.
  - the nature of the canister damage,
  - the canister material and the buffer material, and
  - any building material in the near field
- site-specific conditions
  - groundwater chemistry such as redox conditions and salinity.

Transport mechanisms and retardation mechanisms in the near field and adjacent host rock can be affected by

- the design of and properties of the engineered barriers,
- the design of deposition holes and tunnels,
- the impact on nearby rock of the methods of rock extraction,
- the topography of the repository site, its geological structures and their hydraulic connection to the engineered barriers in the near field,
- the sorption of the nuclides on available solid surfaces.

Within the framework of the design that is required to create and maintain good isolation between the fuel and the groundwater, the conditions in the repository's natural and engineered barriers should be chosen so that the solubility of the radionuclides is limited and their transport retarded.

The solubility limitation assumes a definition of acceptable ranges as regards the chemical composition of the groundwater and a limitation of the influence of materials and impurities in the near field. The large-scale transport of radionuclides through the rock is controlled by the groundwater movements in the repository area, available surfaces for sorption and matrix diffusion along the transport pathways, and the geohydrochemical conditions in the area.

The requirements on the engineered barriers are formulated as limitations of chemical parameters (e.g. redox potential, stability of the bentonite) and transport parameters (e.g. hydraulic conductivity of the buffer, temperatures). The requirements on the repository's host bedrock are formulated as safety-related siting factors and met by adaptation of the repository to local geological structures.

Recipient conditions

If the isolation should be broken, radionuclides that do not decay and are not fixed in the rock will reach the biosphere. A favourable recipient will limit the potential radiation dose to man from the radionuclides that reach the biosphere.

The transport pathways along which radionuclides can reach man are mainly dependent on dilution, water use and land use at the points where the deep groundwater from the repository first reaches the biosphere. Thus, potential radiation doses can be further limited by means of a suitable choice and utilization of the repository site.

However, many changes of importance for the transport pathways in the biosphere are highly complex and take place over a much shorter time span than corresponding changes in the geosphere. A safety function that is based on favourable conditions in the recipient is therefore not as dependable and long-lasting as safety functions based on the bedrock and/or the engineered barriers.

4.3 BARRIER FUNCTIONS – REQUIREMENTS

4.3.1 General

A deep repository for radioactive waste should be designed with a view towards safety, constructability and effectiveness with respect to costs and resources. The safety of the repository can be influenced by choice of deposition method, site choice and adaptation of the repository's design and layout to the properties of the site, by choice of technology and inspection methods for construction of the repository, and by choice of materials, design and sizing of the engineered barriers around the radioactive waste. Other factors that also influence safety are the properties of the radioactive waste and radionuclides contained in it. The availability of suitable rock in Sweden and the properties of the water constitute premises for how the repository is to be designed.

The safety functions of the repository and how they are affected by various processes and conditions in the barriers are discussed in section 4.2. This section summarizes the requirements that are made, or may come to be made, on site/layout and on the engineered barriers, as well as on the methods for their fabrication/application and their inspection. One point of departure for the formulation and quantification of requirements is previous performance and safety assessments. The state of knowledge for understanding and quantifying the processes on which the requirements are based is presented in chapter 5, sections 5.3–5.8. The state of knowledge concerning processes essential to the function and fabrication of the canister has been compiled in chapter 6.

The requirements on the barriers must be quantified once design and fabrication methods have been established. This work is under way for the canister, and the status of the work is described in chapter 6. The times at which material specifications, designs or dimensions are established must be coordinated with the planning of fabrication and handling of the engineered barriers. The quantitative requirements presented in RD&D-Programme 95, especially in chapters 6 and 8, are to be regarded as tentative and may be modified during the course of the continued design process.

The coupling between the characterization of the site and progressive decisions regarding how the site is to be utilized and how the layout of the repository is to be made with regard to safety and constructability will be discussed in a planned special report on the programme for site investigations.

Logically, it should be possible to derive the functional requirements on different barriers directly from the quantitative safety goals for the repository (expressed e.g. in dose limits). However, in the case of repositories with multiple barriers, such a derivation does not give unequivocal requirements on the individual barriers. The requirements on the individual barriers will therefore be formulated primarily on the basis of:

- parameter limitations to make it possible to rule out or neglect certain processes,
- available technology,
- cost and reliability optimization.

Considerations of how convincingly the function of different barriers can be proven and how sensitive different barriers are to variations in the surrounding environment will influence the strived for balance between the protective effects of the barriers.

4.3.2 Repository site and rock as barrier

The site should be selected so that it has good conditions with respect to:

Level 1 – isolation

- mechanical stability of the rock,
- chemical environment in groundwater/rock with respect to the canister and the buffer,
- presence and transport of substances corrosive to the canister,
- prevention of future intrusions and alternative uses,
- groundwater.

Level 2 – retardation

- limitation of radionuclide solubility and transport to the biosphere
- chemical environment in groundwater/rock with respect to fuel dissolution and buffer,
- groundwater.

Level 3 – recipient conditions

- groundwater dilution and food chains.

4.3.3 Canister

The canister must be designed and fabricated so that it

Level 1 – isolation

- is leaktight at deposition,
- can resist the chemical action of
  - oxygen and other oxidants that are introduced during the repository's construction and operating period,
  - substances that can normally occur in reducing groundwaters,
- limits the effects resulting from
  - external and internal corrosion caused by radiolysis products,
  - internal corrosion caused by residual oxygen and water,
- can resist mechanical stresses caused by
  - hydrostatic head at repository depth,
  - the swelling pressure from the buffer material,
  - extra loads during an ice age,
  - rock movements caused by stress redistributions as a consequence of the repository's construction.

Level 2 – retardation

- does not unnecessarily and in a detrimental manner affect
  - the normal properties of the surrounding rock,

- the stability of surrounding buffer material,
- the rate of dissolution of fuel if the isolation is broken,
- transport of radionuclides through buffer and rock,

– as far as possible limits and retards the outward transport of radionuclides from fuel to buffer even if the isolation is broken.

4.3.4 Buffer

The buffer should give the canister a favourable environment for maintaining the isolation, comprise a protective layer between the canister and the rock with respect to mechanical and chemical forces, and, if the isolation of the canister is broken, limit and retard the escape of radionuclides from the repository.

For these purposes the buffer should:

Level 1 – isolation

- completely envelop the canister for a long period of time – "remain in the deposition cavity",
- bear the canister centred in the deposition hole,
- prevent groundwater flow and thereby retard the inward transport of corrodants,
- dissipate heat from the canister,
- resist chemical transformation for a long time,
- not jeopardize the abilities of the canister and the rock to fulfil their functional requirements,
- protect the canister by comprising a plastic protection against rock movements.

Level 2 – retardation

- prevent flow of groundwater and thereby retard transport of radionuclides,
- resist chemical alteration for a long time,
- completely envelop the canister for a long time – "remain in the deposition cavity",
- permit generated gas to escape,
- filter colloids.

4.3.5 Design of the repository and the near field

The favourable properties of the site for preventing and retarding the release of radionuclides to the biosphere should be utilized optimally by adaptation of the layout and depth of the repository to local conditions. Tunnels and deposition holes should be situated in the repository rock so that rock formations unfavourable for safety or construction are avoided.

The repository should be designed and construction carried out so that corrosion of the canister is limited with respect to microbial activity and oxygen and other oxidants introduced during the repository's construction and operating period.

The repository's geometric layout should be chosen in consideration of local rock stresses, temperature limitations and water flow paths. Construction and other works should be carried out so that the barrier properties of the host rock are not unnecessarily degraded.

Backfilling of tunnels and rock chambers should be done to give the rock back some mechanical support and to limit the volume increase of swelling bentonite in deposition positions in deposition tunnels. Plugging of tunnels and shafts should be done to limit the transport capacity of groundwater along the pathways opened up during excavation. Before the surveillance and control of the repository is discontinued, the repository must be sealed to prevent access. Materials used during the construction and deposition phases must be checked with regard to the consequences of their remaining in the repository after closure.

The design of the repository must also permit subsequent retrieval of the waste if this is deemed desirable in the future. At the same time, international requirements on physical protection of the fissile material – safeguards – must be met.

5 STATE OF KNOWLEDGE - LONG-TERM SAFETY

Based on the principles and basic requirements on the barrier system presented in chapter 4, this chapter describes the state of knowledge within the various areas that are of importance for the long-term safety of the deep repository. This overview of the state of knowledge has in some portions been made relatively comprehensive in recognition of the viewpoint expressed in the review of RD&D-Programme 92, that it was difficult to judge SKB's programme without having an idea of how SKB views the state of knowledge. This does not mean that the account is exhaustive in all respects. A more detailed and exhaustive account is given in various areas in SKB's technical reports, in the supporting documentation for safety assessments, in SKB's Annual Reports and in a large number of reports from e.g. Stripa, Äspö and the natural analogue projects.

The chapter begins with a survey of methods for safety assessment and for selection of scenarios that can affect the deep repository. This is followed by an account of the state of knowledge for the spent nuclear fuel, for the buffer and backfill materials and for the bedrock. The properties of the Swedish bedrock with respect to deep geological disposal are described in considerable detail. The intention is to provide a full description of the state of knowledge as a background to ongoing feasibility studies and forthcoming site investigations, as well as to experimental activities at the Äspö Hard Rock Laboratory. Specific chemistry questions, which pertain to several barriers, have been collected in their own section, as have the studies of natural analogues. Also, the state of knowledge for description of the biosphere is presented in a separate section. The concluding section deals with questions concerning, among other things, long-lived waste that is to be disposed of by deep geological disposal in a manner similar to the spent fuel. Canister-related questions are dealt with in chapter 6.

In each section, an attempt is made to define the relevant issues that require further R&D and that will be addressed in future RD&D work.

5.1 METHODS FOR SAFETY ASSESSMENT

5.1.1 General

On the basis of the experience that has been gained through research, experiments, studies of natural analogues etc., and on the basis of previously performed performance and safety assessments, preliminarily optimized disposal systems can be designed for final disposal of radioactive waste. To demonstrate the safety of a disposal system, the processes that are essential to the safety of the repository must be identified, and their change with time analyzed. This must be done in consideration of site-specific conditions, the chosen repository design and layout, and for the design, materials and dimensions chosen for the engineered barriers.

Regardless of whether the assessment covers the entire repository or just parts of it, or whether it is intended to shed light on a specific function or the total safety of the repository, it must be carried out systematically. A complete assessment includes:

- Definition of the purpose of the assessment.
- Definition of given assumptions for the assessment, i.e. types and quantities of radioactive waste, the disposal system and its dimensions, and the location and external environment of the repository.
- Definition of the scope and delimitations of the assessment, and of the safety goals.
- Clarification of both the probable and the less probable or improbable conditions for which the system/facility is to be assessed (scenarios).
- Clarification of the time-dependent processes which are essential for the intended performance of the system/facility in different scenarios.
- Definition of calculation models for quantifying the performance of the repository and the couplings between the models, where possible.
- Quantification of the performance of the repository and essential changes in performance.
- Qualitative assessment of important but non-quantifiable processes or events that can affect the performance of the repository.
- Discussion of the uncertainties in qualitative and quantitative sub-assessments and evaluation of their validity with respect to the purpose of the overall assessment.

The work procedure for performance and safety assessments is shown in Figure 5.1-1 and has previously been presented in RD&D-Programme 92 – Supplement.

In order to enable the assessments to be carried out in a systematic and traceable fashion, the methods for the different steps have been discussed within Sweden, in SKB's cooperation with the major nuclear power countries, and in the international cooperation. By "methods" is meant here both the general systematics that are applied in the work, and the tools – numerical models – that are used to quantify the safety-related processes in the repository. Most of the methods that are employed today have been developed over many years, mainly during the 1980s, and are continuing to be developed.

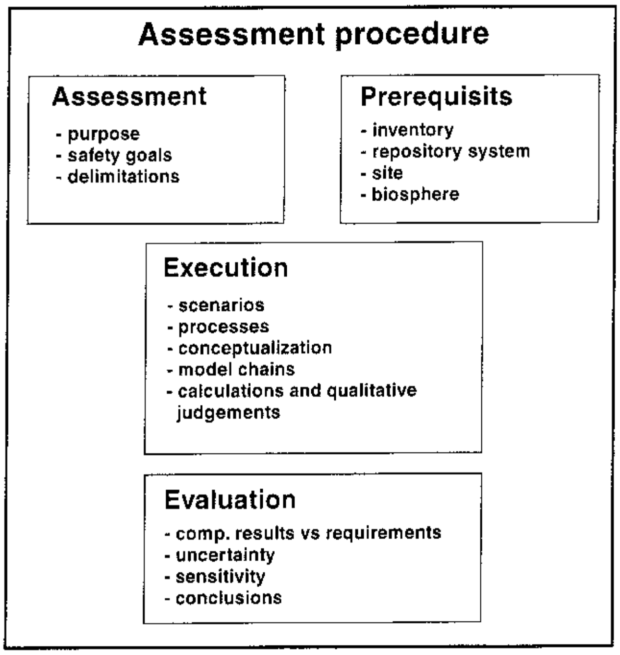


Figure 5.1-1. *Block diagram for how major performance assessments and safety assessments are carried out.*

A large survey of systematics and tools for safety assessments was carried out in October 1989 at a symposium arranged by the IAEA, the OECD/NEA and the CEC in Paris /5.1-1/. The findings of this symposium were evaluated and discussed in the arranging organizations' expert groups, and the results summarized in an International Collective Opinion /5.1-2/. There it was observed

- that safety assessment methods are available today to evaluate adequately the potential long-term radiological impacts of a carefully design 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.

It was also noted that the collection and evaluation of data from proposed disposal sites are major tasks on which further progress is needed, and that the methods for safety assessments can and will be further developed as a consequence of ongoing research work. SKB shares these viewpoints.

In response to the comments and opinions offered on RD&D-Programme 92 and the supplement that was published in 1994, SKB has, for RD&D-Programme 95, made a compilation of the methods which are available to SKB today for carrying out safety assessments. This compilation is available in a separate report, SR 95 /5.1-3/.

Since SKB plans to give accounts of safety on a large number of occasions, see chapter 10, it has been deemed desirable to establish a standardized format for how long-term safety is to be presented. Such a format would both facilitate a progressive updating of safety accounting in various phases and facilitate comparisons between assessments for different sites or under different work phases. For these reasons, SR 95 has been written in such a way that it can serve as a future template for how long-term safety is to be accounted for and comprise an account of currently available methods and numerical tools, including their applicability and quality.

SR 95 begins with a survey of the purposes of a safety report during different phases in the development of a deep repository and the methods available for carrying out the assessments, after which the currently prioritized repository design is presented. The application of SKB's scenario methodology to this design is then presented, along with the safety-related scenarios (calculation cases) with processes of chemical or physical interaction. A detailed examination is made of the tools, i.e. numerical models and conceptual assumptions, available to SKB today and their applicability and quality. The methodology for model couplings and the performance of the calculations are illustrated with material based on the prioritized design and on a site-specific geoscientific body of data from the Äspö HRL and surrounding biosphere.

It should be observed that neither the scenario survey nor the model calculations in SR 95 are complete. They constitute an illustrative method description. In particular the reported set of scenarios (calculation cases) comprise parts of the safety report that are in the process of being developed in preparation for the permit application for the encapsulation plant. As the process of designing the facilities for encapsulation and deep disposal progresses, premises, data and calculations will be revised.

Certain fundamental methodology questions have, however, been taken up for discussion in this RD&D-Programme, see further the sections

5.1.2 Conceptual and numerical models
5.1.3 Numerical coupling between models
5.1.4 Uncertainties and validity
5.2 Scenarios

All numerical models presented in SR 95 have been developed and continue to be developed on the basis of the evolving understanding of the processes included in the models. The state of knowledge within these areas and methods for collection and processing of data are described in the barrier- or subject-specific sections 5.3–5.9 and in chapter 6.

5.1.2 Conceptual and numerical models

Different calculation models comprise important tools in the work with performance and safety assessments. As the computation capacity increases, increasingly powerful tools are being developed for both deterministic and statistical analysis of conditions and processes of importance for safety and performance.

As is evident from the assessment procedure, the processes which can transport radionuclides from the repository to the biosphere are identified at an early stage. On the basis of previously conducted performance and safety assessments, a good understanding has been built up of the processes that are important. Extensive work has been done to clarify these processes and their parameters, and how they are to be conceptualized and quantified in numerical models.

Different views on conceptual models and their role in the safety assessments were presented and discussed in November 1993 in a workshop arranged by the OECD/ Nuclear Energy Agency (NEA) /5.1-4/. A hierarchical structure showing how models can be regarded in relation to natural laws and theories is shown in Figure 5.1-2.

SKB uses the following terminology when describing models:

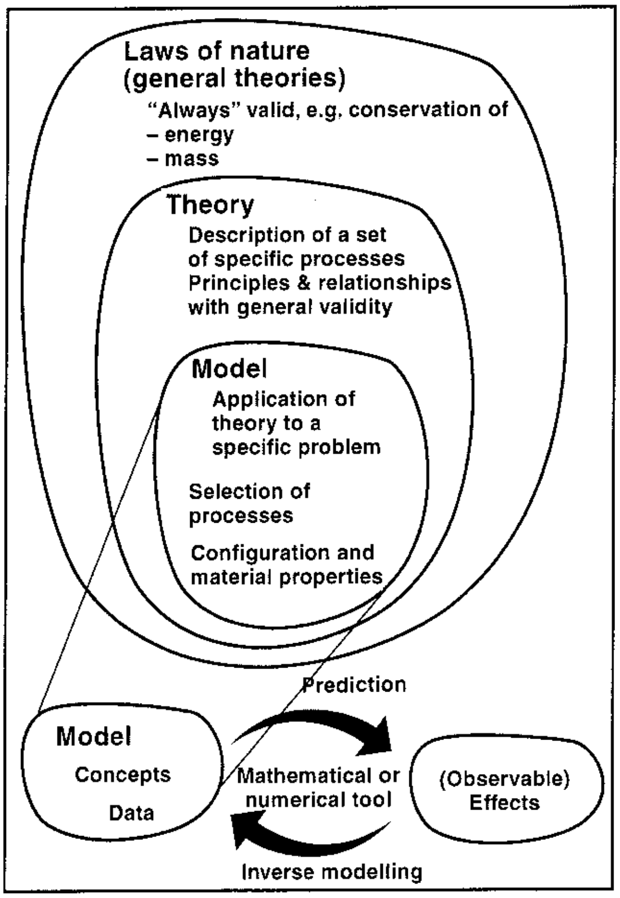


Figure 5.1-2. Schematic structure of natural laws, theories and models /5.1-4/.

conceptual model	describes the assumptions and relationships that are utilized to simulate the events or conditions to be modelled:
– **structural model**	describes the geometric or dimensional conditions that are required for modelling
– **process model**	describes how time-dependent processes are simulated and their boundary conditions,
mathematical model	describes a mathematically formulated approximation of the conceptual (structure/process) model,
numerical model	defines a numerical approximation for quantitative calculations with the mathematical model,
data	defines the parameters/ quantities included in the mathematical or numerical model,
computer code/model	defines the algorithm (for the numerical model) that is implemented on a computer and that is used to generate calculation results.

A calculation model – mathematical, analytical or numerical – is used to

- improve the understanding of a scientific problem,
- analyze which parameters are essential for solving a calculation problem with given assumptions, and for making forecasts in time and space.

SR 95, chapters 10 and 11 /5.1-3/, contains a survey of the state of knowledge for the calculation steps included in an assessment, or the system parts with more or less coupled processes that must usually be evaluated in order to be able to assess the long-term safety of the repository. They have been divided up as follows:

Thermal development
Rock performance
Canister performance
Bentonite performance
Chemical speciation
Groundwater movements
Nuclide transport near field
Nuclide transport far field
Biosphere dispersion
Dose calculation

The safety-related processes and the methods for their quantification are examined there. Any simplifications and pessimistic (conservative) assumptions are discussed. To the extent alternative methods are available for describing a process, the essential differences in conceptualization or modelling are discussed.

Knowledge and experience concerning how well a given numerical model is able to describe the reality we register in the field are discussed in specific sections in the description of the models. Separate validity documents have been prepared for the transport codes NAMMU (far field) and NUCTURAN (near field) and for the biosphere code BIOPATH. In these documents, data essential to the concept of validity have been compiled for the code version in question under the following headings:

- Fundamental theory
- Conceptual model
- Numerical methods
- Verification
- Validation
- Program documentation

The purpose of these validity documents is to determine whether it is possible in practice to standardize the reporting of experience from application of the model and of how reliable different models are for their purpose. Such a standardization would make it simpler to compare experience of the area of application and validity of different alternative models. If these validity documents prove to be useful, most of SKB's more important numerical model tools will be documented in this way.

The knowledge basis for the phenomena that are simulated in the process models – e.g. diffusion, sorption, colloid transport, etc. – is discussed in the sections that describe the barriers 5.3–5.5, or in the chemistry section 5.6.

In addition to the processes discussed in SR 95, other processes may also have to be modelled in order to analyze specific scenarios. For example, the impact of an ice age, or the risks to ore prospectors if they should unknowingly drill through the repository. The assessment models have not been standardized for such odd processes that are not generally utilized in design work or safety assessments. Description and scrutiny of methods employed to quantify or limit the effects of such processes must be done separately.

SKB is now entering a phase when safety assessments and safety reports are directly linked to permit applications and decisions. Models and methods may hereby have to be adapted to the site that is being assessed or to the specific decision the assessment is intended to support. Such adaptation will be done individually for each safety assessment, see chapter 10.

The state of development for characterizing a site and representing it with a geological structure model on the basis of site investigations is discussed in chapter 7.

5.1.3 Numerical coupling between models

Most of the calculations in a safety assessment consist of chains or networks of computer models that are linked together. A handling system called PROPER was developed for such calculations during the 1980s. The system also permits probabilistic assessments.

But with the development of menu interfaces and the ability of computers to handle increasingly complex model chains, PROPER's management system, based on text files, appears outmoded today. A graphics-based management system is therefore being developed at the present time, under the working name MONITOR-2000.

MONITOR-2000 will offer the following advantages compared with today's system:

- Quality assurance will be simplified and improved, in part because there is less risk of mistakes, and in part because the quality of the documentation is better.
- As a consequence of the improvement of the documentation, the format of the reports will be able to be made clearer and more standardized.
- A specially trained operator will no longer be needed to carry out calculations with complex model networks. Everyone who works with modelling will be able to handle even the complex calculations.
- Considerable time can be saved in the work of defining and documenting model calculations. Because these operations will no longer be a bottleneck in the procedure, in terms of either time or manpower, more calculation cases will be able to be carried out.

The development work is planned to be finished in November 1995, which means that MONITOR-2000 can be used in future safety assessments.

To enable the numerical models to be used in coupled model chains, new models will gradually be adapted to the requirements in PROPER as regards handling of input data and formats for data transfer between models.

5.1.4 Uncertainty and validity

A comprehensive analysis must be performed of the uncertainties in the assessment in order to obtain an idea of the assessment's ability to describe reality. The uncertainties introduced into the assessment at the different steps in the analysis sequence described above will be of varying character and quantifiable to a varying degree.

A general survey of our understanding and handling of the uncertainties with reference to their role in the safety assessments is performed in SR 95 chapter 3 /5.1-3/ with background reports. Uncertainties in data and models for the safety assessments is discussed in the account given for the different knowledge areas in SR 95 chapters 10 and 11. Fundamental uncertainties in our understanding of safety-related phenomena and processes and of how they are conceptualized in mathematical models is dealt with in the following sections 5.3–5.9.

Uncertainties associated with scenarios are discussed in connection with the application of the scenario methodology to the disposal system which is described in SR 95 chapter 9.

The assessment of whether knowledge and technology for handling uncertainties is adequate, or whether it must be further refined, is closely linked to how simplifications, pessimistic assumptions and safety margins are handled in each specific assessment. In the same way as for other methodology, a need for supplementary measures connected with each evaluation and account of safety is expected to exist with regard to uncertainties.

5.2 SCENARIOS

5.2.1 General

The purpose of the scenario methodology is to identify and describe scenarios to be evaluated in the safety assessment. To do this, a systematic approach is needed for identifying how the repository will develop with time. The development of the repository is dependent in part on the external conditions or events that can affect the disposal system, and in part on the internal processes that may occur due to the waste, materials present in barriers and host rock, and other materials/contaminants that may be brought down into the repository.

The work of studying and evaluating different approaches for identifying relevant scenarios has been going on for many years. A survey of scenario methodology was published in 1992 by the OECD/Nuclear Energy Agency /5.2-1/. Work has been pursued in parallel in Sweden and other countries to identify the features in and around the repository, and the possible events and processes (Features, Events and Processes) that can make it necessary for specific scenarios to be defined. This material is currently being compiled within the NEA.

The specific question of how human intrusion can affect the safety of the repository has also been dealt within by the NEA /5.2-2/. The value of, and methods for, preserving information on the repository for long periods of time has been studied in Nordic cooperation /5.2-3/. SKB and Swedish government authorities are participating in this joint international work.

The initial step in a safety assessment, after having defined the system to be dealt with, is to identify the scenarios that are of such importance for the performance of the repository that light must be shed on them in the safety report. The scenarios should cover a broad spectrum of possible pathways of development, and together they should provide perspective on the safety margins provided by the system. The different steps in the process of devising these scenarios must be documented to facilitate scrutiny and future surveys and updates. High demands will be made on the completeness of the material and thereby on ensuring that all relevant questions have been examined.

5.2.2 Principal steps in the methodology of devising scenarios

The work of structuring the process of devising scenarios started as a joint research project between SKI and SKB in 1988 and has been described in /5.2-4/. The Features, Events and Processes (FEPs) that are conceivable in a waste repository were sorted there into two groups, either the Process System, consisting of "internal FEPs" which define the performance of the repository (including variations occasioned by parameter deviations) or a group of "External FEPs", each of which can influence the Process System and thereby give rise to different scenarios.

The Process System is defined as a systematic compilation of the phenomena (FEPs) that are needed to describe barrier performance and the mechanisms for radionuclide transport. In order to forecast the development of the repository via the Process System, it must be possible to quantify FEPs thoroughly by means of measurement data, modelling or estimates.

The methodology for devising scenarios has the following principal steps:

- Identification of the Process System plus visualization and documentation.
- Identification of initiating external events or extreme cases.
- Choice of scenarios and calculation cases to treat in the safety assessments.

Identification of the Process System and visualization

The first step is to identify the Process System in text and by visual methods so that all known couplings between component processes are shown. Identification of the FEPs included in the Process System can be done in several ways, and tests have been conducted with different methods during 1991–1994. The following two methods in particular have been used:

- Influence diagrams /5.2-5/
- The Rock Engineering System (RES) method /5.2-6/

Influence diagrams

In an influence diagram, all FEPs are represented by boxes. Couplings/interactions between different FEPs

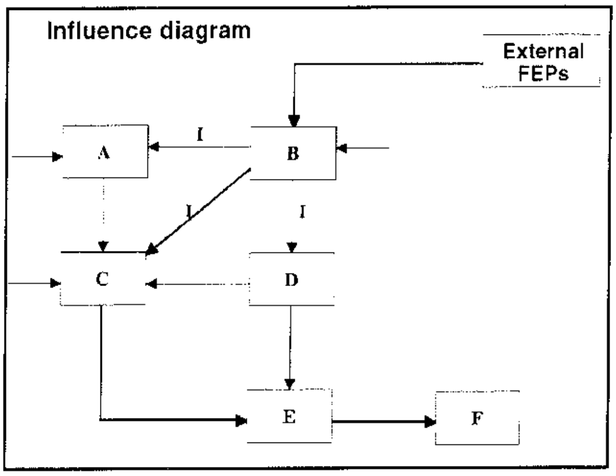


Figure 5.2-1. Schematic illustration of an influence diagram where an initiating event (External FEP) influences the processes.

are illustrated by arrows between these boxes, see Figure 5.2-1. Each box and arrow in the diagram has an identification code and is documented in databases which are directly accessible from the software in which the influence diagram is produced.

The influence diagrams permit a detailed analysis of how different FEPs are linked to or influence other FEPs in the Process System. The methodology has been tested in a study on deep disposal of "Other long-lived waste than spent nuclear fuel" /5.2-5/. The conclusions of this study were that it is fully possible to carry out the systematic analysis of the Process System's different FEPs and to represent this in an influence diagram or in several sub-diagrams for different sub-systems. The databases that are simultaneously produced contain specifications of all influences and notations of the assessments made for each influence. Visually, however, the influence diagrams are complex and do not provide the quick overview of the Process System that would be desirable.

The RES method

The RES method entails identifying the repository's most important sub-systems with their features and representing them on the diagonal in a matrix. The other boxes in the matrix will now represent interactions between these sub-systems, i.e. of the processes that are involved, and the matrix comprises the Process System.

So as not to make the matrices too big (since processing of the matrix would then be too complex), the Process System is divided into smaller parts, see Figure 5.2-2. An example of the contents of a sub-matrix for the far field part of a deep repository is shown in Figure 5.2-3.

Work is currently under way on coupling the various interaction boxes in the sub-matrices to databases in a manner similar to what has previously been done for the influence diagrams in the aforementioned study concerning "Other long-lived waste".

More detailed descriptions of influence diagrams and the RES method, plus comparisons between them, are given in /5.2-7/.

The current scenario work is concentrated on the RES method.

Identification of initiating external events or extreme cases

Through the Process System's RES matrix, it is possible to

- make a survey of the potential pathways that exist for various internal disturbances to propagate through the system via the interaction boxes,
- examine how different External FEPs can influence the Process System and how a possible disturbance can be propagated through the system (the RES matrix may have to be modified for larger disturbances),
- provide a graphic description of how this chain of coupled processes (interaction pathways) affects long-term safety.

This information, together with the external initiating events (External FEPs) that lie outside the Process System, provides a number of scenarios that can be interesting to examine in quantified safety assessments.

However, it is necessary to be aware of the fact that both the creation of FEP lists for influence diagrams and the choice of diagonal elements in RES matrices are dependent on our understanding of the disposal system, and that both the identification of normal repository

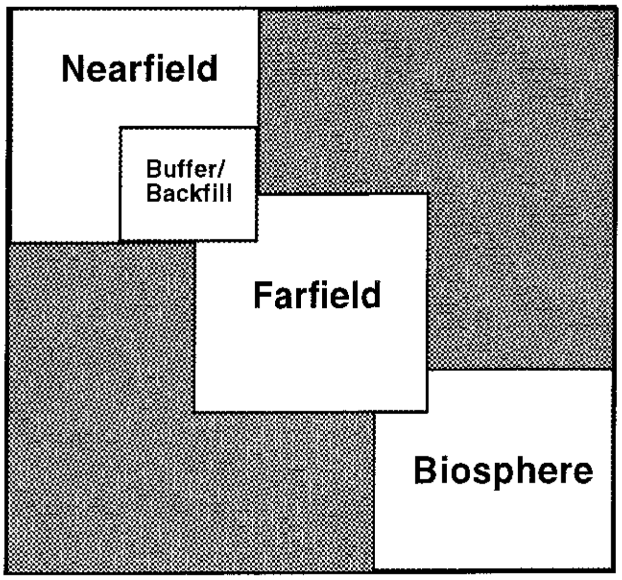


Figure 5.2-2. Example of division of the total system into sub-matrices.

FAR FIELD

process system - far field 1

- Interaction which should be part of the performance assessment
- Interaction present - Influences on other parts of the process system in a limited or uncertain way and/or under special circumstances
- Interaction present - Influences on other parts of the process system can be neglected

CONSTRUCTION/ LAYOUT	1.2 Excavation method	1.3 Excavation method Grouting Reinforcement	1.4	1.5 Displacement effects	1.6 Construction materials Stray materials	1.7	1.8 Drawdown effects	1.9 Repository depth Ventilation	1.10 Tunnel dimension	1.11 Ventilation Blasting gas Gas source	1.12	1.13 Industrial facility Dumps
2.1 Swelling ability Heat	**BUFFER/ BACKFILL/ SOURCE**	2.3 Buffer/backfill penetration into EDZ	2.4	2.5 Buffer into intersecting fractures	2.6 Colloid source Groundwater composition	2.7 Changed flow around holes Changed flow in tunnels	2.8 Resaturation	2.9 Heat generation	2.10 Swelling pressure	2.11 Gas source	2.12 Source term	2.13
3.1 Excavation method Amount of reinforcement	3.2 Volume for buffer/backfill swelling Rock fallout	**EDZ**	3.4	3.5	3.6 Changed R_i and a_r Colloid and particulate generation	3.7 Changed permeability	3.8	3.9 Modified thermal diffusivity	3.10 Fractures affected	3.11 Indiffusion of air Transport path for gas	3.12 Changed R_i and a_r K_i	3.13
4.1 Layout/ construction method	4.2	4.3 Magnitude and geometrical extent	**ROCK MATRIX/ MINERALOGY**	4.5 Fracture characteristics and infilling mineralisation	4.6 Rock-water interaction	4.7 Matrix K Rock compressibility	4.8	4.9 Thermal properties	4.10 Genesis, tectonic history and rock type	4.11 Radon generation	4.12 Sorption Matrix diffusion	4.13 Land-use Potential human intrusion
5.1 Avoid major zones Constructability	5.2	5.3 Mechanical properties and fracture frequency	5.4	**NATURAL FRACTURE SYSTEM**	5.6 Dissolution of fracture minerals Colloid generation	5.7 Flow paths Connectivity Fracture aperture Storage capa.	5.8	5.9 Thermal properties	5.10 Stress magnitude and orientation	5.11 Transport path for gas	5.12 Molecular diffusion a_r Sorption	5.13 Wells
6.1 Depth affected by radexpot. Construction materials	6.2 TDS Ion exchange Infiltration	6.3 Precipitation/ bacterial growth	6.4 Groundwater rock interaction	6.5 Precipitation and dissolution of fracture minerals	**GROUNDWATER CHEMISTRY**	6.7 Density Viscosity	6.8 Density affects groundwater head	6.9 Heat conductivity	6.10	6.11 Chemically generated gas Microbially generated gas	6.12 Sorptivity and solubility Colloids and bacteria	6.13 Water-use Biotopes
7.1 Canister positioning Construction methods	7.2 Saturation Bentonite erosion	7.3 Erosion	7.4	7.5 Erosion and sedimentation	7.6 Mixing	**GROUNDWATER MOVEMENT**	7.8 Equalisation of pressures	7.9 Forced heat convection	7.10	7.11 Two-phase flow	7.12 Transport of dissolved gas Sorbing & nonsorbing tracers Hydrodynamic dispersion	7.13 Recharge and discharge
8.1 Construction methods	8.2	8.3	8.4	8.5	8.6 Solubilities	8.7 Driving force due to pressure gradient	**GROUNDWATER PRESSURE**	8.9	8.10 Effective stress	8.11 Gas solubility Gas law	8.12	8.13 Potential effect on vegetation
9.1	9.2 Temperature in buffer backfill	9.3	9.4 Thermal expansion Thermal conductivity	9.5 Permafrost	9.6 Dissolution and precipitation of minerals	9.7 Viscosity	9.8 Density	**TEMPERATURE/HEAT**	9.10 Thermal expansion	9.11 Gas solubility Gas law	9.12 Kinetic effects	9.13
10.1 Design/layout Construction methods	10.2 Reaction force on swelling pressure Rock fallout	10.3 Mechanical stability Fracture aperture	10.4 Mechanical stability	10.5 Mechanical stability Fracture aperture	10.6	10.7	10.8 Confined aquifers	10.9	**ROCK STRESSES**	10.11	10.12	10.13 Mechanical stability
11.1 Ventilation problems	11.2	11.3 Opening of fractures Heat conduction	11.4 Fracturing Thermal properties	11.5 Fracture aperture	11.6 pH, Eh affected	11.7 Creation of 2-phase flow conditions	11.8 Capillary forces	11.9 Gas law	11.10	**GAS GENERATION AND TRANSPORT**	11.12 Colloid sorption on gas bubbles	11.13 Gas release
12.1 Design/ layout	12.2	12.3	12.4	12.5	12.6 Changed concentrations	12.7	12.8	12.9	12.10	12.11	**TRANSPORT OF RADIONUCLIDES**	12.13 Contamination
13.1 Siting Design/ layout	13.2	13.3	13.4	13.5	13.6 Infiltrating water	13.7 Surface water recharge & percolation	13.8 Land use Climatic & tidal driving force Hydraulic gradient	13.9 Climatic driving forces	13.10 External load Erosion	13.11	13.12	**BIOSPHERE**

Figure 5.2-3. Sub-matrix of the far field in a deep disposal system.

performance and extreme scenarios will contain a measure of expert judgement.

Choice of scenarios and calculation cases to treat in safety assessments

The choice of scenarios to be ultimately included in the safety assessments is made with reference to the questions which the safety assessment is intended to answer. The knowledge and sensitive disturbance pathways discussed above provide guidance in the choice.

The final list of scenarios and calculation cases selected to provide a complete picture of the "threats" to the repository's safety (external and internal FEPs) will also be dependent on "expert judgement". This is a consequence of the fact that neither the background data for nor the understanding of the process that are dealt with in various scenarios permit each scenario to be evaluated with the same accuracy and equally extensive quantification. However, the background underlying the scenario choice will be able to be reported more clearly and be more traceable as a result of the methodology developed as described above.

Besides the scenarios that are intended to depict, through various systematic approaches, realistic possible developments of the repository's safety, other types of scenarios will also be examined, often called "What-If Scenarios", "Worst-Case Scenarios" or "Bounding Scenarios". For example, it is possible to carry out calculations under the assumption that a release from the near field winds up directly in the biosphere without passing through the bedrock, or to exclude in calculations e.g. dispersion or diffusion in the rock matrix. These scenario types are aimed, as the names indicate, at shedding light on the safety-related importance of different barriers or retardation mechanisms. Even though the scenario descriptions are often highly simplified and physically unreasonable, they can provide an upper boundary for the consequences. These unrealistic scenarios are often used to illustrate in a simple manner how robust or how sensitive the safety of the repository is to uncertainties in different barriers or processes.

5.2.3 Ongoing work

The work of establishing practical systematics and traceability in the scenario work continues. The status of the work is described in SR 95 /5.1-3/, where a survey is made in section 3.3 of different ways to establish the scenario systematics. In chapter 9, the RES method is applied to the disposal system that is now being defined in preparation for the siting process and for permit applications for the encapsulation plant.

The systematics in the scenario work is supposed to make it easier to evaluate whether the choice of scenario covers the important development pathways for the repository, and whether the processes essential to safety have been identified. SKB believes that this can be achieved by means of many methods. A crucial criterion for the choice of methodology is whether it meets the stipulated requirements, how practical it is to apply and what kind of overview it provides. Method development within the scenario field will continue both nationally and internationally. Results and experience may lead to changes in the recommended methodology or applications.

5.3 SPENT FUEL

In order for radioactive materials to escape from the repository, a leak must have occurred in a copper canister, for example due to corrosion or mechanical stress. Water can then come into contact with the fuel and water-soluble radionuclides can be released from the fuel. In the case of nuclides that have been released to some proportion from the uranium matrix during reactor operation, such as iodine, this can take place without the fuel itself necessarily being affected. Otherwise, release of radioactivity requires that the fuel either be dissolved or corrode in the water. The goal of the research programme is to investigate how rapidly radionuclides are released from the fuel under different conditions that might prevail in a deep repository, and to describe the release with the aid of suitable models.

Investigations of the fuel's durability in groundwater have been conducted in Sweden since 1977. An overview of results and data obtained within the framework of SKB's programme in recent years is provided in /5.3-1/, together with a comparison with the database that has been gathered during the past ten years.

Several different models for calculating fuel dissolution have been discussed during the years. The hypotheses range from strongly conservative, highly simplified models for calculations in conjunction with safety assessments, to attempts at phenomenologically correct description of fuel dissolution.

The simplest hypothesis is to completely disregard the kinetics of the fuel dissolution and regard it as instantaneous. The release of radionuclides will then be controlled by the individual solubilities of the fission products and the actinides under repository conditions. This description defines an extremely conservative upper boundary for the releases from the canister, and it gives a far-from-correct picture of the fuel dissolution.

An attempt was made in SKB 91 to take into account the dissolution kinetics by using the release of strontium as a measure of the oxidation and dissolution of the fuel matrix /5.3-2/. This model was also very conservative, since the rate of release for strontium that was used was measured in the presence of oxygen from the atmosphere. In the deep repository, only oxidants produced by radiolysis will be present after a few hundred years.

In the model used by AECL Canada, it is assumed that the fuel matrix remains stable and that the solubility of the fuel itself controls the release of radionuclides /5.3-3/. This model is probably the most realistic one in the longer time perspective. Due to the higher actinide content of light-water fuel, compared with CANDU fuel, the effects of radiolysis, particularly α-radiolysis, should be further investigated before a similar model is applied to our conditions.

This section provides a brief account of the importance of a number of important factors such as irradiation history, the mechanism of matrix oxidation, groundwater chemistry, redox conditions, sorption on the buffer mass etc. for leaching of spent nuclear fuel, and thereby also for modelling of the fuel dissolution.

5.3.1 Corrosion of spent fuel

Analytical methods

An experimental programme has been conducted at the Studsvik fuel laboratory since 1982. Short pieces of fuel and cladding have been exposed to corrosion in synthetic groundwater under oxic and anoxic conditions. The filtered solution, the material on the filter and material dissolved in the acid used to rinse the leach vessels have been analyzed. The analysis programme has included laser fluorescence (U), alpha spectrometry (Pu, Cm), gamma spectrometry (gamma-emitting fission products) and radiochemical separation and beta spectrometry (Sr, Tc). The results have provided a good understanding of fuel corrosion, and a summary of the results in the programme has been published /5.3-1/.

The possibility for multi-element analyses provided by mass spectrometry with an inductively coupled plasma source (ICP-MS) can replace all other previously used methods and furthermore provide data for other radioactive elements of smaller radiological importance. Analyses of these radionuclides can be of great importance for understanding the corrosion processes. An ICP-MS instrument has been in operation at Studsvik since 1992. Since the isotope composition in the fuel can vary from sample to sample depending on enrichment, burnup and decay, considerable data processing has been done of sample-related variables /5.3-4/.

The new technology allows uranium contents to be measured with good precision and accuracy under both oxic and anoxic conditions. The same applies to Np and Pu under oxic conditions. For anoxic conditions, on the other hand, precision is lower due to the low concentrations of these actinides. Direct analysis of Am and Cm in the leachate has given unsatisfactory results due to poor measurement statistics and uncertainties in the background level /5.3-5/.

The results from measurements of relative release rates for caesium and technetium in the corrosion tests performed to date are shown in Figure 5.3-1.

Area determination

In SKB's previous fuel corrosion work, leach rates have been expressed as fraction of inventory in the aqueous phase. This provides a general picture of the fuel dissolution process. However, the leach rate is highly dependent on the exposed area of the fuel, which is in turn dependent on several factors related to the fuel's irradiation history. The area of the fuel that is accessible to penetrating groundwater is an essential parameter for determining the corrosion rate in absolute terms. Experiments for determining specific fuel area with BET technology are currently under way. Reproducible values for the fuel area in the in the range 70–120 cm^2/g have been obtained. However, attempts to relate these values to fuel burnup /5.3-6/ do not reveal clear relationships.

Matrix dissolution

In order to be able to predict the release of radionuclides from spent fuel, it is of great importance to clarify how and to what degree this release is related to the dissolution of the UO_2 matrix. This is particularly true for strontium, since this element has been suggested as an indicator of matrix dissolution. Arguments have been offered for and against this hypothesis, however /5.3-7/.

Observations of the release rate for ^{90}Sr indicate a dependence on the migration and segregation that has taken place during irradiation /5.3-1/. Attempts to identify such segregation by means of electron microprobe analysis have not succeeded, however /5.3-8/. It has been suggested that the underlying mechanism is migration of ^{90}Sr's parent nuclides ^{90}Br and ^{90}Rb /5.3-9/. This is supported by the fact that significant quantities of Rb have been detected on surfaces of CANDU fuel with the aid of X-ray photoelectron spectroscopy /5.3-10/.

To test this hypothesis, the migration of Sr was studied by imposing a steeper than normal temperature gradient to the fuel in a power bump test /5.3-11/. Then the concentrations in solution were measured in a series of short leaching experiments and compared with reference samples. The results of both ICP-MS and radiochemical/radiometric methods are shown in Table 5.3-1.

The release rates for Cs, Rb and Sr for reference samples and for samples from the power bump test are shown in Figure 5.3-2.

The results show that rubidium migrates in the fuel, but to a lesser degree than caesium and iodine. No strontium migration of significance, within experimental uncertainty, has been detected. Of the other fission products studied (Mo, Tc and Ba), only molybdenum shows a significantly higher release at higher power density. It is possible that this was caused by oxidation of the fuel during sample preparation.

Flow reactors have been used in a number of studies of the kinetics of dissolution of sparingly soluble solids

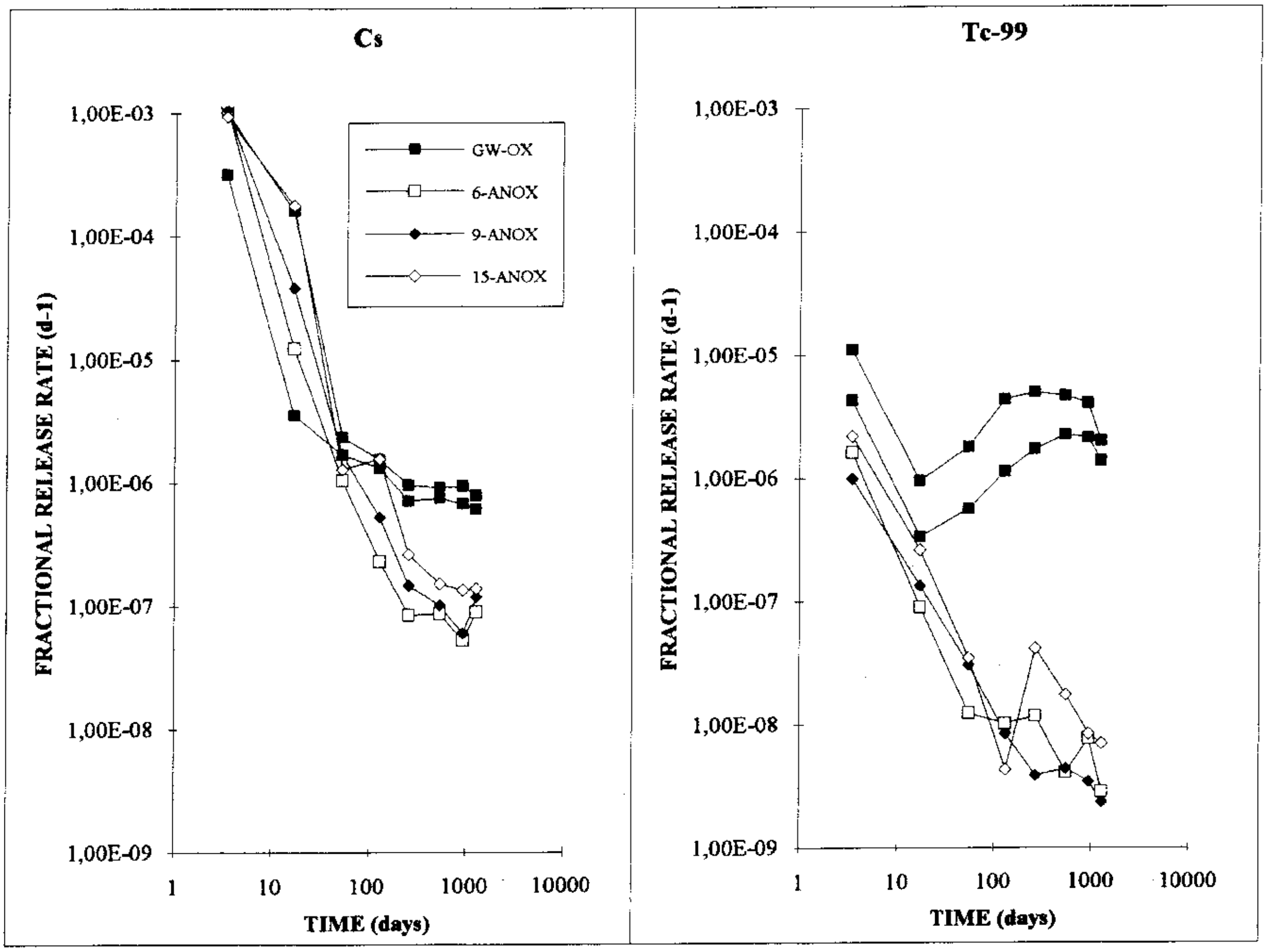


Figure 5.3-1. Fractional release rate per day for caesium (Cs) and technetium (Tc-99) under oxic (GW-OX) and anoxic (6, 9, 15 – ANOX) conditions.

Table 5.3-1: Cumulatively released fraction of the inventory after four contact periods. GS: gamma spectrometry. RC: radiochemical analysis. The > sign means that the nuclide was not detected in the first contact.

		Reference samples		"Power bump"-samples	
Nuclide	**Measurement-method**	**13-1**	**13-2**	**13-3 33,0 kW/m**	**13-4 42,6 kW/m**
Cs-133	ICP	$5{,}20 \cdot 10^{-3}$	$5{,}82 \cdot 10^{-3}$	$2{,}18 \cdot 10^{-2}$	$1{,}16 \cdot 10^{-1}$
Cs-137	GS	$5{,}02 \cdot 10^{-3}$	$5{,}57 \cdot 10^{-3}$	$2{,}08 \cdot 10^{-2}$	$1{,}12 \cdot 10^{-1}$
I-131	GS	NM	NM	$7{,}05 \cdot 10^{-3}$	$7{,}77 \cdot 10^{-2}$
Rb-85	ICP	$4{,}61 \cdot 10^{-3}$	$4{,}82 \cdot 10^{-3}$	$5{,}74 \cdot 10^{-3}$	$3{,}39 \cdot 10^{-2}$
Rb-87	ICP	$1{,}25 \cdot 10^{-3}$	$1{,}29 \cdot 10^{-3}$	$5{,}62 \cdot 10^{-3}$	$3{,}58 \cdot 10^{-2}$
Sr-89	RC	NM	NM	$1{,}70 \cdot 10^{-4}$	$1{,}79 \cdot 10^{-4}$
Sr-90	RC	$2{,}82 \cdot 10^{-4}$	$2{,}93 \cdot 10^{-4}$	$1{,}75 \cdot 10^{-4}$	$1{,}86 \cdot 10^{-4}$
Sr-90	ICP	$2{,}81 \cdot 10^{-4}$	$4{,}06 \cdot 10^{-4}$	$1{,}73 \cdot 10^{-4}$	$1{,}79 \cdot 10^{-4}$
Sr-88	ICP	$2{,}76 \cdot 10^{-4}$	$3{,}51 \cdot 10^{-4}$	$1{,}49 \cdot 10^{-4}$	$1{,}53 \cdot 10^{-4}$
Ba-138	ICP	$3{,}51 \cdot 10^{-4}$	$3{,}67 \cdot 10^{-4}$	$2{,}38 \cdot 10^{-4}$	$5{,}30 \cdot 10^{-4}$
Ba-140	GS	NM	NM	$>2{,}1 \cdot 10^{-4}$	$6{,}14 \cdot 10^{-4}$
Mo-98	ICP	$1{,}24 \cdot 10^{-4}$	$1{,}17 \cdot 10^{-4}$	$3{,}04 \cdot 10^{-4}$	$5{,}88 \cdot 10^{-3}$
Mo-99	GS	NM	NM	$>2{,}8 \cdot 10^{-4}$	$>7{,}5 \cdot 10^{-3}$
Tc-99	ICP	$6{,}50 \cdot 10^{-5}$	$6{,}27 \cdot 10^{-5}$	$7{,}44 \cdot 10^{-5}$	$1{,}60 \cdot 10^{-4}$
U	ICP	$2{,}01 \cdot 10^{-5}$	$2{,}36 \cdot 10^{-5}$	$5{,}64 \cdot 10^{-6}$	$1{,}20 \cdot 10^{-5}$

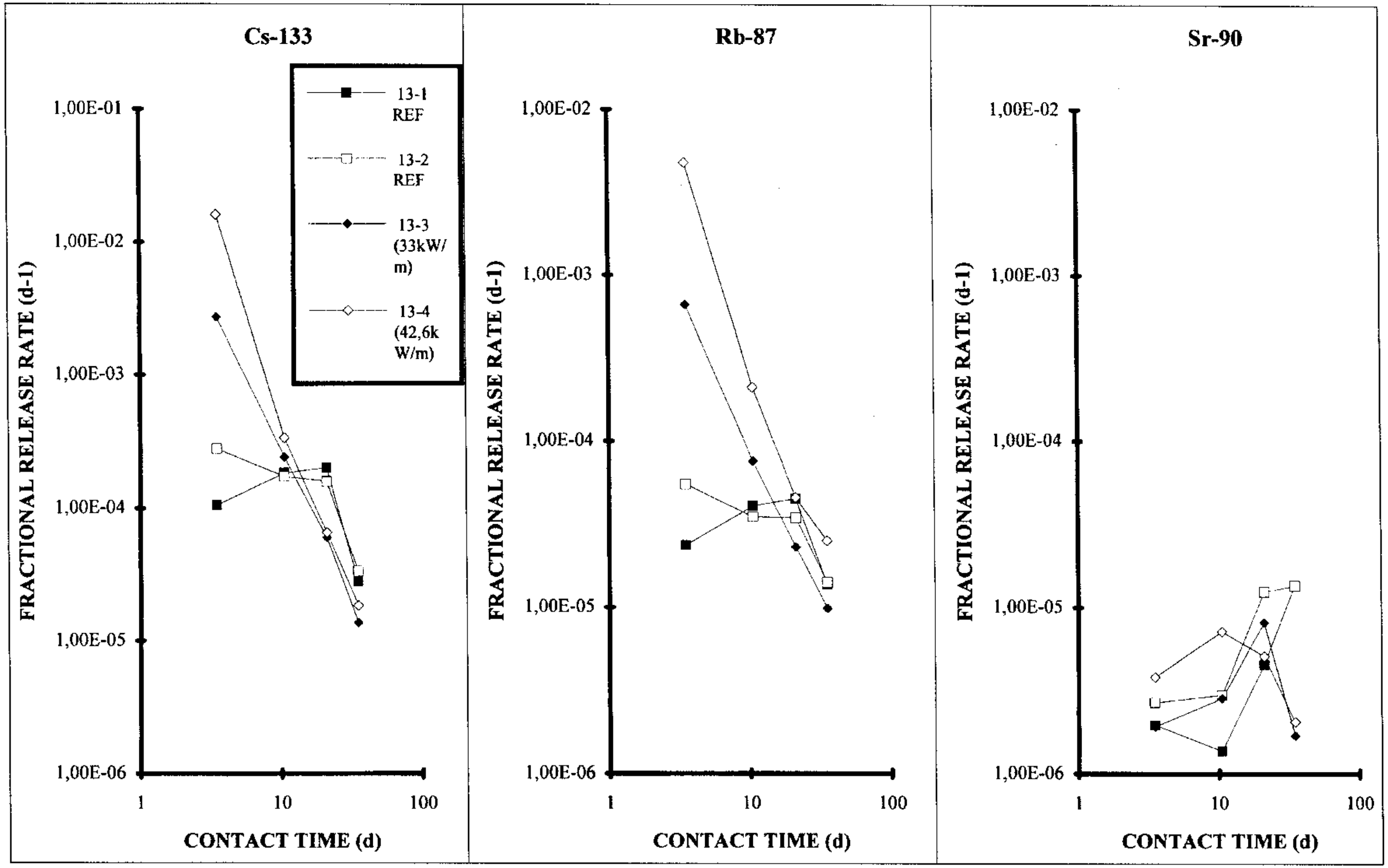


Figure 5.3-2. *Fractional release rate per day for reference samples (13-1 and 2) and "power bump" samples (13-3 and 4).*

/5.3-21/, as well as in leaching studies of uraninite, SIMFUEL /5.3-22/ and fuel /5.3-23/. In this type of reactor, the system will evolve to a steady state if the flow, the composition of the incoming solution, the composition of the solid phase and its surface area remain constant. No precipitation of secondary phases will take place, since the reaction products are flushed out of the reactor before any saturation occurs. An important distinguishing characteristic of this system is that the solid phase that is under observation does not require any manipulation. When the outer layer has been dissolved, new surface will continuously be exposed to the leaching solution. Diffusion of solution species to the surface of the solid phase is also minimized by using thin layers of solid phases in the reactor. The same fuel can be studied under different conditions by changing the leaching solution outside the reactor. Planning to utilize the aforementioned technique in SKB's fuel studies is under way.

5.3.2 Other components in the near field

Investigations of the influence of bentonite on fuel corrosion have been going on since the second half of the 1980s. Preliminary evaluations of experiments with contact times of up to one year have been reported previously /5.3-12, 13/. The evaluation of the release of ^{90}Sr from fuel in contact with compacted bentonite for six years has also been concluded /5.3-14/. Figure 5.3-3 shows the release of strontium as a function of time. After six years the fraction of released Sr is five to ten times lower than in similar experiments in the absence of bentonite. In experiments performed with bentonite containing metallic iron or the Fe(II) mineral vivianite, the fraction of released Sr is lower than in bentonite with metallic copper or without any additives whatsoever.

The leaching behaviour of caesium is different from that of strontium and exhibits an initial pulse of a rapidly released fraction /5.3-13/. The evaluation of the caesium leaching in bentonite does not exhibit any significant increase in released fraction over a six-year period. The small but steady increase in released caesium that was observed in the leaching experiments without bentonite /5.3-1/ is probably concealed by the quantity of initially released caesium, which in the experiments with bentonite remains throughout the experimental period /5.3-14/.

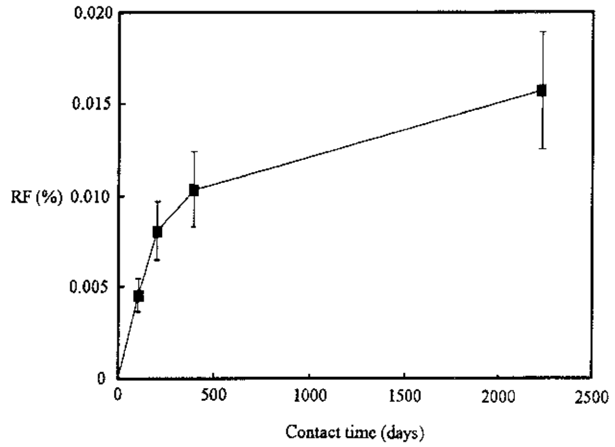


Figure 5.3-3. Released fraction of ^{90}Sr after different contact times with bentonite clay without additives.

5.3.3 Models

Experimental data from dissolution of spent fuel and uranium dioxide under oxidizing conditions suggests a kinetic rather than thermodynamic control of the process. The total dissolution rate in carbonate-containing groundwater can be related to the following processes:

- Initial rapid dissolution of already oxidized UO_{2+x} surface layer. This initial dissolution rate has been determined experimentally /5.3-15/.
- Oxidative dissolution of the matrix in bicarbonate solution. The rate of the process is dependent on the concentrations of oxidants and bicarbonate.
- Precipitation of secondary phases. The rate of this process is dependent on the U(VI) concentration, the fuel area to solution volume ratio, and the composition of the groundwater.

Preliminary calculations indicate that it is possible to develop a quantitative model that takes the above-mentioned processes into account. Approximate calculations show that it is possible to reproduce experimental observations. Additional experiments and modelling work are necessary to strengthen the grounds for the model.

5.3.4 Natural analogues

Safety and performance assessments of a deep repository are based on, among other things, descriptions of the fuel's behaviour over long periods of time. Fuel leaching experiments aimed at kinetic and thermodynamic modelling of fuel dissolution under the conditions prevailing in the deep repository cannot be directly extrapolated to very long times. These time scales can be bridged by studies of alteration processes in natural uraninites under mildly oxidizing and reducing conditions.

The fuel's long-term stability is linked to the reaction paths for the oxidative transformation of the fuel matrix. The oxidation of natural uraninite in air /5.3-16/ shows that at temperatures of up to 300°C, only U_4O_9 type oxides are formed. Experimental and literature data cannot confirm that α-U_3O_7 is formed. With a further increase in temperature, or in the partial pressure of oxygen, a saturation of excess oxygen in the isometric UO_{2+x} structure is probably reached and it is transformed in air to U_3O_8. In the presence of water, uranium oxide hydrates such as schoepite are formed.

Studies of air oxidation of irradiated light-water reactor fuel indicate a similar oxidation mechanism /5.3-17/. First cubic U_4O_9 is formed, which is stabilized by the fission products and reaches an oxygen/uranium ratio of up to 2.4 while maintaining the same fluorite structure. The influence of the fission products is confirmed by a similar behaviour for Gd-doped irradiated UO_2. Further oxidation does not appear to proceed via U_3O_7. This makes it necessary to revise the importance of this phase in the modelling of the long-term corrosion of the UO_2 matrix in spent fuel.

Dissolution experiments have been conducted with natural uranium phases under various redox conditions, combined with detailed characterization of the solid phases before and after the experiments /5.3-18/, in order to obtain supporting data for a thermodynamic and kinetic model for the long-term oxidation of uraninite. These results can be applied to the long-term behaviour of spent fuel. As a part of this work, the solubilities of soddyite, uranophane and becquerelite have also been determined.

5.3.5 Activities in relation to goals in RD&D-Programme 92

- Progressively refine the models for the safety assessment in 1996. – Work is under way on determining the importance of the α-radiolysis for the oxidation/dissolution of the fuel and interim results will be published /5.3-19, 20/.
- 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. – The work is related to the work on α-radiolysis.

5.4 BUFFER AND BACKFILL

5.4.1 Functional requirements

Buffer

The choice of buffer material and the design of the buffer around the canister are based on the fundamental functional requirements in section 4.3. In addition there are economic requirements on good availability of the material type to be used and low occurrence of accessory minerals in the form of sulphur minerals and organic substances. Swelling clay with a high smectite content comes closest to meeting these requirements.

If the fundamental requirements are met, a number of other favourable properties are also obtained, such as:

- limitation of transport of corrodants up to the canister surface (diffusive transport),
- sorption of radionuclides,
- only diffusive transport of radionuclides from a defect canister to the rock in the near field,
- drying-out and filtering of micro-organisms,
- stabilization of the walls of the deposition hole,
- stabilization of the chemical environment (Eh, pH).

The different functions are controlled by the following measurable properties of the buffer:

- water content,
- hydraulic conductivity,
- swelling pressure,
- swelling capacity,
- shear strength,
- rheological properties,
- pore volume,
- diffusion and sorption properties,
- thermal conductivity,
- chemical composition.

Approximate values for satisfying the various functional requirements determine the suitable composition of the buffer, acceptable accessory minerals and impurities, and the limits within which the material's bulk density after deposition and water saturation should lie. Such considerations have governed the choice of buffer material in the BMT tests in the Stripa Project /5.4-1/.

Interaction with near-field rock

The importance of the rock for the bentonite buffer has been analyzed in different studies /5.4-2, 3/. The following requirements on the rock have hereby been specified in order that the buffer should achieve and retain the desired function:

- The rock supplies the buffer with groundwater so that water saturation and swelling occur.
- The near rock has such a composition that the buffer is not affected in a way that leads to serious cementation or mineral alteration.
- The rock does not contain discontinuities of such a nature that openings are formed in which bentonite can penetrate, causing a considerable decrease in buffer density in the deposition hole.

It is judged that these requirements can be satisfied in commonly occurring Swedish crystalline rock at the depths in question for the deep repository. Practical experience has been obtained through the Stripa Project /5.4-4/. Furthermore, water saturation and swelling can be facilitated by compaction of the bentonite blocks with a high water content and by filling the gaps between block and canister and between block and rock with water in conjunction with deposition.

Backfill materials

Different mixtures of bentonite and aggregate are planned to be used for backfilling of tunnels, rock caverns and shafts. The required function of these materials is to:

- counteract swelling of the bentonite out of the deposition hole,
- prevent or limit the flow of water in the tunnel and around canister positions,
- resist chemical transformation during a long period of time,
- not lead to significant chemical transformation of the buffer around the canister.

In those cases where the water transport around the canisters is not affected by the hydraulic conductivity in certain sections of the tunnel, the requirement on mechanical stability will be the most important. Unmixed aggregate can then be considered. Such tunnel sections may have to be sealed at either end.

The properties that determine the performance of the material and that can be measured are the same for bentonite/aggregate mixtures as for pure bentonite. An additional property of importance for backfill materials is compressibility, which determines how swelling-out from the deposition holes can be limited.

5.4.2 Step-by-step development stages and compilation of present-day knowledge

Knowledge of the properties of bentonite clay as a buffer around the canister has been built up over a long period of time through extensive studies in laboratories and in the field. The activities have been pursued iteratively with the following thrust:

- Fundamental knowledge about the buffer:
  - Build-up of broad knowledge
  - Development of calculation models
  - Inventory of materials
- Analysis of individual processes:
  - Interaction with rock
  - Interaction with canister
  - Longevity
- Analysis of coupled processes:
  - Function in deep repository

At present a compilation of knowledge and experience gained so far from the research conducted is in progress. It will comprise three parts, of which Part I, which deals with basic definitions and method descriptions, is finished /5.4-5/. Part II is concentrated on material descriptions and Part III on compilation of models. The purpose of the compilation is to present uniform definitions of physical and chemical concepts and phenomena relating to buffers and backfills, standardized and recommended laboratory and field test methods, and mathematical models for description and prediction of practically significant processes and functions.

5.4.3 Properties of different bentonite materials

The material structure of bentonite buffers has been described qualitatively in a general microstructural model (GMM) /5.4-6/. This model is based on the fact that the buffer material consists of ground, dried bentonite which, on being compacted, forms a mass with a random distribution of bentonite grains and with a random size and shape of interstitial pores. On water saturation and subsequent homogenization, the pores are first filled with water and then with bentonite gel, whereby the end product is a buffer with bentonite grains of high density interspersed with pore spaces filled with a bentonite gel of lower density. Different grades of end products are obtained for Na- versus Ca-based bentonites. The gel-filled pore spaces are assumed to constitute the dominant passages for transport in the buffer, and GMM serves today as a tool for judging the reasonableness of laboratory-determined transport parameters for water and gas.

Mass transport in a water-saturated bentonite buffer is limited to diffusion. This applies to both transport of solutes from e.g. the groundwater in towards the canister and transport of radionuclides out from a defective canister. Diffusion is a very slow process and is therefore an important barrier property of the bentonite clay. It protects the canister against the influence of harmful substances from the outside and is an extra barrier against the escape of radioactive materials if the canister should fail.

In order to predict the degradation of buffer and backfill materials, a model has been developed founded on the mathematical model for conversion from smectite to illite that is based on Pytte's theorem /5.4-7/. The model shows that degradation takes place either in the form of a transformation from smectite to illite via mixed-layer minerals or by new formation of illite. In both cases, the change is assumed to be of the Arrhenius type, i.e. controlled by the activation energy and the temperature, but also dependent on the supply of potassium. Applications to geological examples (Kinnekulle, Hamra and Burgsvik) and laboratory experiments /5.4-8/ support the validity of the model and thereby the conclusion that the illite alteration in the buffer and backfill with crushed rock as aggregate does not constitute a problem on a 100,000-year time scale if a suitable type of bentonite has been chosen. The primary alteration that is foreseen to occur will take place during the period with elevated temperature immediately after deposition.

A chemical influence that has been brought to attention in the ongoing work of modelling of bentonite degradation is that salt enrichment can take place in the buffer nearest the canister in conjunction with wetting when the groundwater has a high electrolyte content. Laboratory tests conducted with samples with a low water content from the start indicate, however, that the phenomenon can be avoided if the density of the water-saturated buffer amounts to about 2.0 g/cm^3 and blocks with a high original degree of water saturation are used in combination with a supply of water with a low electrolyte content /5.4-9/.

The result of an inventory of different buffer materials is presented in /5.4-10/. This study shows that with application of the model for degradation by conversion of smectite to illite /5.4-11, 12/, the lowest initial smectite content should be 50%. The study shows that only the smectite types montmorillonite and saponite with sodium as the main adsorbed ion should be considered. However, a high initial smectite content is very valuable to ensure effective self-healing and homogenization of the buffer. Such bentonites, with a montmorillonite content of 70-90% and with sodium as the dominant adsorbed cation, are commercially available in the USA, Greece and Italy. No commercially interesting bentonite deposit has been found in Sweden. However, the State Mining Property Commission (NSG) has investigated the prospects for exploitation of deposits in the southernmost province of Sweden and found indications of deposits in wellholes. The few clay analyses that have been performed show that smectitic clay minerals exist in a basalt-tuff parent rock. However, very little is known about the extent and thickness of the tuffs.

The preliminary choice of backfill material is mixtures of 10-20% bentonite and the remainder aggregate that has been laid out and compacted in place. For environmental and economic reasons, use of the rock that is excavated in the deep repository, which is crushed to a suitable grain size, is recommended. The properties of such material have been investigated and found to be comparable with the properties of quartz sand /5.4-13/.

A technique for determination of the thermal conductivity of buffer material has been developed and determinations have been performed at different densities and degrees of water saturation. The results have been compared with theoretical predictions and with data from field tests at Stripa /5.4-14/. The results exhibit good agreement when the buffer is near water saturation, but not so good under unsaturated conditions.

5.4.4 Calculation models for various functions

A general model for Thermo-Hydro-Mechanical (THM) processes in water-saturated buffer has been developed /5.4-15/. It can be applied to, among other things, swelling of buffer against backfilled tunnels, settlement of the canister and impact of rock movements on the canister's integrity. The model has been verified on laboratory scale, with the exception of volume shrinkage under pressure and plastic, volumetric strain. These phenomena can be evaluated when ongoing long-term tests are interrupted during the first half of the six-year period.

A preliminary model has been developed for THM processes in unsaturated buffer which for some cases shows acceptable agreement with experimental data but still has serious shortcomings /5.4.16/. It has been applied in the international DECOVALEX project, where it has been used for calculation of PNC's "Big Ben" experiment. The model calculations indicated that the buffer material nearest the heater would dry and shrink in volume. The test shows good agreement with regard to temperatures and the decrease in water content. Since the pore volume was not measured in the tests, the shrinkage could not be evaluated. The measured total buffer pressure was higher than the calculated value /5.4-17/.

For the homogenization process, a numerical model has been sketched and applied to both pure bentonite and a mixture of bentonite and sand with the aid of the model for water-saturated buffer /5.4-18/. The model can now be applied in two-dimensional geometries.

5.4.5 Gas transport

If water enters the canister, hydrogen gas will be formed by anaerobic corrosion of iron. The potential pressurization of the canister and the gas's effect on the water movements around the canister have been studied /5.4-19/ for the purpose of clarifying which mechanisms control the transport of hydrogen gas from the canister and what the consequences of this transport are. The following conclusions were drawn:

- The long-term effect of gas generation is primarily dependent on the gas generation rate and the capacity of the bentonite to permit gas to escape.
- The quantity of gas that can be dissolved in water and is transported away by diffusion is small compared to the expected generation rate, which means that gas phase flow through the bentonite is expected to be the dominant transport mechanism.
- Once the gas has passed through the bentonite, there are enough transport pathways to enable the gas to

continue on through the rock towards the ground surface.

The most important questions to be answered are therefore what the bentonite's gas transport capacity is and whether any permanent weakening of the buffer occurs with time as a consequence of gas transport. Experiments have been conducted /5.4-20/ to determine at what pressure the bentonite opens to allow gas to escape. It was hereby found that:

- the gas passes through the clay in a few pore passages, so that there is very little drying-out caused by gas,
- the clay opens at a pressure corresponding to the hydrostatic head plus 50–90% of the swelling pressure at the densities being considered for the deep repository,
- when the pressure drops the clay closes and self-heals,
- the effect of repeated gas passage cycles is being studied in ongoing experiments.

5.4.6 Bacteria

Oxygen and sulphide are in practice the only substances that could conceivably corrode copper. Deep groundwater is reducing and thereby completely oxygen-free. Furthermore, it can be shown that the oxygen that is trapped in the repository at closure is soon consumed by the reducing substances present both in the backfill material (e.g. pyrites in bentonite) and in the rock (ferrominerals and sulphides). Sulphide occurs at low concentrations in deep groundwaters. Groundwater can contain relatively much sulphate and the bentonite contains quite a bit of soluble sulphate. Sulphate does not react spontaneously with copper, nor is sulphate reduced by the reductants that may be present in a final repository, unless this is mediated by bacteria. Sulphate-reducing bacteria occur at repository depth, and it is urgent to identify what might limit their action. One important limitation is the living conditions in the bentonite buffer.

These questions are being studied both in the laboratory (own experiments) and in-situ (cooperation with AECL in their URL). The results so far indicate that the bacteria do not survive. If this proves true, then the bentonite constitutes a barrier to microbes. A review of the importance of microbes to the long-term safety of the deep repository is provided in /5.4-21/.

5.4.7 Concrete

Concrete is a very useful material for construction and reinforcement. Possible uses in a deep repository are grouting, paving, shotcreting of walls and roof, underground constructions, plugging of tunnels and shafts etc. The use of concrete gives advantages during construction and operation of the repository, but it is necessary to evaluate its long-term properties and the importance of concrete on long-term safety. Modern cements have not been around very long, but knowledge about the minerals that are formed in cured cement and the composition of the pore water has increased considerably /5.4-22/.

5.4.8 Clarified and remaining questions

The aim of the ongoing compilation of accumulated knowledge and experience is to serve as a preliminary basis for design and selection of materials. Subsequent development steps include:

- deepening of knowledge,
- field tests,
- optimization.

The purpose of these activities is to gather detailed background data for design and selection of materials in conjunction with assessment of performance in the deep repository on candidate sites.

Properties of different bentonite materials

Theoretical explanations of the buffer's performance can be provided by a preliminary version of GMM. What remains is to describe in greater detail the bentonite's homogenization after water saturation for redistribution of water and bentonite between bentonite grains with high bentonite density and interstitial pore volumes with low bentonite density, as well as for bentonite-aggregate mixtures where the corresponding homogenization takes place in the pore volume between the ballast grains.

Temperature-induced conversion of montmorillonite to non-expanding illite can be predicted with a developed model. Since the process is slow and difficult to reproduce exactly in the laboratory environment, natural analogues also constitute suitable verification examples. Several Swedish natural analogues have been studied and others, Swedish as well as foreign, are deemed to be able to furnish supplementary evidence of the accuracy of the model for Swedish deep repository contexts.

Knowledge of the quality of different commercial bentonites has primarily been acquired via information from suppliers and own analyses of the product. In conjunction with the future choice of quality for the deep repository, it must be established which quality the supplier can be counted on to be able to deliver in the long term.

Calculation models for various functions

The function of buffer materials, with particular emphasis on rheology and transport of water and ions in deposition holes under water-saturated conditions, can be described and calculated today using developed models. These models have been verified by laboratory tests, with the sole exception of the two plastic alteration processes mentioned previously. Tests are under way for these processes.

For unsaturated conditions, i.e. with regard to processes associated with wetting of buffer and backfill, several processes and boundary conditions relating to water exchange between near-field rock and buffer and water distribution in the buffer remain to be clarified, so that the time for water saturation can be calculated with sufficient accuracy, primarily for use in evaluating buffer tests in the Äspö HRL, and secondarily for performance assessment of final disposal. Thermal, hydraulic, mechanical and chemical processes interact, and the couplings between them are crucial for the results. Calculation models for thermo-hydro-mechanical processes are studied today as coupled models, while chemical processes are regarded separately.

Gas transport

The transport mechanism for gas through a water-saturated and swollen buffer is known, while the effect of repeated gas pressure increases and gas passages requires additional laboratory testing.

5.5 THE BEDROCK

5.5.1 The role of the rock in the deep repository

The barrier functions in the KBS-3 version of a deep repository for spent nuclear fuel and other long-lived radioactive waste are described in chapter 4.

Some central properties of the bedrock, see Figure 5.5-1, are essential to guarantee the performance and long-term radiological safety of the deep repository, and there are certain obvious couplings between these properties:

- long-lasting mechanical stability
- a chemically stable environment with a groundwater which does not contribute to corrosion of the canister material or alteration of the buffer material, and which ensures low solubility and high retardation of the radioactive constituents in the waste,
- a slow and preferably constant groundwater flux which limits the transport of substances having an adverse effect on the waste and the backfill material, or on radionuclide transport.

Besides the above geoscientific properties, it is essential to minimize the risks of future intrusion by:

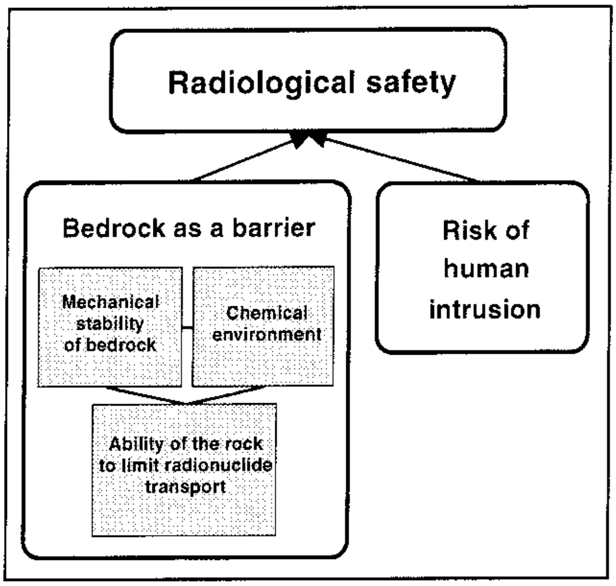


Figure 5.5-1. Properties and conditions in the bedrock that are of importance for the long-term radiological safety of a deep repository.

- avoiding rock volumes or proximity to rock volumes with a potential for exploitable metals and minerals, energy storage and energy extraction.
- locate the repository on such a site and at such a depth that the risk of intrusion by drilling (wells, prospecting, exploitation) is very small.

The Swedish crystalline basement forms a part of the Baltic Shield that stretches from the Kola Peninsula and Karelia over Finland and Sweden to Southern Norway. Most of the Swedish crystalline basement has an age of about 1,700 million years. The Baltic Shield offers a generally suitable environment for safe deep repositories, with respect to the aforementioned desirable properties of the bedrock.

5.5.2 Geoscientific data and uncertainty

Geoscience has, in a historical perspective, been a descriptive science where most interest has been devoted to the genesis and general characteristics of geological formations. The applied direction of geoscience within the field of nuclear waste management makes greater demands on quantification of geological processes and predictions of the future. The great heterogeneity of the rock means that generalizations cannot be made without describing the variability of different rock properties. Furthermore, the properties and their variation can depend on the scale on which a given problem is regarded.

The bedrock with its component parts is deterministic, i.e. it exists and has apparently measurable, predictable properties and structures on all scales. The complex structure and great variability of the bedrock mean, however, that it is impossible to rely solely on observations in a detailed quantitative description. For example, when it comes to hydraulic conductivity in different rock types, the difference in mean values between different rock types is less than the range of variation within a given rock type. For this reason, both deterministic and stochastic calculation models are utilized in geoscientific contexts.

The geological aspects are mainly associated with the performance and safety of the repository. However, geological conditions are also of importance for the actual construction of the repository. There is a connection between the design of the repository, its safety and its placement in the rock. This connection does not have to be such that construction-related difficulties always entail long-term safety risks. For example, clay-filled zones may pose an engineering problem, but not an obvious problem for future radiological safety.

The long-term safety of a deep repository is judged in a more or less site-specific safety assessment, which is based on, among other things, a general geoscientific understanding of the area in question on a scale that is relevant for the repository. The repository site, a square kilometre or so in size, must be considered in its regional-geological context. Site characterization can be both quantifying and qualitatively descriptive.

Regardless of the scale on which the safety-related conditions at the repository site are considered, the studies should be rooted in the safety assessment's need for relevant information. On general scales, judgements are largely of a qualitative character. The borderline between a qualitative judgement on a general scale and a quantitative judgement on a detailed scale is not always obvious.

Uncertainties in geoscientific data and information are associated with:

- measurement accuracy,
- the heterogeneity and anisotropy of the particular property,
- the degree of confidence for mean values and variance,
- the directional dependence and dimensionality of the investigation methods,
- the volume representativity of the investigation methods (scale),
- subjectivity in interpretation.

It is, in other words, important that investigations in rock be conducted step-by-step on different scales and iteratively, i.e. with progressive knowledge accumulation and increasingly detailed information.

5.5.3 General goals of the activities

The geoscientific programme at SKB entails broad knowledge accumulation within geology, geophysics, rock mechanics, geohydrology, geochemistry and groundwater chemistry. The programme also includes further refinement of numerical computer models, and there is a strong coupling to development of instruments and measurement methods.

The work is integrated with efforts within:

- The study site investigations.
- The Stripa Project.
- The Äspö Hard Rock Laboratory.
- Safety assessments.
- Natural analogues.
- The siting programme's general studies, feasibility studies and site investigations.
- Repository design.
- Alternative studies of waste disposal, particularly very deep holes.

The general goals of the geoscientific R&D activities at SKB are:

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

Important interim goals are thereby:

- to deepen understanding of the georelated processes which can affect the long-term safety of a deep repository, and to further refine the capability to predict such changes,

- to further refine models for calculation of groundwater flow 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 mechanics at a pace that corresponds to the need in assessments of the performance and long-term safety of the candidate sites.

- to ensure that suitable measurement technology is available for high-quality collection of such data as are required to characterize rock volumes for the construction of a deep repository.

The following sections present the state of knowledge and results obtained (1993–1995) for mechanical stability, groundwater chemistry conditions and the rock's capability to limit nuclide transport. The presentation largely follows the subdivision into siting factors in the supplement to RD&D-Programme 92 /5.5-87/. An account is also given of the state of knowledge concerning various model tools that are used in SKB's activities when it comes to hydrogeology, transport of solutes in groundwater and rock mechanics.

Certain results from Äspö are spotlighted and discussed in the following subject-by-subject account of geoscientific knowledge. A coherent status report on experience from the R&D work that has been done at the Äspö HRL up to and including the construction phase will be published in separate reports during 1995 and 1996. However, a brief summary of results is presented in the programme for the Äspö HRL (see chapter 12). Experience from site investigations and detailed characterization at Äspö is presented in section 8.2.2.

5.5.4 Structural geology and mechanical stability

A KBS-3-like repository presupposes a crystalline basement host-rock. This section deals with geological and bedrock-structural conditions as well as **the mechanical properties of the crystalline rock (movement proneness).** Figure 5.5-2 shows a schematic division into a number of **processes** that might occur, whose consequences are dependent on structural and mechanical conditions. Stability questions may be associated with natural movements or induced movements, depending on the layout of the repository. Consequences of natural movements may be dependent and/or independent of the repository.

The overall purpose of the R&D work devoted to rock stability is to:

- meet the rock-mechanics functional requirements so that the deep repository can be constructed and function stably during the deposition phase. Working conditions must be safe for both personnel and the waste to be handled.
- meet the requirement that shearing movements in the rock in the 100 000-year perspective will not damage the integrity of the canisters so that unacceptable doses occur.

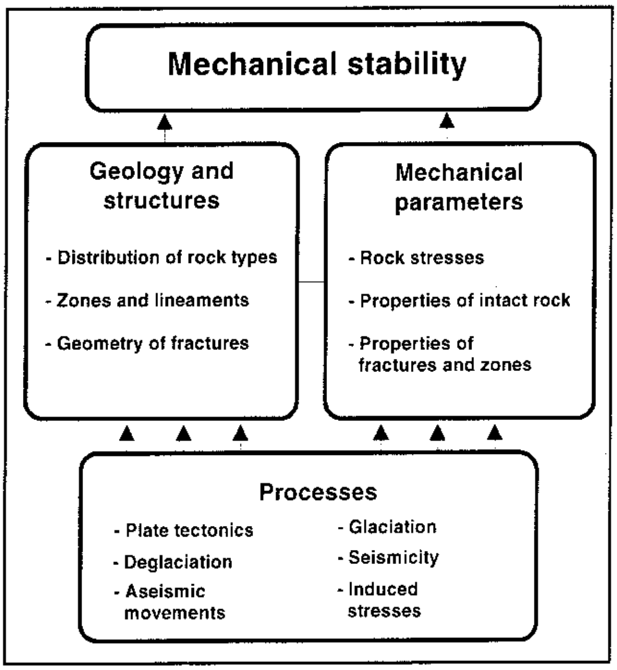


Figure 5.5-2. Processes and factors within bedrock geology, structural geology and movement proneness that may be of importance to safety.

A brief resumé of studies performed at SKB during the years 1993–1995 is given below. The reader is otherwise referred to SKB's Annual Reports /5.6-8 et al./ and reports in SKB's TR series.

Geological and bedrock-structural conditions

The bedrock in Sweden can be divided into three major units: The crystalline basement, the Caledonides and the sedimentary bedrock regions outside the Caledonian mountain chain. Most of Sweden's crystalline basement was formed by means of foldings, alterations and magma intrusions between 500 million and 2 500 million years ago. What we see today of the exposed surface of the crystalline basement lay at great depths when it was formed. Erosion has reduced the thickness of the crystalline basement by 10–30 km.

The Swedish waste management programme has focused on magmatic and highly metamorphic rock types as host rock for a repository. The rock types are often gneissic with a granitic composition. These rock types were studied during the "study site investigations" /5.5-1, 2, 3, 4, 5, 6/.

Basic rock types, particularly gabbro, have been proposed as an alternative, since they are able to offer better sorption properties and conditions for self-healing of fractures. However, the occurrence of large and homogeneous rock volumes with bodies of basic rock types is strictly limited. From a rock engineering point of view, experience from basic rock types does not differ essentially from experience from more acid bedrock types. In terms of hydraulic properties, available results show that basic rock types are generally slightly less hydraulically conductive than granites and gneisses. The range of variation within a rock volume is considerable, however. It is known from investigations in Taavinunnanen, for example, that granitic dykes can occur in gabbro bodies and that the dykes have a much higher hydraulic conductivity, relatively speaking. The relatively low thermal conductivity of basic rock types necessitates utilizing large repository volumes in order to keep temperatures from becoming too high in the near field. Another negative factor is that gabbro is often of interest for extraction of metals (e.g. nickel).

In summary, manifest difficulties are encountered in trying to find large homogeneous rock volumes with basic rock types for a deep repository. A comparison that takes into account hydrogeology, geochemistry and rock engineering does not reveal any obvious advantages for gabbro in relation to granitic rock types /5.5-7, 8, 9, 10, 11/.

The Swedish crystalline basement is intersected by certain very old super-regional zones of deformation /5.5-14/. The Southwest Scandinavian Gneiss Province is bounded on the east by a zone that extends from Skåne to Värmland, known as the Protogine Zone /5.5-12/. Within the Southwest Scandinavian Gneiss Province, the Mylonite Zone stretches from the Norwegian border in Värmland southward via Värmlandsnäs to Varberg. The Mylonite Zone is a sign of severe deformations in the bedrock about 1 000 million years ago. East of the zone the rock types are mainly gneisses of magmatic origin, while west of the zone surface rock types are most common. The Tornquist Zone is a very elongated zone within the Eurasian plate. The zone constitutes a boundary between the Baltic Shield and an expansive depressed area in the western part of continental Europe. In Skåne, the Tornquist Zone takes the form of crystalline basement horsts and intervening deep depressions with sedimentary rock. The Tornquist Zone exhibits recent movements from the Quaternary period /5.5-13/.

When it comes to structural-geological elements in a regional perspective (aside from the above), different conceptual models exist. The morphology of the bedrock surface can be compiled and serve as a basis for a structural and tectonic interpretation /5.5-15/. Several structural models have also been devised working from lineament interpretations based on, for example, the Swedish height data base and satellite images /5.5-14, 60, 61, 105, 106/. Some models suggest, for example, that Sweden should be divided into a few super-regional blocks separated by elongated shear zones /5.5-16, 107/.

Rock-mechanics experience from underground facilities says that siting of the deep repository in a super-regional deformation structure within the Baltic Shield does not have to be ruled out. A deformation zone may contain rock volumes with suitable conditions. This is clarified by site investigations /5.5-80/.

Observations from shield areas around the world provide evidence of the fact that fractures and fracture zones, from a detailed scale to the regional perspective, occur with discrete directional concentrations. The displacements as well as the length and width of the deformation structures, often indicate fractal dimensionality. Intensive development work is under way to increase our knowledge of the bedrock's fracturing /5.5-62, 63, 64/.

Conventional geological and geophysical investigation methods mainly map steeply dipping structures and their appearance on the ground surface. It is, however, desirable to further develop methods for mapping sub-horizontal structures in the bedrock. Further development of seismic reflection investigation and interpretation methods is being pursued within SKB /5.5-67/.

Mechanical properties of the crystalline basement rock

Due to the low temperatures and pressures that prevail in the upper part of the lithosphere, stresses both during the construction phase and on a geological time scale can result in brittle fracturing. Brittle mechanical behaviour may include formation of fractures and zones or propagation and reactivation of already existing fracture zones. Brittle-mechanics deformation can occur on all scales.

The mechanical theories for the behaviour of intact crystalline bedrock are relatively well-developed and tested on a laboratory scale /5.5-16, 17, 18, 19/. Once a fracture or a zone has formed, future movements are controlled more by friction than the properties of the intact rock. Various empirically founded parameters are used to describe the mechanical properties of fractures /5.5-20/. Important properties of the individual fracture are, for example, proportion of contact surface, roughness of the fracture plane, grinding caused by the movement, fracture-filling material, distribution and size of the normal stress with respect to liquid pressure, etc.

Mechanical properties of major discontinuities such as fracture zones are relatively unknown. A compilation of existing information has been done /5.5-21/ based on observations from limited sections of zones or from large-scale load changes, mainly in the mining industry. The knowledge is essential from both a rock-engineering and safety-assessment point of view. The study finds, among other things, that the assumption of flat zone structures with uniform function in calculation models is dubious, and that the geometry and morphology of zones ought to be represented more realistically.

It is essential to be aware of couplings between mechanical, thermal and hydraulic properties in a deep repository. The thermo-mechanical coupling has been examined relatively thoroughly, while the constitutive relationship between mechanical and hydraulic properties in fractures is under development /5.5-22, 23, 24/.

Geodynamic and mechanical processes

Stability in a deep repository is affected by both geological processes and induced movements due to the geological configuration of the repository /5.5-30/.

Tectonics is a collective term for the deformation of the earth's crust and the structural forms arising as a result. The term covers deformation and structural forms from a millimetre to a kilometre scale.

A review has been made of the various tectonic regimes in the Baltic Shield during the last 1 200 million years /5.5-25/, with a focus on different dominant stress fields associated with them. The surface of the present-day crystalline basement exhibits fracture patterns and heterogeneities that have been shaped and developed over many hundreds of millions of years. It is worth pointing out in this context that sedimentary rock strata covered large parts of Sweden with a thickness of up to 3 km during the Devonian-Triassic periods. Fracture patterns with four to five dominant fracture directions usually occur on the basement surface. These are the result of shear and/or tension fractures. Since then the stress field has varied in direction and size through the ages. Analyses show that super-regional stresses have caused the fractured bedrock to be deformed along already existing fractures and zones. Any active loads during the coming 100 000-year period will thus probably reactivate old zones and fractures. The present-day tectonic regime with a relatively passive response to the Mid-Atlantic ocean floor spreading is a calm period (anorogenic) for the Baltic Shield.

Figure 5.5-3 shows the largest principal stress from undisturbed measurements at levels of 400–1000 m /5.5-31/. The stress picture, with a principal compression in the northwesterly direction, verifies the effect of the Mid-Atlantic ocean floor spreading on the rock stresses in Sweden. Similar conclusions are also obtained from focal plane analyses of seismic events in the Baltic Shield /5.5-48, 49, 50/.

This present-day super-regional tectonic state, which was initiated about 50 million years ago when the Atlantic began to form, is expected to persist during the next 100 000 years /5.5-26/. However, as far as the stress situation is concerned, it is necessary to take into account expected changes caused by glaciations and deglaciations /5.5-27, 28, 29/.

Dating techniques have been developed in recent years to determine when the most recent significant movement has occurred in individual fractures or zones. Examples of dating methods are paleomagnetism, radiometric dating, electron spin resonance (ESR) and petrographic evolutionary history. A comparative study of different methods has been performed on fracture-filling material from the Äspö HRL /5.5-30/. The study has yielded valuable experience of sampling technique, measuring accuracies and representative time resolution. In the project in question, the K-Ar analyses were considered to be the most reliable, and the interpretations suggest that the most recent significant fault movements took place more than 300 million years ago. In the near future, different dating methods for the activity of fractures can be expected to become valuable tools for shedding light on the stability of the rock.

Glaciations can affect the safety of a deep repository. In cooperation with Teollisuuden Voima OY (TVO), SKB has surveyed the international state of knowledge concerning future ice ages and when they can be expected, and what changes in the geosphere may then occur /5.5-27/. Interesting questions in this respect are mechanical aspects, groundwater hydraulics and groundwater chemistry.

SKB has had a numerical glaciation model of Scandinavia developed. The three-dimensional model is time-dependent and includes thermomechanical coupling /5.5-32/, see Figure 5.5-4. The glaciation model is driven by changes in the air temperature and predicts the state of the ice mass on a predetermined topography and how the bedrock is affected mechanically. A simple groundwater flow model has also been coupled to the model /5.5-33/ which makes it possible to perform general and regional groundwater simulations during both the glaciation and deglaciation phase, see further section 5.5.9.

Previous stability calculations with an ice load and where changes in water pressure have been taken into consideration show that any movements are absorbed in already existing zones. In a calculation example based on data from the Finnsjön study site, the movement amounts to about 0.05 m in a normal case. In extreme situations with low in-situ stresses, displacements of about 0.5 m could be obtained in zones. It is assumed that canisters will not be emplaced in such zones /5.5-34, 35/.

The aforementioned glaciation model will make it possible to shed further light on the prospects of reactivation in zones and possible hydraulic fracturing of the rock due to high water pressures during a glaciation.

Land uplift following the most recent glaciation is a conspicuous geological process in Sweden, especially along the coast of Västerbotten in northern Sweden where the land is rising relative to the Bothnian Sea at a rate of about 9 mm per year.

At the beginning of the melting phase of the Weichsel Glacial Stage (about 18 000 years ago), the surface of the world ocean was about 120 m lower than today and the surface of the ocean started to rise (eustatic rise). The warmer climate brought large flows of meltwater to the oceans. As the continental ice sheets gradually melted, the pressure of the ice on the earth's crust decreased and a process of land uplift began (postglacial isostatic rebound). The previous highest coastline and the evolutionary stages of the Baltic Sea are shown in Figure 5.5-5.

Viewed in an overall perspective, land uplift has been and is a relatively continuous process in space and time. Within the area covered by a deep repository, differences in the rate of land uplift are completely negligible for the mechanical stability of the repository. Based on detailed Quaternary geological studies, however, it is possible to analyze shoreline displacements that could be a sign of local postglacial movements in existing fractures or zones. SKB is funding some such studies, the results of which are expected to be published within the coming year /5.5-37, 38/.

Our knowledge of the process of land uplift can be expected to be deepened by data from the recently established GPS (Global Positioning System) network. Approximately 20 fixed measurement stations were recently commissioned in Sweden. Fine adjustment, calibration and analysis of measurement accuracy are currently under way /5.5-39/.

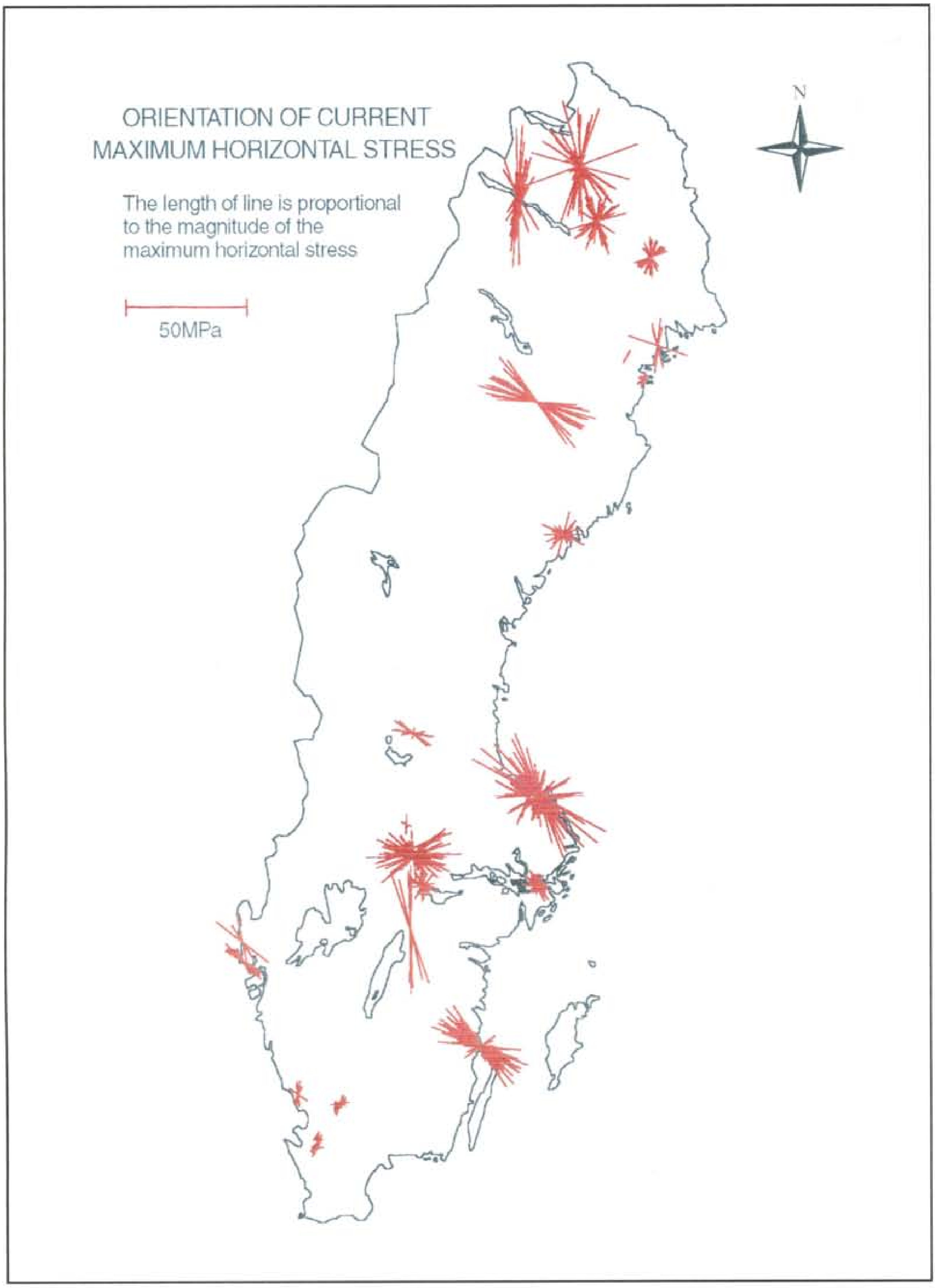


Figure 5.5-3. Rock stress measurements at levels of 400–1000 m below the surface in Sweden (from /5.5-31/).

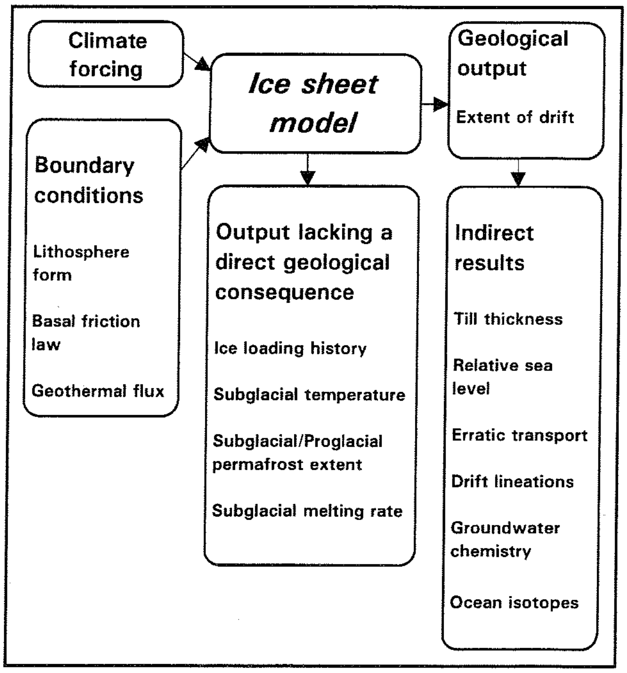


Figure 5.5-4. *Input data and results from SKB's glaciation model (after /5.5-32/).*

Neotectonic and postglacial movements are objects of thorough studies. By "neotectonic movements" is usually meant displacements that have taken or are taking place during the current tectonic regime, i.e. during the time when the Atlantic Ocean has existed. Movements after the most recent deglaciation are called postglacial. It is essential to ascertain whether such movements can lead to new fracturing or seriously alter the hydrogeological or chemical conditions for a deep repository.

Postglacial faults in the northern part of the Baltic Shield have been known and have been the subject of investigations for about 20 years /5.5-40, 41/. During the period 1986–1992, SKB carried out a comprehensive programme of geoscientific research in the Lansjärv area, about 150 km north of Luleå. The field studies were concentrated to certain sections of the faults and also included test pit investigations /5.5-42/. An excursion was arranged to the area with international experts in June 1991. Summarizing reports /5.5-43, 44/, including expert comments, conclude the following:

- The postglacial faults are primarily reactivations of older dominant zones, but the occurrence of some few new fractures cannot be ruled out.

- The causes of the postglacial movements are probably a combination of relatively rapid changes of the vertical loads (associated with deglaciation) and horizontal compression from the Mid-Atlantic Ridge related to continental drift. Landslides and violently disturbed soil strata (seismites) are clear indications of brief instability.

- No clear evidence is available today to suggest that the postglacial faults are still active. Thus, these fault movements were probably caused by specific stresses in conjunction with glaciation and deglaciation and have little or no connection with the present-day stress situation in northern Sweden.

Some researchers claim that postglacial structural elements not only occur in northern Sweden but are common all over the country /5.5-45/. Rock movements are said to have taken place in conjunction with the most recent deglaciation. It is asserted that indications of this can be found in redistributions of boulder accumulations at the surface of the crystalline basement. There is uncertainty as to the depth of the structures, however.

The previously mentioned projects intended to map displacements of old shorelines, as well as various dating projects and use of GPS technology, will shed further light on neotectonic and postglacial phenomena.

Seismicity can be said to be a sign of active tectonics in a geological time perspective. More than 95% of all earthquakes take place at the boundaries of the continental plates. Approximately a million earthquakes with a magnitude of more than 2 on the Richter scale take place annually in the world. Of these, about 10 quakes occur in Sweden. In other words, our country is a seismically inactive area. The biggest earthquakes in Sweden reach a maximum magnitude of about 5.

The Swedish earthquakes are mainly concentrated in two areas. One area extends from Lake Vänern down to the west coast. The other area follows the coast along the Gulf of Bothnia towards Tornedalen and northern Lapland. The majority of the Swedish quakes take place deep down in the bedrock, so that the epicentre of the quake lies about 10–20 km below the ground surface and the movements there are small. Calculations indicate displacement sums of about 10 mm at magnitudes of 5 /5.5-46/. The movements take place as a reactivation in an existing fault structure and within a radius of about 900 m down at the epicentre of the earthquake.

The mechanisms that control the quakes within the continental plates, for instance within the Baltic Shield, are relatively poorly understood /5.5-47/. In Sweden, the discussion concerns whether the earthquakes are controlled by the processes of plate tectonics, the ongoing process of land uplift, or a combination of the two mechanisms /5.5-47/. Seismic measurements /5.5-48, 49, 50/ show that most of the stress that triggers an earthquake has a compression direction of N60W, which is roughly perpendicular to the continental movement from the Mid-Atlantic Ridge. The worldwide database from the World Stress Map Project also exhibits good agreement between crustal plate tectonics and the largest horizontal principal stresses, i.e. compressions. Certain deviations in the stress field occur in the Baltic Shield, especially

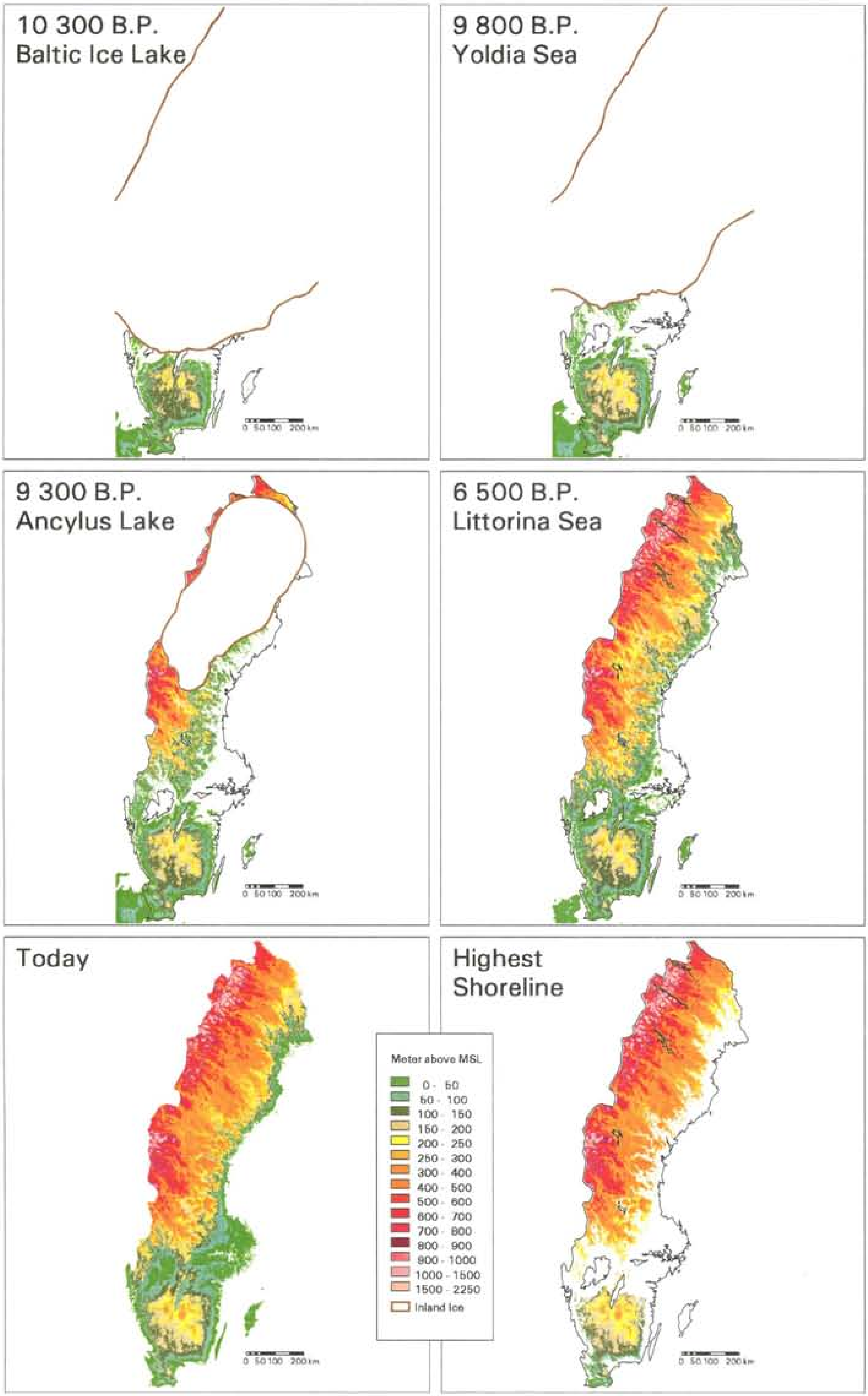


Figure 5.5-5. The different evolutionary stages of the Baltic Sea after the most recent glaciation and a picture of the previous highest coastline. The pictures were produced from SKB's Geographic Information System.

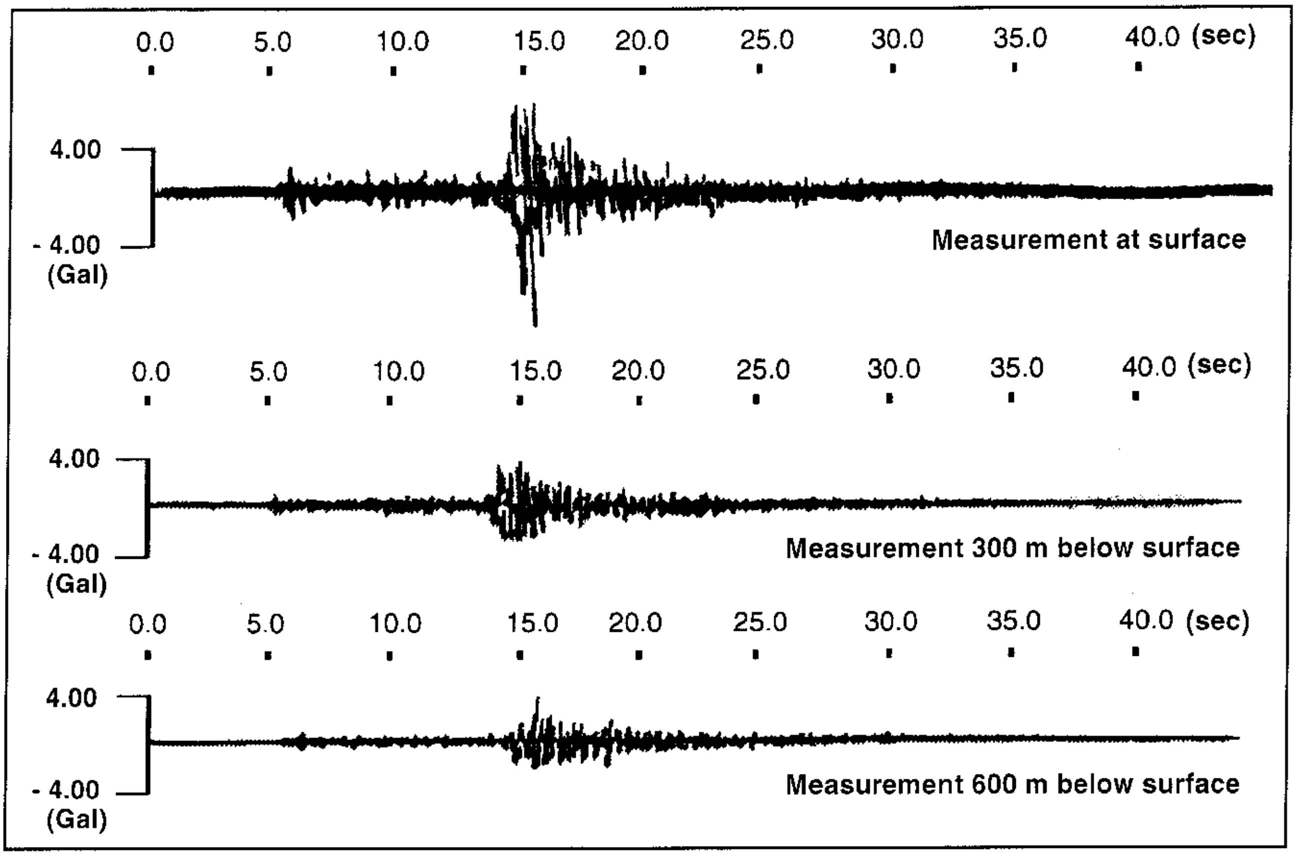


Figure 5.5-6. *An example of the decline of the vibrations with depth. The earthquake took place on 27 January 1991 with a magnitude of 4.6, near the Kamaishi Research Mine in Japan. The seismograms show registrations of the quake made at different depths.*

near the ground surface, which can be explained as effects of the processes of glaciation and postglacial land uplift. A separate seismo-tectonic compilation carried out on behalf of SKB particularly emphasizes the influence of land uplift /5.5-51/.

As far as the effects of earthquakes on underground facilities are concerned, the mechanical stresses on such facilities are in general less than for facilities on the surface of the ground. Many observations, above all from mines, exist to support this contention /5.5-52, 53/. SKB has recently become involved in a Japanese project in the Kamaishi research mine in order to study the seismic and construction-related effects of earthquakes. Kamaishi is situated in a seismically active area with frequent quakes with magnitudes of 5-7. Figure 5.5-6 shows results from seismographs in Kamaishi at different levels below the ground surface. The free energy (the amplitude of the acceleration) is greatest at the ground surface and declines with depth.

In addition to the mechanical effects of seismicity, the geohydraulic and hydrochemical effects are also being studied both at the Kamaishi Mine /5.5-81/ and in other international projects /5.5-54, 55/.

In summary, Sweden is located in a seismically inactive region. There are no signs today that this will change on a 100 000-year time scale, except for the changes in stress conditions that might be caused by a future ice age.

Induced loads implies rock stresses that arise as a consequence of the disturbance caused by the very existence of a deep repository or its construction. Induced stresses cannot be avoided, as they occur necessarily as a consequence of the construction of the deep repository. The questions are where, when and how displacements and stresses will occur. The induced loads, and movements associated therewith, tend to be local and to be initiated in the immediate vicinity of the deep repository.

The creation of cavities in the rock mass causes stress redistribution and stress concentrations. The short-term stress picture around the repository depends on, besides the properties of the rock mass (initial stresses, fracturing and strength), the extraction method (careful or conventional drilling and blasting) and the geometric shape ("rectangular" or "circular" tunnel cross-section). The rock excavation works (drilling, blasting) causes some fracturing of the rock near the tunnels. A high-strength and fracture-poor rock promotes the risk of rock burst.

Heat generation in the stored waste will give rise to an elevated temperature in the vicinity of the deep repository up to about 80°C locally, which then slowly declines. Locally around the deposition tunnels, the thermal load causes increased tangential stresses. On cooling, sections with low stresses can arise with a risk of local instability (rock outfall). Local temperature differences during the heating phase (gradient problems) can lead to increased fracture formation. The time perspective is crucial for the importance of thermo-induced stresses.

In the long run, the rock will strive by means of creep to even out stress concentrations. An ongoing very slow fracturing of the rock near the tunnels can be expected.

The induced stresses are known by experience from similar underground excavations, such as mines and hydropower plants. There are good opportunities to calculate the mean occurrence of such stresses in the facility with the aid of well-documented models. The greatest difficulty lies in quantifying different types of long-term effects. The local variations that occur can be handled, provided quality control is effective during the building phase /5.5-57/.

For further comments and description of the state of knowledge when it comes to the repository's design, layout and construction methods, see chapter 8.

Activities in relation to goals in RD&D-Programme 92

A number of goals were set up in RD&D-Programme 92 for the R&D work on bedrock stability during the period 1993–1998. Here is a brief account of the situation in relation to the goals:

- Carry out a seismotectonic compilation for the Baltic Shield. – A separate report has been written and presented at an expert seminar /5.5-51/.

- Carry out a pedagogical compilation that shows the brittle-tectonic history of the Baltic Shield with the load situations that have taken place. – A separate report has been compiled, and it was also presented at an expert seminar /5.5-25/.

- Make a detailed study of and map the land uplift in different regions of Sweden. – Compilation work is under way following field work. The intention is to initiate the activities based on the results of GPS measurements.

- Carry out detailed studies of former shorelines in some regions of Sweden to clarify postglacial tectonics. – Compilation work is under way following completed field work.

- Increase knowledge of the brittle-tectonic fragmentation of the crystalline bedrock. – Certain studies have been conducted within the framework of the Äspö Project /5.5-19, 59/. Literature studies are under way and rough data simulations have been carried out.

- Further refine methods for dating fracture zone movements. – A separate report on results has been compiled /5.5-30/. The work has been presented at several international conferences. Scientific papers have been published /5.5-58/. Additional measures are planned in Laxemar, borehole KLX 02 and in the Äspö HRL.

- Increase understanding of the representativity of rock stress measurements. – A database of reliable stress measurement results has been compiled within the framework of the siting programme's general studies. Programme work is under way to initiate a separate study.

5.5.5 Groundwater chemistry

The importance of groundwater chemistry and its coupling to the mineralogical composition of the rock and to hydrological conditions are examined in this section.

The goal of the efforts within SKB's hydrochemical and geochemical research activities is to be able to ensure a favourable chemical environment in the repository by

- clarifying geohydrochemical conditions of importance for the repository's long-term performance,
- clarifying the processes that can affect the geohydrochemical conditions over long periods of time,
- supporting the geohydrological groundwater flow models via hydrochemical observations in the bedrock.

A description of the state of knowledge within the area is given in RD&D-Programme 92. For a full account, see this presentation and SKB's Annual Reports. New knowledge beyond this has been obtained within the following areas:

Groundwater types with different origins

Both fresh and saline waters in the rock's fracture systems are mixtures of several water types with different origins /5.5-145/. Multivariate analysis has been used to identify and group the water samples and to identify end members. Groundwater from Äspö and the deep borehole KLX 02 at Laxemar (see section 5.5.7) can be described as a mixture of modern infiltrated fresh water, modern Baltic Sea water and very old saline water, see Figure 5.5-7. The figure shows that water samples taken in various contexts are tightly grouped. For example, the waters sampled during the pre-investigation phase are a mixture between "saline water" and "glacial water". The water sampled in the tunnel has a larger content of "Baltic Sea water".

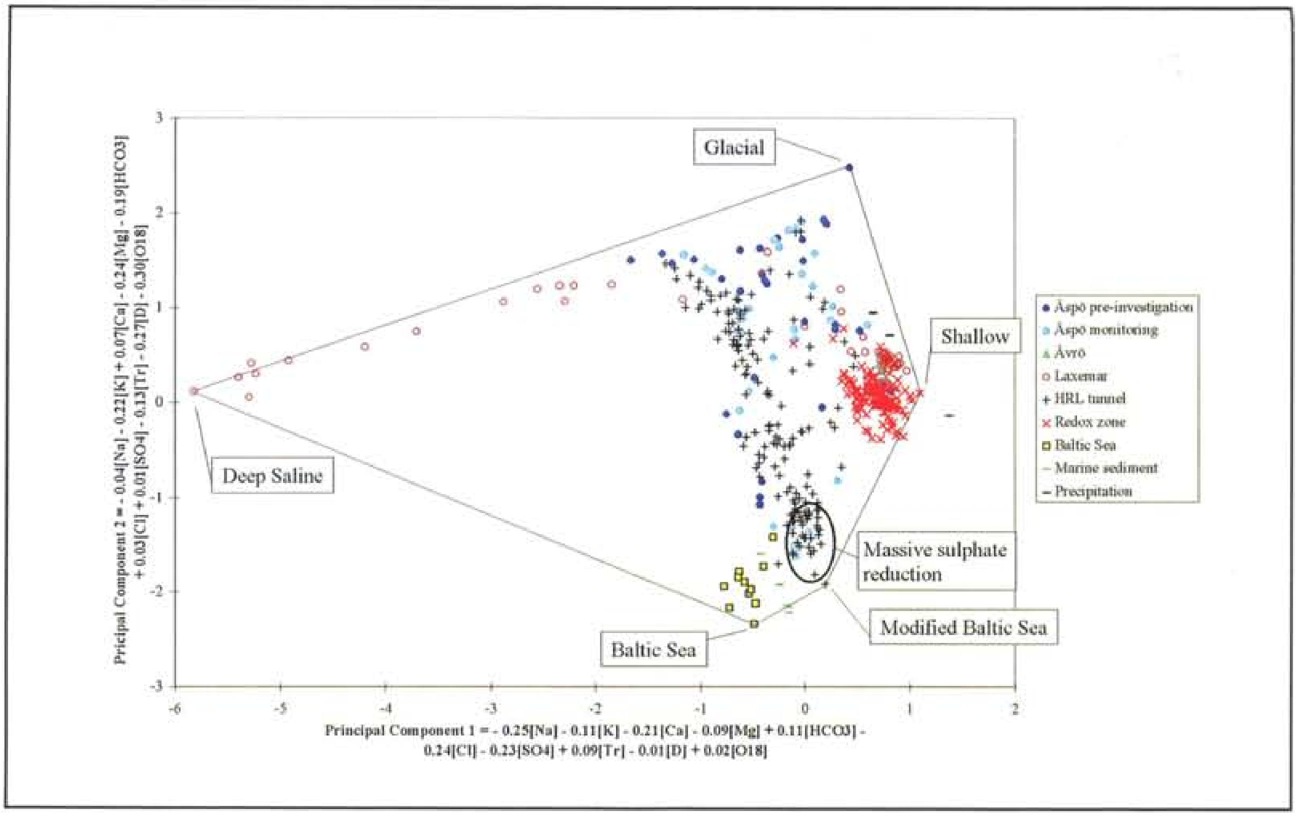


Figure 5.5-7. *Results of principal component analysis of Äspö data. The different selected end members are indicated. The area with massive sulphate reduction is marked. The importance of the different elements is shown in the equations for the axes.*

By making use of the special character of the different water types (i.e. content of certain substances and isotope ratios), it is possible with multivariate technique to determine the proportions of constituent original waters in a water sample with an estimated uncertainty of about ±10% /5.5-146/. This multivariate-based evaluation technique will be further refined in preparation for future site investigations. The goal is to be able to identify favourable and unfavourable geochemical conditions with respect to the function of the different barriers at an early stage of a site characterization.

Groundwater residence time

RD&D-Programme 92 describes how the water's chemical composition, in combination with different stable and radiogenic isotopes, can be used to determine the groundwater's residence time. The origin of the various water types can also be used to determine the water's turnover time. Lately, the combination of the strontium-87, carbon-13 and oxygen-18 conditions in groundwater and fracture-filling minerals has been very useful.

An aggregate interpretation of the data that describe groundwater turnover on Äspö provides the following picture: Down to a depth of about 500 m, the groundwater on Äspö has been affected by conditions prevailing since the most recent ice age. It is possible to trace the earlier stages in the evolution of the Baltic sea in the water composition.

At even deeper levels, the water is insignificantly affected by postglacial events and therefore to be regarded as stagnant in a 10 000-year perspective /5.5-147/.

The very saline water that has been found at a depth of about 1000 m in KLX 02 can be regarded as very old and stagnant /5.5-145/. To further clarify this relationship, more dating methods will be applied, including Cl-36 analysis and noble gas analyses. The results from both Äspö and Laxemar show that the chemical conditions at great depth, 1000 m, have presumably been stable on a time scale of 100 000 years or longer.

To be able to evaluate the groundwater flux in the environs of Äspö in detail, the evolution of the Baltic Sea has been charted /5.5-148/. The changes that can be used to trace previously prevailing groundwater conditions are mainly changes in the isotope ratios of oxygen-18 and deuterium and carbon dioxide pressure. The values of these parameters and others have been listed for the conditions that prevailed before, during and after the glaciation.

This investigation work is an important link to the paleogeohydrological studies and modellings that are expected to go on for many years. The work is aimed at providing a global picture of the variations in groundwater flow and composition that could conceivably occur in conjunction with glaciations, see section 5.5.8.

Redox conditions

In order that the copper canister should not corrode, oxygen-free (anoxic) conditions are required. Changing from an oxygen-rich (oxidizing) to an oxygen-free (reducing) environment requires reactions and processes that consume the oxygen (redox buffer). This usually takes place in the most superficial rock at a depth of a few tens of metres. It has been assumed that the redox buffer capacity, as well as the redox buffer level (Eh value), is dominated by reducing (iron-containing) minerals in the bedrock. (See RD&D-Programme 92 for a more thorough description.)

However, *bacterial oxygen reduction* has been found to be most important when it comes to consuming dissolved oxygen in infiltrated surface water /5.5-149/. The water's content of organic matter has been transformed to hydrogen carbonate and the oxygen has been reduced via bacterial action. If the quantity of organic matter exceeds about 10 mg/l in the infiltrating water, all dissolved oxygen will be consumed near the ground surface. At a large surplus of organic matter, the carbon oxidation proceeds via bacterial reduction of iron(III) minerals and sulphate even under oxygen-free conditions.

Under forced water flux as well, which can be caused by inflow to various parts of the repository, dissolved oxygen will be reduced near the ground surface and will thus not affect the reducing conditions that prevail in the bedrock prior to repository construction.

The occurrence of *bacterial sulphate reduction* has been detected in the tunnel section between Hålö and Äspö. An integrated interpretation of hydrological, chemical and biological data shows that it is probably due to the presence of about 40% or more sediment water (see Figure 5.5-7) that this process has occurred on a large scale /5.5-150/. Chloride concentrations in the range 4000–6000 mg/l and TOC concentrations >10 mg/l correlate positively with high hydrogen carbonate concentrations, low sulphate concentrations and the presence of sulphate-reducing bacteria. The product of this process is sulphide, which can in this way be generated in large quantities, about 100 mg/l, locally. Sea sediment with high organic content can therefore constitute a condition for the occurrence of sulphate reduction, cf. 5.4.6 and 5.6.2.

Water-mineral reactions

The composition of the groundwater is mainly determined by five different processes:

- Equilibrium with minerals on fractures and in the rock matrix.
- Surface reactions: ion exchange and sorption.
- Dissolution and precipitation of minerals.
- Bacterial activity e.g. of sulphate- and iron-reducing bacteria.
- Mixing of waters with different origins.

In order to be able to ascertain the effects of water-mineral reactions, the mix of different types of water must first be determined. With multivariate technique it is possible, as described previously, to quantitatively derive the proportion of different original waters in a water sample. The mix calculations are primarily based on the content of so-called conservative constituents, mainly chloride and stable isotopes. Non-conservative constituents then exhibit a deviation in concentration compared with the mix calculation. The deviation is the result of water-mineral reactions.

The interaction between water and minerals has, over a shorter or longer period of time, led to the composition of the particular original water. In the deep saline water from KLX 02, oxygen and hydrogen isotope data deviate from the linear relationship of the Meteoric Water Line, which applies to all meteoric waters. This is a strong indication of a water-mineral interaction that has gone on for a very long time, perhaps up to millions of years. The extremely high salinity may also derive from a very long residence time.

At prevailing groundwater temperatures, the *new formation of stable secondary minerals* is very slow and equilibrium is only achieved between water and reactive minerals such as calcite. Traditional geochemical equilibrium modelling can therefore not be expected to give a correct prediction of the groundwater's chemistry /5.5-154/, since the water's chemical composition is not in equilibrium with the different mineral phases. This is verified by the pilot test that was performed at Äspö of sampling water from low-conductivity blocks and analyzing main and trace elements /5.5-151/. Table 5.5-1 shows a comparison between samples taken in low- and high-conductivity boreholes in different rock types /5.5-152/.

The fact that there are no great differences between the composition of samples in low- versus high-conductivity blocks and in diorite versus greenstone indicates that the mineral-water reactions on a micro-scale are of subordinate importance compared with mixing and other process that take place on a macro-scale.

Ion exchange equilibria with clay minerals in fractures and fracture zones have a notable effect on water chemistry, above all on the concentrations and proportions between Na, Ca, Sr, Rb and Cs /5.5-147, 153, 154, 155/. The kinetics of the ion exchange reactions are such that the reactions can be studied in the lab and in the field and are affected by changes in groundwater conditions during tunnel construction.

Table 5.5-1. Chemical composition of groundwaters in Äspö diorite and greenstone. The concentrations are give in mg/l, * in µg/l.

Substance	ÄSPÖ DIORITE High conductivity	ÄSPÖ DIORITE Low conductivity	GREENSTONE Low conductivity
flow ml/min	600	30	2.5
Na	2030	1990	2080
Ca	1700	1680	1720
Mg	77	72	68
HCO_3	40	34	24
Cl	6400	6200	6600
Br	34	38	45
SO_4	435	444	450
Sr	26	27	30
Fe	0.44	0.32	0.05
Mo*	50	71	79
U*	0.6	0.07	0.53
La*	0.7	0.56	0.76

The *dissolution* of easily weathered minerals such as calcite, Ca-plagioclase and biotite can also be expected to contribute to changes in the water's composition in a shorter time perspective. These reactions preferably take place near the ground surface where the water's pH value is low and the weathering takes place at a rate that can be studied in the lab /5.5-156/.

Bacterial processes can, as mentioned previously, influence the composition of the water. This, however, requires a good supply of organic matter or another substrate. Bacterial processes that have influenced the water chemistry on Äspö are:

- Reduction of dissolved oxygen that has led to an increase in the hydrogen carbonate concentration and a decrease in the concentration of organic matter /5.5-155/.

- Reduction of iron(III) minerals and accompanying increase in carbonate and iron concentration /5.5-155/.

- Reduction of sulphate and increase in carbonate and sulphide concentrations /5.5-150/.

These rapid processes can lead to new formation of minerals such as calcite, magnetite and pyrite. Calcite and magnetite have been observed in fresh rust (iron hydroxide) precipitates /5.5-153/.

The *fracture-filling mineral composition* bears traces of former hydrochemical conditions. This can be utilized in two different ways: 1. to trace previous groundwater flow patterns 2. to evaluate the transport of radionuclide-like substances. Research aimed at clarifying former flow situations is being conducted within the hydrogeological programme, see section 5.5.8.

The distribution of uranium, thorium and rare earth metals (radionuclide analogues) in fracture-filling minerals and water is described in RD&D-Programme 92, Detailed R&D-programme 1993–1998. Since then the binding to the three most common minerals has been studied by sequential leaching of clay minerals, iron hydroxides and calcite /5.5-153/. These three mineral types constitute the largest contact surface with water in fractures and fracture zones on Äspö.

Ion exchange with clay minerals is the most important retention factor for Cs and Rb, and probably also for Sr (which is also incorporated in calcite). Th and rare earth metals are enriched in clay minerals, but also in iron oxides and calcite. Ba and Ra are found in iron oxide and calcite precipitates.

Incorporation in calcite mineral can, in a long time perspective, be regarded as a completely reversible process, since calcite precipitation and dissolution are fast reactions that are influenced by carbon dioxide pressure, pH and temperature conditions in the rock.

Cs uptake on clay minerals consists of a fast and a slow sorption, previously called reversible and non-reversible, respectively. In a long time perspective, however, it is probable that the slow sorption is also reversible, e.g. the Cs concentrations in water of very high salinity in the borehole KLX 02 are correlated with the Na concentrations, which are controlled via reversible ion exchange /5.5-154/. Rb and Ba are also sorbed reversibly on clay minerals.

Continued work will be focused on better clarifying reversible/non-reversible sorption and its importance on the time scale of the repository.

Activities in relation to goals in RD&D-Programme 92

A systematic processing and sorting of all chemistry data in SKB's database – GEOTAB – into quality classes and type classes has been carried out /5.5-157/. Waters that are representative of the environment where they have been sampled are considered to be of high quality. The quality classification has been done in cooperation with TVO and has thereby been able to utilize a larger database. The type classification is an interpretation of existing data that has been systematically applied to the evaluation of conditions at both Äspö and Laxemar /5.5-145, 146/.

Multivariate analysis is used in connection with the type classification. This increases the reliability of the interpretations in that all evaluations are done in the same way. The quality classification cannot be performed in the same strict manner, however. A certain measure of expert judgement is necessary /5.5-157/.

The distribution of trace elements (radionuclide analogues) in fracture-filling minerals has been determined for the most commonly occurring minerals /5.5-153/. A traditional K_d determination with the same mineral samples is planned.

Different isotope methods have been tested in analysis of fracture-filling minerals, drill cores and water for determination of the groundwater's residence time and history /5.5-147, 153, 154, 155/. Their usefulness is strongly dependent on the complexity and variations in the investigated material. An example of this is the carbon-14 determinations that have been done in the redox zone in the Äspö HRL. Carbon-14 activity is highest in water from the deepest borehole and lowest in the shallowest. Counted in age, the analyses give 3,365 years for a depth of 15 m and 250 years for a depth of 70 m. Even though the C-14 analyses are misleading as a dating method in this case, they have been useful for quantitatively establishing the effects of bacterial oxygen reduction.

The presence of much saline and presumably stagnant water at great depth both on Äspö and at Laxemar is significant for paleohydrogeological and regional groundwater conditions. The hydrochemical conditions are the subject of extensive analyses in this respect /5.5-145, 146, 147, 154/. The work will continue within the framework of the paleohydrogeological programme /5.5-158/.

The effects of acidification on the rock /5.5-159/ and on the water /5.5-110/ have been investigated. In summary it can be concluded that acidification caused by the combustion of fossil fuels does not have any appreciable effect on a deep repository.

Water in low-conductivity blocks in the rock has, contrary to expectation, been found to have a composition similar to that of water in more conductive blocks. Variations in rock type are also of very little importance for chemical composition /5.5-151, 152/.

The size of the wet surface in the flow paths from the ground surface down to the redox zone was expected to be of importance for the rate of oxygen reduction. The redox experiment showed, however, that it is bacterial activity that consumes the dissolved oxygen. Since the reduction rate is then dependent on the supply of oxidizable organic matter, the wet surface is irrelevant in this context /5.5-149/.

5.5.6 Ability of the rock to limit nuclide transport

The rock's ability to limit the transport of various substances is a composite property, related to groundwater flows and flow paths. These are in turn determined by the rock's hydraulic conductivity, the degree of inhomogeneity, the topography of the area, water balance, etc. If water transport takes place without the influence of retentive mechanisms, it is regarded as non-reactive. The non-reactive transport process includes advection (convection), kinematic dispersion and molecular diffusion.

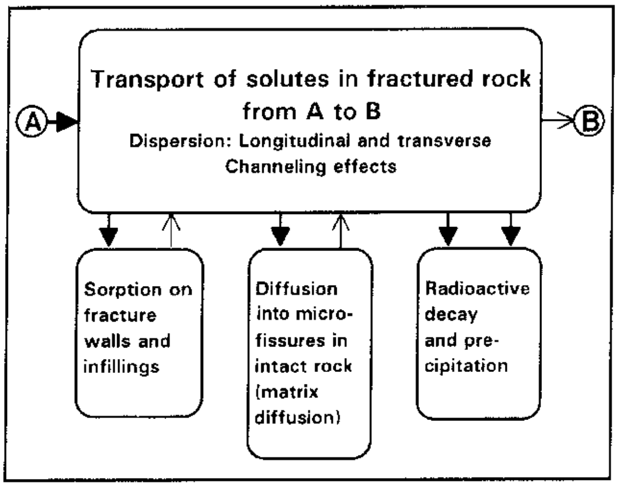


Figure 5.5-8. Schematic illustration showing the retention processes on nuclide transport in the bedrock (after /5.5-82/).

The aggregate retardant effect on nuclide transport is determined by the diffusion into intact rock (matrix diffusion) and by the bedrock's sorption properties. These are in turn determined by the groundwater chemistry, the mineralogical composition of the rock, microfissures in the rock, the character of the fracture pattern and the availability of sorption surfaces, see Figure 5.5-8.

The overall purpose of the RD&D work devoted to the ability of the rock to limit groundwater flow and pollutant transport is to:

- meet the hydrogeological functional requirements so that the deep repository can be constructed and function stably during the deposition phase.
- be able to describe and quantify how the groundwater flows are distributed in the bedrock,
- quantify processes of importance for solute transport,
- describe the rock's retention properties,
- be able to identify bedrock with good retention properties.

A brief resumé of the general state of knowledge with references to studies conducted at SKB during the period 1993–1995 is given below. The reader is otherwise referred to SKB's Annual Reports and reports in the TR series.

Groundwater flow and advection

Many studies have shown that Darcy's law is applicable within a wide range of hydraulic gradients. Darcy's proportionality constant, which is normally called hydraulic conductivity, is related to the density and viscosity of the liquid, and a form factor that is dependent on the geometry of the liquid passage. The most common model for calculating and describing flow in fractured rock, which can be presumed to be laminar, is based on Darcy's law.

During the past decade, theoretical and experimental studies have contributed towards a gradual improvement in our understanding of the flow process in fractured bedrock. Use of geostatistical methods for processing of hydraulic properties and fractal characterization of fracture lengths, fracture apertures, etc., as well as knowledge of two-phase flow and multi-porosity systems, has increased. Geophysical methods have been developed into valuable complements to hydraulic tests, thereby creating a better conceptualization for the modelling work. Greater interest is being devoted to processes coupled to the groundwater flow, i.e. thermo-hydro-mechanical relationships and hydrochemical dependencies.

Hydraulic gradients that control the groundwater flows in the bedrock have been found by experience to exhibit relatively little variation. Ordinary values are 0.1–1%, and the gradient can increase slightly where large topographical differences exist.

Credible boundary conditions are essential in groundwater modelling. The three-dimensional models are becoming more common, which means that increasingly complex flow geometries are being analyzed. The greatest interest in groundwater modelling for a repository is being devoted to levels about 500 m below the ground surface. Based on classical theory for groundwater flow, the flow at a deep repository can be expected to be influenced by regional groundwater systems. The local topography at a repository area with small height differences probably has very little to do with the flow conditions at the repository level. The flow at the deep repository is then determined by regional gradients.

Using the Swedish height database, wavelengths and amplitudes in the Swedish topography are currently being analyzed (a signal analysis approach). The goal of the study is to increase understanding of the depth dependence of the hydraulic gradient and the influence of the topography /5.5-65/.

One of the driving forces for groundwater flow is differences in the density of the liquid. The groundwater in the bedrock exhibits a salinity variation that influences the density and the prevailing pressure conditions in detail. If it were possible to measure vertical pressure variations at model boundaries with greater accuracy to a depth of 1 km, more realistic regional flow descriptions would probably be obtained than flows calculated from boundary conditions based on topographical data.

Another aspect that has a bearing on the salinity and density of the groundwater is any convection flows that might arise due to heating from deposited canisters. Results show that a salinity gradient with depth has an opposing and counterbalancing function on any convection flows /5.5-66/.

Hydraulic conductivity is a central parameter for being able to describe the flow of the groundwater in any geological media. As mentioned above, it is related to the properties of both the flow medium and the liquid. (If conductivity is integrated over the thickness of the water-conducting stratum, the parameter is termed transmissivity.)

In-situ methods for determination of hydraulic conductivity and storage characteristics in fractured rock are constantly being refined. Great efforts are being made in particular within the field of gas and oil prospecting. The analysis methodology has been improved through the Stripa and Äspö projects, particularly when it comes to cross-hole tests (interference tests). There is nevertheless a need for a greater understanding of the dependence of the in-situ methods on scale, rock heterogeneity, dimension of measurements, rock volume covered by measurement, etc. This work will also be pursued in conjunction with the Äspö Project and the tests that have been begun in the deep borehole Laxemar KLX 02. A separate state-of-knowledge compilation with suggestions for development was recently performed by an expert group at the initiative of SKB /5.5-69/.

In RD&D-Programme 92, it was stated that the results of several tunnel investigations have shown that hydraulic conductivities appear to decrease nearest the tunnel walls. Explanations have been sought for this phenomenon in stress redistributions and chemical precipitations. Another explanation is that the decrease in conductivity may be a phase boundary mechanism between air or other gas and water. The process is then regulated by a two-phase flow. Preliminary theoretical calculations support the latter theory /5.5-70/ and SKB has, together with the US DOE (U.S. Department of Energy), initiated experimental activities focusing on two-phase flow at the Äspö HRL /5.5-71/. Furthermore overall understanding of two-phase flow is important for interpretation and conceptualization of groundwater flows based on tunnel mapping.

Knowledge of two-phase flow in fractured rock is also essential for understanding the water homogenization of the bentonite buffer during the deposition phase and when it comes to the conditions for gas migration.

Flow porosity (kinematic) is a central parameter when the advection process or the groundwater's mean flow velocity is to be determined. Relatively few guideline values are given for crystalline rock types in the international literature. Generally speaking, flow porosity is low and on the order of 10^{-5} to 10^{-2} /5.5-72, 77, 78/.

Values of kinematic porosity can be obtained indirectly from hydraulic tests or via tracer tests. In the case of hydraulic tests, permeability is interpreted and fracture apertures are generalized using a relationship saying that the hydraulic conductivity is proportional to the fracture aperture cubed ("cubic law relationship"). In tracer tests, the available flow porosity is interpreted on the basis of a mean velocity and with reference to a kinematic dispersivity.

Within the framework of the Äspö Project, certain in-situ studies have been initiated which will provide a better database on available porosity and its variation in connection with transport modelling of groundwater /5.5-73/. On assignment for SKB, a method has also been developed for fracture porosity determination where fluorescent epoxy is injected into drill cores and mapped/analyzed by means of an image processing system, see also below and Figure 5.5-8.

Kinematic dispersion is a "mixing phenomenon" that is dependent on velocity differences for the flow in a fracture and between different fractures. The flow is mixed through fractures which intersect each other, and changes in the concentrations of solutes in time and space are obtained. It has been found that the classic dispersion equation only seems to be valid after long flow times and for large distances from the point of release. The dispersion for instantaneous releases is, for example, probably controlled by time-dependent dispersion coefficients and not by any scale effect in space /5.5-76, 79/.

Molecular diffusion entails that substances dissolved in the water (solutes) move from areas with high concentrations to areas with low concentrations (Fick's law). Molecular diffusion is generally considered to be subordinate to the effect of kinematic dispersion in non-reactive transport and is not dealt with for the fractures and zones in the bedrock /5.5-79/. The process is, on the other hand, essential for diffusion into the microfissures in the rock, so-called matrix diffusion.

The **storage coefficient** in the bedrock is of importance when the transient flow situation of the groundwater is to be described. The specific storage coefficient is defined as the volume of water removed per unit decline of the hydraulic head and is dependent on the porosity and the compressibility of the water and the rock. It has been found in theoretical studies that the storage coefficient is proportional to the cube root of the transmissivity. For more channelled flows, the corresponding relationship is a simple root expression. Field measurements of hydraulic diffusivity support these theories /5.5-74, 75, 76/.

Databases of storage coefficients for crystalline basement rock are relatively limited. As transient cross-hole tests become more common in fractured bedrock, experience of variability and possible scale effects is expected to improve. Ordinary values of the storage coefficient for Swedish crystalline basement rock lie in the range $5 \times 10^{-4} - 5 \times 10^{-6}$.

Retention

The state of knowledge concerning the bedrock's hydrochemical environment is described more thoroughly in section 5.5.5. The state of knowledge concerning radionuclides and the geochemical environment for the repository's various barriers in its entirety is described in section 5.6. This section provides a brief commentary solely on the retention properties of the radionuclides' transport pathways with respect to sorption mechanisms and diffusion into the rock's microfissures, known as matrix diffusion. Brief comments are also made on conditions for transport with naturally dissolved gases in the groundwater.

Sorption is strongly dependent on the ions' charge, hydrolysis and possible complexes with strong complexing agents. It is therefore essential to know the groundwater's pH, redox conditions and content of complexing agents such as humic and fulvic acids. Ion exchange is an important sorption mechanism for e.g. Cs^+ and Sr^{2+}. The salinity of the water is therefore also of importance, see section 5.5.5.

The minerals that constitute the substrate for sorption have different capacities to absorb radionuclides. Certain minerals are, for example, good ion exchangers, while others are not, and so on. The K_d values that are used in the safety assessment are chosen so that the retention of radionuclide transport is not overestimated. Complexing with humic and fulvic acids can reduce the sorption of some of the radionuclides. The chosen K_d values can be adjusted with this in mind /5.5-89/.

Sorbing radionuclides could in principle be transported with the water if they adhered to colloidal particles in the groundwater. The concentration of colloids in the groundwater is lower than 0.4 mg/l. They consist of inorganic particles, e.g. calcite, iron hydroxide, clay etc. and can of course sorb radionuclides. If the uptake of radionuclides on colloidal particles is reversible, it is of no consequence for radionuclide transport. Somewhere along the streamline the nuclide is then turned over to the rock. If, on the other hand, the nuclide should become sorbed irreversibly, the nuclide will be transported with the particle and, at worst, not be retarded at all by sorption in the rock. Laboratory tests verify that the sorption is largely reversible. The strength of the sorption is roughly equivalent to measured K_d values for corresponding minerals and solutes /5.5-90/.

Thorough analyses of the groundwater show that bacteria also exist at great depth. All species have not been identified, but methane bacteria and sulphate-reducing bacteria have been found. The environment is poor in nutrients. Substances that could conceivably be included in the metabolism of the micro-organisms are, for example, methane, hydrogen, organic matter, carbonate, sulphate, etc. Laboratory tests show that bacteria can take up radionuclides. In principle, radionuclides should be able to accompany bacteria in the same way as they accompany other colloidal particles in the groundwater. However, the concentrations of microbes are very low (less than 50 mg/m^3). The importance of bacteria transport for safety has been analyzed in the same way as for inorganic colloids and the conclusions are that it has a negligible effect /5.5-90/.

Development of more detailed models for sorption and matrix diffusion that will better describe the physical and chemical processes involved is being pursued both in Sweden and in several other countries /5.5-88/.

Matrix diffusion of dissolved radionuclides could take place to the micropores in the rock that surround the larger fractures and zones. At later stages, radionuclides could diffuse back to the flowing water in the fractures. These processes have noticeable effects on the transport of both non-sorbing and sorbing radionuclides. The relatively large surfaces in the micropores are of essential importance for the sorbing nuclides. The most important parameters that determine matrix diffusion are specific area, the diffusion coefficient and the diffusion porosity /5.5-77, 92/.

Gas transport of nuclides is a possible transport mechanism. Transport with naturally occurring gases (geogas) and transport with hydrogen gas generated by canister corrosion are conceivable.

Regarding geogas transport, it has been claimed that trace quantities of metals can be transported with gas bubbles in the rock /5.5-96/. The conclusion is based on analyses of geogas above relatively deeply-lying ore bodies. The gas has contained traces of metal from the ore. But the concentration of metal in the gas compared with the quantity of metal in the ore shows that only very small quantities could be released and transported to the biosphere in this manner. In order for gas bubbles to be formed, the water must be saturated with gas at the pressures that prevail at repository depth. Experience from the Äspö Project shows that the groundwater at the depth in question is greatly undersaturated with respect to dissolved gases, more than 90% of which consist of nitrogen. The depth at which the measured dissolved quantity of gas passes over into a separate gas phase is no deeper than about 50 m /5.5-89/.

Hydrogen gas can be formed by anaerobic corrosion of iron if water should enter a copper-steel canister, see chapter 6. The hydrogen gas can be a carrier of radionuclides that exist in the gas phase, for example ^{14}C, ^{85}Kr. If the hydrogen gas passes the bentonite buffer and backfill, model calculations show that the conditions exist for direct gas transport to the ground surface from the deep repository /5.5-97/. In order not to underestimate the effects of nuclide transport in a pure gas phase in SKB's safety assessments, a direct short-circuit is assumed between the near field and the biosphere. A general knowledge buildup on gas migration and two-phase flow has begun at the Äspö HRL.

Characterization of fractures with regard to nuclide transport and retention

Due to the heterogeneity of the bedrock, flow calculations and retention assessments must be based on certain assumptions. For example, channel flow with constant water chemistry, hydraulic gradient, fracture mineralogy and porosity along the entire flow path can be assumed in a transport modelling.

A transport model can be more or less grounded in geological reality. The characteristics of a fracture, such as porosity, fracture mineralizations, etc., are naturally a result of the mode of formation and development of the bedrock on a geological time scale. In recent years, SKB has started and completed several projects with the overall goal of improving the classification of individual fractures and fracture systems with a focus on radionuclide transport and retention. Fractures are thereby characterized with respect to rock type, tectonic evolution, mineralizations and alterations in the vicinity of the fractures. A fracture characterization with this focus is being applied and tested at the Äspö HRL /5.5-19/.

Both quantitative and qualitative understanding of fracture characteristics will increase as results from underground laboratories become available. This subject area has high priority when experimental activities begin in the Äspö Project.

Figure 5.5-9 provides a summary of geological, hydrogeological and hydrochemical data with a focus on the processes in a transport model that are commented on briefly in the previous sections in this chapter.

Several important results have been presented with regard to hydrothermally altered rock matrix around fractures. For the red-coloured rock near fractures in the Äspö tunnel, for example, higher porosity, lower density and greatly reduced magnetic susceptibility have been found. Furthermore, it was found in the study that calcium had been replaced by potassium and sodium in the altered portions of the rock (albitization). The red coloration in Äspö is due to an oxidation of magnetite minerals to hematite and iron hydroxides. These observations indicate properties in the fracture surfaces that contribute to strong retention of solutes /5.5-83/.

One goal is to improve our understanding of fracture flows with respect to the detailed geometric conditions that prevail in an individual fracture. SKB has had a special laboratory apparatus manufactured, a biaxial cell, where relationships between rock stresses over individual fractures and groundwater flows are studied. Half-metre long drill cores with a diameter of 200 mm can be placed in the cell. The cell is set up at the Chalmers University of Technology, Gothenburg, and image processing analysis of fractures is done at the Royal Institute of Technology, Stockholm, see Figure 5.5-10. One of the findings is that the ratio between mechanical and hydraulic mean fracture aperture is about 1.4 /5.5-84, 119, 120/.

Activities in relation to goals in RD&D-Programme 92

A number of goals were set up in RD&D-Programme 92 for the R&D work for the period 1993–1998 concerning the ability of the rock to limit transport. Following is a brief status report in relation to these goals:

- Further refine methods for description of the geometry and hydraulic properties of fractures. – This is a central field of research in SKB's activities. Compilations are being made continuously, including material from the Äspö Project.

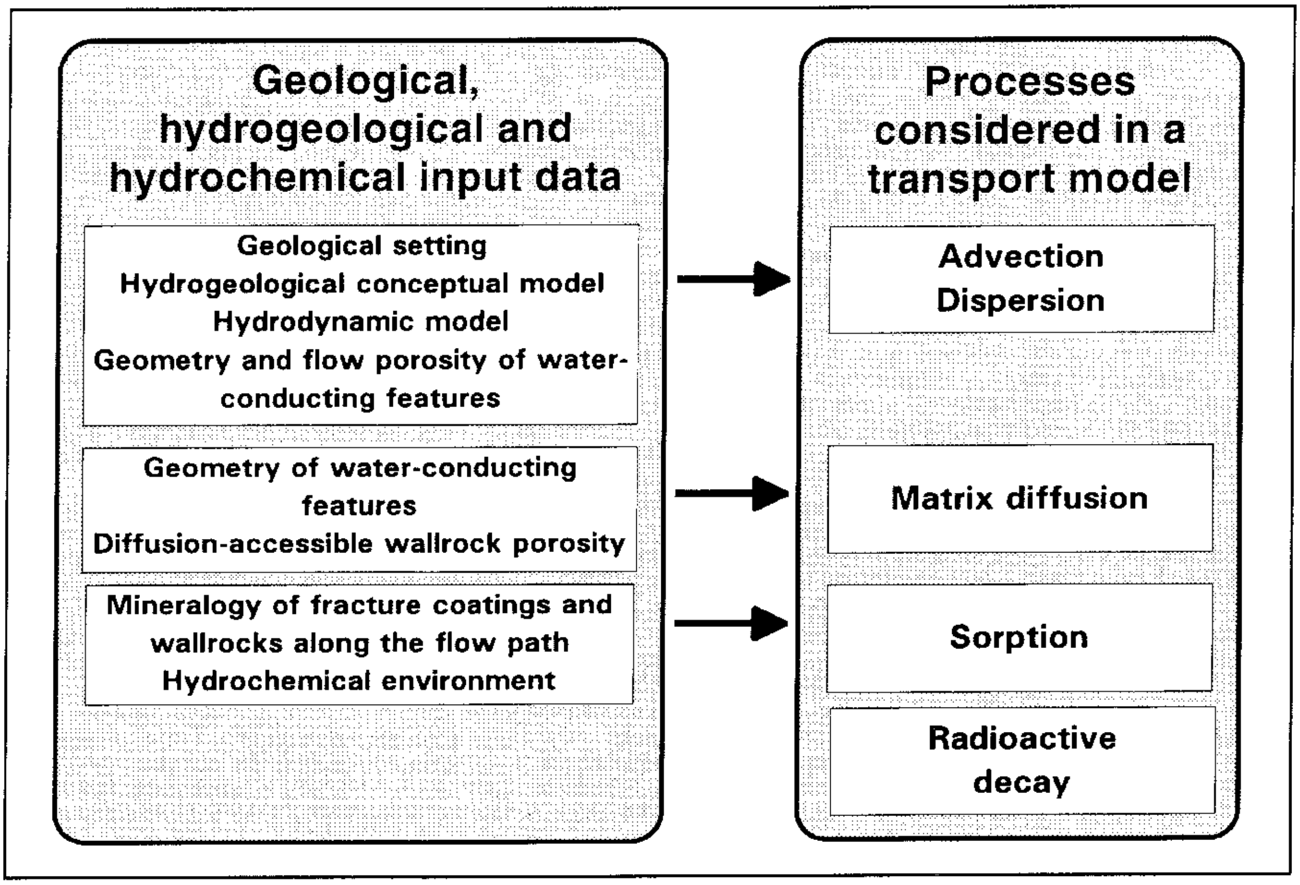


Figure 5.5-9. *Use of geological, hydrogeological and hydrochemical information in modelling of radionuclide transport – an overview of parameters and processes (after /5.5-19/).*

- Further refine in-situ methods for determination and analysis of hydraulic properties in fractured rock. – Method development has taken place in the Äspö Project and in the Laxemar deep borehole. A separate working group recently compiled a state-of-the-art report /5.5-69/.

- Investigate the large-scale dependence of the hydraulic properties on fracture mineralizations, rock stresses and former permafrost depths. – Work is being pursued with a primary focus on the correlation between permeability and rock stresses. The main sources being used for this are SGU's well archive, study site results and the nationwide rock stress database compiled by SKB.

- Investigate and compile indirect signs of the groundwater's flow pattern in fractured rock for both model structuring and verification of models. – A project for characterization and classification of fractures is being pursued in cooperation with NAGRA of Switzerland. The methods are being tested in the Äspö HRL.

- Investigate risks of short-duration pressure changes in the groundwater reservoir at repository level due to earthquakes. – Cooperation has been initiated with PNC, Japan, and SKB is participating in a project for this purpose in the Kamaishi research mine.

- Further refine theories of non-reactive flow transport in fractures and systems of fractures. – Separate laboratory studies have been conducted in cooperation with the Chalmers University of Technology in Gothenburg and the Royal Institute of Technology in Stockholm.

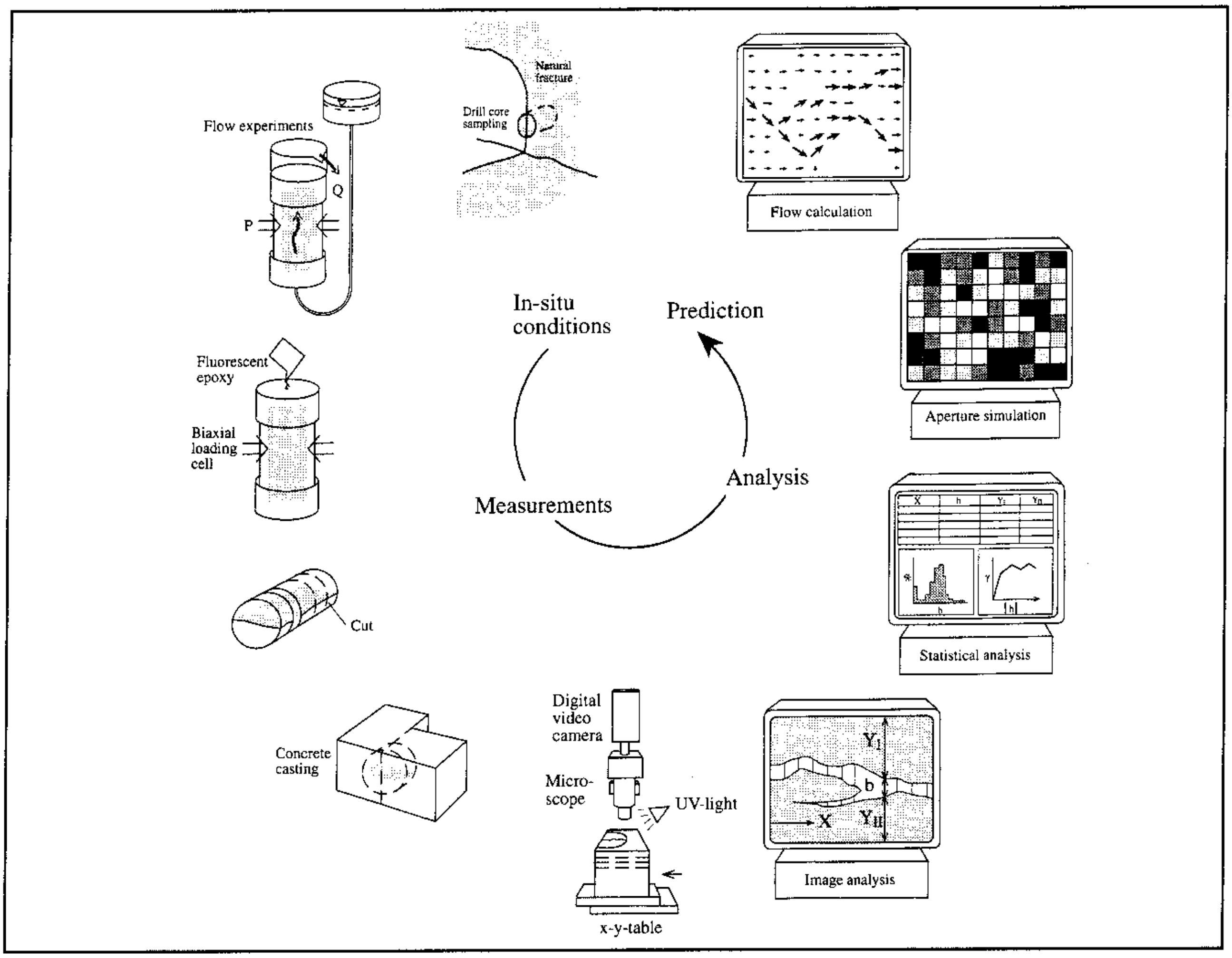


***Figure 5.5-10.** Survey of fracture geometries and fracture flow – project structure.*

5.5.7 Project "Deep drilling KLX 02 – Laxemar"

Sections 5.5.4–5.5.6 deal with, in order of mention, the state of knowledge regarding the rock's mechanical stability, hydrochemical environment and ability to retain radionuclides. Mechanical, hydraulic, thermal and chemical aspects are all being studied in the field in a small integrated project, "Deep drilling KLX 02 – Laxemar". The project, which is being undertaken alongside of the Äspö HRL, was initiated in the autumn of 1992 with the following goals:

- to broaden knowledge of the composition and properties of the rock at great depth and to obtain new information regarding the flow pattern and chemical composition of the groundwater.

The project has further had the following interim goals:

- to identify different possible drilling techniques for exploratory drilling to great depth and to demonstrate such drilling to a depth of about 1500 m below the surface,
- to demonstrate methods for investigations in boreholes within the depth range 1000–1500 m.

The drilling of KLX 02 was carried out during the period October–November 1992 to a depth of 1700 m using the wire-line technique. An intact drill core exists for the 200–1700 m interval. After drilling, the investigative activities have included geological mapping, mineralogy, geophysical measurements, groundwater chemistry and groundwater hydraulics. Equipment has been modified to measure rock stresses at levels deeper than 1000 m /5.5-121–130, 147/. The deep borehole KLX 02 at Laxemar furnishes essential input data to SKB's paleohydrogeological programme, see section 5.5.8.

As the hole has been drilled to 1700 m, it also furnishes important information to the programme within SKB that deals with the alternative repository concept "Very Deep Holes", see section 13.2.

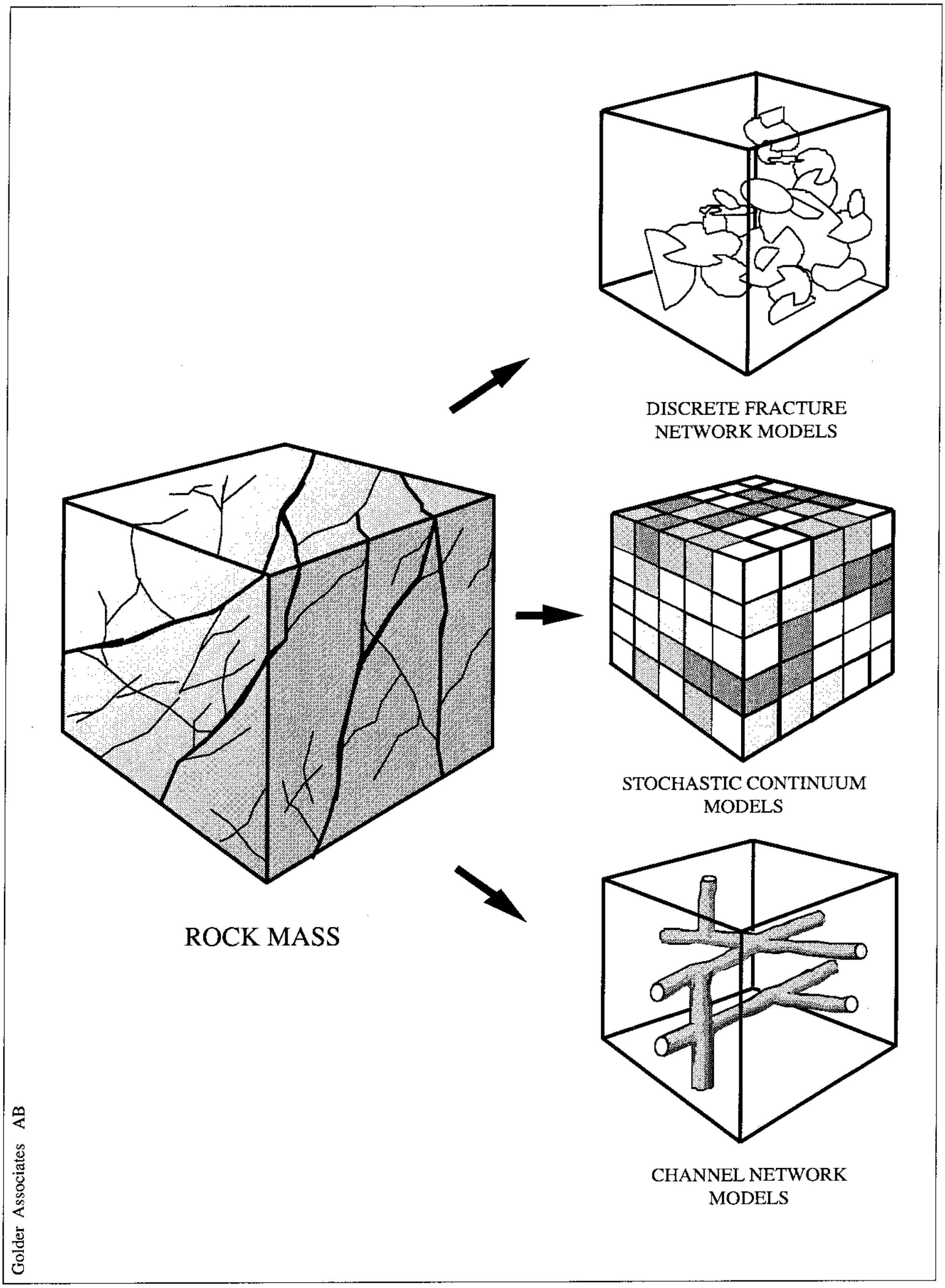


Figure 5.5-11. *Three common approaches to modelling groundwater flow and groundwater transport in fractured bedrock (from /5.5-98/).*

5.5.8 Model tools and model development

Mathematical models for groundwater flow, nuclide transport and rock mechanics comprise important tools in the work of waste disposal. As the computation capacity of computers increases, numerical model codes are being developed alongside of purely analytical equation solutions and empirical relationships. The options available for solving calculation problems with various coupled physical and chemical processes have thus increased in recent years.

SKB's overall view on model concepts and how models are used in its activities is described in section 5.1. The mathematical tool for calculating and describing a chemical or physical process can be a numerical model or an analytical equation solution. Assumptions, simplifications and relationships that are utilized for the calculation are shown by a conceptual model. The conceptual model is thus a description of how the geometric assumptions (the structure) and the component processes are organized and represented /5.5-85/.

SKB's efforts in model development have been concentrated on necessary calculation tools for performance and safety assessments, at the Äspö Hard Rock Laboratory and in the Stripa Project.

Groundwater modelling

Modelling of groundwater flow and groundwater transport of dissolved substances in fractured, low-permeability rock is relatively complex compared with in geologically porous media. Connections between fractures form flow paths which can be very irregular. To describe the groundwater flow in the bedrock, it is necessary to represent the heterogeneous structure of the rock in models and to take into account the scale on which the calculation problem is being regarded. Different types of approaches, conceptual models, are used, and they can be said to represent different idealizations of how the groundwater flows. The conceptual model is the basis of the mathematical model, which contains equations that are solved analytically or numerically. Three different approaches are illustrated in Figure 5.5-11.

Discrete Fracture Network (DFN) modelling is intuitively appealing because the primary flow paths are assumed to be represented by a network of interconnected fracture planes. The model is built up on the basis of a statistical description of the geometric and hydraulic properties of the individual fractures. This requires data that give distributions of the position, length, orientation and transmissivity of existing fractures. Different fracture populations can thereafter be simulated based on these statistics.

Stochastic Continuum (SC) modelling is based on the assumption that properties in the rock (hydraulic conductivity, storage coefficient, etc.) can be described as variations with spatial distribution functions. The stochastic approach contains several geostatistical steps. Stochastic continuum modelling is used to advantage for analyses aimed at studying regional scales (km scale) and for conditional simulation with the aid of e.g. kriging. Classical or deterministic continuum modelling can be said to constitute a special case of stochastic continuum modelling.

Channel Network (CN) modelling represents the flow in fractured rock as limited, discrete and for the most part one-dimensional flow paths, channels, which intersect each other at certain intervals. The approach is based on observations in the field, mainly from tunnels, where groundwater often occurs as flows along channels in fractures in the rock. The properties of the channels are derived from field data. Measurements of channel widths are necessary. Field measurements in tunnel fractures are, however, associated with uncertainties due to the disturbances in the rock caused by the tunnelling work.

A more detailed description of these basic concepts is provided in SKB reports, e.g. /5.5-98/ and /5.5-99/. Further details on the three concepts with a special emphasis on the data requirement are provided in separate reports /5.5-100, 101, 102/.

The choice of model for an analysis is entirely dependent on the purpose of the analysis in question, which geometric scale is intended to be studied and available data. Regardless of model concept, there are a number of common central questions to be taken into consideration in groundwater modelling.

The boundary conditions influence the calculation results. Uncertainties can include the degree to which groundwater movements at repository depth, alongside the regional groundwater system, are influenced by local topography in combination with steeply dipping water-conducting structures /5.5-103/. When density differences in the groundwater, caused by salinity variations, are taken into account in the modelling, this must also be dealt with in the choice of boundary conditions.

Dominant water-conducting structures are handled differently in modelling depending on the model concept. The heterogeneity of the bedrock, with large differences in material properties, gives rise to both conceptual uncertainties and numerical difficulties in the modelling.

Input data to the modellings are obtained from different field investigations in boreholes. These are conducted on a given geometric scale. It may be uncertain which rock volume the actual tests represent, see section 5.5.6. In the modelling it is necessary to scale the information interpreted from field data down or up to fit the numerical calculation. This scale adaptation requires extra care in the modelling /5.5-104/.

Modelling of radionuclide transport

Modelling of groundwater flow and groundwater transport of solutes should preferably take place in a single context. In a safety assessment, however, flow/advection is often dealt with by means of a groundwater model (including dispersion, for certain model concepts), while diffusion, dispersion and retention are dealt with by means of a separate nuclide transport model. Consequently, the flow model is used to evaluate boundary conditions for the transport model in the form of length of transport pathways or alternative water travel times.

The transport processes that were discussed at the beginning of section 5.5.6 are dealt with summarily in the following way in the numerical models:

- **Advection** with the groundwater: The mean velocity of the groundwater is often modelled as the Darcy velocity divided by the flow porosity. The flow porosity is the proportion of the rock that is occupied by the flowing groundwater, and it is less than the total porosity..

- Molecular **diffusion** is described by Fick's law.

- Hydrodynamic **dispersion** is usually modelled with a diffusion term proportional to the velocity of the groundwater; transversal dispersion is often much lower than longitudinal dispersion. With this procedure, the mass flow of a substance along with the flow in the rock is given by the product of the concentration gradient, the groundwater velocity and a coefficient called the longitudinal dispersion length. The latter can be very uncertain to estimate, due to the difficulty of conducting tracer tests in fractured rock over suitable length scales. The scale dependence for the dispersion coefficient determined in the field is sometimes simulated by the use of a constant, called Peclet's number. Peclet's number represents the ratio between a characteristic time for dispersive transport and a characteristic time for advective transport. The parameter is included in the advection-dispersion formulation for particle transport in rock that is often used in safety assessment contexts. For a detailed discussion of this subject, see /5.5-77 or 82/.

- Chemical and physical **retention**: The sorption models are often based on the assumption that actual kinematic behaviour can be simplified and modelled with a linear equilibrium model. For this purpose, distribution coefficients, K_d values, are posited for each nuclide. The K_d values are determined in laboratory tests and give the ratio between the concentrations in the solid phase and in the solute phase. The equilibrium model applies if the concentrations are small and if the time scale for sorption is much smaller than the time scale for transport with advection and dispersion. Physical retention through matrix diffusion in the rock is usually handled in modelling by a double-porosity description of the fractured medium. An exchange system between the fractures and the intact rock (the rock matrix) is included in the analysis. Transport in the fractures is dominated by advection, while transport between the fractures and the intact rock is dominated by diffusion /5.5-77/.

- **Radioactive decay**, including chain decay, is also dealt with in connection with modelling of nuclide transport.

Rock-mechanics modelling

The calculation models for the mechanical properties of the rock, i.e. strength and deformation, can in principle be divided into two main groups:

- Continuum models.
- Discontinuum models.

The continuum models describe the rock as a uniform rock mass where the effects of discontinuities in the rock are included without being able to define them specially. Movements in the rock are described in these models by means of continuum mechanics, which means that only limited effects of movements along discontinuities can be taken into account. The continuum models can be of the differential or integrated type. Finite Element Methods (FEM) and Finite Difference Methods (FDM) are examples of differential methods, while Boundary Element Methods (BEM) are examples of integral methods.

In discontinuum models, the rock mass is described using a coupled model between the intact rock and occurring discontinuities in the form of fractures and zones. These models can describe movements in the rock with deformation mechanisms for sliding along fracture planes, separation of fracture planes and rotation of rock blocks. Distinct Element Methods (DEM) simulate the rock mass as a discontinuum and can be used, for example, to calculate non-linear material behaviours and large deformations which can lead to collapse /5.5-93/.

Coupled models

The deep repository's influence on the rock mass is a coupled phenomenon which, besides chemical reactions, includes thermal, hydrological and mechanical processes. These processes affect each other mutually to a greater or lesser extent. In recent years, interest has increased in developing coupled models in order to be able to describe the conditions in the near field of a repository in particular with greater realism /5.5-94, 95/. Analytical calculation methods can very seldom be employed for the complex relationships to be described in the coupled processes. Development is being pursued with numerical equation solvers, which require extensive verification.

Coupled thermo-hydro-mechanical models have been developed and verified in the DECOVALEX Project (international cooperative project for the DEvelopment of COupled models and their VAlidation against EXperiments in nuclear waste isolation) /5.5-22, 23, 24/. One of the conclusions from the first stage of the project is that the temperature field around a repository can very well be described in pure thermal conductivity terms without reference to hydro-mechanical coupling. Stress and displacement conditions are often consistent between different modelling concepts, while the hydraulic results have shown appreciable differences. The general validity of the results is also very sensitive to the choice of boundary conditions, according to whether the calculation cases are two- or three-dimensional /5.5-112/.

Äspö HRL's international Task Force on Modelling

One of the goals of the Äspö Project is to "test models for description of groundwater flow and nuclide transport". During the pre-investigation phase and the design phase, the modelling work has primarily dealt with groundwater flow. Some work has been devoted to transport of solutes, particularly saline groundwater, and evaluation of a tracer test. Modellings have also facilitated the Äspö HRL's experiment planning with design calculations.

An international Task Force on Modelling of Groundwater Flow and Transport of Solutes has been tied to the

Table 5.5-2. Organizations and model groups with their computer codes in the Äspö HRL's international Task Force on Modelling. The table also shows the calculation tasks performed during the period 1992–1995. (*) T=Task.

ORGANIZATION	MODEL GROUP	CONCEPTUAL MODEL	COMPUTER CODE	CALCULATION TASK (*)
ANDRA France	BRGM(I)	continuum	MARTHE/SESAME	T1
	BRGM(II)	continuum	ROCKFLOW	T1
	ITASCA	channel network	CHANNET/TRIPAR	T1, T3
CRIEPI Japan	CRIEPI	continuum	FEGM/FERM	T1, T3
PNC Japan	PNC/Golder	discrete fracture network	FracMan/MAFIC	T1, T3
	Hazama	stochastic continuum	SETRA/ARRANG	T1
SKB	CFE	stochastic continuum	PHOENICS/PARTRAK	T1, T2, T3
SKB	KTH, Chemical Engineering	channel network	CHAN3D	
		continuum	–	T2
SKB	KTH, Water Resources Engineering	stochastic continuum	TUBA, etc.	T2
SKB	Geosigma	continuum	SUTRA	T2
TVO Finland	VTT (flow)	continuum	FEFLOW	
	VTT (transport)	–	–	T1
UK NIREX United Kingdom	AEA Technology	continuum/discrete fracture network	NAMMU/NAPSAC	T1, T3
US DOE USA	LBL	equivalent discontinuum with inverse modelling	–	T2

Äspö HRL. The Task Force was initiated by SKB at the end of 1992. The active groups and the modellings carried out to date are shown in Table 5.5-2. The tasks have so far included: A large-scale long-term pumping test (Task 1), design calculations for forthcoming tracer tests (Task 2) and hydraulic impact on the Äspö tunnel (Task 3). The model concepts that have been utilized are also presented in the table, where relevant. The model results are presented in the Äspö HRL's international report series /5.5-73, 108–111, 113–116/.

The Task Force is a forum for the organizations that are participating in international cooperation within the Äspö HRL. The Task Force also includes representatives of NAGRA (Switzerland), AECL (Canada) and BMBF (Germany). The emphasis is on gaining confidence in and experience of the methods that are used in performance and safety assessments. The Task Force has also produced a table of the key issues in safety assessments (Issue Evaluation Table), which are related to the ongoing or forthcoming experiments at the Äspö HRL. The table of key issues should also serve as a basis for choosing future modelling tasks.

Paleohydrogeological programme and regional modelling

The safety assessment's reference scenario deals with a climate situation similar to that existing today, whose geoscientific parameters and boundary conditions do not change with time. Effects of possible future periods with permafrost and glaciation are associated with extreme and necessarily speculative scenarios. The questions relate to the performance of a deep repository and the radiological risks to humans, animals and vegetation after the next ice age.

Based on knowledge of the extent of the Weichsel Glacial at different times and indirect climate data from pollen and isotope analyses, SKB has had a time-dependent glaciation model developed /5.5-32, 33, 118/. The model, which exists in a two- and three-dimensional version, is able to handle coupled processes with:

- temperature changes in ice and bedrock,
- growth and flow of the continental ice sheet in the landscape with actual topography,
- generalized mechanical impact on the lithosphere,
- meltwater flows at different stages during a glaciation cycle.

A simple model code for groundwater flow has also been attached to the model for the purpose of being able to handle and describe hydrogeological changes in connection with permafrost, glaciation and deglaciation. Of special interest are, for example, pressure and gradient changes under the ice and in the bedrock. The groundwater model also allows the flow of infiltrated meltwater to be followed in a simplified manner.

The development work with the glaciation model has had two purposes, namely to:

- increase the reliability of the predictive calculation models covering long periods of time that are used by improving understanding of thermo-hydro-mechanically and chemically coupled processes and their development (paleohydrogeological programme),
- create a better platform for handling extreme future scenarios that include permafrost, glaciation and deglaciation.

The glaciation model developed by SKB thus makes it possible to discuss and justify scenarios for the future performance and safety of a deep repository with a good underpinning. Site-specific data should be included in the assessment to as great an extent as possible. The hydraulic aspects are of central interest, but thermal, mechanical and certain hydrochemical aspects can also be dealt with.

A development project with groundwater modelling in a regional perspective is currently under way. The region includes the Simpevarp Peninsula, Äspö and the Laxemar area north of Oskarshamn. The project has a paleohydrogeological orientation, i.e. modelling results concerning groundwater flow and transport of dissolved salts are compared with hydrochemical analyses from the Äspö HRL and the 1700 m deep borehole KLX 02 in Laxemar, see also section 5.5.7. A separate programme for the activities with all their integrated parts has been compiled /5.5-158/.

Data management

A calculation problem within applied geology consists of both quantitative and qualitative data. When evaluating large quantities of data and how they can be used, it is necessary to take into account measurement and interpretation methods, representative dimension and from which geological unit samples have been taken, etc. See also section 5.5.2.

In the international perspective a large development is under way regarding geostatic analysis methods and their practical application, see e.g. /5.5-131–137/. Activities at SKB have mainly been focused on:

- developing and applying methods for statistical inference, i.e. limiting the uncertainties regarding properties in the rock outside the actual measurement points and being able to posit a value for how reliably these properties are described. This is done e.g. by means of variogram analysis and/or by the use of methods for non-parametric geostatistics,
- developing methods for analyzing whether data are statistically stationary or not (a parameter is stationary if both its mean value and variance are independent of "sampling scale" and geological unit),

- developing and applying regression methods, above all multivariate analysis, for hydrochemical data,
- refining and applying methods for hydrogeological classification in a siting process with continuous updating of available information according to the Markov-Bayes geostatistical model.

Studies with the above orientation have been presented in different articles and SKB publications, see e.g. /5.5-117, 138–144/.

Activities in relation to goals in RD&D-Programme 92

A number of goals were set up in RD&D-Programme 92 for modelling tools and model development for the period 1993–1998. Following is a brief status report in relation to these goals:

- Describe conditions for groundwater flow and transport at a deep repository in a regional perspective in the present-day climate situation. – A separate development programme is under way that is exemplified with data from the Äspö-Laxemar region.
- Describe the conditions for groundwater flow and transport at a deep repository in a regional perspective during glaciation and deglaciation. – a time-dependent glaciation model has been developed for Scandinavia /5.5-32/. A finite element code able to handle meltwater and groundwater flows during a glaciation cycle has been coupled to the model. In coordination with the above programme, the models are being tested with a paleohydrogeological orientation.
- Compile and structure geostatistical data that are used in conjunction with safety assessment and constructability assessment. – A separate compilation of factors is included in SKB's supplement to RD&D-Programme 92.
- Further develop how results from the volume representativity and dimensionality of hydraulic tests are to be integrated in a model structure. – A working group, HYDRIS, has compiled different aspects of execution, interpretation and use of test pumping results /5.5-69/. Development work is under way on volume representativity in the model structuring.
- Integrate and take into account general geological and geophysical information in the conductivity distribution for a site-specific stochastic groundwater modelling (indicator simulation). – Exemplifying calculations have been carried out and reported on /5.5-138/.
- Refine models for convection modelling in fractured rock. – Development work is under way.
- Refine rock-mechanics stochastic models and refine the scale dependence within rock-mechanics model structures. – Some model development has taken place within the framework of the international DECOVALEX programme /5.5-24/.
- Refine coupled hydro-thermo-mechanical models. – SKB is involved in several modelling tasks in the DECOVALEX programme.

5.6 CHEMISTRY

The results of the last three years' investigations within the chemistry programme are reported in a number of reports, publications and theses. Summaries and references to this material are found in SKB's Annual Reports (see e.g. /5.6-5/). A review of the rock's ability to influence nuclide transport through various bedrock-related factors is provided in section 5.5.6.

The following review is deliberately concise for the sake of anyone who wants to obtain a quick overview of what has happened since the last RD&D programme. SKB's Annual Reports provide more details and contain references to all background reports.

The chemistry programme with a focus on deep disposal of spent fuel has made substantial progress. Its scope has therefore been narrowed slightly and some investigations have progressed from a general orientation within the chemistry programme to applied studies. Certain general studies such as solubility, sorption, diffusion etc. are now concentrated within the project for other waste (long-lived low- and intermediate-level waste), see section 5.9.

5.6.1 Radionuclide chemistry

Solubility and complexation

Several of the radionuclides that are present in the waste and are important for long-term safety have a low solubility both in the groundwater and in the bentonite clay's pore water. Formation of complexes affects solubility, for example complexes with an actinide as the central atom and hydroxide or carbonate ions as ligands. This must be taken into consideration. Thermodynamic data and models are used to calculate solubility and speciation, and experiments are performed to improve the body of data. SKB has contributed to a large number of measurements of constants for formation of hydroxide and carbonate complexes with, for example, tetravalent thorium, technetium and neptunium, see Table 5.6-1 /5.6-1, 2, 3, 4/. It has been argued that complexation between tetravalent actinides and phosphates is of importance, despite the low concentration of phosphate in groundwater and bentonite. The conclusion is based on

Table 5.6-1. Equilibrium constants for thorium and neptunium(IV) in carbonate containing solution.

Equilibrium reaction	Equilibrium constant log K
Thorium[a]	
$ThO_2(s) + 4H^+ = Th^{4+} + 2H_2O$	9.47±0.13
$ThO_2(s) + H^+ + H_2O + CO_3^{2-} = Th(OH)_3CO_3^-$	6.11±0.19
$ThO2(s) + 4H^+ + 5CO_3^{2-} = TH(CO_3)_5^{6-} + 2H_2O$	42.12±0.32
Neptunium[b]	
$Np(OH)_4(s) = Np(OH)_4(aq)$	-8.28±0.23
$Np(OH)_4(aq) + CO_3^{2-} = Np(OH)_4CO_3^{2-}$	3.00±0.12
$Np(OH)_4(aq) + HCO_3^- = Np(OH)_3CO_3^- + H_2O$	3.23±0.12
$Np^{4+} + 4OH^- = Np(OH)_4(aq)$	46.2±0.3
$Np^{4+} + 4OH^- + CO_3^{2-} = Np(OH)_4CO_3^{2-}$	49.2±0.3
$Np^{4+} + 3OH^- + CO_3^{2-} = Np(OH)_3CO_3^-$	45.2±0.3

a Reference /5.6-3/
b Reference /5.6-4/

Table 5.6-2. Experimentally determined stability constants β for thorium complexes with phosphate /5.6-6/.

Complex	Exp. pH 8.0	Exp. pH 9.0	Literature value[(1)]
$ThHPO_4^{2+}$	$(5 \cdot 10^8)$	$(5 \cdot 10^9)$	$2 \cdot 10^{13}$
$Th(HPO_4)_{2.aq}$	$1 \cdot 10^{15}$	$2 \cdot 10^{17}$	$3 \cdot 10^{26}$
$Th(HPO_4)_3^{2-}$	$1 \cdot 10^{21}$	$1 \cdot 10^{23}$	$8 \cdot 10^{34}$

(1) Moskvin et al. /5.6-7/.

databases that contain older and incorrect constants for complexation with phosphate. New and more exact measurements which SKB has had carried out (see Table 5.6-2, /5.6-5/), show that the phosphate complexes in these contexts are in fact less important.

Regarding the chemistry of plutonium, SKB has prepared new investigations and this is now a prioritized area. The task is not an easy one, since plutonium has a complicated chemistry (several different possible redox states), in addition to the fact that high standards of radiological safety are necessary at the laboratory. As an example, it can be mentioned that the technique that has been developed with SKB's support to study Np(IV) and Tc(IV) can probably not be used for plutonium /5.6-4/.

To be able to use equilibrium calculations, it is first necessary to know that the reactions take place. For this reason, reaction kinetics have also been investigated, for example to show how "high-mobility" pertechnetate, i.e. technetium(VII), and neptunyl, i.e. neptunium(V), are reduced to tetravalent technetium and neptunium with low solubility and high sorption /5.6-8/. It has now been established that such a reduction actually takes place under the geochemical conditions that prevail in a deep repository.

Co-precipitation is still difficult to utilize fully in transport calculations. SKB will continue to follow international developments in the field.

Sorption and diffusion

Sorption and diffusion are processes that influence the migration of the radionuclides. Ion exchange and surface complexation are used as models to describe the uptake of dissolved radionuclides on mineral surfaces in the rock and the bentonite clay. SKB is funding a number of such studies. The results of the experiments have not been used to replace the sorption coefficients (K_d values), but rather to increase understanding of the sorption mechanisms and determine what they are dependent on, i.e. how reliable the sorption can be considered to be /5.6-9/.

There is still no good method for determining the flow-wetted surface area, but studies are being conducted for example in the Äspö programme.

Supplementary measurements of diffusion in concrete and bentonite have been and are still being carried out. Regarding bentonite, the aim is to determine data that describe diffusion and sorption in bentonite.

A literature survey has been conducted of matrix diffusion by radionuclides in rock /5.6-10/.

5.6.2 Organic substances, colloids and microbes

Colloids

The column experiments with colloids and radionuclides have been concluded. The experiments show that colloids can take up and transport radionuclides, at least under special conditions /5.6-11/. However, deep groundwaters have such low contents of colloidal particles, that they cannot contribute to radionuclide transport to any appreciable extent /5.6-12/. In other words, it has been possible to show that the groundwater's natural content of colloidal particles is not of any safety-related consequence. There is, however, reason to look at the substances that come from the repository itself, e.g. clay particles from the bentonite buffer. A special case is transport with gas bubbles, to which more attention will be devoted.

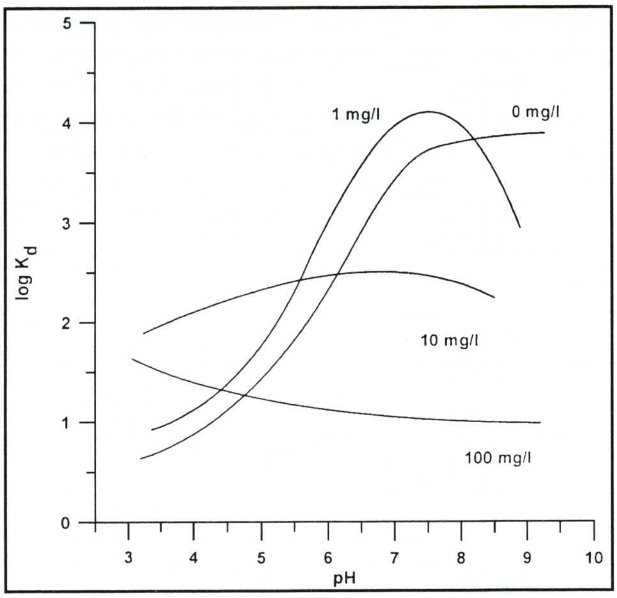


Figure 5.6-1. *Influence of fulvic acids on the sorption of Eu on alumina at different pH-values and concentrations of fulvic acid.*

Organic substances in the groundwater

A series of laboratory experiments with natural humic and fulvic acids has been carried out which verifies that the sorption of certain radionuclides, e.g. Am^{3+}, on minerals is hindered by complexation /5.6-13/. However, the same experiments show that, owing to the low concentrations of natural humic and fulvic acids in deep groundwaters, the size of the reduction of radionuclide sorption is limited or even negligible, see Figure 5.6-1. Similar conclusions have been arrived at in the majority of the countries that are considering emplacing their HLW in a deep repository in granitic rock /5.6-15/.

All organic matter in the groundwater is not humic and fulvic acids, however, and it still remains to determine what other organic substances in the groundwater consist of and what their chemical properties vis-à-vis radionuclides are.

Microbes

It is essential to investigate the importance of microbes. The area has been prioritized since 1992. In principle, microbes can influence a number of conditions of importance for the isolation of radioactive waste, e.g. migration, solubility and gas generation. Of the greatest importance is, however, the capacity of microbes to influence the chemical environment with which the canister and the waste can come into contact. The analyses and studies that have been performed at Äspö HRL support this view. Microbes can be helpful by contributing to the chemical reduction of oxygen and radionuclides, or harmful by reducing sulphate to sulphide. This can affect the waste canister, and for this reason special attention has been devoted to the occurrence of microbes in the chemistry programme.

Studies of microbes have been intensified and sampling has been carried out at Äspö and at most of the places where analogue investigations have been conducted, see Figure 5.6-2. A collaborative effort with AECL and ANDRA in microbial studies of the buffer (sand/bentonite) used in a Canadian "Buffer Mass Test" in the underground research laboratory, URL, in Manitoba

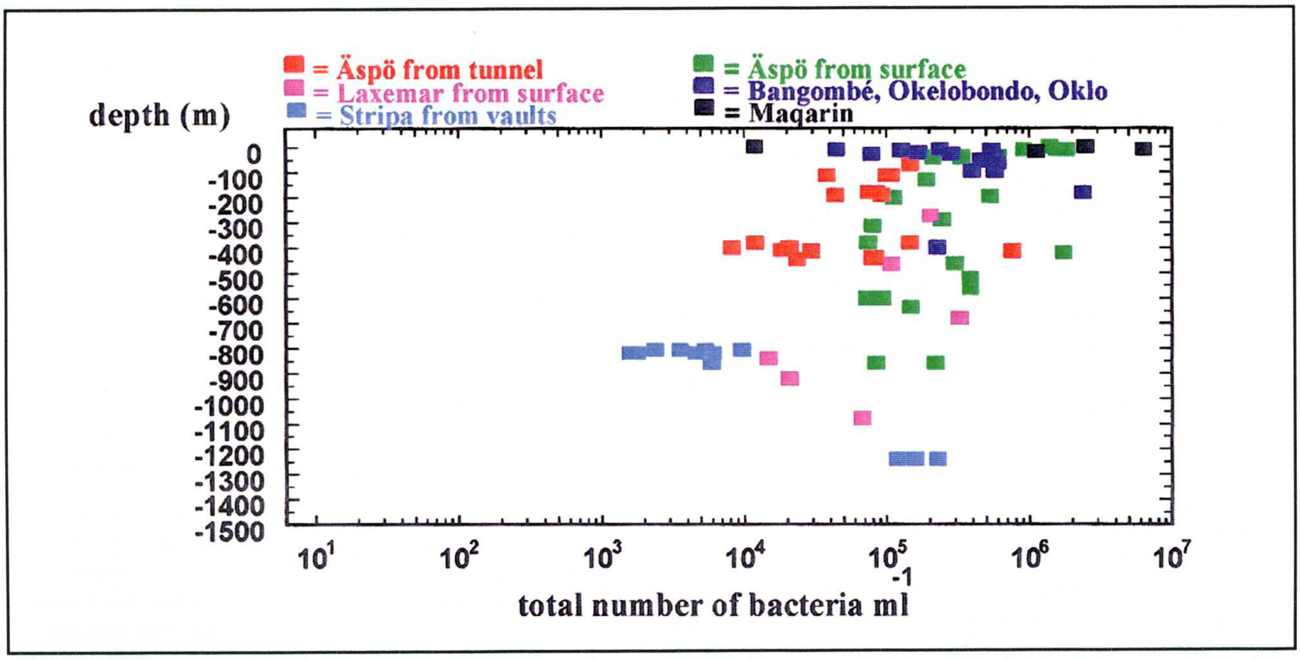


Figure 5.6-2. *The total number of bacteria in groundwater down to 1,240 m, determined with the acridine orange counting technique. Data has been collected over a period of 9 years from 30 boreholes at 49 different sections.*

have yielded essential information on bacteria in a bentonite buffer.

A summary of the present-day state of knowledge regarding microbes and their importance for long-term safety is presented in a technical report /5.6-16/. The following safety-related areas are dealt with in this report:

- Radionuclide transport and bacteria.
- Complexing agents from bacteria.
- Microbial processes in a repository.
- Steel corrosion and bacteria.
- Sulphate-reducing bacteria and copper corrosion.
- Bacterial sulphate reduction by hydrogen gas.
- Consumption of oxygen by bacteria.
- Reducing conditions and bacteria.
- The living conditions of bacteria in bentonite.

5.6.3 Validation experiments

The work of validating models and assumptions in performance and safety assessments is focused today on the in-situ experiments that will be conducted with the CHEMLAB probe in Äspö, see chapter 12. The objective of the CHEMLAB experiments is to test the models that are used to describe dissolution and migration of radionuclides. These experiments are being prepared in the laboratory. The tests which we will later perform in-situ in the CHEMLAB probe are first being tried on a bench scale.

Concrete is frequently used in underground construction, e.g. for paving of floors, shotcreting of walls and roofs, grouting of fractures, structures of various kinds, etc.

Ordinary Portland cement contains alkali hydroxides (NaOH and KOH) and portlandite ($Ca(OH)_2$), which give the pore water a high pH. To ascertain the long-term geochemical influence of concrete, experiments have been conducted where simulated cement pore water has been allowed to run through columns filled with granitic minerals. Calcium-silica-hydrate phases are formed by reactions with silicate minerals. The reaction proceeds very slowly and the new solid phases that are formed tend to clog the columns. The experiments are being performed by BGS (the British Geological Survey) in the UK with the support of NAGRA, NIREX and SKB. The objective is to validate the models that are used. The results of the first phase of the experiments in England have been reported /5.6-19/.

5.6.4 Hazardous substances

The important research work has to do with the radiological toxicity of the waste. A small portion of the substances included in the waste can, however, be classified as chemically toxic hazardous waste. This is particularly true of some metals such as lead, cadmium and beryllium. This has been investigated in conjunction with both the high-level waste (spent fuel) and the low- and intermediate-level waste. Hazardous substances have been inventoried and methods for assessing the safety of a repository with regard to these substances have been tested. SKB has found that the deep repository also provides very good protection against these substances /5.6-20/.

5.7 NATURAL ANALOGUES

5.7.1 Natural analogues and safety assessment

There is now a considerable body of results from analogue studies that can be used to assess the long-term safety of the repository. Such information has previously been used more sporadically for safety assessments such as KBS-3, SKB 91 and SSR-SFR, and most of the applications relate to conditions in the near field, see Table 5.7-1 /5.7-1/.

Table 5.7-1. Processes and analogues studied within SKB's programme /5.7-1/.

Processes in the near field	Natural analogues	Safety assessment
Canister corrosion	7	KBS-3
Bentonite stability	8 (3,4,6)	KBS-3
Concrete influence	6,11,12	
Fuel corrosion	3,4,5,9	KBS-2, SKB-91
Radiolysis	3,4	KBS-3, SKB-91
Formation of redox front	1	
Radionuclide solubility	1,3,6	
Radionuclide migration		
– colloids	1,3,6 (4)	
– organic matter	1,3,6 (4)	SFR
– microbes	1,3,6 (4)	
Processes in the far field		
Radionuclide solubility	1,3	
Radionuclide migration		
– colloids	1,3,5,6 (4)	SKB-91
– organic matter	3 (4,5,6)	
– microbes	3 (4)	
Radionuclide retention		
– absorption	1,2,13 (4,5)	
– co-precipitation	1,2,3,6 (4,5)	
– matrix diffusion	3,5 (4,6)	SKB-91

1. Poços de Caldas
2. Alligator Rivers
3. Cigar Lake
4. Oklo
5. Palmottu
6. Maqarin
7. Copper objects
8. Bentonite deposits
9. Uraninite samples
10. N Sweden (drill cores)
11. Porjus (old concrete)
12. Uppsala (old concrete)
13. Äspö (fracture filling)

These and similar examples are described in a book on which SKB has collaborated together with NAGRA and NIREX /5.7-2/. A more detailed account of some selected applications of analogues within the safety assessment has been prepared in collaboration between TVO and SKB /5.7-3/.

SKB is participating and has participated in a number of international analogue projects and is also represented in the international National Analogue Working Group, NAWG, which has been organized by the EU in cooperation with the US DOE. Its purpose is to discuss and evaluate the results of completed analogue studies.

5.7.2 Cigar Lake

The Cigar Lake project has studied a very concentrated, deep-lying and 1.3-billion-year-old uranium deposit in northern Saskatchewan in Canada. It has been studied by AECL as an analogue to a deep repository for spent fuel since 1984. SKB joined the project in 1989. This collaboration has now been concluded and the results reported /5.7-4/. Cigar Lake provides strong support for the deep repository concept in three areas in particular. The redox conditions in the deep groundwater have kept down the solubility for uranium and prevented it from escaping; the low hydraulic conductivity of the clay in and around the ore has slowed down the release of mobile nuclides; and radiolysis has had a moderate influence despite the long exposure to groundwater. The redox conditions can be described with the usual geochemical models and the release of mobile substances from the near field has been able to be described with mass transport models used in the safety assessment. The model developed to deal with the radiolysis of groundwater in contact with ore should also be able to be applied to the scenario of spent fuel in a damaged canister, see Figure 5.7-1.

5.7.3 Jordan

The Jordan Project is investigating the conditions in and around active hyperalkaline springs in Maqarin, see Figure 5.7-2, and similar fossil springs in central Jordan. Spontaneously formed combustion zones have generated portlandite and calcium silicates, which give the groundwater a pH of 12–13. A whole series of typical cement minerals have been formed as a consequence of the water's reactions with minerals in the area. The project was started in 1990 with funding from NAGRA, NIREX and Ontario Hydro. SKB has been participating since 1991. The results of the first phase have been published. A large part of the second phase involves testing (validating) chemical models for calculating the solubility of radionuclides. There are plenty of minerals in Maqarin, and one can see how the elements Sn, Se, Ni, Pb, Ra, Th and U dissolve in water with a high pH /5.7-5/. The project is now in its third phase and is currently being funded by NAGRA, NIREX, HMIP (Her Majesty's Inspectorate of Pollution) and SKB. The third phase is being coordinated and administered by SKB.

5.7.4 Oklo

The Oklo Project is investigating the natural fossil reactors at Oklo, Okelobondo and Bangombé in Gabon, Africa. The first reactor zone was discovered in 1972. Oklo is quite unique among the natural analogues. During uranium mining and prospecting in Oklo and the surrounding areas, no fewer than 15 different spots have been found with traces of nuclear reactions, i.e. reactor zones. It is the same type of nuclear fission as in a reactor with uranium fuel, and it has produced a large quantity of fission and activation products. This occurred nearly 2 billion years ago. The radioactive isotopes have since decayed and been transformed into predominantly stable products, but traces of the reactions remain, as do many of the materials that were involved. By tracing the daughter products, it is possible to see whether migration has occurred; for example, an elevated concentration of U-235 outside a zone indicates that diffusion of plutonium has occurred in the near field, which consists of clay minerals.

The project "Oklo, Natural Analogue for a Radioactive Waste Repository" has been carried out by CEA with financial support from the EU. SKB and several other organizations from various countries have been able to participate. SKB's interest has mainly been focused on the reactor zone in Bangombé, see Figure 5.7-3. It is suitable to study because it lies relatively close to the ground surface and far from the others (about 20 km), which means that no mining has yet taken place in Bangombé. The results of the first phase of the Oklo Project were published in the summer of 1995.

5.7.5 Palmottu

There is a uranium mineralization at Palmottu Lake in Finland. It forms a 1–15 m thick steeply dipping zone that extends to a depth of about 300 m down into the rock. It was discovered late in the 1970s and was thoroughly investigated, 62 exploratory holes having been drilled. The Palmottu analogue project started in 1988 and has been run by GTK (Geological Survey of Finland), with funding from STUK (Finnish Centre for Radiation and Nuclear Safety). The uranium deposit has been studied as an analogy to a deep repository for spent fuel in granitic rock. An advantage is that the conditions at Palmottu are the same as those that can be expected to exist at the sites chosen for deep disposal in Finland and Sweden. SKB has participated in the project as an "active observer".

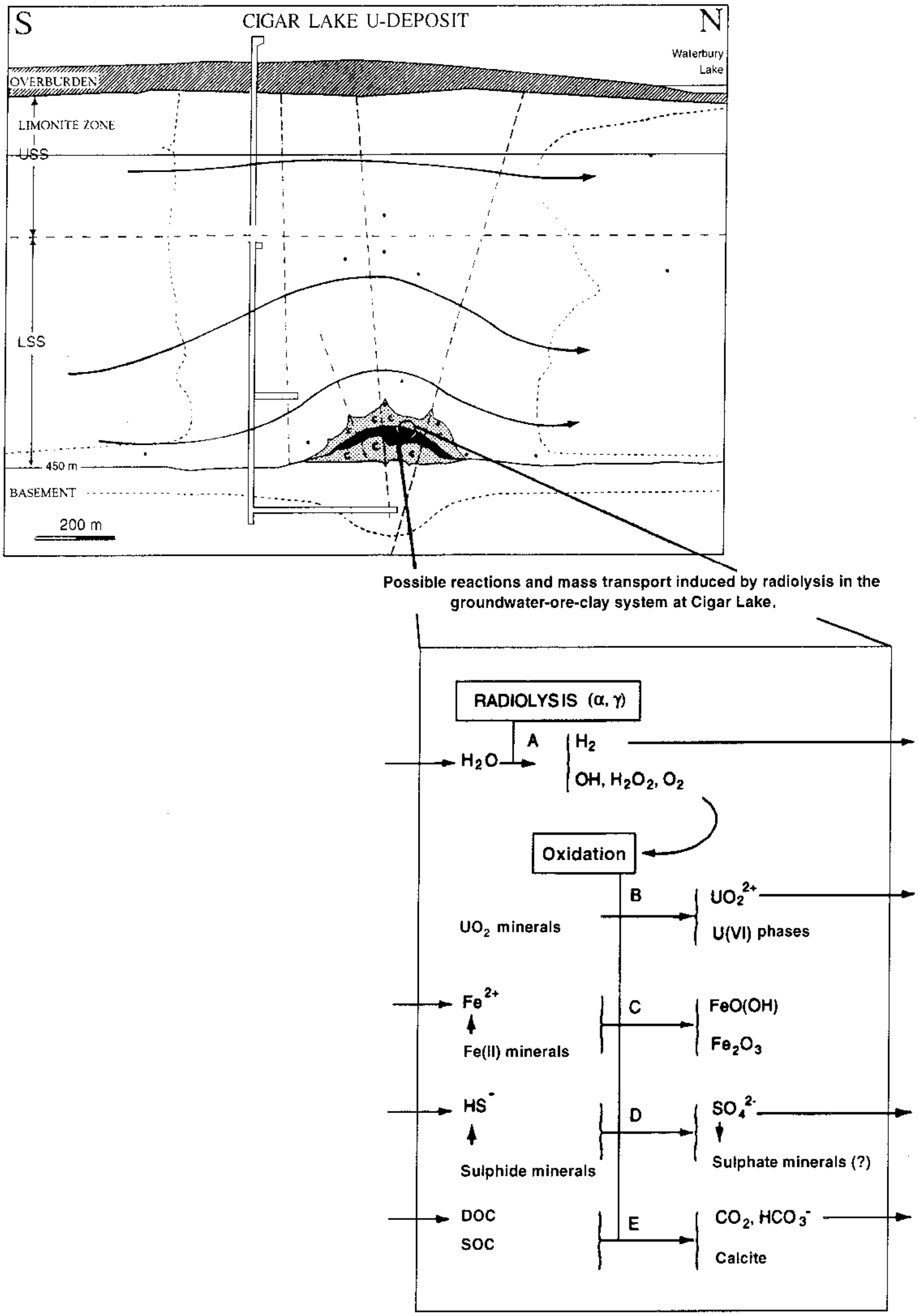


Figure 5.7-1. *Vertical profile through the uranium mineralization at Cigar Lake with the direction of the groundwater flow. The possible oxidation reactions that can be caused by radiolysis in the ore and the surrounding clay are shown schematically.*

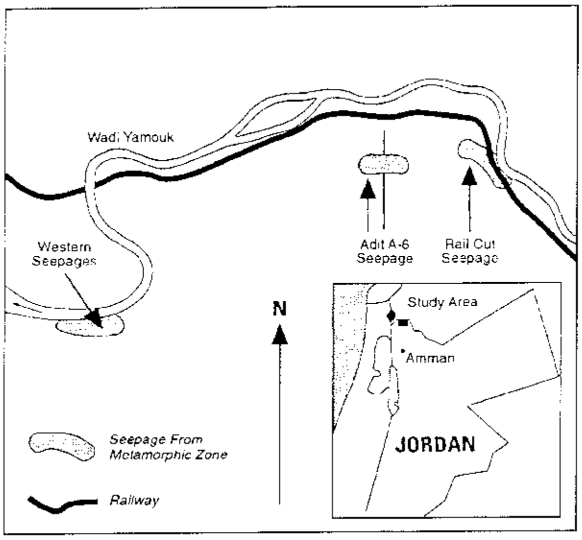


Figure 5.7-2. Location of the Maqarin analogue study site in northern Jordan. The sites of the hyperalkaline springs are shown.

5.7.6 Old concrete

Old concrete structures of Portland cement have been sampled to study how the cement paste changes with time. Wherever possible, an effort has been made to find samples that have been in a water-saturated environment. Where samples have been in air, carbonatization otherwise dominates, caused by the influence of the carbonic acid from the air. Portland cement was introduced at the start of this century. In other words it is not very old, but it is important to be able to discover trends towards changes in the samples, and some of the observations will hopefully be able to be verified by the results from the Jordan Project, which involves considerably longer periods of time. We now know more about which mineral phases are formed in cement and their properties. It should therefore be possible to shed light on what happens in the long term, provided that geochemical and hydrological conditions are given. The studies of old concrete have not been completed, but are expected to continue for some time yet.

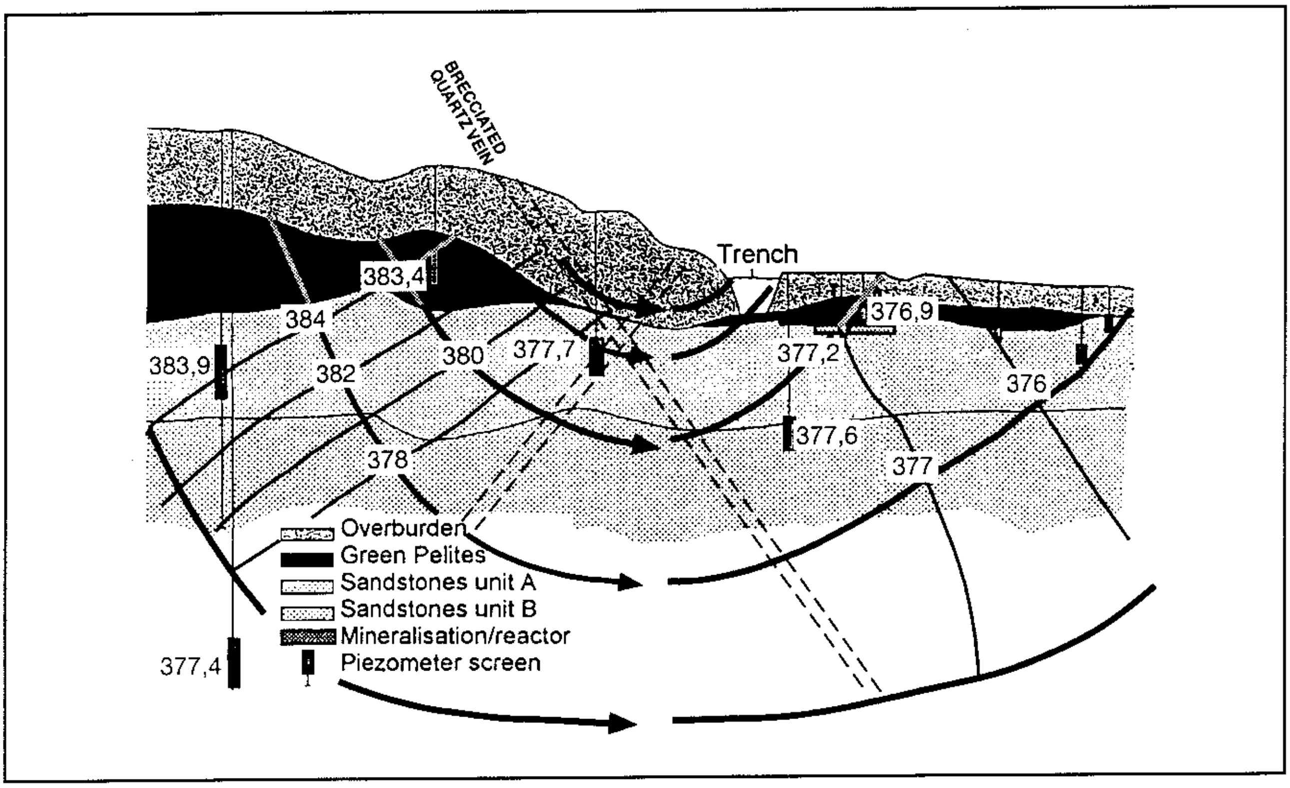


*Figure 5.7-3. The Bangombé site. Piezometer measurements, hydraulic gradient (—) and groundwater flow (**—**) directions are marked in the figure.*

5.7.7 Results in relation to goals in RD&D-Programme 92

- The Cigar Lake Project has been concluded and an independent evaluation has been completed. The latter is finished, but the results have not yet been published.
- SKB has participated in Phase I of the Oklo Project. As planned, we have increased our involvement in the project and are now participating as co-applicants to Phase II of the EU project.
- The third phase of the Jordan Project has begun and SKB is in charge of the project management.
- SKB has participated in the investigations at Palmottu.

5.8 THE BIOSPHERE

5.8.1 General

In order to be able to assess the radiological consequences of possible releases of radioactive materials from a final repository, it is necessary to describe transport from the rock to man (or to other biota). The overriding goal of SKB's studies of the behaviour of radionuclides in the biosphere is to be able to carry out credible consequence calculations in the safety assessments.

The state of knowledge for modelling the transport of radionuclides in the biosphere is relatively good. The goals of a joint international project initiated by the Swedish National Radiation Protection Institute called BIOMOVS /5.8-1/ are to:

- examine how the conceptual models have been converted into calculation models,
- compare different transport models,
- document the methods for defining transport scenarios.

Fundamental questions in biosphere modelling that have been identified within BIOMOVS and are the subject of continued work concern the handling of different scenarios for biosphere evolution and of critical groups. These questions are closely linked to the formulation of the radiation protection requirements.

As far as scenario methodology is concerned, the possibility of applying the RES method to the biosphere has been tried by a working group within BIOMOVS /5.8-2/.

5.8.2 Data needs

The predominant uncertainty in today's biosphere calculations is rooted in the availability and quality of data. Previously, generic data have been used almost exclusively. The reason given for this has been the great changes that occur in the biosphere within relatively short spans of time. This is correct for times after the start of the next Scandinavian ice age. For the next 1000 years in particular, however, site-specific databases and specific assessments of the frames within which the biosphere on a given repository site may change are judged to provide a basis for a meaningful forecast.

Analysis of the uncertainties in generic calculations show a confidence interval of several orders of magnitude /5.8-3/. For modelling which aims at showing that dose limits will not be exceeded, this can be acceptable, even though reported consequences are highly overestimated. If, on the other hand, the modelling aims at an assessment of **expected** impact, such as for example in comparisons between repository sites or in optimization of radiation protection, such results are meaningless, as a rule.

Today's food production system is so complex that even if the levels of radionuclides in the constituent raw materials are known, it is not possible to calculate the intake to a given individual (with known dietary habits) with sufficient accuracy. The large dilution factor (several powers of ten) entailed by production and distribution is often overlooked in the safety assessment.

5.8.3 Model development

SKB's modellings of radionuclide transport in the biosphere have been carried out with BIOPATH, a calculation program developed by Studsvik EcoSafe and included in the BIOMOVS work. BIOPATH has been utilized for KBS-3, SFR and SKB 91 and has been progressively refined through the efforts of SKB, among others. For a detailed discussion of how biosphere modelling is performed and how it is utilized in safety assessments, the reader is referred to SR 95 chapter 8 and /5.1-3/.

At the same time as biosphere evolution and the choice of a critical group will increasingly be coupled to site-specific conditions, the structure of the BIOPATH model will be adapted to local conditions as well. The development work that is planned will be aimed at improving the data on which the transport models rest and trying to validate the models through studies of analogous transport processes.

5.8.4 Results in relation to goals in RD&D-Programme 92

Interim goals in RD&D-Programme 92 were:

- **Try to quantify the uncertainties that stem from changes in the biosphere**
  The results of this work have been reported in /5.8-2 and 3/. The large dissimilarities in dose conversion

factors associated with release in the Baltic Sea and in the interior of the country are worth noting. The ratio between them is more than two orders of magnitude, with better conditions in the Baltic Sea. This is partly due to favourable transport pathways and dilution conditions, partly to long residence times in the sea sediments.

- **Improve the database on which the transport models rest**
  Site-specific data have been gathered and utilized in modelling /5.8-3 and 4/. Models for how K_d values for soils and sediments can be calculated have been developed /5.8-5/.

- **Validate models through the study of analogous transport processes**
  The work within BIOMOVS II provides a good basis for judging the validity of the transport models /5.8-6/, as do studies of the interface between geosphere and biosphere /5.8-7/. Studies of natural analogues have begun with an inventory /5.8-8/.

5.9 OTHER WASTE

According to plans, long-lived LLW and ILW is supposed to be disposed of in a separate part of the deep repository called SFL 3, 4 and 5. Long-lived waste From Studsvik will be disposed of in SFL 3, along with such LLW and ILW that could otherwise go to SFR, but that arises after SFR has been closed and sealed. Decommissioning waste from CLAB and the encapsulation plant will be deposited in SFL 4, and reactor core components in SFL 5. The waste in SFL 3–5 is similar in nature to that which goes to SFR. It is mainly the higher content of long-lived nuclides, e.g. plutonium isotopes and nickel-59, which warrant deeper disposal in a separate repository.

The studies that have been carried out for safety assessments of SFR serve as a basis for the work with long-lived waste. Experience from the ongoing investigation work described below will also be applied to SFR.

5.9.1 Prestudy

A prestudy of the performance of the barriers in SFL 3–5 /5.9-1/ has been carried out. It was begun early in 1993 and concluded late in 1994. The goal was to make an initial preliminary assessment of the near-field barriers against the escape of radionuclides. The point of departure was the conceptual design of the repository presented in PLAN 93 /5.9-2/. Hydrogeological and other site-related premises were chosen with the aid of simplified assumptions and previous experience.

5.9.2 Inventory of the waste

The waste has been inventoried and characterized within the framework of the prestudy, see Figure 5.9-1 /5.9-3/. The quantity of radionuclides of various kinds has been estimated, along with some other components of importance for safety, such as metals, organic matter, concrete, etc. The most common metal in the waste is steel, primarily stainless steel, which will be found in all parts of the repository. Organic matter is concentrated to SFL 3.

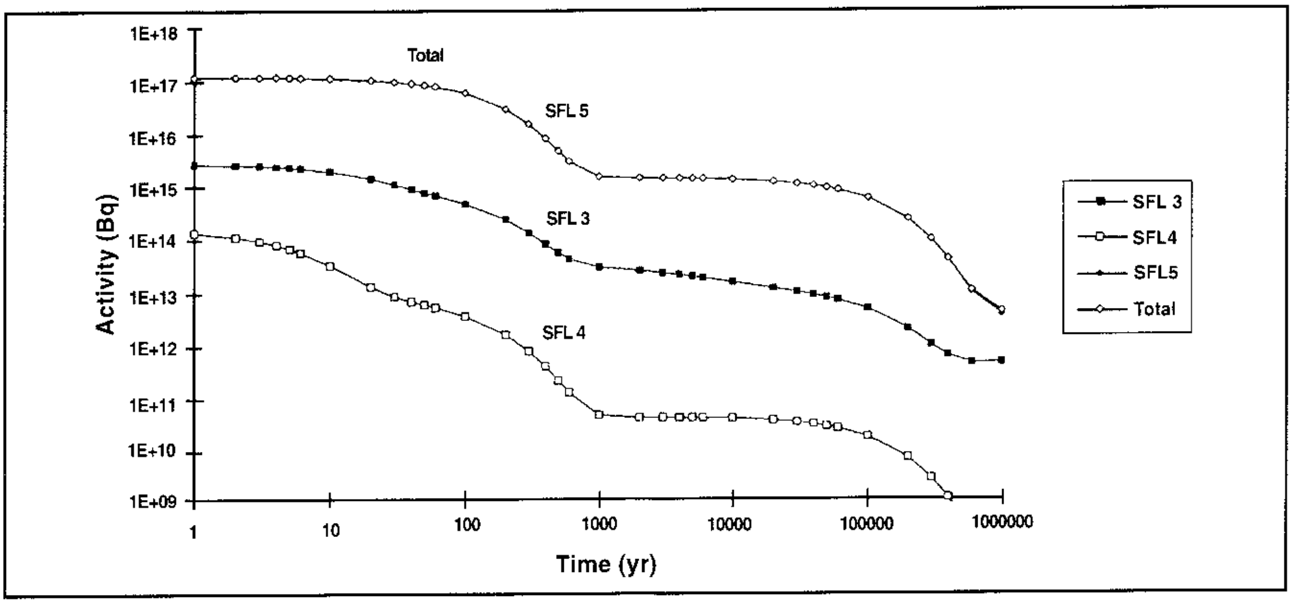


Figure 5.9-1. Content of activity (Bq) in the different parts of the repository as a function of time (from the year 2040).

5.9.3 Laboratory investigations

Laboratory investigations have begun with the goal of producing the necessary data and models to assess the performance of the barriers. The work is long-term and mainly focused on radionuclide chemistry in the repository (solubility, sorption, diffusion, etc.), the influence of organic substances and changes in the concrete. The following investigations are being conducted during 1995:

- Compilation of the state of knowledge in the area of concrete stability under repository conditions.
- Leaching of crushed cement paste with normal versus saline groundwater. The ions being analyzed are: Na^+, K^+, Ca^{2+}, Mg^{2+}, OH^-, SO_4^{2-}, Cl^-, Al_{tot} and Si_{tot}. The leached cement paste is examined after leaching to see which secondary minerals are formed.
- Solubility measurements of Ni, Pu, Eu and Th in cement environment.
- Measurements of radionuclide sorption on concrete (K_d values). The elements studied are: Th, Eu, Cm, Pm, Co, Ra and Ni (tetravalent, trivalent and bivalent elements).
- Measurements of radionuclide diffusion in cement paste and cement mortar. The elements studied are: Ni, Cs and tritium. Both diffusion profiles and diffusion are being studied.
- Measurements of radionuclide diffusion through a mixture of 15% bentonite (Wyoming MX-80) and 85% sand. The elements studied are: Cs, Tc and Ni. The influence of cement pore water on the diffusion is also being measured.
- Studies of how cellulose (crystalline cellulose, industrial products such as wood, paper, cotton, cement additives, etc.) are degraded at high pH values (10–13.5) under anoxic conditions; identification of degradation products, including strong complexing agents of the type polyhydroxycarboxyllic acid (especially isosaccharinic acid).
- Degradation of isosaccharinic acid at high pH values (10–13.5); long-term stability.
- Complexation capacity of isosaccharinic acid, especially with trivalent elements.
- Diffusion of isosaccharinic acid through cement.
- Influence of isosaccharinic acid and degradation products from cellulose on the absorption of radionuclides on cement. Experiments are being performed with trivalent and tetravalent elements and nickel (bivalent).
- Influence of organic complexing agents on diffusion of radionuclides through cement.

Similar work is being done in other countries and SKB has therefore started informal cooperation with ANDRA, NAGRA and NIREX. Results and experience have been applied in the prestudy /5.9-4/.

5.9.4 Performance assessment of near-field barriers

To test the ability of the barriers to retain and limit the release of radionuclides, transport calculations were carried out – simple calculations with a tank reactor model for all nuclides and calculations of advection-diffusion in concrete, bentonite and sand for specially selected nuclides. A couple of toxic metals were also included in the study. Calculations and results are described in connection with the prestudy /5.9-4/.

The prestudy has entailed an initial estimate and characterization of the waste that will go to SFL 3–5. Modern systematic scenario methodology has been tested /5.9-5/ and an initial assessment of the barriers has been carried out. The results show that the metallic waste from the reactors contains the most radionuclides. This waste will be collected in SFL 5. At the estimated time of closure, the activity in SFL 5 will be more than 10 times as high as in SFL 3, which will in turn be more than 10 times as high as in SFL 4. The calculated release of radionuclides is so low that the near field alone would suffice as a barrier. The results suggest that the proposed design of the repository is purpose-suited and provides large safety margins. Several scenarios remain to be analyzed, however, for example long-term evolution of the barriers, consequences of gas generation, ice age scenarios, etc. So far the prestudy has served its purposes: to estimate waste quantities and contents, to test the conceptual design, to focus the basic investigations on relevant issues, and to prepare for future safety assessments.

5.9.5 Results in relation to goals in RD&D-Programme 92

The goals in RD&D-Programme 92,

- to inventory and characterize existing and foreseen waste of this type,
- to modify the design of the deep repository for LLW and ILW, and
- to prepare for and gather data for necessary safety assessments,

have been pursued within the concluded prestudy and continue to be pursued in the ongoing second phase. The purpose of the latter is to gather data for a safety assessment that will form a part of the safety report in support of an application for a permit for detailed characterization for the deep repository. Laboratory experiments have been started as planned and active cooperation with NAGRA, NIREX and ANDRA has provided good guidance and a valuable exchange of information on chemical conditions etc.

6 STATE OF KNOWLEDGE - CANISTER AND ENCAPSULATION

6.1 PREMISES

Before the spent fuel is emplaced in the deep repository, it must be encapsulated in a durable canister. The programme for canisters and encapsulation mainly comprises development and fabrication of canisters and design and construction of an encapsulation plant. This work is designated "Encapsulation Project".

The canisters must be designed and fabricated in such a manner that they remain intact for a very long time in the conditions that will prevail in the deep repository. This means that they must not be penetrated by corrosion when exposed to the groundwater present in the rock, or be broken apart by the mechanical stresses to which they may be subjected in the deep repository.

To achieve this, it is planned that the canister will consist of an insert of e.g. steel, which provides mechanical strength, and an outer copper canister which provides corrosion protection. Copper corrodes very slowly in the oxygen-free water present at depth in Swedish bedrock. Studies have shown that the canister will probably remain intact for millions of years, which provides a considerable margin of safety (which is needed due to uncertainties and variations in the design assumptions).

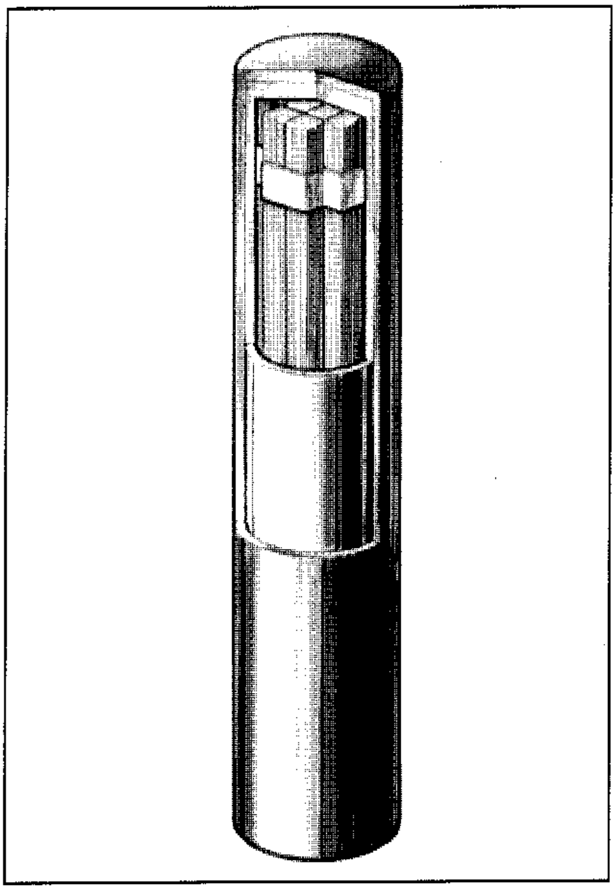


Figure 6-1. Copper canister with pressure-bearing steel insert.

Previously, other designs of the canister have also been studied, for example a solid copper canister which is fabricated by means of hot isostatic pressing, and a copper canister in which the void around the fuel is filled with lead. However, both of these alternatives require that encapsulation be performed at high temperature, which is avoided with the currently planned design of the canister. The canister's insert has positions in which the fuel assemblies are placed and the lid is sealed without causing any great heating of the fuel. This "cold process" limits the risks associated with encapsulation, which has been crucial for the choice of this option, since the long-term performance of the three canister types is equivalent.

Encapsulation is planned to take place in a new plant connected with CLAB. In the encapsulation plant, fuel from CLAB's storage pools will be placed directly in canisters. After having been checked and dried, it will be placed in the canister. Before the lid is placed on the inner steel container, the air in the canister will be replaced with inert gas. Then the copper canister will be sealed by the welding on of a lid. The requirements on this weld in terms of leaktightness and the ability to demonstrate leaktightness are exacting. The lid is planned to be sealed by means of electron beam welding.

In designing the encapsulation plant, a great deal of emphasis will be placed on radiation protection for the personnel and the environment. This means, among other things, that the actual encapsulation procedure will be performed by remote control in heavily radiation-shielded compartments, called hot cells. A large part of the handling of canisters will also be done by remote control. Experience from CLAB and SFR, as well as from various foreign facilities, will be drawn upon.

In a later phase, other long-lived waste will also be treated in the encapsulation plant. Examples of such waste are core components, e.g. control rods, and other reactor internals that have become activated by neutron bombardment during reactor operation. These components are planned to be embedded in concrete. Certain preparations for the later installation of such equipment will be made when designing and building the encapsulation plant.

The encapsulation project also includes expansion of the storage capacity of CLAB. With the current storage pools, it is estimated that CLAB will be full by around 2004, and additional capacity is needed to be able to accommodate all fuel from the Swedish nuclear power stations. This part of the project is not described in this programme.

This chapter provides a status report on the encapsulation project and an overview of the state of knowledge concerning the canister and the encapsulation process. Chapter 7 presents a programme for SKB's forthcoming work in the area.

6.2 DEVELOPMENT AND DESIGN OF CANISTER

6.2.1 General

The work of designing the canister is planned to take place in steps through the compilation of basic premises, requirements on properties and criteria for sizing and design. These compilations, combined with experience from practical trials of canister fabrication and sealing, will then serve as a basis for the final choice of canister design.

Basic premises, requirements on properties and criteria will be established with the aid of the results of assessments of both long-term safety in the deep repository and of safety in the operation of the encapsulation plant and the transport system. The starting points for this work have been discussed in chapter 4. A status report on this work is presented in the following sections.

6.2.2 Requirements on performance and properties

Safety during operation of the encapsulation plant and transport and deposition of canisters, as well as long-term safety of the deep repository, impose requirements on the performance of the canister. It must be possible to fabricate, fill, handle and seal the canisters in a safe manner, and they must remain intact and impervious for a long period of time after deposition.

Based on these fundamental premises, the requirements on the properties of the canister can be broken down into three parts: long-term safety and performance in the deep repository, fabrication and handling, and economy and environment.

Long-term safety and performance in the deep repository

The fundamental principle for safety in the deep repository is to isolate the spent fuel. This isolation is achieved by enclosure in leaktight canisters, which are deposited in the crystalline bedrock on a selected site. This requires that the canisters be leaktight when deposited and remain leaktight over a long time. The canister must therefore be capable of withstanding the mechanical and chemical stresses to which it will be subjected.

Safety in the deep repository is based on the multibarrier principle, which means that safety must not be dependent on a single barrier. In the event of canister failure, the other barriers must prevent or retard the dispersion of radionuclides to acceptable levels. Material choice and design of the canister must then not have an adverse effect on the performance of the other barriers.

The canister must thus meet two primary functional requirements in order to provide the necessary isolation in the deep repository:

a. The canisters must **retain** their integrity over a long time, which in turn imposes requirements on

   - initial integrity,
   - corrosion resistance, and
   - strength.

b. The canisters **must not have any harmful effect** on the other barriers in the deep repository, which imposes requirements on

- choice of material that does not adversely affect the buffer and rock,
- limitation of heat and radiation dose in the near field,
- design so that the fuel remains subcritical even if water enters the canister, and
- limitation of the bottom pressure on the bentonite.

Industrial safety, fabrication and handling

Safety during operation and maintenance work at the encapsulation plant must be high. The canister's design must meet the requirements made by both normal and abnormal operating cases in the plant. It must also be able to withstand handling accidents without subjecting the personnel and the plant to unacceptable exposures or leading to unacceptable releases of radioactivity.

During transport to the deep repository, the canister is placed in a transport cask which provides protection against external damage.

For the work of deposition in the deep repository, the canister must be designed so that it can be transferred from the transport cask to the deposition equipment. Emplacement in the deposition holes must be able to be performed with the necessary precision and safety.

In the event of retrieval of canisters after the initial deposition stage, it must be possible to lift the canisters and place them in a transport cask.

The overall property requirements with respect to fabrication and handling are that the canister should be designed to be able to be

- fabricated and inspected in serial production with adherence to stringent quality requirements,
- handled, sealed and inspected in the encapsulation plant,
- transported to the deep repository,
- handled and deposited in the deep repository, and
- retrieved (if required) from the deposition holes.

Economy and environment

The safety goals should be fulfilled with observance of good resource management and in consideration of the environmental consequences of canister fabrication and the encapsulation procedure.

The selected canister material must not have any harmful effect on the environment.

6.2.3 Criteria for sizing and design

With the above basic premises as a guide, criteria for sizing and designing the canister have been compiled. The criteria may be modified later in reference to the final design of the repository.

Integrity

Integrity over the long term is derived from three sources, namely: initial integrity, corrosion resistance and strength and specified as follows.

a. Initial integrity
The canisters must be fabricated, sealed and inspected with methods that guarantee that less than 0.1% of the finished canisters will contain undetected defects that could entail initial leakage or that could lead to early canister failure.

b. Corrosion resistance
The canister must not be penetrated by corrosion in order to prevent water from entering the canister during the first 100,000 years after deposition. As is evident in section 6.3.2, the maximum corrosion depth after 100,000 years is estimated to be about 5 mm in typical Swedish groundwater and with the selected canister design /6-1/. In recognition of uncertainties in the data and other factors, a suitable safety factor should be included when determining the wall thickness. Moreover, a minimum total thickness of the canister wall is required so that the corrosion rate will not increase as a result of radiolysis.

c. Strength
The canisters must be designed to withstand deposition at a depth of 400–700 m, which entails an evenly distributed load of

- max. 7 MPa hydrostatic pressure and
- approx. 10 MPa pressure from bentonite and rock.

Designing for these "normal" and evenly distributed loads is done in observance of the customary safety margins.

Check calculations must also be carried out on the canisters for possible uneven loading (bending) due to uneven density distribution or water absorption in the bentonite, causing an uneven build-up of swelling pressure. Uneven loading (shearing) might also be caused by rock movements.

Check calculations are also performed for increased hydrostatic pressure (20–20 MPa), which can arise during a future ice age.

The design calculations must show that the canisters can withstand the prescribed stresses. The required safety margins may be lower for these cases, however.

Material changes that might have a long-range effect on strength are taken into account in the design process. Mechanisms that could cause such material changes are:

- Internal corrosion (due to radiolysis of any trapped air molecules).
- Grain growth in the canister material.

The radiation dose received by the canister during 100,000 years is several orders of magnitude lower than that required for the material properties of copper and steel to be affected.

Impact on the barriers in the deep repository

The canisters must be designed so that they do not have a detrimental effect on the performance of other barriers in the deep repository. This requirement means that the following must be taken into account when designing the canister.

a. Choice of material that does not affect buffer and rock
The canister material and the corrosion products must not appreciably degrade the performance of the buffer. Moreover, the canister material and canister design must be such that they do not appreciably degrade the performance of the other barriers in the event of canister penetration. This could conceivably occur as a result of unacceptably high gas evolution.

b. Limitation of heat transfer to surrounding bentonite
The canisters must be designed so that the temperature in the bentonite nearest the canister does not exceed about 90°C. This entails a limitation of the canister's maximum surface heat output.

c. Limitation of radiation dose to bentonite
The canister must provide sufficient radiation protection so that the contribution made by radiolysis products to the corrosion of copper is small compared with the corrosion caused by residual oxygen in the system. This limits the permissible surface dose rate to approximately 500 mGy/h /6-2, 3/.

d. The canister must be designed so that the fuel remains subcritical even in the event of water penetration
This influences the design of the canister insert and the need for a filler material.

e. The canister's bearing pressure against the bentonite must be limited so that the canister does not sink down through the bentonite
Calculations show that this can be achieved for the different canister designs studied.

Fabrication and handling

The determination of the design requirements for fabrication and handling of canisters can be divided up in the following manner:

a. Fabrication and inspection of unfilled canisters
- Reliable methods must be developed for fabrication and inspection of the necessary canister production with satisfactory quality. Depending on the choice of method, this leads to analysis of the planned fabrication chain with respect to quality requirements and necessary inspections, such as:

- microstructure of the material,
- porosity and surface finish,
- strength properties and
- inspection of fabrication welds etc.

b. Transport to the encapsulation plant and arrival inspection
- Canister with "packaging" must withstand transport without damage.
- Methods must exist for examination and inspection in order to determine the canister's condition after transport.

c. Handling of canister in encapsulation plant
- The insert in the canister must be designed so that existing types of fuel assemblies can be emplaced using the equipment in the plant.
- The lid for the insert must be designed to meet the requirements of the encapsulation process for a change of atmosphere in the canister and leaktightness on sealing of the copper lid.
- Radiation from the canister does not have to be limited for the sake of handling in the encapsulation plant, since the canister is provided with extra radiation shielding for different work operations.
- The canister must be designed for the necessary handling and lifting operations.
- The canister must be provided with clear identification.

d. Sealing and postweld machining
- The copper lid must be designed so that it can be sealed using electron beam welding and so that inspection of the weld can be performed.
- The weld area must be designed to meet requirements on surface finish after machining.
- The canister must be designed so that the outside can be decontaminated if necessary.

6.2.4 Reference canister

The work with the canister and the encapsulation process has focused on studies and development of technology that does not contribute to heating of the fuel during encapsulation. Such technology facilitates the encapsulation process and reduces the radiological risks for the operating personnel. The canister is planned to be composed of two components: an outer corrosion protection of copper and an inner pressure-bearing container of steel so that it will fulfill its function in the deep repository.

The development work on the canister and the encapsulation process has resulted in a slightly modified design for the inner container. Previously it was planned to fill the space around the fuel assemblies with sand or glass beads in an inner container in the form of a steel cylinder. This involved a technically complicated work operation. To avoid this, an alternative with a cast inner container for the copper canister has been studied, see Figure 6-2. This container replaces both the steel cylinder as a pressure-absorbing component in the canister and the insert that was required to guide the assemblies. The inner container is cast in one piece with holes for the different types of fuel assemblies. It is assumed that it will be cast in steel, iron or perhaps some other material. This design comprises the reference for the continued work.

The exact size of the canister and choice of material grades will be studied in the continued work and will be chosen with a view towards the criteria described above. A copper canister with a 50 mm wall thickness made of oxygen-free copper with a low phosphorus content is being used for the time being as a basis for the continued work. The bottom and lid are joined to the shell by

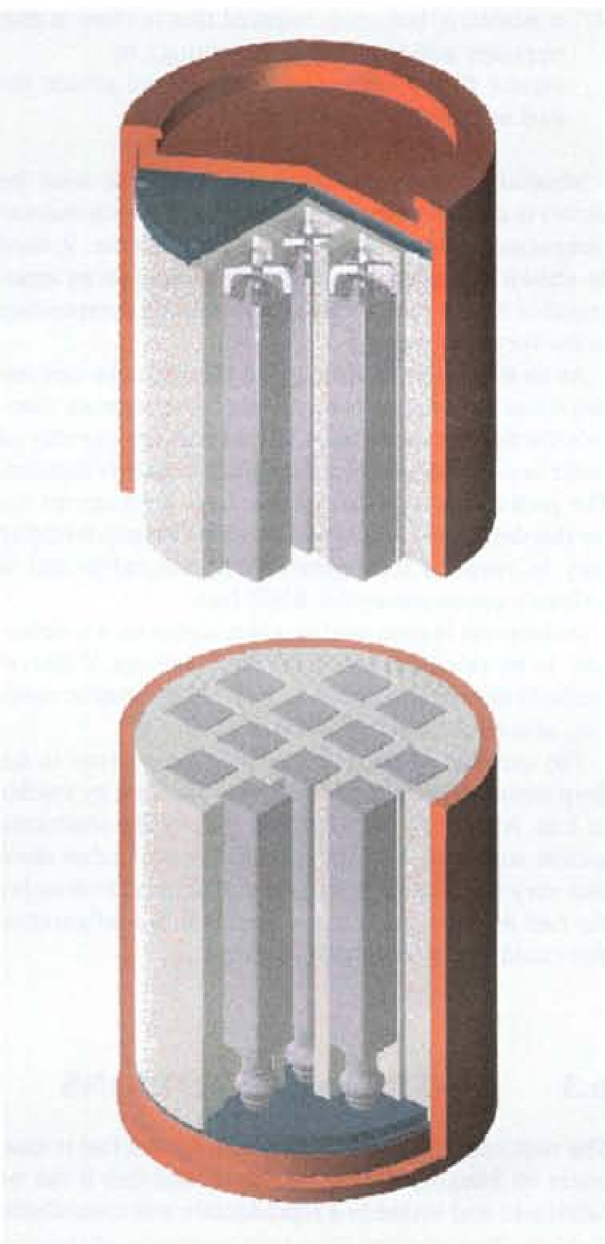


Figure 6-2. Copper canister with cast insert.

electron beam welding. The canister insert is of cast steel and has a minimum wall thickness of 50 mm.

Several alternative designs of the copper canister have been studied and will be further studied.

- Copper canister with inner component of cast steel (reference canister).
- Copper canister with inner component in the form of a steel cylinder, sized to withstand the external pressure.
- Copper canister with an inner component of spheroidal graphite iron with the same design as the reference canister.
- Copper canister with an inner component of cast bronze with the same design as the reference canister.

The reference alternative for the time being is a canister as described above with a cast steel inner component. Trial fabrication of a cast steel inner container is planned within the near future.

The alternative of an inner steel cylinder has been investigated thoroughly and trial fabrication has been carried out, see section 6.4. The weakness of the design is that the large empty space inside the canister must be reduced with a filler material to avoid the risk of criticality if water should enter the canister. Handling and addition of filler material complicates the encapsulation procedure. This operation can be avoided entirely if a cast inner component is used.

The alternative of a cast inner component of spheroidal graphite iron (Swedish Standard SS 0717) is identical in all essential respects to the reference alternative. Spheroidal graphite iron has better castability, but its weldability is poorer. The latter does not have to be a disadvantage, since there are good prospects for casting the inner component in one piece, which would greatly reduce the requirements on weldability.

A bronze inner component would eliminate all possibilities of hydrogen gas formation in conjunction with corrosion, if the copper shell should be penetrated. This is judged to facilitate performance assessment for a leaking canister, but the canister cost is higher for this alternative.

Background to choice of copper canister

Several materials possess good corrosion resistance in water. This may be due to the thermodynamic stability of the material itself or of a surface film of corrosion products (passivation), or to a low corrosion rate. Complete immunity to corrosion in groundwater can be guaranteed by the use of noble metals and ceramics, for example aluminium oxide and titanium dioxide. Noble metals are out of the question for cost reasons, and ceramic canisters are difficult to fabricate and seal. The brittleness of the material also makes them sensitive to handling accidents. Previous investigations have also indicated uncertainties in the estimation of the risk of delayed fracture in these materials.

Canisters made of metals which corrode in water (even at a low corrosion rate) will have a limited lifetime, which is determined by the thickness of the canister. The life of canisters of passive materials will be dependent on the passive film remaining stable and undamaged over long periods of time. Copper is immune to corrosion in oxygen-free water, and the life of a copper canister will be determined by the quantity of dissolved corrosive substances, mainly sulphides, in the groundwater that is transported to the canister surface (and by the form of the corrosion attack).

Water at great depth in Swedish bedrock is oxygen-free and contains very small quantities of sulphide. A copper canister will therefore be capable of achieving a very long life in the repository environment.

Due to the low solubility of copper and copper sulphide in water, the canister will have an insignificant effect on the buffer, even after very long periods of time.

6.2.5 Canister size

The size of the canister will be determined with a view to the limitations of the handling, transportation and deposition systems, and well as to the fact that the temperature on the canister surface in the deep repository must not exceed 90°C. The latter factor leads to certain degrees of freedom in the balance between canister size, fuel heat output (which is dependent on burnup and decay time) and the geometric configuration of the deep repository (mainly the spacing between the canisters). Certain variation calculations for canister size were carried out in the PASS study /6-4/, after which a reference canister for up to 12 BWR assemblies or 4 PWR assemblies was chosen. Further studies are under way to determine a suitable canister size. They include, among other things, analyses of how sensitive the temperature is to variations in the thermal conductivity of the bentonite and the rock, rock temperature, repository configuration and deposition schedule.

6.2.6 Criticality questions

The canister must be designed so that there is no risk of criticality in connection with handling of the canister with fuel in the encapsulation plant or in the long run in the final repository if water should enter the canister. The reference canister holds 12 BWR fuel assemblies or 4 PWR assemblies. With the enrichments that are used today, a critical configuration can be achieved if the fuel's burnup is low enough. Various methods can be used to avoid the risk of criticality /6-5/:

- Only fuel with a guaranteed minimum burnup is placed in the canister.
- The void between the fuel assemblies if filled with material that reduces the available water moderator.
- Neutron-absorbing materials are used in the canister and filler material.

Preliminary calculations have been performed of the reactivity in the canister for different fuel and filler configurations /6-6/. Calculations for a canister without insert show that for fuel with the maximum enrichment approved for CLAB:

- a minimum burnup is required that is close to that normally achieved (burnup crediting), or
- careful filling with particulate material around the fuel assemblies is necessary.

Mindful of long-term criticality safety, it must be shown in the latter case that the material does not become compacted or dissolve with time. Furthermore, it must be shown that reactivity does not increase to an unacceptable degree due to changes in isotope composition in the very long term.

As an alternative to filling with particles, the canister insert can be designed from the start with separate channels for the fuel assemblies. In this way the quantity of water around the fuel in a damaged canister is reduced. The preliminary calculations that have been carried out for this design /6-7/ show that a smaller burnup crediting may be required for modern PWR fuel, while this is probably not necessary for BWR fuel.

In the event burnup crediting is resorted to, it is necessary to be able to measure the fuel's burnup. Different methods have been compiled for this /6-8/. Similar needs may also exist for safeguard reasons.

The question of criticality outside the canister in the deep repository has been brought to the fore by studies in Los Alamos /6-9/. SKB has previously conducted similar studies /6-10/. The results of these studies show that very improbable processes are required to dissolve the fuel and then get it to precipitate in a configuration that could lead to criticality /6-11/.

6.3 MATERIAL QUESTIONS

The requirements on the copper canister are that it must retain its integrity over a long time, and that it can be fabricated and sealed in a reproducible and controllable fashion. The important long-term properties of the copper material are its corrosion properties and its creep properties. The grain size of the material is also important for inspection of the sealed canister, and impurities and alloying elements are of importance for weldability.

The main requirement on the canister insert is that it should lend sufficient mechanical strength to the canister.

6.3.1 Investigated materials

Copper canister

The principal strategy has been to select copper which is as pure as possible. This choice has been determined by the requirements on corrosion resistance. Pure, oxygen-free copper has, however, been found to have reduced creep ductility at elevated temperatures. This phenomenon can be avoided by the addition of 40 to 60 ppm phosphorus. The addition of phosphorus also led to an increase in the recrystallization temperature from 140°C to 250°C. The increase of the recrystallization temperature limits grain size. No increased sensitivity to stress corrosion has been observed at phosphorus concentrations in this range /6-12, 13/.

A pure copper with the following levels of additives and undesirable impurities is being used for the time being as a reference material for tests of fabrication and sealing of the copper canister:

P: 40–60 ppm, O: <10 ppm, S: <6 ppm, H: <2 ppm.

The desired maximum grain size in the material is about 250 μm.

Alternative copper alloys have been discussed, for example CuCr, CuZr and CuSn. Chromium and zirconium cause a very large increase in the recrystallization temperature, perhaps 400°C, but are very reactive in a copper melt, and it is therefore difficult to accurately control the quantities added. The use of these alloys would considerably complicate the fabrication of the copper canister. Tin is an alloying element that is often used in copper. Concentrations in the range 0.1–0.2% cause an increase in the recrystallization temperature by 175–200°C and could be an alternative to phosphorus. The possibilities of using alternative alloys will be further explored.

Inner container

Sufficient compressive strength and ductility can be achieved with several different materials, which also have sufficient toughness to withstand handling accidents. Low-alloy pressure vessel steel provides fully adequate strength for a self-supporting structure (steel tube). For a cast insert with inner support, different types of cast metals can provide acceptable properties, e.g. cast steel (SS 1306), cast iron (e.g. SS 0717) or bronze. The choice of material will be based in part on forthcoming fabrication trials.

6.3.2 Results of material investigations

Material investigations have only been carried out for the copper material. For the steel component, sufficient information is available in the literature.

Corrosion

General

The corrosion properties of copper are well known. An appraisal of the state of knowledge with regard to copper corrosion is provided in /6-14/. The conclusions are that it is highly unlikely that general or local corrosion will be a limiting factor for the life of the canister in the deep repository. Areas identified for further investigation include stress corrosion cracking.

New information has emerged during the period suggesting that the possibility of bacterial corrosion should be further investigated as well. Furthermore, the possibility of local sulphide corrosion due to formation of sulphide "whiskers" has been discussed in the expert review of RD&D-Programme 92 /6-15/.

Water chemistry

The outside of the canister must be resistant to corrosion so that water does not enter the canister during the stipulated period of time in the repository environment. The corrosion attacks may be of a general or local nature and are controlled, as far as external corrosion is concerned, by the chemical environment in the canister's immediate near-field, i.e. the composition of the bentonite pore water. This in turn is controlled by the interaction between the bentonite and the groundwater in the surrounding rock. Preliminary assumptions have been used to describe the hydrochemical environment during different periods in the deep repository. Table 6.2-1 shows typical values for carbonate-containing groundwater with low chloride concentration equilibrated with bentonite.

Table 6.2-1. Typical values (mmol/dm^3) for carbonate-containing groundwater with low chloride concentration equilibrated with bentonite.

Na^+	Ca^{2+}	Cl^-	SO_4^{2-}	HCO_3^-/CO_3^{2-}	pH
90	9.2	1.8	33	3	9.3

The chloride concentration in the groundwater varies within very wide limits, ranging from about 0.15 mmol/dm^3 to about 1.5 mol/dm^3. The chloride concentration is not affected by the bentonite, which means that equivalent values are also obtained in the bentonite pore water. Dominant cations are Na and Ca, which thus also vary with the chloride concentration. For other dominant anions, the range of variation in groundwater lies within the bentonite pore water's range, which will control the levels of these ions for the foreseeable future.

Virtually all investigated groundwaters have been found to be reducing. A typical Eh value is -300 mV. This applies to undisturbed conditions. After the sealing of the deep repository, there will be $<3 \cdot 0^{-4}$ mol/dm^3 dissolved residual oxygen in the groundwater. The sulphide concentration is projected to be 10^{-5} mol/dm^3.

Conclusions regarding corrosion resistance are not judged to be appreciably affected by the local variations that can be accepted for the choice of repository site.

Forms of corrosion
The following factors must be taken into account in judging corrosion in the repository:

- External corrosion in groundwater under oxidizing conditions as long as initial oxygen remains around the canister.
- External corrosion in groundwater under reducing conditions.
- External corrosion caused by radiolysis products.
- External corrosion caused by microbial activity.
- Stress corrosion cracking.
- Internal corrosion caused by residual oxygen, residual water and radiolysis products in the canister.
- Internal stress corrosion cracking.

External corrosion: oxidizing/reducing conditions
For copper, corrosion processes that can lead to canister penetration have been studied and the required wall thickness has been calculated for relevant cases. The calculations have been based on conservative assumptions. However, in view of uncertainties in the data and the risk of unknown local variations in conditions, the wall thickness should be chosen for the greatest depth of corrosion damage obtained in these calculations cases, with the addition of a safety factor.

Modelling of corrosion processes in copper shows the following corrosion depths after 100,000 years for a probable case and a conservative case /6-1/:

	Probable case	Conservative case
– General corrosion		
In the presence of oxygen	0.5 μm	5 μm
Oxygen-free conditions	5 μm	0.4 mm
– Pitting corrosion		
In the presence of oxygen	250 μm	2.5 mm
Oxygen-free conditions	10 μm	2 mm
– Maximum corrosion depth	~270 μm	~5 mm

Oxygen is expected to be present in the deep repository from the operating period up to several hundred years at the most /6-16/. This oxygen is consumed by reactions with copper or minerals in the near field. After that the repository is oxygen-free.

Corrosion due to formation of sulphide "whiskers" has been investigated by means of literature studies and attempts to model whisker formation /6-29, 30/. There is no evidence in the literature to suggest that this type of corrosion takes place in water. The possibilities of nucleation and growth from a surface chemistry point of view should perhaps be further investigated.

Corrosion caused by radiolysis products
Radiolysis calculations /6-2/ show that at the canister thicknesses being considered, the contribution to corrosion from radiolytically produced oxidants will be negligible.

Microbial corrosion
Since RD&D-Programme 92, the previous hypothesis presented in KBS-3 regarding bacterial corrosion has been questioned. In KBS-3, the supply of organic matter in the near field was believed to limit the growth of bacteria. This may not be the case. Ongoing investigations of the growth of sulphate-reducing bacteria in compacted bentonite show that the bacteria cannot survive at densities above 1500 kg/m^3 (see section 5.6.2). If this is true, microbial corrosion could not have any decisive effect on the life of the canister.

Stress corrosion cracking
In order to obtain a copper with good resistance to stress corrosion cracking, a high-purity oxygen-free grade of copper was originally suggested, since the risk was then judged to be non-existent /6-17/. More recent results have shown that this is not the case, but that stress corrosion cracking is unlikely under repository conditions /6-3/. This has been confirmed by subsequent investigations, which show that the tendency towards stress corrosion cracking in the repository environment is small /6-13/.

Internal corrosion caused by residual oxygen/water
The quantities of encapsulated water and residual oxygen are small and will lead to only insignificant corrosion attack on the steel container if the corrosion occurs in the form of general corrosion or pitting.

Internal stress corrosion cracking
Investigations of the sensitivity of the steel canister to stress corrosion cracking caused by radiolysis products shows that stress corrosion cracking cannot be ruled out if the residual air content exceeds 10 vpm /6-18/. Since the canister is under pressure from the outside, the consequences of stress corrosion cracking are limited and do not have to lead to any appreciable reduction in strength for the steel canister.

Corrosion of steel canister after canister penetration

The consequence of corrosion on the internal steel parts after penetration of the canister's outer copper shell have been investigated in recent years. This applies in particular to the rate and mechanism of corrosion, pressure buildup caused by the growth of corrosion products, galvanic effects, formation of HNO_3 inside the canister due to radiolysis, and the consequences of hydrogen gas production in the repository. Both experiments and modelling have been performed. The results of these studies are presented in /6-19, 20, 21, 22, 23, 24, 25/. The consequences of hydrogen gas production for the bentonite are discussed in 5.4.5.

The main conclusions from the studies are:

- Experiments show that the corrosion rate assumes a value of 0.1–1 µm/y after a period of 1,000–2,000 hours, which corresponds to a hydrogen gas production of about 1.6 m^3/y (NTP). The corrosion rate is dependent on pH and ionic strength: lower pH and higher ionic strength increase the rate, but it always ends up within the above range. The corrosion rate is in principle independent of other factors, for example samples in a humid anaerobic atmosphere show the same corrosion rate as samples immersed in water.

- The value of the corrosion rate is dependent on the formation of a protective film of magnetite on the steel. This film acts as a true passive film, i.e. its low solubility/reactivity prevents water from coming into contact with the steel and therefore smothers the anodic dissolution. The magnetite film consists of two layers: a hard electrochemically formed inner layer, and a precipitated outer layer. The primary protection is provided by the inner layer, which is difficult to remove. Mechanical damage to the magnetite film will therefore not affect the corrosion rate of the steel.

- Calculations show that in the event of a defect in the copper shell, corrosion will take place over a large portion of the surface of the steel. The corrosion products formed will therefore not create such stresses in the copper canister that the defect in the copper will be widened.

- For a canister containing 0.01% air initially, about 0.02 g of nitric acid will be formed by radiolysis of water and air. However, the model used contains some uncertainties for systems containing mixtures of air, inert gas and water.

- Galvanic coupling between the outer copper and the inner steel canister can only occur if there is a continuous solution between the metals. This can only arise if there is a defect in the copper shell and water enters the gap between the canisters.

Under aerobic conditions, substantial galvanic corrosion of the steel could occur. However, the combination of the low transport rate for oxygen through the bentonite and the low electrical conductivity of the clay prevents any significant galvanic corrosion during the repository's oxygenated period.

Under anaerobic conditions, the copper-steel coupling could increase the corrosion rate in proportion to the increased area, since the copper acts as if it were a piece of non-corroding steel. This could increase the corrosion rate by a factor <2. However, experiments show that the previously mentioned magnetite film is reaction-controlling, i.e. galvanic effects will not influence the steel's corrosion rate.

The present-day state of knowledge on steel corrosion is good and no further studies are planned.

Cast iron is an alternative material for the inner canister today. The conclusions that apply to steel do not necessarily apply to cast iron. The corrosion properties of cast iron will therefore be investigated.

Creep ductility

For fabrication reasons, the reference design has a radial clearance between canister and insert of about 2 mm. The canister is subjected to a uniform external pressure of 14 to 17 MPa in the deep repository. This load deforms the copper canister and presses it against the insert. A series of creep tests has been conducted to check the creep ductility of copper. The results of these experiments showed that pure oxygen-free copper had poor creep ductility when tested in the temperature range 200–250°C (the expected temperature in the repository is less than 90°C). The reason was assumed to be precipitations of sulphur. Sulphur has very low solubility in copper, and because it deposits at the grain boundaries it has a detrimental effect on the creep ductility of copper. To solve the problem, a new grade with better creep properties was tried, Cu-OFP with reduced sulphur content (<6 ppm) and an addition of about 50 ppm phosphorus. The phosphorus contributed to considerably improved creep ductility, but the mechanism behind this is still not clearly understood. The results of the material testing are summarized in /6-26/.

With the current design of the canister, with a radial clearance of 2 mm between the copper shell and the steel insert, the requirements on ductility are limited. Maximum strains in the copper wall are calculated to be less than 4%.

Grain size

The grain size in the material has a certain influence on its creep ductility due to the fact that at smaller grain sizes, undissolved sulphur is distributed over a larger grain boundary area with lower surface concentrations as a result. Grain size is also of importance for which creep mechanism is dominant /6-27/.

The grain size and shape within the material also affects the resolution that can be achieved in ultrasonic investigations. The ultrasonic signal is spread and attenuated if grains are large or irregular. Consequently, the coarse structure in ingots must be broken down during rolling or extrusion in order to optimise the possibilities for ultrasonic inspection. The requirements on grain size are thereby dependent on the requirements made for the smallest defect size to be detected, see section 6.5.3. The objective during fabrication trials has been to obtain a maximum grain size of about 250 μm. This has been judged to provide fully adequate possibilities for ultrasonic testing of the parent metal and of welds.

A fine-grained structure within the copper reduces the canister's sensitivity to intergranular corrosion. The grain boundary area is greater, which means that larger attacks are required for penetration to occur. Since there may be a risk that the grain microstructure becomes coarsened during the time in the deep repository, investigations of grain growth have been initiated.

The proposed addition of phosphorus (about 50 ppm) increases the recrystallization temperature for copper (the temperature for 50% softening) from 140°C to 250°C. This limits the grain size. An equivalent or greater increase of the recrystallization temperature can be achieved by additions of chromium, zirconium or tin. Chromium and zirconium are difficult to add to a copper melt with sufficient control and would therefore greatly complicate fabrication. Additions of tin at concentrations of 0.1 to 0.2% increase the recrystallization temperature by 170–200°C.

Weldability

Both pure oxygen-free copper and oxygen-free copper with 50 ppm phosphorus have very good weldability by means of electron beam. Elevated concentrations of oxygen and phosphorus have been found to affect welding mainly by leading to increased porosity.

6.3.3 Summary

The choice of material in the copper canister is determined by requirements on corrosion resistance, ductility, weldability and the ability to fabricate with a suitable grain size. An oxygen-free copper with approximately 50 ppm phosphorus is being used as a reference material for the time being. The corrosion properties of this material are well known. Its susceptibility to stress corrosion cracking is low. Several possible materials are available for the inner container. The choice will be determined in part by the fabrication method.

Activities in relation to goals in RD&D 92

A number of goals are set up in RD&D 92 for the R&D work on canister materials during the period 1993–1998. A brief status report on progress towards these goals is presented below.

Progress in the following corrosion studies:

- Investigate the premises for stress corrosion cracking of copper. – The work has begun and a status report has been produced /6-13/.

- Investigate corrosion and radiolytically induced stress corrosion cracking on the inside of the canister. – Reports have been produced during the period on hydrogen-generating corrosion of the steel insert /6-23, 24, 25/ and the risk of stress corrosion cracking caused by radiolysis /6-18/.

- Investigate local corrosion in a mildly oxidizing environment. – Reports have been produced on the development of the redox potential in the near field /6-16/ and the interaction between bentonite and copper /6-1/. An initial mathematical model for modelling of pitting of copper under oxidizing conditions has been produced /6-28/.

6.4 CANISTER FABRICATION

The reference canister can be produced using several different methods. The following methods have been identified for the copper canister:

- Tube extrusion.
- Pressing/rolling.
- Hot Isostatic Pressing, HIP.
- Electrodeposition.
- Spray forming.

In recent years, trial fabrication has been carried out with the first two methods, which are commercially available in full size. Certain investigations and minor tests have also been carried out for the next two methods.

Extrusion, pressing/rolling and casting are being studied for fabrication of the canister insert.

This section presents an account of how trial fabrication is progressing, the status of alternative fabrication methods, a review of quality aspects and a summary of how the different fabrication techniques compare with each other.

6.4.1 Results of trial fabrication

Trial fabrication of four canisters was begun in 1994. The canisters will have an insert fabricated from a steel tube

Figure 6-3. Full-scale canister produced during fabrication trials with self-supporting inner steel canister.

/6-31/. A full-scale canister with self-supporting inner steel canister is shown in Figure 6-3. A new trial series of two to three canisters with cast inserts is planned to start in the autumn of 1995.

Copper canister

Two methods have been tested for the copper canister:

- roll forming or press bending of tube halves with subsequent joining of the two halves by electron beam welding,
- extrusion of whole tubes.

In both cases, a bottom is then welded on by means of electron beam welding.

The copper alloy selected is not a standard material. This, in combination with the unusual (for copper) ingot dimensions, meant that there were no standard products available to buy for the tests. In one case, the casting machine was rebuilt to permit addition of phosphorus, and in another case a new continuous casting mold was built. However, the planned annual volume of copper is great enough to make it worthwhile for manufacturers to modify their plants for serial production.

Rolling mills with a capacity for sufficiently large plates for roll forming or press bending exist at a number of locations in Europe. In order to obtain suitable grain size (currently estimated at about 250 μm) during rolling, a large reduction during rolling is striven for (about 5). If the reduction is smaller, the end result is less certain, since a narrower range of variation is required for other parameters during rolling, such as rolling temperature. Tests performed at different suppliers' plants have shown that it is possible to obtain the desired grain size with less reduction as well. A total of six plates have been fabricated with a grain size in the range 180 to 360 μm.

Roll forming and press bending are conventional methods for fabricating tube from plate. The most important factor in the forming operation is to fabricate the tube halves with such precision that they can be finish-machined without excessive machining allowance and can be joined together by means of electron beam welding. Both methods meet this precision requirement. Roll forming was preferred for trial fabrication, since it leaves the least damage on the surface of the copper plate. This is desirable since it results in better material yield, due to the fact that the starting thickness of the plate can be reduced. Forming of the plates gave good results, with a straightness and a roundness over the length of the plate within a few millimeters of tolerance.

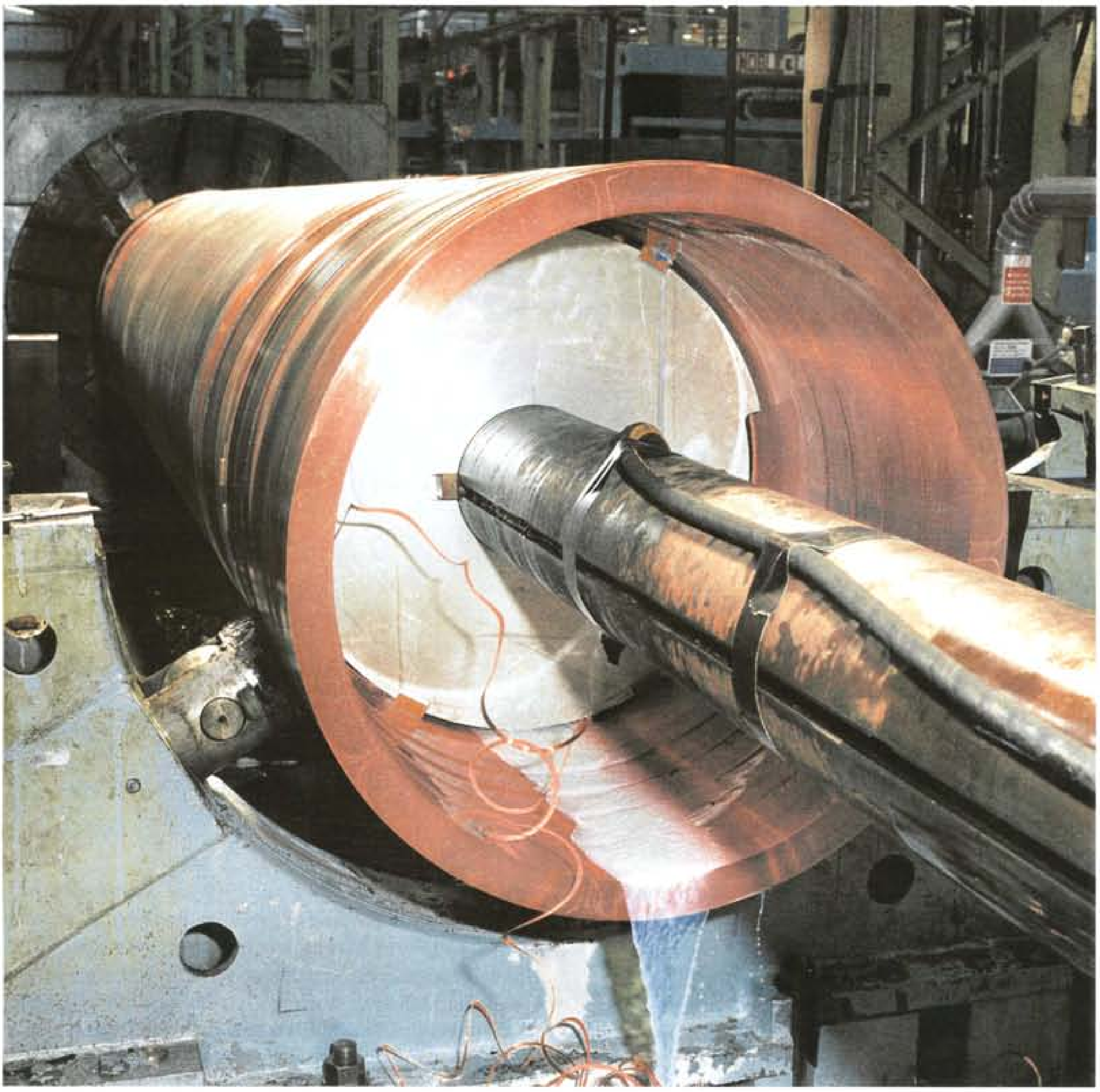


Figure 6-4. *Rough machining of inside diameter of extruded copper tube.*

Extrusion of copper tube of the size in question has never been done before. In this case as well, one of the difficulties lies in controlling the grain size of the material. The trial fabrication was carried out at an extrusion temperature of 800°C, since there was some uncertainty as to what press pressure would be required. It turned out that only about one-third of the capacity of the press was utilized at this temperature. As far as straightness and roundness are concerned, the results of the trial fabrication were very good. However, the grain size in the material was coarser than had been hoped for. On average the grain size was about 800 to 1000 μm, with single grains of up to 2000 μm. This indicates that grain growth is taking place. Trial extrusions carried out without any other cooling than natural convection indicate that the grain size can be reduced by lowering the extrusion temperature. Furthermore, it is possible to cool the material during extrusion. See Figure 6-4.

Canister insert

A canister insert based on a steel tube and a cast insert are being studied. The steel tube insert has been fabricated both by pressing/rolling and by extrusion. Conventional methods were used in both cases and the trials have been carried out without problems.

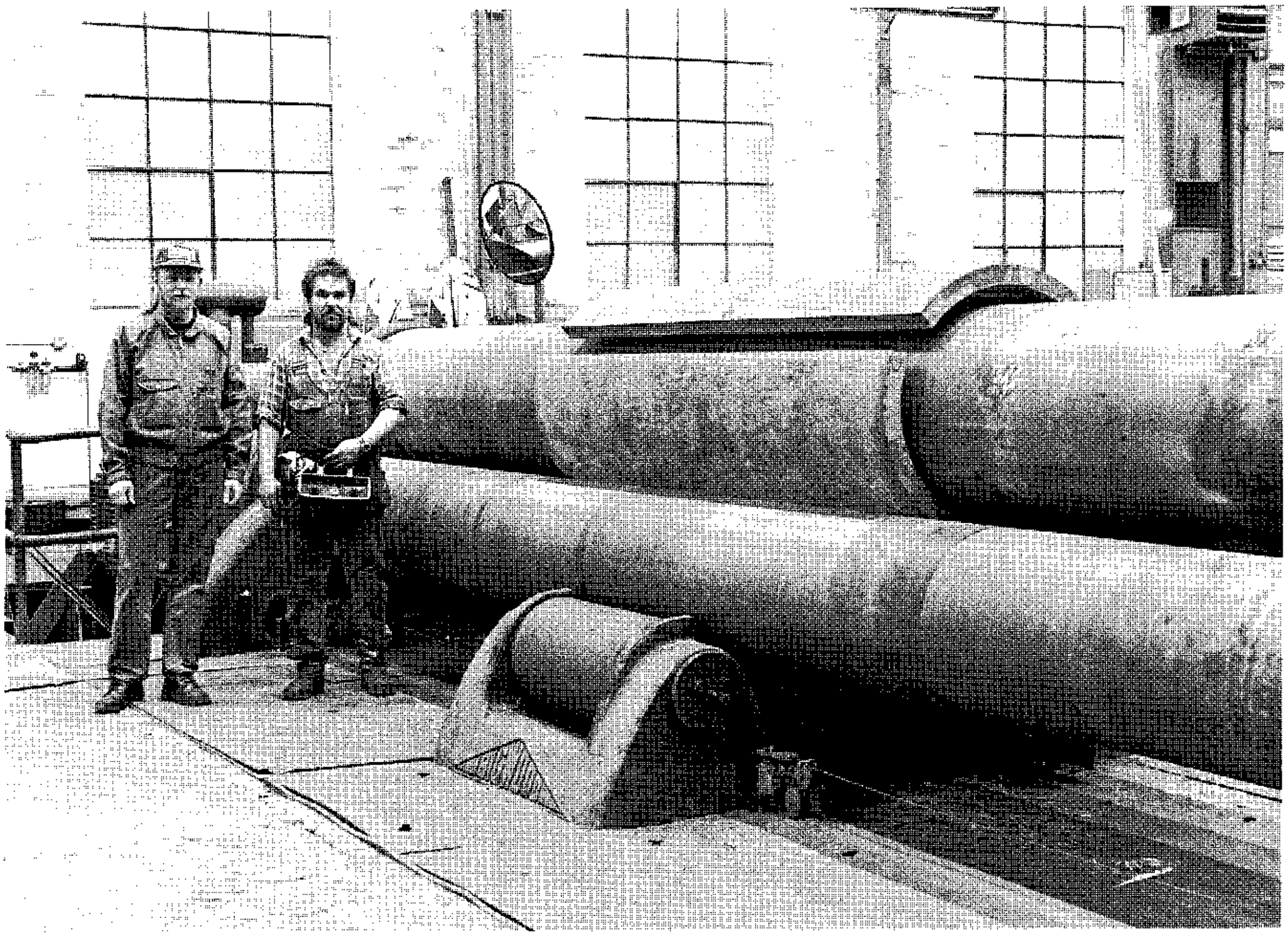


Figure 6-5. Canister insert formed from steel plate for canister with self-supporting canister insert.

Preparations are under way for trial castings of the cast insert with channels for the fuel. Some development work may be needed on the casting process, but the method is judged to be feasible.

From the viewpoint of price and fabrication, it is preferable to fabricate the insert from cast iron, since the alloy has very good castability. If steel or a bronze is chosen, the price will be higher due to poorer material yield and higher material prices.

6.4.2 Studies of other methods

Hot isostatic pressing (HIP) and electrodeposition are also being studied as alternative fabrication methods for the copper canister.

Hot isostatic pressing, HIP

In hot isostatic pressing (HIP), copper powder is compacted to full density at elevated temperature and pressure. The prospects for fabrication of full-sized copper canisters with HIP have been investigated by ABB Atom and IVF /6-32/. No practical tests have been conducted by SKB since 1982-83. However, the aforementioned investigations indicate that it would be possible to achieve full density and ductility in copper materials with a grain size of 4 to 40 µm with press times of about 1 hour at 550°C and 100 MPa.

Two alternative approaches are being discussed. In the first case, only an empty canister is fabricated from copper powder. In view of the length of the canister and the relatively thin canister walls, it may be difficult to achieve the desired straightness and roundness. Alternatively, the copper canister can be fabricated with the steel component as an integral part by pressing the copper powder around the steel container. With this method it would be easier to achieve the desired straightness and roundness of the canister. The disadvantage of this approach is that there will probably be high residual stresses in the copper shell.

In order to obtain good mechanical properties with HIP, surface oxides of the powder particles must be removed. This can be done with hydrogen gas at about

350°C. Reduction of the surface oxide occurs rapidly at this temperature. However, the reduction requires that a sufficient quantity of hydrogen be supplied to the system. Furthermore, water formed by the reduction reaction must be removed. Oxygen dissolved in copper diffuses slowly compared with hydrogen in copper. In-diffused hydrogen combines with oxygen inside the copper grains to form water vapour, which then diffuses very slowly out into the gas phase and leaves the system.

In the trials carried out in 1982-83, nearly theoretical density and mechanical properties could be achieved, compared with forged copper, with good commercial powder grades in large-scale trials /6-33, 34, 35, 36/. The oxygen and hydrogen concentrations in the powder material were substantially higher than in oxygen-free copper, however. For this reason, the material was judged to be susceptible to "hydrogen sickness" if it was heated to high temperatures (about 400°C). This temperature is well above the temperature in the repository, but is reached in the joint area if the material is welded. Metallographic examination also revealed clear signs of hydrogen embrittlement after heat treatment at 400°C for one hour /6-37/.

Subsequent investigations have shown that good mechanical properties can be achieved even after heat treatment if the powder can be handled in virtually oxygen-free conditions from atomization to HIP (6-38). Despite the careful handling of the powder, it was found that impurities on the powder grains, probably dust, could not be avoided. This had no influence on the mechanical properties of the material, but could affect weldability. In that case, this must be investigated.

No isostatic presses are available today with large enough dimensions to carry out full-scale trial fabrication.

If the HIP process is chosen as a fabrication method, it is probable that investments will have to be made in both press equipment and an installation for the production of copper powder via an atomisation process.

Electrodeposition

In electrodeposition, copper is precipitated directly on the insert by means of electrolysis.

Good results were obtained in initial tests at the model level. Copper was deposited on several miniature canisters. Before the tests are continued on a larger scale, the mechanical properties of the electrodeposited copper will be investigated. Preliminary investigations have started and will continue for the next few years.

The structure of electrodeposited copper can be controlled to a great extent by the process parameters and subsequent heat treatment. Creep tests have shown that the material has poor creep ductility compared with the reference material. The creep tests were, however, conducted on material whose structure was not representative of the material that is foreseen for the canisters. Creep testing on more representative material will therefore be carried out. Electrodeposition has the advantage that no gap exists between the canister and the insert. This reduces the requirements on the mechanical properties of the material. The material must also be tested with respect to weldability and inspectability by means of ultrasonic inspection.

Spray forming

With this method, tubes are formed by spray-depositing finely dispersed molten metal onto a recipient. The method is used in industrial tube fabrication. Sandvik Steel manufactures tubes of high-alloy special steels with diameters of up to 400 mm.

6.4.3 Quality assurance and control methods

The work of designing a quality assurance programme for canister fabrication and sealing has been commenced in conjunction with the ongoing trial fabrication of whole canisters.

The principle of trial fabrication has been to obtain full traceability for all parts of the canister with respect to, for example, analysis and testing protocols, repair reports and heat treatment. The intention has also been to identify and remedy any defects as early as possible in the operation sequence.

At the same time, different test methods (mainly conventional) are being compiled for use.

6.4.4 Summary

Trial fabrication of full-sized canisters has shown that both forming from rolled plate and extrusion are possible methods for fabricating copper canisters on a full scale. In the case of forming from rolled plate, it was possible with available ingots and equipment to satisfy the requirements on the microstructure in the material to an essential degree. In the case of extrusion, the results were promising although they did not fully achieve the original objectives. But there are good prospects for achieving the desired grain size in the material by means of modified process parameters and the use of controlled cooling during extrusion.

Regarding other fabrication methods, HIP and electrodeposition, studies are under way to determine the value of carrying out trial fabrication tests on a full scale. Besides judging material properties, the evaluation also includes estimating the costs of serial production of canisters and the flexibility provided by a given method in choice of suppliers, etc.

The operating period for the encapsulation plant will extend over a time span of more than 40 years. The possibility cannot be excluded that several different fabrication methods will be used during this time if alternative methods are developed and become available.

Activities in relation to goals in RD&D 92

- Completing the studies concerning lead casting on a model scale. – The work during the period has been focused entirely on copper/steel canisters of alternative designs. Work on the alternative lead-filled canister has been given lower priority for the time being.
- Supporting development of fabrication technology, welding technology and methods for nondestructive testing. – Trial fabrication of four full-sized canisters by means of different methods is in progress.

6.5 SEALING METHOD

It must be possible to seal the copper canister to high standards of reliability and leaktightness, as well as inspectability. The objective is that no more than 0.1% of all canisters may contain undetected defects that could lead to leakage.

To fulfill the stringent requirements for the sealing of the copper canister, a method is being developed employing electron beam welding to seal on the top copper lid. The same method is also employed to attach the bottom of the copper canister. All development efforts are currently being concentrated on this method. Alternative methods that have been proposed are friction welding and diffusion bonding.

Various methods for the nondestructive testing of the weld are being developed in parallel in order to verify conformance with the stipulated requirements. The formulation of defect detection requirements is also under way.

6.5.1 Results of trial welding

During the period 1986–1992, within the framework of the EUREKA Project, SKB participated in the development of an electron beam welder designed to be used without high vacuum in pressures up to atmospheric pressure. After the project had been concluded, the equipment that was developed within the framework of the project was used to develop welding technology for sealing of the copper canister.

A development programme for welding of copper under reduced pressure was conducted during 1992 and 1993 /6-39/. Welding trials with both horizontal and vertical electron beams were conducted on oxygen-free pure copper and on oxygen-free copper with a low phosphorus content on flat workpieces. The main purpose of the work was to determine the optimum pressure for lid welding in the welding chamber over the pressure range 5 Pa to 100 kPa. See Figure 6-6.

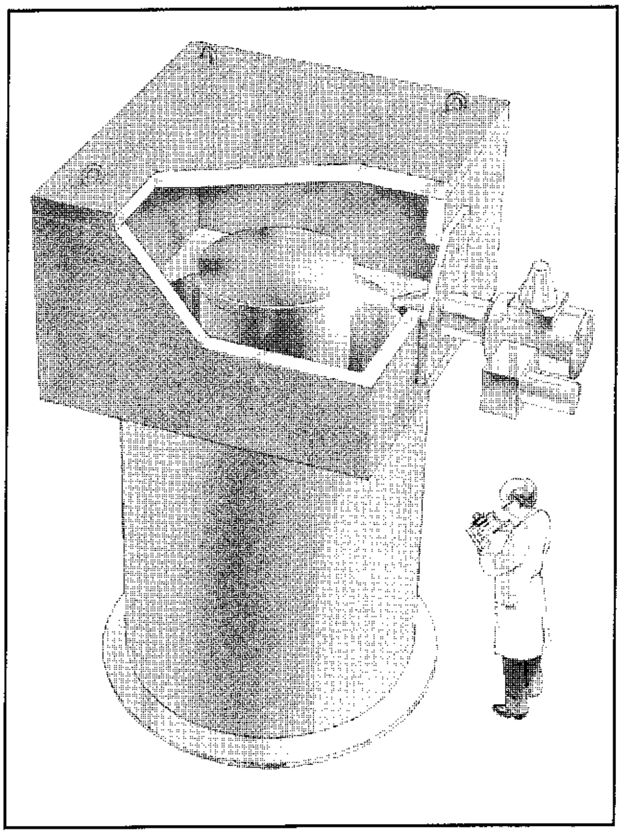


Figure 6-6. Electron beam welding of copper lid on copper cylinder.

The influence of the helium pressure in the welding chamber and of the distance between the workpiece and the electron gun was investigated in detail for several different geometries of the electron beam and for outputs of up to 60 kW at an acceleration voltage of 100 kV. At helium pressures under 1 kPa, it was found that a penetration of more than 70 mm could be achieved without root defects over a large range of welding parameters. This was also true over the entire length of the weld.

The weldability of oxygen-free pure copper and oxygen-free copper with a low phosphorus content was very similar, although the tendency towards microporosity was slightly greater for the phosphorus-containing material.

The results were transferred during 1994 to full-scale welding of copper tubes of sufficient length to provide realistic conditions for lid welding in the encapsulation plant. It was found that certain changes of the lid design were necessary in order to obtain a good result.

After these changes had been made, welding could be performed with satisfactory results /6-40/. The trial series included first welding of five lids and was then concluded with welding of the bottom and lid on a 2.4 m long canister of full diameter. The welds were subsequently examined for defects and found not to be completely free of defects. However, the scattered pores that were detected in the weld have been deemed to be acceptable, see also section 6.5.3.

The changes made in lid design have certain negative consequences for the practical handling of the canister in the encapsulation plant. Further modifications of the lid are therefore planned.

Temperature measurements were made on the canister shell during welding of the trial series. The long-term stability of the equipment was tested and the high-voltage equipment was modified to reduce the incidence and consequences of discharges.

6.5.2 Investigation of other welding methods

Due to the high thermal conductivity of copper, joining of the lid to the canister using methods where the metal is melted locally will require high power density. The methods that could provide this are electron beam welding and laser welding. Laser welding of copper is not possible with available welding equipment.

Alternative methods could be processes where joining takes place without the copper having to be heated to the melting point. The most interesting methods in this context are friction welding, diffusion bonding and brazing. Friction welding is done with a very high energy input and produces rapid joining. The other methods require relatively long soaking times. This means that the whole canister with fuel must be heated to a joining temperature of several hundred degrees. Sealing of the canister therefore takes a much longer time than with other methods.

Friction welding

Friction welding has been investigated both theoretically and in practical tests. The first investigation was conducted by TWI /6-41/, which conducted weld tests on copper tubes on a scale of 1:4. The tests were considered successful producing fine-grained material in both the weld and the heat-affected zone, apparently without defects. Based on these tests, full-scale friction welding was deemed possible. A study by IVF /6-42/ confirmed the TWI conclusions.

Equipment for full-scale tests is not available, but is judged to be fully possible to manufacture. For the time being, friction welding is therefore regarded as a reserve alternative that can be considered if electron beam welding does not provide satisfactory results.

Diffusion bonding

In conjunction with the tests of hot isostatic pressing (HIP) of copper powder, diffusion joining (500°C, 150 MPa) of the lid was also carried out as an integral part of the tests /6-33, 34, 35, 36/. Several different methods for edge preparation were tested in order to remove oxides from the copper surfaces, but no real grain growth over the joint surface was observed. The joint surface most closely resembled in appearance a grain boundary that ran through the entire wall thickness. The grain-boundary-like joint could be a zone of weakness and a starting point for intergranular corrosion. No systematic investigation of the process parameters (pressure and temperature) was conducted at this time.

6.5.3 Nondestructive testing

Fracture toughness measurements performed on welded material show that copper possesses exceptionally high fracture toughness. The fracture mechanism for a given defect in the weld will be dominated by plastic collapse. This means that the residual stresses in the weld are of no importance and that only external load need be considered. Under the conditions in question, an entrained defect corresponding to two-thirds of the wall thickness with a considerable length could be approved /6-43/. Defects of this size can be detected without difficulty by ultrasonics, even in very coarse-grained material.

In view of the material's high damage resistance, the largest permissible defect will be determined by how much reduction of the wall thickness can be tolerated from a corrosion viewpoint. As was evident from the discussion in 6.2.2, the safety margins against corrosion penetration in 100,000 years are very large. In the case of deeper-lying defects, very large defect sizes can be tolerated, while near-surface and surface-breaking defects could serve as starting points for crevice corrosion. This applies to the time immediately after deposition, which in this context means the period up to 300 years post-deposition. Work is under way to determine acceptance criteria for defects in welds and parent metal.

Investigations of creep crack growth are under way. The results so far indicate that no growth can be observed within the measuring accuracy. Further tests will be performed to obtain a better basis for a judgement.

Method development for nondestructive testing is currently being conducted with ultrasonics and digital radiography.

Tests have been conducted with ultrasonic pulse-echo on copper blocks with artificial defects. The frequency range was 2.25-5 MHz and the material was hot-rolled copper with a grain size of 180–250 μm. The sensitivity that could be achieved corresponded to 0.5 mm side-drilled holes /6-44/. A total of seven lid welds have been tested with ultrasonics. All were inspected using pulse-echo, compression waves from above the weld, most with manual scanning /6-45/. Specimens from two lids were inspected by digital radiography, as a reference for ultrasonic inspection. The number and distribution of defects varied considerably between the welds. This proved to be useful for testing the sensitivity of the different methods used.

The most important observations were:

- Digital radiography can be a useful technique either as a main alternative or as a complement to ultrasonic testing. Based on the preliminary work that has been done, the technique appears to be feasible for detecting defects down to around 1 mm in diameter.

- The pulse-echo technique was able to detect defects in the weld down to 2 mm in diameter with a signal-to-noise ratio of 6 dB. The detectability of defects was not always directly related to size.

- The use of gel as a coupling medium, instead of immersion in liquid, sometimes gave inadequate coupling, and water as a coupling medium is preferable.

- For deeper-lying defects, detectability is fully adequate with both digital radiography and ultrasonic inspection. The possibilities of supplementing these methods for the detection of surface-breaking defects should be further explored.

6.5.4 Summary

Results obtained show that electron beam welding is a feasible method for fabricating and sealing copper canisters. Fully satisfactory results have not yet been obtained from the work of development of methods for nondestructive testing; this will require further efforts. The changes in lid design that were necessary may complicate the handling in the encapsulation plant. Testing of alternative lid designs will be necessary.

Activities in relation to goals in RD&D 92

- Supporting development of fabrication technology, welding technology and methods for nondestructive testing. – Development of welding technology has been carried out to a level where lid welds can be made with electron beam welding without root defects in the joint area, even during termination of the weld. Development of methods for nondestructive testing has proceeded according to plan with both ultrasonics and radiography.

6.6 ENCAPSULATION PLANT

The plan to build an encapsulation plant in connection with CLAB was presented in RD&D-Programme 92. Preliminary design of the plant is currently in progress. Support documentation for an application for a permit to erect the plant will be finished during 1997. The support documentation will include an environmental impact assessment for the encapsulation plant. Important parts of this are a plant description and a preliminary safety report.

6.6.1 Siting and EIA process

An addition to SKB's Central Interim Storage Facility for Spent Nuclear Fuel, CLAB, at the Oskarshamn Nuclear Power Station is planned to house the encapsulation plant. Such a co-siting has clear advantages in comparison with other siting alternatives in terms of fuel handling logistics, resource utilization and environmental impact.

- Some existing systems and plant parts in CLAB can be used for the encapsulation process. The extent to which this is possible is dependent on whether an integral or free-standing solution is chosen for the addition. The construction needs for the encapsulation plant will be reduced if the decision is made in favour of an integrated solution.

- Access to other nuclear engineering infrastructure is good at the Oskarshamn NPS and CLAB.

- The experience of fuel handling and operation and maintenance of associated service systems possessed by the personnel at CLAB will be best put to use if the work is coordinated organizationally and physically at one location.

- The shipments to the deep repository can be carried out with a simpler container system when the fuel is encapsulated than if fuel first has to be shipped from CLAB to another location for encapsulation. The number of transport units is 40–50% greater, however, since there will be fewer fuel assemblies in a canister than in a fuel transport cask.

- The encapsulation plant can be accommodated within SKB's CLAB property site. This means that the environmental impact will be minimal. New land does not have to be used, and no new roads or cooling water installations are needed. Polluting emissions to the atmosphere and discharges to water are expected to lie within the limits allowed at the CLAB facility (the Oskarshamn NPS and CLAB).

A joint consultation group for EIA matters for encapsulation at CLAB was formed during 1994 with participation by the Municipality of Oskarshamn, the County Administrative Board in Kalmar County, the Swedish Nuclear Power Inspectorate, the National Radiation Protection Institute and SKB. To obtain an early overview of the environmental impact an encapsulation plant might have, the EIA work was initiated with a pilot study. The purpose of the pilot study was to provide an overview of the project and to shed light on what alternatives exist with regard to siting and design. The results are presented in a pilot study report /6-46/.

6.6.2 Encapsulation process

The encapsulation plant will be designed and built primarily to house the encapsulation process. In addition, it must be possible at a later stage to add a process line for conditioning of core components. In designing the plant, special consideration will be given to matters related to operation and maintenance as well as to industrial and radiological safety.

The functional parts planned to be included in the plant, besides the actual encapsulation process, are:

- Conditioning of core components.
- Areas for materials handling.
- Service systems.
- Areas for operating staff.
- Areas for maintenance activities.

The encapsulation process will be designed and engineered to deliver well-fabricated and carefully inspected disposal canisters containing fuel to the deep repository. In designing the process, special attention will be given to matters related to industrial and radiological safety. The work of designing a suitable process for encapsulation of the fuel can be divided into functional parts where different technical solutions will be considered. The work of designing the plant has begun and preliminary descriptions of the encapsulation process /6-47/ and the layout of the plant /6-48/ have been produced. The following general description of the encapsulation process will serve as a basis for the continued design work. See Figure 6-7.

Transfer of fuel

The fuel is transferred in storage canisters from the storage section of CLAB via the existing fuel elevator to a new pool area in the encapsulation plant.

Preparation of fuel

Identification of the fuel as well as measurements, and presumably some form of sorting, will be carried out in the handling pool. Water serves as a coolant and radiation shield.

Handling cell for fuel

The fuel is lifted out of the water up to the handling cell, where it is dried and placed in a disposal canister, see Figure 6-8. In this part of the plant, where the fuel is handled freely, special requirements must be met to prevent the release of radioactivity. The cell is built with radiation-shielded walls and special requirements on airtightness and ventilation. This type of compartment is usually called a "hot cell". A handling machine that incorporates proven technology and meets stringent requirements on reliability and safety is chosen for handling in the cell. Special attention is given to accessibility for service and maintenance.

Other encapsulation functions are located at separate process stations.

Transfer of filled canisters

Transfer of filled disposal canisters will take place in an area underneath the handling cell and the various process stations. Alternative transportation systems are being studied and the choice will be made based on stringent requirements on reliability and safety. During transfer the canister is sealed so that radioactivity cannot be released from the fuel.

Three stations for sealing of the canister

The first process station will contain the functions required for replacement of the atmosphere in the canister and, if necessary, filling around the fuel assemblies. Finally, the steel lid is fitted and sealed. After that there is no risk of radioactivity being released from the fuel. This station is also designed to meet the requirements for a hot cell.

The copper lid is placed on and sealed in the welding station. The design of the welding station will be based on the current work of developing an electron beam welding method. The material in the copper lid and the copper cylinder are fused together by an electron beam in a vacuum chamber.

The last of the three stations will house equipment for the inspection of the lid weld and for the machining of the weld area on the canister, as well as for being able to remove an improperly welded lid. The result of the sealing operation will be inspected by means of non-destructive testing, using technology that is being developed along with the sealing method.

Post-weld treatment and buffer storage

A routine check for surface contamination is planned, and decontamination of the outside of the canister will be possible in a special process station.

A buffer store for filled and sealed canisters is planned so that the canisters can be delivered to the deep repository at a suitable pace. Handling in the buffer store will be done with a radiation-shielded handling machine. The

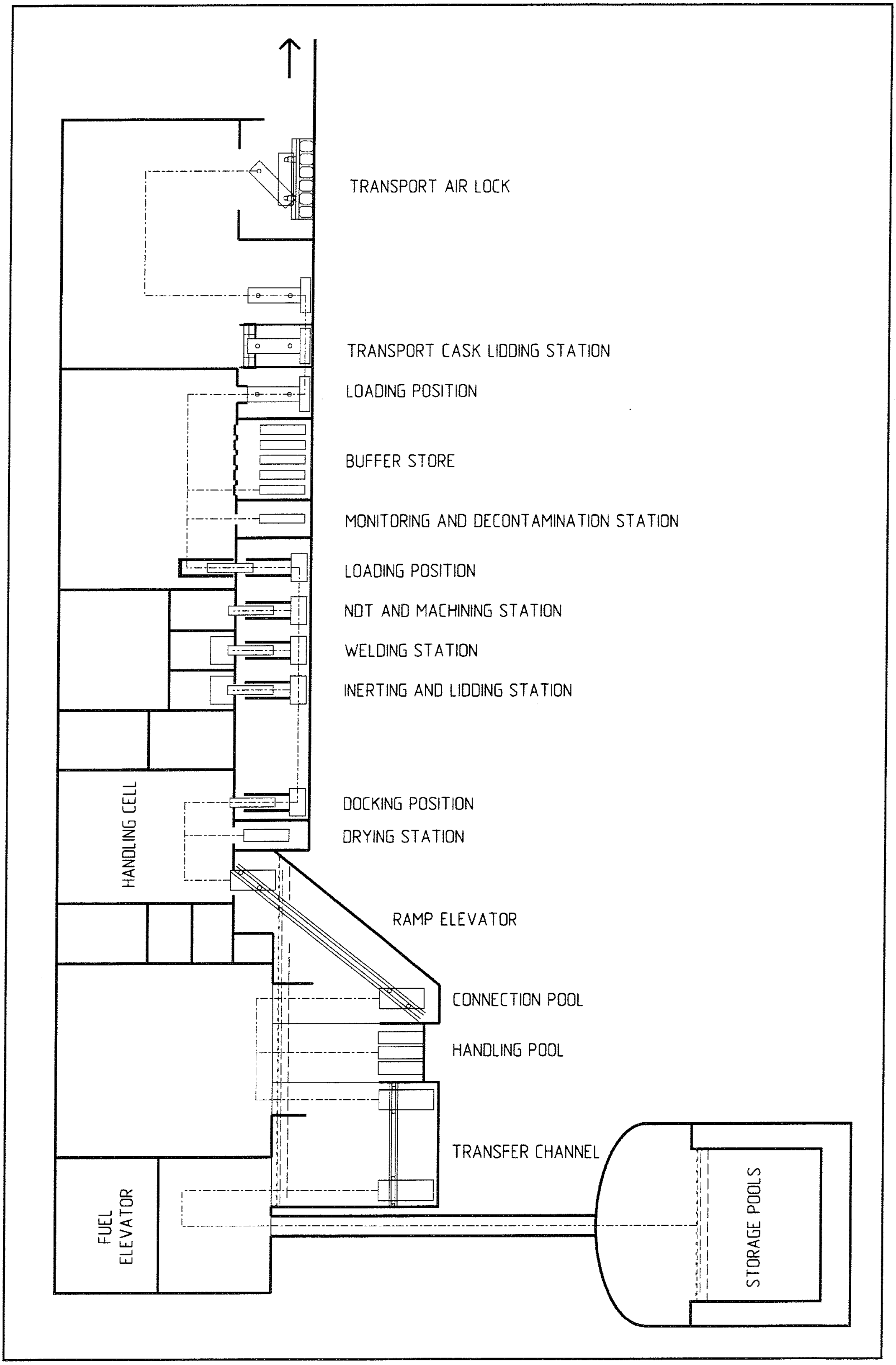


***Figure 6-7.** Overview of the different steps in the encapsulation process.*

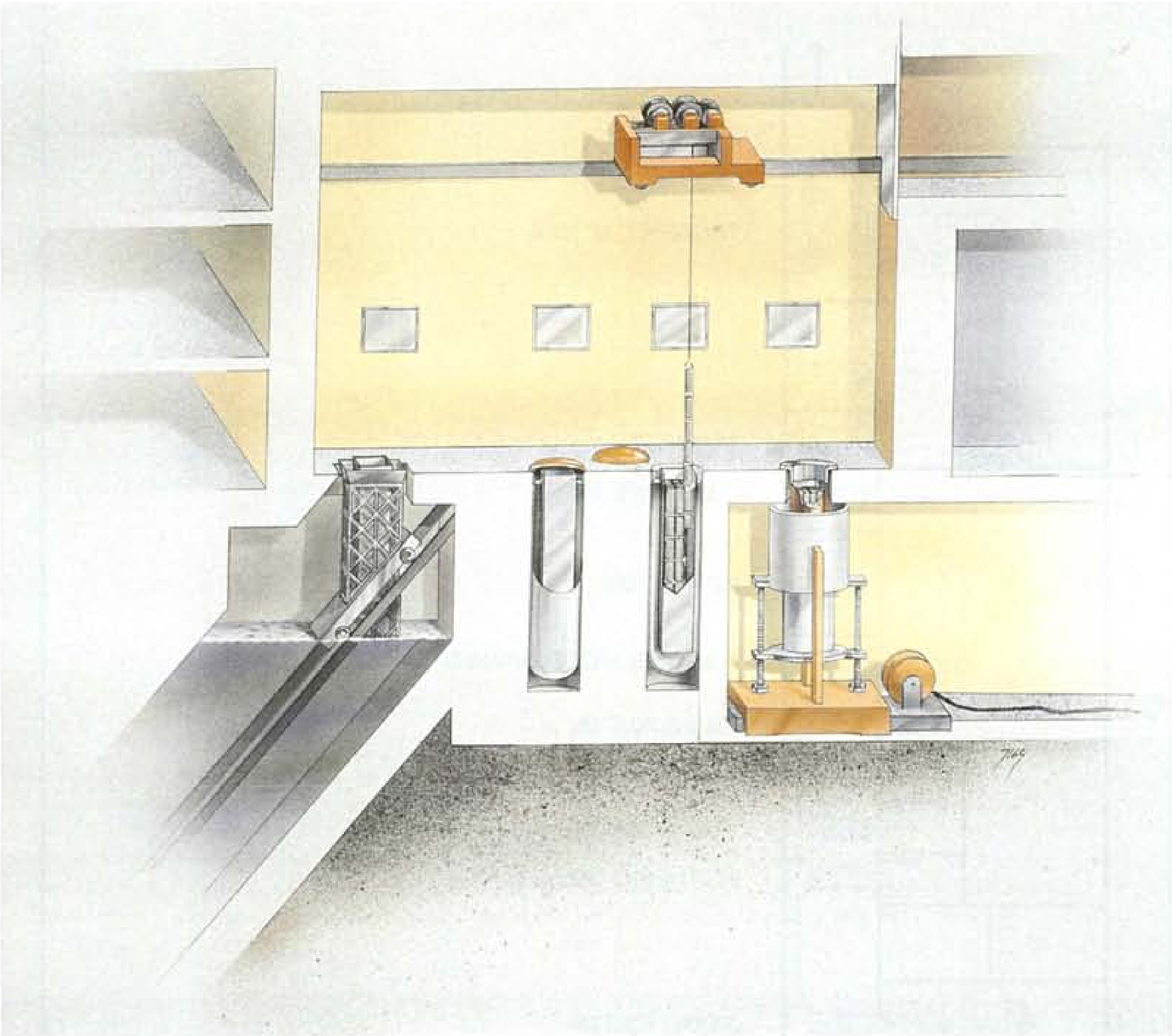


Figure 6-8. Handling cell where the fuel is lifted out of the water, dried and placed in disposal canisters.

buffer store will be connected to a docking station for transport casks.

Handling of failed canisters

If inspection of the weld reveals that it is not approved, the canister is taken back to the welding station, where it is rewelded. In the event rewelding cannot be done or is not successful, the canister goes back into the process for removal of the lid and extraction of the fuel, which is placed in a new canister.

7 PROGRAMME FOR CANISTER AND ENCAPSULATION

7.1 PREMISES AND GOALS

Premises

The programme for canister and encapsulation primarily comprises development and fabrication of the canister for spent nuclear fuel and design and construction of an encapsulation plant. This work is called "Encapsulation Project".

In a later phase, other long-lived waste will also be treated in the encapsulation plant. Examples of such waste are core components, such as control rods, and other internals parts from the reactor vessel that have been activated by neutron bombardment during operation of the reactors. These components are planned to be embedded in concrete. Certain preparations for subsequent installation of such equipment are being made in conjunction with the design and construction of the encapsulation plant. The waste will be shipped in transport casks of the same type as those used today for shipments of spent fuel from the nuclear power plants to CLAB.

Goals of the work with canister and encapsulation plant

The copper canister will be designed to meet the requirements for containment and isolation of the fuel for the necessary time in the deep repository.

The design of the copper canister will be developed so that it can be fabricated and sealed in a reliable manner. Properties of essential importance for long-term safety will be taken into consideration in the choice of material grades and methods for fabrication, sealing and inspection.

The encapsulation plant will be designed and built to deliver well-fabricated and carefully inspected canisters with fuel to the deep repository. In designing the process, special attention will be given to matters related to industrial and radiological safety.

SKB's planning for the work with canister and encapsulation plant is based on the following activities:

Canister:
- Development and design of the copper canister.
- Development of sealing method.
- Development of fabrication method.

Plant:
- Planning of encapsulation plant.
- Construction and installation.
- Commissioning and trial operation.

The planned work sequence is described in chapter 7.3, based on the subdivision into phases for step-by-step design of the plant. Planned activities for the work with the canister are then described in greater detail in the following chapter.

An investigation of the possibility of testing the vital equipment for canister sealing in a pilot plant is currently under way. The necessary background information for a decision is expected to be available during 1996. In the following description of the programme, it is assumed that a decision will then be made on continued work with a pilot plant.

Important interim goals in the planning are to:

- Finalize the design of the canister.
- Prepare an EIA document and a safety report for the encapsulation plant.
- Build a pilot plant for canister sealing.
- Compile a permit application for an encapsulation plant at CLAB (pursuant to the Act – Concerning the Management of Natural Resources and the Act on Nuclear Activities).
- Evaluate and report results from the pilot plant.
- Obtain a Government permit for start of construction of the encapsulation plant.
- Start serial fabrication of canisters.
- Start trial operation of the encapsulation plant.
- Prepare a final safety report in support of an application for a permit for initial operation.
- Obtain an operating permit and start initial operation of encapsulation of fuel.

Support documentation for a permit application is expected to be ready at the start of 1998. It is projected that licensing, construction, commissioning and trial operation can then be carried out so that the first canisters are ready for delivery to the deep repository in 2008.

7.2 ALTERNATIVE SITINGS AND PROGRAMME

The work with environmental impact assessment, EIA, for the encapsulation plant will examine various aspects

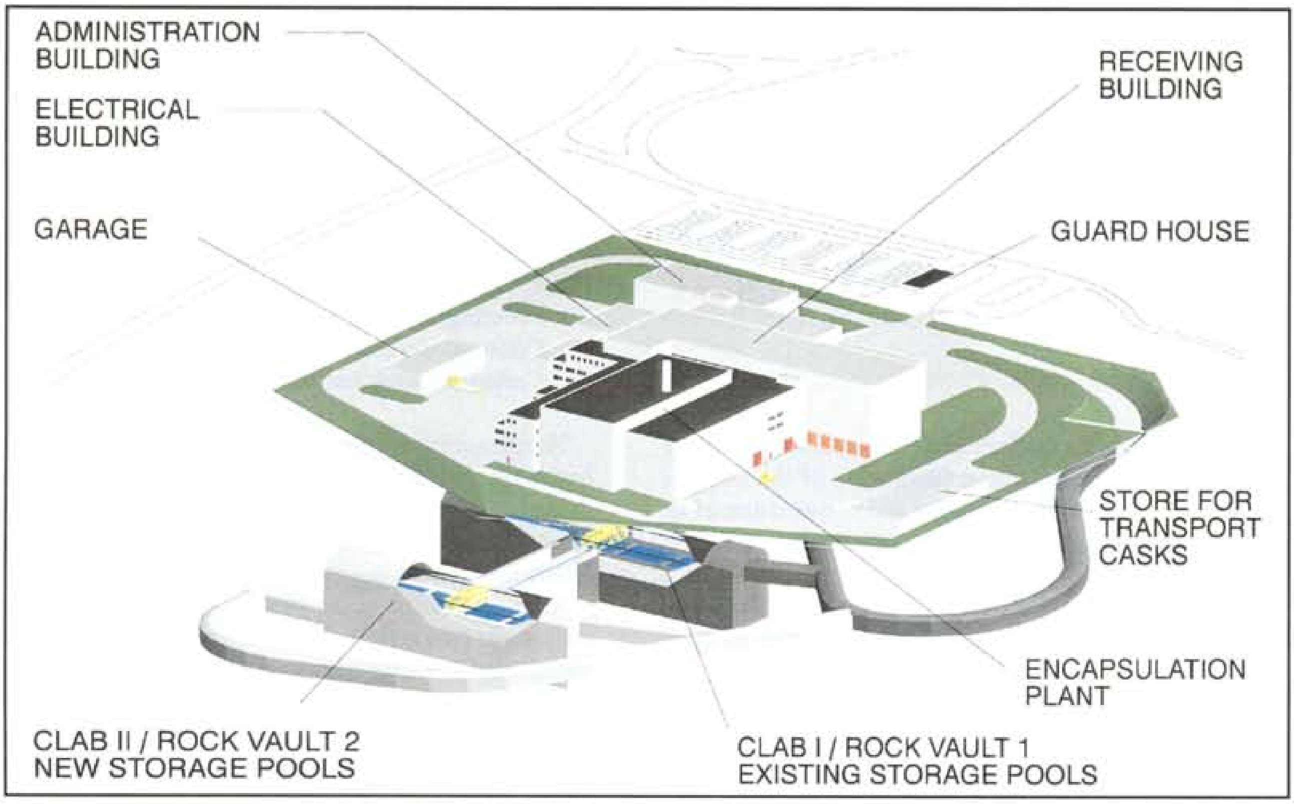


Figure 7-1. *Overview of planned encapsulation plant at CLAB and new rock cavern for fuel storage.*

of siting alternatives. The main strategy is to build the plant adjacent to CLAB. An alternative is location adjacent to the deep repository.

A zero alternative will also be examined in the EIA work, even though a zero alternative in the true sense of the word does not exist in this case. Continued storage in CLAB for a given period of time pending the availability of another method for final disposal of the fuel can, however, be considered. This is defined as the zero alternative and is described in the following section.

Continued storage in water pools in CLAB does not entail any radical changes in relation to the present-day operating situation. This means that CLAB will be kept in operation with cooling and purification of water and with ventilation. Regular maintenance of active equipment will be required. Then when fuel can no longer be received, the workforce can be reduced somewhat.

As a basis for an assessment of how long CLAB can be kept in continued operation with appropriate maintenance, a study needs to be done of various mechanisms that might affect the plant and the fuel, such as corrosion. Further, a study is needed of the expected maintenance requirement, in part to estimate the costs of continued operation. Finally, various environmental consequences of continued operation will be compiled. A preliminary assessment is that storage for 100 years or more should be technically feasible.

For the alternative of continued storage for 50 years or more in CLAB, the possibility should also be considered of transferring the fuel to dry storage containers of a type similar to that used in Germany, for example. This would reduce the need for surveillance, while eliminating the need for cooling and purification of pool water and for ventilation. A study of consequences and costs for this alternative is also planned to be included in the study.

7.3 DESIGN OF THE ENCAPSULATION PLANT AND LINKS TO THE WORK OF CANISTER DEVELOPMENT

Design of the encapsulation plant follows a programme for step-by-step design. Conceptual design studies have been carried out and the results presented in a prelimi-

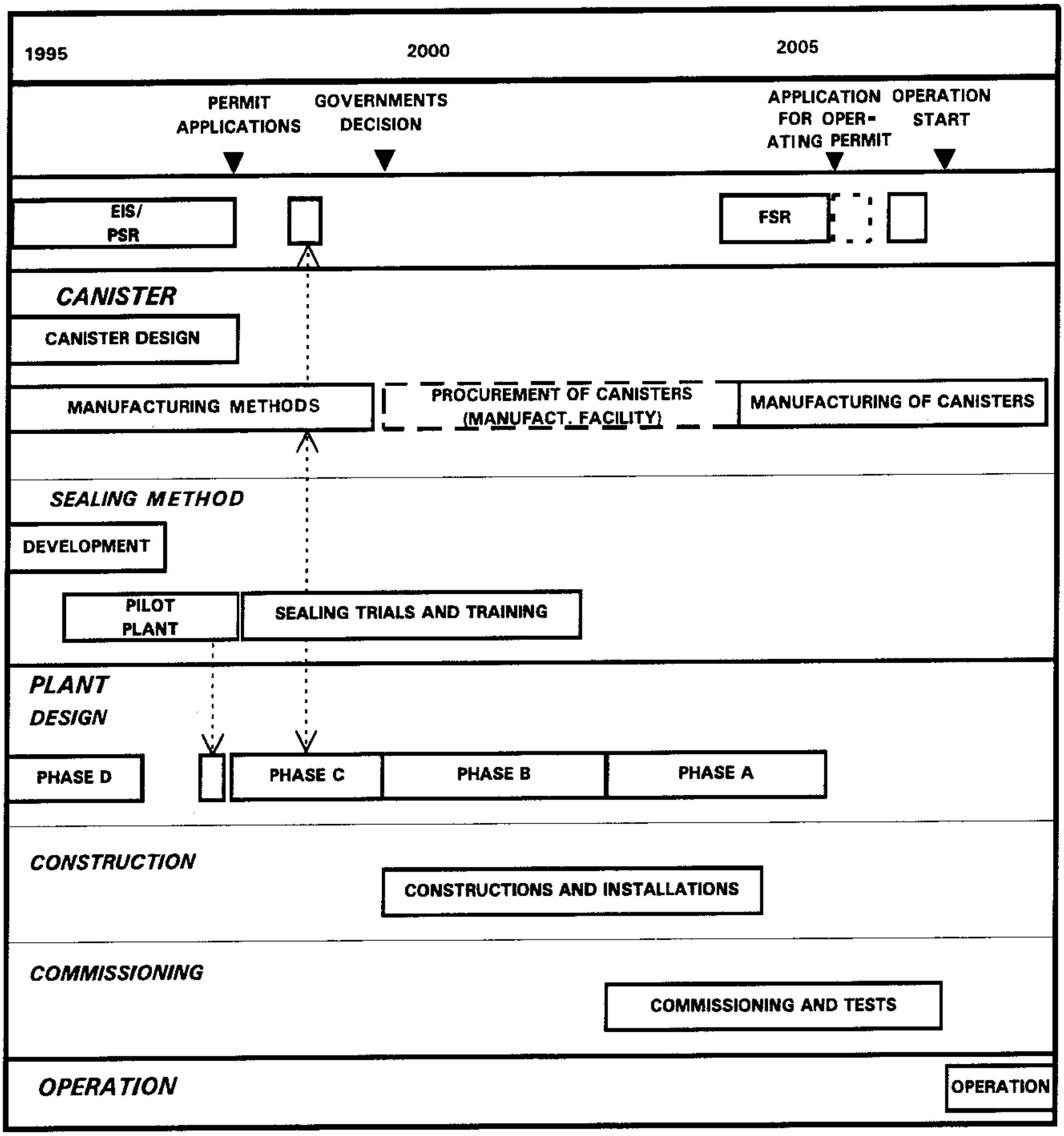


Figure 7-2. Overall time schedule 1995–2008.

nary plant description. The continued work includes the following phases with associated summaries of plant design and canister development:

- Basic design (Phase D).
- General engineering (Phase C).
- Engineering (Phase B).
- Documentation (Phase A).

Design of the canister and development of methods for fabrication, inspection and sealing follow a separate programme which is linked to the various phases in the step-by-step design process for the encapsulation plant.

The design work includes planning for the future installation in the plant of equipment for embedding core components in concrete. Some equipment and building sections for handling of canisters are also planned to be able to be used for the similar handling of packages for core components. The preparations will be different for different parts of the process, depending on opportunities for future installation and commissioning of this supplementary equipment.

Transport casks for copper canisters with fuel and packages with core components are also included in the design work.

A general description is given below of what is planned to be done in the different design stages with regard to both plant design and development and trial fabrication of the canister. A more detailed description of the work with canister design, sealing method and plant design is given in the following sections.

Each step in the development work for the canister follows a methodology that basically includes the following steps.

- Problem formulation and inventory of alternatives.
- Decision on choice of line of action.
- Planning and implementation.
- Compilation and evaluation of results.

In making a decision on choice of line of action, consideration is given to the risks that the chosen method will not lead to the desired result and the consequences this may have for subsequent steps in the work with canister and plant design.

Basic design (Phase D)

The basic design phase will serve as a basis for SKB's decision to apply for a permit for the encapsulation plant.

Plant design
In this phase, the constituent systems are studied and optimized. Performance and safety assessments are carried out. A plant description and system descriptions are written as a basis for the preliminary safety report and the environmental impact assessment.

Canister
The following activities are foreseen to be carried out during phase D for the work with the canister:

- Supplementary tests for canister material.
- The main principles for the design of the canister are determined.
- Testing of fabrication and inspection methods for the canister.
- Fabrication of full-sized trial canisters.
- Investigation of premises for serial fabrication.
- The sealing method is tested on trial canisters.
- Inspection methods for the weld joint are tested and developed.
- Alternative joint designs are tested.
- A pilot plant for lid welding is built.

Miscellaneous
A design of transport casks for copper canisters is finalized as a basis for the design of the handling equipment in the encapsulation plant.

General engineering (Phase C)

The general engineering phase will mainly serve as a basis for procurement of construction works and the various systems in the encapsulation plant, as well as for planning of the forthcoming construction phase.

Plant design
The building design is progressed with the guidance of the descriptions of the constituent systems and equipment. The building structure is designed for existing loads, radiation shielding requirements etc. and with a view to installation routes for equipment. Preliminary general arrangement drawings are prepared.

Supplementary descriptions of equipment for the different systems are drawn up and other documents to be included in the procurement documentation are prepared.

Canister
The following activities are foreseen to be carried out during phase C for the work with the canister:

- The detailed design of the canister is determined.
- Trial fabrications of canisters are carried out.
- Trial weldings are carried out in the pilot plant.
- A method for serial fabrication is settled upon.

Miscellaneous
A target specification for transport casks is prepared as a basis for tendering documents.

Engineering (Phase B)

In this phase the work documents for the building work and for manufacture and installation of the different systems in the plant are prepared.

Plant design
The suppliers carry out detailed engineering of equipment for the various systems.

The plant layout is determined on the basis of documentation from the system suppliers. General documents are prepared for the building structure and working documents are drawn up for the construction works.

Canister
Continued trial fabrications at suppliers and trial weldings in the pilot plant are foreseen during phase B.

- Necessary agreements on serial production are concluded with suppliers.

Documentation (Phase A)

Plant design
Relational documents are drawn up and final system descriptions are written as a basis for commissioning and a final safety report.

Canister
The results of trial fabrication and of trial welding in the pilot plant are compiled and documented.

7.4 DEVELOPMENT AND DESIGN OF CANISTER

7.4.1 Criteria for sizing and design

The work of designing the canister is planned to take place in steps through the compilation of basic premises, property requirements and criteria for sizing and design, in accordance with the procedural description in chapter 6. These compilations, combined with experience from practical trials with canister fabrication and sealing, will then serve as a basis for the final choice of canister design. The planning of the work on the different questions that influence canister design is described below.

Long-term safety and performance in the deep repository

Initial integrity
The probability of undetected defects that could lead to early canister failure will be analyzed on the basis of selected methods. The value given in 6.2.3 for what can be accepted may be changed on the basis of e.g. safety assessments for the deep repository.

Strength
Load premises as given in section 6.2.3 will be reviewed with a view towards the planned layout of the deep repository and the scenarios dealt with in the safety assessment. Safety factors will be determined for normal loads, along with criteria for assessment of extreme loads.

Corrosion resistance
Conclusions regarding corrosion resistance are not expected to be influenced to any great extent by the local variations that can be accepted for the choice of repository site. In view of uncertainties in the data and the risk of unknown local variations in conditions, however, the wall thickness of the canister should be chosen taking into account the greatest depth of corrosion damage that has been obtained in these calculation cases, with the addition of a safety factor. This principle for determination of copper thicknesses will be established.

Knowledge of how the corrosion properties of copper are affected by microstructure and alloying elements will be supplemented as needed depending on the choice of fabrication methods.

Heat transfer
Calculations of heat transport in the deep repository will be compiled as a basis for determination of a limit value for heat transfer from the canister to the bentonite.

Radiation dose
The limit for radiation dose from the canister will be finally determined based on what can be accepted with a view towards the risk of radiolysis of the water surrounding the canister.

Criticality
The canister will be designed so that the fuel remains subcritical even in the event of water penetration. Criteria for verifying calculations will be determined.

Chemical impact
The material in the canister must not chemically affect the performance of the other barriers. The copper material is not foreseen to have any such effect. No further studies are, therefore, planned in this area.

Mechanical impact
The canister's bearing pressure against the bentonite will be limited so that mechanical stability is preserved. When the design of the canister has been determined, a verifying calculation will be performed.

Reliability, fabrication and handling

Fabrication and inspection of canisters
A reliable method for fabrication and inspection of the necessary canister production with adequate quality

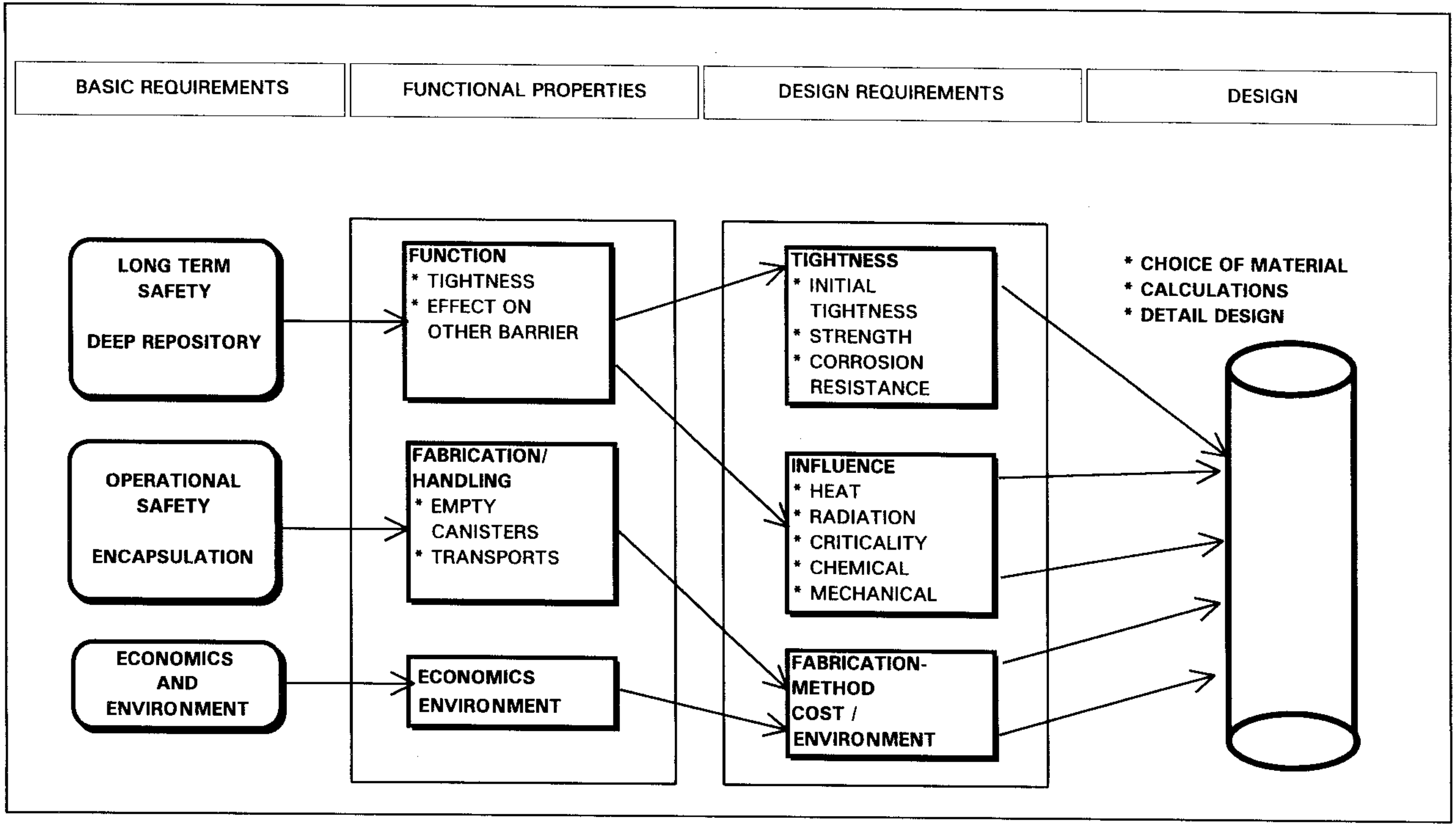


Figure 7-3. *Overview of steps in the canister design process.*

must be developed. Depending on the choice of method, this leads to analysis of the planned fabrication chain with respect to quality requirements and requisite inspections, such as:

- Microstructure of the material.
- Porosity and surface finish.
- Strength properties.
- Inspection of fabrication welds.

Transport of canisters
The canister with "packaging" must withstand transport without damage. Methods must be devised for examination and inspection for the determination of the canister's status after transport.

Handling of canister in encapsulation plant
The insert in the canister must be designed so that existing types of fuel assemblies can be emplaced with the equipment in the plant.

- The lid for the insert must be designed to meet the requirements of the encapsulation process on change of atmosphere in the canister and leaktightness on sealing of the copper lid.
- Radiation from the canister does not have to be limited for the sake of handling in the encapsulation plant, since the canister is provided with extra radiation shielding for different work operations.
- The canister must be designed for the necessary lifts.

Sealing and postweld machining
The copper lid must be designed so that it can be sealed using electron beam welding and so that inspection of the weld can be performed. The weld area must be designed to meet requirements on surface finish after machining. The canister must be designed so that the outside can be decontaminated if necessary.

Handling of filled canisters
Requirements for the integrity of the canister in the event of accidents during handling in the encapsulation plant or during transport in casks must be established.

Economy and environment

A compilation of economic and environmental criteria will be made. Different canister designs will be costed and evaluated with respect to economy and safety. The environmental effects of the chosen canister materials will be analyzed.

7.4.2 Design of the canister

When the design requirements have been established, the final work of determining the design of the canister will be carried out. This work can be divided into the following steps.

Material selection and tests

Mechanical testing and creep testing will continue to the extent required to:

- verify the outcome of trial fabrication,
- qualify materials from alternative fabrication methods.

Furthermore, testing of welded metal to verify the material properties of the weld will continue during the period.

The sensitivity of copper and steel to stress corrosion cracking will be further investigated.

The stability of the microstructure over long periods of time will be studied more closely.

As a basis for the final choice of material, the following will be done:

- Review of the target specification for the copper material and for the material for the inner container.
- Compilation and evaluation of possible materials for the canister.

Sizing

This work consists of several main parts, namely:

- Determination of the inside cross-sectional area and length with regard to the dimensions of existing fuel assemblies and the number of assemblies per canister.
- Design of necessary inserts or guides for the fuel assemblies in the canister.
- Sizing of wall thicknesses for steel and copper based mainly on the sizing criteria regarding mechanical stability and radiation.
- Check calculation for accident cases during handling and transport.
- Check of overall dimensions with regard to requirements on limitation of heat transfer and mechanical impact on the bentonite.

Criticality calculations

A programme to verify that the fuel will remain subcritical even in the event of water penetration will be carried out. The calculations will, with a progressively increasing degree of detail, shed light on the influence of the following factors, among others:

- Geometry and material properties of the chosen insert with variations within given tolerance limits.
- Possible presence of neutron absorbers in the insert material as well as of material defects such as cavities.
- External conditions such as neutron reflection from steel, copper, bentonite and water.
- Large changes in geometry caused by water penetrating into the canister, with consequent corrosion of the insert material.
- Possible crediting for fuel burnup.
- Change of isotope composition and reactivity in the very long time perspective if credit has been taken for fuel burnup.

For long-term safety, criticality analyses will also be carried out for cases with other hypothetical distributions of uranium and plutonium outside the canister in the deep repository.

Detailed design of lid

The inner steel lid will be designed in detail with a view to the requirements that must be met for lifting and attaching the lid and possibly additional requirements from the encapsulation process.

The copper lid will be designed in detail with a view to the requirements that must be met for lifting the lid and possibly the entire canister and for the execution and inspection of the sealing weld.

Chemical environment in the canister

In the event of water penetration, the chemical environment in the canister is affected by the groundwater, the buffer material and other materials in the canister's near-field, as well as by the materials in the fuel and in the canister itself. The work of studying the effect of such material mixes is continuing.

Gas formation in the canister in the event of water penetration is affected by the choice of material for the inner container. Studies have been made of inner containers of steel. Further studies of hydrogen-generating corrosion of cast iron are planned.

7.4.3 Alternative canister designs

The previously studied design of the copper canister with molten lead fill around the fuel assemblies constitutes the reserve alternative to the copper canister with inner steel container. No studies of the lead-filled canister are planned for the time being. If the results of the development work during the next few years should show that the canister with inner steel container does not meet the stipulated requirements, work on the lead-filled canister can be resumed. However, this may affect the time schedule for the encapsulation plant, depending on when such a decision is made.

7.5 DEVELOPMENT OF FABRICATION METHOD

The canister's copper shell is planned to be made by fabrication of a copper cylinder which is provided with a copper bottom. The bottom and copper cylinder are joined together by an electron beam weld. An inner container is placed in the copper shell prior to delivery to the encapsulation plant.

Two methods for the fabrication of copper cylinders have been tested and evaluated (see chapter 6). These trials show that copper cylinders of the planned dimensions can be fabricated with the necessary quality. Studies of other conceivable methods for serial production of the copper shell for the canister have also been made.

Inner containers in the form of steel cylinders have been trial-fabricated for use in the first canisters. Trial fabrication of cast inserts of steel and iron is planned for the near future.

So far, the studies of fabrication methods have been focused on determining whether it is possible with available methods to produce canisters of the planned design. The continued work will now be focused on development of the fabrication and inspection technology so that canisters with suitable and quality-assured properties can be fabricated industrially.

Fabrication of additional test canisters is planned to be carried out in several steps. These test canisters will be used to test the sealing method and used later in the inactive trial operation of the encapsulation plant. In the next step, development of the fabrication and inspection technology will continue with additional canisters until the permit application for the encapsulation plant is ready.

The trial fabrications planned at present are:

Cast inserts: steel, spheroidal graphite iron, possibly bronze

Trial fabrication of cast inserts will be carried out to explore the feasibility of full-scale fabrication of inserts. In the case of cast steel, which can be welded with satisfactory results, the canister insert can be fabricated in two parts, each of half length, which are welded together.

Rolled and welded copper tubes

Copper canisters for cast inserts will be fabricated, in part to gain experience from several suppliers of rolling feedstock and in part to gain experience from several fabricators of forming tube from formed plate.

Extruded copper tubes

The extrusions done thus far have given satisfactory results with respect to roundness and straightness. However, the extrusions were performed at high temperature (800°C), which may be the reason for the observed grain growth in the material. Extrusion at lower temperature (600°C) can produce a better microstructure in the material. Furthermore, the copper shell for the cast insert will have a larger diameter than the previously extruded tubes.

In parallel with these continued tests using already tested methods, tests may be carried out with the HIP method and with electrodeposition. Further studies will be made before a decision is taken on this.

Hot isostatic pressing

For the HIP method, studies are planned on subjects such as:

- availability and quality requirements for copper powder.
- necessary vacuum for obtaining an acceptable product, and the feasibility of achieving this on a 5 m long canister.
- fabrication of specimens for material tests, examination of material structure and functional testing (ultrasonic testing, weld testing).

The results of these studies and material tests will decide whether trial fabrications are to be done. In view of the availability of HIP plants, only one canister of reduced length but full diameter can be fabricated for the time being. Fabrication of a full-length canister requires investment in a new HIP press.

Electrolytic deposition of copper

The current manufacture of electrolytic copper material for testing will continue. Testing will include material testing, examination of material structure and functional testing (ultrasonic testing, weld testing). The results of these tests will decide whether the programme will be expanded to fabrication of model canisters of full diameter.

Evaluation

With these trial fabrications and alternative studies as a basis, evaluation will be made of which methods can be suitable for serial fabrication. An assessment of the costs of serial fabrication of canisters with different methods, and of the flexibility in choice of suppliers provided by the method, is also required as a basis for the evaluation.

Fabrication methods that provide equivalent end products will be evaluated according to the cost required to achieve the right quality with a given method. This means that all fabrication requirements on the canister must have been quantified in order to evaluate a method correctly. In the evaluation of different fabrication methods, tried-and-tested methods may be preferable to new and unproven solutions.

The operating period for the encapsulation plant will extend over a time span of more than 40 years. The possibility cannot be excluded that several different fabrication methods will be used during this time if alternative methods are developed and become available.

7.6 DEVELOPMENT OF SEALING METHOD

A method for sealing, employing electron beam welding is required to meet the stringent requirements on sealing of the copper canister. Methods for nondestructive testing will also be developed and used to verify that the seal complies with stipulated requirements.

The work of developing of the sealing method is planned to proceed in the following steps.

Electron beam welding

The equipment that has heretofore been used in tests needs to be further developed in order to meet the requirements made in the encapsulation process, when it must be possible to seal one canister per day. This work has begun with the power supply unit, where a new unit is currently under development to achieve higher reliability.

A test series with an inclined weld as an alternative to the horizontal weld used up to now is planned next. This would simplify the lid design and reduce the risk of operating disturbances and problems when welding in a radioactive environment.

Inspection methods

To guarantee satisfactory integrity of the canister, rigorous in-process inspection of the metal containers and post-sealing inspection of the canister will be carried out. Methods for these inspections will be determined and, if necessary, developed simultaneously with the determination of methods for fabrication and sealing of the canister.

The ultrasonic testing method, particularly processing and interpretation of signals, will be developed. This work has begun and will continue in connection with continued welding work. The plan includes design, fabrication and testing of equipment for serial fabrication of sealing welds.

Use of x-ray (radiographic) testing as an alternative or complement to ultrasonic testing has been conducted on pieces of the test lids with good results. As in the case of ultrasonic testing, the activities planned for radiographic testing during the next few years include design, fabrication and testing of equipment suitable for use in the encapsulation process.

Eddy-current testing is another method whose suitability will be studied and evaluated for use in the inspection work.

Acceptance criteria will be formulated for nonconformances found in the testing at the different steps in canister fabrication and sealing welding.

7.7 PILOT PLANT FOR CANISTER SEALING

A pilot plant is being considered for development and testing of equipment for sealing welding and nondestructive testing. The equipment for this plant will be designed with the intention in mind that it is suitable for incorporation in the encapsulation plant's welding station. The experience will be utilized in determining the final design of this part of the encapsulation process.

The equipment manufactured for this testing purpose may otherwise be put to use in a plant for the fabrication of canisters where, depending on the fabrication method, such equipment may be needed for welding the bottom on the copper canister.

The main reason for the construction of a pilot plant is to obtain a solid basis for the continued design of the encapsulation plant. Without this verification of the practical function of the equipment, it would be necessary to design and build the plant with large and costly flexibility to permit any necessary modifications to be made during trial operation.

The work on the pilot plant is planned to proceed in the following steps:

- Preparation of function specifications.
- Basic design.
- Detailed design and procurement.
- Construction and installation.
- Commissioning.

An initial trial series should be carried out, evaluated and the results reported before a decision is taken to start construction of the encapsulation plant. The plant will then be used for continued operating tests of equipment and for training of operating and maintenance personnel.

7.8 CONSTRUCTION AND OPERATION OF THE ENCAPSULATION PLANT

A decision by SKB on construction of the encapsulation plant is planned to serve as a basis for the applications for the necessary permits for construction and operation. The forms of procurement and construction of the encapsulation plant are planned to be established in conjunction with this decision. Preparations can be made for procurement of building and installation works during the licensing process so that it is possible to commence the building works as soon as the permit has been obtained.

The building and installation works are of the same character as the previous works for CLAB. After the commissioning of the various items of equipment, a period for the integrated functional testing of the entire encapsulation process is planned (inactive trial operation). A series of canisters will undergo the various handling steps before irradiated fuel is admitted to the encapsulation plant for an initial trial operation period (active trial operation). The results of trial operation will then be compiled, evaluated and reported on. The reports will then serve as a basis for an application for a permit for routine operation of the encapsulation plant.

In a future stage, the encapsulation plant will be augmented with installations designed to embed core components in concrete. Preparations will be made in the initial construction phase for the necessary building and installation works so that the operation of the encapsulation process will be disturbed as little as possible during this stage. Commissioning and trial operation will be carried out in the same way as described above for the encapsulation process.

Transport casks for canisters and core components will be fabricated, tested and approved before they are put into operation. The first cask will be produced at an early stage for integrated functional testing with the transport system and handling equipment in the encapsulation plant and the deep repository.

7.9 SAFETY, QUALITY AND SAFEGUARDS

Safety reports

Matters pertaining to radiological safety in the encapsulation plant will be analyzed and reported in two steps.

Preliminary safety report, PSR
The safety of the plant will first be described based on the design work carried out during the period 1994–97. The level then achieved will be termed layout D. The report should describe safety-related matters for the handling process, especially possible accidents and their consequences.

Final safety report, FSR
Safety will then be reported for the finished plant. The background material for the report will consist of the plant description and final system descriptions, layout A.

The results of inactive trial operation with fabrication of a number of canisters will be reported in 2006 as a complement to the FSR for regulatory review of operating permit applications under KTL.

Matters pertaining to the long-term performance of the canister are dealt with in the safety assessment for the deep repository.

Performance and risk assessments

The encapsulation plant with constituent systems will be analyzed with respect to normal operating functions and the risks of disturbances that can be foreseen. This work will be pursued in several steps during the design process. The operating cases and associated consequences that emerge in this manner will be evaluated and remedied with appropriate design changes.

Quality assurance

The overall objective of quality assurance within the encapsulation project follows "SKB's guidelines for quality assurance work".

The goal of the quality work in the project is to ensure that:

- the plant and the canisters have the quality that is needed for safe performance and handling of the canister and for safe operation of the plant,
- the documentation that is prepared during the licensing, design and building process is correct, clear, traceable and available for scrutiny,
- the requirements of the authorities on quality assurance for nuclear facilities are met,
- the project work is efficient with clear descriptions of goals as well as the responsibilities and powers of the organization, and that
- information on the progress of the project work is disseminated to all concerned.

A project manual has been compiled as a tool for facilitating the work in the project and as an instrument to ensure that the project work is conducted with such good quality that established project and quality goals can be achieved. Consideration has been given to "Quality systems – Model for quality assurance in design/development, production, installation and servicing" (SS/ISO 9001) where applicable.

Quality assurance, canister production

A quality programme comprising inspection methods and acceptance criteria will be established for the various work operations in the fabrication and sealing of canisters. The final properties of the canisters will be documented via the inspections performed on them. Data from the inspection records will be transferred to the safety assessment together with data on the fuel assemblies that have been encapsulated in each canister. Special routines for keeping of records on finished canisters, i.e. inspection documentation for the canister with fuel assemblies, must be devised for this purpose. A method will be developed to mark the canister for identification during handling and final disposal.

Safeguards

The handling of spent nuclear fuel is subject to the IAEA's (International Atomic Energy Agency) rules on safeguards. Oversight is also exercised through the offices of SKI and Euratom.

The planned encapsulation process will therefore be analyzed with respect to requirements on safeguards for the spent fuel. Measures will be adopted in the design work to conform to such requirements as e.g. radiation monitoring of fuel assemblies, surveillance of possible exit routes, identification of canisters and installation of surveillance equipment. Quarters must also be provided in the plant for Euratom's inspectors.

Each canister will be provided with an identification that is unique for that canister, and a record will be kept of its fuel contents. Methods and routines must be devised for this.

Matters pertaining to safeguards will be duly dealt with and described to SKI and the international bodies.

8 STATE OF KNOWLEDGE - DEEP REPOSITORY

This chapter describes how far the work of planning a deep repository for spent nuclear fuel and other long-lived waste has progressed. The account is general, and more detailed information on the results achieved to date can be obtained in the series of reports published by SKB. Plans for future work are presented in chapter 9.

So far the work with the deep repository has mainly included the following activities:

- preparation of plans for the design, construction, operation and closure of the deep repository,
- execution and compilation of siting studies (general studies, feasibility studies) as a basis for the future selection of a site for the deep repository,
- preparation of plans for site investigation and evaluation.

The main components in the siting work are illustrated schematically in Figure 8-1, including what has been or is currently being done for each component.

In relation to RD&D-Programme 92, it is taking longer to carry out the first stages in the siting process than was assumed at that time. Another change is the Government decision of May 1995, which shifts the weight in the regulatory scrutiny of the siting of the deep repository to the application for a permit for detailed characterization (see chapter 9). The current situation regarding the above-mentioned activities is as follows:

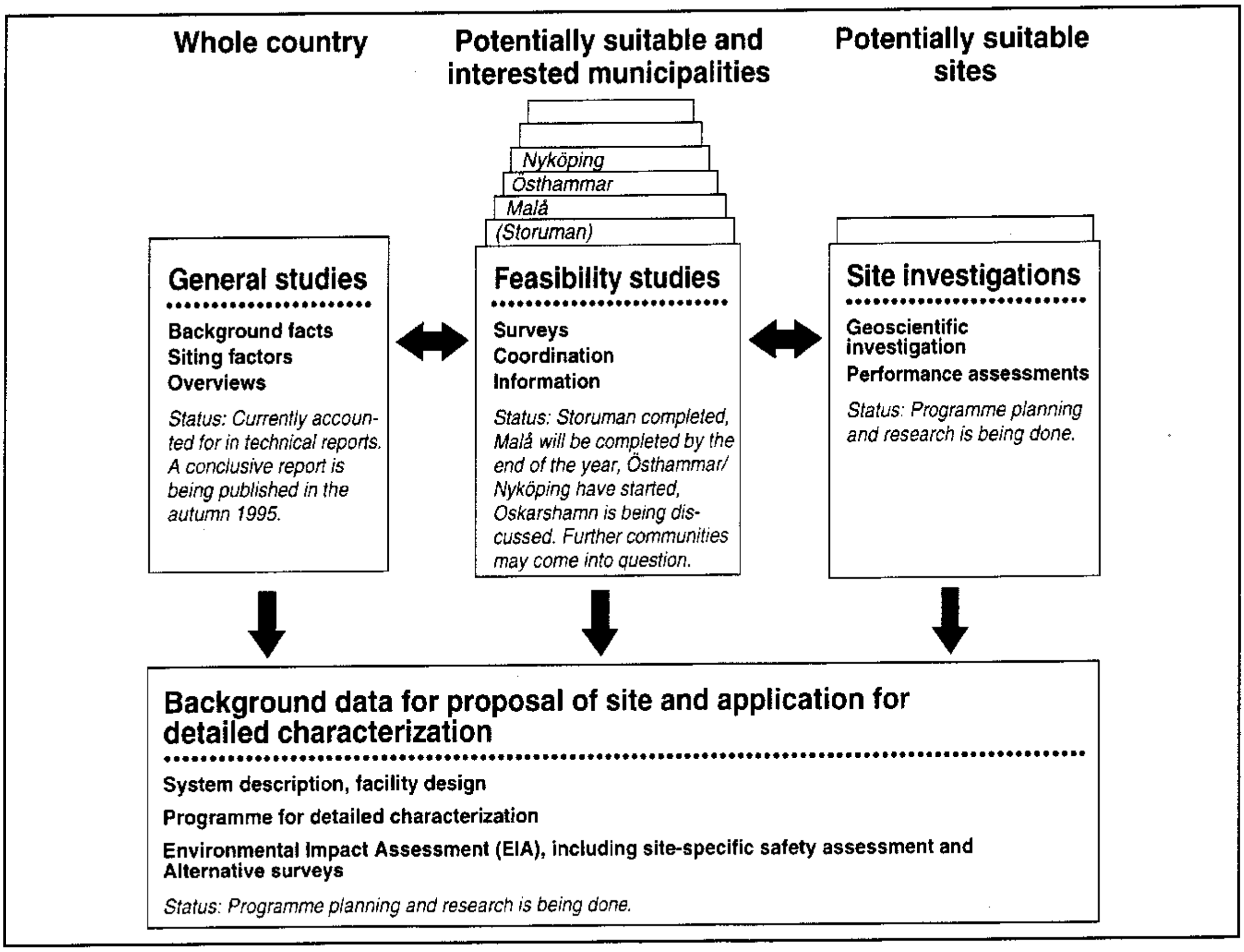


Figure 8-1. Main components in the siting work, plus completed and ongoing activities.

The planning work for the design and operation of the deep repository lies well in phase with RD&D-Programme 92. Carefully worked-out plans now exist for how the deep repository in general terms can be designed and how transportation can be arranged. The next planning stage with a higher level of detail will start when potential sites have been reported, i.e. during the site investigation stage.

General studies of the whole country and a feasibility study have been carried out. Other feasibility studies are under way or will presumably begin in the near future. It is now estimated that the site investigations will begin in 1996 at the earliest (see chapter 9), which is a couple of years behind the schedule presented in RD&D-Programme 92.

The preparation of a general site investigation plan is in phase with other siting activities, which means that the general scope of investigation and a plan for its implementation will be ready in good time before site investigations begin.

8.1 DESIGN, CONSTRUCTION, OPERATION AND CLOSURE OF A DEEP REPOSITORY FOR SPENT NUCLEAR FUEL AND OTHER LONG-LIVED WASTE

A deep repository is a medium-sized industry with facilities both above and below ground. How these facilities are to be designed so that they provide optimal performance from a technical, environmental and safety point of view requires long-term and careful planning.

Key questions for achieving a well-functioning deep repository are:

- how should the surface and underground facilities be designed?
- what building methods are suitable for the underground facilities?
- how should deposition be carried out?
- how can the deep repository be closed?
- how can provisions be made for retrievability and monitoring?
- how should a good working environment be ensured?
- how can it be ensured that all spent fuel is really deposited in the deep repository and that it is not secretly taken away (safeguards)?
- what are the environmental effects of construction and operation?
- what are the effects of the facility on the community where it is sited?
- how can shipments of radioactive waste and backfill materials be arranged?

The present-day state of knowledge on these questions is described below. An account of shipments to the deep repository is given in section 8.4. The present-day situation regarding alternative disposal concepts is described in chapter 13.

8.1.1 Facility design

The central activity at the deep repository is to receive canisters with spent nuclear fuel and deposit them in selected positions approximately 500 m down in the rock. During regular operation, certain other radioactive waste will also be deposited in the deep repository. Doing this requires:

- geoscientific investigations, planning and design, tunnelling, blasting of rock caverns, boring of deposition holes, etc.,
- transport down under ground and emplacement of the canister and surrounding bentonite buffer in deposition holes,
- transport down under ground and emplacement of other radioactive waste in rock caverns,
- post-emplacement tasks such as instrumentation, backfilling of deposition tunnels and rock caverns, inspection, etc.

In support of these central activities, the following machinery and equipment is needed at the deep repository:

- deposition system, machines,
- transport plant (track etc.),
- receiving station above ground,
- machine shop,
- facility for stocking and preparation of bentonite and backfill materials,
- buildings for offices, admissions control, canteen and information,
- installations for utilities (ventilation, water, sewage).

Chapter 9 describes the planning of these activities. In general, it can be said that tried-and-tested methods and machines exist for most of the functions that are planned to be included in the surface facilities. A schematic illustration of a possible layout of the surface facilities is shown in Figure 8-2, in which a flat industrial area has been assumed. There are ample opportunities for adapting the layout to the local topography and other conditions on the actual site. The space requirement for the surface facilities is estimated to be about 0.3 km^2.

Requirements and principles for the different functions of the deep repository have been studied and compiled in Facility Descriptions /8-1, 2, 3/, which are predicated on general assumptions regarding the conditions

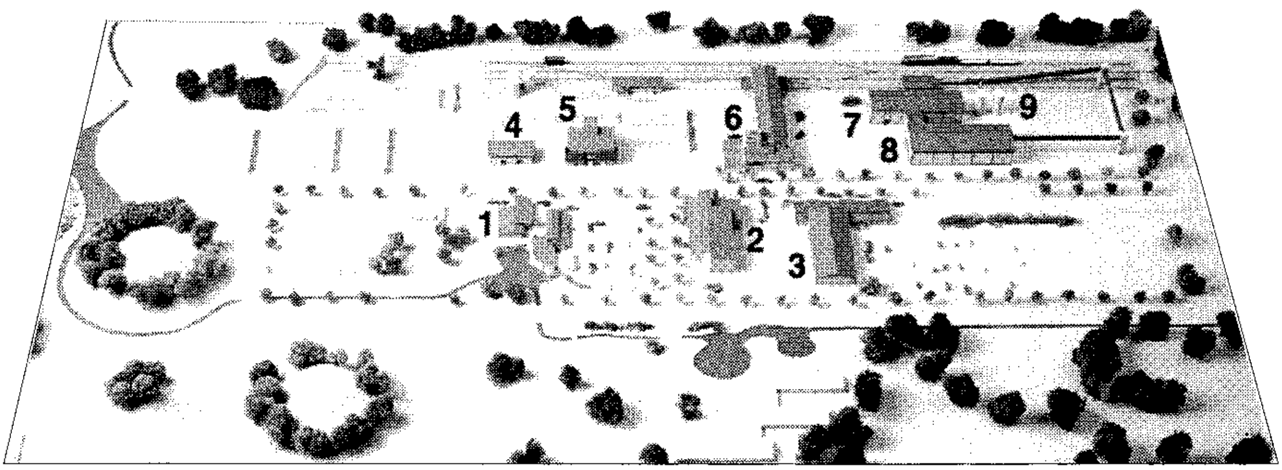


Figure 8-2. Schematic layout of the deep repository's surface facilities. 1. Information and restaurant; 2. Offices and workshop; 3. Personnel quarters and stores; 4. Utilities building (water/heat); 5. Ventilation building; 6. Operations building (reception and inspection of containers); 7. Production building (for bentonite blocks etc.); 8. Sand store and 9. Bentonite store.

where the deep repository may come to be sited and on general rock engineering assumptions based on the results of the investigations conducted by SKB on study sites. The design and layout of buildings, land areas, rock caverns, tunnels, shafts etc. above and below ground are exemplified in the documentation. The deposition of spent fuel is exemplified for the KBS-3 reference concept, see section 2.2, while the deposition of other long-lived waste is exemplified with an SFR-like method. Three different descriptions have been done for the three fundamentally different access systems:

- straight ramp,
- shaft,
- spiral ramp.

Which access system is most suitable depends on technical factors as well as local conditions. There are, for example, good prospects for siting the surface facility so that both technical requirements on ground conditions and competing land-use interests are taken into account, at the same time as the underground facility is sited optimally from a safety point of view. This can be done by separating the surface and underground parts laterally by means of a ramp or, in the shaft alternative, by excavailing a tunnel of equivalent length at the deep repository level. Figure 8-3 shows a schematic drawing of how the deep repository might be configured with a ramp and how the underground facilities can be displaced laterally in relation to the surface facilities.

The facility descriptions also assume that the entire deep repository is located on a single level. During the 1980s, a scheme was outlined with deposition galleries on two levels separated by 100 m /8-4/. The option of planning for more than one level remains open, but will probably only be considered in special cases when the horizontal extent of good rock is limited.

Studies have indicated alternative layouts that make better use of the tunnels than is possible with one canister per hole in vertical holes beneath the tunnel floor. The alternative is horizontal emplacement in holes bored in the tunnel wall, whereby holes can be bored in both directions /8-5/. A further development is a system with two canisters in each hole /8-6, 7/.

Final disposal of long-lived low- and intermediate-level waste is planned to take place in a special part of the deep repository located separate from the repository sections with high-level waste. Since there are different types of low- and intermediate-level waste, three disposal caverns are needed (named SFL 3–5), see Figure 8-4. These caverns are designed so that the different needs of the different waste types for handling and disposal are taken into account.

One waste type is long-lived waste from Studsvik. This includes some of the waste which Studsvik collects from research (internal and external), industry and medicine. This waste is conditioned and packaged at Studsvik, see Figure 8-5. It also includes some older, already packaged waste. This type of waste will be lowered into separate compartments in an underground concrete structure. The remaining spaces in the compartments will probably be filled with concrete. The backfill material in the space between the rock and the concrete structures may, for example, be sand, crushed rock or bentonite. Operational waste from CLAB and the encapsulation plant that arises after SFR has been closed will also be deposited in the concrete structure. Accordingly, only some of this waste can be called long-lived. A large portion consists of waste of the type that is deposited today in the final repository for radioactive operational waste (SFR).

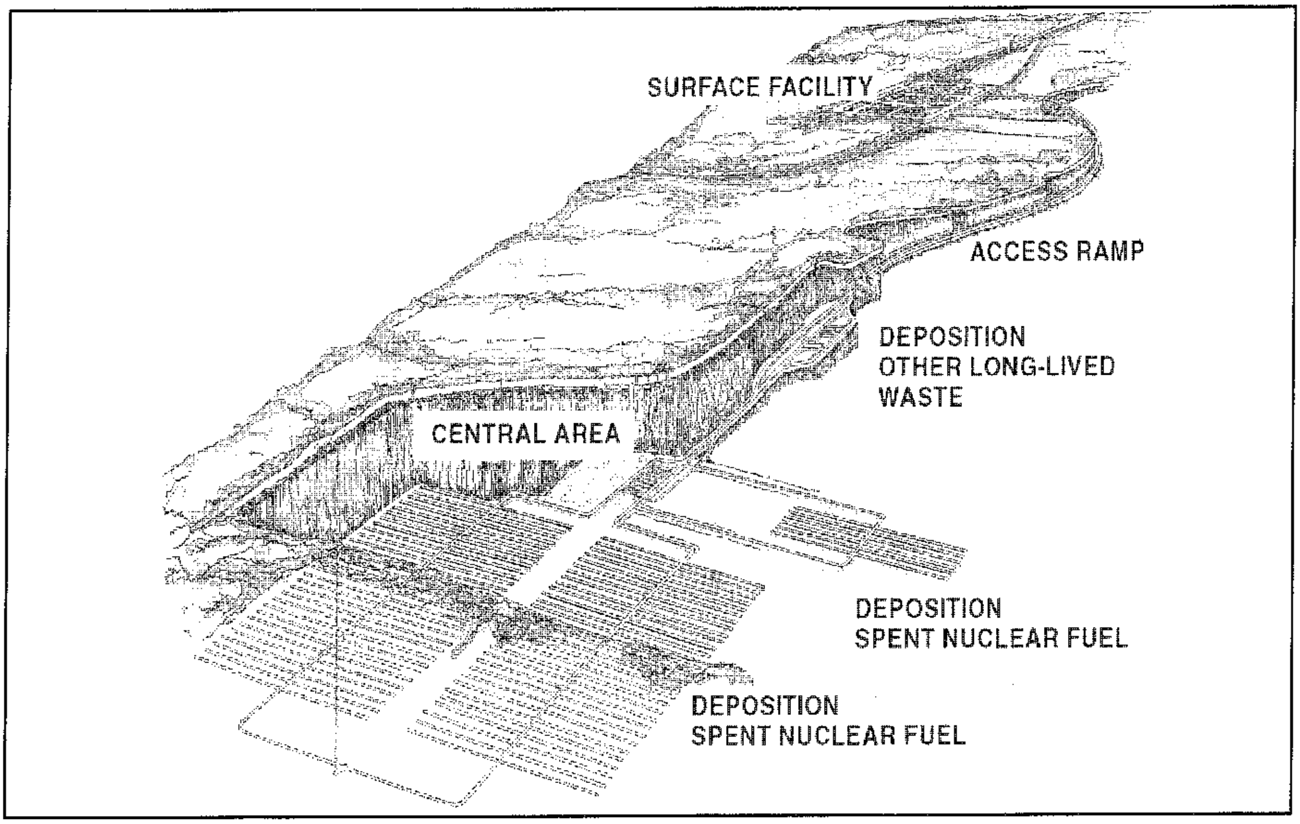


Figure 8-3. *Schematic layout of the deep repository.*

Another waste type consists of core components and reactor internals from the power-generating reactors. It consists for the most part of stainless steel. The intention is to package the components in concrete containers which are also backfilled with concrete. The waste will then be deposited in rock caverns with floors and walls of concrete. The rock caverns will probably be backfilled with sand or crushed rock.

Decommissioning waste from CLAB and the encapsulation plant, as well as CLAB interim storage canisters (if they have not been decontaminated) and transport casks, are types of waste that arise very late in the programme. The deposition chambers for these types of waste consist of the transport tunnels and cavities that are left after closure and sealing of other repository sections for low- and intermediate-level waste.

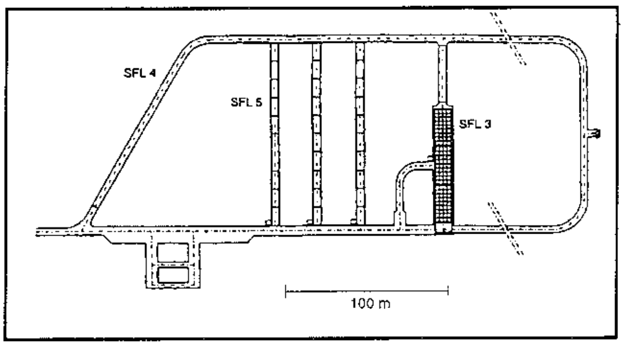


Figure 8-4. *Overview of SFL 3–5.*

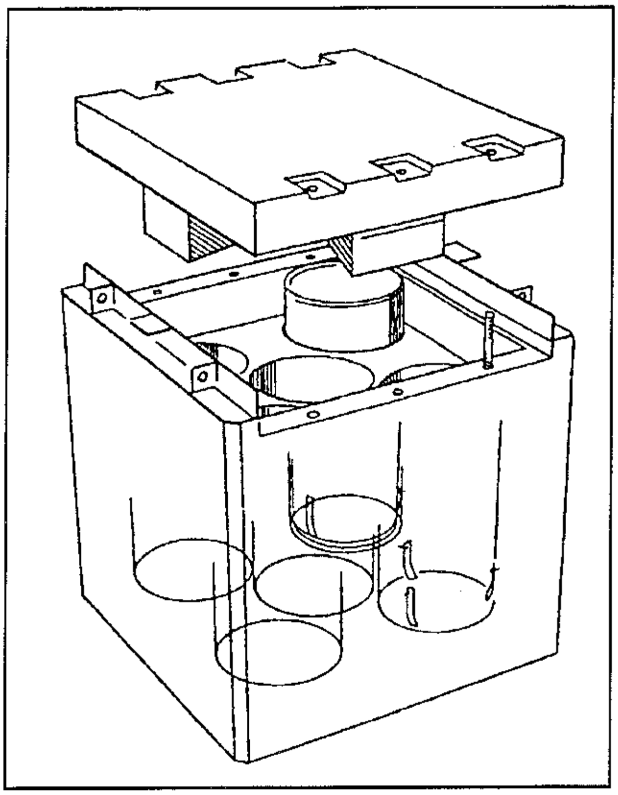


Figure 8-5. *Packaging of low- and intermediate-level waste at Studsvik. Five steel drums with double lids (85 litres) are placed in five holes in a concrete container (1.2x1.2x1.2 m).*

The total volume of low- and intermediate-level waste that is planned to be deposited in the deep repository is estimated to about 25,000 m^3. Reactor components and much of the Studsvik waste are counted as long-lived waste. More than half the volume consists of operational waste and decommissioning waste, which in principle could be deposited in the SFR repository in Forsmark.

Waste types and packages are described in a progress report from SKB /8-8/. The contents of the progress report are summarized in the final report on the prestudy of SFL 3–5 /8-9/. The design of the different repository chambers is described in SKB's Plan reports /8-1/.

8.1.2 Construction methods

A general survey of the state of knowledge regarding construction methods is done every year as a part of the PLAN work /8-10/. Generally speaking it can be said that the different methods for building in rock that may be used in the deep repository are well known. There are, however, knowledge gaps regarding the advantages and disadvantages of different construction methods with respect to the repository's performance after closure.

Recently, two tests of mechanized excavation methods (tunnelling with TBM and shaft boring using raise-boring equipment) have demonstrated the possibilities of the latest technology for hard, crystalline rock /8-11/. A test at Äspö gave experience from full-face boring of an inclined tunnel at repository depth and confirmed the expected reduction in the amount of rock reinforcement required compared with conventional tunnelling. The second test with full-face boring of deposition holes confirmed that the reverse raise-boring method can be used with good results for boring of deposition holes, see Figure 8-6. The cuttings are removed dry from the borehole by means of vacuum suction.

The valuation of and choice between tunnelling methods will hinge on how the benefit that is probably provided by TBM in the form of less reinforcement is weighed against the drawback entailed by the larger tunnel cross-sectional area. The latter is a consequence of the round cross-section.

An important question in this context is the use and testing of grouts for sealing against water inflows. Regardless of whether a large or small discontinuity is to be sealed, good knowledge is required of the rock, the grout and the injection technique in order to obtain good grouting results. Due to the complexity of this problem, practical grouting work has mainly been based on experience and only to a small extent on theoretical knowledge.

It is possible today to partially describe the grouting process in a rock mass on the basis of investigations and experiments in the Stripa Project /8-12/ and the Äspö HRL /8-13/. However, several essential factors remain to be studied before grouting can be described more accurately. Such development is described in greater detail in chapter 9.

8.1.3 Deposition technology

The reference concept for deposition from RD&D-Programme 92 is unchanged. In brief, it entails that the canisters with the spent nuclear fuel are transported down under ground in transport casks. There the canister is transferred to a deposition machine, which drives the canister up to the deposition hole and lowers it down into the hole. Before this the hole has been lined with blocks of highly compacted bentonite, leaving an inner space for the canister. When the canister is in place, bentonite blocks are placed above the canister and the hole is filled out with a mixture of sand or crushed rock and bentonite. The same material is then used for backfilling of the entire deposition tunnel.

It is assumed that the blocks of highly compacted bentonite around the canisters can be fabricated in the form of "pineapple rings", if these prove to be convenient to handle during deposition. Two different compaction methods have been tried: isostatic compression and uniaxial compression. The isostatic method was employed for fabrication of blocks for tests in Stripa. The state of knowledge for the axial pressing method is that 10–20 kg heavy blocks can now be produced using the same method as that used for pressing refractory brick /8-14/. The technology is based on a high compaction rate, which requires a coarse-grained bentonite material. At present, blocks that are suitable for manual application in KBS-3 holes can be fabricated with a dry density of 1.7–1.9 g/cm^3 and with a water saturation of 50–85% /8-14/. With moderate gaps in the deposition hole to provide room to build up the bentonite buffer and to permit emplacement of the canister, this block quality is sufficient to result in a buffer density of about 2.0 g/cm^3 in the water-saturated and swollen state. The buffer properties of the blocks are as good as those of blocks compacted from bentonite powder.

For backfilling of tunnels and rock caverns, crushed rock is being studied as a ballast material instead of sand with rounded grains, which was the main alternative in the KBS-3 report.

Preliminary results indicate that a suitable mixture for backfilling may be 10–20% bentonite and crushed rock. The mixture is deposited in layers and compacted with a vibratory roller or vibratory tools at the top against the roof. It is estimated that this material will have a hydraulic conductivity after water saturation of less than 10^{-9} m/s, which is in the same order as low-conductivity rock /8-15/.

Alternative backfill material being studied is crushed rock or till without bentonite, which is also deposited in layers and compacted on the site. Although the hydraulic conductivity of these materials will be higher than that of sound rock after compacting /8-16/, other factors can be advantageous.

Mishaps requiring some form of remedial action cannot be ruled out during deposition. Normally it should be possible to remedy the problem and continue deposition, but it is also conceivable that the problem cannot be

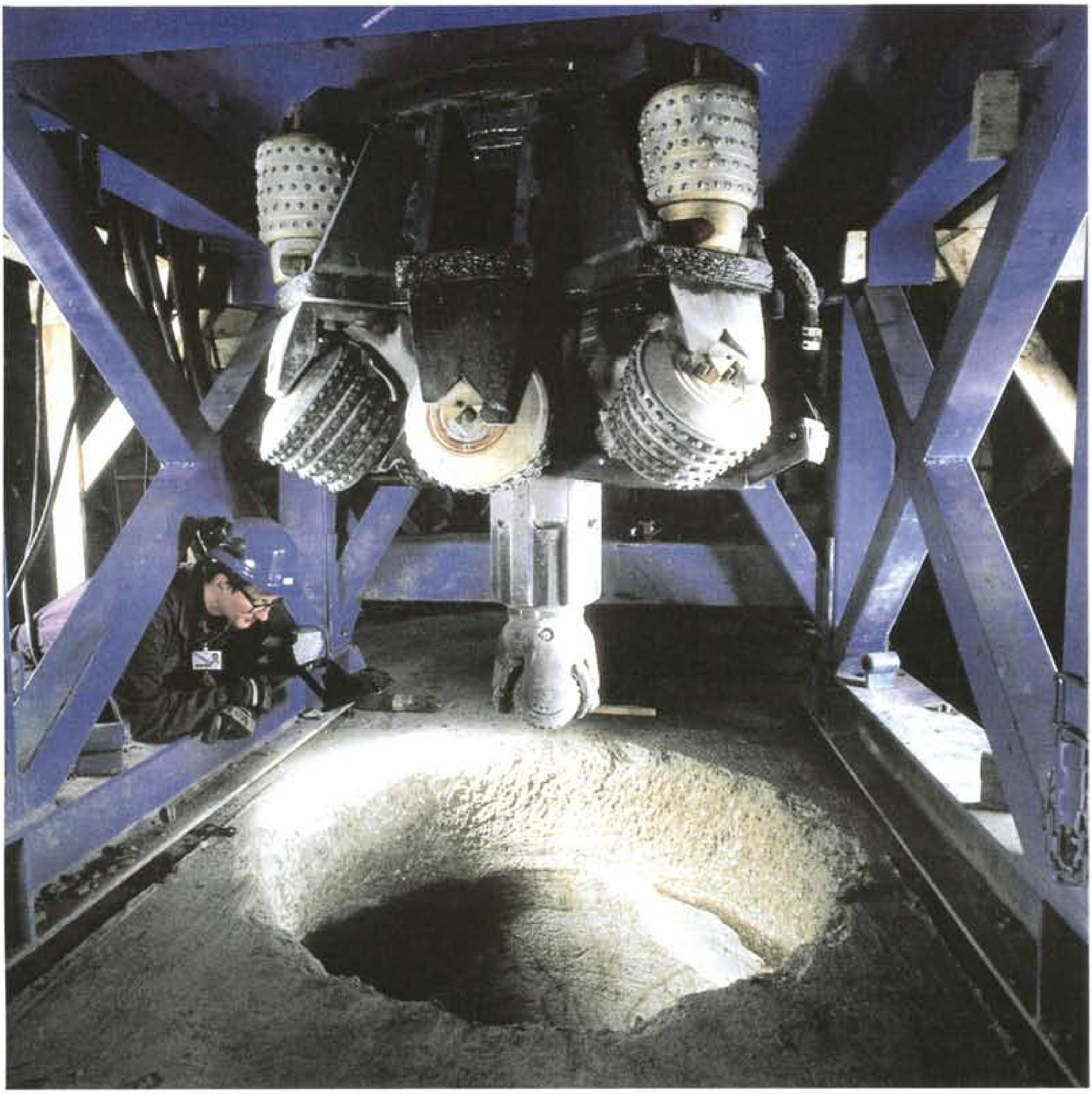


Figure 8-6. Test boring of a deposition hole. The photograph shows the cutterhead and the top of a borehole with a diameter of 1.5 m.

remedied except by removing the canister. For this reason, the option of reversing the deposition process is included as one of several functions in the design and testing of the deposition equipment.

Packages with other long-lived waste are transported down under ground in their transport containers and deposited in rock caverns using the same method as that used in SFR and with similar equipment. Since experience from SFR is good, no alternative method is being studied.

The overall assessment is that the equipment that is needed for deposition of canisters and packages can be built with duty-proven components and tested in the customary fashion.

8.1.4 Closure, retrieval and monitoring

The repository must be designed so that it is safety over a very long time, even if monitoring and supervision cease after closure and sealing. This can be achieved by backfilling and plugging tunnels and shafts. After closure the site is restored as nearly as possible to the conditions prevailing prior to establishment of the facility.

Closure

After deposition of canisters with spent nuclear fuel, the deposition tunnel is backfilled with a mixture of bentonite and ballast material and temporary walls are set up in the tunnel mouth. When all canisters have been deposited, the temporary wall can be demolished and the transport tunnels backfilled. The aim of backfilling is to limit possible transport pathways for groundwater. A special investigation has shed light on the technical possibilities of keeping the deposition tunnels open during different periods /8-17/. The conclusion is that it is preferable for technical reasons to close the deposition tunnels at an early stage.

Plugs are set at fracture zones and other places to further prevent or greatly restrict water flow in the tunnels, but also in the rock nearest the tunnels (the excavation-disturbed zone). Boreholes must also be plugged. Solutions for how the plugs can be designed have been presented in RD&D-Programme 92. Methods for plugging of tunnels and boreholes have been tested at Stripa (tunnels and boreholes), the Ranstad plant and SFR (boreholes).

Retrieval

A method and equipment are also being developed for retrieval after the canister has been deposited and the deposition tunnel backfilled. The purpose is to be prepared for a retrieval if this should prove necessary for some reason. As mentioned previously, the possibility of reversing the deposition process is therefore one of several functions included in the design and testing of the deposition equipment.

Monitoring

During the entire time the deep repository is in operation, there is an organization on the site for a similar type of monitoring as at other nuclear facilities. For canisters deposited during initial operation there will be programmes for measuring different parameters (pressure, temperature, moisture content, radiation level, etc.) e.g. in deposition holes and deposition tunnels. The details of such a programme will be worked out in the continued planning work and be based on experience obtained during the experiments in the Äspö HRL. Shafts, ramp, access tunnels and general areas in the deep repository will naturally be kept open and accessible throughout the operating period. Before the repository is closed, the initially deposited canisters will therefore have been observed for several decades to ensure that everything works as intended. It will also be possible to retrieve the initially deposited canisters and inspect them if additional verification is desired.

When all waste has been deposited, the generation that is then responsible for the facility can decide how to proceed regarding closure and future monitoring. The programme currently being run by SKB includes, as evident from the above, devising methods for complete backfilling and closure of the repository. The type of long-term monitoring that will be done of the deep repository and the site is primarily a question for each future generation to decide for itself. What can be done now is to stipulate and describe the technical possibilities for monitoring of the site and the conditions in and near the deep repository. In this context it is necessary to analyze what effect monitoring measures can have on both short-term and long-term safety.

8.1.5 Working environment

An introductory study concerning the radiological working environment in the deep repository has been conducted in conjunction with the feasibility study at Storuman /8-18/. Generally speaking, it is found that all types of tasks that will exist in the deep repository and along the transport pathway from the encapsulation plant and coastal harbour already occur today in various industrial contexts.

The activities at the deep repository will initially be dominated by rock excavation and underground construction, which in working environment terms can be compared with the development phase in a mine. Even when deposition of canisters has got under way, excavation will be carried out as the underground facilities are gradually extended. The working environment in those parts of the facility where excavation is not conducted can be compared to the working environment in power stations and similar underground facilities.

Experience shows that underground construction work entails greater risks of work injuries than many other industrial settings. A great deal can be done, and has been done in recent years, to reduce these risks. Technology improvements, strict safety procedures and good experience feedback are examples of important components in the work of occupational safety. The systems in the deep repository will be designed with this in mind. The same applies to the execution of the surface construction works.

Monitoring of the radiological working environment in conjunction with transport and handling in the deep repository will comply with the customary standard for nuclear activities. The fundamental principle is that radiation protection should be optimized and that the individual's protection should be safeguarded by dose limits. By "optimization" is meant that all measures that are defensible in terms of economic and societal considerations should be taken in order to reduce the total radiation dose. People's subjective anxiety about the risks posed by waste should also be taken into account.

During transport from encapsulation to the deep repository, the canisters with spent fuel are enclosed in transport casks with steel walls a couple of decimetres thick. The main purpose of the transport casks is to shield off the radiation from the canisters so that the casks can be handled without danger to the transport personnel. The casks also provide mechanical protection that prevents the canisters from being damaged during transport. The transport casks remain hermetically sealed until they have reached the deposition level in the deep repository. In the mouth of the deposition tunnel, the transport vehicle docks with a deposition machine, which opens the transport casks and removes the canister for further conveyance to its deposition hole. A few persons will be occupied with this work. Of those who work at the deep repository, only this personnel category can be expected to receive a measurable dose exposure from the waste being deposited. Extensive measures are taken to minimize the radiation doses to the personnel. The equipment will be remote-controlled or equipped with radiation shields, which means that the personnel do not come into direct contact with the canisters.

The dose exposure must naturally be limited in conformance with SSI guidelines for persons working with ionizing radiation. In practice, it can be assumed that the doses will be considerably lower than the maximum values counted on in the design phase. For example, experience from SFR at Forsmark shows that the radiation doses are a tenth of the maximum levels assumed when the repository was put into service /8-18/.

Since the deep repository is an underground rock facility, there is another source that can emit radiation, namely the radon gas released from the bedrock itself. Unlike the radiation from the waste, which only concerns the personnel who work with handling and deposition, all personnel who work under ground are exposed to radon. Today, the limit value for radon is 2,000 Bq/m^3 in all underground facilities (workplace for 2,000 hours/year). In the future it is probable that the limit value will be set at 200 Bq/m^3 of radon in the air, regardless of whether the workplace is situated above or below ground. To achieve this in a uranium-rich bedrock, extensive technical measures are required, notably with regard to ventilation.

In summary, it can be concluded that waste handling at the deep repository can be designed to stringent industrial safety standards. The personnel doses can be kept far below applicable limit values. Parts of the facility will be zoned according to radiation level. Dosimetry will be introduced. No airborne activity (except radon from the rock) or surface contamination will be present.

8.1.6 Physical protection and safeguards

Through international agreements, such as the Nuclear Non-Proliferation Treaty, the Euratom Treaty and other bilateral agreements, Sweden has undertaken to use nuclear materials for peaceful purposes only, and to keep records of all handling of nuclear materials. This also includes nuclear fuel. To ensure that the fuel is used in the intended manner, all handling is governed by physical protection and safeguards. The physical protection consists of the guarding and other security measures taken to protect the fuel from theft or sabotage. The safeguards exercised by SKI, IAEA and Euratom are aimed at ensuring that fuel is not diverted for use in nuclear weapons.

Through safeguards in connection with CLAB, the encapsulation plant and transportation to the deep repository, records are kept of which fuel has been placed in each canister. At the deep repository, it must be checked and certified that the right canister has been received and that it is deposited and remains in the deep repository.

A well-established safeguards policy exists for handling of spent fuel at the nuclear power plants and in CLAB. It includes documentation of all measures carried out as well as frequent and sometimes unannounced inspections by officials from the IAEA and Euratom. Such inspections are facilitated by the fact that the fuel is always accessible for direct verification and measurement. After the fuel is encapsulated, it is more difficult to measure the individual fuel assemblies in the canister directly. The safeguard measures must therefore mainly be based on identification of the canisters and continous recording of their handling.

It is judged that existing safeguard technology can be used for the most part for the encapsulation plant. Some development work may be needed, however, mainly to secure accurate identification of canisters.

When it comes to deep disposal, some questions arise in connection with the fact that the canisters are not available for direct identification and with the time scale of deep disposal.

The IAEA arranged a consultation meeting on these matters in 1988. At the meeting, the conclusion was drawn that spent fuel in a deep repository remains accessible, which means that it will have to be covered by safeguards in the future as well. The scope of the necessary safeguards was not discussed, however. Recently the IAEA took the initiative to preparing a background document for a safeguards policy for deep geological disposal of spent nuclear fuel. The background document was prepared within a support programme called SAGOR (Safeguards for Final Disposal of Spent Fuel in Geological Repositories). The work was begun in 1994 and has priority within the IAEA. From Sweden, experts from SKI and SKB are participating in the work, which is expected to be finished in 1996–97 and to result in a proposed safeguards policy for deep repositories.

Generally speaking, it can be said that encapsulated spent fuel deposited in a deep repository at a depth of 500 m is not easily accessible and that the measures that would be required to retrieve it and then reprocess it into plutonium are laborious, time-consuming and expensive. Furthermore, plutonium in spent fuel has a composition that renders it unsuitable for weapons purposes. Taken together, this means that the need for safeguards at the deep repository should be limited in the long-term perspective.

In order to reduce the risk of inadvertent human intrusion in a repository for radioactive waste, an attempt should be made to preserve knowledge of such repositories for future generations. If information is available, future generations will also be able to make informed decisions regarding possible changes of or in the repository.

In order to preserve the information concerning the deep repository for as long as possible, the important information (source term data, geographic location, design of the repository and background material for the safety assessment) should be kept filed on a durable medium. This information should be kept at several archives (local, regional, national and international). These conclusions were presented in a Nordic report in 1993 /8-19/.

8.1.7 The influence of repository depth on repository performance

In its comments on RD&D-Programme 92, SKI called for a systematic analysis of how repository depth influences the isolation function of a KBS-3 repository. Such an analysis is reported in /8-20/. This study takes into account depths down to 2,000 m, which lies within the limit of what can be defined as "mine depth", and describes the trends different factors can be expected to exhibit when the depth is changed. At great depth, i.e. 1,000 to 2,000 m, the most important positive factors are:

- lower groundwater flow (due to lower hydraulic conductivity, lower hydraulic gradient and higher salinity) and thereby reduced supply of corrodants and reduced leaching of radionuclides in the event of canister failure,
- longer transport pathway and longer travel time for radionuclides,
- reduced impact of glaciation and permafrost,
- reduced risk of human intrusion,

while the most important negative factors are:

- reduced constructability due to higher rock stresses and higher water pressure,
- greater difficulties with geological and hydraulic description of rock volumes,
- reduced buffer effect due to increasing salinity in the groundwater,
- higher degree of uncertainty and increased risk of unexpected events in conjunction with construction and operation.

At great depth, the rock stresses become unfavourable for a KBS-3-like repository, which presumably necessitates deposition of canisters in rows in tunnels. The temperature increases with the depth, which means that the waste must be spread out over a larger rock volume so that the maximum temperature will not be exceeded. In summary, it can be said that the advantages that are gained by a locating the deep repository at greater depth do not offset the growing difficulties that are encountered.

8.1.8 Environmental effects

When a site investigation begins, a plan must be drawn up of what is to be included in an environmental impact assessment. The latter should be compiled together with concerned municipalities, regulatory authorities etc. (EIA process).

In conjunction with the feasibility study at Storuman, an orientation study has been carried out of what the possible effects could be on land use and the environment of the construction, operation and closure of the deep repository /8-21/. The results are in large part general and independent of where in Sweden the deep repository is located.

The environmental study in Storuman has examined what a deep repository means for:

- impact on land use, natural and cultural environment,
- air pollution, noise and vibration,
- accidents and fire,
- impact on water,
- impact on flora and fauna,
- management of natural resources,
- site restoration.

In summary, it is concluded that the deep repository can be designed in such a way that it causes little environmental impact compared with ordinary industrial activities. There is no industrial process, few chemicals are present and the backfill materials, quartz sand (or crushed rock) and bentonite clay, do not contain any polluting substances. In connection with investigations, construction and operation of the facility, some drawdown in the groundwater table can occur in the same way as at mines. The most conspicuous environmental impact may be the rock waste dumps, and perhaps the construction of a railway or road up to the repository site. Rock spoils should therefore be re-used wherever possible. The placement of the facility, and any railway/road, in the terrain must be planned so that the impact on the environment, outdoor recreation activities, forestry and agriculture, hunting and fishing is little.

8.1.9 Societal effects of the deep repository

Establishment and operation of a deep repository will affect the locality and the region in different ways. The deep repository means jobs, population increase, support for the infrastructure and new opportunities for local business. However, it also means something new and unknown, which can be perceived as a threat and create anxiety.

The societal effects of a deep repository have been studied in the feasibility studies for Storuman and Malå. Surveys have examined the scope of direct and indirect job opportunities /8-22, 23, 24, 25/, psychosocial effects /8-22/, impact on tourism /8-26, 27/ and outdoor recreation /8-28/ as well as experience from the siting of other facilities of a somewhat similar character /8-29/ (nuclear power plants, SAKAB, oil refinery, mines).

A fully operational deep repository would entail about 200 direct and 100 indirect jobs. The population increase as a result of the deep repository would, on average, be about 500 persons, according to a forecast from Umeå University /8-22/. According to the same study, more than 30%, or about SEK 5 billion, of the total cost of SEK 15 billion could be absorbed locally and in the surrounding region.

How employment varies during the construction and operation of the deep repository, what type of skills will be in demand, and how the personnel are to be recruited has also been described in conjunction with the feasibility studies /8-22, 23/.

Projected employment is based on a relatively good idea of the scope of the future deep repository. As far as other societal consequences are concerned, for example impact on tourism or psychosocial effects, the evidence is uncertain. It is also possible to make use of experience from the siting of other similar facilities or estimates based on extrapolations of trends with different assumptions /8-26, 27/. It is unavoidable that these and similar issues are coloured by subjective judgements and evaluations /8-30/.

If a deep repository is sited at another location in Sweden, it is likely that the effects on employment and population will be roughly as estimated for Storuman and Malå. A siting in direct connection with an existing nuclear installation or other large industrial enterprise may slightly modify the personnel requirement, since coordination could reduce the need for certain personnel categories.

8.2 INVESTIGATION AND EVALUATION OF SITES

The general acceptance that radioactive waste can be disposed of safely in Swedish crystalline bedrock is based on knowledge of the properties of the waste, the properties and performance of the engineered barriers, and the properties of the host rock. The rock's most important properties in this context are that it offers:

- a chemically favourable environment,
- a mechanically stable environment,
- retardation of nuclide transport,
- an obstacle to human intrusion.

Even if the bedrock generally offers good conditions for a deep repository, it is the local conditions in the bedrock that determine the suitability of a site. The geological criteria that are important for the performance of a repository are described in RD&D-programme 92, Supplement /8-31/. Before site investigations begin, SKB intends to present an investigation programme that will describe how essential information on these criteria can be obtained. Key issues in designing the programme are:

- what is the experience from Swedish and foreign site investigations?
- what knowledge (data) is needed about the site and the bedrock in order to carry out design of the facility and safety and environmental impact assessments?
- what investigation methods and instruments are to be used?
- how should data management and processing be handled?
- what quality procedures and quality programme should exist?

8.2.1 Experience from site investigations

For the purpose of summarizing the results of the study site investigations, summary reports were published for most of the study sites during 1991 /8-32–37/. A summary evaluation and account of experiences from these investigations is currently in preparation.

Methodology and practical experience from abroad is ascertained in different ways. Among other things, SKB has ordered summarizing experience reports from the site investigations carried out in Canada and Finland /8-38, 39/. These countries have nuclear waste programmes similar to the Swedish programme, and similar geological conditions.

In Canada, AECL has been conducting geological investigations since the 1970s for the purpose of examining conditions in the Canadian bedrock. They have investigated a number of granites and gneisses, and a gabbro. During the past ten-year period the geoscientific investigation activities have primarily been concentrated to Whiteshell in central Canada, where an underground research laboratory, URL, has been established. Experience of potential importance for the Swedish site investigation programme has been published in /8-38/. In 1994, AECL published an extensive EIA on the Canadian nuclear waste disposal concept. One of the background reports provides a detailed description of the geoscientific data and how site evaluation and site screening will be carried out /8-40/.

In Finland, TVO has conducted site investigations on five sites. The results of these are described in /8-39/. The goal is to begin at the start of the next century what SKB calls detailed characterization, i.e. construction of investigation tunnels and/or shafts on a site. Preliminary site investigations have been carried out on five sites, and at present supplementary site investigations are being conducted on three sites.

Extensive work on devising and applying methods for site investigations has been done in the Stripa Project 1980–1992 /8-41, 42/ and at the Äspö HRL. Results from the latter are reported below.

In summary, there is a large body of experience of site investigations both in Sweden and abroad that can serve as a basis for designing SKB's site investigation programme. Other important considerations are data needs to serve the site assessment criteria, and adaptation to the local geological conditions.

8.2.2 Experience from the Äspö Laboratory

An important part of the activities at the Äspö Laboratory during the period 1986–1994 has been to test and develop methodology for site investigations and detailed characterization. One of the purposes has been to study to what extent investigations from the ground surface and in boreholes are capable of describing conditions at a depth of 500 m of importance for safety. The pre-investigation phase (site investigations) was concluded in the autumn of 1990 /8-43, 44/. An evaluation of the investigation methods used and their practical application has been published /8-45/.

A strategy for the site investigations and follow-up of these investigations has been published /8-46/. On the basis of the site investigations, several models were devised of the bedrock around Äspö. They included e.g. rock types, geological structures, groundwater chemistry, geohydrology and mechanical stability /8-44/. To test the models, detailed forecasts were also made of the expected data that would be collected during the construction of the laboratory /8-47/.

The final evaluation of the results from the Äspö investigations and the reliability of the predictions made will be reported for the most part during 1996, but the following preliminary conclusions have been reported /8-48/ and are summarized below:

Existence/absence of processes, structures and geological phenomena and geometric description

- The rock types that were identified during the site investigations have been encountered during the construction work to a depth of 460 m. No other rock types have been identified and none are absent.
- The site investigations showed that the different rock types on Äspö Island were very heterogeneously distributed and no detailed spatial distribution of the rock types was reported. This heterogeneity has now been further verified.
- The large fracture zones, with a width of more than 5 m, which were identified in the site investigations have been encountered in the tunnel. No new, unknown large fracture zones have been encountered.
- The geometry of the zones has for the most part been described correctly. One zone of minor importance for construction with a complex and varied dip was found to dip easterly instead of westerly, as had been interpreted on the basis of the site investigations.
- On the whole the site investigations were reliable for describing geometric phenomena.
- Using a specially developed nomenclature for Certain, Probable and Possible, a gently dipping fracture zone was regarded as being geohydrologically Probable and geologically Possible. This zone has not been found, however.
- The small fracture zones (width <5 m) in the north-northwesterly direction that were identified as being significant water conductors have been confirmed.
- Saline groundwater has been encountered largely as expected.
- The chemistry investigations have indicated a new process that was not taken into account in the interpretation of the site investigations, namely bacterial oxygen and sulphate reduction, see section 5.5.5.

All in all, the site investigations on Äspö have proven to be reliable in identifying the existence or absence of geological characteristics, structures and chemical-physical processes.

Parameter estimates

The work of comparing parameter values established from site investigations with parameter values obtained in conjunction with the construction work is under way; the last data that was used for the comparison was obtained in the summer of 1995.

Preliminary conclusions indicate that geological and geohydrological characteristics predicted on the basis of the site investigations have been determined with the foreseen degree of accuracy. Higher salinities have been measured in the tunnel that was foreseen in the predictions. This suggests that groundwater from greater depths, where salinities are higher, have been raised up to the tunnel faster than estimated. The deviation may be due to deficiencies in data used in the modelling of the transient process in the groundwater regime during the construction period.

Miscellaneous

Besides purely geoscientific results, other results have also been obtained. The practical work of carrying out a large number of technical and scientific experiments in coordination with ongoing construction works has yielded invaluable know-how. The work includes practical measurement technology, data management, quality procedures of all kinds including quality control, project management, documentation, etc.

Much of the results and experience obtained at Äspö concerning site investigations and detailed characterization has been published in technical reports and progress reports. The final report will be published in 1996. The Äspö work provides an important basis for achieving high quality in planning and execution of site investigations and detailed characterization.

8.2.3 Data from site investigations

A site investigation should:

- lead to a geoscientific understanding of the site and its regional environs with respect to current conditions and naturally ongoing processes,
- provide geoscientific parameter data for site-adapted layout of the deep repository with its tunnels and shafts and for assessment of the deep repository's long-term performance and radiological safety,
- provide data of importance for an environmental impact assessment of the deep repository, including transportation.

Data and results from the site investigation are thus used in performance and safety assessments for calculations and descriptions of the natural and engineered barriers for a deep repository on the investigated site. Important factors include the chemical environment, stability of the bedrock, conditions of importance for how radionuclides can move through the bedrock, risk of human intrusion and biosphere conditions. Data are also used for modelling and evaluation of the long-term radiological safety of the deep repository for different scenarios and with alternative model and parameter choices. See further in chapter 10.

In the planning and design work, site data are used to determine layouts for the different parts of the deep repository, with adaptation to local site and bedrock conditions. Important data are rock type, fracture frequency, positions and characters of fracture zones, hydraulic conductivity, size and orientation of rock stresses and mechanical properties of the bedrock. Further, construction analysis is performed, i.e. calculations and analyses of the properties and limitations of the rock with respect to construction technology and occupational safety.

Data from the region are used to propose and analyze alternative transport routes and modes of transport for the radioactive waste and the backfill material.

Site selection and plant design should be done so that conflicts with competing interests are minimized. Consideration should thereby be given to nature, environment, cultural monuments, recreation, hunting, fishing, other outdoor activities, important natural resources, agriculture and forestry, existing and planned land use. General data concerning these conditions are gathered and analyzed in feasibility studies. In a site investigation, deeper studies regarding the aforementioned conditions are conducted, including inventories of fauna and flora in interesting areas. The results of data analyses are presented in an environmental impact assessment. Section 3.4 describes how the EIA process may be designed.

8.2.4 Method and instrument development

In parallel with the development of the site investigation programme, development of new measurement technology has been pursued, along with modification or adaptation of existing such technology /8-49, 50/. The development work that has been done in the past few years is described in the following.

The experience and evaluation of investigation methods obtained and done within the Äspö Project are worthy of special note /8-45/. The fundamental conclusion is that SKB has access to well-proven technology and know-how, which is reassuring for the coming site investigations. Further development is still being conducted in some areas, however.

An important area where progress has been made is seismic reflection. The usefulness of this method for investigating the crystalline bedrock down to repository depth was previously unclear. At the same time, there has been a need for a method that is capable of identifying horizontal fracture zones and rock type contacts, which has warranted further development of the method. SKB has re-evaluated data previously obtained at Finnsjön using improved processing methods with good results, and has devised strategies for how the method could be optimized for the intended purpose /8-51/. However, the method should be tested on a full scale before being employed in site investigations.

The capability to orient fractures in boreholes has taken a large step forward with the acquisition of a new borehole TV logging system. The system permits the long-sought-after presentation of high-quality flattened-out pictures of the borehole wall. It can be used down to a depth of 1500 m in 56 mm or larger boreholes. Image quality is very good, and the software is powerful for processing and calculation of fracture orientations, see Figure 8-7.

As previously reported, drilling methods have been developed to reduce the disturbance caused by the drilling fluid, especially on the chemistry of the groundwater. Difficulties related to drilling caused by heavily fractured (and thereby unstable) formations with fulfilment of the requirements on minimal chemical disturbance is an area in which work to find different solutions is continuing.

A method for length calibration in borehole measurements is under development. It is based on drilling calibration rings into the borehole wall in connection with drilling. In subsequent borehole measurements, depth can then be calibrated against these calibration rings with the aid of a sensor installed in the measuring probe. The method has been tested with a prototype down to a depth of 400 m.

Regarding the borehole radar method, development of yet another antenna type with higher frequency is under way for the purpose of improving resolution, at the expense of a smaller range. The higher frequency should also lead to better measurement conditions in an environment of low resistivity.

Experience and lessons concerning measurements at great depth with different methods have been obtained and more will be obtained in SKB's 1700 m deep hole KLX 02 at Laxemar, see section 5.5.7. Regarding drilling activities, it can be mentioned that the use of the wire line technique was of central importance for efficient drilling of a deep hole such as this one. Experience from measurements of different parameters during drilling and during the subsequent investigations in the borehole will be utilized in the planning of deep boreholes in the site investigations. It can be mentioned that the highly saline groundwater from 1100 m and down (8%) has necessitated changes in the methodology for execution and evaluation of hydraulic tests.

Methodology for measurement of groundwater head and groundwater levels has been developed within the Äspö Project. The methods used have both advantages and disadvantages, and a thorough evaluation of the experience from Äspö will be carried out before a programme is set up for site investigations.

8.2.5 Data management and data processing

For many years, SKB has collected geological site data in a database named GEOTAB. In response to changed requirements and new technical methods, a new database has been developed called SICADA, where fundamental requirements include both efficient input and output of data and high quality and traceability.

Making models of how a rock volume is structured is a complex and laborious process. The modelling work entails iterative testing of various possibilities for correlation between data from different boreholes and from the ground surface. A computer-based visualization tool is now being developed to streamline this work. The tool is based on a 3-D CAD program. With this tool it will be possible to enter and study new boreholes and repository layouts during the different phases of the site investigations. The usefulness of the visualization tool for presentation of the structure of the rock volume is also important. See also section 12.4.3.

The development of the visualization tool will go hand in hand with the development of the aforementioned database so that effective links are established between them. There will also be a link to SKB's GIS database, where surface information on different scales will be available, so that these information sets can be co-processed.

8.2.6 Quality procedures and quality programme

The site investigations must conform to high standards of quality. Simplified, this means "doing the right things" and doing them "in the right way". Doing the right things is mainly ensured by establishing different programmes, so that needs and goals can be scrutinized and the results checked against the goals or given criteria. Doing things in the right way is mainly a question of devising procedures and making sure they are followed in the execution of different activities. Furthermore, the requirement on traceability is important. This is fulfilled mainly by following established procedures. The aforementioned geodatabase has a key role in this context.

Administrative procedures for checking and approving programmes, instructions, data results etc. are in the process of being developed. The quality assurance philo-

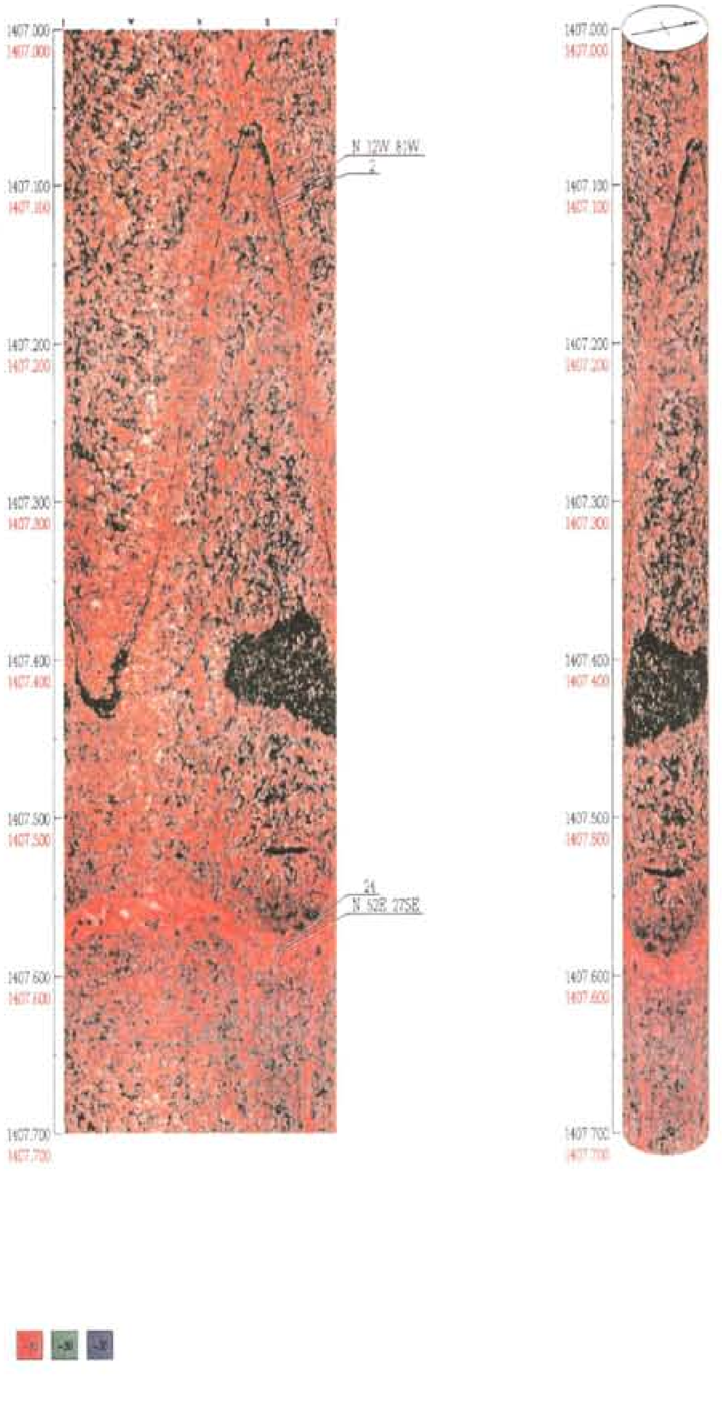


Figure 8-7. Picture of the borehole wall at a depth of about 1400 m, obtained from SKB's new borehole TV system.

sophy is that the person who performs a task is the one best suited to check and verify that it is carried out in accordance with established procedures. The work requires thorough documentation. At present, no separate quality-controlling organization is planned. Quality audits will be performed on a random basis.

8.3 COMPLETED SITING STUDIES

SKB intends to gather supporting data for the siting of Sweden's deep repository with the aid of feasibility studies in five to ten municipalities and site investigations of specific sites in two municipalities. Furthermore, general studies are being carried out to obtain a general background on the fundamental conditions all over the country or in parts thereof. A summary account of the general studies has been published from this material /8-52/, as has a feasibility study /8-30/.

Key questions in the siting of a deep repository are:

- what requirements are made on a deep repository site?
- what is a suitable procedure (siting process) for selecting a site?
- what are the prospects for a deep repository site in different parts of Sweden?

8.3.1 Brief history

Studies for the purpose of accumulating comprehensive knowledge on the Swedish crystalline basement rock and the conditions that could influence the performance of a deep repository were started 20 years ago, in 1975, by the AKA Committee /8-53/. Since then, extensive work has been done studying processes and conditions that could influence the performance of a deep repository. Examples of the subjects of such studies are: seismotectonic conditions, post-glacial faults, analyses of the suitability of certain rock types, geohydrochemical conditions, the influence of ice ages, and rock stresses.

The results have been published by SKB and other collaborating organizations as they have become available. A compilation and evaluation is provided in the national general study /8-52/.

Much of the knowledge we have today concerning conditions at a depth of 500 m in the bedrock comes from the study site investigations which were conducted during the years 1977–1985. During this period, 85 cored holes were drilled with a total length of more than 45 km. The boreholes were mapped with the aid of different types of measurement methods. Special care was devoted to determining the hydraulic conductivity of the rock and the chemical composition of deep groundwaters.

Besides the study sites, major research efforts have also been conducted at Stripa and Äspö, as mentioned previously. Altogether, bedrock conditions have been studied in deep boreholes at 12 sites scattered throughout the country. Reports of the results of these investigations have been published in a number of contexts, most recently in RD&D-Programme 92 – Siting of a deep repository /8-54/.

An important observation is that suitable, or less suitable, areas are not confined to a special part of the country or a special geological setting. Rather, it is the local conditions in the area and in the surrounding region that determine the suitability of an area.

SKB's conclusion, which is described in RD&D-Programme 92, Supplement /8-31/, is that "considerable potential exists for finding repository areas with excellent conditions for isolating the radioactive material. It is therefore reasonable and realistic to start with municipalities who actively wish to participate or otherwise show an interest, and which are situated in those parts of Sweden that may have good prospects for siting a deep repository, and further investigate the prospects for siting a deep repository there".

In the supplement to RD&D-programme 92, SKB presented a detailed account of the siting process and the siting criteria which serve as a basis for site selection. The Government states in its decision from 18 May 1995 /8-55/ that "the siting factors and criteria stipulated by SKB should, in the Government's opinion, be a point of departure for the continued siting work".

8.3.2 General studies on a national scale

In the aforementioned Government decision concerning SKB's programme, the Government states the following regarding the siting work:

"The Government finds, in agreement with most of the reviewing bodies, that SKB ought to present its general studies and site-specific feasibility studies collectively for the purpose of providing background and premises in the siting work. The Government believes that such collective accounts ought to be presented in future research and development programmes in accordance with Section 12 of the Act on Nuclear Activities."

General Siting Study 95 /8-52/ comprises SKB's collective account of general studies on a national scale in accordance with the Government decision. The study is mainly based on the comprehensive background material which SKB has accumulated as a part of the research and development work that has been pursued since the end of the 1970s.

In General Siting Study 95, a number of national databases are presented and evaluated which may be of importance for the siting of the deep repository. The General Siting Study concludes that scientific, technical and societal factors cannot be reported on a national scale with the degree of detail that is necessary for the siting of the deep repository. Furthermore, the information that is reported generally pertains to conditions on the ground surface and not at the depths that are considered for the deep repository, 400–700 m below the surface. For this reason, the suitability of an area cannot be assessed until feasibility studies, site investigations and detailed characterization have been completed. The general study can primarily identify conditions in different parts of the country that may be unfavourable for a deep repository.

For the siting factor **Long-term radiological safety,** the following databases have been evaluated: rock types, topography, well data and the so called highest coastline. Furthermore, geological deformation zones, lineaments, future ice ages and earthquakes have been evaluated, along with the possibility of unintentional human intrusion and recipient conditions. Unnecessary consumption or blockage of natural resources should also be avoided, if possible. Areas situated in unusual rock types, or where there is a potential for ore, are thereby of less interest. By avoiding these areas, the probability of future unintentional intrusion in the deep repository is also reduced.

For geological reasons it is deemed unsuitable to site the deep repository in the Scandinavian mountain range, Skåne or Gotland. Moreover, the Swedish mountains are of national interest for reasons of nature conservation and outdoor recreation. Siting in the crystalline basement beneath Öland is deemed to be technically possible, but unsuitable in view of the general resource management provisions of the Act Concerning the Management of Natural Resources.

The siting factor **Technology** sheds light on how practical feasibility can be affected by different conditions. When it comes to transportation to the deep repository, it is found that access to harbours and railway lines is good throughout Sweden and that this factor therefore does not constitute a limitation on the siting of the deep repository viewed on a national scale. The execution of rock investigations, plant design and safety assessment is facilitated by the fact that the geoscientific characteristics of the site are easy to understand and interpret. A number of attempts at assessing possible regional differences in interpretability on a national scale are described. Even though it is advisable to first seek out sites that are easy to interpret, similar reliability can also be obtained in the results on more complex sites, but then at the price of more extensive investigations. This is thus a question of optimization, among many others, in the siting process.

The repository should also be sited with respect to the siting factor **Land and environment**, in observance of the Act Concerning the Management of Natural Resources. Siting in areas that are expressly protected by law is neither necessary nor desirable and should be avoided. Areas that are of national interest in other contexts can, however, not be categorically ruled out, since the deep repository can in many cases be designed in such a manner that the purpose of the national interest is not adversely affected.

As far as the siting factor **Society** is concerned, a large number of conditions are dealt with in the report. It is concluded that these conditions should be evaluated on a more detailed scale, i.e. during feasibility studies, site investigations and detailed characterization.

General Siting Study 95 does not change SKB's earlier conclusion that there are many areas in Sweden that could be suitable for a deep repository. Siting should continue to be concentrated on bedrock that occurs widely in Sweden, preferably granitic rock types, or older metamorphic sedimentary bedrock. This type of "interesting" bedrock can be found in large parts of the country.

This does not mean that areas with gabbro, or areas where the basement rock is overlain by palaeozoic sedimentary rock, need necessarily be ruled out in the siting process. These areas will be specially assessed if it turns out that feasibility studies are of interest in municipalities with this type of bedrock. Furthermore, it is technically feasible to site the repository under large lakes or beneath the sea. There is, however, no particular reason to either seek or avoid such a siting.

Besides the already concluded, ongoing or planned feasibility studies, one or more additional feasibility studies are deemed desirable. It should in this context be an advantage in the continued work and in discussions with different municipalities that this collective presentation of the general siting studies now exists. It should provide a better opportunity than before for all parties concerned to become acquainted with the background and general prospects for the siting work at different places in the country.

8.3.3 Feasibility studies

General

A feasibility study examines the prospects for a deep repository within a municipality. The studies are primarily based on existing material.

The following questions are dealt with:

- What are the general prospects for siting a deep repository in the municipality?
- Where might suitable sites for a deep repository be located in view of geoscientific and societal aspects?
- How can transportation be arranged?

- What are the important environmental and safety issues?
- What are the possible consequences (positive and negative) for the environment, the local economy, tourism and other enterprise within the municipality and the region?

SKB requires no formal permits to carry out a feasibility study. However, in practice the feasibility studies are conducted in consensus between SKB and the municipality in question.

A feasibility study should provide a broad body of facts for decisions by both the municipality and SKB. Both parties can then decide for themselves whether they are interested in starting a site investigation. The same facts are made available to all stakeholders, who thereby have an opportunity to present viewpoints long before any decisions need to be taken on the siting of the deep repository.

The purpose of the feasibility study is thus to investigate whether prospects exist for locating a deep repository in the municipality, and to provide a factual basis for decisions on whether to continue investigations. Questions regarding the principles of disposal, advantages and disadvantages of the chosen concept, and methods for evaluating long-term safety are dealt with in other contexts and are not examined in the feasibility study. These matters are, however, taken up in the dialogue and the discussions that are held with all concerned parties in connection with a feasibility study.

Current situation

SKB has conducted a feasibility study in the municipality of Storuman. The final report was published in February 1995 /8-30/. The feasibility study shows that good prospects may exist for a deep repository in the municipality. The municipality arranged a (local) referendum on 17 September 1995 regarding the question: "Should SKB be allowed to continue searching for a final disposal site in Storuman Municipality?" The result was 28% Yes, 71% No and 1% Blank votes. Voter turnout was about 73%. This means that SKB will cease its work in Storuman.

Another feasibility study has been under way in Malå since the winter of 1994. All investigations have been published as they have been completed and a summarizing status report was published in May 1995 /8-56/. The purpose of the status report is to give the municipality, its reference group and other interested groups in Malå and in the region a basis for discussion and viewpoints before the final report is written. The final report is expected to be finished in the autumn of 1995.

During the late spring of 1995, discussions were held with Tranemo Municipality concerning a feasibility study. The municipal executive committee has, however, decided not to take up the matter for discussion.

Discussions concerning a feasibility study have also been held during the past 3-year period with the municipalities of Arjeplog and Överkalix. Both have, however, decided not to pursue the matter further.

A study /8-57/ has examined the prospects for feasibility studies in municipalities with nuclear activities. For Oskarshamn, Nyköping and Östhammar, the existing body of geological data is considerable and suggests a possibility of good siting potential. For these municipalities, SKB believes it is of primary interest that feasibility studies are conducted so that the background material for site selection has the necessary breadth.

Some uncertainty concerning the suitability of the bedrock exists for the municipality of Varberg. Among other things, modern geological map material is lacking for parts of the municipality. Supplementary geological mapping is therefore required at an early stage in order to obtain material equivalent to that available for the other nuclear municipalities. Nevertheless, SKB deems it desirable that a feasibility study is also made of the Varberg municipality.

As far as the municipality of Kävlinge is concerned, both geological and technical conditions show that a siting of the deep repository in the municipality would be complicated. SKB therefore deems it not to be of interest to carry out a feasibility study in Kävlinge Municipality.

The geographic location of municipalities where feasibility studies have been concluded, are in progress or are being discussed, or where discussions have been discontinued, is shown in Figure 8-8.

Results

The completed and ongoing feasibility studies show that a municipality-sized area can often be a suitable point of departure for a survey of a region's prospects for siting of a deep repository. Comprehensive yet relatively detailed geological maps are often available on this scale, as are geophysical and other geoscientific maps. Utilizing the administrative unit of a municipality facilitates surveys of socioeconomic consequences, as well as assessments of municipal comprehensive plans and other sources of information on current or planned land use.

A feasibility study is done in consultation or collaboration with a municipality. This means that the municipality has an opportunity to call for surveys within the framework of the feasibility study. In Storuman, the municipality or its reference group have called for several surveys that shed light on the consequences of a deep repository for community development, the psychosocial effects and impact on outdoor recreation and the environment. These questions are important on the local plane, but tend to be forgotten in a larger context. The feasibility studies have therefore generally provided a forum for highlighting and exploring these aspects.

The general conclusion from the feasibility studies in Storuman is that good technical prospects exist for siting,

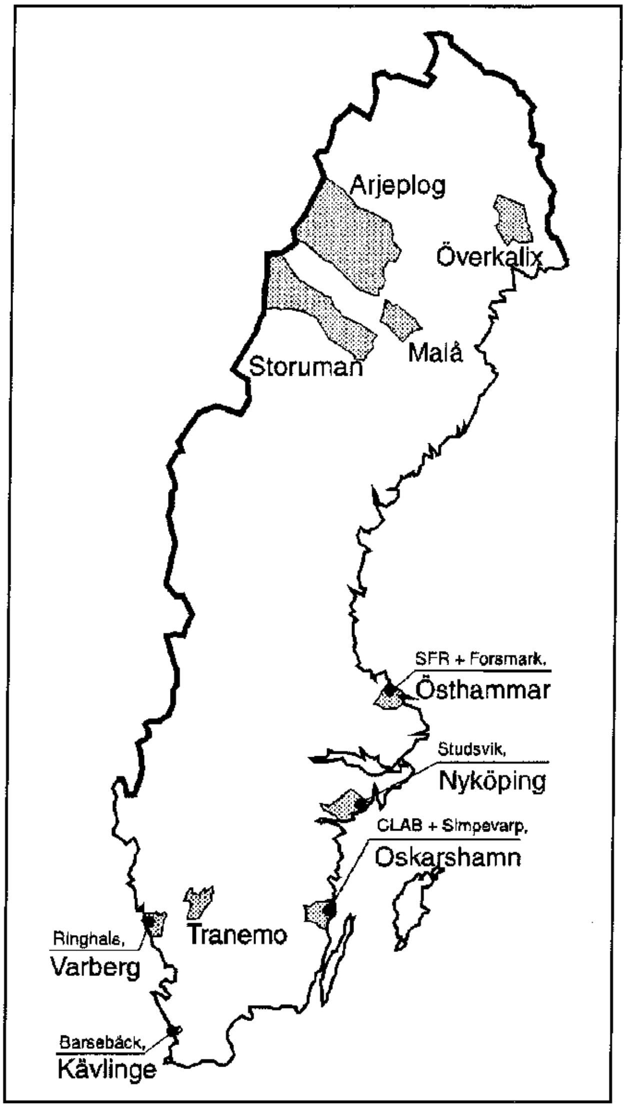


Figure 8-8. *Municipalities where feasibility studies have been concluded (Storuman, Malå), started (Nyköping, Östhammar) or are being considered (Oskarshamn, Varberg). Discussions concerning feasibility studies have been held in Arjeplog, Överkalix, Tranemo and Kävlinge, but these municipalities are no longer candidates.*

building and operating a deep repository. It is not possible to draw any conclusion regarding long-term safety since such an assessment requires knowledge concerning conditions in the bedrock at greater depth. Based on conditions at the ground surface, the feasibility study has identified two areas in the eastern part of the municipality that may have good potential for a deep repository. Both areas consist of homogeneous, fracture-poor granite. The areas are not encumbered by any obstacles to siting with respect to land use and environment, and they are situated in the vicinity of existing road and rail links that can be used for shipments to the deep repository.

8.4 SHIPMENTS TO THE DEEP REPOSITORY

In connection with the feasibility studies of the inland municipalities of Storuman and Malå, the transport aspects have been considered and studied. These studies have included cargo types and quantities, mode of transport, transport routes and safety aspects. Since these matters have only been dealt with superficially in previous RD&D programmes, and as a consequence of the knowledge that has been accumulated in conjunction with the aforementioned feasibility studies, the transport aspects are discussed more thoroughly in the following.

8.4.1 Cargo types and quantities

The transportation system must be able to handle two different types of cargoes: heavy units containing encapsulated fuel and bulk cargoes in the form of bentonite and possibly sand. A survey of the cargo quantities the transportation system will have to handle and which transport modes may be suitable has been done in the feasibility studies for Storuman and Malå /8-59, 59, 60, 61/. The results of these studies are presented below.

The heavy units are specially designed transport casks for encapsulated fuel and similar containers for concrete moulds containing other waste. It is assumed that the shipments of encapsulated material will go from the harbour at Simpevarp. The transport steps from the encapsulation plant to the deep repository are shown in Figure 8-9. The total number of units shipped to the repository is estimated at about 300 per year. The containers are shipped in return for refilling.

The largest cargo quantities to be transported to the deep repository will, however, consist of about 15,000 tonnes of bentonite per year. Depending on the choice of backfill material, 45,000 tonnes of sand may also have to be shipped to the deep repository. The figures are preliminary. The need for buffer and backfill materials is determined by the design of the underground facility and the development of the backfilling method, including changes in the mixing ratio between bentonite and ballast.

8.4.2 Mode of transport for radioactive materials

Transport from CLAB takes place by ship to a suitable harbour for further overland transport to the deep repository.

Transport from the harbour to the deep repository takes place by rail or road. The feasibility studies for siting of the deep repository include an inventory of suitable

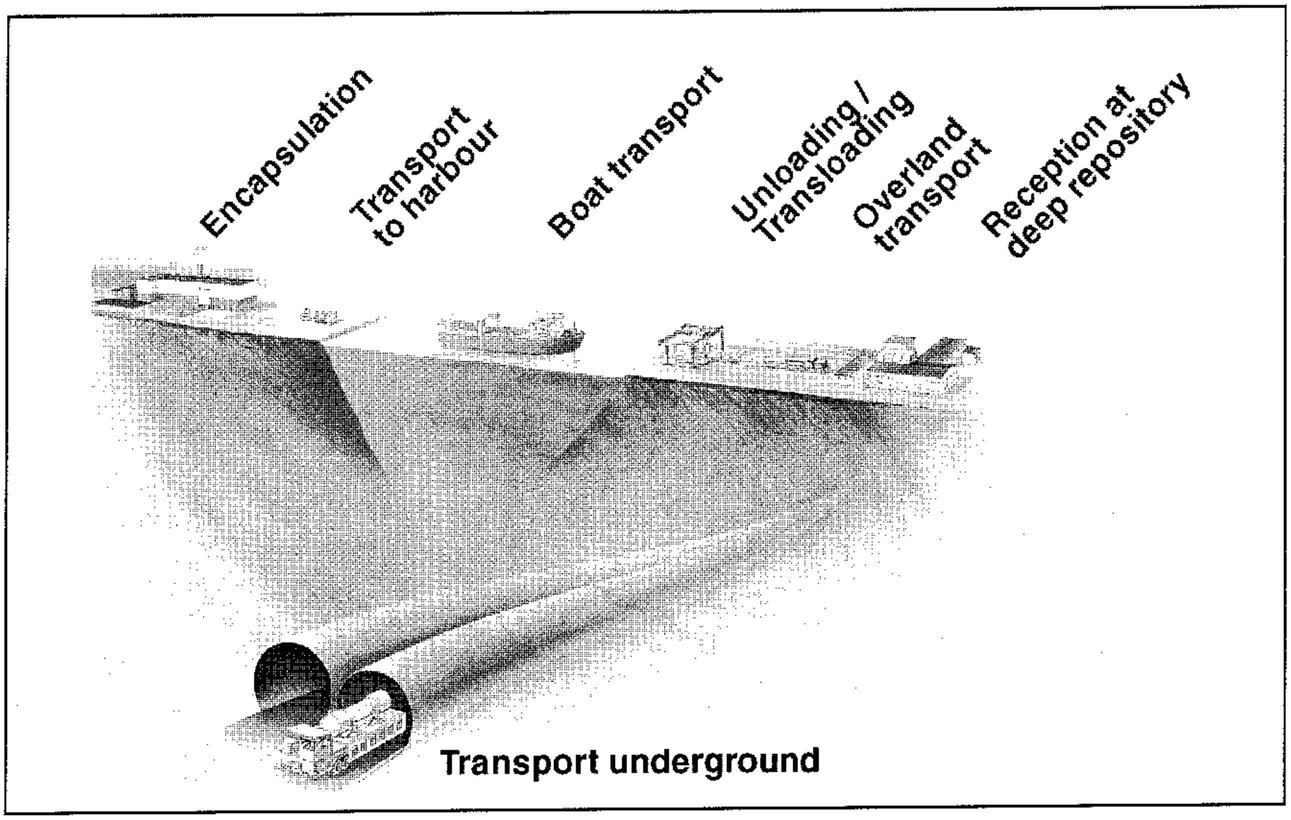


Figure 8-9. Transport steps from the encapsulation plant to a deep repository located in the interior of Sweden.

harbours linked by road or rail to the potential repository site.

The fundamental technical difference between rail and road transport is that a railway is built for higher axle loads and that the load can be distributed over a larger ground area.

To avoid transloadings it is an advantage if the same transport mode can be used over the entire route. Depending mainly on local conditions, it may nevertheless prove advantageous or less costly to change transport mode for the final transport link.

8.4.3 Transport safety

Nuclear fuel, both new and used, consists of a solid ceramic material encased in metal tubes or "cans" of a zirconium alloy. The fuel assemblies will be enclosed ("encapsulated") in canisters that are hermetically sealed. The risk that radioactive materials will leak out during handling or transport is virtually non-existent. On the other hand, some radiation does penetrate the canister walls, which is why the fuel canisters must be transported in radiation-shielding casks. At the same time, the transport casks provide strong mechanical protection of the canisters during transport. A transport cask with canister is estimated to weight 50–70 tonnes.

Most of the "other waste" is expected to be embedded in concrete moulds. This type of waste also requires some radiation shielding and is therefore transported in steel transport containers, very similar to those used in SFR.

The transport casks/containers shield off the radiation to such a low level that they can be handled without any special protection during loading and unloading of ships, vehicles and trains. Since the casks/containers have to be able to withstand severe external stresses, the transportation system does not have to be designed to give additional mechanical protection to the cargo. On the other hand, the casks and containers with contents are classified as dangerous goods according to international regulations and must be marked, separated and kept under surveillance in accordance with the regulations that apply to radioactive goods.

The transport casks and containers are designed in compliance with the requirements imposed by the IAEA. They have to protect the enclosed canister against damage while shielding off the radiation that is emitted by it so that the casks can be handled safely during loading and unloading. The strength of the casks/containers is then such that they are able to withstand forces in excess of those that could occur in connection with conceivable accidents, such as collisions and falls, without breaking. What is important in an accident situation is that the radiation-shielding capacity of the cask/container is largely retained, i.e. that the heavy steel body remains intact around the enclosed waste.

Safety during transport between the encapsulation plant and the deep repository is thus focused primarily on the following:

- the risk that accidents and incidents will occur during transport must be minimized,
- if an accident of some kind nevertheless does occur, it must not cause release of radioactive material,
- the radiation levels on the outside of the transport casks and containers must be below the applicable limit values so that they can be handled without risk to the personnel.

If these conditions are fulfilled, the shipments will not pose any danger to persons in the vicinity of the repository or along the transport routes that are used.

Part of the safety is the physical protection, which is aimed at preventing theft or sabotage of the casks/containers. It consists of a combination of physical and administrative measures which protect the cargo and provide for detection and alarm if anything abnormal occurs. The measures include guarding, communication with a transport command centre and the like.

8.4.4 Experience of today's shipments

SKB's present-day system for sea transport has been in use since 1982. Some 80–100 casks with spent fuel have been transported annually to CLAB since 1985, and roughly an equal number of transport containers with radioactive operational waste have been transported annually from the nuclear power plants to SFR in Forsmark. Experience from these shipments is good. No disturbances or accidents of safety-related importance have occurred. The radiation dose to the handling personnel has been kept at a low level. Thus, for example, the crew of the ship M/S Sigyn, the ship used for this work, has not received any radiation doses in excess of the normal background radiation to which everyone is exposed.

Other countries, such as the UK, France, Germany, Japan and the USA, have many years of experience of rail, road and sea shipments, with just as good experience as far as accident statistics are concerned as Sweden.

9 PROGRAMME FOR DEEP REPOSITORY

The main lines in the programme for siting of the deep repository remain the same from RD&D-Programme 92 and its supplement /9-1, 2/. A modification of the time schedule to adjust for developments during the past three years is necessary, however. During the period 1996–2001, the main task will be to prepare supporting documentation for an application to site the deep repository at a specific site. The most important immediate goal is to start site investigations.

9.1 GOVERNMENT DECISION REGARDING THE SITING PROCESS

The Government decision from May 1995 /9-3/ regarding the supplement to RD&D-programme 92 has entailed a number of clarifications of importance to the siting work:

- The siting factors and criteria stipulated by SKB in the supplement to RD&D-Programme 92 "should, in the opinion of the Government, be a point of departure for the continued work".

  This means, among other things, that the criteria for safety, technology, land and environment, and society which SKB has identified can serve as a basis for continued siting studies and for further definition of acceptance criteria in site investigations. The Government's decision also means that SKB's endeavour to comply with high environmental and safety standards while seeking local collaboration and understanding has the Government's support.

- Municipalities in which SKB conducts feasibility studies can obtain up to SEK 2 million per year to "follow and assess, and furnish information in matters relating to final disposal of spent nuclear fuel and nuclear waste". This means that concerned municipalities are now directly assured of their own resources for participation in the siting work. In the first feasibility studies, the costs of the municipalities have been covered by SKB.

- The Government decision states that "Applications for permits pursuant to Chapter 4 of the Act Concerning the Management of Natural Resources and Section 5 of the Act on Nuclear Activities to construct a final repository for spent nuclear fuel and nuclear waste should contain material for comparative assessments that shows that site-specific feasibility studies in accordance with SKB's account have been conducted at between 5–10 sites in the country and that site investigations have been conducted at least two sites, plus the reasons for the choice of these sites. The Government finds, judging from SKB's account, that the planned site for detailed characterization comprises a first step in the construction of a final repository for spent nuclear fuel and nuclear waste."

  As far as SKB can judge, this means that an application for a siting permit for the deep repository must be submitted for consideration under both the Act Concerning the Management of Natural Resources (NRL) and the Act on Nuclear Facilities (KTL) when SKB wishes to commence detailed characterization of a selected site. This licensing procedure will thereby be the first major decision-making occasion as far as the design and siting of the deep repository system is concerned. (In RD&D-Programme 92, such a comprehensive licensing review was not planned to take place until after detailed characterization.) If a permit is obtained, the next major decision-making occasion should be issuance of a permit for commissioning of the deep repository, i.e. before the first waste canister is deposited.

  In their scope and focus, detailed characterization and construction of the deep repository are heavily dependent on the local conditions that will prevail on the site in question. These conditions are not fully revealed until a site investigation is undertaken. It should therefore be possible to describe in the application for a siting permit the different phases in the work of detailed characterization and construction of the plant. A permit based on such an application can then contain provisions setting forth specific conditions for the progress of the work from one phase to the next. The permit can also regulate in which order results and data from the different steps of the detailed characterization are to be reported to the Nuclear Power Inspectorate and other concerned authorities. This means that even though detailed characterization and construction now comprise a single main stage, the work progresses in steps with scrutiny on the part of the safety authorities after each step.

- The Government assumes that the county administrative boards in the counties affected by feasibility studies, site investigations or detailed characterization will assume "a coordinating responsibility for the contacts with municipalities and state authorities that are needed for SKB to prepare supporting documen-

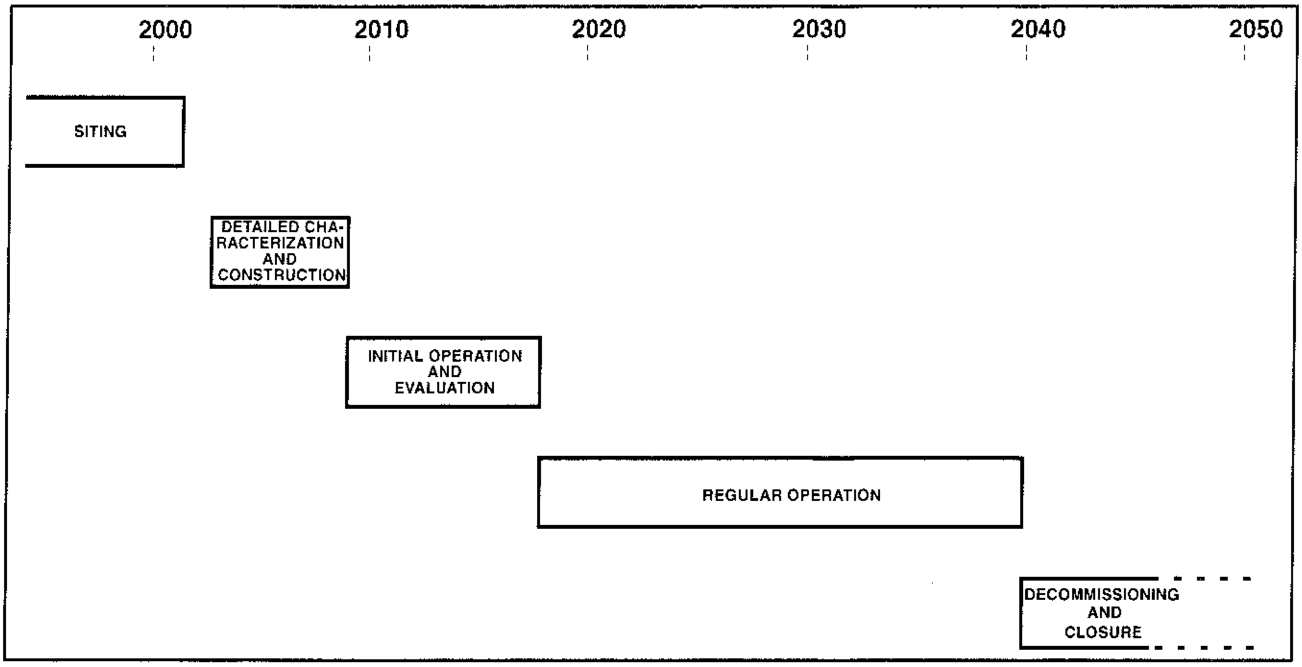


Figure 9-1. General time schedule for the main stages in the deep repository project. A safety report and a new permit are required before each new stage.

tation for an EIA for review pursuant to Chapter 4 of the Act Concerning the Management of Natural Resources".

This means that the coordinating responsibility for the EIA procedure is placed on a regional authority, which should contribute to satisfactory consideration of any viewpoints from local (municipal) and regional (neighbouring and transit municipalities) concerned parties.

9.2 STAGES OF THE DEEP REPOSITORY PROGRAMME

The Government decision places the main weight in the authorities' review of the siting of the deep repository on the point when SKB applies for permission to start detailed characterization. This lends added importance to the site investigation phase in relation to what was assumed in RD&D-Programme 92. More resources and time must therefore be allocated to this phase, which is reflected in the revised time schedule below. On the other hand, it should be possible to reduce the scope of the regulatory review following completed detailed characterization. Despite the delays in the schedule in the initial stage, it should be possible to stick to 2008 as the earliest date for the start of deposition.

The Government decision also entails that there is reason to revise the subdivision into stages that was presented in RD&D-Programme 92. The new subdivision is coupled to regulatory reviews of the siting and operation of the deep repository. In this process there are five well-defined stages after which safety accounting and a permit are required before the next stage can begin. The different stages are:

Stage 1. Siting.
Stage 2. Detailed characterization and construction.
Stage 3. Initial operation and evaluation.
Stage 4. Regular operation.
Stage 5. Shutdown and closure.

A general time schedule for the aforementioned stages is presented in Figure 9-1.

How fast the siting process progresses is dependent on technical, societal and political factors, of which the latter are difficult to put a time to. In general it can be said that the greatest uncertainties are in the initial stage. This is particularly true of the time for the start of site investigations, where delays due to societal or political factors can arise. A discussion of the uncertainty inherent in the schedule can be found in chapter 15.

To move forward in the siting process it is therefore urgent to focus early during the next six-year period on activities on specific sites. Then the design and investigation activities change character from being general

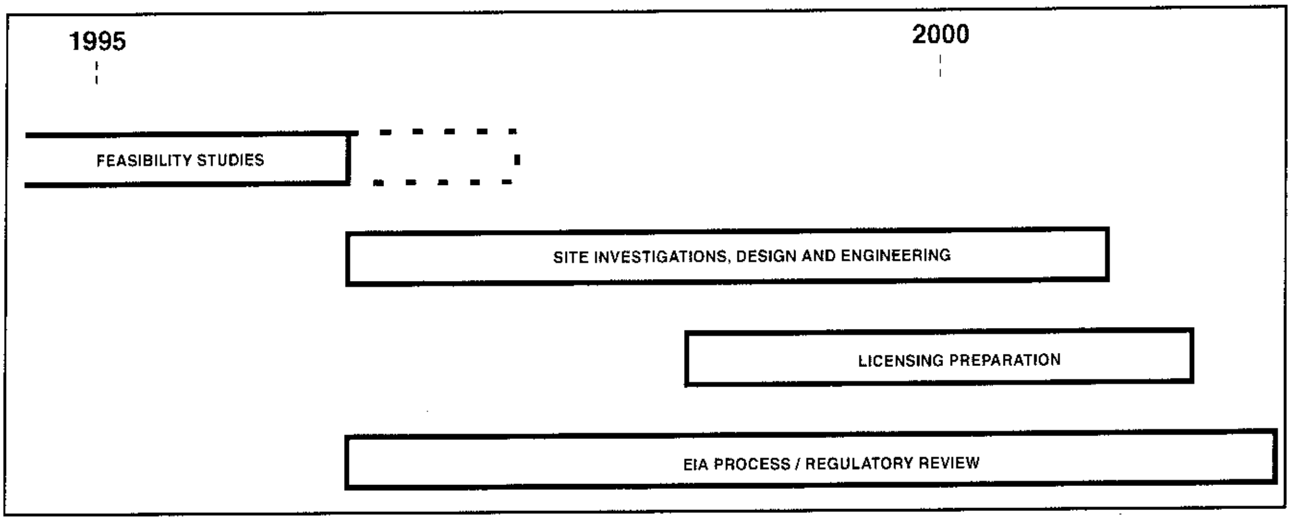


Figure 9-2. *Schedule for phase 1, siting.*

programmes and plans to concrete site-specific tasks. The main activities within each stage are as follows:

Stage 1

Siting

The stage consists in collection of the background material that is required to select a site for the deep repository. The background material consists of general siting studies /9-4/ covering the whole country, feasibility studies in 5–10 municipalities /9-5/ and site investigations in at least two municipalities. In parallel, work is pursued with plant design and engineering, performance and safety assessments, environmental impact assessment and EIA consultation. The stage is concluded with a compilation of the necessary supporting documents to accompany an application for a siting permit under the Act Concerning the Management of Natural Resources (NRL) and an application for a permit to construct the deep repository under the Act on Nuclear Activities (KTL). A permit under the Water Act and the Planning and Building Act will also be required, as well as other municipal permits.

A more detailed account of the activities in Stage 1 is given later in this chapter.

A schedule for the siting stage is shown in Figure 9-2. Site investigations are estimated to take 4–5 years to carry out, based on the outline of the investigations presented in section 9.4.

Stage 2

Detailed characterization and construction

This stage involves design and construction of surface and underground facilities and general investigations of all repository sections. The section that will be used for initial operation (operation stage 1) is investigated in greater detail. The data are obtained mainly from tunnels down to repository depth and from boreholes and measurements from these tunnels. Connecting roads and railway are also built. Recurrent scrutiny is foreseen on the part of the Swedish Nuclear Power Inspectorate and the National Radiation Protection Institute in particular, based on the requirements stipulated in the siting permit for different steps in the work of constructing the deep repository. Stage 2 is concluded with an application for a permit (KTL) for initial operation.

Stage 3

Initial operation and evaluation

The stage consists of deposition of about 10% or 400 of the total of about 4,500 canisters containing spent nuclear fuel. During both initial and regular operation, deposition takes place in repository tunnels at the same time as new tunnels are excavated. Excavation of the repository will therefore continue throughout both the initial and regular operating periods.

Experience from initial operation is evaluated over a period of several years.

At the same time, detailed characterization of repository areas for regular operation is carried out (area for deposition of spent fuel and area for deposition of other waste). Provided that the evaluation of the initial operation turns out favourably and it is decided to proceed, and provided that the detailed characterization confirms the existence of suitable bedrock conditions, an application is made for a permit (KTL) for regular operation.

Stage 4

Regular operation

During regular operation (operation stage 2), approximately 4,000 canisters of spent nuclear fuel are deposited. Other radioactive waste will also be deposited during this stage in a separate section of the deep repository. The stage is concluded with an application for a permit (KTL) for closure of the facility.

Stage 5

Shutdown and closure

The surface facilities are dismantled and demolished and the underground facilities are backfilled and plugged. Before this is done, it has been possible to observe the initially deposited canisters for several decades. The scope of the future monitoring of the repository and/or the disposal site is decided by each generation for itself.

9.3 PROGRAMME FOR SITING STUDIES

The outline of the siting programme was recently presented in the supplement to RD&D-programme 92 /9-2/. The Government decision /9-3/ states that "Applications for permits pursuant to Chapter 4 of the Act Concerning the Management of Natural Resources and Section 5 of the Act on Nuclear Activities to construct a final repository for spent nuclear fuel and nuclear waste should contain material for comparative assessments that shows that site-specific feasibility studies in accordance with SKB's account have been conducted at between 5–10 sites in the country and that site investigations have been conducted at at least two sites, including the reasons for the choice of these sites". As is evident from the account in chapter 8, SKB has commenced the work of gathering the background material that will be required. The goal for the coming six-year period (1996–2001) is to collect all this material to be submitted in support of an application for a permit to site and construct a deep repository on a selected site. The most important immediate goal is to carry out feasibility studies and commence site investigations.

A collective account of general siting studies and feasibility studies has been requested by the Government in the forthcoming research and development programme.

As evident from chapter 8, a collective account of nationwide general siting studies is being presented in conjunction with this research programme /9-4/. The purpose is to provide background and premises for the siting work and summarize the general siting studies which SKB has carried out over the years. This does not mean that such studies are finished once and for all. There may be a need for regional studies in conjunction with future feasibility studies in order to shed light on e.g. geological conditions, land use, transport options or socioeconomic conditions in a particular region.

9.3.1 Feasibility studies

SKB's siting programme includes between 5–10 feasibility studies.

The feasibility study of Storuman is finished and the feasibility study of Malå is in its final phase. Feasibility studies are now being started in Nyköping and Östhammar and are being discussed in the municipalities of Oskarshamn and Varberg. Feasibility studies may also be considered for several other municipalities.

The scope of future feasibility studies will be based on experience from the feasibility studies in Storuman and Malå. Some reduction in the scope of the investigations can be foreseen, however, since much of the general material that has been obtained in the Storuman and Malå studies can also be utilized in future studies.

The programme for future feasibility studies, which is based on the recently completed feasibility studies, will be tailored to the specific conditions existing in each municipality. Investigations and fact-finding surveys are being conducted within the following areas:

- bedrock,
- land and environment,
- transportation,
- societal impact.

For the most part, existing data and available information are compiled, e.g. from economic and geological maps, geoscientific databases, municipal comprehensive plans, etc.

Bedrock

- compilation of previously made geoscientific studies,
- existing rock types and their properties,
- lineaments and fracture zones,
- stability of the bedrock, seismic conditions,
- hydrogeological and hydrochemical conditions,
- ore potential, potential for industrial minerals,
- compilation of rock engineering experience in the region.

The general survey of the municipality's geoscientific prospects is initially focused on identifying areas that may be unfavourable for a deep repository. For areas where no conspicuously negative conditions have been identified more detailed studies are made for the purpose of deciding whether one of them may be suitable for further studies.

Land and environment

Together with the municipality, a survey is made of plans for utilization of land and water. The comprehensive plans are studied, as are economic maps and land ownership. Further, a municipality-specific compilation of different environmental and safety aspects is done.

The work results in the identification of parts of the municipality in which there are essential limitations on the siting of a deep repository from the community planning and environmental points of view.

Transportation

The study describes possible transport modes and transport routes for encapsulated spent nuclear fuel and other waste from the encapsulation plant to the municipality. Possible transport modes and transport routes for backfill materials are also examined.

Societal impact

Under this heading, the possible consequences of a deep repository siting for the economy and community life in the municipality and the region are explored. The orientation and scope of the survey is determined to a large extent by the particular municipality.

9.3.2 Site investigations

As mentioned previously, the Government decision of May 1995 /9-3/ provides that SKB's proposal for the site and layout of the deep repository will be reviewed when SKB applies for permission to conduct detailed characterization involving construction of a tunnel or shaft. The review will encompass both the Act Concerning the Management of Natural Resources and the Act on Nuclear Activities. The fact that the regulatory reviews are concentrated to this occasion means that extensive background material must be compiled during the site investigation phase.

SKB plans to carry out at least two site investigations. Key questions which these investigations will answer for a particular site are:

- What are the rock conditions like?
- What are the prospects for underground construction?
- What are the prospects for long-term safety of a repository?
- What competing land-use interests are there?
- What might the environmental consequences be?
- How can the facility be linked to existing roads and railways?
- Where is it suitable to locate the surface and underground facilities?
- How should these facilities be designed and configured?
- How can transportation be arranged?

To answer these and other questions, a body of data will be gathered for each site by means of the following activities:

Geoscientific investigations, whose purpose is to gather site-specific data in order to indicate preliminarily where and how the deep repository can be situated in the bedrock and for the assessment of long-term safety. The programme for geoscientific investigations is described in greater detail in section 9.4.

Design of the deep repository for the purpose of determining the layout of the deep repository's facilities above and below the ground, adapted to geographical and geological conditions on the investigated site. SKB's plan for the design process, divided into different phases, is presented in the supplement to RD&D-programme 92 /9-2/. Plans for the continued work with the deep repository's design and methods for construction and operation are presented in section 9.5.

Performance and safety assessments, which are supposed to provide the material needed to obtain a permit under the Act on Nuclear Activities to begin detailed investigation and build the deep repository. A programme for them is described in chapter 10.

Surveys of societal issues and land and environment issues which shed further light on the deep repository's possible impact on society, land use and the environment in the concerned municipality and region. The surveys build further on the material gathered in the feasibility study. One purpose is to furnish data for adapting the layout and placement of the deep repository facilities within the investigated area so that environmental impact and other disturbances are minimized.

Environmental impact assessment (EIA) and EIA consultation aimed at gathering comprehensive and clear background information in accordance with the requirements made on environmental impact assessments in the laws relevant to the deep repository. EIA is dealt with in chapter 3.

As has been stated above, SKB expects to be able to begin site investigations within the next few years. Before a site investigation begins, the following documentation will be available:

- A survey showing how SKB, through evaluation against guiding criteria, has arrived at the conclusion that the area may have good prospects as a disposal site and is therefore of interest for further investigation. (**Final report feasibility study** with underlying reports.)

- A compilation which – with the aid of general siting studies, completed feasibility studies and other relevant material – places the area in question in a national context and gives the reasons why SKB deems it appropriate to begin site investigations in the area. (**Report** with explanation why the site has been selected.)

- An agreement between SKB and the municipality on the outline of and forms for the site investigation phase. (**Execution programme** with description of the forms for **EIA consultation**.)

- A programme for geoscientific site investigations specifying the goals, measurement methods and evaluation methodology for the investigations, as well as the acceptance criteria against which the site is evaluated. (**Geoscientific site investigation programme.**)

The execution programme and the site investigation programme must be based on specific local conditions and are therefore necessarily in part different for the various sites being investigated. The site investigation phase must be carried out step-by-step, and as results are obtained re-evaluations are made of whether it is worthwhile continuing the investigations and, if so, the scope of investigations.

9.4 GEOSCIENTIFIC INVESTIGATIONS

A programme for geoscientific site investigations is in preparation. It will be published in good time before site investigations are begun. The current situation is described below. The programme is based mainly on experience from bedrock investigations within the Swedish nuclear waste programme (including Stripa and the Äspö HRL), but also on foreign experience /9-6, 7, 8/.

Besides the site investigation programme, this section also presents plans for refinement of methods and instruments as well as data management and quality assurance.

9.4.1 General

A geoscientific site investigation entails collection of site-specific data for description of bedrock and groundwater conditions and properties. The purpose is to identify a site and to evaluate its suitability.

A site investigation has the following main goals:

- The investigations should provide a geoscientific understanding of the site and its regional environs with respect to present-day situation and natural ongoing processes.

- The investigations should provide the necessary geoscientific data for a site-adapted design of the deep repository and for assessment of the deep repository's long-term performance and radiological safety.

The main strategy is that SKB should carry out two site investigations and that this should be done in two of the municipalities where feasibility studies have been conducted. The investigation programme is centred on prioritized areas that have been identified in these feasibility studies. The site investigations are conducted in parallel, but will probably be staggered about six months apart.

A site investigation results in a consolidated geoscientific description of the site and its regional environs. Data and results from the site investigation are used in performance and safety assessments for calculations and descriptions of the performance of the natural and engineered barriers for a deep repository located on the investigated site. The results are presented in a safety report in support of an application for a permit for detailed characterization.

In the design work, site data is used to establish layouts for the different parts of the deep repository, with adaptation to the geographic and geological conditions of the site and the bedrock. Furthermore, construction analysis is carried out, i.e. calculations and analyses of the properties and limitations of the rock with respect to construction technology and industrial safety, etc.

Site evaluation is a collective term for the interactive process that consists of the parts mentioned above for evaluation of an investigated site with respect to its suitability for a deep repository, see Figure 9-3. Not only safety and technology but also other siting aspects that fall under the category "environment and society" are dealt with in such an evaluation. The environmental impact assessment is supposed to provide an overall picture of the effects of the planned deep repository.

9.4.2 Execution

The scope of a site investigation is dependent on the investigated geographic area, the investigated depth and the contents of the investigations. The geographic terms area, sub-area and site are used in talking about scope, see Figure 9-4. A site investigation has begun when drilling of deep investigation boreholes has been commenced.

A site investigation is carried out in two main stages: initial and complete site investigation. The main purpose of the initial site investigation is to ascertain with relatively limited measures whether the judgements from the feasibility study are correct. The initial studies also aim at identifying where within a stipulated area the potential for a deep repository is greatest and thereby where the continued investigations should be concentrated.

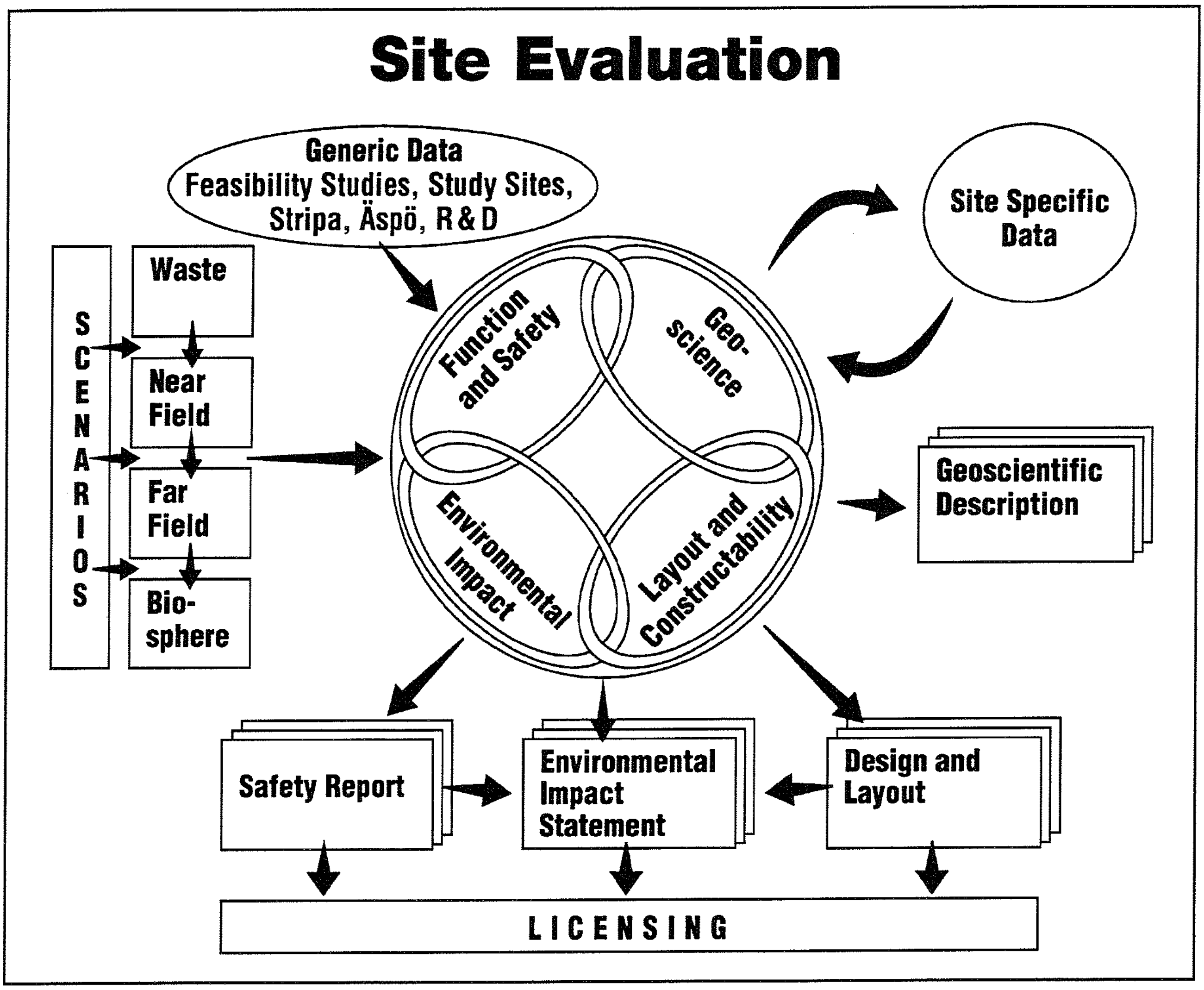


Figure 9-3. Scope of site evaluation.

Initial site investigation

The initial site investigation aims at identifying a suitable site on the order of 5 km^2 within areas which the feasibility study has indicated as suitable for further studies, see Figure 9-5. Supplementary regional studies are conducted during this stage in order to place the study area in its regional geological context. This is a task which can also be performed as a part of or supplement to a feasibility study.

Subsequently a second stage is initiated for the purpose of confirming the choice of site by mapping of fracture zones, rock type boundaries and other geological conditions at depth. Seismic reflection surveys and the first deep exploratory drillings are carried out during this stage with analyses and measurements of groundwater chemistry, hydraulic conductivity and rock stresses as key parameters. The design work is begun and preliminary performance and safety assessment are carried out. Provided that the initial investigations and analyses indicate suitable conditions, the investigation programme proceeds with complete site investigations.

Complete site investigation

The complete site investigation is carried out in two steps, partly because certain investigations require undisturbed conditions and partly to provide an opportunity for stepwise geoscientific evaluation as well as stepwise facility layouts, construction analyses and performance assessments. Execution of investigations is foreseen as follows, see also Figure 9-6:

Step 1 Aims at verifying and supplementing the picture of how major fracture zones and rock types (especially dykes) are distributed by means of a drilling and measurement programme. This is primarily done as a basis for a site-specific

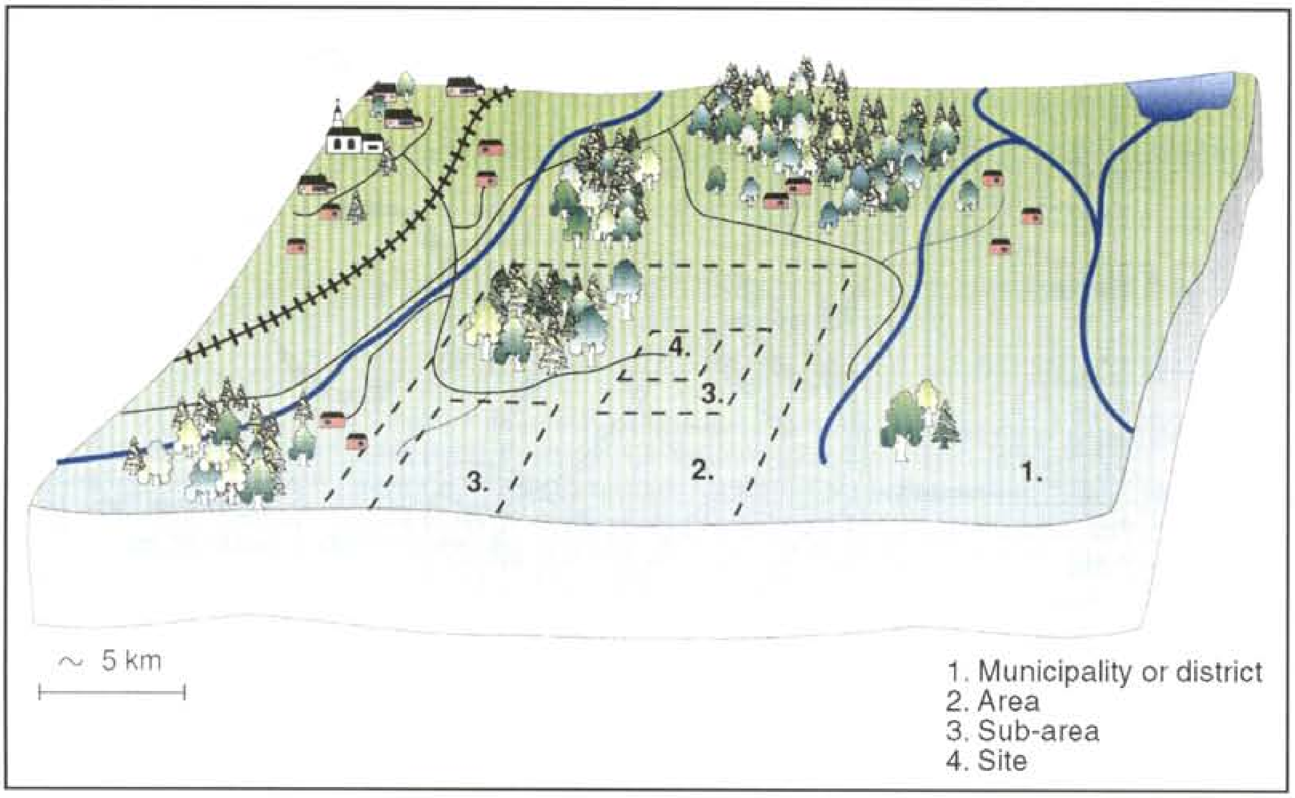


Figure 9-4. Definitions of geographic terms used for site investigations.

preliminary layout of repository sections. A suitable repository depth is designated based on this material. Further, a preliminary description is made of the frequency and characteristics of minor fracture zones, as well as the properties of the rest of the rock mass, as a basis for calculations of groundwater movement.

Step 2 Aims at investigating in greater detail the properties of the rock mass within the repository area, and answering specific questions identified during Step 1, by means of a second drilling and measurement programme.

Geoscientific evaluation

The geoscientific evaluation will be presented in the form of models within the subject areas geology, hydrogeology, groundwater chemistry and rock mechanics. The evaluation and presentation technology that has been developed at the Äspö HRL will provide guidance in this work. The descriptions will be progressively developed during the execution of the site investigations, with continuous feedback to planning and execution of new investigation steps.

9.4.3 Methods and instruments

Knowledge and experience from site investigations is presented in section 8.2, and in particular with regard to methods and instruments in section 8.2.4. Experience from previous site investigations, not least from the Äspö HRL's pre-investigations, has been taken into account when considering site investigation methods and instruments, as has experience from other countries with similar conditions and potential disposal media. The conclusion is that SKB has access to well-proven technology and know-how for forthcoming site investigations. However, it should also be noted that further development is needed or is under way within some areas.

Planned work relates not only to continued development of methods and instruments, but includes also plans for how geoscientific investigations can be carried out efficiently and with high quality. Since the investigation for the deep repository in many areas requires odd and/or more advanced technology that what is available on the ordinary geotechnical market, one must consider not only technology development but also technology availability. For this reason, SKB has for many years organized stocking and servicing of instruments and measurement systems. Even at a low intensity of field

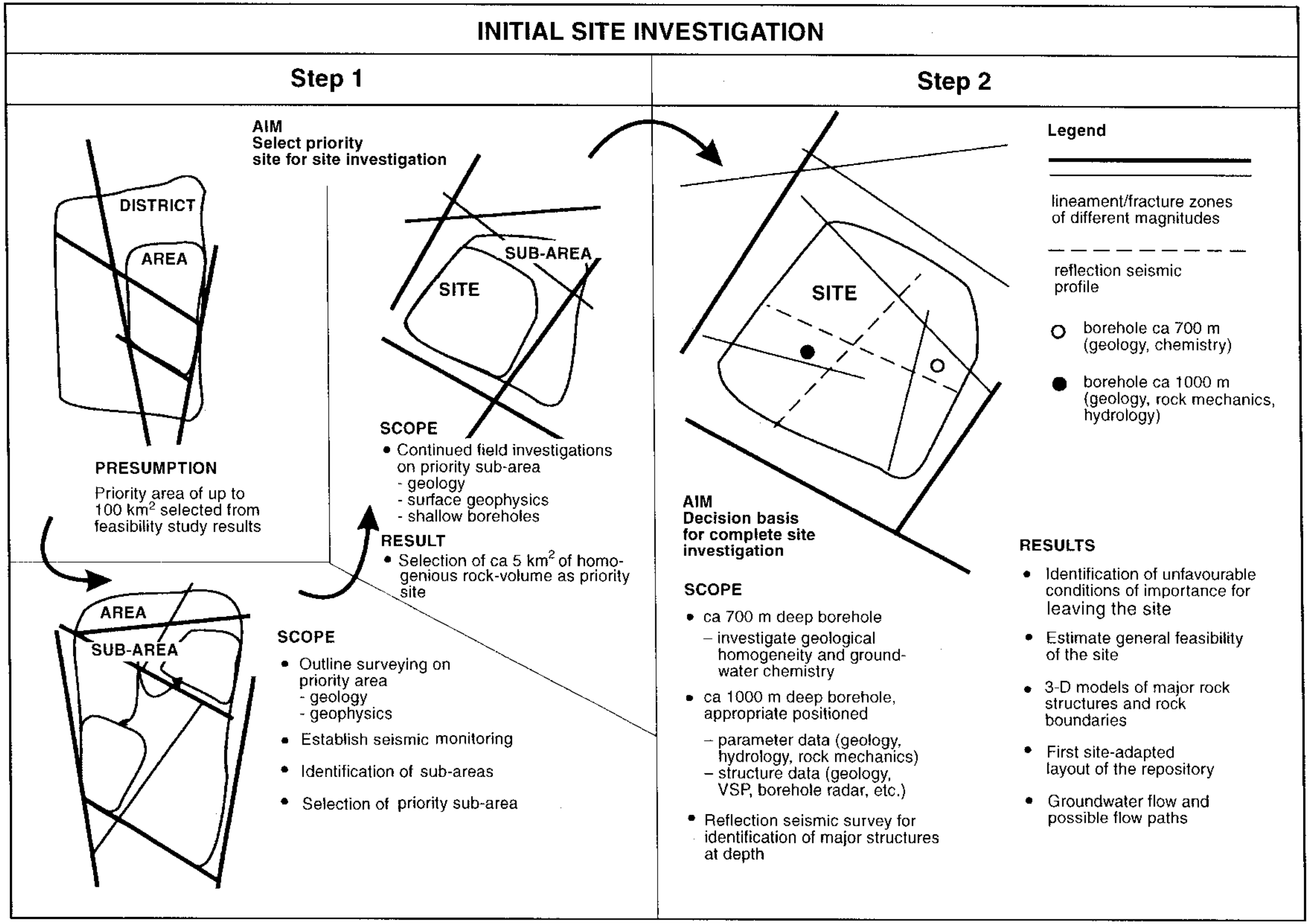
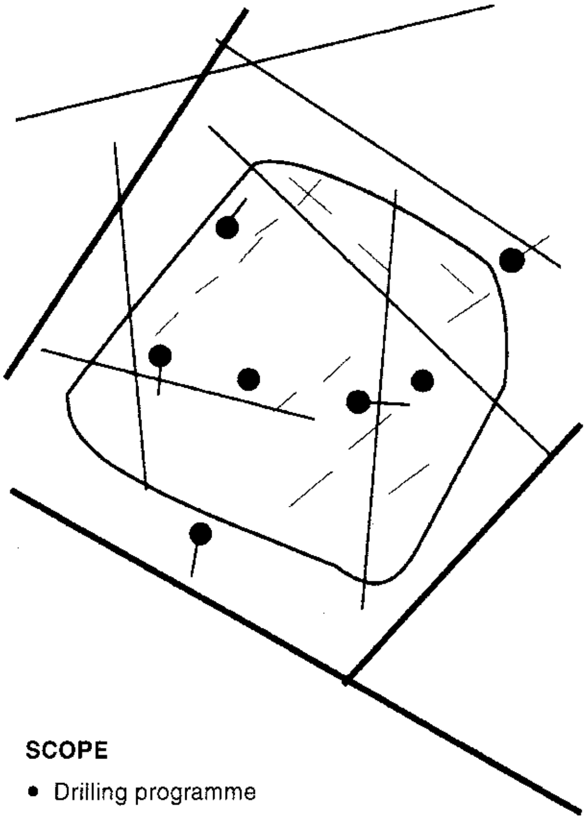


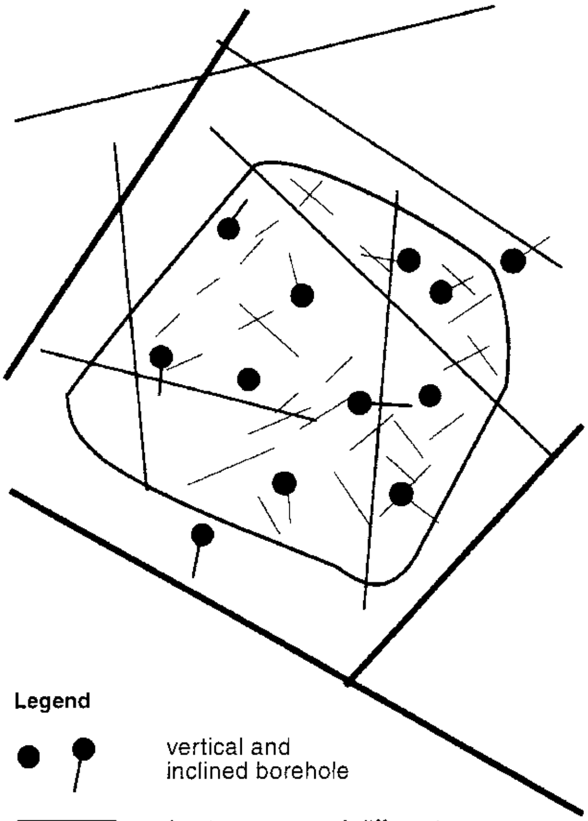
Figure 9-5. Conceptual sketch of scope of initial site investigation.

COMPLETE SITE INVESTIGATION

Step 1 | Step 2

AIM
Basis for application for detailed investigations





Step 1

SCOPE

- Drilling programme
  - extension and properties of rock types, including dykes
  - position and properties of fracture zones of major importance
  - frequence and properties of fracture zones of less importance

RESULTS

- Basis for performance assessment and design
  - preliminary layout (D2) adapted to site conditions
  - decision of repository depth, tunnel/shaft
  - construction analysis
- Basis for groundwater modelling
  - groundwater flow velocities
  - identification of discharge areas

Step 2

Legend

vertical and inclined borehole

fracture zones of different magnitude/importance

SCOPE

- Drilling programme
  - extended knowledge and increased detailness of parameter data on rock types and structures (extension, properties etc.)
  - in-situ measurement of groundwater flow
- Regional environment
  - supplementary investigations for verification/ adjustment of groundwater flow model
- Special issues
  - Complementary investigations and studies

RESULTS

- Basis for performance and safety assessment and design
  - analyses of longterm radiologic safety for different modelling alternatives and parameter sets
  - adjusted site-adapted layout of the repository

Figure 9-6. *Conceptual sketch of scope of complete site investigation.*

work, a certain level of maintenance must be upheld. Routines are currently being reviewed and improved in preparation for forthcoming site investigations.

For most measurement methods, site investigations on two sites in parallel require two or more sets of equipment of the same kind. This will lead to additional equipment procurements over the next few years. Furthermore, existing equipment is being modernized to increase its availability.

Some ongoing and planned development efforts are described below. Others may follow.

Surface geophysics

Surface geophysics will play an important role in the initial part of the site investigation. It will vary from site to site, depending on the geological conditions. On sites with an extensive soil cover, for example, geophysics has to be relied on to a greater extent for identification of fracture zones than where the surface of the bedrock is well-exposed. In the case of sub-horizontal zones, geophysics, and mainly seismic reflection, is the only means of identification prior to the start of drilling. As mentioned in section 8.2.4, it is a method where progress has been made and where field testing on a full scale is the next planned step. Besides seismic reflection, other surface geophysics methods, such as magnetic gradiometry and regional resistivity surveys, are planned to be used within the same field study.

Surveying

Various kinds of surveying are central to geoscientific investigations. As far as geographic positioning is concerned, GPS (Global Positioning System) will be used in the site investigations. This technology is making rapid strides and is becoming more and more commercially available for various applications. Its potential usefulness to SKB is associated with the setting-out of measurement profiles, coordinate determination of measurement points, positioning of boreholes, etc. The method is intended to be used in conjunction with the surface-geophysics field studies mentioned above.

Drilling technology

In section 8.2.4, it was noted that drilling through fracture zones is an area where the technology needs to be improved. It is a fundamental rule that no stabilizing additives to the drilling fluid are accepted when drilling at a potential repository site. Exceptions from this rule must, however, be made sometimes in order to be able to continue the drilling. So far, cementation has mainly been used. There are also ideas for other ways to tackle this instability problem. An inventory and feasibility study of these drilling-related questions is under way. The goal is to find standard solutions for various stability problems and for different requirements on what disturbances can be accepted.

The documentation of the drilling work and logging in conjunction with drilling, as well as of the borehole as such, will be streamlined. The drilling work produces a large quantity of peripheral information which, if properly documented, can help to shed light on various kinds of questions.

Tests during drilling

Hydraulic tests and water sampling during drilling were tried during the pre-investigations for the Äspö Project. The technology will be re-examined and improved, not least when it comes to wire-line drilling. The greatest improvement is likely to be achieved for water sampling, where sampling of undisturbed samples is striven for.

Hydrological documentation and hydraulic tests

A hydrogeological investigation programme encompasses several measurement, testing and monitoring methods. Some instrument and method questions should be further studied, tested or developed to optimize this investigation programme with respect to the representativity of the measurement data and resource requirements.

For determination of hydraulic conductivity, hydraulic injection tests are performed with SKB's mobile umbilical hose systems. To meet requirements on availability, these systems must be modernized to some extent. A simpler method for determination of hydraulic conductivity, which may be able to reduce the scope of the relatively time-consuming injection tests, has been developed in Finland. This method will be studied further, including testing in one of SKB's boreholes. Moreover, fabrication of a new pipe string system for pumping tests and injection tests was recently initiated.

The measurements performed in borehole KLX 02 at Laxemar, Oskarshamn, have brought attention to the influence of salinity on measurements of hydraulic properties. This borehole contains groundwater with a salinity of up to 8%, which increases the density of the water so markedly that it affects both evaluation and execution of the tests. Since high salinities can occur at other sites situated below the highest coastline, see Figure 5.5-5, measurement and evaluation methods will be developed during the next few years so that any site investigations in these areas are able to handle such conditions.

The method for flow logging during pumping will be modified slightly to provide better data at great density contrasts. In addition, use of an accurate absolute pressure probe is planned to obtain a better picture of the hydraulic driving forces.

The point dilution method with downhole probe for in-situ measurement of groundwater flow will be tested and evaluated, along with another measurement method developed in Finland for the same purpose.

Groundwater monitoring can be done in several ways. The method developed for the Äspö HRL will be evaluated thoroughly before it is decided which method is to be used in the site investigations.

Length calibration of boreholes

The importance of surveying measurement points was mentioned above. Exact coordinate determination in boreholes requires highly accurate measurement of both borehole deviation and depth (borehole length) to the measurement point. To better be able to compare and co-evaluate different measurement parameters, greater accuracy in length measurement is needed than is possible today. As mentioned in section 8.2.4, development is under way of a length calibration system where the boreholes are marked during drilling with rings at selected depths. The measurement probes are equipped with sensors to detect these rings. The work will continue with development of ring-setting tools for 56 mm boreholes, integration of detectors on the measurement probes and fine-tuning of the methodology.

Borehole geology and geophysics

Based on SKB's new borehole TV system (section 8.2.4), new routines for borehole geological documentation will be developed where core mapping and geophysics are integrated.

9.4.4 Data management and quality assurance

An efficient database is of central importance for the administration and control of data. The development of SKB's central geodatabase (SICADA) will be concluded in all essential respects during 1995. Quality assurance of data is intimately linked with an efficient database, but such a database is not a sole guarantor for this. The main motto of quality assurance was discussed in 8.2.6, "doing the right things", doing them "in the right way" and with adequate documentation.

A "Quality programme for site investigations" will be the comprehensive quality assurance document that specifically regulates the quality procedures within this area. The document exists in a first draft. For site investigations, quality assurance and traceability of data will be of essential importance, along with efficient documentation and reporting procedures.

Quality assurance will be integrated in regular programmes and plans. In connection with the site investigation, the individual who does the work is also responsible for ensuring that it has been done in the right way (according to instructions) and that the reporting document and/or the data set has been checked. Nonconformances and errors must be documented. Spot checks and quality audits will be made. Clear rules for responsibilities and powers in the organization are other important quality aspects.

For certain instructions, manuals that have been issued within the Äspö HRL will be used directly or with minor adjustments. In other cases new ones must be written. Another component of quality assurance is the technical documentation of hardware and software. For administrative organization of all manuals and technical records, a library system for this has been set up, SKB MD (method documentation).

9.5 DESIGN

The aim of the process of designing the deep repository is to achieve optimal function with respect to safety, environment and technology. SKB has begun the design work by preparing general plant descriptions which give examples of some possible ways to design the deep repository (section 8.1.1). To a large extent the design work can be based on experience from nuclear facilities and underground projects. There are, however, questions or requirements which pertain specifically to a deep repository, and which require special measures.

9.5.1 Design specifications

The plan for designing the deep repository is based on a number of design specifications consisting of technical principles, requirements on the disposal site etc. as follows:

- the waste is encapsulated elsewhere and the waste packages are transported to the deep repository,
- the total weight of package and transport protection is under 100 tonnes,
- the package with transport protection is transported down under ground, where the transport cask/container is opened for the first time,
- the waste units are deposited in crystalline rock at a depth of about 500 m,
- excavation and other construction works under ground can be done with conventional methods and using conventional building materials,
- the requirements on environment and occupational safety are comparable to those made in other underground projects.

9.5.2 The design process

The planning is based on the execution of the design process in defined steps as presented in RD&D-Programme 92, Supplement /9-2/. The division into design phases described there is influenced to some degree by the fact that site licensing for the deep repository takes place after the completion of the site investigations. This means, among other things, an earlier start of construction of the surface facilities. Construction of the deep repository proceeds in stages, and the engineering of different areas in the facility is carried out accordingly with a certain delay as solutions and design etc. are finalized as a basis for investment decisions.

The following division into design phases is used:

Conceptual design (Phase E)

The Plant Description comprises the first coordinated proposal for a design of the deep repository and serves as a basis for the subsequent design work. It also serves as one of the premises for the choice of rock excavation method, preparation of proposal documents for the rock and construction works and preparation of system function programmes for layout-determining systems within the underground facility such as ventilation, rock drainage and power supply. The plant description also sets forth the specifications for design of machines and vehicles.

All layout-determining alternatives or sub-solutions are developed to detailing degree E for subsequent detailing to level D, providing the alternative has not been dismissed as less interesting before then. The complete set of plant descriptions in detailing degree E serves as the point of departure for the continued design work.

Basic design (Phase D)

This phase is planned to encompass two distinct subphases as follows:

The first sub-phase, which is expected to start about 1.5 years after site investigations on the site have commenced, includes geographically and geologically tailored proposals for facility designs. Type of access (ramp or shaft) to repository level is chosen with respect to the conditions on the site in question. The results consist of situation plans above and below ground and layouts of the tunnel system, including access ramp/shaft. Individual buildings and the like are not studied.

A second sub-phase, which is estimated to be able to start about 3 years after site investigations have commenced on the site, includes a preliminary solution of the facility's layout adapted to the specific site. The preliminary solution comprises a processing of results from earlier work and is based on, among other things, preliminary operating and design-basis data. The needs of the investigation activities are taken into account. Layout drawings are prepared that define each facility section with regard to function, execution and interrelationship. The results serve as a basis for preliminary principal documents, including preliminary blasting drawings. This material serves as part of the supporting documentation for an application for a permit for detailed characterization and construction.

In a future situation, when the time comes to make a decision about construction of the deposition areas, the first measure will be a revision based on achieved results as a basis for further design.

General engineering (Phase C)

This phase is aimed at a final plant design. The facility is adapted to final operating and design-basis specifications.

The protection requirements must be stipulated and taken into account. The surface industrial area must blend in well with the terrain and be linked to the local infrastructure. The facility must be designed in consultation with concerned authorities. The results of this phase will serve as a basis for the procurement of construction and rock works and various systems, plus the preparation of principal documents and construction documents.

Engineering (Phase B)

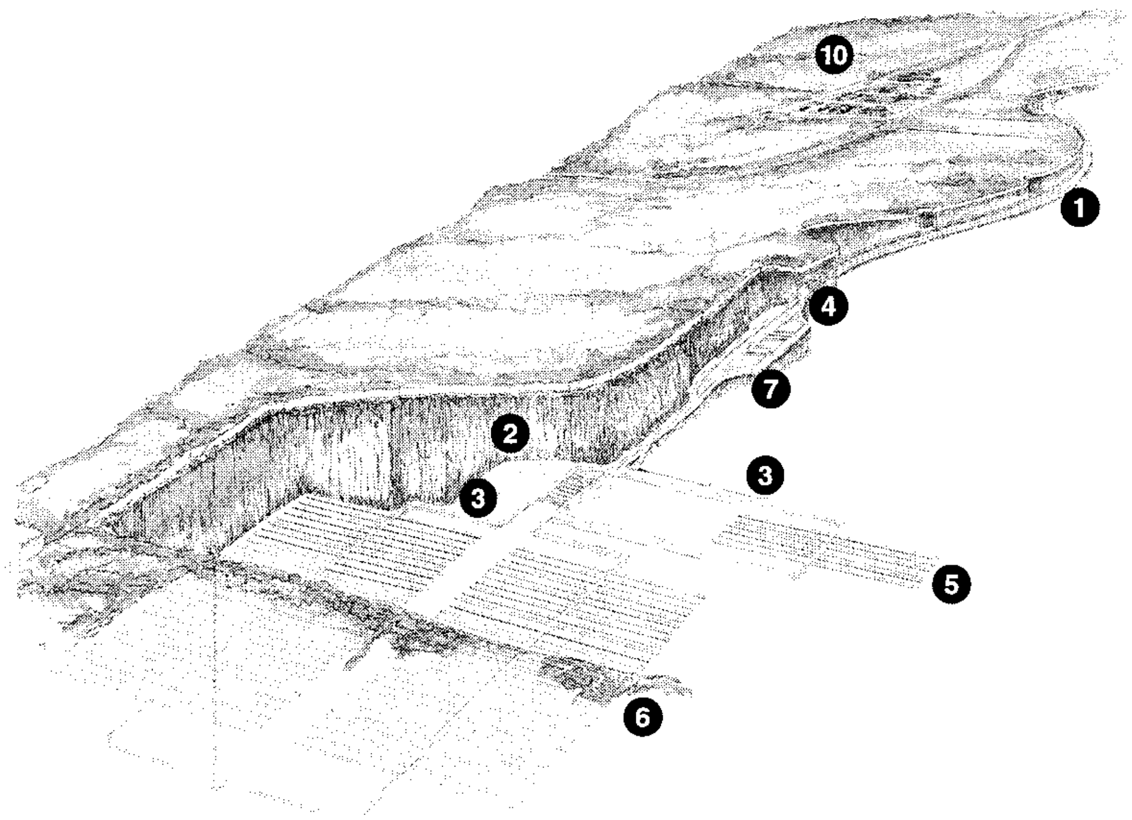
Phase B is aimed at a finally approved plan and preparation of working documents for construction, manufacture and installation.

Documentation (Phase A)

Phase A compiles the final documentation as a basis for a permit for start of operation.

9.5.3 Planned design measures up to application for permit for siting and excavation for detailed investigation

Preliminary design measures are currently being pursued for different parts of the deep repository based on the subdivision described in RD&D-Programme 92 and subsequently detailed more through the definition of different "modules", as shown by Figure 9-7. The intention is that it should be possible to examine different technical solutions for each module, so that a "kit" is created for the subsequent site-adapted design work. Depending on local conditions, different technical solutions may prove suitable for different sites.



Planned modular subdivision of the deep repository:

1. Access ramp to repository level.
2. Central area under ground.
3. Transport tunnels to deposition area for spent fuel.
4. Transport tunnel to deposition area for other long-lived waste.
5. Deposition area for initial operation.
6. Deposition area for regular operation.
7. Deposition area for other long-lived waste.

(8). Investigation activities.
(9). Infrasystems.
10. Surface facility.

Figure 9-7. *Modular subdivision of the deep repository.*

The work includes the following parts:

- Programme, layout and engineering work for land, buildings and rock caverns. Design work for hoists and lifting devices.
- Design work regarding systems (electric power, ventilation, etc.)
- Design and engineering work regarding machines and vehicles.
- Special studies of importance for the layout and engineering of the deep repository.

A prerequisite for start of site-related design is that the following choices can be made for the site to be studied:

- Deposition method.
- Access ramp or shaft.
- Design and application of bentonite barrier.
- Transportation system to deposition level.
- Tunnelling method.
- Diesel or electric power for machines and vehicles.

In large part, the work with a site-adapted layout will first consist of combining and adapting selected modules for each studied site. Later, when more extensive data are available detailed solutions are devised. Besides site-specific data, the design process is guided by system information and results from safety review.

Table 9-1. Topics of interest for further studies.

Phase E

Alternative conceptual designs
Deposition methods.
Down-transport of waste containers on ramp.
Transport of waste containers on the level.
Excavation methods.

Topics where a standpoint, priority or choice needs to be analyzed
Design of different repository areas.
Infrasystems.
Protection aspects.
Material development.
Construction methods and civil engineering solutions.
Quality assurance.
Nuclear safeguards.

Design-basis specifications
Waste types.
Process description.
Technical standard.
Mechanical equipment.

Phase D

Studies that may be addressed:

Topics where a standpoint, priority or choice needs to be analyzed
Design of different repository areas.
Infrasystems.
Protection aspects.
Material development.
Construction methods and civil engineering solutions.

Design-basis specifications
Process description.
Technical standard.
Mechanical equipment.

The need for machines and vehicles has been identified in conjunction with the writing of plant descriptions. Each machine type and vehicle type has been studied with a view towards its intended function. The intention is that these studies should serve as a point of departure for the necessary engineering work. The design work for machines and vehicles is then pursued step-by-step in agreement with the working model for the design process. Preliminary data of importance for the overall layout and function of the repository include space requirements, room profiles, interrelationship with other equipment and supply systems.

The described design model for the deep repository entails that the work of designing the facility follows a traditional pattern with application of conventional and proven technology.

Up until the site investigations begin, the design work is based on relevant plant descriptions, general data on the Swedish bedrock and information from investigations of various study sites. Activities during the next few years have the following purposes:

- serve as a basis for selection of best generic technical solutions,
- serve as a basis for site-specific design,
- help to limit the number of alternatives for continued studies,
- identify factors that can affect the local placement of the deep repository and planned geoscientific characterization of the rock volume for the deep repository.

The objective is to eliminate alternatives that are not deemed to be of interest at an early point. Important concrete goals are choice of type of access (ramp or shaft) to the deposition level and deposition method.

Interesting alternatives and different types of decisions and recommendations are presented in a manner that permits subsequent scrutiny. Table 9-1 shows topics which are or may be of interest for studies.

9.6 TECHNOLOGY FOR CONSTRUCTION, OPERATION AND CLOSURE OF THE DEEP REPOSITORY

Construction, operation and closure of the deep repository can be designed to a considerable extent on the basis of known technology and experience within the nuclear power industry and underground facilities. There is, however, a need for adaptation or development of the technology with a view towards the requirements and conditions that apply to the deep repository.

9.6.1 Construction

The different rock construction methods that may be used in the deep repository are well-developed with respect to the safety that is needed in order that tunnels and rock caverns will serve as safe worksites for the personnel.

Analyses, evaluation and comparison of different construction methods are carried out as a part of the design process. The influence of different building materials on mechanical stability, chemical conditions and hydraulic properties in the deep repository are examined by means of construction analysis.

The development of a construction analysis strategy can be divided into several different steps as follows:

- compilation of requirements in design and construction,
- review of important construction-related factors,
- predictability for these factors,
- verification.

Important factors that are dealt with in the construction analysis include:

- Mechanical stability:
  - Load cases.
  - Structure-controlled instability.
  - Passage of zones of weakness.
  - Rock burst.
- Groundwater:
  - Rock properties.
  - Water seepage during driving.
  - Groundwater head.
  - Groundwater drawdown.
  - Corrosion of installations.

Certain parts of the construction analysis will be verified by comparing what the analysis predicts with observations in the Äspö HRL.

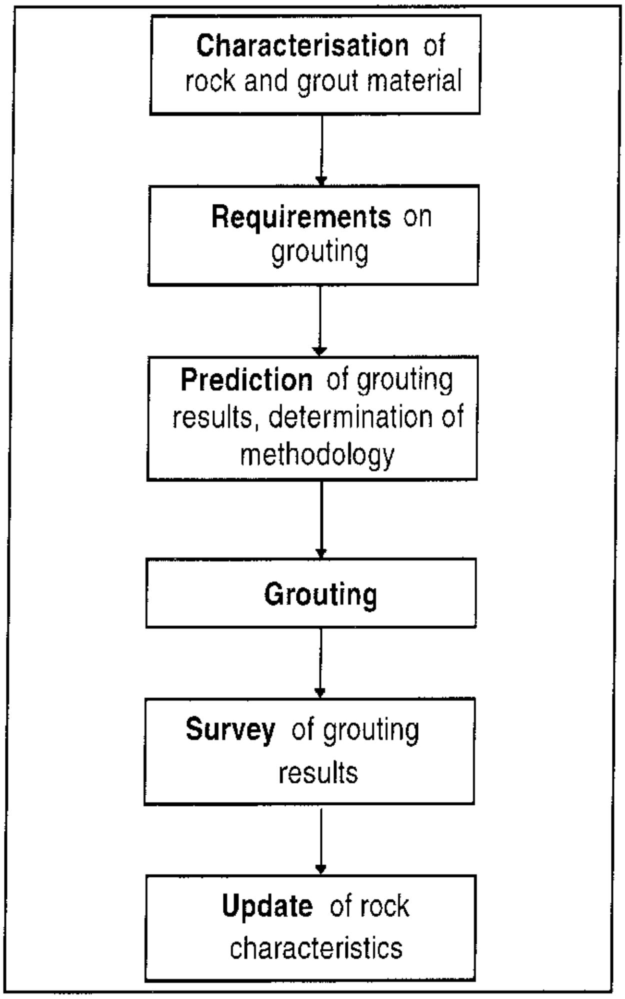


Figure 9-8. *The grouting process.*

Grouting

Knowledge of methods for grouting of fractures to prevent large seepages of water into the deep repository needs to be improved. Present-day knowledge is largely based on practical experience, while theoretical understanding is less satisfactory.

Planned measures for grouting are aimed at a better understanding of important factors and the size of constituent parameters, as well as modelling of the grouting process. Verification is done by means of tests under realistic conditions in the Äspö HRL.

Another important task for the next few years is to work out a strategy for passage of heavily water-conducting fracture zones at repository depth.

The practical measures that are required at the passage of these zones with a shaft/ramp/tunnel include sealing against water inflow and reinforcement so that the passage remains stable. Normally, post-excavation grouting is also needed in such cases as a part of the permanent reinforcement. The required properties of the grouting materials are determined primarily by what is needed for a successful sealing result, i.e. rapid hardening and high strength, and secondarily by the goal of limited chemical effects on the deep repository. The "respect distance" from canisters with spent fuel is determined by this consideration.

Smaller but highly water-conductive fracture zones should not pose any serious problems to mechanical stability. Within the deposition area, however, the quantity and type of grouting material is important, as is control of the spreading area.

Fine, discrete fractures are not mechanically important but can conduct water and therefore need to be grouted to achieve sufficiently low water inflows during the deposition and operating periods. If the groundwater has a high radon content, the seepage needs to be limited so that the need for ventilation air will not be too great.

Regardless of type of fracture or fracture zone, the grouting work is planned to be done in the steps shown by the flow scheme in Figure 9-8.

9.6.2 Development of machinery and equipment

Most tasks in the deep repository can be done with standard models of machinery or equipment available on the market. In some cases, however, the requirements are so special that special versions must first be developed and prototypes tested before they can be built for use in the deep repository. This may be the case when it comes to:

- Short tunnel boring machine (TBM) in the event full-face boring of deposition tunnels is chosen and the flexibility of the layout must be retained,
- boring of deposition holes,
- rail-borne transportation system on ramp if the rail-borne principle is chosen,
- truck for transport on ramp if the rail-free principle is chosen,
- hoisting mechanism for transporting heavy loads in a shaft if the shaft alternative is chosen,
- lifting yoke for various waste units and for various locations in the deep repository,
- transport vehicle for spent fuel on the deposition level,
- deposition vehicle for spent fuel,
- bentonite vehicle in case canisters and bentonite are handled separately,
- retrieval of canister after completed deposition and backfilling,
- freeing of deposited canister after the bentonite buffer has become water-saturated.

The general development plan for machinery and equipment that will be followed during the coming six-year period (1996–2001) includes the following steps:

Preliminary studies (Phase E)

- Conceptual sketches.
- Evaluation and comparison of different transportation and deposition systems, and elimination of alternatives.

Site investigation phase (Phase D)

- Preliminary size calculation.
- Design of scale working model, e.g. 1:10.
- Testing of model.
- Preliminary study of sub-systems to determine development need.

With few exceptions, the process of design, manufacture and testing of prototypes for machinery and equipment is not expected to start until during the next six-year period. The current situation and plans for the coming six-year period are described below for the machinery/equipment that needs to be specially developed for the deep repository.

TBM

In the event a short machine is needed, one can be designed on the basis of an existing conceptual sketch. Preliminary dimensioning is the first step. The machine will not be needed until construction of the deep repository is begun and should be designed and manufactured then for direct use in the deep repository without any model or prototype tests.

Boring machine for deposition holes

A boring machine has been tested in full-scale boring in Olkiluoto /9-9/, and it is deemed possible to design and manufacture a machine for production boring based on experience gained to date. A preliminary study will be conducted during the period for the purpose of analyzing the space requirement for the machine and how it affects the size of the deposition tunnel.

Lifting yoke

The need for lifting yokes will be determined and standardized models will be developed during the coming six-year period. Model-scale units may be manufactured and tested.

Rail-borne transport on ramp

A conceptual sketch for a system exists if this solution is chosen. Model-scale tests may be carried out.

Rail-free transport on ramp

A conceptual sketch exists for a truck design. Model tests with a truck are not judged to be necessary. Model tests of loading and unloading of transport casks/containers may possibly have to be carried out during the six-year period.

Hoisting mechanism in shaft

A hoisting mechanism for the loads to be handled is not available as a standard product today. Moreover, there are additional special requirements on an extra brake system, which needs to be designed. If transport of the waste packages by shaft is chosen, development will follow the steps outlined above based on the conceptual solutions presented by hoist manufacturers. Tests have been conducted in Germany with a hoisting mechanism prototype that exhibits good potential for meeting the requirements of the deep repository. Experience from this work may reduce the need for model and prototype tests.

Transport vehicle on deposition level

The reference concept is that a special vehicle is used for transport of the waste packages on the deposition level, but other alternatives that include the same vehicle from the ground surface to the deposition position are also conceivable. Regardless of choice of system, conceptual sketches need to be developed into preliminary dimensioning drawings. The sub-systems for lifting and docking to the deposition machine may have to be tested on a model scale, which will then be done during the six-year period.

Deposition vehicle

This vehicle will transfer waste packages from the transport cask/container to the disposal position and will probably be the most complex machine in the deep repository. Several of the sub-systems are judged to need to be individually tested before the whole machine is built, but the scope of the tests must first be analyzed, which is planned to be done during the six-year period.

Bentonite vehicle

The vehicle will first handle the bentonite rings before the canister is placed in its position, and then the blocks that are placed on top of the deposited canister. Development is being conducted in parallel with the development of the deposition vehicle.

Machines for retrieval

The cycle for retrieval after water saturation and swelling of the bentonite buffer encompasses first freeing of the canister and then gripping and hoisting plus possible cleaning of the canister from bentonite and placement in a transport cask. For this a new unit for freeing of the canister is needed. Only an interchangeable lifting yoke is required for the deposition vehicle. It is deemed possible to use the same transport casks and transportation system for retrieval as were used for deposition.

9.6.3 Application of buffer and backfilling

The bentonite buffer around the canisters is planned to be applied in block form. After deposition, the bentonite swells as it becomes saturated with water and fills out the whole space between canister and rock wall. The technology for block compaction is under development.

In the reference concept, backfilling of the deposition tunnels is planned to be carried out with in-situ compaction of deposited layers of backfill material. A problem to which attention was drawn in the Stripa Project is that methods have to be developed for compaction of the backfill material against the roof of the tunnel.

Development of method for compaction of bentonite blocks

The continued work of scaling-up of the block compaction method is planned to be concentrated to begin with on uniaxial compression of large blocks which can either have the form of segments or whole rings 1.5–1.6 in diameter. In an initial stage the technology is being developed for blocks with a diameter of about 1.0 m. This is planned to be done with a specially-made mould that is used in an existing press in a workshop in Ystad. This press is also powerful enough to compact full-sized blocks in a later stage.

Besides assessment of the suitability of the blocks for handling and use in the deep repository, the presence of any irregularities in the blocks due to the uniaxial compression method will also be clarified.

The isostatic pressing method used in the Stripa tests will also be evaluated so that it can be compared with the uniaxial compression method. The isostatic pressing method will probably also have to be tested on full-sized blocks in order to enable a detailed comparison of the methods to be done.

Regardless of compaction technology, a packing and handling method for the compacted blocks remains to be developed. This work is being pursued in parallel with development of the compaction technology.

Development of method for backfilling

Different methods for emplacement and compaction of backfill material in tunnels are planned to be tested on full scale in the Äspö HRL. The following materials are planned to be used:

- uncrushed TBM muck,
- crushed TBM muck,
- mixtures of crushed TBM muck and bentonite in proportions of 10, 20 and 30% bentonite.

The principal compaction method is vibratory compaction of horizontal layers and up towards the roof of sloping layers. Emplacement of precompacted blocks of mixtures with 30% bentonite up against the roof is also intended to be tried.

The compaction properties, hydraulic properties and compression properties of these materials will be tested in the laboratory for optimization of method and quality and prediction of the results of the field tests.

9.6.4 Closure

Four different requirements are made on closure and plugging for different needs in the deep repository:

- prevent or greatly limit the flow of water in the tunnel itself,
- limit the axial flow in the disturbed zone around tunnels and shaft,
- prevent or greatly limit the flow of water in drilled investigation holes,
- impair unauthorized intrusion in the repository.

The programme for the coming six-year period (1996–2001) includes the following:

Sealing of tunnel

Two principles for sealing of tunnels will be studied during the coming six-year period: plugs of bentonite/rock aggregate and concrete plugs with O-rings of bentonite against the rock wall. The alternative of swelling concrete without an O-ring will also be evaluated. Dimensions and material grades will be analyzed with respect to long-term durability and chemical effects on the surrounding medium. Verification tests are planned as a part of the buffer tests in the Äspö HRL.

Limitation of the axial flow in the disturbed zone

Slots into the rock wall will be further studied with respect to shape and location. The theoretical premises will be tested by modelling and calculations for different rock conditions. Formulations of filler material in the slots and supporting plug designs for keeping the slot material in place will be analyzed. Opportunities for development and verification of design solutions will be provided in the Äspö HRL in conjunction with different buffer tests.

Sealing against fracture zones

It is essential that water flow between fracture zones and tunnels in the deep repository be limited. For this purpose, the effects of different plug designs with and without grouting of the fracture zone and its near-rock will be studied. Important in this context is the long-term durability of the sealing effect.

Plugs against the ground surface

The uppermost 100 m or so of shaft and ramp against the surface will be plugged in order to impair human intrusion. Concrete will be used to build a rock-like seal while asphalt or similar material appears suitable for sealing against water transport past the plug. Different designs will be analyzed for sealing capacity and durability.

Plugging of boreholes

Existing technology for plugging of short (a few hundred m) diamond-drilled boreholes in sound rock will be developed for longer holes (1000 m) during the six-year period. In addition, technology will be developed for filling-out of volumes where the hole has been widened by e.g. flushing-away of loose rock. The development work will extend to both material selection and how plugging is to be done.

9.6.5 Monitoring

The technical options for monitoring of a completely or partially sealed deep repository will be explored during the coming six-year period. Effects of different types of monitoring measures on short-term and long-term safety will thereby be taken into consideration.

9.7 TASKS PERTAINING TO REPOSITORY SECTION FOR OTHER WASTE

The parts of the deep repository that will receive long-lived low- and intermediate-level waste (see section 8.1.1) will not be taken into operation until around 2020. The technology that is needed can largely be based on experience from SFR. An updated inventory of this waste must be completed for the safety assessment that will form a part of the safety report accompanying the application for a permit to site the deep repository. The purpose of the safety assessment is to serve as a basis for designing the repository and for deciding what further studies need to be done.

To achieve good design and function, a number of questions need to be answered, of which the following are some of the most important:

- What requirements are made on the buffer function of the backfill?
- Where is backfill needed and what should it consist of?
- Are concrete structures needed, and if so what is their required longevity?
- What requirements are made on the barriers at the boundaries of the different repository sections?
- How will the waste be handled during deposition, i.e. what are the requirements on remote control and radiation protection?

Work has started on answering some of the above questions. Since all of the questions affect each other, the safety assessment and the design, they will be studied as an integrated system. Ongoing and planned measures for characterization of the waste are described in sections 5.9 and 11.9.

9.8 OPERATING SAFETY AND SAFEGUARDS

The design of the repository as a whole, and particularly the design of the deposition technology, will largely be guided by the requirement of high operating safety.

The canisters of spent nuclear fuel arrive at the deep repository in transport casks, which according to the reference concept are transported down under ground without being opened and up to the mouth of the deposition tunnel. There the transport cask is docked with the special deposition vehicle and the canister is transferred to the deposition vehicle, which is equipped with radiation shields against the tunnel mouth. Finally the deposition vehicle is driven up to the deposition hole and deposits the canister in the bentonite-lined deposition hole, which is then filled with bentonite up to the tunnel floor.

Other long-lived waste is planned to be handled in the same manner as in the rock vaults in SFR, where transport containers are driven into the deposition chamber. The transport container is opened and the packages are lifted out and transported to their deposition position.

An assessment of operating safety in these handling operations will be included in the environmental impact assessment and as a supporting document for the permit application, see also chapter 10. As a basis for this, an assessment is made of the technical performance of the handling systems in the deep repository. In this way more or less probable incidents and accidents can be identified. This makes it possible to carry out the operating safety assessment and to introduce safety-enhancing measures.

Over the next few years, SKB will participate in the IAEA's work of formulating a safeguard policy for the deep repository (see chapter 8). The IAEA's recommendations will provide guidance in the process of designing the deep repository and its transportation system.

9.9 TRANSPORTATION

Transportation from the encapsulation plant at CLAB to the deep repository will be carried out with exclusively conventional technology. A safety report will be produced as a part of the environmental impact assessment for the deep repository. The choice between different transport modes and equipment will be a practical question linked to the siting of the deep repository. No research on transportation is expected to be conducted in the next few years.

One necessary development measure is to modify the reference transport cask so that it is adapted to the latest concept for the canisters. A renewed study will show how the mechanical properties of the cask are affected by the heavier canister, and thereby how the cask should be modified to meet the desired safety requirements.

This study will be carried out during the coming three-year period. The actual licensing of the cask is, however, a question that lies considerably further ahead in time.

10 PROGRAMME FOR SAFETY ASSESSMENTS ETC.

10.1 OVERVIEW

SKB's programme for siting, designing and building the deep repository and the encapsulation plant was described in RD&D-Programme 92 and its supplement. Important occasions for coming decisions/permits in this programme are presented in chapters 7 and 9 and summarized in Table 10-1.

Comprehensive background material will be presented in support of the decisions and licensing procedures shown in Table 10-1. Much of this material concerns radiological safety. This material is presented in the form of safety reports, which at the same time comprise a part of the required environmental impact assessment.

Due to the many nuclear facilities that have been taken into operation, the reporting of their operating safety has been standardized and harmonized. There is no corresponding standard for reporting of long-term safety after closure. However, since long-term safety will be reported on a number of occasions during the development of the Swedish system for radioactive waste, a proposed template for safety reports has been presented in a separate report, SR 95 /10-1/. Such a template is intended to simplify the follow-up of how the safety assessments continuously incorporate a progressively more detailed body of data, and how this influences the assessments of safety. SR 95 also describes the methods and analysis tools that are available today for carrying out the assessments of long-term safety.

The scope of the safety reports and other safety-related material supporting the first three decision-making occasions is discussed below. Subsequent decisions lie so far ahead in time that it is not meaningful to discuss the requirements on necessary background material today.

10.2 APPLICATION FOR PERMITS FOR ENCAPSULATION PLANT

Permits under Act Concerning the Management of Natural Resources (NRL) and Act on Nuclear Activities (KTL) to site and construct the encapsulation plant for

Table 10-1. Coming safety assessments.

	Background material for decision on permit for:	
Safety assessment	**Encapsulation plant**	**Deep repository**
Encapsulation plant SR-I	Siting Construction	
Deep repository SR-D		Siting Construction, incl. detailed geoscientific investigations and some construction
Initial operation (Stage 1)	Start encapsulation – spent fuel Initial operation – stage 1	Initial operation – Stage 1 – deposition, progressive excavation of deposition tunnels
Regular operation (Stage 2)	Supplementary extension Regular operation – stage 2	Regular operation – Stage 2 – deposition, progressive excavation of deposition tunnels
Decommissioning	Decommissioning	Possible supervised storage, closure

spent nuclear fuel etc. will be applied for prior to the start of construction of such a plant.

In the decision after the supplement to RD&D-Programme 92 /10-2/, the Government states that:

- *SKB should not commit itself to any specific waste management and disposal method until a consolidated and detailed analysis of the associated safety and radiation protection aspects has been presented,*
- *technical requirements on individual barriers and components in the repository system [should not be] established until the system's overall safety has been described in a satisfactory fashion,*
- *a decision (to construct the encapsulation plant) should not be made until a safety assessment of the repository system as a whole has been presented (...) and been found to be suitable,*
- *a consolidated assessment should be included as supporting documentation in any applications for permits (under NRL and KTL) for the encapsulation plant.*

To comprise the consolidated and detailed analysis of the associated safety and radiation protection aspects requested above, the supporting material for the permit application should include:

- Safety report for the encapsulation plant.
- Account of system for transportation of encapsulated fuel from encapsulation plant to deep repository.
- Safety report for deep disposal of encapsulated spent nuclear fuel after closure.

In addition, background material will be compiled for the safety aspects of the zero alternative according to the EIA requirements and for a plant for dry storage of retrieved encapsulated fuel (if any).

The safety report for the encapsulation plant covers the plant and the activities that are needed to encapsulate spent nuclear fuel etc. It will deal with the encapsulation process for the spent fuel assemblies and other core components, as well as activities in CLAB occasioned by encapsulation. The report will have an outline similar to that of the preliminary safety reports that were compiled prior to construction of nuclear power plants and CLAB.

The report is supposed to describe safety in and outside the plant during normal operation and in the event of accidents, plus the arrangements that have been made to ensure that relevant radiation protection regulations can be complied with. The safety report must further specifically describe:

- the requirements that have been imposed on the canister with regard to radiation protection and safety, for example safety against criticality, integrity and mechanical strength,
- the quality of encapsulation that can be obtained with the selected fabrication method,
- the testing that will be utilized to show that the requirements have been met,
- how defective canisters will be handled.

The account of the transportation system describes the planned transportation system and the scope of requisite shipments of encapsulated fuel from the encapsulation plant to different alternative disposal sites for the deep repository. The account at this stage is mainly a survey of the transport options.

The requirements are in principle the same as those for the system that is in operation today for shipments to CLAB and SFR. The present-day safety report for these shipments covers the essential aspects of the deep repository shipments as well.

The weight of the copper/steel canisters and the strength requirements on them occasion specific analyses of the transport cask.

The transport casks will licensed according to the IAEA regulations for Type B packages, which means that specified requirements regarding radiation shielding and strength must be met, both during normal operation and in accident situations. The licence for such a transport cask stipulates the limits on what it may contain, in other words the requirements on the structure of the cask are dependent on what is to be transported in it. Such licensing must take place in good time before the estimated start of the shipments, see further under transportation in sections 10.3 and 10.4.

Transportation under ground is dealt with in the safety report that describes the operation of the deep repository.

The report on the **long-term safety of the deep repository** covers the deep repository for spent nuclear fuel and its post-closure safety.

The report deals with the canister's long-term safety function (to isolate the spent fuel from groundwater and to retard the dispersion of radionuclides if this isolation function is lost) under both expected and reasonably unfavourable conditions in the repository. Long-term safety for other types of waste that are admitted to the deep repository will be described in conjunction with permit applications for the deep repository. The report should specify which environmental conditions were assumed to exist when the functional requirements for the canister were established and compare these conditions with the conditions that normally exist in the bedrock proposed to host the deep repository.

Since the siting process is not expected to have reached a point in time when data from candidate sites are available, the safety assessments will be based on typical conditions in Swedish crystalline bedrock. Geochemical conditions in the deep groundwater will be chosen on the basis of the general knowledge that exists concerning groundwater in Swedish crystalline rock and of SKB's experience from previous investigation sites, Stripa and Äspö. The safety- and construction-related siting factors described in the supplement to RD&D-Programme 92 will be specified in greater detail, and the sensitivity of the safety assessment to variations in these factors will be examined.

When a candidate site has been located, an initial investigation stage is planned to be utilized for a review of the characteristics of the site with respect to the siting factors, see chapter 9. In this report, the comparison between assumed chemical conditions and actual conditions on the site in question is also planned to be updated.

The safety evaluation of the zero alternative according to EIA requirements entails an analysis of the possibilities – and safety consequences – of extended operation in CLAB. The analysis is based on the safety report for CLAB. Experience of long-term storage of zircaloy-clad fuel will be compiled and reviewed, as will experience of dry storage of fuel.

A safety evaluation of a facility for dry storage of retrieved encapsulated nuclear fuel will be presented to indicate the feasibility and safety of long-term storage of retrieved canisters. The permit application is not planned to include the dry storage facility, however.

10.3 APPLICATION FOR PERMITS FOR DEEP REPOSITORY

An application is submitted after surface-based investigations on two candidate sites have been carried out, but before construction of an access ramp/shaft to repository depth has been begun.

The application pertains to a permit under NRL to site the deep repository and to carry out detailed geoscientific characterization to confirm the suitability of the rock to host a deep repository for long-lived radioactive waste. Since certain rock facilities which are necessary for the detailed characterization may be included as parts of a deep repository later on, the Government observed in its decision after the supplement to RD&D-Programme 92 /10-2/ that the detailed characterization comprises a part of the construction of a nuclear facility. For this reason, the application will also pertain to a permit under KTL to site and begin construction of a deep repository on the site.

The supporting material concerning the safety aspects will therefore include:

- Safety report for transport of encapsulated spent nuclear fuel etc. between the encapsulation plant and a deep repository situated on the site proposed for detailed characterization.
- Safety report for operation of the deep repository on the site proposed for detailed characterization including retrieval of encapsulated fuel.
- Comparison between the investigated sites with respect to safety and constructability.
- Safety report for deep disposal of long-lived waste on the prioritized site.

The safety report for the transportation system will be a further elaboration of the report that was submitted in support of the permit application for the encapsulation plant. The material will be more detailed with respect to transport routes and with respect to transportation of different waste types.

The contents will include descriptions of the organization of the transport activities, transport casks and containers, ships and vehicles, routes and road choices, scope of the shipments, radiation protection during transport, accident preparedness, physical protection, and accident analysis of the different steps.

The safety report for operation of the deep repository will be a preliminary safety report for the facility and activities that are required to dispose of spent nuclear fuel and other long-lived waste, i.e. to

- receive shipments of encapsulated nuclear fuel and other long-lived waste,
- handle and deposit the waste packages in a safe manner with respect to radiation protection and safety,
- backfill the deposition tunnels and chambers to the requisite quality, and
- retrieve previously deposited encapsulated nuclear fuel if this should be found desirable.

The assessments of the long-term safety of the deep repository can lead to requirements on the methods for building the repository or for fabricating barrier components.

The report will describe safety inside and outside the plant during normal operation and in the event of accidents, plus the arrangements that have been made to ensure that relevant radiation protection regulations can be complied with. The safety report must further specifically describe:

- the requirements that have been imposed on deposition positions and deposition caverns with respect to radiation protection and safety,
- the requirements that have been imposed on engineered safety barriers around different waste packages with respect to radiation protection and safety,
- the quality that can be obtained with the chosen technology for construction and fabrication of the near field and the engineered barriers, including the testing that will be utilized to show that the requirements have been met and how any defective positions and barriers will be handled.

Comparison between the investigated sites

The process of investigating and evaluating the geological conditions on the candidate sites and gradually narrowing the focus of the investigations to parts with good potential for constructability and safety will be described in the programme for the site investigations (cf. chapter 9).

The integrated safety assessments will be preceded by analyses of groundwater movements in the area in order to utilize the disposal site and locate the different repository sections so that the safety potential of the site is effectively exploited. Geological structure modelling, modelling of groundwater movements and identification of areas where deep groundwater reaches the biosphere, will be done to an equivalent level on both candidate sites in order to permit comparison.

Depending on site-specific conditions, geometric boundaries in the modelling of groundwater or nuclide transport may be chosen differently on the candidate sites. An important part of the comparison is thereby the analysis of the safety-related importance of uncertainties or alternative interpretations of the geological structure of the candidate sites, i.e. the potential of the sites to provide safe final disposal is compared taking into account remaining uncertainties.

If the safety potential of the two sites is sufficient and largely equal, other siting factors besides the safety aspects will also be weighed in.

The report on the long-term safety of the deep repository will present a site-specific assessment of the safety of a deep repository situated on the candidate site recommended for detailed characterization. Canister data will be obtained from the specifications in the preliminary safety report for the encapsulation plant.

The placement of the different repository sections in the host rock and the layout of the deposition tunnels etc. will be done on a site-specific basis and linked to investigations and evaluation of the candidate sites (cf. chapter 9). The safety-related significance of the important differences between the two investigated candidate sites will be assessed in variation analyses.

The safety assessment will cover both spent fuel, stage 1 and stage 2, and other long-lived waste. The outline of the safety report will follow the same template as the report on the long-term safety of the deep repository submitted in support of the permit application for the encapsulation plant.

10.4 APPLICATION FOR PERMITS FOR OPERATION – STAGE 1

Permit applications for operation are submitted after construction of the encapsulation plant has been completed and tests have shown that the plant is ready to be put into operation, and after detailed geoscientific characterization on the repository site has shown that the site provides sufficient volumes, with ample margin, for a repository and possesses such characteristics that the safety requirements can be met. The planned construction stage 1 will provide additional background data on the site. After construction stage 1 has been finished, detailed characterization will continue within the area for stage 2.

The applications pertain to a permit according to KTL for production of encapsulated fuel and a permit for commissioning of stage 1 of the deep repository, comprising 5–10% of the spent nuclear fuel.

The safety-related material is planned to include:

- Safety report for operation of encapsulation plant.
- Safety report for shipments of encapsulated nuclear fuel.
- Safety report for operation of stage 1 of the deep repository.
- Safety report for the deep repository after closure, stage 1.

The safety report for operation of the encapsulation plant comprises a final safety report for the encapsulation plant.

The safety report for shipments of encapsulated nuclear fuel comprises a final safety report for the transportation system for encapsulated fuel.

In this phase, transport casks for encapsulated fuel are licensed in accordance with the IAEA regulations for Type B packages. Transport casks for other waste types are licensed in connection with the application for a permit for operation stage 2.

The safety report for operation of stage 1 of the deep repository comprises an account of safety in connection with the handling of the encapsulated waste in the deep repository.

The report is supposed to specify the progressive excavation method that will be used for the construction of phase 1, including

- the grounds for choice of location and acceptance criteria for deposition tunnels, deposition holes and tunnel plugs,
- how planned tunnel routing and canister deposition are adapted in the event that rock units of inferior quality are encountered, and
- planned quality control.

Finally, the safety report should also describe methods for application/manufacture, acceptance requirements and inspection methods for bentonite buffer and backfill.

The radiological consequences of the activities are to be reported under normal conditions of excavation and operation and in connection with possible accidents. Safety in retrieval of canisters must also be described.

The safety report for the deep repository after closure, stage 1, comprises an account of the long-term safety of the deep repository for spent fuel according to the same principles as were established in previous analyses, i.e. that judgements and calculations should be based on available site- and system-specific data, and be supplemented where necessary with forecasts or generic information.

The assessment will be based on

- canister properties in accordance with the final safety report for the encapsulation plant,
- the acceptance requirements defined in the safety report for operation of the deep repository as far as properties of rock and engineered barriers are concerned,
- a biosphere description and structural model of the repository revised after the detailed characterization.

Once the working methods in the repository have been defined, it will be possible to describe in greater detail the possible consequence of material left behind in the tunnels. The closure and sealing of deposition tunnels will also be able to be described in greater detail, in terms of the construction, placement and function of the plugs. Since the performance of the first stage of deposition will be followed up at the same time as detailed geoscientific characterization for stage 2 continues, the impact of the continued activities on the safety of the repository will also be evaluated.

10.5 OTHER PERMIT APPLICATIONS

After the excavation and deposition of stage 1 have been completed, experience and results will be compiled for a total evaluation. The safety reports will be revised in accordance with actually measured conditions in the host rock and engineered barriers. The need to make changes in methods for construction of barriers or rock chambers and for inspection of these will be evaluated.

The results will comprise a basis for a decision to apply for a permit to continue deposition in stage 2.

11 PROGRAMME FOR SUPPORTIVE R&D

11.1 GENERAL

This chapter contains a detailed programme for the supportive research and development that is needed during the six-year period 1996–2001 in order to implement the main strategy of the RD&D-Programme. The different sections in this chapter presume knowledge of the corresponding sections in chapter 5.

The supportive research and development aims at:

- building up a good understanding of the phenomena and processes that may be of importance for long-term safety in the deep disposal of radioactive waste,
- refining the numerical models for the processes in the repository that are essential for its safety,
- building up the requisite generic databases and comparison material so that the performance and safety can be evaluated in conjunction with the planned site investigations.

In RD&D-Programme 92, SKB judged that sufficient understanding and capability to quantify the safety of the deep repository existed to enable SKB to begin siting and construction of the facilities required for deep disposal. The development work that has been carried out since then has gradually narrowed its focus to technology for practical site investigation, the construction of barriers, the examination of other waste forms than spent nuclear fuel, and the evaluation of the dependence of safety on extreme events such as ice ages, major earthquakes, intrusion, etc. This process is also seen in the continued programme.

In view of the strong link that exists between the encapsulation project and canister-related R&D work, the latter is described in chapter 7. A similar link also exists between RD&D for fabrication and inspection of other engineered barriers and the deep repository project. These parts of the programme are described in chapter 9.

The RD&D-programme associated with the Äspö HRL is described in chapter 12. During the construction phase, the RD&D activities there have been strongly linked to the excavation and characterization of the Äspö HRL. The continued programme requires coordinated planning between the different underground experiments. Chapter 12 therefore includes both previously started and partially completed characterization, testing and experimental activities on Äspö and the larger integrated demonstration tests that will be conducted.

The amount of non-project-linked R&D activities will continue to diminish, but this does not mean that they can be discontinued altogether. Both demonstration trials and practical tests, and site-specific bedrock investigations will continue to identify questions and problems that require further deepening of knowledge and databases. Deepened knowledge also contributes to an ability to describe the probable evolution of the repository without unfavourable simplifications and to quantify the safety margins.

Furthermore, it is important to maintain a high level of competence within areas that are essential for safety and implementation.

11.2 SPENT FUEL

Currently available experimental data from fuel leaching can be used to determine an upper limit for the release of radionuclides from spent fuel in contact with groundwater.

The goals for the coming 6-year period are:

- Progressive refinement of our understanding of how radioactive materials are released from spent fuel under different conditions.
- Improvement of currently available models prior to forthcoming safety assessments.
- Development of a realistic model for the release of radionuclides from the fuel prior to application for a permit for siting and detailed site characterization for the deep repository.

11.2.1 Deepened understanding of how radioactive materials are released from spent fuel

Corrosion of spent nuclear fuel and release of fission products and actinides to the groundwater is a process that is dependent on a number of factors such as the burnup of the fuel, composition of the groundwater, etc. A better understanding of the corrosion process requires an improved quantitative determination of the properties of the fuel and of external conditions that are of importance for the corrosion process. This requires continued development of measurement methods and means of analysis. Examples of important parameters are the surface area of the fuel, the extent of radiolytic oxidation and the redox potential. Several of these parameters must be measured in situ during ongoing corrosion experiments. The high radiation field around the fuel makes it difficult to perform these measurements, since there is a possibility that measurement probes and electrodes will be affected by the radiation. Development work during the coming three-year period will be devoted to devising specific experimental set-ups to follow parameter changes such as redox potential and pH during the fuel corrosion experiments.

Corrosion of high-level fuel

The emphasis in the corrosion experiments is gradually shifting from large series of leaching experiments of BWR and PWR fuel in oxidative environments to experiments under anaerobic and reducing conditions. The environment in the solution will be controlled by means of an inert gas atmosphere or reductants, such as Fe(II) sources. The experiments are expected to require more complicated experimental set-ups in a radiation environment.

Studies at elevated temperature during fuel corrosion to promote the formation of secondary phases are in progress.

Due to the relatively high ionic strengths of certain deep groundwaters, there is a need for measurements of fuel corrosion in groundwater with moderate salinity. The experimental results will be used to validate model calculations as far as possible, based on improved databases for weak complexes and solubilities of tetravalent actinides.

Flow experiments

Flow reactors have been utilized for studies of the kinetics associated with the dissolution of sparingly soluble solids. Planning is under way to employ this technique in SKB's fuel studies.

11.2.2 Improvement of currently available models

Solubility of actinides and fission products

There is a large spread in thermodynamic data for solubility-limiting phases and species of U(IV), Np(IV) and Pu(IV). This is especially the case for the solubility of $PuO_2(s)$ under reducing conditions and in the presence of carbonate. Experimental determination of the solubility product of plutonium oxide in carbonate solutions under reducing conditions will therefore be carried out.

Solubilities and complexation equilibria for tetravalent actinides are usually determined at high ionic strengths in order to circumvent the problem of their strong hydrolysis and low solubility. Attempts will be made during the period to improve the procedures for extracting data at the ionic strengths relevant for repository conditions.

Model development

Development of models will be directed towards integrating the mechanisms and rate of formation of radiolytic oxidants and the role of oxidative layers and gradients. The development of the basic kinetic theories makes it possible to integrate this fundamental information into the spent fuel stability models.

Data concerning the rate and quantity of oxidizing radiolytic species produced will be used, together with information on how they interact with the fuel surface, to develop a quantitative kinetic model for fuel dissolution. Furthermore, experiments are planned aimed at understanding and quantifying the kinetics of secondary phases precipitation under repository conditions, i.e. determination of the critical U(VI) concentration and the surface/volume ratio that initiates precipitation, and its influence on the spent fuel surface.

Natural analogues

Corrosion of spent nuclear fuel over long periods of time can only be predicted by combining data from laboratory experiments of relative brief duration with studies of natural analogues. The work during the period will be focused on a better understanding of the structure and the crystal chemistry of uranium oxides and of the alteration sequence of uranyl compounds. This includes work to gather more reliable thermodynamical data on some of the possible secondary phases, in order to use these data in a model of fuel corrosion over long times.

11.2.3 Realistic model of release from the fuel

Radiolysis

As mentioned previously, radiolysis can be a very important process for fuel dissolution, since it can disturb the original reducing conditions in deep groundwaters. Experiments aimed at determining the quantity of oxidants produced by radiolysis are under way and will continue. Data from these experiments will be used in a more elaborated model of radiolytic oxidation/dissolution of the uranium dioxide matrix.

Characterization of spent nuclear fuel

Experiments to determine the quantity of oxidants produced by radiolysis have shown that oxygen seems to be consumed in the system without a corresponding oxidation of uranium dioxide to U(VI) being observed. Attempts will be made to determine how this oxygen is consumed by the fuel matrix. This entails determining the oxygen/metal ratio in the fuel before and after leaching and examination of the fuel with x-ray diffraction and electron microscopy. Extensive development of methods will be required, and the work is foreseen to continue throughout the six-year period.

11.3 BUFFER AND BACKFILL

Functional requirements on buffer and backfill, as well as the present-day state of knowledge and remaining RD&D questions, are presented in section 5.4.

Goals

The goal is to obtain the data needed for safety assessments in support of forthcoming permit applications, for prediction and planning of field tests in the Äspö HRL, and for the process of designing the deep repository that begins when site data from candidate sites become available.

Important interim goals during the period are:

- conclude ongoing compilation of knowledge and experience from previous buffer research,
- devise an improved material model for simulation of the water saturation process in the buffer,
- improve knowledge and modelling of (hydrogen) gas transport through the buffer, especially in connection with repeated cycles of gas release.

Programme

The work is focused on further accumulation and deepening of knowledge of subjects that are essential to the long-term performance of the buffer and on verification of present-day knowledge in large-scale field tests. Further investigations will be made of the water saturation process. Preparations for designing the deep repository will continue through collection and compilation of knowledge for analyses and optimization of the dimensions of the buffer and for choice of bentonite content and aggregate in the backfill material.

Properties of different bentonite materials

The work to compile knowledge and experience from the buffer research into a handbook continues. Part I is finished and concerns fundamental definitions and method descriptions. Part II provides *material descriptions* with a survey of practically important data for buffers and backfill materials and a description of methods for production of such materials in bulk and block form. Methods for quality assurance are presented. Part III will be a compilation of *mathematical models* for predicting transport processes of various kinds and for analyzing rheological phenomena, e.g. creep.

The *microstructural model* for bentonite will be further refined through microscopic studies. The model will be applied to the backfill material for describing the homogeneity of the bentonite phase in the pores in the aggregate material. The model will be further developed to simulate swelling, compression, creep, passage of gas, transport of water and diffusion.

The knowledge of buffer and backfill materials will be used in the process of *designing the deep repository* in order to determine dimensions, methods and quality. Knowledge of certain aspects of the long-term performance of the buffer will be deepened as a basis for optimization. A plan for long-term assurance of the desired bentonite quality will be worked out during the period.

An important factor for optimization is the *durability* of montmorillonite, which is the principal mineral in the buffer and therefore determines its properties. A kinetic model exists for conversion of montmorillonite to illite. The continued work will focus on improving the accuracy of calculations of the conversion rate and describing and evaluating alternative conversion processes and their importance in relation to the original quality of the bentonite. The purpose is to be able to specify in detail the required bentonite quality for different repository conditions.

Investigations of *chemically induced changes* in smectite-rich clays will be conducted by means of laboratory and field tests in parallel with studies of natural analogues. Field tests will be conducted in the Äspö HRL.

Studies of *natural analogues* will continue. Particular attention will be devoted to alteration processes at high pHs, the importance of temperature, and cementation. Parts of the work will be done, as before, in cooperation with different international institutions.

Calculation models

The *material model* and the ABAQUS code, which describes the properties of the buffer and various thermo-hydro-mechanical (THM) processes in the water-saturated state, needs to be revised to include volume creep and plastic, volumetric strains, among other phenomena. These properties will be investigated in long-term laboratory tests. The ABAQUS code will be updated in these areas.

For a period following deposition, the buffer material will be unsaturated until water from the rock has been absorbed and has completely filled out all pores in the buffer. The preliminary material model for simulation of THM processes and *water saturation processes in unsaturated buffer* will be revised and improved to permit evaluation and prediction of the processes in the full-scale buffer tests in the Äspö HRL. Laboratory tests on unsaturated bentonite samples are planned. The results of these, in combination with theoretical studies, will be used for an improved model which will be applied to verification tests in both the laboratory and field environment.

Valuable review and evaluation of different models is obtained through participation in the international co-operation project VALUCLAY (SKB, AECL and PNC), which is aimed at comparing and exchanging information on THM processes in unsaturated clay buffers.

Gas transport

The process of transport through the bentonite buffer of the hydrogen gas that may occur in the event of holes in the copper shell and water entering the canister has been described and a conceptual model of the process has been set up. To better quantify the transport capacity and verify the process over a long duration, laboratory experiments with low gas flows and repeated cycles of penetration and self-healing of the bentonite are planned. The results will be utilized to improve the modelling of the transport capacity of different buffer grades.

Chemistry of buffer and backfill

A programme of measurements of *diffusion* in bentonite is under way within the chemistry programme, and experiments will be conducted with *bacteria* in the bentonite buffer, see section 11.5.

The studies of *concrete and cement* are being conducted with the objective of describing the stability and development of the concrete itself with time, and to predict the possible influence of the concrete on the repository and the materials incorporated in it. Analyses will be made of old concrete, and chemical experiments with concrete are being considered. Additional experiments involving the influence of concrete on the surrounding rock are planned.

11.4 THE BEDROCK

A summary description of the state of knowledge concerning the barrier function of the bedrock at a deep repository is provided in section 5.5. Against this background, the following programme is planned for the period 1996–2001. The programme only includes bedrock-related questions within SKB's supportive R&D activities. The programme for the Äspö HRL is presented separately in chapter 12, and many geochemistry-related questions are dealt with in sections 11.5 and 11.8. Development of investigation methods and instruments is discussed in section 9.4.3.

11.4.1 Structural geology and mechanical stability

Site-specific rock-mechanics questions at a deep repository facility are mainly dealt with at the Äspö HRL. More fundamental strength-related activities and tectonic assessments are included in the supportive R&D activities.

Goals

The aim of the activities is to quantify and explore the consequences of long-term tectonic effects, including earthquakes and glaciation cycles. Important interim goals during the period are thereby to:

- compile the vertical load situations that have prevailed and the consequences these loads have had on the Swedish crystalline basement on the geological time scale,
- compile and statistically process data on displacement sums from discontinuities (fractures/zones) in the Swedish crystalline basement,
- compile knowledge on what seismic effects mean for a deep repository.

Programme

Geological and structural conditions in the bedrock

Determination of the extent and thickness of former sediment layers on the sub-Cambrian peneplain with the aid of fission tracer dating. – The study is intended to provide a basis for discussion of former vertical load situations and their consequences for the Baltic Shield during the Phanerozoic in relation to ice loads existing during the Quaternary period.

Refine methods for identification of *subhorizontal structures in the bedrock* – mainly by improving interpretations of seismic reflection survey methods.

Mechanical properties of the crystalline basement rock

Gain a better understanding of *the representativity of rock stress measurements* by compiling available measurement data.

Perform and interpret *rock stress measurements at depth* down to 1500 m in the KLX 02 borehole at Laxemar in the municipality of Oskarshamn.

Determine the *normal stiffness in the field of fractures*. – The field work will be carried out with slightly modified equipment for rock stress measurements by means of hydraulic fracturing. The method will be tested in the deep hole at Laxemar.

Contribute towards *development of rock mechanics theory* when it comes to *creep movements* of tunnel walls and assessments of *effective stresses* in fractured rock.

Geodynamic and mechanical processes

Gain a better understanding of the fracturing of the crystalline bedrock in a tectonic, historical perspective, by means of, among other things, field studies of maximum displacements for discontinuities (fractures and fracture zones) on different scales.

Explore the potential for *hydraulic fracturing* and/or propagation of existing fractures and fracture zones in conjunction with a *glacial cycle*. – The glaciation model developed by SKB offers a means to evaluate these effects.

Study *aseismic movements* and ongoing *land uplift*, with the aid of recently installed permanent stations in the Swedish GPS network. – The shoreline displacement that is observed is due to isostatic land uplift and eustatic sea level rise.

Gain a better understanding of the causes of the *earthquakes* that occur in the Baltic Shield.

Compile experience of the earthquakes that have occurred at the *Kamaishi Research Mine* in Japan and generalize the effects, if any, on an underground facility. – The work is being done within the framework of a bilateral agreement between SKB and PNC, Japan.

Further refine *methods for dating of previous fracture zone movements.*

Further refine quantitative criteria for *classification of discontinuities* in the bedrock. – The criteria will be applied to determine exclusion distances to repository sections and canister positions. The work should be seen as a continuation of the method developed in the Stripa Project.

Investigate the possible *effect of a deep repository on the strength of the rock* in a regional perspective. – The question concerns whether the deep repository is to be viewed as a horizontal plane of weakness acting in concert with or independent of regional fracture zones. The project is closely linked to the above classification project.

11.4.2 Groundwater chemistry

Site-specific geohydrochemical questions related to a deep repository are mainly being dealt with in the Äspö Project. SKB's supportive R&D activities include fundamental knowledge accumulation on chemical processes.

Goals

The research is aimed at further clarifying the hydrochemical processes that can affect the various barrier functions of the deep repository. Important interim goals during the period are thereby to:

- prepare a final programme for hydrochemistry in connection with the site investigations,
- report on how the K_d concept can be used and how reliable the concept is in calculations of nuclide transport.

Programme

Further refine *mathematical/statistical methods* for being able to describe and predict the hydrochemical conditions regionally, locally and in the repository during different phases of a site evaluation. – Mathematical/statistical methods are used to classify hydrochemical data in the database. At present, conditions on previously investigated sites are predicted and then compared with actually measured results. An effort is also made to utilize available hydrological data.

Determine the technical content and logistics for *hydrochemical investigations* and surveys *in future site investigation programmes.* – Hydrochemical, geological and geohydrological investigations and evaluations together comprise the core of a site characterization. An integrated programme that links these efforts together is described in 9.4. The contents of the hydrochemical programme must be optimized with respect to the stage goals, but also with respect to the logistics of the entire site characterization programme. An important part of the work is to plan the practical execution of the various activities and estimate the time and resources needed for them.

Model and visualize *hydrology-chemistry-isotope data* in a clear and graphic manner. – Coupled modelling of hydro-chemistry-isotope data is being done within the framework of international cooperation on Äspö. Effective systematization and standardization of these activities requires modelling tools and presentations that can be utilized by several different users. The Rock Visualization System (see section 12.4.3) is an excellent basis for further processing and modelling. The tool will be utilized in forthcoming international cooperation on Äspö (see chapter 12) and in future site characterizations.

Investigate and describe *hydrochemical conditions in low-conductivity rock* and refine sampling and analysis techniques. – The hydrochemistry of low-conductivity rock is similar to that of conductive rock, see 5.5.5. This is still a preliminary conclusion that requires more work both on the sampling and analysis side and on the modelling side. The work is aimed at answering questions about the value of equilibrium models in describing the chemistry of the groundwater-rock system.

Describe the *regional hydrochemical conditions around Äspö* during the most recent glaciation. – Regional and palaeohydrogeological studies are planned to be included in the palaeohydrogeological programme and in the international cooperation on Äspö, see section 5.5.8 and chapter 12. Similarities and dissimilarities in water chemistry between the investigated sites will be analyzed.

Conclude the development of the "*in-situ K_d concept*". – The in-situ K_d concept entails that the naturally occurring elements that are analogues to the radionuclides in the spent fuel are analyzed in water and in fracture-filling minerals. The results are used to interpret the transportability of these substances and to calculate an in-situ K_d

value. This value is compared with the result of laboratory measurements. The work has been going on for several years and is linked to how fractures are classified and characterized, see section 12.5.2. The results are evaluated for the purpose of supporting the use of the K_d concept in nuclide transport calculations.

11.4.3 Ability of the rock to limit radionuclide transport

This section mainly deals with regional flow and transport conditions. It also deals with physical and certain chemical processes that are linked to the groundwater-hydraulic system. For nuclide transport, the reader is generally referred to sections 11.5 and 11.8. Site-specific research pertaining to the geohydrology and groundwater transport of solutes is mainly being done at the Äspö HRL and in the Laxemar Project (see chapter 12 and section 5.5.7). Hydrogeological investigation methods are dealt with mainly in chapter 9.

Goals

The purpose of the work is to gain a better understanding of how the groundwater flow is distributed in the bedrock and how solutes are transported, retained and fixed. Important interim goals during the period are thereby to:

- determine how hydraulic tests should be performed and interpreted in coming site investigations,
- devise methods for characterization and measurement of field parameters for nuclide transport and retention in fractured rock.

Programme

Groundwater flow and advection

Further refine in-situ methods for determination and analysis of hydraulic properties of fractured rock. – The methods must take into account both stationary and transient single-hole and cross-hole tests.

Compile available *databases on kinematic porosity and dispersion in fractured rock.*

Further refine methods for determination of *absolute pressure and hydraulic gradients in rock* and their variation with depth below the ground surface. – This is important information for siting a deep repository in relation to the recharge and discharge areas.

Further refine methods for judging possible temperature-induced convection flow in relation to the natural groundwater flow, and then with reference to the salinity of the groundwater.

Process *well data* from the Protogine Zone, the Mylonite zone and any other interpreted super-regional discontinuity, and evaluate these data in relation to wells in the surrounding bedrock. The project is aimed at shedding light on whether there are significant differences between the hydraulic properties in different areas.

Investigate *mixing processes between saline and fresh groundwater* in rock in relation to land uplift and the evolutionary stages of the Baltic Sea. – The purpose of the study is to improve knowledge of groundwater flux in areas below the previous highest coastline. The project is linked to SKB's palaeohydrological programme.

Retardation

Develop borehole-based in-situ methods for determining *parameters for transport of solutes.* – Transport parameters such as K_d, flow-wetted surface area and kinematic porosity are required in safety assessments. Tracer tests in the field, which can determine these parameters for a rock volume on the block scale, are relatively large undertakings. Several boreholes are required, for example. It is desirable to develop tests that can be performed as single-hole tests. A possible point of departure may be combined tracer and hydraulic tests with injection of tracer followed by a pumping phase. The idea should be tested with model simulations and the project should be seen as a complement to the TRUE tracer experiments that will be carried out at the Äspö HRL.

Keep track of the current state of knowledge concerning *geogas transport.* – Besides the migration-oriented aspect, the possibilities of using information on geogas as an indirect sign of conductive fractures or fracture zones should also be considered.

Characterization of fractures with respect to nuclide transport and retardation

Further refine *methods for description* of the geometry of fractures and their hydraulic and retarding properties.

Investigate the *large-scale dependence of hydraulic properties on rock stresses.* – The databases for Äspö and the Laxemar area are probably sufficient to correlate hydraulic information with rock stress measurements.

11.4.4 Modelling tools and model development

Calculation models are primarily being developed within the framework of SKB's work with safety assessments and at the Äspö HRL. Supplementary activities are described below.

Goals

The purpose of the programme presented below is to support the further refinement of the modelling tools and to strengthen the means of testing their validity. Important interim goals during the period are thereby to:

- compile experience and results from the palaeohydrological programme,
- summarize how thermo-hydro-mechanical coupled models can be utilized to describe processes in the near field of the deep repository,
- gather background material for selection of models for groundwater flow and nuclide transport with reference to different scales.

Programme

Refine methods for taking into account the effects of *calculation scales in groundwater models.* – Generalizing groundwater models are based on input data that represent certain measurement volumes. An area can simultaneously be modelled with data representing different information densities and resolutions. Possible effects of this on the modelling results will be examined.

Refine different model concepts for *transport of radionuclides* for handling advective flow and retarding processes in the bedrock.

Develop *rock-mechanics modelling* where greater attention is paid to scale dependence and statistical distribution of pertinent parameters. – The rock-mechanics modelling technique primarily employs a deterministic approach with relatively few input data from direct field measurements. The consequences of changed premises in a rock-mechanics calculation problem are usually ascertained by means of sensitivity tests. If relatively extensive field measurements are carried out on a site, statistical data analyses and stochastic modelling are possible.

Refine *thermo-hydro-mechanically coupled models.* – SKB will continue to be involved in validation and verification of thermo-hydro-mechanical models, for example within the framework of the international programme DECOVALEX II.

Carry out SKB's *palaeohydrogeological programme.* – The glaciation model will be refined and more exhaustive climatic data will be collected data with precipitation figures from interglacial periods. Regional modelling will be exemplified for the Äspö-Laxemar area.

Refine methods for *input data management to stochastic models.*

Refine classification methods for *weighing together the geoscientific factors* that are included in the siting process. – This work is to be regarded as a continuation of the projects that have been carried out with a focus on hydrogeological decision-making theory.

11.5 CHEMISTRY

Goals

The goal of the chemical investigations is to improve the data-base and the understanding of essential processes in a deep repository by

- measuring and updating the compilations of basic chemical data on solubility and complexation of radionuclides inside and outside the repository,
- determining concentrations, stability and mobility of radionuclides in the form of colloids, organic complexes and microbes,
- determining the retention of radionuclides in rock and backfill material due to sorption and diffusion,
- evaluating and compiling the importance of the chemical influence of e.g. microbial processes and cement in a repository.

Experiments will also be carried out with the objective of testing (validating) the models that are used to describe retention, release and transport of radioactive materials from a final repository.

Programme

Solubility, complexation and kinetics

The checking and expansion of the database continues. The plutonium tests are being given priority in the current work. Field tests have been conducted and further experiments are being prepared. Other substances may be tested if safety assessments show a sensitivity to specific uncertainties in the databases.

Organic complexes, colloids and microbes

Developments in the area of *colloids* in deep groundwaters will continue to be followed and the concentrations will be checked by samplings and analyses. The importance of the fact that colloids are generated in the near field and the possibility of interaction between gas bubbles and particles will be further explored.

Developments in the area of *humic and fulvic acids* in deep groundwaters will also continue to be monitored, and natural concentrations will be determined by samplings and analyses. It remains to identify what the other organic substances dissolved in the groundwater consist of. Furthermore, studies are planned of the organic substances that may come from materials in the repository.

The work with *microbes* will continue to include sampling and analysis of deep groundwaters, plus laboratory tests and tests with microbes in repository environments, e.g. in buffer and backfill.

Sorption and diffusion

There are good and useful compilations of parameters that describe sorption and matrix diffusion of radionuclides in rock. However, as methods for sampling and measurement are improved there may be reason to redo or supplement earlier measurements. This need is being examined, but for the time being existing data will be used.

The testing of advanced models for *surface sorption* will continue. The goal is not primarily to replace the use of sorption coefficients (K_d values), but to increase our understanding of the mechanisms for sorption on mineral surfaces.

Models and parameters that are used to calculate *diffusion of radionuclides in the bentonite buffer* are based to a high degree on profile measurements. In many cases this is quite adequate, but not for relatively highly mobile ions such as Cs^+, Sr^{2+} and I^-. Data on these ions will therefore be augmented by measurements of mass diffusion and sorption in compacted bentonite in order to provide a basis for more accurate calculations and better models for diffusion under stationary conditions. The experiments will be evaluated during 1997.

Validation experiments

Experiments will be conducted in the laboratory to test (validate) the models and assumptions that are used in the safety assessment to calculate radionuclide migration in the near field and the rock. Some of this work is also aimed at preparing later in-situ tests in the Äspö HRL.

Attempts will also be made to validate geochemical models. Aqueous solutions with a composition simulating concrete pore water are passed through columns of crushed minerals. The purpose is to test (validate) geochemical calculation models that describe how the rock is affected by concrete over a very long time. The experiments will extend through 1997.

11.6 BIOSPHERE

Goals

The goal of SKB's studies of the behaviour of radioactive materials in the biosphere is to be able to carry out credible consequence calculations in the safety assessments.

Interim goals in this process are:

- to quantify the uncertainties that stem from the fact that the biosphere is constantly changing,
- to compile site-specific evaluations of the potential and limitations of the candidate sites for changes in the biosphere,
- to improve the body of data on which the transport models rest,
- to validate the models by studies of analogous transport processes.

Programme

Evolution of the biosphere

The greatest uncertainty in the biosphere is associated with the natural evolution of the ecosystems during the periods of time for which the deep repository is expected to have a safety function. Examples of processes that will become important in a 1000-year time perspective are:

- eutrophication of lakes and cultivation of the old sediments (also due to post-glacial land uplift),
- erosion of soils by wind and water,
- redistribution of sediments in lakes and watercourses.

Man exploits the ecosystems for purposes such as food production and changes them for purposes such as to increase crop yields. This can also be said to constitute a kind of evolution. This human influence can be of great importance for the consequences of radioactive materials that may reach the biosphere, especially if such phenomena as urbanization, large-scale hydroponics, dam construction or the greenhouse effect are included in the picture.

As the siting process progresses, the work will become site-specific and focus on trying to identify conditions that can limit the range of evolutionary options on the sites selected for site investigations. Examples of such conditions are climate, soils and topography that set natural limits on man's exploitation of nature.

Transport to man through production and distribution of food

Production methods for food comprise one of the factors that influence how different parts of the ecosystems are linked to man's internal environment. An example from today's situation is irrigation of cropland, the scope of which must be changed within a few hundred years due to accumulation of salts etc. in the soils.

Normally, transport in the ecosystems leads to great dilution of nuclides. Some enrichment occurs, but in most cases the cycle time is short, i.e. the enrichment ceases after a year or so and is not significant over a ten-year period.

In order to map these transport pathways, site-specific studies of candidate sites will be done within the framework of the site characterization programme. The studies will be adapted to each site, but will cover the occurrence of natural radioactivity in superficial waters, groundwater conditions, soil types, land use, biota, population, etc.

Acceptance criteria

The choice of radiological acceptance criteria is of great importance for how biosphere analyses are to be carried out. The Swedish authorities are expected to issue regulations during 1995/1996. This work and the work within international bodies such as the IAEA and ICRP will be monitored.

11.7 SAFETY ASSESSMENT METHODS

Safety assessments require both systematic methods, for e.g. checking the coverage of possible scenarios or uncertainties, and valid models for quantification of the essential processes in the repository. A number of methods and models have been developed and applied to carry out assessments of long-term safety in the deep disposal of radioactive waste. They are discussed in chapter 5. The application of these methods/models to a specific repository design and site is exemplified in SR 95 /4-11/.

Standardized methods facilitate the execution of the safety assessments and of the validation. However, since both the information base and the purpose of the safety assessments can vary widely, it is expected that both methods and models will constantly have to be modified or refined.

Planned RD&D activities for the safety assessments are aimed at

- reviewing and improving understanding of important processes in the deep repository and its vicinity, and how these processes are to be modelled in the performance and safety assessments,
- refining the safety assessment's numerical models for quantification of the post-closure performance of the deep repository,
- adapting methodology and models for the assessments to the specific safety assessment to be performed.
- Supportive RD&D activities for the first purpose have been presented in the sections on fuel, buffer, geoscience, chemistry and biosphere above.
- The programme for other supportive RD&D activities during the coming six-year period is discussed below under the headings "Method development" and "Model development".

11.7.1 Method development

Goals

The methods that are used to carry out and report the safety assessments must satisfy the need for systematics in the work and in the documentation, provide an overview and an understanding of how the work has been done, and be able to be further refined and adapted to various purposes with the safety reports.

An interim goal during the period is to carry out appropriate method adaptation so that the safety assessments presented in chapter 10 can be carried out with the necessary quality and traceability.

Checking how the methods have been applied in each individual safety assessment is an essential part of checking the reliability of the safety assessments.

Programme

Scenario methodology

The RES method has been applied to visualize the interactions that are essential to the long-term performance and safety of the repository, to systematize the documentation of phenomena that can be of importance for the performance of the repository, and to provide support in the work of checking how well the selected calculation cases cover the various possible options for the evolution of the deep repository.

Experience from SR 95 and the safety report in support of the permit application for the encapsulation plant (SR-I) will influence the direction of the continued work on the scenario methodology.

A compilation of background material, organized in accordance with the RES matrix, is in progress and will be presented in the documentation for SR-I together with evaluations of selected scenarios. In coming safety reports, these scenarios will gradually be adapted and updated with reference to changed premises and site-specific information.

Uncertainties

A *compilation* of different factors that influence confidence or uncertainties in a safety evaluation is being made in preparation for SR-I. The purpose of the work is to clarify different types and sources of uncertainties to permit a systematic handling of the uncertainties in the safety assessments and to shed light on their relative importance. However, since SR-I is not based on a specific facility site, the discussion will be theoretical in certain respects. The idea is that this compilation of uncertainties will become increasingly concrete as the premises are defined and site-specific information replaces the generic data and as the uncertainties are related to the specific questions different safety assessments are intended to provide answers to.

The work of preparing *validity documents* for the assessment models continues. Specific documents are being prepared for important calculation models. The documents contain i.a. a discussion of area of application, theory, conceptualization, mathematical modelling, verification, supporting experiments and observations.

An essential part of the uncertainties may lie in the *conceptual description* of different phenomena. Different models can quite simply give different results. A check of how big the differences in the results can be is obtained through the Äspö HRLs Task Force, where at present up to 11 groups are applying different models for groundwater movements and nuclide transport to the same examples from the Äspö Project. The work within this Task Force will continue. A working group recently established by NEA will also discuss uncertainties stemming from different conceptualizations.

Quality assurance

Control of traceability and quality in the safety assessments will follow SKBs quality manual and the specific quality manuals being prepared within the project for the encapsulation plant and deep repository.

Questions of specific interest for safety assessments are:

- validation of models used in the safety assessments,
- version management of computer models and input data,
- documentation and traceability of calculations.

The work of establishing QA procedures is in progress.

Reporting

SR 95 constitutes a draft of a standardized template for what safety reports should contain and how the reports should be organized. This template may have to be revised in response to viewpoints from the review of SR 95 and experience from future safety accounts.

Hydrology calculations in three dimensions and stochastic calculations make high demands on good visualization of the results. Efforts are planned to improve the presentation of results in preparation for future safety accounts.

11.7.2 Model development

Goals

The numerical models that are used for the safety assessments and their databases must be kept updated with respect to changes in our understanding of the fundamental processes and developments in the computer field. They must be harmonized with each other so that they can be coupled in calculation chains.

An interim goal during the period is to have a practicable set of models available for the execution of each safety assessment mentioned in chapter 10.

Programme

Numerical coupling of models

Development of a modern menu-based user interface (MONITOR 2000) for the program package for coupled probabilistic calculations (PROPER) is in progress and is estimated to be concluded during 1995. Program maintenance and periodic updating is foreseen during the period.

Models for radionuclide inventory and residual heat

Radionuclide inventory and residual heat have been calculated with the codes CASMO and ORIGEN 2. A summary of the experience gained from completed verifications and validating measurements is planned for the safety report prior to excavation and operation of stage 1 of the deep repository. Some modification of ORIGEN 2 may be required for the safety assessments of other waste.

Models for fuel dissolution/transformation

Fuel dissolution in contact with groundwater has been limited in previous calculations via the solubilities of the radionuclides or, as in SKB 91, been based on measured data from leaching tests and estimated availability of radiolysis oxygen.

During the coming period a kinetic model for fuel dissolution will be developed. It will be based on the experimental data from the fuel studies, see section 11.2.

Chemical modelling in the near field

A realistic description of the chemistry in the near field is important for solubilities and nuclide transport. A model that describes how the components in the buffer material affect the chemistry in and nearest the canister is ready. The work will continue in order to include the canister material and fuel as well. The studies of the influence of cement and concrete will continue.

The EQ3/6 code is used for chemical equilibrium calculations. SKB is not planning any development work of its own for this. Collection and validation of important thermodynamic data is continuing, see section 11.5.

Models for nuclide transport etc. in the near field

SKB is using two different calculation models for nuclide transport in the near field: Tullgarn and NUCTRAN.

Tullgarn is, in principle, fully developed. The development work on NUCTRAN will continue. Important areas for this development are: supplementary work with divided solubilities and calculated canister penetration time for the safety report in support of siting of the encapsulation plant, and adaptation of the model for calculations of nuclide transport from SFR 3-5 for the safety report in support of detailed characterization.

Additional model development is planned for gas transport through bentonite.

Models for transport in the far field

SKB uses a number of different calculation models for groundwater flow and transport of radionuclides.

The general FEM program for groundwater flow and nuclide transport calculations, NAMMU, is expected to be able to be used for large-scale studies in the future as well. Program maintenance and development is being carried out by AEA Technology, and SKB has good opportunities to influence future development.

HYDRASTAR is a tool for stochastic continuum modelling of groundwater movements. It will be further refined, e.g. with regard to documentation, presentation of results and user-friendliness. It should be possible to use steady-state and transient pressure head measurements for systematic calibration of groundwater models. Future work should result in a systematic technique for this with HYDRASTAR. Furthermore, different statistical groundwater models for description of the heterogeneous nature of the rock can be tested to see what influence this has on the final results. Some development work is foreseen to make this possible.

Radionuclide transport in the rock is represented in SKB's model chain by a so-called "streamline" model. Flow paths from different parts of the deep repository are generated by hydrology models.

The coupling between hydrology and transport models is intended to be improved, however. Today's streamline model, FARF31, requires constant parameters for the flow path, e.g. flow-wetted surface area and longitudinal dispersion. The methodology for varying these properties along the streamline will be developed and the database will be expanded.

CHAN3D is a model for calculation of radionuclide transport where the rock is considered to consist of a few water-conducting channels with mixing. The model is intended for calculations on a regional scale. Development of the model is planned during the next few years, aimed mainly at improved handling of fracture zones and dispersion.

PHOENICS/PARTRACK has been utilized by SKB to shed light on the effects of density variations on groundwater movements. Density contrasts arise due to thermal effects or saline groundwater, which is often encountered deep in the bedrock. The method in PARTRACK indirectly describes the processes that influence nuclide transport in fractured rock. Improvements are foreseen when it comes to handling of chain decay and methodology for relating field data for radionuclide retention to parameter values in PARTRACK.

Models for radionuclide dispersal in the biosphere

The transport of radionuclides in the biosphere and consequential doses to man are calculated with BIOPATH, a "compartment model" that can be adapted to arbitrary initial recipients of the deep groundwater.

The calculation model has been developed and maintained by Studsvik EcoSafe, which is also participating in the international comparison and evaluation of biosphere models that is taking place within BIOMOVS. The further refinement of BIOPATH that is foreseen during the coming 6-year period is mainly aimed at adapting and testing the model on the specific biosphere conditions existing on the candidate sites.

11.8 NATURAL ANALOGUES

SKB is giving priority to participation in three major international analogue projects during the next three years: Jordan, Oklo and Palmottu. In addition there will be a few smaller studies with limited aims and scope. The goal of the analogue studies is to gather data to validate (test) assumptions and models for describing long-term processes of importance for the safety of the deep repository.

SKB will continue to participate in the international Natural Analogue Working Group organized by the EU. The results of the various investigations are scrutinized and discussed in NAWG by international experts in the field. In this way, both the quality of the results and their usability in the safety assessment are evaluated.

11.8.1 Jordan

The presence of hyperalkaline springs in Maqarin (active) and in central Jordan (fossil) is being studied as an analogue to concrete in a deep repository. The project is now in its third phase, which is expected to continue until the end of 1996. Phase III is being funded by HMIP (Her Majesty's Inspectorate of Pollution, UK), NAGRA, NIREX, and SKB. The goals of the ongoing third phase are as follows:

- Determine the origin of the groundwater in Western Springs (area in Maqarin) and the chemical composition of the water.
- Petrographical and chemical analysis of textures and mineral phases resulting from reaction of hyperalkaline groundwater with the rock.
- Examine how the hyperalkaline water has affected the accessibility of the host rock to matrix diffusion by radionuclides.
- Test calculation models that couple mass transport with chemical reaction and are used to assess the influence of cement in a repository.

- Test chemical models of solid solutions to describe the solubility of trace elements (Sn, Se, Ni, Pb, Ra, Th and U) in the groundwater at high pH.
- Investigate zeolites from Maqarin to check whether information in the literature on zeolite formation due to hyperalkaline geochemical conditions is accurate.
- Investigate whether the clay minerals at Maqarin and in central Jordan are stable at high pH.
- Investigate whether colloids are formed and whether colloids or natural organic matter transport trace elements.

11.8.2 Oklo

The first phase of the Oklo Project will be concluded and the results reported during 1995. A second phase has been planned that will extend through 1998. The organization has been changed, but as in the case of phase I, funding is being sought from the EU. The French CEA is in charge of both phase I and phase II (but different branches of the organization). SKB also intends to take part in the second phase.

The investigations are being conducted at three locations in Gabon where remains of natural reactors have been found: Oklo (an opencast mine), Okelobondo (an underground mine) and Bangombé (a prospecting area). SKB's interest will continue to be concentrated on the reactor zone in Bangombé.

The overall goal of the EU programme in the field of natural analogues, as well as the goal of other participants, is to test models, calculation programs and databases used in the safety assessment. That is why they wish in phase II to concentrate on quantitative evaluation of processes that influence retention or migration of radionuclides. The project has formulated a number of questions to which answers will be sought:

- Which fission products from the nuclear reaction zones are still present?
- Where in the zone, the near field or the rock have they been retained and in what form?
- Which retention mechanisms have been active and for how long?
- What influence have the geological events had on retention?
- Which transport mechanisms have been active and when?

The term "geological events" refers here to, for example, an intrusion of magma that affected a couple of the zones with a hydrothermal pulse, and further the fact that, for instance, the zone in Bangombé can be reached by oxidative weathering due to the fact that it has come closer to the ground surface with time. Beyond the goal of answering the above questions, phase II of the Oklo Project also has the following goals:

- To develop models for the processes that are important for the function of a repository, i.e. processes that affect materials in the repository or radionuclide transport,
- To test these models on data from the well-characterized natural system, with due consideration given to limitations in time and space.
- To improve our knowledge of the ability of the natural materials to retain radionuclides.
- To improve the databases that are used in the safety assessment.
- To identify slow processes (thousands of years) that could influence the transport of radionuclides.

The focus of phase I was basic and more generally geoscientific, i.e. to investigate and describe the reactor zones discovered at Oklo, to follow the traces of the migration of the fission products from the reactor zones, and to provide a hydrogeological description of the areas as they appear today. The results of phase I form the foundation on which phase II will build.

11.8.3 Palmottu

The GTK in Finland wishes to pursue the continuation of the analogue studies at Palmottu in Finland as an international project. They have therefore expanded participation in the project and are seeking funding from the EU. According to the plans, the new Palmottu Project will begin in 1995 with publication of a final report in 1999. Palmottu has the advantage of offering the same bedrock and other conditions as those generally encountered in Sweden.

The goals of the new Palmottu Project are as follows:

- Provide a quantitative description of the uranium-thorium deposit situated in granitic rock near Palmottu Lake.
- Examine the relative importance of processes that control water flow in crystalline rock.
- Investigate and model the influence of geochemical oxidation and reduction on the mobility of radionuclides in crystalline rock.
- Examine the importance of different mechanisms for retardation of radionuclides.
- Investigate the importance of repeated glaciations (ice ages) on the properties of the rock.
- Use knowledge and data from the studies to develop and refine models used for performance and safety assessment.

11.8.4 Other natural analogues

Other natural phenomena are also being investigated besides those discussed above. There is, for example, natural evidence of retention of caesium. Caesium ions are assumed in the safety assessment to be fairly mobile, but this is contradicted by the results of from tracer experiments. This may be due to the fact that caesium is sensitive to the minerals that constitute the substrate for sorption. For example, clay particles bind caesium virtually irreversibly. Accordingly, it is urgent to find examples of caesium transport in the rock that can lead to a more realistic (less conservative) calculation model.

11.9 OTHER WASTE AND SFR WASTE

All LLW and ILW that will be taken to the deep repository is lumped together in the category "other waste". Aside from the slightly higher content of long-lived isotopes in some of the waste, it has a great deal in common with the SFR waste. The waste consists of filter resins, sludge, scrap and trash that has been packaged in containers of concrete and steel, and when necessary conditioned with concrete. Owing to the similarities, the experience gained from SFR can to a large extent be used for design and performance assessment of SFL 3–5. Supplementary investigations for SFR and new studies for SFL 3–5 can in many cases be conducted in common.

11.9.1 Other waste

Goals

The goal of supportive RD&D in the field of other waste is to prepare for future safety assessments and to test proposals for the design of this part of the deep repository. To achieve these goals, information is required on the quantity and composition of the waste, and on how the waste has been packaged etc.

Interim goals for the period are thereby:

- carry out and report the results of phase II in ongoing prestudies,
- compile the background material for a safety assessment and design work in support of an application for a permit for detailed characterization for a deep repository.

Programme

International cooperation

Long-lived LLW and ILW exist above all in countries where spent fuel is reprocessed, such as France, the UK, Germany, Japan, the USA and Belgium. During reprocessing, some long-lived radionuclides end up in the LLW and ILW. A number of countries therefore have advanced programmes for planning, repository design, research and safety assessment relating to long-lived LLW and ILW. SKB is following this international development work. Informal cooperation has been established with organizations in France (ANDRA), Switzerland (NAGRA) and the UK (NIREX).

Prestudy, phase II

Relying solely on international development work in the field and on experience from SFR is not enough to prepare a safety assessment for other waste. It is necessary to carry out experimental investigations that are not being done elsewhere. There are models that need to be tested and adapted to our conditions. Some of the experiments take a long time and need to be started in good time. This was an important reason for the prestudy that was carried out in 1993–94. The prestudy has guided the planning of further experiments. The goal of phase II is to prepare for a safety assessment that is planned to get started towards the end of 1996.

Phase II started in October 1994 and includes the following parts:

- Compile tables of the radionuclide content and composition of the waste to be used in the safety assessment.
- Compile a chemical database with information on water chemistry (in the repository), composition and chemistry of the concrete, radionuclide sorption, diffusion and solubility, organic complexes and colloids.
- Analyze alternative scenarios (e.g. ice age) and the importance of the water flow in the rock, colloids, microbes and gas generation.
- Compile data on the properties of the barriers, the waste packages, concrete structures, the rock in the near field, and backfill consisting of concrete, bentonite or sand.
- Compare different design alternatives.
- Develop and test transport models.

The final point ties in with the programme for safety assessment of the deep repository for spent fuel. The goal is to use the same calculation models for all parts of the deep repository.

The investigations are dominated by chemical questions, such as the stability of the concrete, the influence of the concrete on the surrounding media, corrosion, organic complexes, solubility, sorption and diffusion of radionuclides, etc. Some of the studies are also intended as supplementary investigations for SFR. This particularly applies to the breakdown of cellulose and the resulting degradation products.

With the exception of the analysis of the importance of hydrogeological conditions that is included in phase II, there is no separate programme for supportive RD&D for SFL 3–5 in the geoscientific field; this is included in the general programme for the deep repository. Some adjustments of calculation models for water flow and transport may possibly be necessary so that analysis of SFL 3–5 is covered.

At present, the plan does not include field studies and in-situ investigations. They will have to wait until a later phase. For the time being, experience from above all SFR is being utilized. Some of the results from the Äspö HRL can also be utilized.

Old concrete and natural analogues where cement-like conditions have prevailed for a long time can yield information on the changes that take place in the concrete. Studies of old concrete based on Portland cement and studies of hyperalkaline geochemical conditions in Jordan are included in SKB's RD&D programme, see section 11.8. There are geochemical models that describe the influence of alkaline water on rock minerals, but they need to be further substantiated. The laboratory tests being conducted by the BGS (British Geological Survey) are a step in the right direction, see section 11.5.

11.9.2 SFR waste

According to the existing operating permit for SFR-1, stage 1, renewed safety assessments have to be carried out every tenth year, as long as the facility is in operation. Furthermore, a renewed safety account has to be submitted in support of an application for a permit for closure of the facility. Another stipulation in the operating permit is that, for as long as the facility is in operation, a special monitoring programme must be maintained for gathering of site-specific information.

Generic knowledge will be obtained from, among other things, the studies and investigations described above that are being or will be conducted for SFL 3–5 and from the international development work that is being followed within the project. The fact that the work with SFL 3–5 is being pursued in close cooperation with SFR ensures that information and knowledge will be exchanged between the two facilities/projects.

In preparation for the future safety assessments and closure of the facility, a special monitoring programme is being conducted for the purpose of gathering site-specific knowledge. The monitoring programme includes registration of groundwater head, rock deformations, water seepage quantities and chemical composition of the groundwater, as well as recurrent inspection of the rock.

In the silo repository, movements of the concrete silo and water absorption in the bentonite fill in the gap between the concrete silo and the rock (swelling pressure) are being monitored. Wetting is also being studied in the upper part of the gap fill for the purpose of ascertaining how much of the bentonite has to be replaced when the top of the silo is sealed. The monitoring programme is subjected to annual review, and the results are applied to the coming year's programme. A total review of the design, scope and relevance of the programme is carried out every three to five years. At these occasions, the knowledge and experience obtained from other projects within SKB as well as from international projects is taken into consideration.

The need for increased knowledge of the long-term properties of certain waste materials has been identified. Current knowledge of the long-term properties of organic matter is limited. This is particularly true of the degradation products of cellulose and its possible complexation with radioactive materials such as Pu. A separate study supported by tests of cellulose degradation is currently being conducted, and the most important degradation product (isosaccharinic acid) has been identified. The long-term properties of this and other substances are being studied, and for the time being the quantity of organic matter in the different repository chambers is being checked and limited administratively.

The management and disposal of large and individual components, mainly scrap metal, is being examined in a separate project, where different waste forms and their distribution between different repositories are being studied.

12 PROGRAMME FOR THE ÄSPÖ HARD ROCK LABORATORY

12.1 INTRODUCTION

The Äspö HRL constitutes an important part of the work of designing a deep repository and developing and testing methods for investigating a suitable site. A proposal to build an underground rock laboratory was put forth in R&D-Programme 86 /12-1/ and was received very positively by the reviewing bodies. In the autumn of 1986, SKB initiated the field work for the siting of the underground 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 laboratory on southern Äspö, about 2 km north of the Oskarshamn Nuclear Power Station. After regulatory review and approval, work on the facility was commenced in the autumn of 1990.

The Äspö HRL has been designed to meet the projected needs of the planned research, development and demonstration activities. The underground part takes the form of a tunnel from the Simpevarp Peninsula to the southern part of the island of Äspö. On Äspö, the tunnel runs in two turns down to a depth of 450 m, see Figure 12-1. The total length of the tunnel is 3,600 m. The last 400 metres were excavated with a tunnel boring machine (TBM) with a diameter of 5 metres. The first part of the tunnel was excavated by drill-and-blast. The underground excavations are connected with the surface facilities by a hoist shaft and two ventilation shafts. On the surface is the Äspö Research Village with offices, stores and hoist and ventilation building /12-2/, see Figure 12-2.

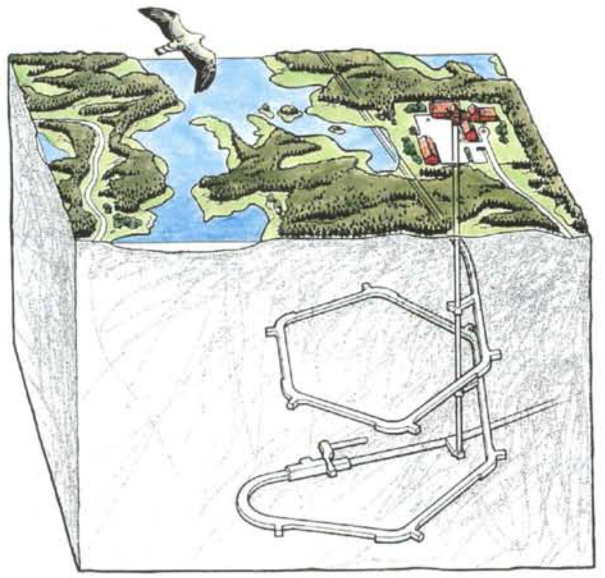


Figure 12-1. General layout of the Äspö HRL.

The work at the Äspö HRL has been divided into three phases: the pre-investigation phase, the construction phase and the operating phase. During the **pre-investigation phase**, 1986–1990, siting of the Äspö HRL took place. The natural conditions in the bedrock were described and predictions were made with respect to the geohydrological and other conditions that would be observed during the construction phase /12-3/. Planning for the construction and operating phases was carried out.

During the **construction phase**, 1990–1995, extensive investigations, tests and experiments were carried out in parallel with the civil engineering activities. The tunnel was excavated to a depth of 450 m and construction of Äspö Research Village was completed. Äspö Research Village was completed and put into use during the summer of 1994. The underground civil engineering works were completed in the summer of 1995.

The **operating phase** began in 1995. This programme describes the investigations, experiments and tests planned to be carried out during the operating phase. A preliminary programme for these studies was presented in a background report to RD&D-Programme 92 /12-4/.

12.2 GOALS

One of the fundamental motives for SKB's decision to build the Äspö HRL was to provide an opportunity for research, development and demonstration in a realistic and undisturbed rock environment down to the depth planned for a future deep repository. In the planning and design of activities to be performed at the Äspö HRL during the operating phase, priority is being given to projects which aim to:

- increase scientific understanding of the deep repository's safety margins,
- develop and test technology which reduces costs and simplifies the repository concept without sacrificing high quality and safety,
- demonstrate technology that will be used for the deposition of spent nuclear fuel and other long-lived waste.

To meet the overall schedule for SKB's RD&D work, the following stage goals were set up for the activities at the Äspö HRL in R&D-Programme 89 /12-5/:

Figure 12-2. Bird's-eye view of Äspö Research Village.

1 Verify pre-investigation methods.
2 Finalize detailed characterization methodology.
3 Test models for groundwater flow and radionuclide migration.
4 Demonstrate construction and handling methods.
5 Test important parts of the repository system.

The start of the operating phase has occasioned a revision and focusing of the goals of the Äspö HRL based on experiences gained to date. For the operating phase, the stage goals are given the following wording:

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.

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

3 Test models for description of the barrier function of the rock
- Refine and test at repository depth methods and models for describing groundwater flow, radionuclide migration and chemical conditions during the repository's operating period and after closure.

4 Demonstrate the technology for and function of 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 and show that high quality can be achieved in the design, construction and operation of system components.

The Äspö HRL comprises an important part of the work being pursued within SKB's RD&D-Programme. The quality standard for the work is very high, and a general ambition is that the Äspö HRL should be developed into an internationally leading centre for research, development and demonstration concerning facilities for the deep disposal of high-level waste.

12.3 RESULTS – CURRENT SITUATION IN RELATION TO THE STAGE GOALS

SKB considers the first stage goal, Verification of pre-investigation methods, to have been fulfilled now that the construction phase has been completed and reports on the investigations and research done in connection therewith have been published. Final reports on the results and experience gained from investigations during the pre-investigation and construction phases, and the comparison of predictions and actual outcomes, will be published during 1995 and the beginning of 1996. A summary of experience to date and preliminary conclusions from this work is presented in section 8.2.2. The results as a whole show that the methods that are available for investigating rock are well-suited to gathering the knowledge and the data on the bedrock that are needed to construct a deep repository and demonstrate that it fulfils the safety requirements.

As far as the second stage goal is concerned, Finalize detailed characterization methodology, a great deal of experience has been gained concerning the application of many different investigation methods under ground, particularly as regards coordination of detailed characterization with tunnel construction. The investigation methods used for detailed characterization in conjunction with construction of the Äspö tunnel will be described in a report similar to the one that summarized the experience from the pre-investigation phase /12-6/. This report gives an account of the experience gained to date for each method used. A programme for detailed characterization on a selected site needs to be devised on the basis of the experience from work done thus far on Äspö in accordance with the guidelines set forth in chapter 9.

Great progress has been made towards the third stage goal, Test models for groundwater flow and radionuclide migration, especially as regards modelling of groundwater flow. Under the auspices of the Äspö HRL, three-dimensional models have been developed that can describe flow of groundwater with varying salinity within a volume sufficiently large to be representative of a deep repository /12-3, 7, 8/. The ability to obtain representative data for describing groundwater flow and radionuclide transport is being studied in an international Task Force, where data from Äspö are being used in several fundamentally different groundwater flow models, see section 5.5.8 for a more detailed account /12-9–17/. In general, good agreement has been obtained so far between the models and reality /12-18/. Continued work is planned within this area, in particular to obtain additional data to test models for transport of radionuclides that react with the minerals in the rock (sorption), resulting in a slower transport of the radionuclides than of the flowing groundwater.

Construction of the Äspö HRL has yielded valuable experience when it comes to the fourth stage goal, Demonstrate construction and handling methods, especially as tunnelling has been carried out using both conventional drill-and-blast technique and boring with a tunnel boring machine (TBM). This provides a basis for selection of excavation method in the deep repository.

12.4 FINALIZING DETAILED CHARACTERIZATION METHODOLOGY, PROGRAMME FOR 1996–2001

12.4.1 General

The purpose of detailed characterization for a deep repository for spent nuclear fuel is to:

- confirm that a suitable repository volume is available,
- provide sufficient data for the safety assessment that is needed to obtain a permit for construction of a deep repository, and
- furnish data so that the repository system can be optimized with respect to engineered barriers and geometric configuration.

Within the framework of the stage goal, detailed characterization of the disturbed zone around blasted and bored tunnels will be carried out, along with development of interactive computer systems for interpretation of measurement data and design of the repository plus instruments for investigations under ground.

12.4.2 ZEDEX – A study of the disturbed zone for blasted and bored tunnel

Background

Excavation of a tunnel or a cavity in the rock changes the properties of the rock in the vicinity of the tunnel, the so-called excavation-disturbed zone (EDZ), which may be of importance for pre-closure conditions in the repository and the long-term performance of the repository. To obtain a better understanding of the properties of the disturbed zone and the importance of different excavation methods, ANDRA, UK Nirex and SKB have joined forces to conduct a joint study of the disturbed zone. The project is called ZEDEX (Zone of Excavation Disturbance EXperiment). The results are expected to be of importance for the choice of excavation method or combination of excavation methods that will be used in the deep repository.

Objectives

The objectives of the ZEDEX Project are:

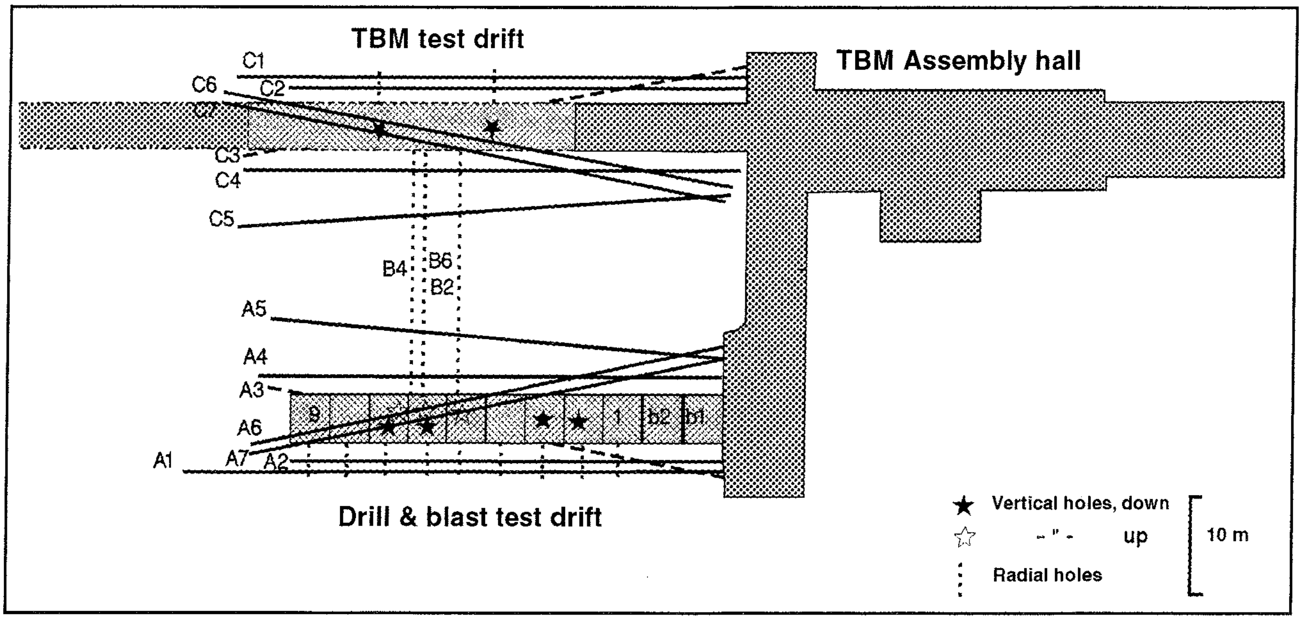


Figure 12-3. Location of the ZEDEX Project's experimental tunnels and exploratory boreholes.

- to obtain a better understanding of the mechanical behaviour of the EDZ with respect to its origin, character, magnitude of property change, geometric extent and dependence on excavation method,
- to perform supporting studies to improve understanding of the hydraulic significance of the disturbed zone, and
- to test equipment and methodology for quantifying the properties of the EDZ.

Execution

The project involves investigation of the rock around two parallel tunnels at a depth of about 420 m. One of the tunnels consists of the first 40 m of the TBM-bored tunnel and the other of an approximately 40 m long blasted tunnel. Both tunnels have a circular profile and are 5 m in diameter.

A number of exploratory boreholes were drilled both along and radially from both of the tunnels, see Figure 12-3. Measurements in the exploratory holes were carried out before, during and/or after excavation of the tunnels. The properties of the disturbed zone were quantified by measurements of:

- acceleration,
- rock movements,
- seismic velocity,
- resistivity,
- observation of natural and induced fractures,
- temperature,
- acoustic emission from microcrack formation,
- rock stresses.

Furthermore, seismic and radar methods were used to describe spatial changes in the properties of the rock. A detailed description of the project is given in /12-19/.

ZEDEX was begun in April 1994 with drilling of exploratory holes and measurements in them. TBM boring through the test area was carried out in June 1994. Investigations for the blasted tunnel were carried out during the period November 1994 to May 1995. These will be followed by evaluation and writing of a final report, which is estimated to be finished at the beginning of 1996. Preliminary results indicate that the impact on the rock caused by TBM boring is very slight /12-2, 20/.

Certain supplementary measurements and a deepened analysis and numerical modelling of obtained results to further increase understanding of the hydraulic and mechanical properties of the disturbed zone are planned during 1996.

12.4.3 Rock Visualization System

Background

A rock model for a site is built up gradually by collection and processing of data. These data will be collected in SKB's geodatabase SICADA (see also 8.2.5). In addition, geological and geophysical map material will be collected in SKB's GIS system. Experience from the investigations at the Äspö HRL shows that it is important during the characterization of a site to have access to an interactive 3D CAD system that enables testing in 3D at different possible connections of observations in bore holes and on the ground of during the work of building a model of the rock's structure. A rapid visualization of the structural model makes it possible to optimize further investigations on the basis of existing information. In the design of a deep repository, the 3D model of the rock will be used to adapt the configuration (layout) of the facility optimally to the geometric structure of the rock. Since several different groups will work with data from a site investigation, it is important to have a common 3D CAD system where all data can be processed and visualized.

To meet these requirements, SKB began development of the interactive 3D CAD tool Rock Visualization System at the end of 1994.

Objectives

The objective is to further develop a commercially available CAD system into a tool

- which can, in a computer model, simulate the principal structure of the rock with respect to rock type units and discontinuities (fracture zones),
- which can be used to represent and adapt the layout of an underground facility to the rock conditions,
- which can simulate results of investigations, existing and planned boreholes and parts of the facility different scales, and
- which enables the computer model to be built up in an efficient and user-friendly way, making use of up-to-date information from the project's database.

Execution

Rock Visualization System is being made as a software application for the commercially available 3D CAD tool INTERGRAPH MicroStation.

12.4.4 Hydrotest equipment for underground measurement

Background

For surface-based measurements, SKB has access to what may be called standard equipment for hydraulic injection and pumping tests. For underground-based measurements, i.e. measurements performed in boreholes from tunnels or shafts, other requirements must be met regarding measurement technology, ruggedness and adaptation to the spatial limitations at measurement sites. During the construction phase of the Äspö HRL, hydraulic tests have been performed with different types of equipment, yielding valuable experience. The critical requirements in underground tests have been linked to high pressures and large water flows, a combination that puts greater demands on equipment the deeper down in the rock the measurement site is located. Experience from work done so far indicates the need for flexible, easy-to-handle and reliable hydrotesting equipment that can quickly be set up at measurement sites under ground.

Objectives

The objective is to develop hydrotesting equipment that can be used for most of the hydraulic tests that will be performed within the framework of the Äspö HRL's different experiments or in a future deep repository.

Execution

A target specification will be prepared based on suggestions and requests from those in charge of future experiments and experience from underground tests performed to date as regards technical and economic constraints.

To achieve stipulated requirements on flexibility and ease of handling, the equipment will in principle consist of modules. Of these, the rig for insertion of different borehole probes will be the most important module. The control and measurement module must be able to handle several different probes for different testing methods. The intention is that the system should also be prepared for future measurement methods and probes. Another important requirement is ruggedness and ability to withstand ambient conditions.

12.4.5 Test and development of investigation methodology for detailed characterization

Background

A programme for detailed characterization will be devised within the framework of the deep repository project. Several different investigation methods have been tested in conjunction with the construction of the Äspö tunnel, and the usefulness of these methods in the detailed characterization for a deep repository is currently being evaluated. The preliminary results from Äspö show that there is a need for further refinement of these methods to improve the quality of collected data, increase the efficiency of the methods and improve their reliability in a demanding underground environment.

Objectives

To test existing and new methods where uncertainty exists regarding the usefulness of the methods for detailed investigations. The methods to be tested are selected based on their potential use within the detailed characterization strategies developed within the deep repository project.

To refine important methods in a detailed characterization programme for improving data quality, efficiency and reliability.

Execution

A programme for testing and refining investigation methods will be devised when the results from the construction phase of the Äspö HRL have been evaluated. In the case of the seismic and radar methods, it may be necessary to evaluate the usefulness of the methods and their dependence on different tunnel and borehole layouts and to test commercial technology that has never before been tried by SKB. It may also be necessary to test new methodology for documentation of tunnels and shafts.

12.5 TEST OF MODELS FOR DESCRIPTION OF THE BARRIER FUNCTION OF THE ROCK, PROGRAMME FOR 1996-2001

12.5.1 General

The rock surrounding the repository constitutes a natural barrier to the release of radionuclides to the biosphere. The most important functions of the rock are to protect the engineered barriers by ensuring long-term stable chemical and mechanical conditions and limit transport of corrodants and radionuclides by means of slow and stable groundwater flux. In safety assessments, the barrier function of the rock is described with different models.

Within the framework of this stage goal, projects will be carried out for the purpose of evaluating the usefulness and reliability of different models and developing and testing methods for determination of parameters included in the models. An important part of this work is being done in the Äspö HRL's international Task Force on Groundwater Flow and Transport of Solutes, see also 5.5.8. In the Task Force, the modelling work is tied to ongoing and coming experiments at the Äspö HRL, which enables the experiments to be planned and evaluated better. The work of the Task Force will continue during the period with specific tasks, where different model groups work with the same set of field data. The results will then be evaluated by participants from the international organizations participating in the Äspö HRL. The lessons that can be drawn for future site investigations and safety assessments will be emphasized in the evaluation.

Studies will also be conducted of the disturbance entailed by the construction and operation of a repository in order to ensure that this disturbance does not have a negative impact on the long-term safety of the repository.

12.5.2 Fracture Classification and Characterization, FCC

Background

The fracture system in the bedrock is of varying origin and appearance. The geological history of the rock has affected the fractures through the action of movements and alterations of fracture-filling minerals. This history has given rise to fractures of varying character with respect to e.g. direction, length, fracture-filling minerals and alteration of the wall rock (see section 5.5). It is probably possible to link the geological character of the fractures to their occurrence and properties as water conductors and transport pathways for radionuclides.

Objectives

The objective of the project is to develop relevant concepts and data on fracture properties that can be handled in radionuclide transport modelling and that are based on a sorting of the fractures into a few relevant classes.

Execution

A preliminary fracture characterization and classification based on geological mapping was carried out during 1994. The results show that a classification into simple and complex fractures can be done and that different properties can be attributed to these classes /12-21/. This is being followed by sampling for quantification of the properties of the fractures by means of laboratory analyses.

The project is now being pursued in cooperation between SKB and NAGRA and the final report will be published during 1996.

A continuation of the project with a focus on studying the natural occurrence of cesium, strontium and other radionuclides in different types of fractures will be considered.

12.5.3 Tracer Retention Understanding Experiments – TRUE

Background

The processes that are of importance for the ability of the rock to limit the transport of various substances are described in section 5.5.6. Most substances dissolved in the groundwater are transported more slowly than the average flow velocity of the groundwater. This is due to a number of different processes that cause retardation of the solutes relative to the flowing groundwater. Important processes are dispersion and retention. Retention (retardation) is caused by the following mechanisms:

- Radionuclides sorb on mineral surfaces past which the groundwater flows.
- Radionuclides diffuse out from water-conducting fractures into the stagnant water in micropores in the rock and are sorbed on the mineral surfaces there.

Non-sorbing substances are also retarded due to the fact that they remain in the stagnant water in the micropores and are in this way kept from being transported in the flowing water in water-conducting fractures. Radionuclide retention according to this pattern is important and is often referred to as "matrix diffusion".

To gain a better understanding of radionuclide retention in the rock and create confidence that the radionuclide transport models that are intended to be used in the licensing of a deep repository for spent fuel are realistic, a programme has been devised for tracer tests on different scales. The programme has been given the name Tracer Retention Understanding Experiments (TRUE) and is described in /12-22/.

The basic idea is to carry out a series of tracer tests of gradually increasing complexity. In principle, each tracer test will include a series of activities starting with geological characterization of the test area, followed by hydraulic and tracer tests, and ending with the injection of epoxy or a similar substances into the test volume, which will then be excavated and the rock samples analyzed with respect to flow paths and tracer concentration. The first tracer test cycle, which will be carried out on a small scale, will be of limited duration and be aimed primarily at technology development. Subsequent tests will have a longer duration to enable different retention mechanisms to be studied. Tracer tests on both the block scale (10–100 m) and detailed scale (1–10 m) will be conducted in the same rock volume. This will provide a basis for understanding scaling ratios and testing models for radionuclide transport on the 50 m scale.

The experimental programme is designed to generate data for conceptual and numerical modelling at regular intervals. Regular evaluation of the test results will provide a basis for planning of subsequent test cycles. This should ensure a close integration between experimental and model work. Detailed plans for the tracer tests will be worked out during the course of the programme.

Objectives

The objectives of the TRUE programme are:

- to deepen knowledge of radionuclide transport and retention in fractured rock,
- to evaluate the usefulness and feasibility of different model concepts,
- to develop and test methods for determining important transport parameters, and
- to determine values of important transport parameters in-situ.

Execution

The TRUE programme has been divided into stages, and a detailed plan has so far been drawn up for the first stage, which is estimated to be completed at the beginning of 1997. The detailed plans for stage 1 are described in a Test Plan /12-23/. The first stage is primarily concentrated on development of technology for tracer tests on a detailed scale and for characterization of the pore volume by injection of epoxy and development of weakly sorbing tracers.

The first stage includes identification and characterization of a fracture or fracture zone at a distance of 10–20 m from an existing drift. After that, tracer tests will be carried out with several different non-sorbing tracers under different boundary conditions, see Figure 12-4. The tests will be conducted in close cooperation with the Äspö HRL Task Force, which will be kept informed of the results of the tests and be given an opportunity to influence the design of the tests. This guarantees the close cooperation between experimentalists and modellers that is needed in order to refine models in this area.

Development of technology for characterization of the pore volume will be conducted in parallel with the field work. One or more pilot tests will be performed before epoxy or another suitable substance is injected into the fracture in which the tracer tests have been conducted. Development of weakly sorbing tracers and testing of their properties on Äspö rock will be carried out in preparation for stage 2.

Planning of the execution of tracer tests on a block scale (10–100 m) will be commenced in 1996. Detailed planning of stage 2 of the detailed tracer tests will be carried out on the basis of the experience gained in stage 1.

12.5.4 REX – Redox Experiment on detailed scale

Background

The groundwater's redox conditions under natural conditions are well known /12-24/. During the tunnel construction and operating phase, the increased water flux can influence the redox conditions beyond what applies under undisturbed conditions. The effect of the rapid transport of oxygenated water down into the rock has been examined in the Block Scale Redox Experiment

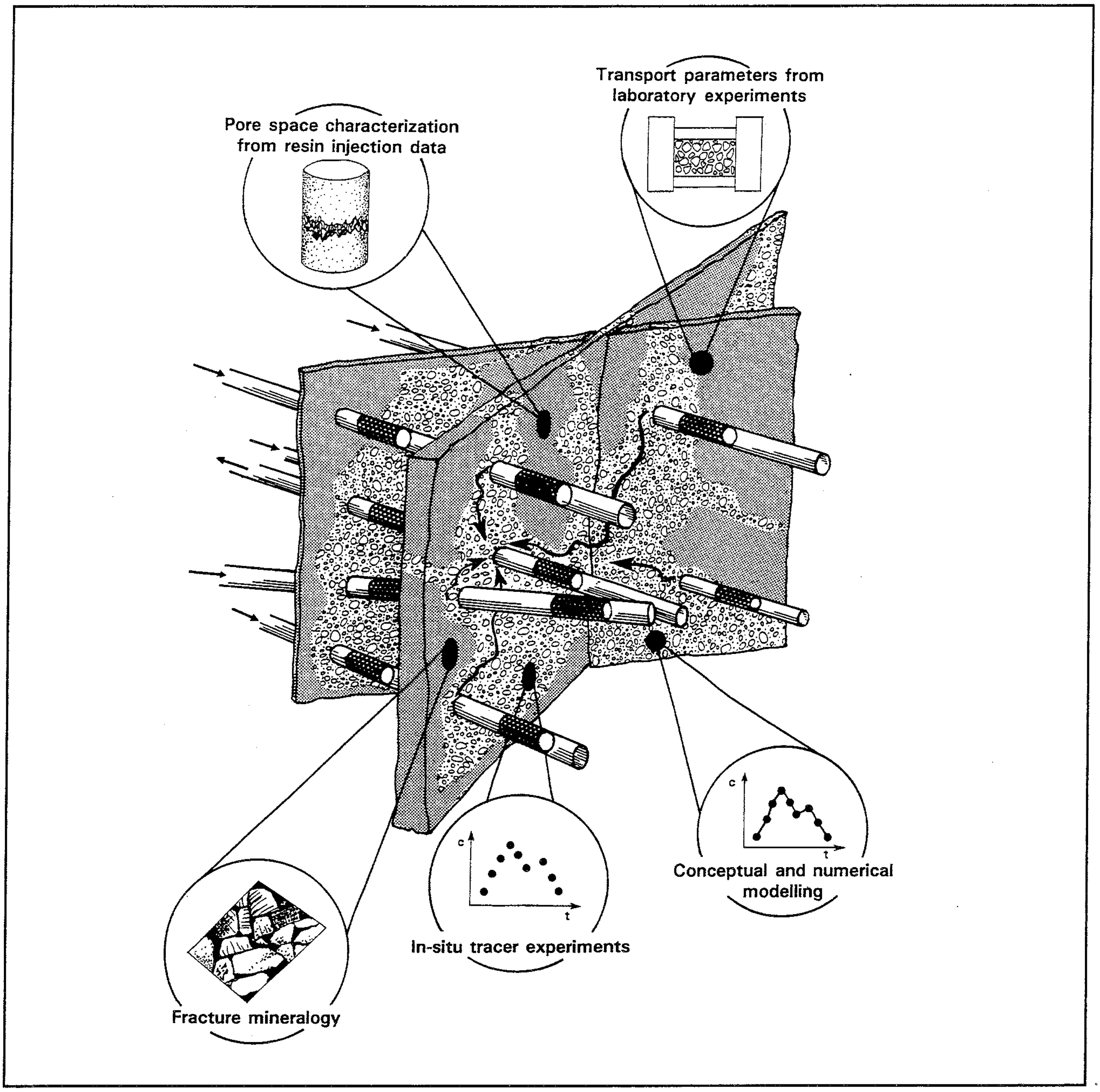


Figure 12-4. *Schematic illustration of TRUE.*

which was conducted between 1991 and 1993 in the Äspö tunnel.

The results show that in this and other cases where the concentration of organic matter in the inflowing surface water exceeds 10 mg/l, bacterial processes will consume the oxygen near the ground surface. An expected oxygen breakthrough at a depth of 70 m failed completely to materialize, even though surface water was transported down at the expected rate /12-25/.

Large quantities of oxygen are trapped in the repository at the time of closure. This oxygen is consumed in different ways via reactions with rock, backfill material and bentonite clay. There is also a hypothetical possibility that the oxygen will attack the copper canister and cause corrosion during a limited period of time. To analyze this potential risk, it is important to be familiar with the reaction kinetics for reduction of oxygen in the rock and in the bentonite.

REX – Redox Experiment on detailed scale – is aimed at clarifying and demonstrating how oxygen is consumed in contact with rock. For the time being, this is an SKB-initiated project where other organizations participating in the Äspö HRL have shown an interest and will probably participate actively.

Objectives

The objectives of the REX Project are:

- to define a half-life for oxygen in contact with rock,
- to deepen knowledge of the mechanisms that give rise to reduction of oxygen,
- to quantify the reaction rate for reduction of oxygen in contact with rock minerals in water-conducting fractures.

Execution

The work will begin with laboratory tests where oxygenated water is allowed to react with rock material. Both static tests and column experiments will be carried out. The results of the laboratory tests will determine the parameters of the field tests that will be performed in the Äspö HRL. Before the work in the field is begun, the experimental set-up for the field test will be tried out in the laboratory.

12.5.5 Radionuclide retention

Background

Most radionuclides have a strong affinity for adhering to different surfaces, i.e. a high K_d value. Numerical values that can be used in the safety assessments have been arrived at via laboratory measurements. However, it is difficult in the laboratory to simulate the natural groundwater conditions in the rock when it comes to redox status and concentrations of colloids, dissolved gases and organic matter. Consequently, it is difficult to obtain reliable values in the laboratory for e.g. dissolution or retention of radionuclides in the cases where they are strongly dependent on the properties of the groundwater.

Objectives

The objectives of the investigations are to:

- validate models and check constants used to describe radionuclide dissolution in groundwater, the influence of radiolysis, fuel corrosion, sorption on mineral surfaces, diffusion in the rock matrix, diffusion in backfill material, transport out of a damaged canister and transport in an individual fracture.
- 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.

Execution

With the aid of a borehole probe designed to be used for different kinds of retention measurements, data can be obtained for the in-situ properties of the groundwater in the rock. The CHEMLAB probe is currently being manufactured and will be put into use in 1996. The principles of the design of the probe are shown in Figure 12-5. It is estimated that different types of measurements will be made for more than 5 years to come. Before the in-situ experiments are conducted in the CHEMLAB probe, the experimental set-ups will first be tested in a normal laboratory. A detailed programme for the experiments to be performed will gradually be worked out starting at the end of 1995.

Experiments planned for the CHEMLAB probe include radionuclide diffusion in bentonite and in rock, migration of redox-sensitive nuclides, solubility, desorption, migration from buffer to rock (film resistance), radiolysis and dissolution of spent fuel.

Diffusion in bentonite and concrete is planned for ^{85}Sr, ^{134}Cs, ^{125}I in the same run and ^{99}Tc in a separate experiment. Standard Portland cement and bentonite MX 80 will be used in the diffusion cells. This experiment is estimated to take nine months to conduct.

Migration of redox-sensitive nuclides, Tc(VII) and U(VI), is planned to be carried out in a natural overcored fracture. The experiment, combined with determination of solubilities for Tc(IV) and U(IV), is estimated to take ten months.

Desorption of radionuclides, Tc, Sr and Cs, is planned for determination of reversibility for radionuclide sorption on rock surfaces. The experiment is estimated to take more than six months to carry out.

The film resistance experiment, a study of the transport of nuclides from the bentonite barrier out to a water-conducting fracture, is planned with Sr, Cs and I as nuclides. It is estimated that the experiment will take five months.

A radiolysis experiment is planned with Tc and is estimated to take more than a year.

Experiments with spent fuel will require careful radiological monitoring of both execution and associated transport. The exact duration cannot be estimated until these questions have been explored. It is reasonable to assume an experimental period of 3–5 years.

The sequence of these experiments has not been determined.

12.5.6 Hydrochemistry modelling

Background

Combining chemistry models with hydrology models is a task that can be carried out in different ways. Ongoing international work within the framework of the Äspö Geochemical Modelling Project aims at checking the

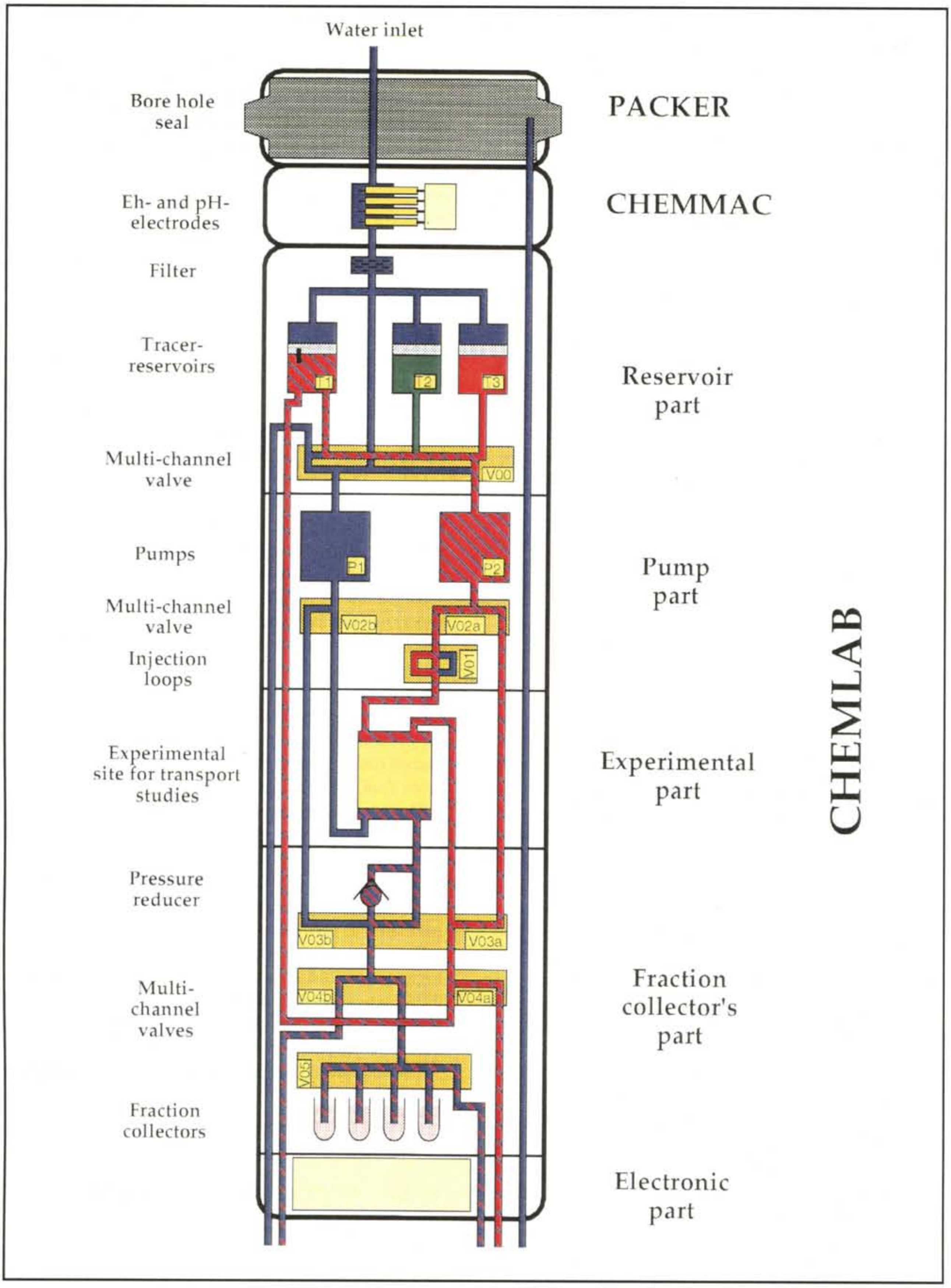


Figure 12-5. Schematic illustration of the CHEMLAB probe.

consistency of hydrogeological groundwater flow models and hydrogeochemical mixing models. The work also includes clarifying the chemical processes that are of importance for palaeohydrogeological evaluations and predictions of future groundwater conditions.

The most compelling reason to perform the palaeohydrogeological modellings on Äspö data is that the groundwater exhibits such a clear difference in salinity and isotope pattern. The reason is that Äspö lay below the surface of the sea from the end of the most recent ice age, about 12,000 years ago, until about 3,000 years ago. During all of that time, no topographically induced groundwater flows changed the chemistry of the rock, so that the conditions that prevailed at the time the ice sheet retreated have been preserved. The only noticeable effect is that salt water has infiltrated into the conductive zones.

Objectives

The overall objective is to obtain a better understanding of the changes in prevailing groundwater systems that might occur in the future and to develop methodology for integrated hydrochemical modelling.

Execution

The modelling work is being carried out by model groups who work for the organizations that are participating in the Äspö Project. The work is being done on investigation data that have mainly been collected during the pre-investigation phase. Reports have been submitted at work meetings in June 1994 /12-26/ and 1995. The work between 1994 and 1995 has been carried out in accordance with two plans published together with a palaeohydrogeological programme in SKB's framework for regional studies /12-27/. A proposal for a concluding modelling exercise is to carry out performance assessment on a hypothetical repository on Äspö with a focus on the stability of the chemical conditions and the geochemical conditions that influence the transport of the weakly sorbing nuclides in particular.

12.5.7 Degassing of groundwater and two-phase flow

Background

Unsaturated or two-phase flow, i.e. simultaneous flow of gas and water, can occur in a deep repository due to 1) release of dissolved gases occurring naturally in the groundwater at the low pressures that occur in the vicinity of drained tunnels, 2) penetration of gas (air) into the rock from ventilated tunnels or from buffer and backfill material, 3) penetration of gas generated due to corrosion or biological processes, and 4) radioactive gases formed by radioactive decay of the spent fuel (see also section 5.4.5 and 5.5.6).

Knowledge of two-phase flow in fractured rock is essential to understand the observations of hydraulic properties that are made from tunnels, the interpretation of experiments conducted near drifts and the performance of the buffer and backfill material, especially in conjunction with backfilling and closure of the deep repository.

Objectives

The objectives of the investigations of two-phase flow in the rock and the importance of degassing of the groundwater are:

- To show to what extent degassing of groundwater at low pressures influences measurements of hydraulic properties in tunnels and boreholes situated under ground,
- To study and quantify other processes that give rise to two-phase flow near tunnels or rock caverns, such as air invasion into the rock and evaporation,
- To show under what conditions two-phase flow will occur and determine how significant it is, e.g. the gas content of the groundwater and the properties of the fractures.
- To get a measure of the time scale required for resaturation of the deep repository after closure.
- To develop equipment for in-situ measurement of water saturation.

Execution

Changes in the hydraulic properties of the rock due to degassing will be observed by measuring the inflow to a borehole at different pressures. Degassing and resultant two-phase flow will show themselves as non-linearities in the saturated relationship between pressure and flow from the borehole, see Figure 12-6. Gas will then be injected into the borehole to simulate gas transport from waste that generates gas as it corrodes. The tests in the Äspö HRL will be complemented by relatively extensive laboratory tests. A detailed Test Plan has been worked out for the project /12-28/.

The results of an initial pilot test have been reported /12-29/.

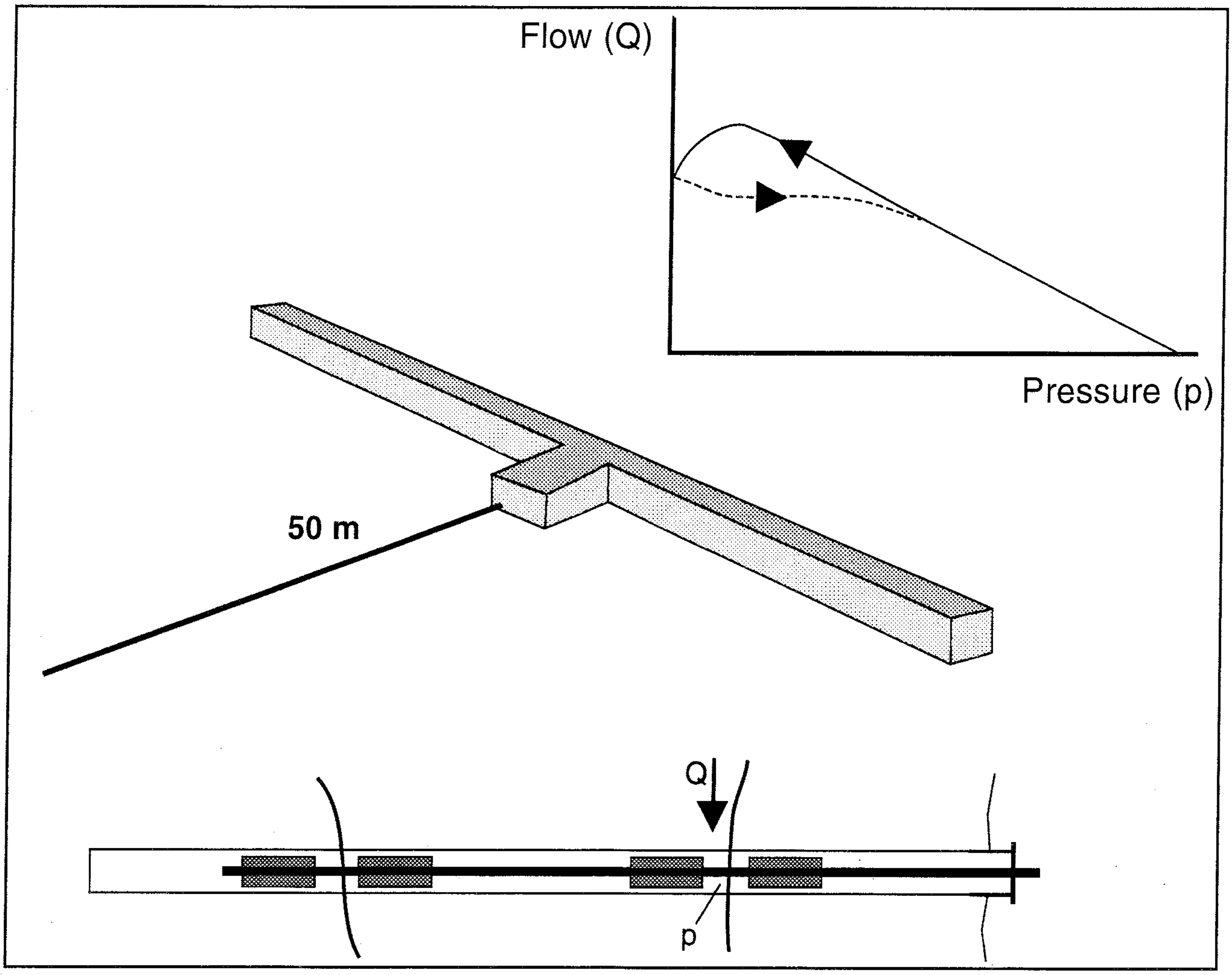


Figure 12-6. Principle for study of the influence of degassing of groundwater in boreholes. The graph indicates expected results if the hypothesis is correct.

12.6 DEMONSTRATE TECHNOLOGY FOR AND FUNCTION OF IMPORTANT PARTS OF THE REPOSITORY SYSTEM, PROGRAMME FOR 1996–2001

12.6.1 General

The Äspö Hard Rock Laboratory provides an opportunity to test, investigate and demonstrate on full scale various components of the deep repository system that are of importance for long-term safety. It is also important to show that high quality can be achieved in design, construction and operation of a deep repository. Within the framework of the stage goal, a full-scale prototype of the deep repository will be built to simulate all steps in the deposition sequence. As a preparation, different backfill materials and methods for backfilling of tunnels will be tested. In addition, detailed investigations of the interaction between the engineered barriers and the rock will be carried out, in some cases over long periods of time.

12.6.2 Testing of different backfill materials

Background

A programme for construction of a prototype repository on Äspö was drawn up in 1994. The programme was published as a Progress Report /12-30/ in the Äspö HRL report series. The programme contains three main parts:

- testing of different materials for tunnel backfill,
- construction of a prototype repository at the end of the TBM tunnel, and
- testing of buffer material under different extreme conditions.

The three main parts of the programme will be conducted in the form of separate projects.

It has been assumed in the PLAN work and safety assessments that material for tunnel backfill will consist of a mixture of quartz sand ("Bornholm sand") and bentonite that is deposited in layers and compacted with vibratory equipment. At the top against the roof, the mixture is simply blown in, which however results in a lower density in that part of the backfill. The possibility of replacing quartz sand with crushed rock from the excavation process has been analyzed /12-31/. The conclusion is that the necessary functional requirements (see section 5.4) can also be met with excavated rock as the aggregate material. In another study /12-32/, the conclusion is drawn that backfill material consisting solely of sand or crushed rock without bentonite meets the requires on safety against radionuclide migration if the tunnel does not intersect major fracture zones and the water flow through the tunnel is small. Tests will be performed with different backfill materials in order to confirm the models for the hydraulic and mechanical properties of the backfill material in a deep repository on a full scale. In conjunction with the testing of different backfill materials, a method for compacting and retrieving (digging out) backfill material in tunnels will also be developed and tested.

In the planned deep repository, deposition of waste will proceed one tunnel at a time. When deposition has been completed along a tunnel and the tunnel has been backfilled, it must be sealed with a temporary plug. Tests with different backfill materials will also include development and testing of methods for emplacement, design and function of a tunnel plug.

Objectives

The objectives of the Backfill and Plug Test are to:

- test different tunnel backfill materials,
- develop and test technology for backfilling of tunnels, especially in the upper part of the tunnels,
- develop and test technology for retrieval of tunnel backfill,
- test the interaction of the backfill and the near rock in a blasted tunnel,
- gather data as a basis for choice of backfill material in the prototype repository test,
- develop and test technology for emplacement and design of tunnel plugs,
- test the mechanical and hydraulic function of a type of tunnel plug,
- improve knowledge of the hydraulic properties of the excavation-disturbed zone (EDZ).

Execution

The test is planned to be conducted in the drill-and-blasted ZEDEX tunnel, which will be filled with different types of compacted backfill material. The inner part will be filled with TBM muck without bentonite. Both existing TBM muck and TBM muck crushed to a suitable particle size will be used. The outer part of the drift will be backfilled with TBM muck mixed with bentonite. Different methods will be used to compact the backfill material in the top and bottom parts of the tunnel. A plug will be constructed at the tunnel opening to seal off the backfill material, see Figure 12-7.

The backfill material and the near rock will be instrumented with sensors for measurement of water pressure, total pressure, movements and water saturation. The axial hydraulic conductivity of the backfill material and the near rock and the mechanical properties of the backfill will be measured. The hydraulic function of the plug will be tested.

Various tests and technology developments are necessary in preparation for the full-scale tests of buffer and backfill material. The following work was begun in 1995:

1 Development and testing of technology for emplacing, compacting and digging out backfill material in tunnels.
2 Laboratory testing of the compaction properties of the backfill materials and their hydraulic and mechanical properties after compaction.
3 Development and testing of field measurement instruments and sampling technology for buffer and backfill material.

Details of the initial tests are described in a Test Plan /12-33/.

12.6.3 Prototype repository

Background

In preparation for the construction of a deep repository it is important to have an opportunity to test as many of the components of the system as possible on a full scale before deposition of spent fuel takes place. In this context it is of particular importance to test the interaction between the engineered barriers and the rock in a realistic environment. This is planned to be done in a prototype repository situated at the end of the TBM-bored tunnel in the Äspö HRL.

Design, construction and testing of the prototype repository on Äspö aim at simulating all steps in the deposition sequence from detailed characterization of the host rock to water saturation of the backfilled deposition holes and tunnel. The prototype repository will not simulate handling and deposition of spent fuel. Instead, copper canisters with electric heaters will be used. A more detailed description is provided in /12-30/.

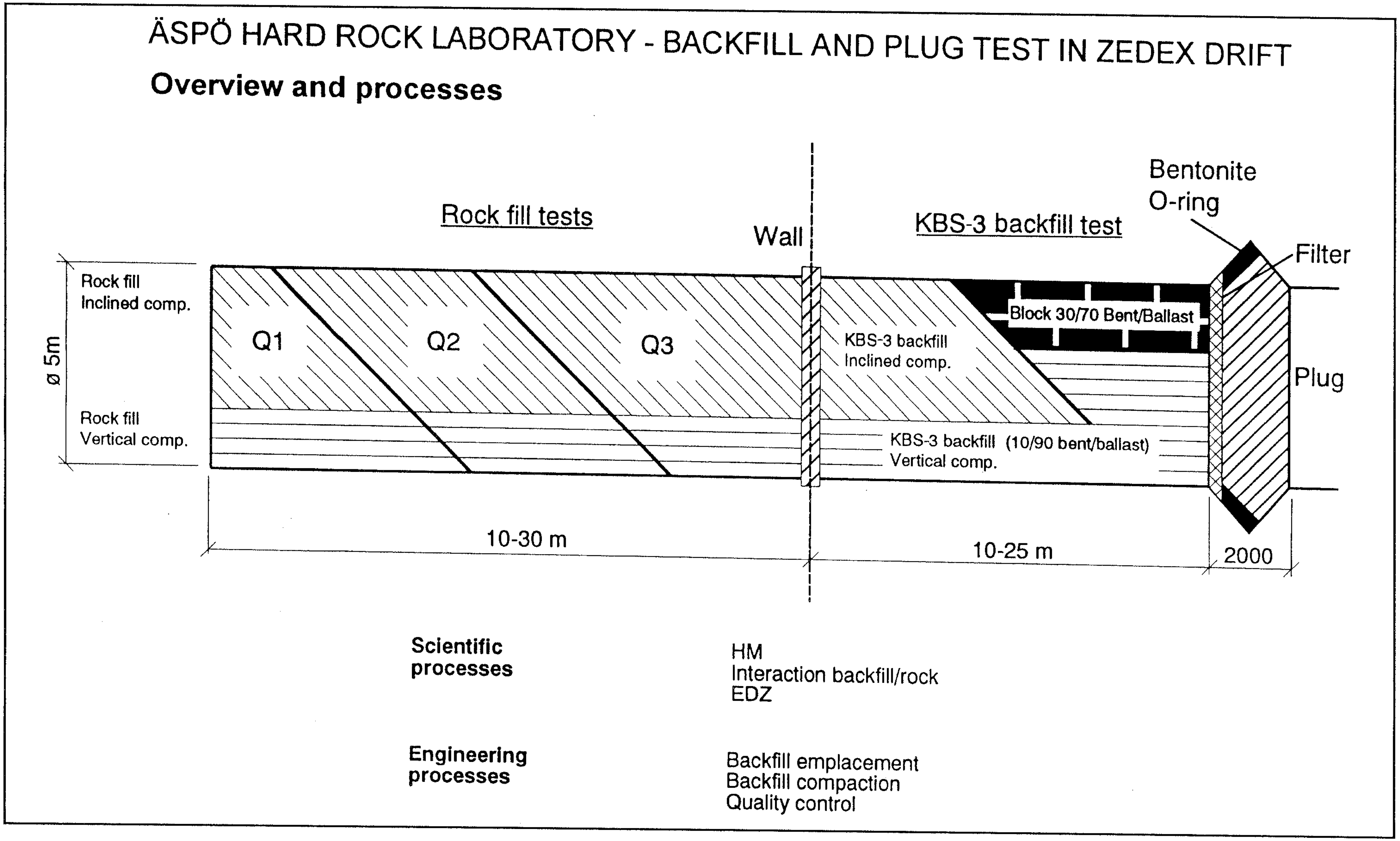


Figure 12-7. Overview of the test configuration for the backfill and plug test and the processes to be studied.

Objectives

The objectives of the project are:

- to translate scientific knowledge and state-of-the-art technology into engineering practice that can be applied in a real repository.
- to test and demonstrate the integration of the different steps in the deposition sequence in a realistic environment.
- to show that the deposition sequence can be carried out with satisfactory quality in relation to relevant criteria.
- to develop and test appropriate criteria and quality systems.
- to test and demonstrate the integrated performance of the prototype repository.
- to demonstrate methods for design, construction, boring of deposition holes, characterization of the near field, backfilling, plugging, monitoring and retrievability.

Despite a relatively long project time (10–15 years), the prototype repository cannot be used to demonstrate the long-term safety of a deep repository.

Execution

The basic idea of the project is to design, construct and test the function of a prototype repository on Äspö. This will be a dress rehearsal for the construction of the deep repository. The project will include characterization of the rock in the near field, design of repository components, boring of deposition holes, deposition of canister dummies with electric heaters, closure, water saturation of bentonite in the deposition holes and the backfill material in the tunnel, and digging-out and retrieval of the canister dummies. The project will include performance and safety assessments in connection with different phases in the deposition sequence.

The prototype repository will be built in the inner part of the TBM-bored tunnel at a depth of 450 m in the Äspö HRL. The preliminary plans call for 4 full-scale deposition holes spaced at a distance of about 6 m, which is the same as planned in the KBS-3 concept. The proposed configuration is shown in Figure 12-8. The TBM tunnel above the two inner deposition holes will be completely backfilled. The results of the test with different backfill materials will be used as a basis for choice of backfill material in this test. A tunnel plug will separate this part from the outer part of the test tunnel in which a steel tube with an inside diameter of 2.5 m will be fitted. The steel tube will be held in place by the three tunnel plugs shown in Figure 12-8. The tube and the plugs will be designed to withstand the combination of full water pressure and swelling pressure from the buffer and backfill material. The steel tube provides space for cabling and instrumentation, permits sampling of buffer and backfill material, permits measurements of deformations in the vicinity of the deposition holes, and reduces the quantity of backfill material and the time required to achieve water saturation.

12.6.4 Long-term test of buffer material performance

Background

The performance of the buffer material was previously tested at Stripa during a time span of up to 6 years /12-34/. These tests were done in water with low salinities, under relatively low water pressures and moderate temperatures (about 80°C). One of the results was that an expected elevated chloride concentration was found in the inner part of the buffer nearest the heater. (The increase in chloride concentration was, however, too small and irregular to be able to be definitely interpreted as an enrichment phenomenon.) A test with French clay was conducted at a temperature of about 170°C, which resulted in a severe deterioration in the buffer properties of the bentonite nearest the heater as a result of a combination of salt enrichment, dissolution of quartz in the bentonite material and cementation (formation of "clay stone"). Different conceivable mineralogical and chemical processes have been studied in laboratory tests. The results show that the negative effects on buffer performance are negligible if the buffer is designed with the density (about 2.0 g/cm^3 after water saturation) that is foreseen in KBS-3, where bentonite blocks are pre-compacted with a high degree of water saturation and the temperature is kept to a moderate level (below 100°C). The current state of knowledge is described in section 5.4. It is essential to verify the laboratory results and models developed under realistic deep repository conditions over long spans of time, possibly up to 20 years.

Objectives

To test the performance of the bentonite buffer in the deep repository environment during a long period of time (possibly up to 20 years).

To test models and confirm results from laboratory experiments concerning conversion of smectite to illite, salt enrichment and influence of high pH.

To rule out the occurrence of unidentified but possible processes in the deep repository environment.

Execution

The long-term tests of bentonite performance are planned to be performed in 4–7 m deep boreholes with a diameter of about 300 mm. Compacted bentonite blocks

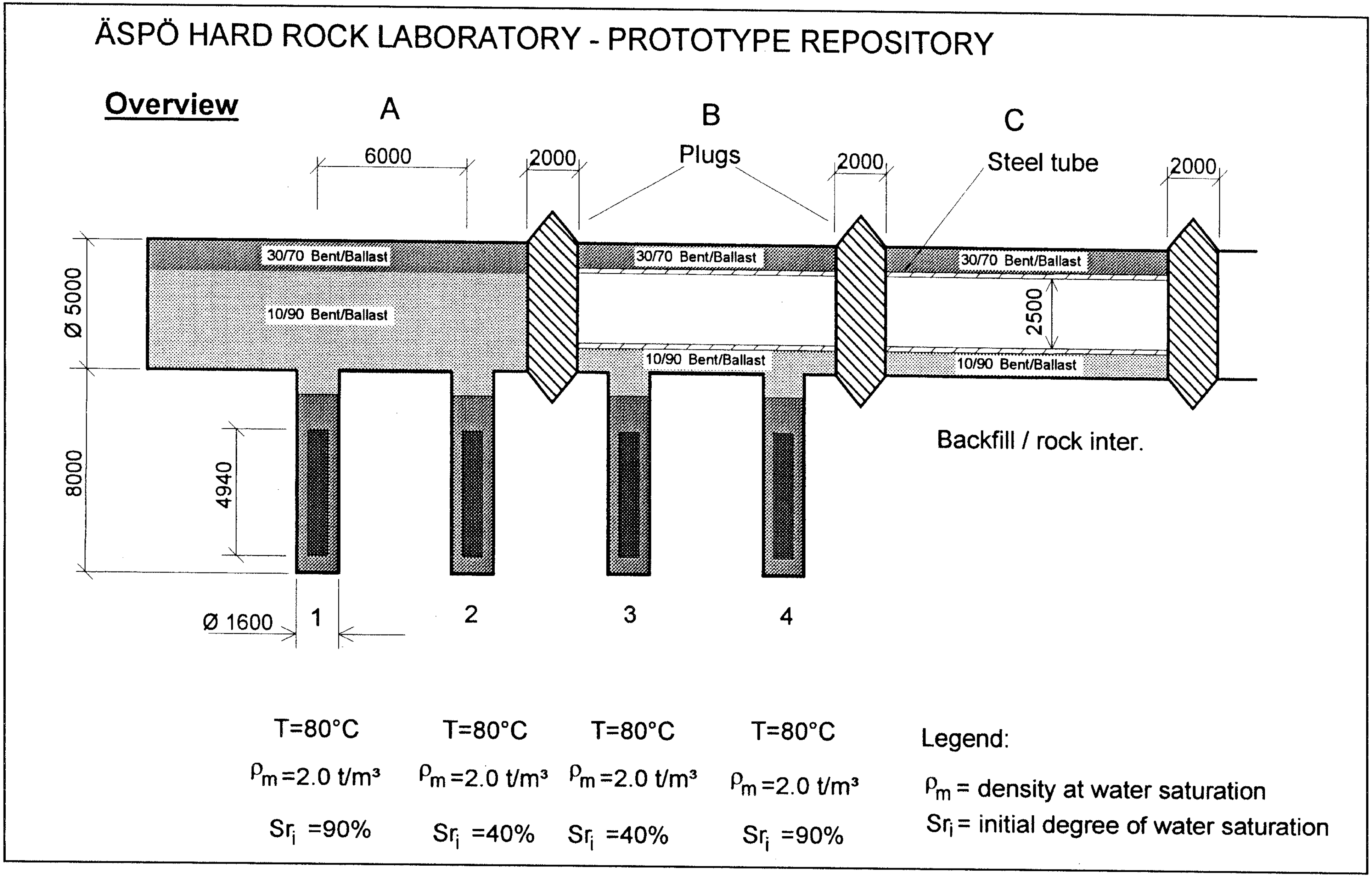


***Figure 12-8.** Preliminary design of Äspö prototype repository.*

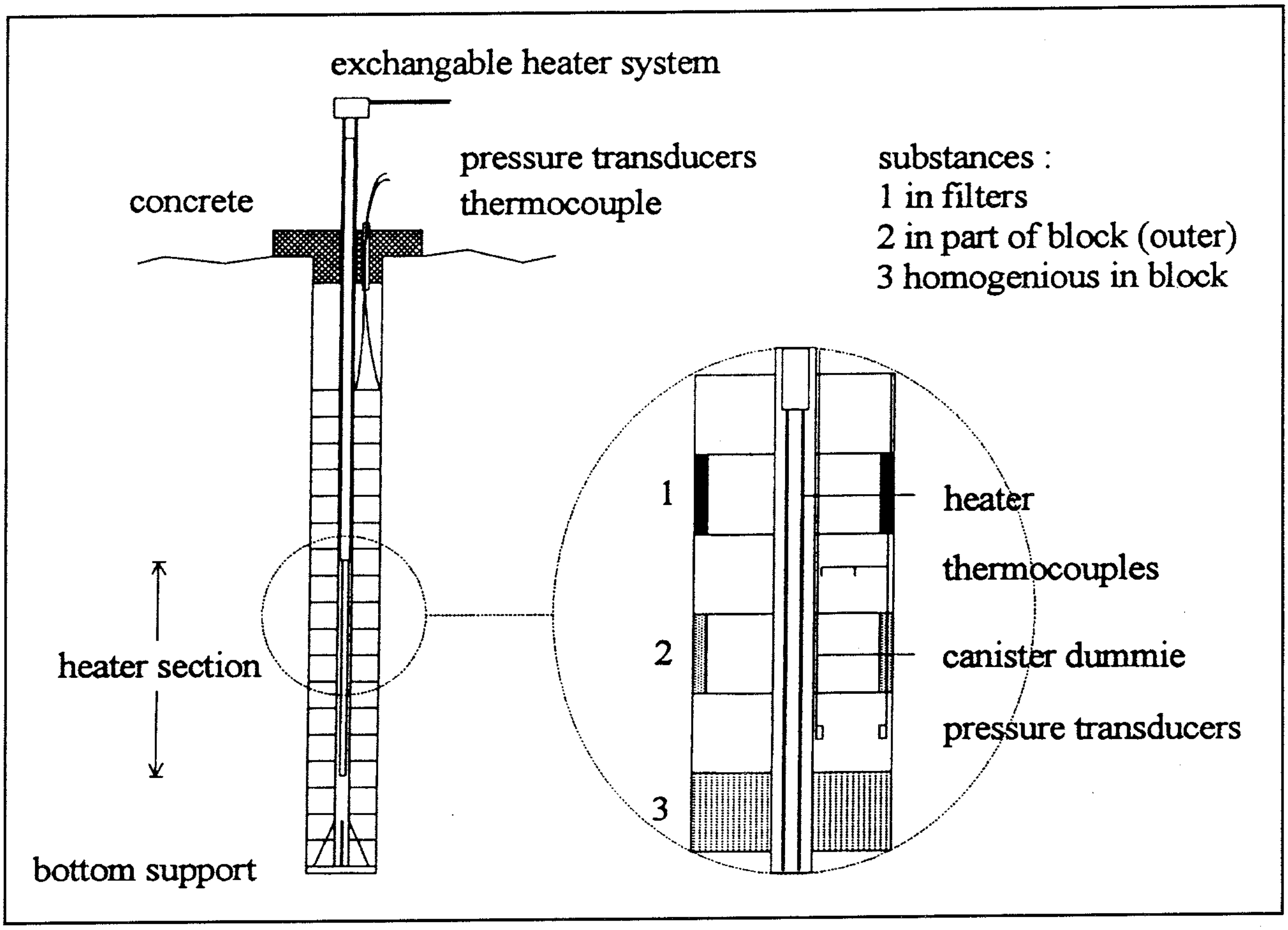


Figure 12-9. Preliminary design of long-term tests of bentonite performance.

with a heater in the middle are lowered into the holes, see Figure 12-9. The holes are instrumented for measurement of temperature, pressure and thermal conductivity (water content) during the duration of the test. To test the buffer under conditions representative for the reference concept, the temperature is driven to a maximum level of about 100°C and the holes are filled with MX-80 bentonite with a high degree of water saturation. For testing under deviant conditions, tests are combined with salt enrichment, cementation and mineral alteration in the same hole. Since these processes are temperature-dependent, the tests are conducted at temperatures higher than 100°C. Tests with high temperature alone are conducted separately.

In an initial phase, a hole in accordance with the reference concept is instrumented and measurements of buffer performance in the hole are performed during 1 year. Then the buffer is taken up by overcoring of the test hole. Samples are taken and sent to the laboratory for quantification of parameters of interest, such as water saturation, chemical composition of buffer and pore water, mineral composition, swelling pressure, hydraulic conductivity and shear strength. This hole is utilized as a pilot test for testing of instrumentation and experimental concept before the long-term tests are begun.

It is estimated that the long-term tests will be done in a total of 6 holes, where the tests will be interrupted progressively after 3, 5, 10 and max. 20 years, respectively. When the test in a hole is interrupted, the buffer is taken up by overcoring and sampled in a similar manner to that done for the pilot hole.

12.6.5 Fracturing during tunnelling by TBM

Background

Mechanical mining methods, such as TBM boring, cause much less damage to the rock wall than drill-and-blast. This is because mechanical mining methods create fractures with less penetration into the rock wall than blasting. Laboratory tests have shown that the extent and shape of the fractures can systematically be related to the properties of the rock and the machine parameters. Fracturing in the tunnel wall or in the walls of the deposition holes is of importance for the hydraulic properties in the near field of the canister.

Objectives

To develop a model for fracturing in crystalline rock in connection with mechanical mining to be able to describe the properties of the rock nearest the opening.

Execution

Based on results of previous laboratory tests with "indenters", a conceptual model for fracturing during mechanical mining has been presented. This model will be further refined on the basis of verifying studies with laboratory set-ups and studies of samples from the TBM-bored part of the Äspö tunnel. Representative drill core samples will be selected for detailed studies of the micro-fissure system created by the steel cutters and grippers on the TBM.

12.6.6 Location of suitable near fields

Background

A suitable rock environment for deposition of canisters is one where the near field is mechanically stable, the chemical conditions are reducing and the water flux is low.

Site investigations for a repository aim at finding a suitable location for a disposal site. The subsequent detailed characterization is intended to confirm that a repository volume is available. As the repository is excavated, canister positions can be accepted or rejected, one by one. A methodology for this is being tested at the Äspö HRL and is expected to provide a basis for planning, designing and constructing a deep repository.

Objective

To develop a statistical methodology for estimating the portion of the length of the deposition tunnels that can be used for emplacement of canisters, based on geological, rock-mechanical, hydrogeological and other information from different investigation phases.

Execution

The first phase included estimating the number of suitable canister positions in a given rock volume that meet requirements on mechanical stability, chemically reducing environment and low water flux. Calculations have been carried out with a Markov-Bayes geostatistical model developed by SKB /12-35/.

The calculations in the first phase were based on data from investigations on the ground surface and in boreholes from the ground surface. In the next phase, data from detailed characterization under ground will be incorporated in the calculations and the calculation methodology will be refined.

12.6.7 Test of grouting methodology

Background

The programme for development of grouting technology is presented in section 9.6 and is aimed at gaining additional knowledge of processes and factors that affect the consumption of grouting material and the grouting result obtained, plus development of a practical technique for dealing with conditions that can arise in the deep repository and that cannot be solved in a reliable manner with the commercial methods available today. Know-how and technology need to be verified by tests under realistic conditions.

Objective

To verify know-how and technology for grouting/reinforcement of large transmissive discontinuities and strongly water-conducting discontinuities of moderate thickness and extent.

Execution

For characterization of rock from the grouting viewpoint, development of a purpose-suited hydraulic testing method is planned. Equipment will be tested in the Äspö HRL under well-controlled forms, and a study of the accuracy of the characterization will be carried out by grouting of boreholes.

The stabilizing and sealing effect of the pre-grouting will be studied by investigation of suitable sections in the Äspö tunnel where only pre-grouting has been carried out.

Field tests concerned with transport mechanisms in fractured rock with varying fracture apertures will eventually be carried out, but not until a theoretical basis is available in the form of conceptual models for prediction of the grouting result.

12.7 TIME SCHEDULE FOR EXECUTION OF THE TESTS

A time schedule for execution of the tests in shown in Figure 12-10.

12.8 INTERNATIONAL PARTICIPATION

The activities at the Äspö HRL have attracted great international interest. Agreements on participation exist with Atomic Energy of Canada Limited (AECL); Power Reactor & Nuclear Fuel Development Corporation (PNC), Japan; Central Research Institute of Electric Power Industry (CRIEPI), Japan; Agence Nationale pour la Gestion des Dechets Radioactifs (ANDRA), France; Teollisuuden Voima Oy (TVO), Finland; UK Nirex, UK; United States Department of Energy (USDOE), USA; Nationale Genossenschaft für die Lagerung von Radioaktiver Abfälle (Nagra), Switzerland; and Bundesministerium für Bildung, Wissenschaft, Forschung und Technologie (BMBF), Germany. The agreements with PNC and CRIEPI have been renewed for a second 4-year period.

The organizations participate in the practical work via their own personnel who participate in the performance of various experiments on the site. Several of the participating organizations have planned additional investigations and experiments, which have been specified in the agreements signed with the organizations. These experiments are being conducted in addition to the programme presented here and add considerable value to SKB's activities.

Scientific coordination takes place via an International Joint Committee (IJC, see below) and Technical Evaluation Forums (TEFs). A considerable part of the joint work concerns models for groundwater flow and radionuclide transport. A Task Force with members from the participating organizations regularly compares and evaluates the results of different experiments with the aid of calculation models. The international participation provides an opportunity to try different approaches e.g. for describing groundwater flow and radionuclide transport in rock. The results of the international cooperation are presented in a separate report series, Äspö International Cooperation Reports.

12.9 EXECUTION, ORGANIZATION, INFORMATION

The research at the Äspö HRL is carried out, like SKB's other research work, via contracts to universities, research institutions, consultants, industrial companies and other Swedish and foreign researchers. This makes it possible to maintain a high level of expertise and quality in all the research and to choose the most qualified experts for different investigations and experiments. Different alternative methods or models can be tried for certain issues.

The aims and contents of the programme are discussed in a programme group within SKB's Development Division. With the RD&D-Programme as a foundation, annual planning reports are written which describe the work for the coming year in relatively great detail. International cooperation is coordinated in an International Joint Committee (IJC). Technical Evaluation Forums (TEFs) are arranged in conjunction with IJC's meetings to furnish advice and viewpoints on programmes and results. In addition to the members of IJC, technical experts from each organization participating in the activities of the Äspö HRL take part in TEFs.

The Äspö HRL comprises a department within SKB's Development Division. The activities at the Äspö HRL are conducted for the most part in project form. The different research projects are headed by project managers, and the Äspö HRL's personnel are responsible for the organization and execution of the work on Äspö.

Information on the Äspö HRL is furnished in a number of ways. The public and nearby residents are given information on the site. A special visitor's niche has been arranged in the access tunnel where SKB's activities and the Äspö HRL are presented. Äspö Research Village also has exhibitions describing the activities and results of the Äspö HRL. Furthermore, general information is provided in the exhibition hall in the village of Simpevarp and along a "nature path" out on Äspö Island. Both are open to the public.

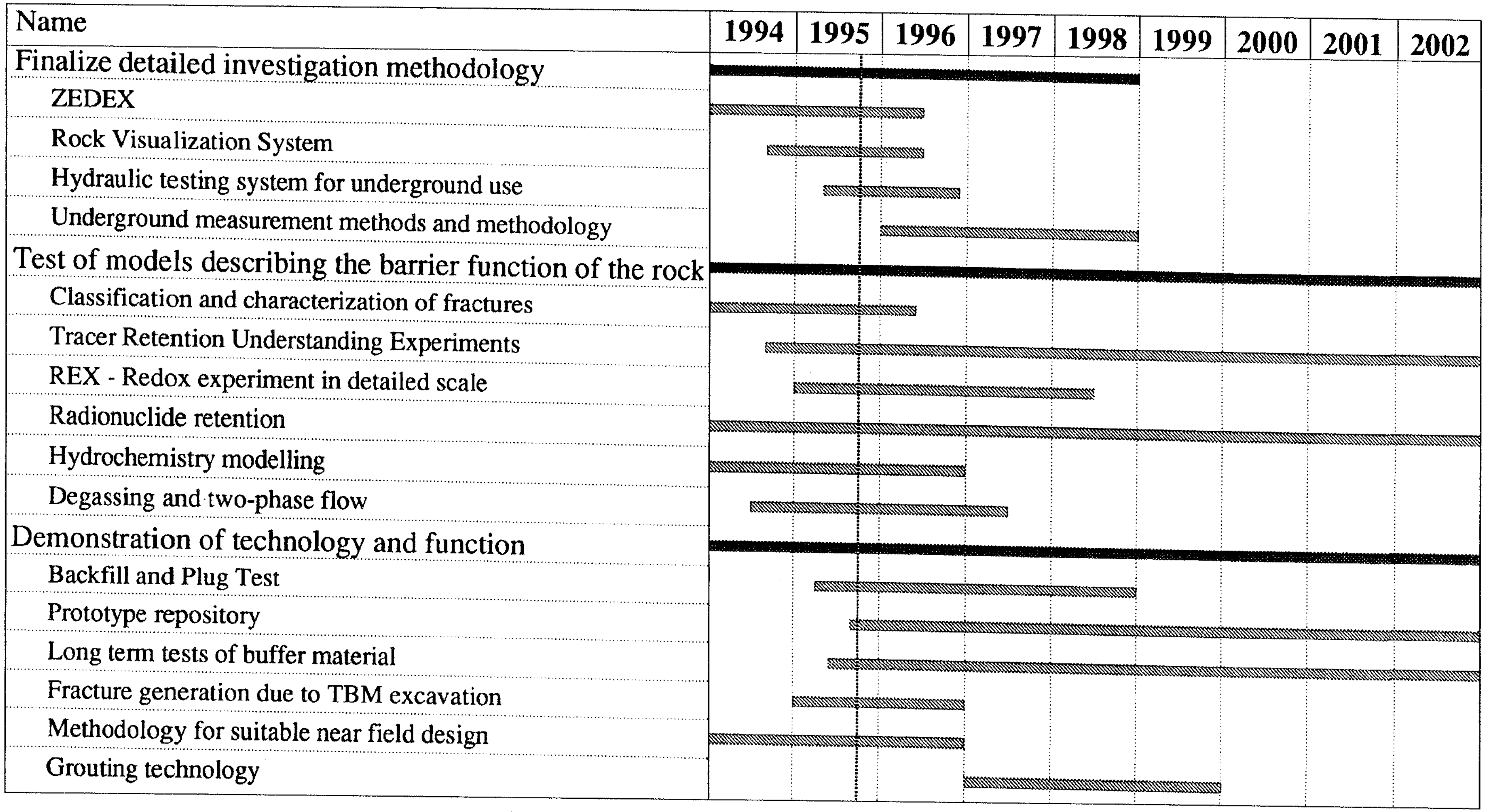


Figure 12-10. *Time schedule for execution of the tests at the Äspö HRL.*

13 ALTERNATIVE METHODS

Deep geological disposal of long-lived radioactive waste is generally accepted among international experts as a good method that can be implemented in a way that satisfies fundamental ethical and environmental requirements. At the same time as the research and development work on direct disposal of spent nuclear fuel is being pursued by construction of facilities for encapsulation and an initial stage of deep disposal, however, good reasons exist to allocate some resources to the follow-up of alternative methods to SKB's main line. 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 alternative final disposal methods. Through SKB's extensive international cooperation network, broad insight is ensured in the major programmes that are in progress in other countries. For certain specific lines of development with possible applications in the longer term, however, a limited domestic effort is warranted. In this way Sweden can build up domestic competence in the field which will allow Sweden to gain insight into the broader programmes being pursued in other countries. In the Government decision on RD&D-Programme 92, the Government stipulated that SKB should, in the present programme, give an account of its assessment of the alternative methods that are being considered.

The greatest interest in the scientific and general debate concerning alternative methods to deep geological disposal has been devoted in recent years to transmutation of the long-lived radionuclides in the waste. This method requires reprocessing of the spent nuclear fuel and separation of the long-lived nuclides from the rest of the waste (partitioning). The situation in this area is described in section 13.1.

Other alternatives to deep geological disposal are not the subject of any direct interest, having been more or less dismissed. Supervised storage is the only exception for the time being. Supervised storage can be carried out in many ways, and is currently being practised in CLAB for the Swedish spent nuclear fuel. However, supervised storage does not fulfil the long-range goal expressed in the law's requirement on final disposal in a safe manner.

Among various alternative methods for deep geological disposal considered in Sweden, some interest has been expressed in disposal in very deep boreholes. The current situation for this alternative is described briefly in section 13.2.

13.1 PARTITIONING AND TRANSMUTATION (P&T) OF LONG-LIVED RADIONUCLIDES

13.1.1 Background

The option of partitioning and transmutation (P&T) of long-lived nuclides in the high-level waste has attracted renewed interest in recent years. Relatively large programmes are under way or planned in France, Japan, Russia and the USA. Sweden is following developments in this field, and contacts have been established with these programmes by Swedish researchers. SKB is funding limited research activities in this field at Swedish universities.

Extensive studies of partitioning and transmutation were conducted in several countries during the 1960s, 1970s and early 1980s. This led to broad knowledge in certain parts of the field, but interest waned as the great interest in breeder reactors and in nuclear waste reprocessing and plutonium recycling declined. A renewed interest came at the end of the 1980s. A Specialist Meeting on Accelerator-Driven Transmutation Technology for Radwaste and other Applications was held in Saltsjöbaden outside Stockholm on 24-28 June 1991 at the initiative of, among others, the then National Board for Spent Nuclear Fuel, SKN /13-2/. After this meeting, reviews of current projects were conducted at SKB's initiative and reports from these reviews were presented in 1992-1993 /13-3, 4, 5/. An update of developments since then is presented in a technical report that is in publication /13-6/.

13.1.2 Radionuclides of interest for P&T

In view of their potential toxicity (and without taking into consideration the geological barrier in a deep repository), the actinides are the most important radionuclides that should be considered for P&T of any kind. There are, however, also some long-lived fission products that may be of importance. A comparison between the radiotoxicity of actinides and of other long-lived fission products in spent fuel is presented in Table 13-1. The table shows that the relative toxicity of actinides is

Table 13-1. Comparison of the radiotoxicity for actinides and some long-lived fission products in spent nuclear fuel after 100 years cooling time.

Nuclide	$t_{1/2}$ (years)	Abundance[1)] (g/ton IHM)	Spec. act.[1)] (Bq/ton IHM)	Dose factor[2)] (Sv/Bq)	Spec. dose (Sv/ton IHM)
^{237}Np	2.1×10^{6}	689.5	1.8×10^{10}	1.7×10^{-6}	3.0×10^{4}
^{241}Am	4.3×10^{2}	972.7	1.2×10^{14}	1.7×10^{-6}	**2.1×10^{8}**
^{243}Am	7.4×10^{3}	145.5	1.1×10^{12}	1.7×10^{-6}	1.8×10^{6}
^{245}Cm	8.5×10^{3}	6.9	4.4×10^{10}	1.7×10^{-6}	7.3×10^{4}
^{238}Pu	8.8×10^{1}	142.8	9.1×10^{13}	1.3×10^{-6}	**1.1×10^{8}**
^{239}Pu	2.0×10^{6}	5641.	1.3×10^{13}	1.3×10^{-6}	**1.6×10^{7}**
^{240}Pu	6.6×10^{3}	1611.	1.4×10^{13}	1.3×10^{-6}	**1.7×10^{7}**
^{241}Pu	1.4×10^{1}	9.5	3.6×10^{13}	2.5×10^{-8}	9.0×10^{5}
^{242}Pu	3.8×10^{5}	439.5	6.4×10^{10}	1.3×10^{-6}	8.0×10^{4}
^{79}Se	6.5×10^{4}	5.2	1.4×10^{10}	5.0×10^{-9}	68.
^{93}Zr	1.5×10^{6}	770.4	7.2×10^{10}	7.1×10^{-10}	51.
^{99}Tc	2.1×10^{5}	849.	5.3×10^{11}	1.7×10^{-9}	883.
^{107}Pd	6.5×10^{6}	181.1	3.4×10^{9}	1.7×10^{-10}	0.6
^{126}Sn	1.0×10^{5}	11.8	1.2×10^{10}	1.7×10^{-8}	206.
^{129}I	1.6×10^{7}	215.1	1.4×10^{9}	2.5×10^{-7}	345.
^{135}Cs	2.3×10^{6}	262.7	8.6×10^{9}	5.0×10^{-9}	43.

[1)]Reference: SKI Technical Report 90:18, J O Liljenzin, April 1990.
[2)]The dose factor is the ratio between the yearly maximum dose for workers (50 mSv) and ALI.

ALI = Annual Limit of Intake according to ICRP
IHM = Initial Heavy Metal

several orders of magnitude greater than that of long-lived fission products.

13.1.3 Transmutation

The basic principle of transmutation of elements is that they are bombarded with neutrons (protons, heavier ions and photons have also been considered) and that (long-lived radioactive) atomic nuclei are transformed via nuclear reactions into short-lived or stable ones. A high neutron flux is required in order for this transformation to take place with a high yield. This can be accomplished in different ways, depending on which elements are to be transmuted. A basic prerequisite for the process to work efficiently is that the nuclides to be transmuted can be separated from other nuclides that would otherwise compete for the neutrons and thereby greatly reduce the effective yield of the process.

The machines with the highest potential for generating the requisite neutron flux at the present time are high-flux reactors and accelerator-driven subcritical reactors.

There are several proposals as to how P&T should be carried out. However, the fundamental question of the purpose and goal of transmutation has not been answered clearly in the different proposals. Several different answers regarding the purpose are given in the argumentation for increased R&D in the field, such as:

- destruction of active waste from conventional nuclear reactors,
- production of nuclear energy without use of conventional nuclear reactors,
- destruction/denaturation of existing surplus materials for nuclear weapons, i.e. Pu^{239},
- fast production of nuclear weapon materials, i.e. H^3, U^{233} and Pu^{239}, or
- introduction of a new fuel cycle based on thorium, which would result in a lower content, turnover and handling of plutonium and heavy actinides than e.g. the fuel cycle for a light-water or a breeder reactor.

If the primary purpose is to achieve a substantial general reduction of the radiotoxicity of the waste that has to be disposed of, development of new technology is required, e.g. an accelerator-based system. Achieving this requires a major technical breakthrough.

Agreement on the purpose of P&T is necessary in order to bring about international collaboration in the P&T field.

13.1.4 Reprocessing and separation

Transmutation of long-lived nuclides requires some form of reprocessing. Through reprocessing, all uranium and plutonium remaining in the spent fuel is recovered. These elements comprise about 96% of the spent fuel (aside from the cladding). Normally, burning of uranium in various P&T concepts is not planned. The uranium will instead be used as fuel in e.g. light-water reactors. Reprocessing on an industrial scale is based today on the PUREX process, which was developed in the 1950s. With time, improvements in operating conditions and a reduction in the amount of secondary waste have led to reduced losses of uranium and plutonium in this process.

A further reduction of these losses can be expected if TBP (tributyl phosphate) in the existing PUREX process is replaced with a monoamide. Monoamides consist solely of the atoms C, H, O and N and are therefore completely incinerable, while incineration of TBP produces phosphoric acid or inorganic phosphates.

At present, neptunium is not recovered in the reprocessing of spent fuel. However, it is known that nearly all neptunium can be made to accompany the uranium by careful dosing of nitrous acid in the primary separation step. By adding a uranium-neptunium separation step in the uranium purification circuit, it is then possible to recover both elements in pure form.

All americium, curium and heavier actinides, together with neptunium and traces of uranium and plutonium, accompany the high-level liquid waste (HLLW) in the current PUREX process and finally end up in the vitrified waste. Current research in Europe (partially funded by the EU) and Japan is aimed at complete recovery of all actinides from HLLW before vitrification takes place. As already mentioned, neptunium can be recovered by a suitable adjustment of the operating parameters in existing PUREX plants. A complete recovery of all actinides requires expansion for further treatment of HLLW.

To simplify such treatment of HLLW, it is necessary to develop suitable new extractants. Primary candidates are various diamides and TPTZ derivatives (TPTZ = tripyridyl triazine). A suitable diamide would extract all actinides and lanthanides (but no other fission products) from HLLW without the need to reduce the acid concentration. Separation of actinides from lanthanides could then be done using TPTZ derivative. Both types of extractants are completely incinerable.

A group of researchers at Los Alamos have proposed ultracentrifugation of molten metals, molten salts or aqueous solutions as an alternative to the usual chemical separation of fission products from actinides.

13.1.5 Recycling and losses

The efficiency of transmutation of a given element can be indicated by the reduction factor by which the quantity of the element in the waste to be disposed of can be reduced by P&T. This factor is in turn determined by the number of times the element must be recycled through the transmutation reactor and by the efficiency of the chemical separation in each cycle. The smaller the quantity that can be transmuted (i.e. the less the burnup) in each cycle, the more cycles are required. The more cycles that are required, the greater the losses to waste are at a given separation efficiency per cycle. In order for the whole process to be worth the trouble, so to speak, the reduction factor should be greater than 100, i.e. more than 99% of the element should be transmuted.

Figures 13-1 and 13-2 show two slightly different idealized fuel cycles. In the one cycle (13-1), reprocessing of the nuclear fuel and of the elements to be transmuted are done separately. In the other (13-2), all reprocessing and separation (partitioning) are done in the same plant. Waste - losses - occur in the different fabrication and separation steps. The total losses to waste for each cycle are illustrated by Figures 13-3 and 13-4. These examples show clearly that the losses in fabrication and separation and in reprocessing must be very small in order to keep the total loss to waste down. For example: with 5% burnup per cycle and 0.5% losses in reprocessing, the total losses in separation and fabrication must be less than 0.025% per cycle in order to get a reduction factor greater than 100. It is very doubtful whether plants can be designed and operated on an industrial scale with such small losses.

13.1.6 Ongoing P&T programmes in other countries

Several countries are conducting P&T research programmes. In France, Japan and Russia, these programmes have political and institutional support as a complement to a future nuclear energy strategy. In the USA, numerous research teams are involved in various activities. Furthermore, a number of other countries are conducting research of small or moderate scope focusing on various P&T options.

French SPIN programme

Recycling of plutonium in LWRs began on a regular basis in France in 1990.

The reprocessing plants in France employ water-based processes for recovery of uranium and plutonium from spent nuclear fuel. The development of separation technology is therefore based on liquid extraction, since this fits in with the existing system. The liquids are also required to contain only C, H, O and N to guarantee complete incinerability.

P&T of long-lived nuclides is one of the goals of the French law passed at the end of 1991. In response, CEA (Commisariat á l'Energie Atomique) started the SPIN programme (SeParation – INcineration). The purpose of this programme is to reduce the volume and radioactivity

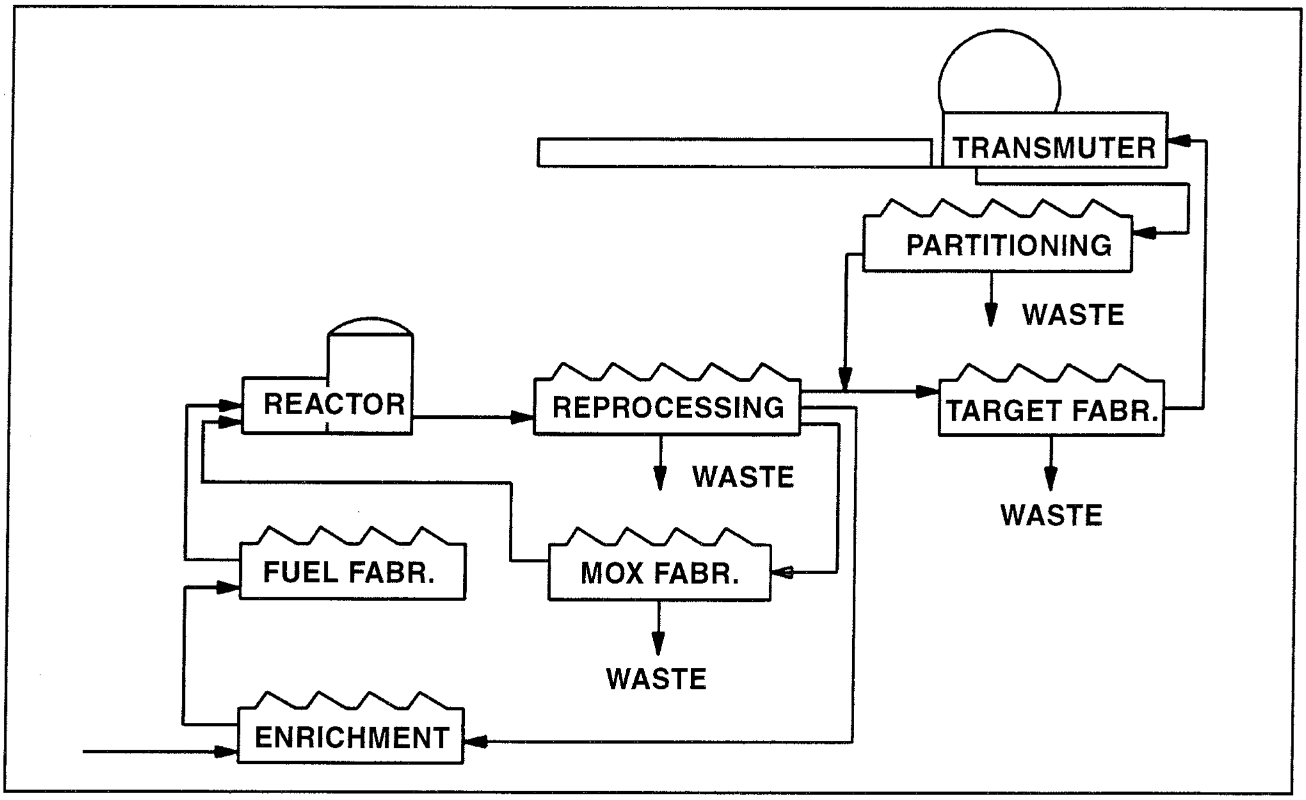


Figure 13-1. *Fuel cycle with heterogeneous transmutation/partitioning operations.*

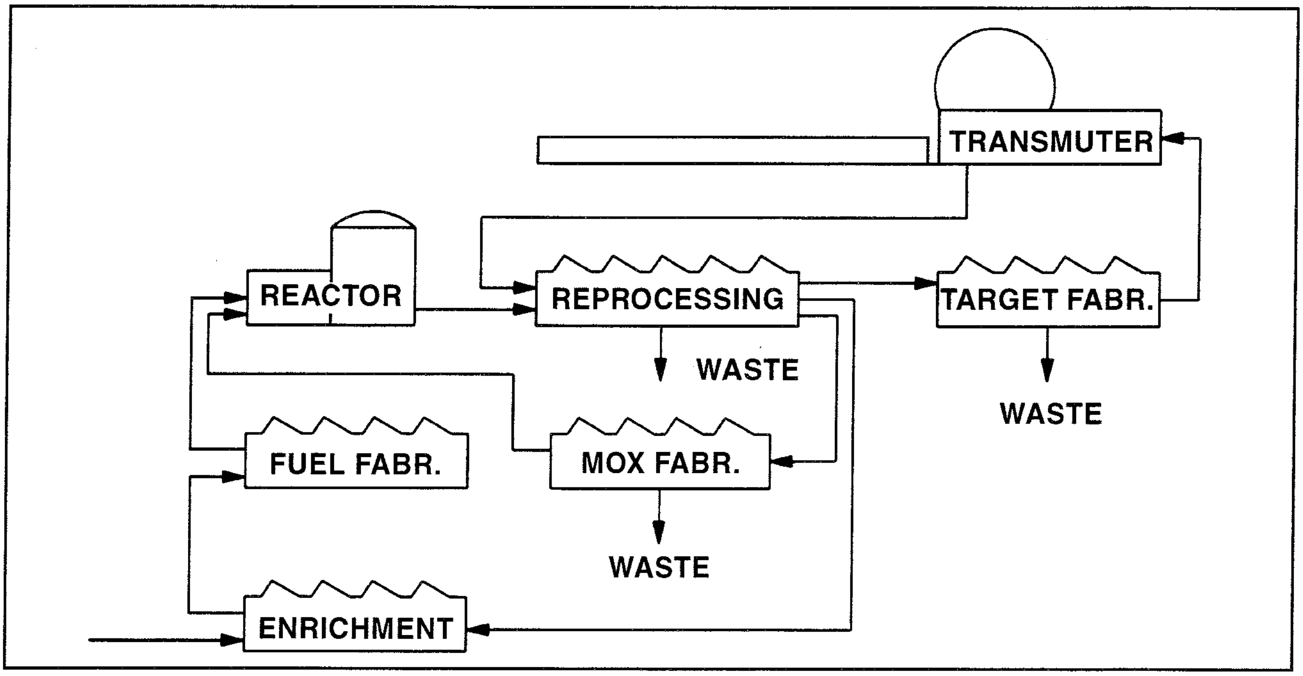


Figure 13-2. *Fuel cycle with homogeneous reprocessing of spent fuel and transmutation targets.*

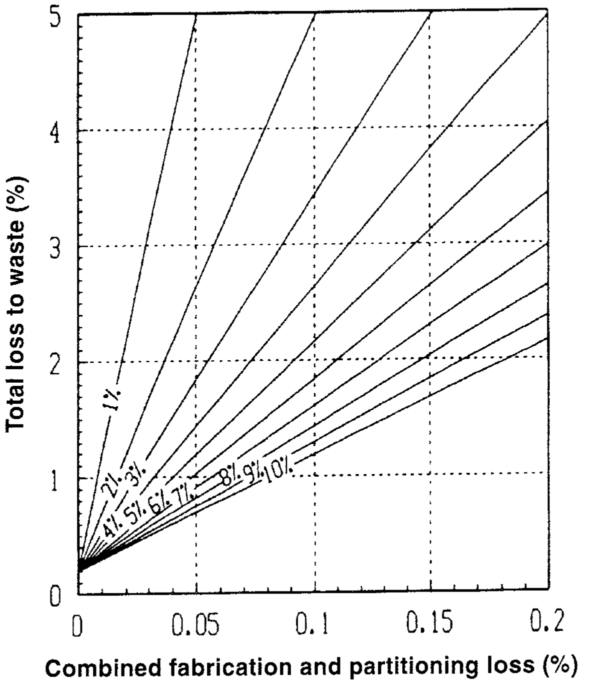


Figure 13-3. *Relation between total loss to wastes and single pass losses in transmutation with separate partitioning for some single pass target burn-out values (assumed reprocessing loss 0.2%).*

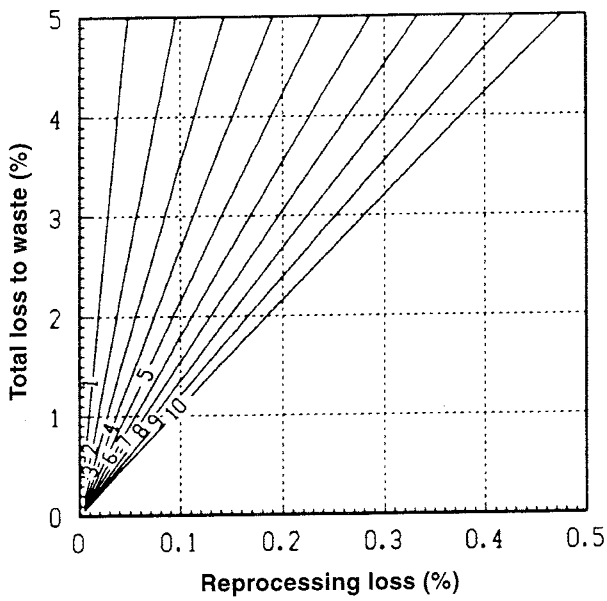


Figure 13-4. *Relation between total loss to wastes and single pass losses in reprocessing with homogeneous reprocessing for some target burn-outs (%) and a fabrication loss of 10% of the reprocessing loss.*

of the waste to be disposed of in a geological repository. The SPIN programme is divided into two sub-programmes – a short-term one called PURETEX and a more long-term one called ACTINEX /13-7/.

PURETEX is concerned with improvements of the traditional PUREX process. Plutonium and neptunium are separated more efficiently by use of a monoamide which is completely incinerable and does not contribute to secondary waste in the same way as TBP.

The ACTINEX programme is concerned with selective and specific separation of all actinides and long-lived fission products. The separated actinides are planned to be transmuted in PWRs or in breeder reactors. Three main strategies for partitioning are being studied:

- Co-extraction of actinides and lanthanides with diamide in the first step and then extraction of the actinides from the lanthanides in a second step employing TPTZ.
- Development of new extractants, e.g. picolinamides, which extract the actinides selectively and leave the lanthanides in the liquid phase. This method is more advantageous, since it is not necessary to extract large quantities of lanthanides.
- Separation of americium at higher oxidation states than three.

The technical feasibility of using U-Pu and U-dioxide for recycling and transmutation of minor actinides has been demonstrated in irradiation experiments within the SUPERFACT project. Additional experiments are planned within this project.

Japanese OMEGA programme

In 1987, the AEC (Atomic Energy Commission) of Japan approved the OMEGA programme (Options for Extra Gain from Actinides) /13-8/ – a programme for R&D on P&T technology. The transmutation methods being considered are based either on an accelerator or on a nuclear reactor.

Most of the R&D within the programme is being conducted at JAERI, PNC and CRIEPI. Phase 1 of the programme will be completed in 1996. Phase 2 is planned to be carried out during 1997-2000 and will include technical tests and demonstrations of the concepts selected on the basis of phase 1.

A conceptual study of accelerator-driven transmutation systems is being conducted at JAERI. Both systems with solid fuel and a spallation target of solid material and systems with fuel in the form of a molten salt and a spallation target of liquid material are being studied. In both cases, fast neutrons are used for the transmutation. PNC has also studied systems based on an electron accelerator and gamma-neutron reactions with Cs and Sr isotopes. This method is judged to be inefficient. However, most of the Japanese efforts appear to be focused on use of existing or new reactors (actinide burners) for transmutation.

The four groups of separation processes developed by JAERI during phase 1 are based on DIDPA (diisodecyl phosphoric acid) and DTPA (diethylenetriamine pentaacetic acid). They will be tested on real HLLW in the NUCEF plant at the research centre at Tokai. Dissolution of enriched uranium for criticality testing began in the autumn of 1994 and active operation is expected to start during 1995.

Russian studies

Studies of P&T are being conducted in Russia as well /13-9/. Several research centres under the Ministry of Atomic Energy in the Russian Federation are participating in these studies. The nuclides they plan to transmute are minor actinides and the fission products Tc^{99} and I^{129}. Burning of actinides is mainly planned to be done in fast reactors, but accelerator-driven systems are also being studied. Both specially designed and traditional fast reactors are being considered. Preliminary results show that sodium-cooled reactors of the BN type can be utilized effectively for burning of minor actinides and plutonium. For safety reasons, the concentration of minor actinides in MOX fuel must be less than 5%. Under these conditions, a breeder reactor (type BN-800) can transmute up to 100 kg of minor actinides per year, which is equivalent to the output of three light-water reactors (type VVER) of 1000 MWe each. A preliminary study shows that a reactor for the P&T option is twice as expensive as direct disposal of solidified waste. It is considered impossible to predict the cost for an accelerator-driven P&T plant.

Studies in the USA

Several strategies for P&T of nuclear waste have been presented in the USA. Funding of P&T development by the responsible government agencies is meagre or non-existent. The purpose of P&T in the USA is therefore unclear. The following reasons have, however, often been given:

- destruction of actinides and long-lived fission products from current or former nuclear energy programmes,
- development of a proliferation-proof nuclear energy system,
- rapid destruction of surplus plutonium for nuclear weapons,
- a nuclear energy system without long-lived radioactive waste.

A P&T programme called ADTT (Accelerator-Driven Transmutation Technologies), primarily based on the use of a linear accelerator in combination with different concepts for a subcritical reactor, is being conducted at Los Alamos National Laboratories. One of the concepts is called ABC (Accelerator-Based Conversion) and is aimed at transmutation of weapons-grade plutonium. A very high burnup of plutonium is expected to be achieved without reprocessing by utilizing neutrons from both the accelerator's spallation target and from the reactor. Another sub-project called ADEP (Accelerator-Driven Energy Production) may be the main direction for transmutation research at LANL and is based on the thorium-uranium fuel cycle where Th^{232} is transformed into the fissile isotope U^{233}. The process provides both energy gain and transmutation. The proposed reactor technology applies experience gained from the MSRE (Molten Salt Reactor Experiment) at Oak Ridge during the 1960s. The goal of the ADEP concept is a future energy source that produces smaller quantities of long-lived radionuclides than present-day nuclear reactors. A third sub-project is called ATW (Accelerator Transmutation of Waste) and aims at transmuting actinides and long-lived fission products from light-water reactors and from waste generated by the nuclear weapons programme. In this sub-project, great importance is attached to the requirement that the separation methods must not "liberate" pure fissile material that could be diverted to weapons manufacture. In both the ADEP and ATW concepts, the idea is to use a subcritical graphite-moderated molten salt reactor. A steady flow of molten salt, from which the fission products are separated, is taken from the reactor.

The PHOENIX concept at Brookhaven National Laboratory is based for the most part on existing technology, with some development. Transmutation is planned to be done with fast neutrons. PHOENIX is described in /13-3/.

Work within the EU

Strategic studies of P&T were carried out on behalf of the EU at three European research institutes during the programme period 1991-1994. CEA in France investigated the potential and cost of P&T of long-lived nuclides. Siemens in Germany analyzed advanced reactors ("converters") for transmutation and their safety. ECN in the Netherlands has processed nuclear data for transmutation studies.

The new EU programme "Nuclear Fission Safety" contains several research areas for P&T. It is still unclear how much money will be allocated to these areas. Swedish groups are participating in two of the projects recently adopted by the EU.

Other international organizations

Several other international bodies are also involved in the P&T work. The OECD/NEA is coordinating a series of workshops for information exchange between its member states.

The IAEA organized a meeting with a Technical Committee on safety and environmental aspects of P&T at the end of 1993 /13-10/. The meeting recommended that the IAEA should study the safety, environmental and non-proliferation aspects of P&T. Special areas that should be studied are:

- definition of toxicity criteria,
- possible reduction of toxicity through P&T, and
- priority list for radionuclides to be subjected to P&T.

Since these areas are closely linked to how a geological repository works, experts within geological disposal research should participate in the work.

13.1.7 Some conclusions

It can be concluded that the technical means exist today for transmutation of a considerable portion of the plutonium that is present in spent nuclear fuel. In the case of certain other long-lived nuclides, the potential exists for transmutation (on a limited scope) after moderate development. However, development of proposed more advanced methods for transmutation on an industrial scale would require in some respects a technical breakthrough, many years of research and large resources. Just the fact that transmutation requires reprocessing makes the costs of a waste management scheme based on transmutation of long-lived nuclides (at present-day cost levels) considerably higher than direct disposal of unreprocessed nuclear fuel.

It is difficult to find any economic motive or short-term safety motive for transmutation in comparison with currently industrially established systems for management of spent nuclear fuel (direct disposal and reprocessing/vitrification plus final disposal). The assessment is rather that radiation doses to the personnel in conjunction with handling and treatment will be considerably higher for the transmutation case. Even if the development of transmutation is successful, there will be a need for deep disposal of such long-lived waste as will inevitably arise from the relatively complicated treatment process.

Transmutation using neutrons from nuclear reactors or spallation neutrons (generated by accelerator technology) is still being considered. Nuclear reactors have come considerably further in development than spallation sources. This applies to both thermal and fast neutrons. However, if actinides with very small reaction cross-sections for neutrons are to be transmuted, it may turn out that this can only be done with spallation neutrons.

In determining the goals of the partitioning process that is needed, it is necessary to take into consideration the strong link that exists between the efficiency of partitioning and transmutation and acceptable losses to secondary waste streams. Water-based processes, molten salt-based processes and metallurgical processes are being considered for partitioning. Present-day knowledge concerning water-based processes is extensive and is based on more than 40 years of experience. Metallurgical and, to a large extent, also molten salt-based processes are in the research stage. It is therefore too early to draw any definite conclusions regarding these partitioning processes. In general it can be said that more research on partitioning technology is included in new R&D programmes than before. However, very large resources must be allocated to this area in the future if P&T is to be a realistic alternative to the present-day nuclear fuel cycle.

The introduction of P&T will not eliminate the need for geological disposal of radioactive waste. Long-lived radiotoxic materials cannot be transmuted in large quantities without some losses to secondary waste streams. In an overall assessment of the P&T option it is therefore important to include an estimation of the risks of releases from the treatment process and final disposal of the secondary waste, so that the total risk of releases to the biosphere (in the short and long term) is not greater than the most pessimistic forecast of the risk of releases from direct deep disposal of spent fuel. Even though there is an international consensus that current fuel cycle scenarios, including geological disposal, provide adequate protection for man and the environment, there is still great interest in studying how a further reduction of the long-term potential danger can be achieved and at what cost. The most serious criticism levelled at geological disposal is the limited ability to predict future effects. However, it is important to consider the question of whether funding for the development of P&T will conceal the fact that the absolute risk of radioactive releases from proposed deep repositories and waste forms is already very low.

Development and application of P&T is more likely in a scenario with continued use and expansion of nuclear energy than in the opposite scenario. It is being discussed whether expansion of nuclear energy with the support of P&T could be a reasonable goal or whether P&T and expansion of nuclear energy should be judged independently of each other. There is very little incentive for the development and introduction of P&T within the framework of current nuclear energy programmes with "once-through use" of the uranium fuel. A decision in favour of more long-range exploitation of nuclear energy would put the situation in a different light. Such a trend - on the global scale - would in the long run probably lead to an increase in the current costs of nuclear fuel supply, which would in turn make other fuel cycle scenarios more economically competitive.

P&T is still being evaluated in many countries and by international organizations in terms of its potential as a long-term energy resource in combination with its potential for destruction of long-lived nuclides. It is difficult to see how P&T could be developed solely on a national basis, considering the great amount of work that is needed. International collaboration is therefore essential. This in turn requires reasonable agreement on the purposes of P&T, which at present appear to be very different in e.g. Europe and the USA.

The development of P&T may possibly be of interest in the long term to the Swedish nuclear waste programme as well. It is therefore important to continue to follow the development work being pursued in other countries. Research in this area spans over several key disciplines in the nuclear energy field such as reactor physics, neutron physics, nuclear chemistry, materials science and engineering, nuclear reactor safety, nuclear reactor technology and radiation protection. The cutting-edge technology involved makes the field attractive to young engineers and scientists. Efforts in this field can therefore contribute to maintaining and rejuvenating research competence within the aforementioned disciplines, which is also beneficial to the nuclear energy field as a whole.

SKB for its part intends, within the framework of studies of alternative waste strategies, to give continued support to domestic research aimed at a deeper understanding of safety matters, material matters and matters of importance for assessment of the entire process. This research should, as before, be pursued in close contact with international development work in the field.

13.2 GEOSCIENTIFIC CONDITIONS AT GREAT DEPTHS

The Project on Alternative Systems Study (PASS) was concluded in 1992 /13-11/. Alternative systems for deep disposal were studied in this project and comparisons were made with the reference alternative according to the KBS-3 method. One of the alternatives studied was disposal in very deep (2000-4000 m) boreholes.

No further development in the technical sense has taken place since then for disposal at great depths. On the other hand, general research is still being pursued for the purpose of knowledge-building. As a part of this research, a study is being conducted of the geoscientific conditions at great depths in the rock. The primary goal is to improve knowledge of parameters and processes of importance for the safety-related and engineering premises for waste disposal at great depths. The scope and outline of the study are described in greater detail in /13-12/. At the same time, developments within the field of drilling technology for very deep boreholes are being followed.

As a first step, literature studies have been conducted to compile and evaluate available data on important geoscientific parameters. For the concept of disposal at great depths as well, radiological safety is dependent on mechanical stability, a chemically stable environment with oxygen-free conditions and a low groundwater flux.

Table 13-2 shows subject areas and examples of parameters that have been studied. Compilations and evaluations of available data are presented in separate reports /13-12, 13/. The requirement that the results should be relevant to conditions in Swedish crystalline rock has limited the compilations primarily to information from the Baltic Shield and from other continental basement areas.

Table 13-2. Examples of geoscientific parameters for which data from the depth interval 1000–5000 m has been compiled.

Subject area	Examples of studied parameters
Geology – lithology	Rock types, degree of metamorphosis, structure
Geology – tectonics	Fractures (frequency, orientation, mineralization, etc.) Fracture zones (frequency, orientation, morphology)
Hydrogeology	Pressure, conductivity, frequency of conductive fractures
Hydrochemistry	Chemical composition and origin of the groundwater Isotopes
Rock mechanics	Rock stresses, strength
Geophysics	Porosity, density etc. Geothermal parameters Seismic parameters

Data that have been compiled pertains to the depth interval 1000-5000 metres. Direct measurement data from such great depths can only be obtained from deep boreholes and mines. Several countries are conducting, or have conducted, extensive research-oriented deep drilling programmes. These programmes have generated a large quantity of geoscientific data from the depth interval mentioned and have yielded valuable experience of drilling and investigation activities. Of special interest for Sweden are deep drillings carried out in crystalline bedrock in the former Soviet Union and in Germany. Data and experience from the very extensive Soviet investigations have been shared with SKB via cooperation with Russian organizations. The results have been published in a special report /13-14/. The material includes, among other things, geoscientific information from a more than 12,000 m deep borehole situated on the Kola Peninsula, within the Baltic Shield.

The deepest mines in the world reach down to a depth of about 4,500 metres. There are a number of mines with depths down to about 2,500 m in Canada and South Africa, for example. The geoscientific information from the mines is extensive, and in some respects unique. One example is the knowledge of rock mechanics that has been gained from mining at great depths. However, assimilating this knowledge from the mines and applying it within other areas requires considerable analysis work and thorough studies of individual practical cases.

Besides direct data from measurements at great depths, the compilations also incorporate indirect data from geophysical investigations performed from the surface. Seismic measurements are the most important source of information. The seismic information includes both data registered in conjunction with natural or induced seismic activity in the earth's crust and data from active seismic investigations.

As a second step in SKB's ongoing programme concerning disposal at great depths, an integrated interpretation of the information is planned aimed at devising descriptive models for important parameters and processes. Important questions that must be taken into account in this interpretation process include the following:

- The reliability of data collected from great depths.
- The feasibility in a general sense of devising generalized relationships and descriptions on the basis of the random sampling represented by deep boreholes.
- The feasibility in a general and site-specific sense of assessing conditions at great depths based on observations at the surface or at depths where the bedrock can be investigated by means of conventional methods.
- The presence and importance of chemical and physical processes that do not occur at or near the surface.

Furthermore, some resources will also continue to be devoted to following developments of drilling technology. Special encapsulation requirements for disposal in very deep holes also need to be examined.

14 DECOMMISSIONING OF NUCLEAR FACILITIES

14.1 BACKGROUND

When a nuclear power plant is taken out of service, parts of it are contaminated with radioactivity. This means that decommissioning and dismantling 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 and medium-sized nuclear power plants have already been decommissioned in various countries. Several medium-sized nuclear power plants are currently being decommissioned, for example in Japan, the USA, Germany, Belgium and the UK. Decommissioning and dismantling of several large nuclear power plants is also under way in the United States, for example Trojan, a 1000 MW PWR. Other reactors that have been taken out of service have been mothballed so that they can wait 30-50 years before the actual dismantling is carried out.

Experience of dismantling in Sweden is limited to the dismantling of the R1 research reactor in Stockholm and several smaller plants at Studsvik. Considerable experience of a similar kind has, however, been obtained from the thorough decontamination and refurbishment of Oskarshamn I, the steam generator replacements at Ringhals-2 and Ringhals-3, and from other repair and refurbishment jobs at the nuclear power stations.

The completed decommissionings and a number of studies show that the methods for decommissioning and dismantling nuclear power plants are available today. A report from the OECD/NEA /14-1/ 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 refurbishment work at the Swedish nuclear power plants. It is only for dismantling of the reactor vessel and its 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 projects through a cooperative programme organized under the auspices of the OECD/NEA, where SKB is in charge of the secretariat and programme coordination.

14.2 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 should be restored (remediated) after some time so that it can be used without radiological restrictions. This is to 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 /14-2/, 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 system 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 than 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 of Swedish nuclear power plants" /14-3/. This report shows that decommissioning can be commenced approximately one year after the last reactor has been shut down at a nuclear power station. The plant mothballing for 30-50 years before the actual dismantling work is commenced is also shown to be a feasible alternative. In the case of immediate dismantling, personnel that know the plant are on hand. 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 of the nuclear power station site – for example if it is to 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 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 for determining the need for making allocations to a fund for decommissioning.

The most important means for achieving these goals are

- follow-up of international development work,
- follow-up of experience from maintenance and rebuilding work at the nuclear power plants, and
- certain special studies and tests.

14.3 CURRENT STATE OF KNOWLEDGE

14.3.1 Sweden

A comprehensive survey of the technology for decommissioning and dismantling the Swedish nuclear power plants and the costs associated with this was made in /14-3/. The conclusion of this study is that decommissioning is not expected to entail any difficult technical problems. Most of the technology that is needed for the future decommissioning of the nuclear power plants is already available and is used routinely in maintenance, repair and rebuilding work at the nuclear power plants. Special equipment only has to be developed for dismantling of the reactor vessel and for demolition of heavy concrete structures. A great deal of work has been done in these areas abroad, and it is of great importance to follow this up. But special research efforts are not warranted in Sweden.

A number of major rebuilding jobs have been carried out at the nuclear power plants in recent years. This work will provide valuable experience for the future work of decommissioning.

A study has been conducted of the possibility of removing and transporting the reactor pressure vessel (RPV) intact and disposing of it in SFR, as an alternative to prior segmentation of the RPV /14-4/. This method offers a possible means for reducing radiation exposure and costs and should be included as an alternative in detailed planning of decommissioning.

Large quantities of slightly contaminated waste are obtained during decommissioning. Similar waste is obtained from repair work at the nuclear power plants. Different methods for managing and disposing of this waste are now being tested, such as decontamination and melting, and direct disposal in SFR.

The costs of decommissioning have been presented in /14-3/. The Swedish costs are low by international standards, which is in part attributable to the efficient system that has been developed for transportation and final disposal of nuclear waste in Sweden, enabling large components to be handled without the need for extensive segmentation.

14.3.2 International work on decommissioning

The most important work in the field of decommissioning 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 or so reactors have been fully decommissioned to stage 3, i.e. they have been dismantled and the radioactive components have been removed from the site. In addition, a large number of plants have been taken out of service and decommissioned to stage 1 or 2. Most decommissioning projects have involved experimental reactors or small power reactors. Not until recent years have some medium-large and large reactors 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 particular 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.

The OECD/NEA's Cooperative Programme on decommissioning

A special programme has been organized within the OECD/NEA for an exchange of information and experience between ongoing decommissioning projects. Most major decommissioning projects in the world are included in this programme. At present the programme includes a total of 29 projects in 10 countries. A summary of the projects included in the programme is given in Table 14-1. Nineteen 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 /14-1/. 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), Niederaichbach (Germany) and BR-3 in Belgium. 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 disposal site.

The decommissioning work for JPDR and Niederaichbach is also nearly finished. In both cases the reactor vessel has been cut up by remote control. In the JPDR project, extensive testing and development of different dismantling methods has been carried out. Segmentation of the RPV after extensive decontamination is also being studied in BR-3.

SKB is in charge of programme coordination for the OECD/NEA programme for the OECD/NEA and thereby has an opportunity to follow the technical aspects of the different projects closely.

The EU's research programme

The EU has had a cooperative research programme within the field of decommissioning since 1979. So far the studies have primarily concerned different dismantling techniques as well as questions pertaining to activity content and waste management /14-5/. 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 /14-6/.

In the recently concluded third 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). In the planning of the fourth programme, the EU has concluded that decommissioning is a mature technology and can be based on commercially available methods. The scope of the research has therefore been reduced considerably.

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 an intact RPV.

Examples from some other countries

Decommissioning of several commercial reactors is being carried out in the USA, for example Trojan,

Table 14-1. 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 MW_t	Stage 3
Gentilly-1, Canada	Heavy water reactor, 250 MW_e	Stage 2
NPD, Canada	Heavy water reactor, CANDU, 25 MW_e	Stage 1
Rapsodie, France	Sodium-cooled fast reactor, 29 MW_t	Stage 2
G2, France	Gas-cooled reactor, 45 MW_e	Stage 2
AT1, France	Reprocessing plant for fast reactor fuel	Stage 3
Niederaichbach, Germany	Gas-cooled heavy water moderated reactor, 106 MW_e	Stage 3
Lingen, Germany	BWR, 256 MW_e	Stage 1
MZFR, Germany	Heavy water reactor, 50 MW_e	Stage 3
Garigliano, Italy	BWR, 160 MW_e	Stage 1
Japan Power Demonstration Reactor (JPDR), Japan	BWR, 13 MW_e	Stage 3
Windscale Advanced Gas Cooled Reactor, Great Britain	AGR, 33 MW_e	Stage 3
BNFL Coprecipitation Plant, Great Britain	MOX fuel fabrication	Stage 3
Shippingport, USA	PWR, 72 MW_e	Stage 3
West Valley Demonstration Project, USA	Reprocessing plant for LWR fuel	Stage 3
EBWR, USA	BWR, 100 MW_t	Stage 3
Tunney's Pasture, Canada	Isotope handling	Stage 3
BNFL B204 Primary Separation Plant, Great Britain	Reprocessing facility	Stage 2
JRTF, Tokai, Japan	Reprocessing facility	Stage 3
Greifswald, Germany	PWR, 5x440 MW_e	Stage 3
Bohunice A1, Slovakia	Gas-cooled heavy water moderated reactor, 150 MW_e	Stage 1
Vandellos 1, Spain	Gas-cooled reactor	Stage 2
HDR, Germany	BWR, 25 MW_e	Stage 3
WAK, Germany	Reprocessing plant	Stage 3
EL 4, France	Gas-cooled, heavy water moderated reactor	Stage 2
Building 211, Marcourle, France	Vitrification plant	Stage 3
Fort St Vrain, USA	High-temperature gas-cooled reactor	Stage 3
Fernald, USA	Uranium hexafluoride plant	Stage 3

Rancho Seco, Maine Yankee and Fort St Vrain. All of these projects are being conducted at a relatively slow pace, one reason being the fact that sufficient disposal capacity is not available.

In Germany, Gundremmingen A, Kahl and a number of smaller reactors are currently being decommissioned. Furthermore, planning is under way to decommission the four reactors at Greifswald. Several reactors, for instance Lingen, have been placed in stage 2 for dismantling at a later date.

In France, all the Magnox reactors, plus two water-cooled reactors, have been taken out of service and will be decommissioned to stage 2. The complete dismantlement of one reactor in the near future for demonstration reasons is being discussed.

In the UK as well, most of the Magnox reactors have been taken out of service. Work is under way to decommission them to stage 2. They are not planned to be dismantled for another 100 years or so.

14.4 RESEARCH PROGRAMME 1996-2001

The schedule for carrying out the necessary R&D work on dismantling is closely linked to the schedule for the decommissioning of the Swedish nuclear power plants. As indicated above, dismantling of the first plant 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. Since a great deal of development work is currently being done abroad, there is no justification for starting any large Swedish development efforts during the coming six-year period.

Completed 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 managing and disposing of an intact reactor vessel (see above),
- transportation and final disposal of other large components,
- technology for segmentation of reactor internals,
- technology for demolition of the biological shield,
- management and disposal 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.

The most important activities planned for the next few years include to systematically follow up and take advantage of the opportunities that arise to gain experience from the repair and rebuilding work that will be carried out. Separate studies are planned of certain large projects, such as the renovation of Oskarshamn I and completed steam generator replacements.

Some large components have arisen from completed rebuilding work that must be dealt with as waste. A systematic review of different methods for handling these components is being conducted and tests are planned of e.g. transport and disposal, or alternatively decontamination and melting.

As far as other areas are concerned, R&D activities being conducted abroad will be followed up. There may be some more systematic work in the other areas during the latter part of the period, in which case the possibility of conducting tests in the decommissioned Ågesta reactor should also be evaluated.

The follow-up activities will be conducted as before through the programme coordination function within the OECD/NEA programme, as well as 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 (free) release at the nuclear power plants. However, the low free release limits make measurement and classification very laborious. Before decommissioning is commenced, it is essential that rules and methods for free 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 stand completed. Since the waste from decommissioning is equivalent in many respects to some of the waste from the operating period, experience from SFR 1 can serve as a basis for the design of SFR 3. This is described in /14-7/. The time from preliminary planning and design to a finished facility has been estimated to be about 7 years, which means that this work will not be commenced until a few years into the next century.

For the dismantling 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.

15 EXECUTION OF THE PROGRAMME UNCERTAINTIES IN THE TIME-SCHEDULE; COSTS

This chapter presents some comments on how the programme will be executed, some uncertainties that exist regarding the time schedule and an estimate of the costs for the work during the next few years. In this connection, a brief summary is also given of the costs for the R&D work carried out within previous programmes.

15.1 EXECUTION

SKB's organization has been adapted to suit the needs of the programme presented in RD&D-Programme 92, which is the basis of the programme presented here. SKB is organized in three divisions: Facilities, Deep Repository and Development. The work involved in designing and siting the encapsulation plant is coordinated by a special project within the Facilities division. Deep Repository is responsible for siting and designing the deep repository. The Development division is in charge of supportive R&D, of operation and RD&D at the Äspö HRL, of R&D on alternative methods, and, in consultation with the facility projects, of the work involved in assessments of the long-term safety of the repository.

SKB's own staff for execution of the programme is relatively limited. It is primarily concentrated to resources for management, planning and follow-up of the work. In addition, SKB has its own staff for execution of central parts of assessments of the long-term safety of the repository. R&D work is pursued in cooperation with universities, colleges, consulting firms and individual experts in Sweden and abroad. Design, engineering and construction tasks are contracted out to consulting, engineering and contracting firms. Bidding and procurement procedures are normally followed for large assignments. A list of the institutions and companies that have participated in SKB's work is presented in SKB's Annual Report /15-1/. See also Appendix 1.

An important ingredient in maintaining a high standard of quality in the work is international cooperation. This is pursued in various forms and on a large scale. It is worth noting here that nine foreign organizations from eight countries are currently participating in R&D work at the Äspö HRL. The majority of the projects concerned with natural analogues are being conducted in broad international collaboration. Representatives from SKB are participating in committees and working groups in the nuclear waste field within the OECD/NEA and within the IAEA.

15.2 UNCERTAINTIES IN THE SCHEDULE

A target date given in the programme is the start of deposition of encapsulated nuclear fuel in 2008. The time schedules presented in chapters 7 and 9 and elsewhere mainly take into consideration engineering activities, which are relatively simple to schedule. As is explained in chapter 9, siting is in practice governed to a large extent by societal and political factors whose influence on the time schedule is difficult to assess. A general assessment is given below of uncertainties in the schedules and their consequences in terms of whether work will be finished ahead of or behind schedule.

A general time-controlling factor is whether results from the research at the Äspö HRL and other research can keep pace with information needs within the encapsulation and deep repository projects. For example, the results of the pre-investigations for the Äspö HRL must be evaluated before they can be taken into account in the site investigation programme for the deep repository.

15.2.1 Siting of the deep repository

The siting of the deep repository is a controversial question that attracts interest in some people and alarms others. The debate can cause politicians to make sudden about-faces in where they stand on siting a repository in their municipality. Anxiety among the population in a municipality being considered for siting of a repository or along the transport route must be met with objective information and by great openness in the siting process. People in the concerned municipalities must be given time to digest the idea of a deep repository and absorb the information given to them. The speed with which the siting process proceeds is therefore in practice largely dependent on public-opinion and political factors. It is difficult to estimate how long this process may take. In the schedule it is assumed that the feasibility studies can be conducted in a rational manner technically speaking and that site investigations can start around the beginning of 1997.

If further feasibility studies should be needed in late 1996, the site investigations can be postponed a year or so. All things considered, the date for the start of site investigations seems to be critical for the timing of the entire siting programme.

It is estimated that site investigations will take 4-5 years to perform. The determining factor is the scope of the investigations, which is in turn affected by safety and performance requirements as well as by local conditions. For example, the schedule is dependent on the availability of existing data at the start of the investigations. It may take a year or so extra to investigate a site where a modern geological survey is lacking. On the other hand, if the investigations are instead performed on a site that has previously been thoroughly investigated, for example one of SKB's study sites, the investigation time can be shortened by a year or so.

The siting phase will be concluded with SKB applying for a permit for detailed characterization of one of the investigated areas. According to the Government decision of May 1995 /15-2/, a detailed characterization is to be regarded as part of the process of erecting a nuclear facility. The site must therefore be licensed according to the Act on Management of Natural Resources as well as the Act on Nuclear Activities. If the licensing procedure has a favourable outcome, a licence is obtained for erection of a nuclear facility.

The process of regulatory review up to a Government decision regarding permission for detailed characterization has been estimated in the schedule to be just over one year. This is taking into account the fact that the comprehensive EIA process that has preceded the application has made everyone involved thoroughly familiar with the background data submitted in support of the application.

All things considered, it is realistic to assume that the question of where the deep repository is to be sited can be decided a couple of years into the next century. This is provided, however, that site investigations can start around the beginning of 1997. Examples of events that could lead to a considerable delay in the schedule are, for example, that feasibility studies and site investigations are delayed for political reasons, that more site investigations are needed than planned due to the fact that the first ones do not reveal sufficiently good conditions, or that the decision process for the necessary government permits takes longer than foreseen.

15.2.2 Encapsulation plant

The siting work for the encapsulation work has come farther. Work is under way to gather material for an environmental impact assessment for a siting at CLAB. This work is being carried out in consultation with the municipality of Oskarshamn and the county administrative board in Kalmar County. The greatest uncertainties in the time schedule for the encapsulation plant stem from its connection with the siting of the deep repository. Prior to the decision on the encapsulation plant, the siting of the deep repository should have progressed so far that it is probable that deposition of canisters can begin as soon as the encapsulation plant has been put into operation. Delays in the deep repository project may therefore influence when the decision to build the encapsulation plant can be taken.

The most significant technical uncertainties in the encapsulation project are connected with the choice of method for fabrication of the copper canister and for sealing and inspection of the canister. Before an application for a permit to build the encapsulation plant is submitted, SKB must show that a suitable method exists for serial fabrication of canisters, and carry out extensive canister sealing trials. Problems within these areas can entail delays of a year or so.

Once the decision to erect the plant has been made, the time required for construction and commissioning of the plant is estimated to be about 7.5 years. The uncertainty in this estimate is judged to be less than one year.

15.2.3 Detailed characterization preceding initial operation etc.

A detailed investigation entails driving tunnels and/or shafts down to repository depth and carrying out various types of bedrock investigations. At the same time, the surface facilities with associated road system and/or railway are built. The phase is concluded with an application for an operating permit. This application contains an updated safety report based on the results obtained during the detailed characterization and a final safety report for initial operation.

Once the Government and the Municipality have given their approval to the siting of the deep repository, it can be expected that the time required for the continued work will be mainly dependent on technical factors. In the time schedule, 5-6 years is the length of time estimated for the detailed characterization, which includes shaft/tunnel to repository depth, excavation of all rock caverns in the central portion of the underground facility and detailed characterization of the repository area for initial operation.

A period of 4 years is normally required for surface facilities and the connecting road/railway. Even if construction of considerably greater scope should be necessary, for example if many kilometres of connecting railway are needed, it should be possible to fit this into the 5-6 years indicated in the time schedule. This, however, requires that a decision on such construction be made at the beginning of the phase. If the decision should be delayed, the whole phase may be delayed by several years. Conversely, an early decision on construction, in combination with uncomplicated geological conditions and rational investigation methodology, can shorten the phase by a year or so.

15.3 COSTS AND PRIORITIES

The costs for execution of SKB's programme are presented annually in a PLAN report /15-3/. These costs will not be presented in detail here. However, in assessing the programme it can be of some interest to have an idea of the magnitude of the costs and the planned priorities in the RD&D work. Priorities will change as results come in, and it is difficult to make an exact estimate for each individual area over a long period of time. As in preceding programmes, a rough preliminary estimate is made of the costs for the coming six-year period.

Table 15-1 contains a preliminary cost estimate for the programmes described in chapters 7 and 9 for the encapsulation plant and the deep repository.

Table 15-1. Preliminary estimate of costs for encapsulation plant and deep repository – MSEK in August 1995 prices.

Year	1996	1997	1998	1999	2000	2001
Encapsulation plant	90	90	80	80	150	180
Deep repository	110	150	180	230	180	130

Table 15-2 contains a preliminary cost estimate for the programmes described in chapters 11-14 for supportive R&D, the Äspö HRL, other waste, alternative methods and decommissioning studies.

Table 15-2. Preliminary estimate of costs for R&D work during the period 1996–2001 – MSEK in August 1995 prices.

Year	1996	1997	1998	1999	2000	2001
Supportive R&D	50	48	46	44	42	40
Äspö HRL	60	60	70	70	50	50
Other waste incl. SFR	5	5	5	5	5	5
Alternative methods	3	3	3	3	3	3
Decommissioning	1	1	1	1	1	1

Table 15-3 gives an approximate percentage breakdown of the costs for supportive R&D for the different subject areas.

Table 15-3. Approximate breakdown of costs for supportive R&D 1996–2001.

	Share %
Safety assessment methods	14
Fuel	31
Other material	10
Geoscience	21
Chemistry	12
Natural analogues	8
Biosphere	4

An indication of the precision in the above estimates can be obtained by comparing the actual cost outcome with previous estimates made in R&D-programmes 86 and 89.

In R&D-Programme 86 /15-4/, the costs for the programme period 1987-1992 were estimated at about SEK 600 million in August 1986 prices. Of this amount, SEK 175 million was said to be the estimated cost for what was to be the Äspö HRL. The money actually spent for the programme during the same period amounted to about SEK 900 million in current money prices or about SEK 695 million in August 1986 prices. (This includes about SEK 80 million in foreign contributions to the phase 3 of the Stripa project.) The costs for Äspö during the period were about SEK 215 million in August 1986 prices. Thus, the costs for Äspö during the six-year period 1987-1992 were slightly higher, and for other RD&D slightly lower, than estimated at the beginning of the programme period. The overall outcome is roughly in agreement with the original estimate, which did not contain a foreign share in Stripa.

In R&D-Programme 89 /15-5/, the costs for the programme period 1990-1995 were estimated at about SEK 760 million in August 1989 prices. The actual outcome is expected to be about SEK 1,040 million in current money prices or about SEK 840 million in August 1989 prices. (Here again, the costs of foreign participation in Stripa and Äspö are included at about SEK 67 million.) The costs for Äspö during this period are about SEK 500 million (of which about SEK 280 million is construction cost) in current money prices or about SEK 400 million in 1989 prices. For the programme period 1990-1995, it appears as if the total outcome will be roughly as predicted at the beginning of the period. (The project costs for the encapsulation plant and the deep repository since 1993 are not included in these values since they were not included in the 1989 programme.)

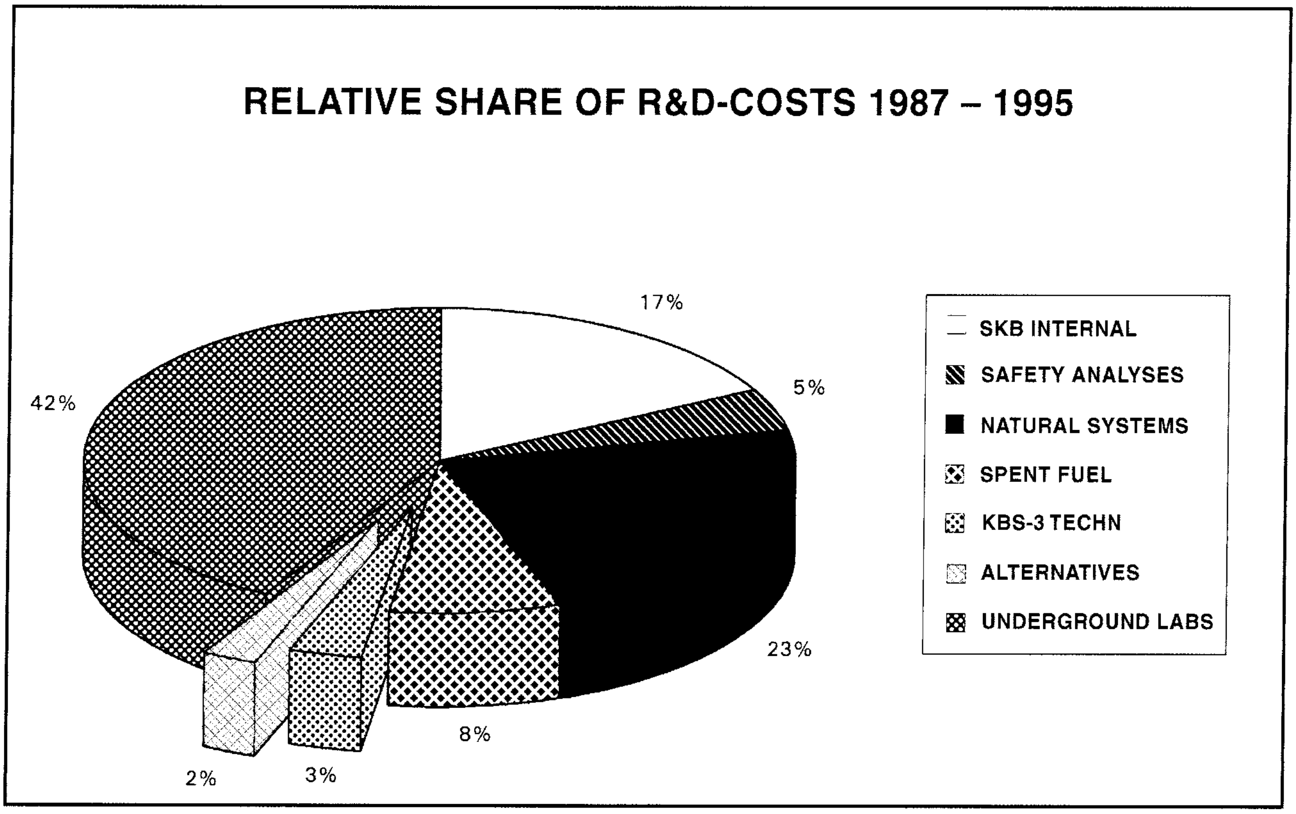


Figure 15-1. *Relative breakdown of R&D costs during the period 1987–1995.*

It may also be of interest to have a look at the breakdown of the actual outcome of R&D costs.

Figure 15-1 shows an overall breakdown of R&D costs within SKB's programmes, including international projects headed by SKB during the period 1987-1995, i.e. the two overlapping six-year periods included in the 1986 and 1989 programmes.

The R&D work that has been done pertaining to safety assessments and natural systems has provided general knowledge that is applicable to deep disposal of different waste forms in Swedish crystalline basement. The same applies to the most of the work at Stripa, Äspö and URL. Only a limited portion of the Stripa activities were specifically concerned with KBS-3. Most of the work on fuel is applicable to deep disposal of unreprocessed nuclear fuel in Swedish crystalline rock, while only some of it is specific for KBS-3.

The conclusion is that by far most of the R&D work done during the period 1987-1995 is generally applicable to deep disposal of spent nuclear fuel in Swedish crystalline basement – cf. section 3.1.2. Only a limited portion has related to specific methods such as KBS-3.

The cost outcome has been broken down into the following categories in the figure:

- **SKB internal =**
  salaries, office rents and utilities, travel costs etc. for the personnel within SKB that have been involved in the programme as well as for computers and documentation,

- **Safety analyses =**
  external costs for development of methods and for evaluation and execution of safety assessments,

- **Natural systems =**
  external R&D costs for geoscience, natural analogues, chemistry, biosphere, instrument development but excluding Stripa and Äspö,

- **Spent fuel =**
  external R&D costs for investigations of spent fuel,

- **KBS-3 technology =**
  external R&D costs directly attributable to the KBS-3 concept, i.e. canister, buffer and design,

- **Alternatives =**
  external R&D costs attributable to other concepts than KBS-3, including alternative treatment methods,

- **Underground labs =**
  costs for Stripa, Äspö and SKB's participation in URL, Canada; construction costs Äspö not included.

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