SKB

TECHNICAL REPORT

95-37

SKB ANNUAL REPORT 1995

Including Summaries of Technical Reports Issued during 1995

Stockholm, May 1996

SVENSK KÄRNBRÄNSLEHANTERING AB

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FOREWORD

The Annual Report on SKB's activities during 1995 covers planning, construction and operation of facilities and systems as well as research, development, demonstration work and information activities.

SKB has an operating and well integrated system for handling of all radioactive residues within Sweden. The central repository for low and medium level waste, SFR, and the central interim storage facility for spent nuclear fuel, CLAB, are in operation and can take care of all radioactive waste produced inside Sweden for a long time ahead.

For the remaining facilities – the encapsulation plant and the final repository for spent nuclear fuel – comprehensive research, development and planning activities are well under way. The aim of the programme is to start the permanent disposal of spent nuclear fuel around the year 2008. Work is undertaken for the development of encapsulation technology on an industrial scale and for design of an encapsulation plant. The siting process for the final repository for spent fuel has started with feasibility studies in a few Swedish municipalities in order to evaluate the potential technical conditions and requirements and the influence on the region.

International co-operation and exchange of information in all fields of the back-end of the nuclear fuel cycle are important and of great value for SKB's work. We are pleased to note the extensive international interest for participation in our Äspö Hard Rock Laboratory. We hope this Annual Report will be of interest and that it will enhance the international information exchange.

Stockholm in June 1996

SWEDISH NUCLEAR FUEL AND WASTE MANAGEMENT CO – SKB

Sten Bjurström
President

ABSTRACT

This is the annual report on the activities of the Swedish Nuclear Fuel and Waste Management Co, SKB. It contains in part I an overview of SKB activities in different fields. Part II gives a description of the research and development work on nuclear waste disposal performed during 1995.

Lectures and publications during 1995 as well as reports issued in the SKB technical report series are listed in part III. Part III also contains listing of consultants which have contributed to the SKB work and of post-graduate theses supported by SKB.

Part IV contains the summaries of all technical reports issued during 1995.

SKB is the owner of CLAB, the Central Interim Storage Facility for Spent Nuclear Fuel, located at Oskarshamn. CLAB was taken into operation in July 1985 and to the end of 1995 in total around 2 300 tonnes of spent fuel (uranium weight) have been received. Transportation from the nuclear sites to CLAB is made by a special ship, M/S Sigyn.

At Forsmark the Final Repository for Radioactive Operational Waste – SFR – was taken into operation in April 1988. The repository is situated in crystalline rock under the Baltic Sea. The first construction phase includes rock caverns for 60 000 m^3 of waste. At the end of 1995 a total of 18 500 m^3 of waste have been deposited in SFR.

SKB is in charge of a comprehensive research and development programme on geological disposal of nuclear waste. The total cost for R&D during 1995 was 139.8 MSEK of which 17.9 MSEK were investments in the Äspö Hard Rock Laboratory.

Some of the main areas for SKB research are:

- Groundwater movements.
- Bedrock stability.
- Groundwater chemistry and nuclide migration.
- Methods and instruments for in situ characterization of crystalline bedrock.
- Characterization and leaching of spent nuclear fuel.
- Properties of bentonite for buffer, backfilling and sealing.
- Radionuclide transport in biosphere and dose evaluations.
- Development of performance and safety assessment methodology and assessment models.
- Operation of an underground research laboratory.

Geological site-investigations are a substantial part of the programme. In the Äspö Hard Rock Laboratory methodologies for characterizing rock are refined and evaluated. In March 1996 there are 9 foreign organizations participating in the Äspö HRL project.

SKB is planning to build an encapsulation plant for spent nuclear fuel and a deep repository for the encapsulated fuel and other long-lived waste. The encapsulation plant is proposed to be built adjacent to the CLAB facility. In the encapsulation plant the spent fuel will be encapsulated in a copper/steel canister. During 1995 conceptual design work was continuing for the facility. Also development work for the manufacturing and closing of the copper canister was performed.

Siting activities for a deep repository included feasibility studies in the municipalities of Storuman, Malå, Östhammar and Nyköping. Technical, geoscientific and socioeconomic studies were performed. Extensive local involvement and discussions is an important part of the siting activities. After completion of the feasibility study at Storuman the inhabitants voted against continued work on siting in Storuman.

Cost calculations for the total nuclear waste management system, including decommissioning of all reactors, are updated annually. The total cost is estimated to 59 billion SEK.

Consulting services from SKB and associated expert groups are available on a commercial basis. From the start of these services in 1985 and up to the end of 1995 more than 120 assignments have been accomplished in a variety of areas.

Information activities are an integrated and important part of the Swedish radioactive waste management system. During 1995 successful public information activities have been carried out using mobile exhibitions in a tailor-made trailer and on the SKB ship M/S Sigyn. SKB's school programme has been very well received in a large number of schools.

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SKB ANNUAL REPORT 1995

Part I

Overview of SKB Activities

CONTENTS PART I

1 GENERAL BACKGROUND

1.1 THE SWEDISH NUCLEAR POWER PROGRAMME

The nuclear power programme of Sweden consists of 12 nuclear reactors located at four different sites and with a combined capacity of 10 000 MW net electric power. Main data and location of the 12 units are shown in Figure 1-1. The nuclear power plants generated about 47% of the total Swedish electric power produced in 1995.

Swedish reactors

Reactor		Power MWe	Commercial operation	Energy availability in 1995 %
Oskarshamn 1	BWR	445	1972	–
Oskarshamn 2	BWR	605	1974	84
Oskarshamn 3	BWR	1160	1985	90
Barsebäck 1	BWR	600	1975	78
Barsebäck 2	BWR	600	1977	76
Ringhals 1	BWR	795	1976	82
Ringhals 2	PWR	875	1975	88
Ringhals 3	PWR	915	1981	62
Ringhals 4	PWR	915	1983	88
Forsmark 1	BWR	970	1980	92
Forsmark 2	BWR	970	1981	92
Forsmark 3	BWR	1160	1985	93

1.2 LEGAL AND ORGANIZATIONAL FRAMEWORK

The nuclear power plants are owned by the following four companies:

- Vattenfall AB is the largest electricity producer in Sweden and owns the Ringhals plant.
- Barsebäck Kraft AB (subsidiary of Sydkraft AB) is the owner of the Barsebäck plant.
- OKG AB is the owner of the Oskarshamn plant. Sydkraft is the major shareholder of OKG.
- Forsmark Kraftgrupp AB (FKA) is the owner of the Forsmark plant. Vattenfall has 74.5% of the shares in FKA.

The Swedish Nuclear Fuel and Waste Management Company, SKB (SKB = Svensk Kärnbränslehantering AB) has been formed by these four power utilities. SKB shall develop, plan, construct and operate facilities and systems for the management and disposal of spent nuclear fuel and radioactive wastes from the Swedish nuclear power plants. On the behalf of its owners SKB is responsible for all handling, transport and storage of the nuclear wastes outside of the nuclear power production facilities.

SKB is also in charge of the comprehensive research programme in the waste field which the utilities are responsible for according to the law. Finally SKB handles stockpiling of uranium for the Swedish nuclear power industry and provides assistance at the request of its owners in fuel cycle services.

The total central staff of SKB is about 85 persons. The organization is shown in Appendix 1. For the bulk of the work a large number of organizations and individuals outside SKB are contracted. As a whole about 700 persons are involved in SKB waste handling and research work.

SKB is the organization that has the lead operative role in the Swedish waste management programme both with respect to planning, construction and operation of facilities and systems and with respect to research and development. The role has its roots in the legislation briefly described below. Figure 1-2 gives an overview of the most important laws and the corresponding authorities involved.

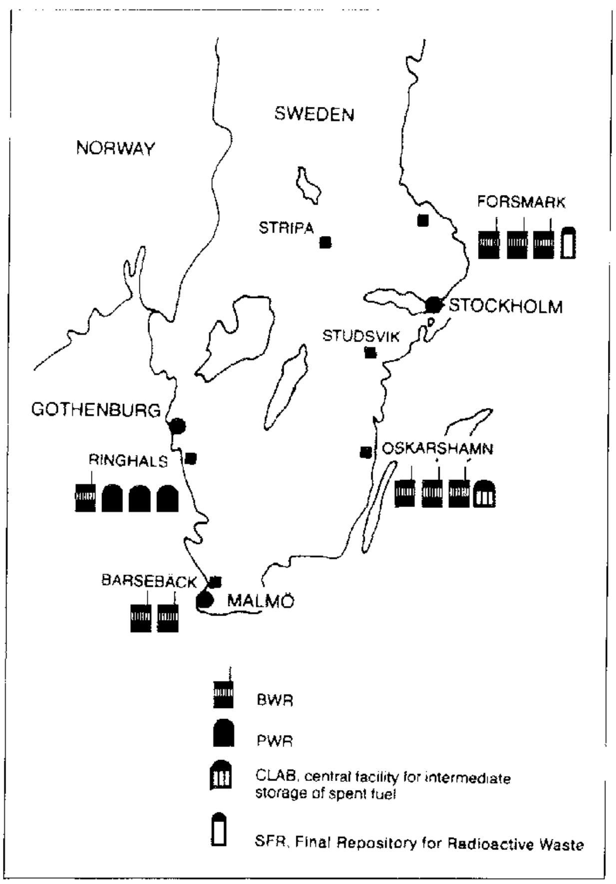

Figure 1-1. The Swedish nuclear power programme.

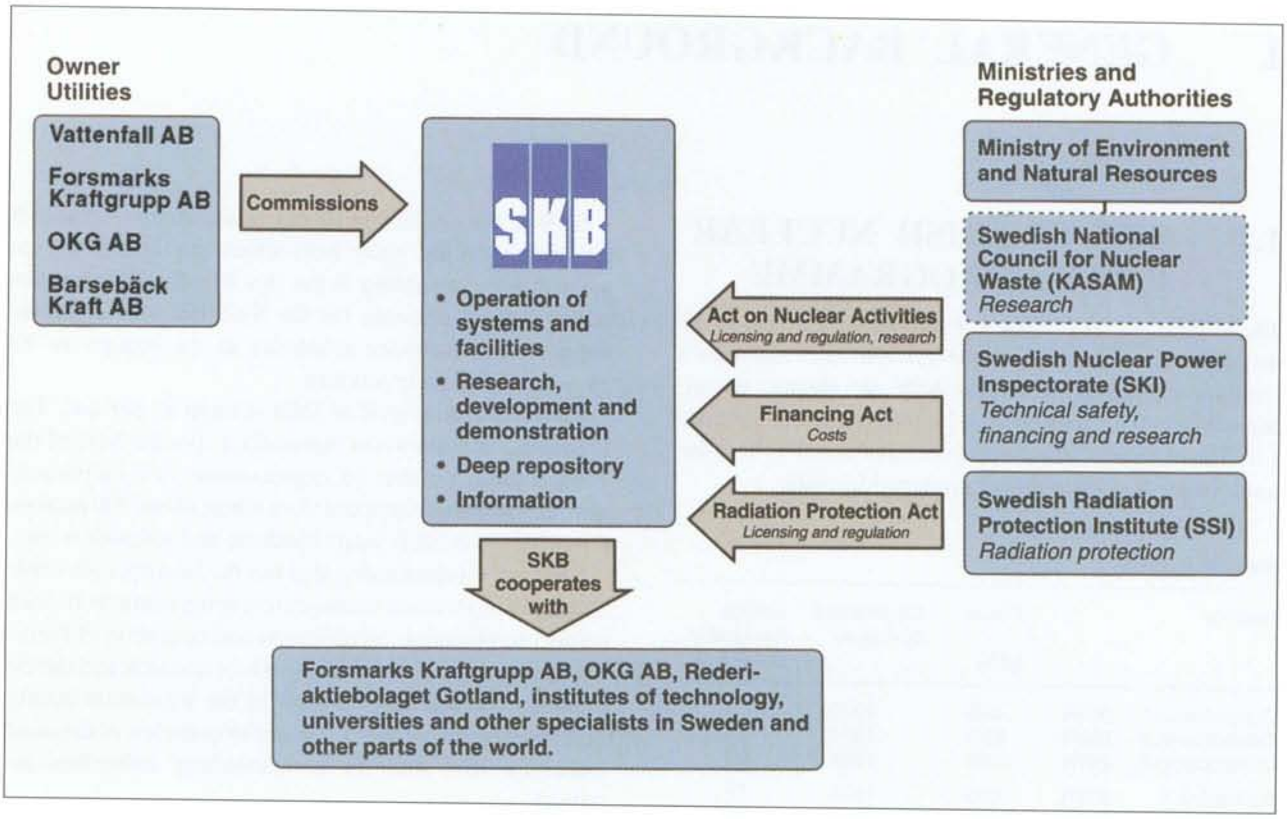

Figure 1-2. *Legal framework for activities of SKB.*

There are three important laws which regulate the nuclear activities.

- The Act on Nuclear Activities.
- The Act on the Financing of Future Expenses for Spent Nuclear Fuel etc.
- The Radiation Protection Act.

The Act on Nuclear Activities /1-1/ puts the primary responsibility for the safety on the owner of a nuclear installation. The owner is thus responsible for safety during design, construction and operation of nuclear facilities, for the handling and final disposal of nuclear wastes and for the dismantling and decommissioning of the facility. The responsibility also includes the necessary research and development in the waste management field. According to the act a programme covering necessary future measures must be submitted to the authorities every three years. The first programme was submitted in September 1986, the second in September 1989, the third in September 1992 and the fourth in September 1995.

The authority for supervision of the safety provisions in the Act on Nuclear Activities as well as the SKB research programme is the Swedish Nuclear Power Inspectorate (SKI). The Swedish Radiation Protection Institute (SSI) is supervising provisions of the Radiation Protection Act.

The SKI is also supervising the adherence to the Act on Financing of Future Expenses for Spent Fuel. According to this law the waste management activities including future decommissioning of all reactors are financed from funds built up from fees on the nuclear power production.

The fees are revised annually by SKI, which proposes the fees for the next year to the government. The average fee on nuclear electricity has since 1984 been 0.019 SEK per kWh.

The radiation protection act contains basic rules for protection against ionizing radiation for

- those who work at nuclear installations and other facilities with potential radiation hazards,
- the general public who lives or stays outside such installations or facilities.

The competent authority in these matters is the Swedish Radiation Protection Institute (SSI).

The authorities have separate funds for the research needed to fulfil their obligations.

1.3 THE SWEDISH NUCLEAR WASTE MANAGEMENT SYSTEM

A complete system has been planned for the management of all radioactive residues from the 12 nuclear reactors and

Table 1-1. Waste categories.

	WASTE CATEGORY	ORIGIN	WASTE FORM	PROPERTIES	QUANTITY
1	Spent fuel	Operation of nuclear reactors	Fuel rods encapsulated in canisters	High activity level. Contains long-lived nuclides	4 500 canisters (7 800 tU)
2	Transuranic-bearing waste	Waste from the Studsvik research facility	Solidified in concrete	Low- to medium-level. Contains long-lived nuclides	c. 2 000 m^3
3	Core components and internals	Scrap metal from inside reactor vessels	Untreated or cast in concrete	Low- to medium-level. Contains some long-lived nuclides.	c. 10 000 m^3
4	Reactor waste	Operating waste from nuclear power plants etc.	Solidified in concrete or bitumen. Compacted waste	Low- to medium-level. Short-lived	c. 90 000 m^3
5	Decommissioning waste	From dismantling of nuclear facilities	Untreated for the most part	Low- to medium-level. Short-lived	c. 100 000 – 150 000 m^3 *

* The amount of decommissioning waste to be disposed of will depend on how much material that will be decontaminated.

from research facilities. The system is based on the projected generation of waste up to the year 2010.

Residues generated by the operation of the reactors are spent nuclear fuel and different kinds of low- and medium level wastes. Furthermore, in the future decommissioning waste will be generated when the reactors and other facilities are dismantled.

The types and total quantities of various nuclear waste categories currently estimated to be generated are given in Table 1-1. The basic strategy for the management of the waste categories is that short-lived wastes should be deposited as soon as feasible, whereas for spent fuel and other long-lived wastes an interim storage period of 30–40 years is foreseen prior to disposal.

The main features of the planned system for nuclear waste management in Sweden are shown in Figure 1-3.

The Swedish Final Repository for Radioactive Operational Waste, SFR, was taken into operation in 1988. The capacity at present is 60 000 m^3. SFR may later on be extended to accommodate waste also from the decommissioning of the nuclear reactors. For spent fuel a central interim storage facility, CLAB, was taken into operation in July 1985. This facility has a current capacity of 5 000 tonnes of spent fuel.

The spent fuel will be stored in CLAB for about 40 years. It will then be encapsulated in a corrosion-resistant canister and deposited at depth in the Swedish bedrock. According to the time schedule presented in the RD&D-Programme 92 SKB plans to expand the CLAB facility with an encapsulation plant in order to make encapsulated fuel available for disposal around 2008.

The construction of the deep repository will be made in steps. A first stage of the repository, for 5 – 10% of the fuel, is planned to be put in operation in 2008. The next stage for the full repository will only be built after a thorough evaluation of the experiences of the first stage and a renewed licensing. The site for the deep repository has not yet been chosen.

For the transport of spent fuel and other kinds of radioactive wastes a sea transport system is in operation since 1982.

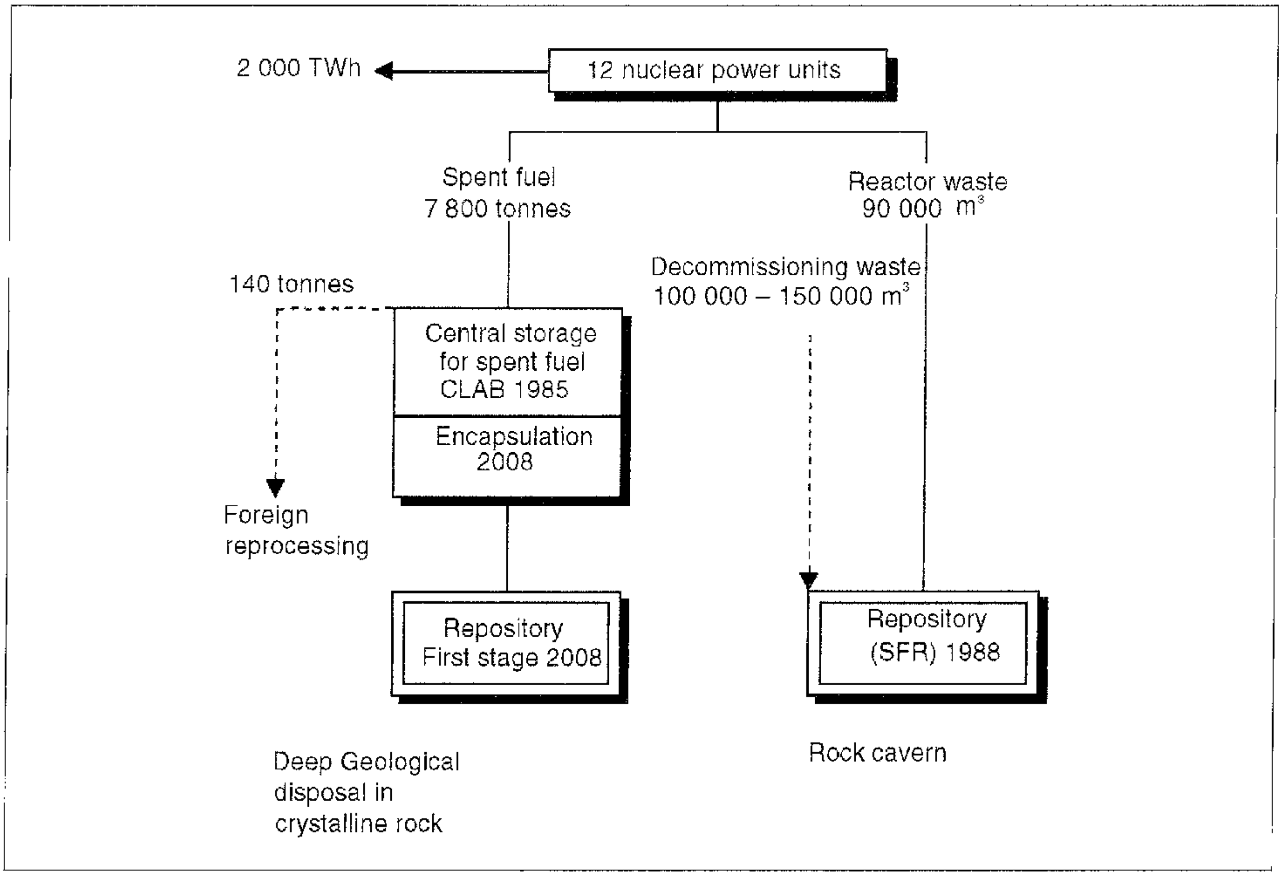

Figure 1-3. *Main system for management of radioactive waste in Sweden.*

2 CENTRAL INTERIM STORAGE FACILITY FOR SPENT NUCLEAR FUEL, CLAB

2.1 GENERAL

The Central Interim Storage Facility for Spent Nuclear Fuel, CLAB, located on the Simpevarp peninsula adjacent to the Oskarshamn nuclear power station, was taken into active operation in July, 1985, see Figure 2-1.

The facility has five underground pools with a storage capacity of 5000 tonnes of uranium (tU). The receiving building and the buildings for auxiliary systems and offices are located on ground level. The facility is designed to receive at least 300 tU per year, equivalent to about 100 fuel transport casks, and some 10-20 casks containing highly active reactor core components, see Figure 2-2. For the operation SKB has contracted OKG Aktiebolag, one of SKB's shareholders, operating three reactors at the site.

2.2 OPERATING EXPERIENCES

By the end of 1995 CLAB had been in operation for 10,5 years and the performance of the facility has been excellent since the start of operation. Improvements have gradually been introduced along with the experiences gained. In total around 2300 tU from the 12 Swedish reactors have been shipped to the facility and placed in storage.

In 1995, 64 casks containing spent fuel assemblies from the Swedish reactors were received together with 3 reactor core component canisters. The total fuel quantity shipped to CLAB during the year amounted to 196 tU. In parallel to the fuel receiving activities 281 BWR assemblies have been transferred from old canisters to new compact storage canisters, see section 2.3. By the end of the year about

Figure 2-1. The Oskarshamn Nuclear Site. CLAB in the foreground.

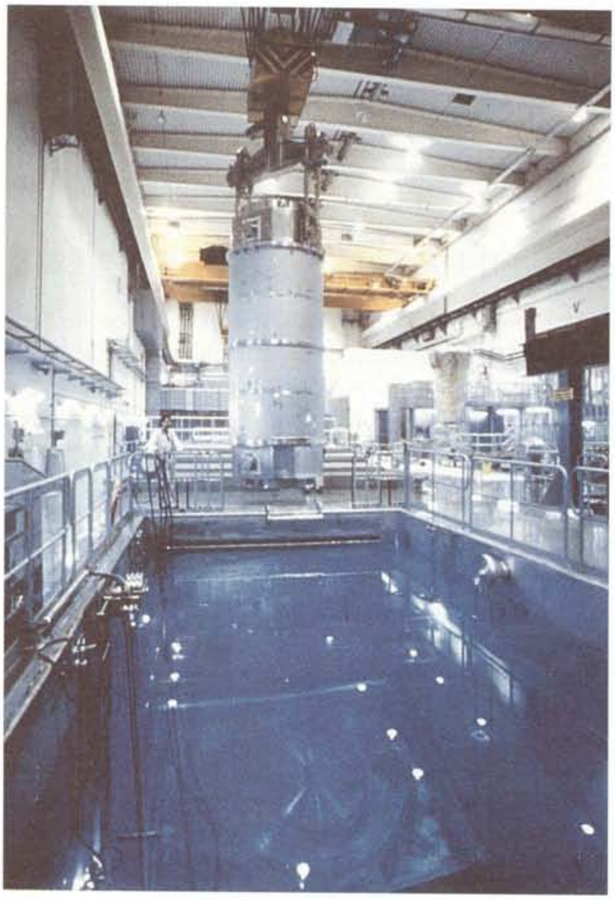

Figure 2-2. Fuel cask with protective shirt being moved from the cooling cell to the unloading pool.

2/3 of the PWR fuel and 1/3 of the BWR fuel was stored in the new canisters.

The total occupational dose in 1995 was 84 mmanSv, which is 19 mmanSv less than in 1994 and corresponds to about 30% of the value calculated in the safety assessment made during the design phase.

The release of radioactivity to the environment during 1994 has been negligible, less than 0.01% of the permissible release from CLAB and the three adjacent reactors together.

The flexibility of the plant has been demonstrated by the fact that other transport casks than the normally used standard cask have been used for shipments to CLAB at several occasions. E.g. a cask built in the 1960's is used for the transfer of post irradiation examination residues from the Studsvik Nuclear Research Centre. The operating procedures and involved equipment have been quite easily adapted to the different casks.

The experiences from more than ten years of operation are continuously used in the project work on the Encapsulation Plant, see section 6, which according to current plans, will be built wall to wall with CLAB.

The control system of one of the fuel handling machines has been exchanged as a part of a continous plant modernization programme.

The public's interest in CLAB has increased and in 1995 around 12 000 persons visited the facility.

2.3 INCREASED STORAGE CAPACITY

The storage capacity of the pools was originally 3000 tU, which would cover the need until 1996. Preparations for a future expansion with additional caverns and pools were made during the construction of the facility in the early eighties. A study performed in 1988 showed that there was a great advantage if the expansion could be postponed by better utilization of the space available in the existing pools.

This has been achieved by using new compact storage canisters with borated stainless steel as neutron absorbing material, allowing the number of fuel assemblies in a canister to be increased from 16 to 25 and from 5 to 9 per canister for BWR respectively PWR fuel, see Figure 2-3. The new canisters have been in regular use since 1992 for all fuel arriving from the reactors and for fuel unloaded from the old type canisters. These used old canisters are decontaminated and conditioned before being shipped away from the facility. Due to the better storage capacity, a new cavern with pools will not be needed until around 2004. In 1995 the design work on this expansion of CLAB started aiming at construction start around the turn of the century.

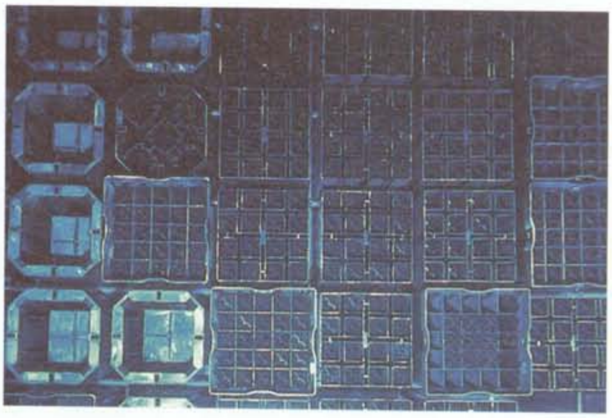

Figure 2-3. BWR-fuel canister of the old and new types containing 16 respectively 25 fuel assemblies. The octagonal canisters are used for highly radioactive reactor core components.

3 TRANSPORTATION

3.1 GENERAL

The sea transportation system consists of the specially designed ship M/S Sigyn, 10 transport casks for spent fuel, 2 transport casks for core components, 27 IP-2 containers (ATB) for transport of low- and intermediate level waste and 5 terminal vehicles. One of the vehicles is specially designed for operation in the SFR repository, see Figure 3-1.

SKB has engaged the shipping line Rederiaktiebolaget Gotland to operate Sigyn.

3.2 OPERATING EXPERIENCES

In 1995 the ship M/S Sigyn sailed around 38 500 n.m. during 157 days. The transports of spent fuel and reactor waste from the Swedish reactors to the CLAB facility and to the repository, SFR, have been performed without disturbances. In total 57 transport casks with spent fuel, 97 IP-2 containers (ATB) with reactor waste and 51 ISO containers with low level waste have been transported with the transportation system during the year. Like earlier years, no measurable dose rates have been registered to the ship's crew.

During the first half of 1995 two transport cask for spent fuel were rebuilt at the CLAB facility with extra over-shielding for transport of high burn up fuel. The work was finished in August and the two casks were taken into operation during autumn 1995.

A new license for the spent fuel cask has been approved by French and Swedish authorities allowing transport of fuel elements with higher enrichment and high burn up.

When the ordinary transport schedule has permitted, M/S Sigyn has been used on commercial basis for transports of heavy equipment. During 1995 seventeen different transports with heavy equipment have been transported with M/S Sigyn.

During the summer period M/S Sigyn was used, like earlier years, as a floating exhibition of the Swedish nuclear waste handling system making a voyage along the Swedish coast and visiting 20 harbours, including the capital Stockholm.

Figure 3-1. Unloading of waste from M/S Sigyn to SFR.

4 FINAL REPOSITORY FOR RADIOACTIVE OPERATIONAL WASTE, SFR

4.1 GENERAL

The Final Repository for Radioactive Operational Waste, SFR, was put into active operation in April, 1988. It is a repository for low- and intermediate level waste, built in the bedrock under the Baltic Sea close to Forsmark nuclear power plant. 60 metres of rock covers the repository caverns under the sea bed, see Figure 4-1. The first stage of SFR, which is in operation, includes buildings on ground level, tunnels, operating buildings and disposal caverns for 60 000 m^3 of waste. A second stage for approximately 30 000 m^3 is planned to be built and commissioned after the year 2000.

The total amount of waste from the Swedish program up to year 2010 has been calculated to about 90 000 m^3.

All waste materials are conditioned at the power plants and CLAB or at the nuclear research centre, Studsvik. Ion exchange resins are incorporated in either cement or bitumen. Scrap from maintenance work are treated in the same way, if required.

At the end of 1995 a total of 18 500 m^3 of waste have been deposited in SFR. All waste producers have delivered waste. The experience from the operation has been good and the radiation doses to the personnel have been very low.

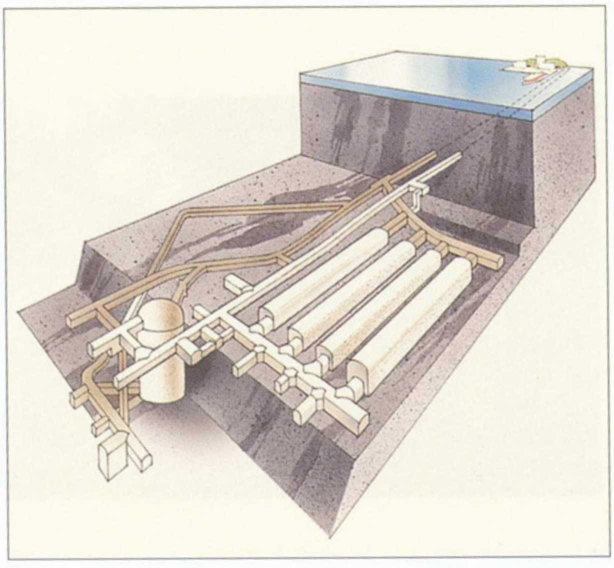

Figure 4-1. Overview of tunnels and storage chambers in the first construction stage of SFR.

4.2 DESIGN AND CONSTRUCTION

The SFR has been sited under the sea in order to minimize the groundwater flow in the repository area. Engineered barriers are used in order to further reduce the ground water flow inside the caverns and through the waste.

There are different caverns for ILW and LLW in SFR. The ILW-packages containing most of the activity are disposed of in a concrete silo structure and surrounded with a low permeable buffer material, bentonite. The space between the waste packages and the concrete construction in the silo are subsequently filled with a porous concrete.

Waste containing a minor part of the activity content are disposed of in 160 m long caverns with various cross sections.

The cavern with the largest cross section, BMA, is equipped with machines for remotely controlled handling, similar to those used in the silo, see Figure 4-2.

LLW is handled with an ordinary, but shielded, forklift truck.

4.3 WASTE ACCEPTANCE

As stipulated in the operational permits all waste that is deposited in SFR should belong to a waste type that has received an approval by the safety authorities. A procedure for the description and approval of waste types has been developed.

All relevant information about each waste package is documented and collected in a computerized waste register. Before the waste is transported to SFR, the contents of the waste register is transferred to a SFR-data base.

The procedure for waste acceptance has been very time consuming. In 1995, 38 waste types (of a total of about 50) were accepted for disposal. In 1995 disposal has been carried out in the rock chambers and in the silo.

4.4 SAFETY ASSESSMENT

In May 1992 a complementary operational permit was granted, which allows also the disposal of waste in the silo on a regular basis and the subsequent grouting around the waste. As a basis for this permit, SKB had in August 1991 presented a deepened Safety Assessment to the authorities. This was in accordance with the conditions of the original operating permit from 1988.

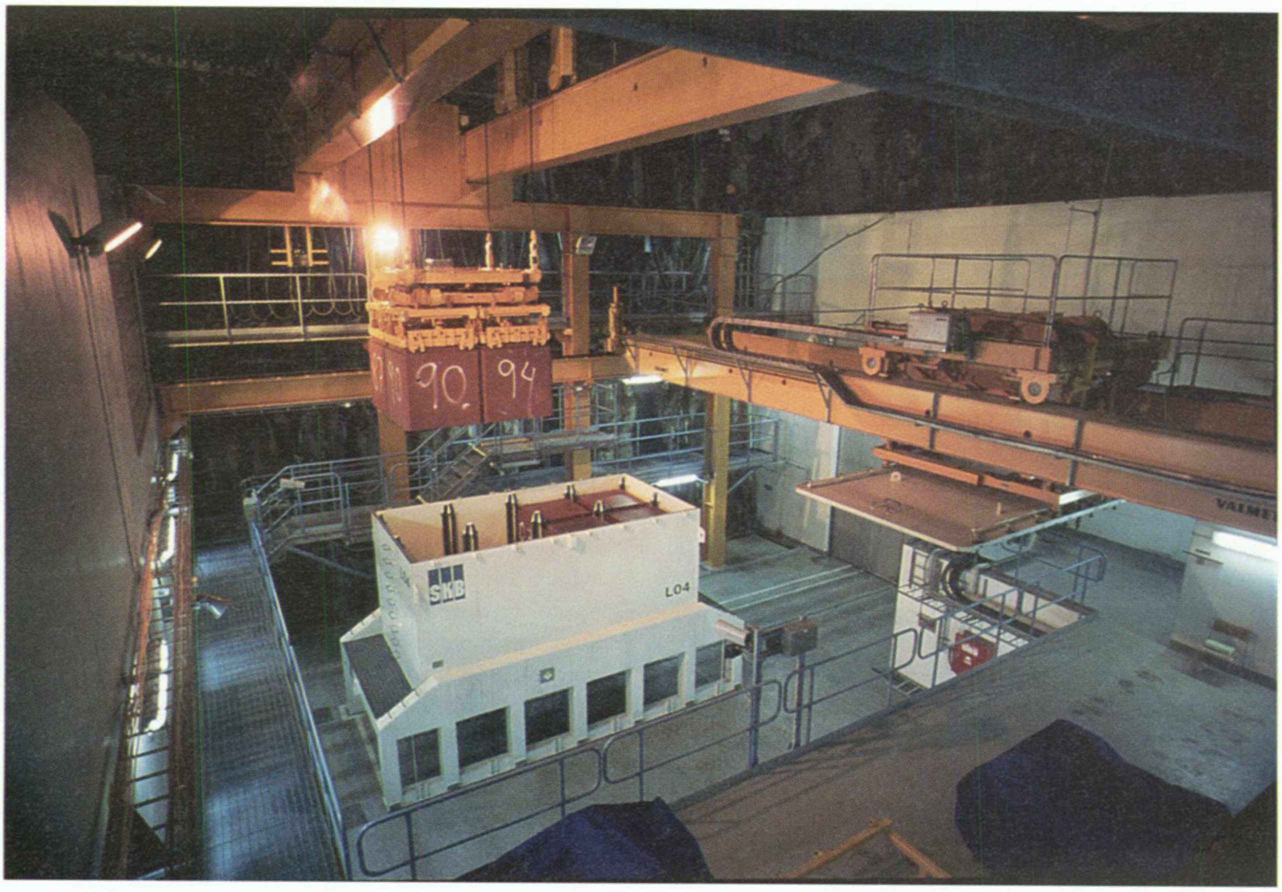

Figure 4-2. The operational waste is transported in special transport containers. In SFR the waste packages are unloaded with remote-controlled handling equipment.

Some areas that are covered in detail in the deepened safety assessment are the effects of gas production, the effect of complexing agents from the degradation of cellulose and the change in the hydrological regime due to land rise. Also a systematic scenario analysis is included. The results of this deepened safety assessment confirmed the results of the Final Safety Report.

4.5 OPERATION

The operation of SFR has been subcontracted to Forsmark Kraftgrupp AB (FKA), the operator of the nuclear reactors at Forsmark, and is closely integrated in the local organization. The staff for operation and maintenance of SFR consists of about 16 people.

In full operation the facility has an annual disposal capacity of about 6000 m^3. Up till the end of 1995 a total of 18 500 m^3 of waste has been deposited. During 1995, 2960 m^3 was deposited in SFR compared to planned 2500 m^3 according to a firm price agreement which have been established between FKA and SKB.

For 1996, 2000 m^3 of waste is foreseen to be transported and deposited in SFR.

The operating experience is good both with regard to handling and availability.

All activities down in SFR are directed and supervised from the operations centre that is located in a building

Figure 4-3. Each year SFR receives more than 20 000 visitors. An exhibition facility has been built underground in a cavern adjacent to the repository caverns.

underground, centrally in the repository area. The operations centre contains equipment for remote control of all handling machines, overhead cranes with waste and of the auxiliary systems, etc.

A new data base system for SFR has been developed and will be taken into operation in 1995 and 1996.

A 10-year programme for corrosion preventing activities in the facility was started up in 1994 and continued in 1995.

5 DEEP REPOSITORY PROJECT

5.1 GENERAL

Siting and construction of a deep repository for final disposal of spent nuclear fuel and other long-lived waste is one of the main remaining tasks within the Swedish nuclear waste programme. In the RD&D-Programme 92 /5-1/ plans were presented for the work to start implementing the first stage of deep disposal by about the year 2008. In 1995 the Government clarified several issues in relation to the siting process and stated that siting criteria and planned studies as described by SKB should form the basis for siting. During 1995 the activities of SKB have been focused on:

- Further development and description of siting criteria and the siting process.
- Background and national overview studies concerning different aspects of siting a deep repository.
- Feasibility studies in cooperation with interested and potentially suitable municipalities.
- Technical studies of the repository system.
- Planning of a site-investigation programme.
- Environmental impact assessment studies.

5.2 THE PLANNED SITING PROCESS

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 a central role in this context.

In Sweden SKB has been conducting studies, such as extensive study site investigations, of geological conditions at depth in the Swedish bedrock since the mid 70s. Furthermore a number of safety assessments for deep repositories have been performed over the years. Based on these results and experiences SKB in 1992 began the work of actually siting the deep repository for spent nuclear fuel.

The work is now underway and much information and experiences have been gathered over the past few years.

Siting criteria have been reported to the government and they form the basis for siting activities. The criteria are structured into four main categories: Safety, Technology, Land & Environment and Society.

General siting studies covering the entire country have been published. They provide the general background and general conditions. They are also used to exclude certain parts of the country from further interest for siting.

Feasibility studies examine the prospects for a deep repository in potentially suitable and interested municipalities. The existing and planned land-use as well as the environmental factors and the societal aspects are clarified relatively thoroughly in the feasibility studies. Judgements of siting factors for safety and technology are based on general knowledge and data. The feasibility study provides a basis for judging if and where areas with good potential exist in the municipality. Geoscientific conditions, transport aspects and impact on local business and industry and the local community are analyzed and described.

Two feasibility studies have been completed in the municipalities of Storuman and Malå in the northern part of Sweden. In Storuman a local referendum resulted in a vote against further investigations. In Malå the municipality is planning a review of the results before possibly organizing a referendum. Feasibility studies are now also underway for two municipalities (Nyköping, Östhammar) in the southern part of Sweden. The plan is to make in total 5-10 feasibility studies of municipalities having potentially good conditions and being willing to participate in a study.

Site investigations are planned at a later stage for at least two sites in the country. They will be located in areas judged to be of particular interest on the basis of the feasibility studies and the general siting studies. A site investigation entails more extensive studies, including bedrock investigations in boreholes, and is estimated to take 4–5 years.

When at least two complete site investigations have been conducted, all relevant material from the siting work is compiled in an application under the Natural Resources Act for siting permission and the Nuclear Activities Act for permission to start constructing of a repository.

5.3 SITING ACTIVITIES

During 1995 siting activities have mainly involved general siting studies of the whole of Sweden, the completion of the two feasibility studies in Storuman and Malå in northern Sweden and the start of two feasability studies in southern Sweden in the municipalities of Nyköping and Östhammar.

Siting process

In RD&D-Programme 92 SKB gave an account of its strategy for the siting of a deep repository for the long-lived radioactive waste, including the spent nuclear fuel.

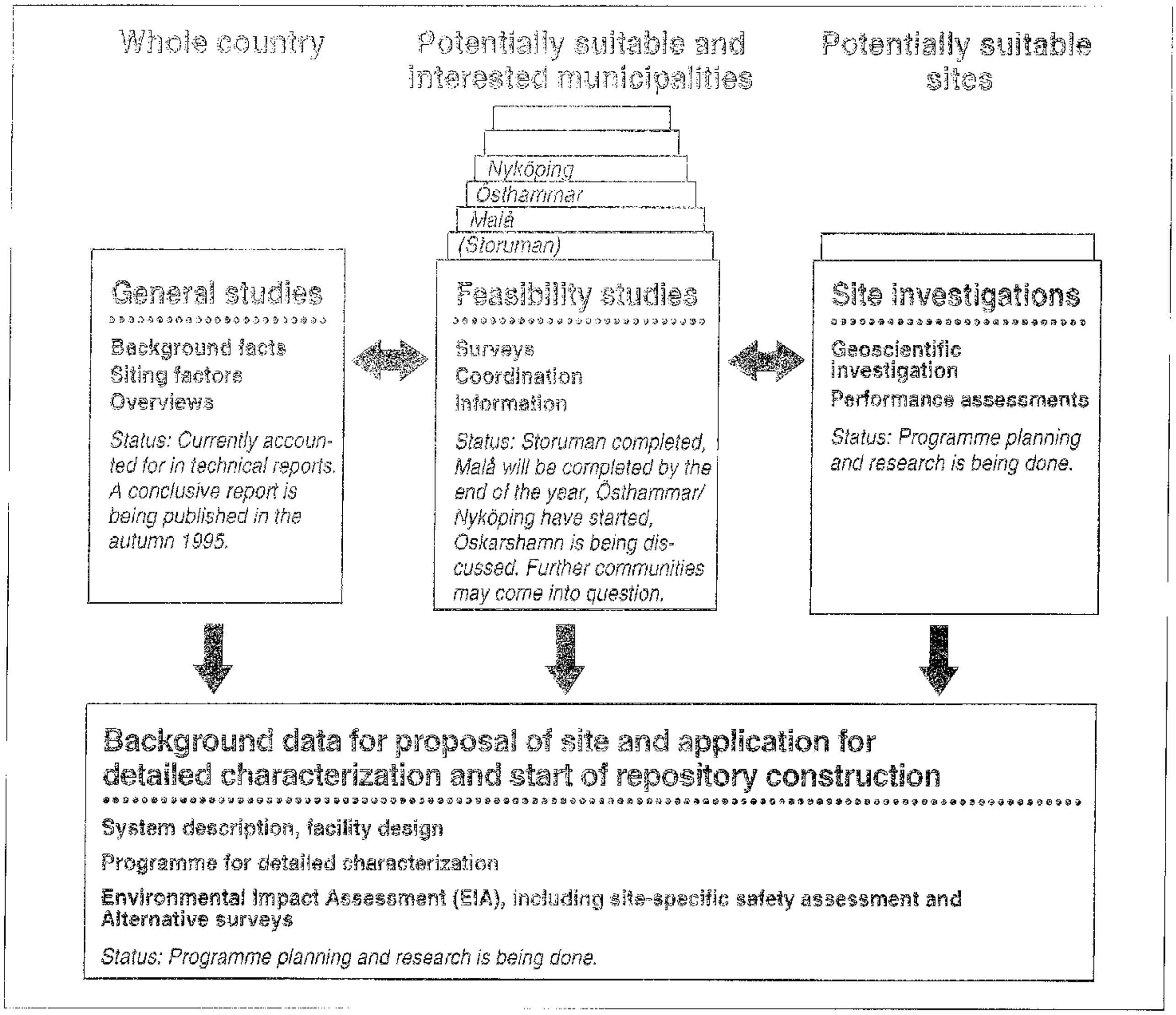

Figure 5-1. Main components of the siting process.

The strategy was accepted in all essential respects by the regulatory authorities and the Government. However, there was some criticism on certain unclear points in the programme regarding the criteria and methods that can form a basis for the selection of sites suitable for a final repository. In August 1994 SKB therefore presented the siting criteria in the RD&D-Programme 92 Supplement /5-2/ A summary of the main groups of siting factors is presented below.

In May 1995 the Government stated that the siting factors presented constituted a good base for the siting work. The Government also stated that the different steps of the siting program with feasability studies, site investigations and later detailed site investigations, were acceptable. The possibilities for the municipalities to gain knowledge, inform its inhabitants and independently review SKB's work was regulated so that each municipality subjected to feasability studies can recieve up to 2 MSEK/year from the national waste fund for these purposes. The role of the County Administration in the siting process was clarified by the Government. Their responsibility is to help coordinate contacts between SKB, municipalities with feasability studies and their neighbour municipalities as well as safety authorities and other concerned parties.

Siting criteria

The question of whether an area is suitable for siting of a deep repository is judged against the following main groups of siting factors:

Safety

Siting factors of importance for the long-term safety of the deep repository.

Technology

Siting factors of importance for the construction, performance and safe operation of the deep repository and its transportation system.

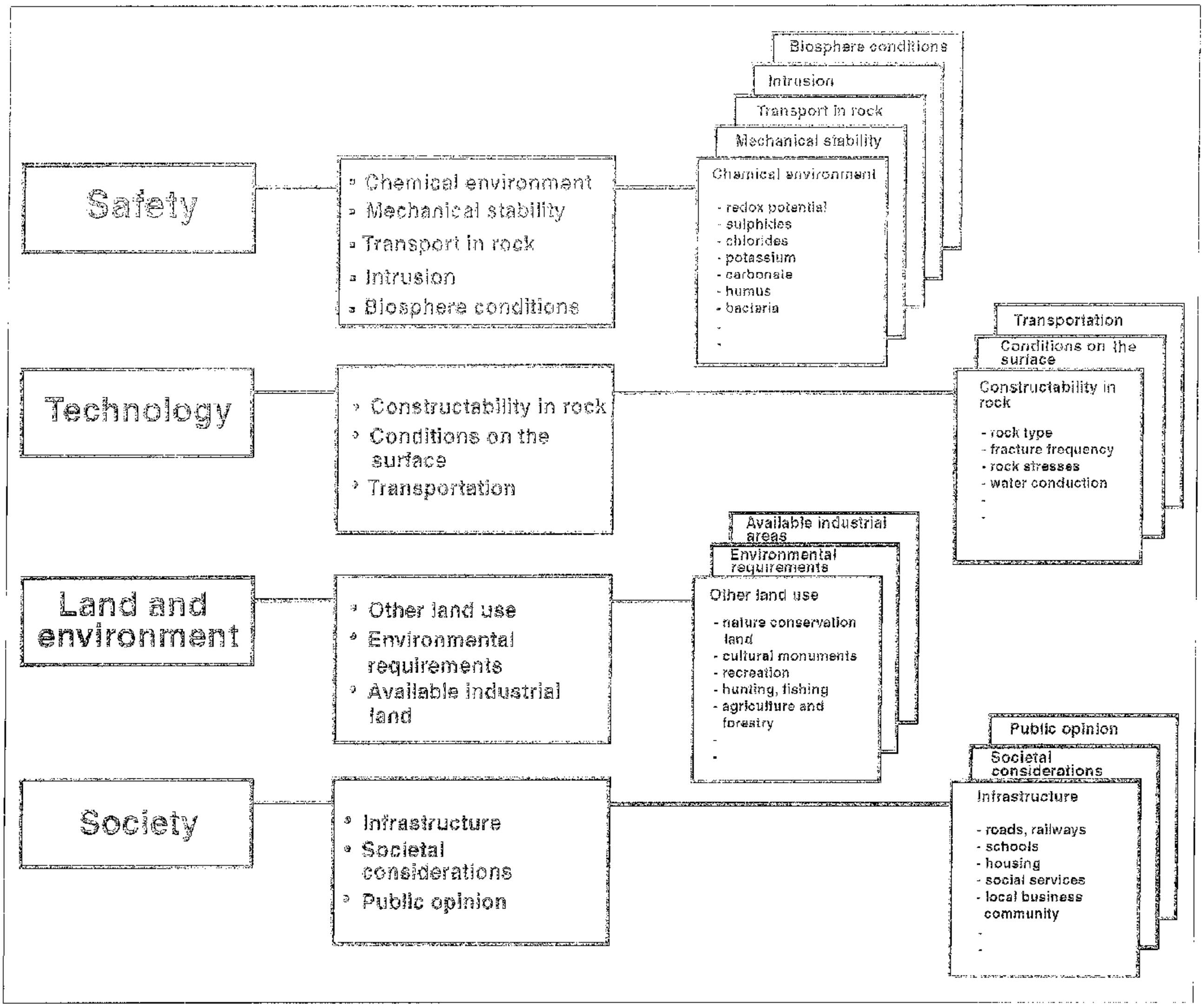

Figure 5-2. Structure for siting factors and criteria.

Land and environment

Siting factors of importance for land use and general environmental impact.

Societal aspects

Siting factors connected to societal considerations and community impact.

Figure 5-2 shows how each main group contains a host of criteria and factors that determine the suitability of a site for a deep repository. Some of the factors are absolute criteria that must be met if a deep repository is to be built on a given site. Examples of such absolute criteria are that the groundwater shall be oxygen-free at repository depth, that mineral deposits may not exist within the deep repository, and that the site may not be situated within a national park.

However, most factors fall along a scale of favourable unfavourable, which means that they are important in an overall assessment of the suitability of the site, but are not by themselves crucial in deciding whether a site is suitable. Examples of unfavourable conditions are heterogeneous bedrock, long distance to an existing road/ railway and competing land-use interests.

General siting studies

The general studies constitute background facts for the municipality- or site-specific investigations. With the aid of the general studies it will be possible to place the relevant sites in their national and regional contexts. Information is entered and stored in SKB's GIS system, which now comprises one of Sweden's largest databases of this type. The general studies will also provide a picture of different areas in the country which are, for different reasons, less suitable for siting of a deep repository. They cannot, on the other hand, provide any specific guidance in the work of finding suitable sites. This requires studies on a more detailed scale and a dialogue with, among others, concerned local and regional politicians and population.

In the general studies, comprehensive background material on geological, technical, environment-related and societal conditions is compiled. These studies have been published continuously as a part of the research and development work which SKB has been conducting since the late 1970s /5-2/. The studies include, among other things:

- General facility description and general background data for future environmental impact assessments.
- General survey of transportation system, including transshipment in harbour and transport by road or rail.
- Compilations of geographically related information on a national and/or regional scale concerning bedrock, topography, nature conservation areas, mineral deposits, major regional fracture zones, earthquake frequencies, etc.
- Surveys, analyses and forecasts of e.g. effects of glaciation on the bedrock and on seismotectonic conditions for different parts of the Swedish bedrock.

Results from geoscientific general siting studies published during 1995 are described in Section 16.3.

In the government decision of May 18, 1995, concerning SKB's programme, the government made the following statement with regard to 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 /5-4/, contains SKB's overall report on general siting studies carried out on a national scale in accordance with the government decision. The report is mainly based on the extensive background material that SKB prepared as a part of the research and development work which has been conducted since the late 1970's.

In General Siting Study 95, important siting factors have been described and applied on a national scale. For each such siting factor, separate conclusions have been reported in detail.

All of the factors, of which the long-term radiological safety is the central factor, must be taken into account in the siting of the deep repository. However, to make an overall evaluation of this factor, site-specific data on the bedrock must be available. Such data can only be obtained by carrying out extensive investigations at sites which can only be selected on the basis of partially incomplete information. This is a characteristic which distinguishes the siting of underground facilities in general, and a deep repository in particular, from other types of industrial siting, where information on all of the vital factors is relatively easy to obtain.

An important part of the work on preparing General Siting Study 95 has been to explore the possibilities and limitations which are related to a general siting study on a national scale.

A general siting study on a national scale cannot provide information on scientific, technical and societal factors with the necessary level of detail in order to identify sites which are suitable for the siting of a deep repository. The information presented in this study in general also covers conditions at the surface and not at the depths envisaged for the deep repository, 400–700 m below the surface. Thus, the suitability of a site is best evaluated through feasibility studies and site investigations and evaluations which must be carried out in connection with pertinent permitting of regulatory authorities.

However, General Siting Study 95 makes it possible to identify areas which are less suitable, or of less interest. Areas which are of less interest, on a national scale, cannot be definitely excluded since, on the local scale of a particular area, there may be many sites of interest which may be omitted when generalizations are made on the national scale. The general siting study also examines several scientific, technical and societal conditions in different parts of Sweden. These conditions provide a basis for assessing interest in, and for carrying out more detailed siting studies (feasibility studies).

A number of national databases which, in one way or another are, or may be significant for the siting factor of long-term radiological **Safety** have been described and evaluated in the General Siting Study 95. This includes rock types, topography, well data and the Highest shoreline. Furthermore, geological deformation zones, lineaments, future ice ages and earthquakes have been included. The possibility of unintentional intrusion and the selection of discharge areas are also discussed.

The siting factor, **Technology**, describes how the feasibility may be affected by different conditions. After the waste is encapsulated, it must be transported to the deep repository. Various alternatives exist, depending on the location of the deep repository in relation to the encapsulation plant. There is no actual restriction on how the deep repository can be located in terms of the means of transportation. The preferred means of transportation would require railroads or harbours. It is easier to carry out investigations of the bedrock, facility design and safety assessment if the geoscientific conditions of the site are easy to understand and interpret. A number of attempts have been made to, on a national scale, assess possible regional differences in interpretability. Even if it is suitable to primarily seek out sites which are easy to interpret, a similar reliability in results might be obtained at a more complex site, although more extensive investigation of the site would be required. This is thus a question of optimization, among other issues. The deep repository, as an engineering and construction project, benefit from the same factors which benefit the long-term radiological safety. Thus, there is no conflict between safety and simplicity of implementation with regard to the actual construction work.

The factor, **Land and Environment**, must also be taken into account in the siting of the repository, bearing in mind the stipulations of the Act concerning the Management of Natural Resources etc. Furthermore, there are areas which are protected against exploitation by law, e.g. national parks, nature reserves etc.

With regard to the siting factor, **Society**, is concerned there are a large number of conditions which are treated in the report which can largely be evaluated on a more detailed scale in connection with feasibility studies and site investigations.

Several conclusions from this work was presented in the General Siting Study 95 and a few of them are:

An overall evaluation of the applicable siting factors shows that it is unsuitable to locate the deep repository in the Scandinavian mountain range, Skåne and Gotland, primarily for geological reasons. Furthermore, the Scandinavian mountain range is an area of national interest with regard to nature conservation and outdoor activities. Siting of the repository in the bedrock below the island of Öland is considered to be technically possible, although unsuitable in terms of the management regulations of the Act concerning the Management of Natural Resources etc.

Siting of the repository in areas which are directly protected by law is neither necessary or desirable and must be avoided.

The unnecessary use, or blocking of natural resources must, if possible, be avoided. Areas where unusual rock types occur or where there is a potential for mineral resources, especially bedrock consisting of acid volcanic rock types, are therefore of less interest. By avoiding these rock types, there is less possibility of future, unintentional intrusion into the deep repository as well.

The conclusion which was previously drawn by SKB that there are many areas in Sweden which appear to be suitable for the siting of a deep repository has not been altered by General Siting Study 95. In the future, siting should also focus on bedrock commonly found in Sweden, preferably granite-like rock types, or old, highly metamorphosised sedimentary bedrock. This type of "interesting" bedrock exists in large parts of Sweden.

It is not necessary to exclude areas containing gabbro, or areas where the bedrock is covered by sedimentary rock, in connection with siting. These areas can be assessed especially if feasibility studies will be carried out in municipalities with this type of bedrock.

As far as transportation and communication are concerned, the availability of harbours, railroads or airports is good. Thus, on the national scale, there is no real limitation of possible areas in terms of these factors.

As a basis for concrete discussions with different communities concerning feasibility studies, SKB intends to use regional general siting studies which are based on the General Siting Study 95. In such regional general siting studies, it is of particular interest to preliminarily identify areas within one region which are expected to have suitable conditions with regard to industrial experience, availability of industrial land and proximity to harbours or railroads as well as where it is expected that there is bedrock with a good potential for fulfilling technical and safety-related requirements.

General Siting Study 95 contains background material and a description of general conditions in different parts of Sweden.

The background material often contains large variations on a local scale, which cannot be specified on a national scale. Thus, in all probability, suitable areas can be found both in regions which, on a national scale, appear to be of greater interest, as well as in regions which, on a national scale, appear to be of lesser interest. This general approach can be applied to several of the databases or situations which are described in the report.

Feasibility studies

An important part of the siting programme involves feasibility studies of municipalities which have expressed a potential interest in hosting a repository facility or are not negative to such studies.

Feasibility studies entail evaluation of siting prospects together with the positive and negative environmental and societal consequences of such a siting. The main purpose is to provide an adequate information basis such that the municipality and SKB can decide whether or not to continue with site investigations. An essential prerequisite to continue further is that there is sufficient interest from both the municipality and SKB.

SKB plans to carry out between five to ten feasibility studies mainly based on existing geoscientific and other data. During 1995 SKB has worked with feasibility studies at four municipalities, Storuman, Malå, Nyköping and Östhammar.

During the late spring of 1995, discussions were held with Tranemo Municipality concerning a feasibility study. Following expressions of a negative public opinion the municipal executive committee has, however, decided not to take up the matter for further 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 /5-3/ 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. The municipality executive committee later decided not to participate mainly because of a negative public opinion following the occurrence of a minor earthquake within the 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 5-3.

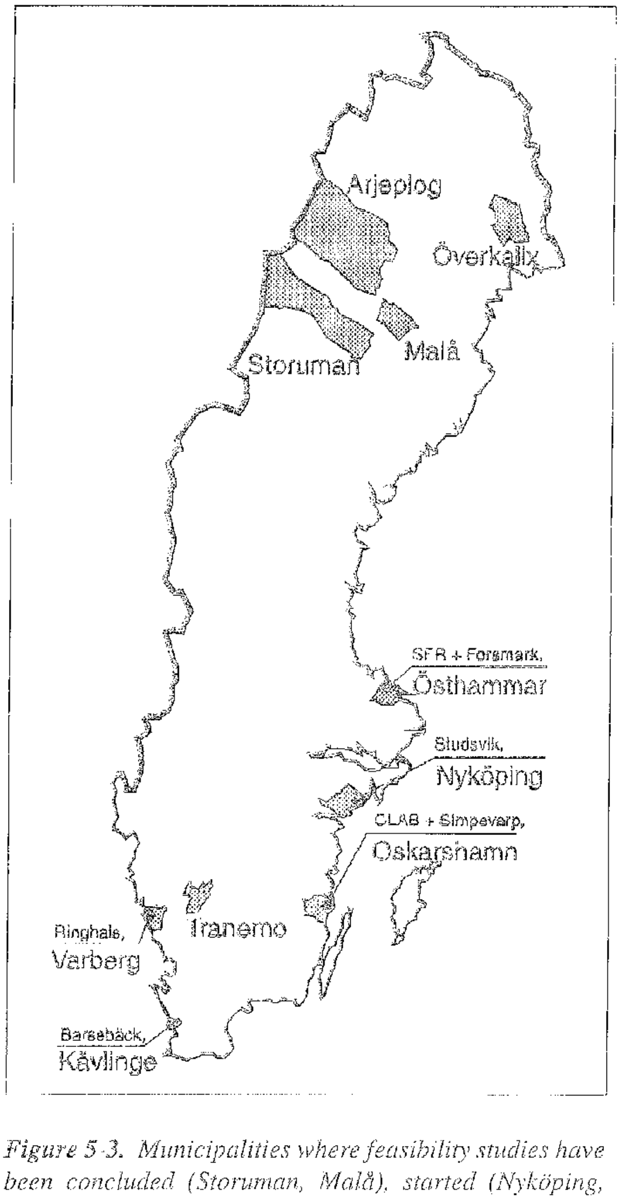

Figure 5-3. Municipalities where feasibility studies have been concluded (Storuman, Malå), started (Nyköping, Östhammar) or are being considered (Oskarshamn). Discussions concerning feasibility studies have been held in Arjeplog, Överkalix, Tranemo, Kävlinge and Varberg, but these municipalities are no longer candidates.

Feasibility study of the Storuman municipality

The feasibility study in Storuman was the first to be conducted. It began during the second half of 1993 with the conclusion of an agreement between the municipality and SKB. SKB has been in charge of the execution of the feasibility study. The municipality has had insight into and been able to influence the feasibility study through a steering group with two representatives for the municipality and two for SKB.

Altogether about 30 reports have been published within the framework of the study. They are all in Swedish. A final report published in January 1995 summarizes the results of the various studies and presents the collective evaluation of the results. This report is also available in English /5-5/. The purpose of the studies has been to describe, in as much detail as possible, the prospects for siting a deep repository in the municipality of Storuman, and to shed light on the possible positive and negative consequences of such a siting.

The final report concludes that potentially favourable sites exist within the Storuman municipality but also that a comparison with sites identified in other feasibility studies must be made before any site is selected for more comprehensive studies.

The Storuman municipality was planning an independent examination of the feasibility study. However, during February 1995 the Municipal Council unexpectedly decided to hold a local referendum on whether SKB should be allowed to continue studies in the municipality or not. The question posed was thus not that of a yes or no to the deep repository itself.

SKB informed the public that it would not run a campaign or otherwise attempt to influence the result of the referendum since it was a matter for those living in the municipality. However, SKB declared that it was ready to provide further information, answer questions and take part in discussions and in public hearings. Several such activities also took place before the referendum.

The referendum was held in Storuman on 17 September 1995. The result was clearly negative, with 71% answering No and 28% Yes. 70% of those entitled to vote did so, which incidentally was a considerably higher turnout than for the election to the EU Parliament which was held at the same time.

Directly after the referendum, SKB announced that the local office would be closed and that Storuman no longer would be considered as a possible site for a deep repository.

The feasibility study of Storuman helped to clarify important aspects of the siting process for SKB and the authorities, as well as for Storuman and other municipalities participating in feasibility studies. The Government also approved the concept of including feasibility studies in the siting process in its decision of May 1995 that was mentioned earlier in this chapter.

Feasibility study of the Malå municipality

Malå is the second municipality subject to a feasibility study. The geographical location of Malå is indicated in Figure 5-3. It is in many respects a typical rural municipality of the sparsely populated inland region of northern Sweden. Land area covered is about 1750 square kilometers, most of which is forested. More than half of the in total about 4 000 inhabitants live in the Malå community, forming the municipality centre. Forestry and mining used to dominate the economy in the region. The decline in labor needs in these sectors over the last decades has resulted in low numbers of employment opportunities and a decreasing population. To some extent, however, this recession has been mitigated by a diversification of the local economy.

The feasibility study started in early 1994, with the conclusion of a contract between Malå municipality and SKB. Prior to that, SKB had conducted a brief evaluation, indicating that a feasibility study was of interest. Furthermore, the local political discussion of the matter had resulted in the Municipality Board passing a decision in favour for conducting the study.

SKB believes that the siting process should proceed in an atmosphere of openness and free discussion. Active participation by municipalities and other local interests is encouraged. The organizational arrangements for the feasibility study have reflected this standpoint, and allowed insight by the municipality into planning and performance of the work. Local information activities have formed an important part of the study. An office was established in Malå when beginning the feasibility study, and has been open to the public on a daily basis.

The feasibility study has entailed a broad investigation program, aimed at providing information for evaluating the siting prospects with respect to siting factors adopted, and describing the environmental and societal consequences of such a siting. Investigation work has been done in the form of studies, based on existing information.

Compilation and analysis of geoscientific data has constituted a large portion of the investigation work. Bedrock geological maps, interpretations of the major fracture zone network, inventories of mineralizations and collection of available groundwater chemistry data from wells are examples of result produced. The data available are superficial and largely refer to conditions observable at surface. Overall interpretation of the geoscientific information shows that areas within the municipality where major granitic plutons occur may offer favourable siting conditions. Two such areas, 100 km^2 and 55 km^2 in size respectively, have been identified where conditions are judged to be particularly promising, see Figure 5-4. Data indicate that the predominantly granitic bedrock in these areas is homogeneous, with a low fracture content and relatively few fracture zones. It is therefore considered that possible, further siting studies should focus on the indicated areas. Important issues to address at an early stage of site investigations would be depth of the granite bodies, fracture zone geometry, ground water chemistry at depth and rock stress conditions.

Operating a deep repository in Malå would imply transporting encapsulated used fuel and other long-lived waste, from the intermediate storage facility (CLAB) near Oskarshamn. The vast experience that exists from transportation of radioactive material, including used fuel, on a production basis demonstrates the technical and administrative capability to conduct such transports, without exposing the environment to risks related to the material being transported. The availability of safe transport methods is a fundamental requirement for the siting program, and is of course a prerequisite for considering Malå municipality. Transport-related issues addressed within the feasibility study have concerned possible modes and routes for the land transport from a suitable harbor to the municipality, for encapsulated waste as well as backfill material and other goods. Shipment by railway or road are both found to be feasible alternative, but improvements of the present infrastructure would be required for both alternatives.

Malå municipality offers good siting premises with respect to current and planned land use, as well as a variety of land preservation interests. This applies also to the areas that are of prime interest from a geoscientific point of view. The fact that these areas are hundreds of times larger than the actual area requirements for the surface facilities accompanying a deep repository would contribute to flexibility for considering local interests in a siting process. Factors that would require special attention in this respect include reindeer herding, which is actively pursued on most land within the municipality.

Besides evaluating technical and environmental siting prospects, the feasibility study has addressed societal issues. The views and opinions as to whether establishment of a deep repository is good or bad for the local society vary widely and in fundamental respects. Therefore, the surveys conducted as part of the feasibility study have aimed at providing background information rather that attempting any kind of ready-made answers. The current situation in terms of population, business and industry, labor market, services and municipal finances has been compiled. Possible future development has been outlined and the societal impact of implementing a deep repository project has been estimated. Out of the total,

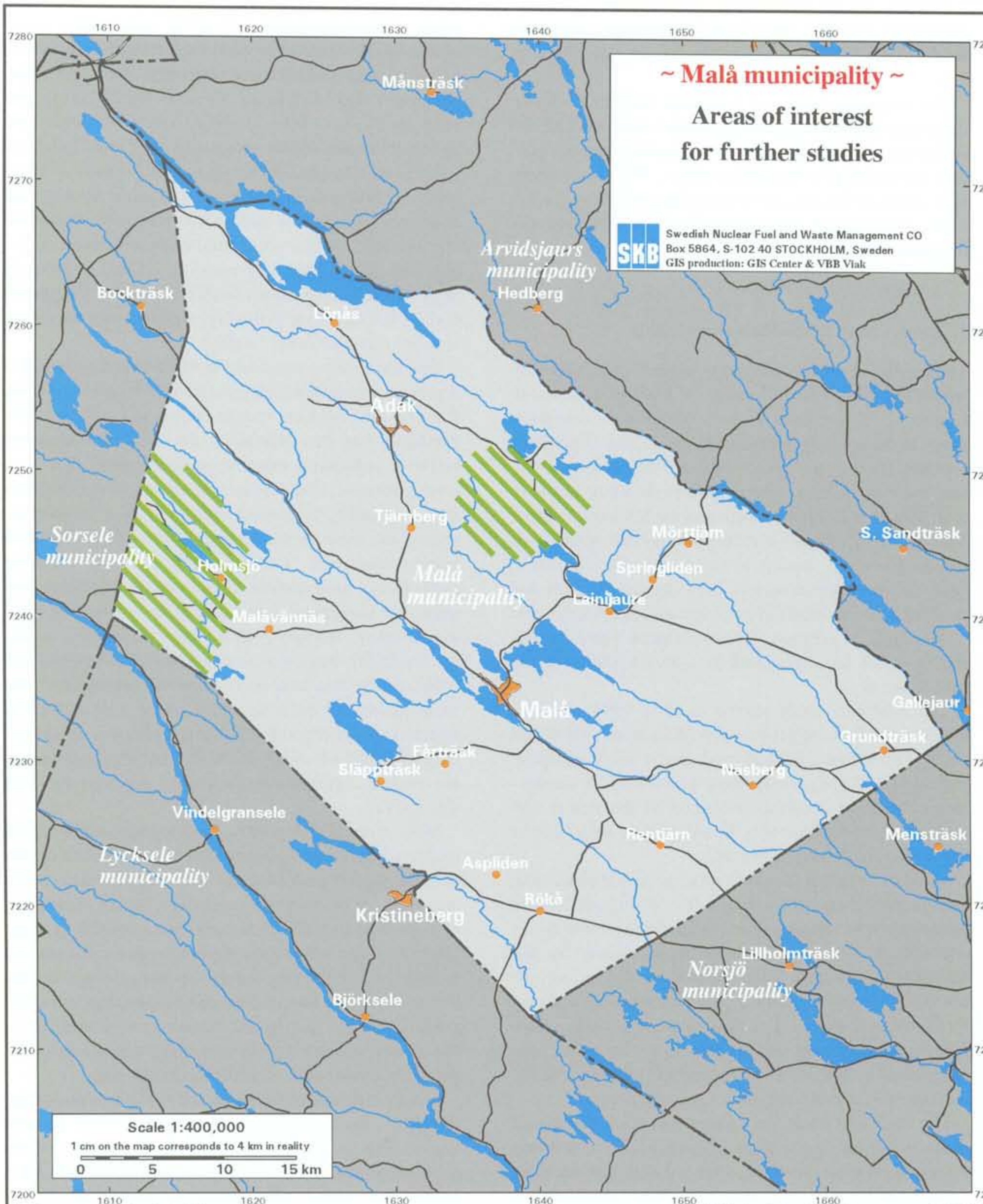

Figure 5-4. Malå municipality – areas of interest for further studies.

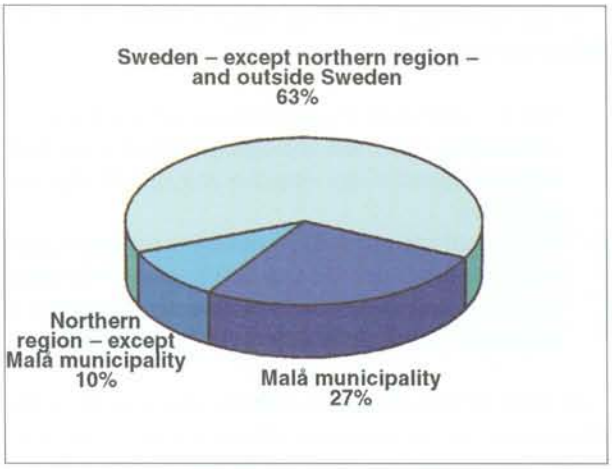

Figure 5-5. Estimated, geographical distribution of investments related to a deep repository in Malå municipality. The total cost is estimated to SEK 15 billion.

approximately SEK 15 billion investment related to the deep repository, it is found that some 27%, or SEK 4 billion, can be absorbed locally, see Figure 5-5. It is anticipated that a deep repository, when fully operational, will provide 200 direct jobs and at least 100 indirect jobs within the municipality. Considering these estimates in relation to today's situation, with total of some 1 500 employed within the municipality, it is clear that a repository would significantly influence local society in terms of population, employment and economy.

At present, investigations within the feasibility study have been completed. Results have been documented in a series of task reports (in Swedish). A final report, summarizing all results and presenting SKB's overall conclusions from the study, was published in March 1996 /5-6/ (will be translated to English). The municipality of Malå has decided to conduct an independent review of the feasibility study. The objective is to both examine SKB's investigation work and to provide a platform for broad discussion of the matter on the local level. A local committee with representatives for the political parties, unions and a wide range of other local organizations has been appointed. Furthermore, independent expertise in fields of concern is being consulted.

Feasibility study of the Nyköping municipality

The feasibility study in Nyköping began early in the fall 1995, some months after the overview report was published regarding the possibilities for siting of a repository in the municipalities that host the nuclear power plants /5-3/.

An SKB information office was inaugurated in Nyköping in the end of October 1995. Here the citizens of the municipality can get answers to their questions, get information material and the latest news in the study etc., see Figure 5-6.

Figure 5-6. Interior from the Nyköping office.

The Municipality Board of Nyköping has created three different reviewing groups to follow the SKB work. One is a political group with members from all political parties in the municipality, one is a group with municipality officials and the third has members from about 25 organisations (environmental groups, political groups, athletic groups etc).

The scientific work is underway and will continue during 1996 and the first part of 1997.

Feasibility study of the Östhammar municipality

The feasibility study of Östhammar has been conducted in parallel to the Nyköping study. The feasibility study of Östhammar was preceeded by a formal decision by the municipal council to participate in the study. The decision was taken with a strong majority.

The Östhammar municipality hosts the Forsmark nuclear reactors as well as the Final Repository for Radioactive Operational Waste (SFR). Thus there exist considerable experience within the municipality regarding the siting and operation of nuclear facilities.

During the fall of 1995 detailed working plans for the feasibility study was presented to the municipality for review. The outcome of the review was ready in early 1996 and the study has started. An information office, similar to the one in Nyköping, has also been established.

5.4 TECHNICAL STUDIES CONCERNING THE CONSTRUCTION OF A DEEP REPOSITORY SYSTEM

Global thermo-mechanical calculations have been made indicating that very high thermal loads can be accepted in the repository before high thermally induced stresses are developed at ground surface, and that the thermo-mechanical regime thus is not the design-determining factor in the present KBS-3 concept with an increased thermal load per canister. The design is still determined by the temperature limitation in the interface between a canister and the bentonite.

An analysis of the importance of the structure and stress regime on the performance of the rock around the tunnel floor and deposition hole has started by generation of fracture networks and modelling of stress conditions taking Äspö HRL as an example. The analysis also concerns investigation of existing know-how of formation of discontinuities and how the hydraulic regime may adjust to a change in the mechanical regime.

The properties of the zone closest to the simulated deposition holes bored in the research tunnel at Olkiluoto have been investigated by several methods. Also the surface roughness of the three holes have been examined.

A constructability analysis with Äspö HRL data has indicated that conventional tunnelling would consume twice as much cement and iron for grouting and rock support as TBM tunnelling.

Present knowledge of grouting as well as development needs have been analysed and a programme for development of the theoretical knowledge has been compiled.

Backfilling and plugging has been studied in both a large in situ experiment in the Äspö HRL and by theoretical analysis of plug design and plug locations. One theoretical result is that the axial flow in the zone closest to the tunnel is most effectively reduced if plugs are located along the tunnel in good rock with low hydraulic conductivity.

The studies of deposition alternatives with radiation shielded canisters up to the deposition hole and with canisters and bentonite buffer in one package have continued for both vertical and horizontal emplacement of the canister, and also the studies of ways of retrieving a canister in a buffer that has saturated and swollen.

A more detailed presentation of the studies mentioned above are given in Chapter 14.

5.5 SITE INVESTIGATION PROGRAMME

As a preparation for the forthcoming site investigations for candidate repository sites, planning work is going on in the following fields:

- development of the geoscientific investigation programme,
- preparation of techniques and routines for data management,
- preparation of instruments and methods, including development, refinement and investment, etc.

A general base for the planning work are the experiences from earlier site investigations, including the Äspö Hard Rock Laboratory (HRL), conducted by SKB.

A site investigation for the deep repository 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.

In SKB RD&D-Programme 95 the framework of the planned site investigation programme was presented. The investigations will be carried out in two main step, Initial site investigations and Complete site investigations. The main strategy is that SKB should carry out two site investigations. The site investigations and site evaluations will be conducted as an iterative process with close interactions between geoscientific investigations, performance assessment and design work. An environmental impact assessment is supposed to provide an overall picture of the planned deep repository.

Routines for data management have taken step forwards by the establishment of a new geological data base system (SICADA), which will be a central tool for fulfilling QA requirements of the site investigations. Work on technical documentations, manuals and QA-routines for all methods and tools which will be used in the site investigations is going on.

The new borehole-TV system, presented in the Annual Report 1994, has been very successful and used i the Äspö HRL as well as in other countries. The manufacturing of a new pipe string system for hydraulic pumping and injection tests is underway. Further information on the work conducted in the field of instruments and methods are presented in Chapter 14.

5.6 ENVIRONMENTAL IMPACT ASSESSMENT

SKB plans for the deep repository include the preparation of an Environmental Impact Assessment (EIA), see Figure 5-7. An EIA is formally required for certain facilities in accordance with the Swedish Act on the Management of Natural Resources and the Act on Nuclear Activities. The deep repository is such a facility.

The first, formal EIA for the deep repository has to be presented before licences of the detailed site characterization. In 1995 the SKB continued to prepare what could be referred to as a preliminary work, based on a repository design, not linked to a given site.

The studies include the impacts expected from all the activities associated with the repository project, from the site investigation, via the detailed site characterization, construction of the repository, operation of the repository

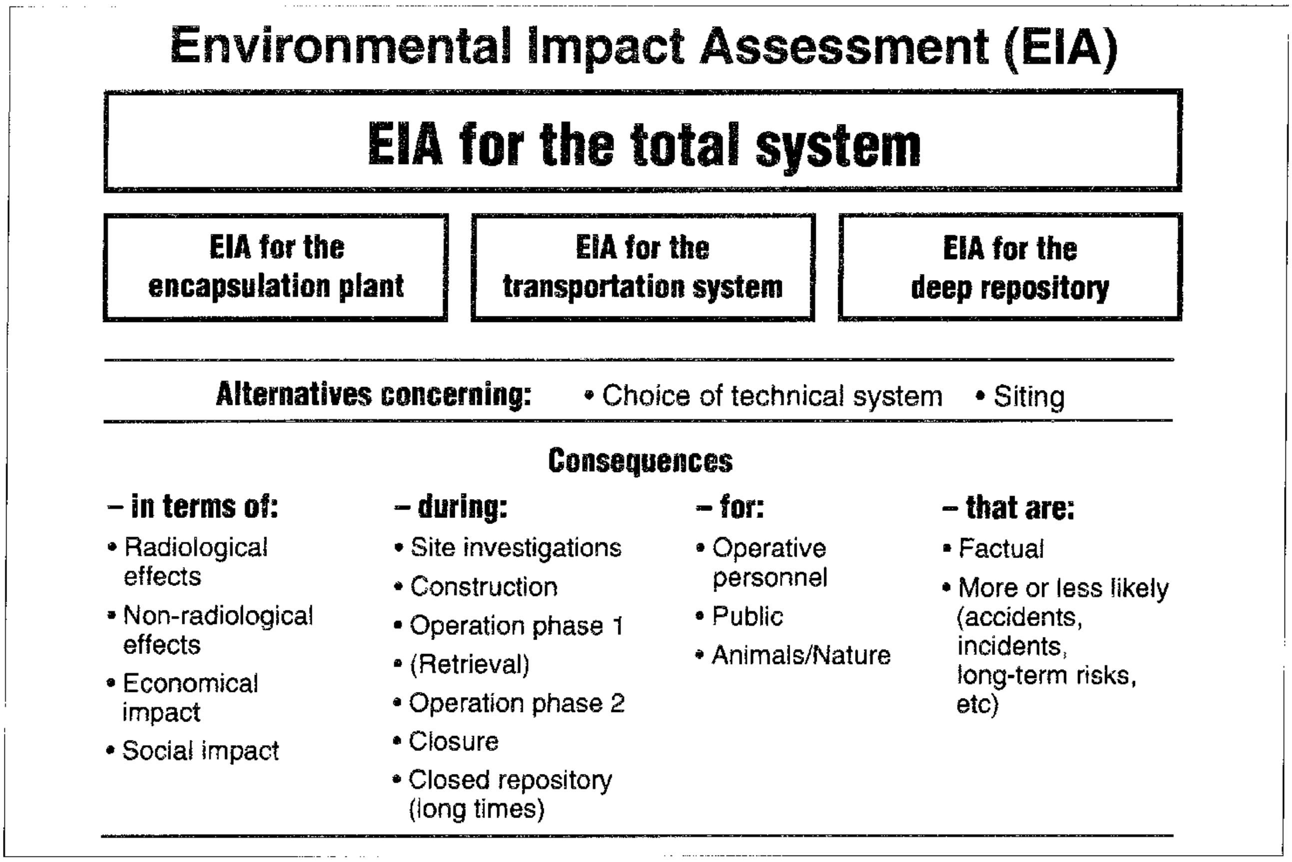

Figure 5-7. Schematic structure of an environmental impact assessment.

system through the final closure and the long-term post-closure phase.

The content of an EIA should be developed gradually in a process where all parties concerned (SKB, municipality, regional and national authorities etc.) try to agree upon aspects and items to be covered in the EIA. As mentioned in section 5.3 the County Administration has a responsability for that necessary contacts are taken to ensure that all parties have been involved in the EIA process on the regional level.

During 1995 there have also been discussions on having an EIA process on the national level. Since all affected municipalities and County Boards must be coordinated and that they have the same type of information etc, a national EIA coordinator has been suggested. To initiate such a coordinating function a formal proposal was in December 1995 sent to the Government by KASAM (The Swedish National Council for Nuclear Waste), the Nuclear Power Inspectorate, the National Institute of Radiological Protection, the County Administration in Kalmar and the municipality of Oskarshamn.

6 ENCAPSULATION PLANT PROJECT

6.1 GENERAL

SKB's development and testing of methods for the encapsulation of spent nuclear fuel is now in a advanced stage. In a few years an application for construction of an encapsulation plant is planned to be submitted to the government. The programme for canisters and encapsulation mainly comprises development and fabrication of canisters and design and construction of an encapsulation plant.

The spent fuel elements are stored in water pools in the CLAB facility. Before the fuel will be disposed of in a deep geological repository it must be encapsulated in a canister. In the repository the canister is one of the essential barriers. It will keep the fuel elements separated from the groundwater for a very long time. The canister will also provide radiation shielding and protection during the handling of the fuel in the deep repository.

Other designs of canisters have earlier been studied, such as a homogeneous copper canister, made by Hot Isostatic Pressing (HIP) technique, and a copper canister filled with lead around the fuel elements. Both these methods require that the encapsulation is done at a high temperature. This can be avoided with the present copper canister with an inner steel container. This fact has been decisive for the choice of canister design as the long time function is equal for the three types of canisters.

The encapsulation is planned to be made in a new facility to be built in connection to CLAB. This siting gives advantages in comparison with other sites to be considered for the encapsulation plant. Advantages are e.g. logistics for the handling, use of existing resources and minimal impact on the environment.

6.2 DEVELOPMENT OF CANISTER DESIGN

Canister design

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 in the groundwater present in the rock, or be broken 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 overpack of copper, 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 remain intact for millions of years, thus providing a considerable margin of safety (which is needed due to uncertainties and variations in the premises).

The work of designing the canister takes 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. 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.

Safety during operation and maintenance work at the encapsulation plant must be high. The canister's design must meet the requirements posed by both normal and abnormal operation in the plant. It must also be able to withstand handling accidents that could cause unacceptable exposures to the personnel and the plant or lead 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 handling and disposal 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 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 near-field 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. 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 only. 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 requirement of imperviousness can be subdivided into initial integrity, corrosion resistance and strength. The initial integrity implies that the canisters must be fabricated, sealed and inspected with methods that guarantee that very few (max. 0,1%) will contain undetected defects that could entail initial leakage or that could lead to early canister failure. The canister must not be penetrated by corrosion so that water can enter the canister during the first 100,000 years after deposition. The maximum corrosion depth during such a period is estimated to be about 5 mm in typical Swedish groundwater. In recognition of uncertainties in the data and other factors, a suitable safety factor should be included when determining the wall thickness. The canisters must be designed to withstand deposition at a depth of 400-700 m.

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 means that the canister material must not affect buffer and rock and the heat transfer to the surrounding must be limited as well as the radiation dose to bentonite. The canister must be designed so that the fuel remains subcritical even in the event of water penetration. The canister's vertical pressure against the bentonite must be limited so that the canister does not sink down through the bentonite.

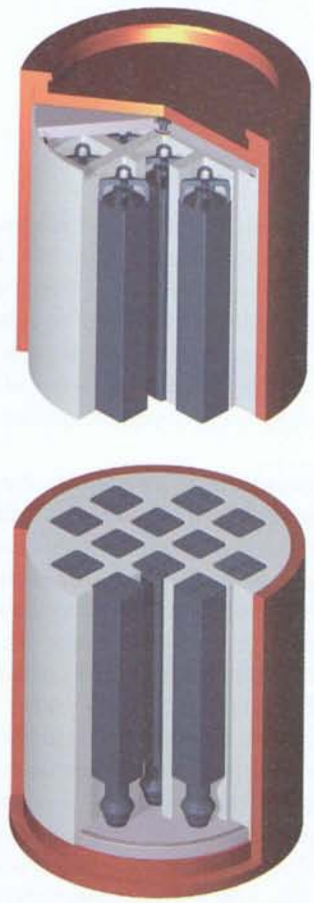

Figure 6-1. *Copper canister with cast insert.*

Reference canister

The work with the canister and the encapsulation process has focused on studies and development of technology that does not require 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 fill its function in the deep repository. One canister can hold either 12 BWR fuel elements or 4 PWR fuel elements.

A copper canister with a 50 mm wall thickness made of oxygen-free copper with a low phosphorus content is being used as reference for the continued work. The bottom and lid are joined to the shell by 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. 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 selection of material grades will be studied in the continued work and be chosen with a view towards the criteria described above.

The selection 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 proneness to stress corrosion cracking is low. Several possible materials are available for the inner container. The selection will be determined in part by the fabrication method.

As an alternative for the insert, a cast inner component of spheroidal graphite iron is considered, see Figure 6-1. It will be identical in all essential respects to the reference alternative. Spheroidal graphite iron has better castability, but its weldability is lower. 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 third alternative, 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.

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. Based on this a reference canister for up to 12 BWR assemblies or 4 PWR assemblies was chosen. Further studies are under way to determine the 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. The preliminary data for the reference canister is 1050 mm in diameter, length 4850 mm and weight 25,000 kg.

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. 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.

Canister fabrication methods

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, see Figure 6-2, 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 fulfil the objective. But there are good prospects for achieving the desired grain size in the material by means of modified process parameters and possibly cooling during extrusion.

Regarding other fabrication methods, Hot Isostatic Pressing (HIP) and electrodeposition, studies are under way to determine the value of carrying out trial fabrication on a full scale with them as well. 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. It is thus possible that several different fabrication methods will be used during this time if alternative methods are developed and become available, see Figure 6-3.

Figure 6-2. *Machining of coppercylinder with 50 mm wall thickness.*

Canister sealing methods

To fulfil the stringent requirements on sealing of the copper canister, a method is being developed employing electron beam welding of the 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.

Results obtained show that electron beam welding is a feasible method for fabricating and sealing copper canisters. In 1995 the tests for a horizontal weld were completed. This method is feasible but requires an outer support of fusion zone. This support ring has been integrated with the lid in the present design. Due to the tight tolerances for the lid fitting to the canister this can create problems. It was therefore decided that tests with inclined weld should be tested in the next step. The test equipment is now redesigned for testing of inclined welds in 1996.

Methods for nondestructive testing of the weld are being developed in parallel in order to verify that it complies with the stipulated requirements. Tests have been conducted with ultrasonic pulse-echo on a total of seven lids. Specimens from two lids were inspected by digital radiography, as a reference for ultrasonic inspection. The number and distribution of detected defects varied considerably between the various welds. This proved to be useful for testing the sensitivity of the different methods used.

Figure 6-3. *First test canister delivered to CLAB.*

Figure 6-4. *Overview of the planned encapsulation plant adjacent to the CLAB facility.*

6.3 DESIGN OF THE ENCAPSULATION PLANT

General Plant Description

The encapsulation plant is planned to be built directly adjacent to the CLAB interim storage facility, see Figure 6-4. The plant shall be designed and constructed mainly for the encapsulation process. Further it shall be possible to extend the facility with a process line also for treatment of core components. 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.

Encapsulation Process

The encapsulation process is being designed and engineered 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. The work of designing a suitable process for encapsulation of the fuel can be divided into functional parts where different technical solutions are considered. The work of designing the plant is in progress and descriptions of the encapsulation process and the layout of the plant have been produced. The following general description of the encapsulation process gives the status of the design work.

The fuel is transported in storage canisters from the storage section of CLAB via the existing fuel hoist to a new pool block in the encapsulation plant. 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 in this step.

In the next step the fuel is taken from the water to the handling cell, where it is dried and placed in a disposal canister, see Figure 6-5. In this part of the plant, where the fuel is handled freely, special requirements must be met to prevent the escape of radioactivity. The compartment is built with radiation-shielding walls and special requirements on airtightness and ventilation. A type of 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.

Transport of filled canisters is planned to take place in a channel that connects under the handling cell and the various work stations. Alternative transport systems are being studied and the choice will be made based on stringent requirements on reliability and safety. The canister is sealed during transport so that radioactivity cannot escape from the fuel.

Three work stations are planned for sealing of the canister. The first work station will contain the functions required for the exchange of air to inert gas in the canister and, if necessary, packing around the fuel assemblies. Finally, the steel lid is fitted and sealed. Thereafter there is no risk of radioactivity escaping from the fuel. This work station is also designed to meet the requirements made on a hot cell. The copper lid is placed on and sealed in a 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 melted together by an electron beam in a vacuum chamber. The last of the three stations will house equipment for inspection of the lid weld and for machining of the weld area on the canister and for removal of any improperly welded lid. The result of the sealing operation will be inspected by means of nondestructive testing using a technology that is being developed at the same time as the sealing method.

A routine check of surface contamination is planned, and decontamination of the outside of the canister will be possible in a special work station. Then the canisters are placed in a temporary store from which they can be delivered to the deep repository at a suitable pace. Handling in the temporary store is planned to be done with a radiation-shielding handling bell. The temporary store will be connected to a docking station for transport casks.

If inspection of the seal reveals that the weld 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.

Pilot Plant

Plans were in 1995 outlined for a pilot plant for development and testing of equipment for sealing welding and nondestructive testing. The plant is now planned to be established in the community of Oskarshamn in an existing building at the wharf. The plant is planned to be in operation in the beginning of 1998 and the design work has started.

The equipment for the 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 prac-

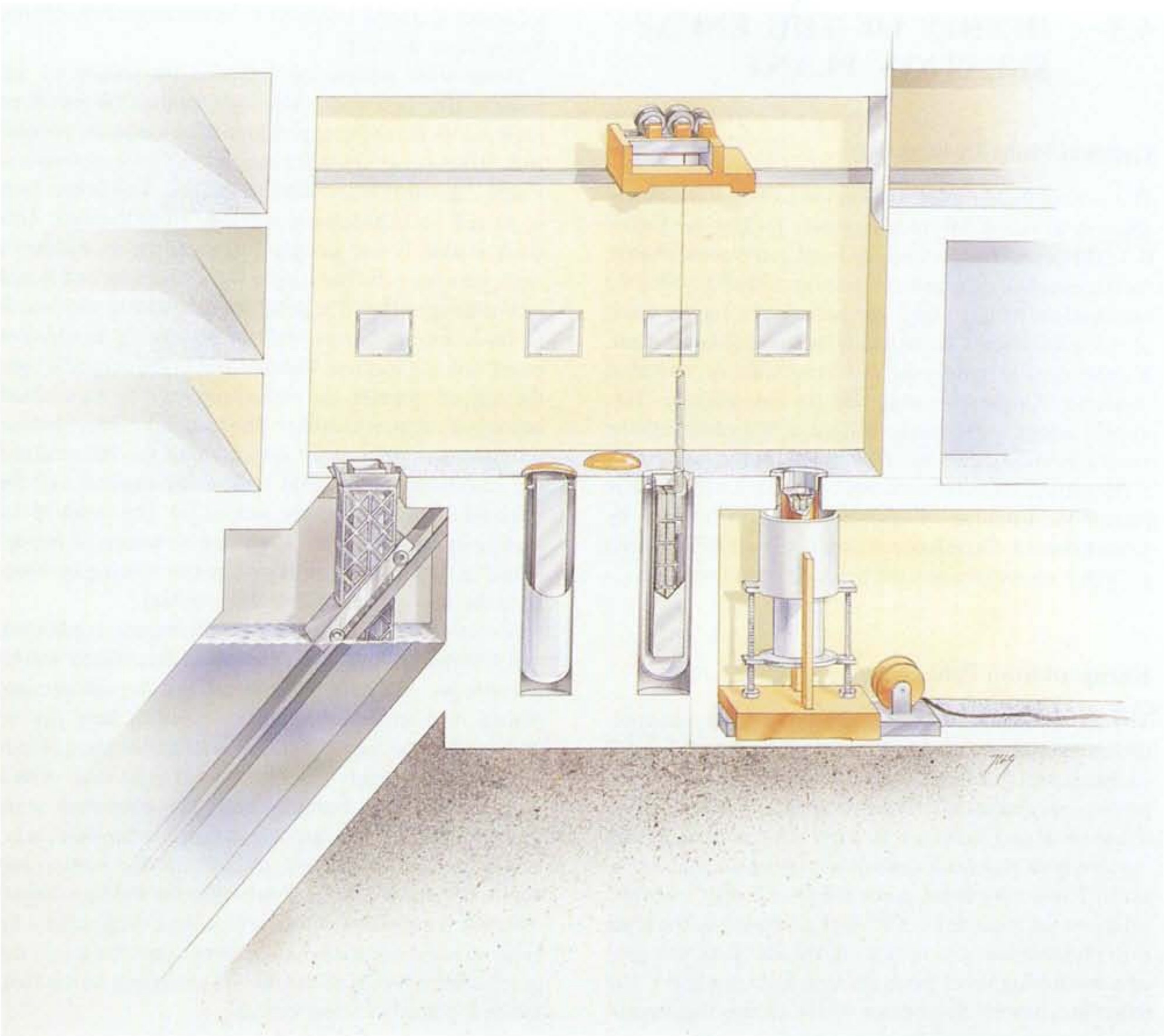

Figure 6-5. *General design of the handling cell.*

tical 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.

6.4 ENVIRONMENTAL IMPACT ASSESSMENT

In 1994 a forum for Environmental Impact Assessments (EIA-forum) was established. It is chaired by the County Administration of Kalmar and has representatives from the Oskarshamn community council, the nuclear power inspectorate (SKI), the radiation protection institute (SSI) and SKB. During 1995 the various aspects of construction and operation of an encapsulation plant were discussed in the forum. The results are documented in a planning report which gives the base for SKBs work with the Environmental Impact Statement report (EIS).

The community has also engaged a local coordinator for information transfer to the local population and for knowledge build-up in the community. This work is financed by the Waste Funds. The community has arranged public meetings and politicians have also attended the meetings with the EIA-forum in order to get information. It is planned for an exhibition in the community hall in 1996 to display the plans for the encapsulation plant and the result of the work in the EIA-forum.

7 SUMMARY OF RESEARCH, DEVELOPMENT AND DEMONSTRATION ACTIVITIES

7.1 GENERAL

According to the 12 § of the Act on Nuclear Activities the owners of the nuclear power plants are responsible for conducting the research, development and other measures necessary for the safe handling and disposal of radioactive wastes arising from the nuclear power production. A programme for conducting the necessary activities must be submitted to the pertinent authority every third year. By the end of September 1992 SKB submitted its third RD&D-programme to the Swedish Nuclear Power Inspectorate – SKI. The government decision on the programme was given in December 1993. The government decided that the programme should be supplemented by SKB. The supplement thus requested was submitted by SKB and a government decision on the supplement was given on May 18, 1995. A brief summary of main points in that decision is given in section 12.2.

In late September 1995 SKB submitted its fourth RD&D-programme to SKI. The programme consists of one volume and covers the period 1996–2001 in some detail. A brief summary is given in section 12.3. The programme was sent by SKI for review to about 60 organizations. The SKI review is expected to be available in the spring of 1996 and the review by KASAM in mid-1996 before the government decides on RD&D-Programme 95.

Supporting R&D work to refine 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 construction techniques. 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.

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.

7.2 SAFETY ANALYSIS

The activities in the area of safety assessment during 1995 have been focused on the RD&D-programme and the development of a standard template for the reporting of long term safety of deep repositories. Important developments have been implemented on the scenario methodology and on the practical handling of numerical calculations with model chains.

A template for safety reporting

Safety reports serve as a basis for a series of decision steps in the implementation of the deep disposal scheme. Safety reports must satisfy three fundamental requirements:

- The scope of the decision that is to be based on the safety report must be clearly defined, and the safety assessments made must be relevant to this scope.
- Prerequisites, methods, analyses and results (incl. uncertainties) must be reported to such an extent and in such a way that an independent review can be carried out.
- It must be demonstrated that safety is adequate in relation to given acceptance criteria or that the safety potential is sufficient to permit a transition to the next implementation phase.

In order to create a continuity in safety reporting and a uniformity between reports, a template was developed for giving an account of long-term safety, SR 95. The template is intended also to facilitate the preparation and review of safety reports, and to simplify comparisons between how, for example, an expanded body of data influences the assessment of safety and uncertainty.

A synopsis of the template has been proposed and is exemplified with illustrative text describing the underlying premises and the status of the assessment capability today.

The template is organized in 4 major parts:

- Premises and scope.
- Description of the deep repository system.
- Evolution of the repository system with time.
- Evaluation and conclusions.

Monitor 2000

During 1995, a new user interface for PROPER, a master program for probabilistic performance assessment calculations called Monitor 2000, has been developed. Monitor 2000 is a graphic interface which will replace the former text-based interface. The introduction of Monitor 2000 will considerably reduce the operator time required to set-up and further handle complex model calculations. Furthermore, the quality assurance of the calculations will be improved through built-in controls in the new interface. Comprehensive reports of complex calculations will be automatically generated, improving the traceability of performed calculations.

Scenario methodology

Due to the very long time frames evaluated in post closure performance assessments of repositories for radioactive waste, there is a need for a detailed discussion of how the performance of the repository might be affected by

- the features of the repository system or hostrock,
- the processes that can take place in the repository, or
- the external events that can change the repository environment.

Although there will always be a need for expert judgement when talking about future possible developments, the need for a structured approach was felt already in the late 1980s. The main advantages sought for was

- a better overview of how the various subsystems of the repository interacted with each other,
- a methodology that would support the control that all relevant features, events and processes (FEPs) – and all relevant interactions – were taken into account,
- a systematic structure that could be used for the logical documentation.

As a consequence a number of FEPs lists were developed in different national groups. An effort to make them available internationally is made by OECD/NEA. The work within SKB have been concentrated to develop a number of so called interaction matrixes for the subsystems of the Swedish repository concept.

Modelling of repository performance

To achieve a quantified evaluation of the performance or safety of a repository system a number of numerical models have been developed and tested. This work is continuously going on as relevant measurements or experiments are made. Recent progress on the work is reported in Chapters 13, 15, 16 and 18 in part II of this Annual Report.

7.3 SUPPORTING RESEARCH AND DEVELOPMENT

7.3.1 General

Chapter 16 in Part II summarizes activities both on general development of understanding and databases in areas of importance for repository safety and on specific supportive research actions that have been initiated to clarify unresolved issues. The chapter contains sections on Spent Fuel, Buffer and Backfill, Geoscience, Chemistry, Natural Analogue Studies and Biosphere. The R&D-work on canister – material, design and fabrication – is reported on in Chapters 6 and 13.

7.3.2 Engineered barriers

In the studies on the behaviour of spent nuclear fuel in repository conditions results on fuel characterization, on fuel corrosion modeling, on alfa radiolysis, and on fuel natural analogues have been obtained. Reproducible values of specific surface area of the fuel available for the corrosive attack have been obtained by using the BET methodology. The content and composition of metallic fission product particles containing Mo, Tc, Ru, Rh and Pd in the fuel has been analyzed. The concentrations of Mo and Tc obtained during their dissolution in groundwater were of the same order of magnitude as those obtained in corrosion tests with spent fuel.

Thermodynamic data for solubilities and the speciation of the lanthanides in repository conditions have been selected and validated, making possible the evaluation of their behaviour during spent fuel corrosion.

Special procedures were developed to analyze the very low concentrations obtained in the leaching and diffusion experiments of spent fuel in contact with bentonite. Results on the leaching and diffusion of neptunium and uranium have been obtained. Long term radiolysis experiments have confirmed the previous results on a clear deficiency of oxidants in the overall redox system, indicating the importance of the redox buffering capacity of the spent fuel matrix. A kinetic model accounting for the redox capacity of the spent fuel matrix has been under development and has been included in the EQ 3/6 code package.

From experimental and modelling studies on uranium minerals as natural analogues for the stability of the spent fuel matrix, a unified model for the dissolution of uranites has been developed. The stability of becquerelite and of the U(VI) silicate phases soddyite and uranophane as end products for the oxidative alteration of the spent fuel matrix have also been determined experimentally.

Studies of buffer and backfill include testing and modelling of physical behaviour and heat conductivity of water saturated and non-saturated bentonite. From laboratory investigations with the present model on thermo-hydro-mechanical processes (THM) in buffer and surrounding rock it has been concluded that the model is relevant for predictions of the repository performance and that the model is ready for testing in full-scale experiments. The preliminary model on unsaturated conditions has been found to be useful in some cases but not mature enough for complete predictions of THM processes.

In april 1995 an experimental program on gas transport through bentonite was started. The program will investigate the transport parameters for a gas phase transport through compacted and saturated MX-80 bentonite under pressure conditions expected in the repository. The first experimental runs have been completed and the program will continue during 1996.

The laboratory work on replacing quartz sand by crushed rock as ballast in backfill has been completed with the result that crushed rock is judged to give a material that serves equally well as the quartz sand based material.

Compaction of large bentonite blocks with a diameter of 1.0 m by uniaxial compaction has shown promising results.

The second part of the Buffer and Backfill Handbook has been prepared. It describes buffer and backfill materials and their preparation and application.

7.3.3 Geoscience

The general geoscientific programme comprises activities which among others attempt to quantify probable impacts of earthquakes, glaciation and land uplift. These activities emphasize long-term geodynamic processes in the Baltic Shield, such as postglacial faulting and glacial impacts on hydrogeology and ground water chemistry.

During 1995 the geoscience programme has included research on the occurrence of sedimentary strata which has covered the subcambrian bedrock in the Baltic shield. Fission track analyses make it possible to follow the temperature development of the bedrock in a geological long-term perspective. The primary results from these analyses of apatite and sphene indicate that the subcambrian peneplane in south and central Fennoscandia might have been covered with Devonian rocks of considerable thickness. The temperature levels indicate sediments with a thickness of approximately 3 000 meters. The duration of the Phanerozoic sedimentary cover has been calculated to 250 million years. It is worth mentioning that this sediment cover equals three times a continental ice sheet load.

Special interest has been devoted to tectonic indicating markers from the margins of the Baltic shield. In the absence of onland late Mesozoic and Cenozoic geological formations the tectonic history of the Baltic Shield over the past 100 million years can most readily be reconstructed from the thick sedimentary basins that surround Fennoscandia on three sides. Changing patterns of sediment thickness accompanying active tectonics, as observed on high resolution multichannel seismic reflection lines, record the boundary conditions of deformation internal to the Baltic Shield. Tectonic activity around Fennoscandia through this period has been diverse but can be divided into four main periods. The highest levels of deformation on the margins of Fennoscandia were achieved around 85 Ma, 60–55 Ma and 15–10 Ma, with strain-rates around 10^{-9}/year. Within the Baltic Shield long term strainrates have been around 10^{-11}/year, with little evidence for significant deformations passing into the shield from the margins. Fennoscandian Border Zone activity, which was prominent from 90–60 Ma, was largely abandoned following the creation of the Norwegian Sea spreading ridge. There is subsequently very little evidence for deformation passing into Fennoscandia. The current tectonic regime is of Quaternary age although the orientation of the major stress axis has remained approximately consistent since around 10 Ma. The past pattern of changes suggests that in the geological near-future variations are to be anticipated in the magnitude rather than the orientation of stresses. However, you have to superimpose the effects of continental ice sheets.

7.3.4 Chemistry

The chemistry program comprises investigations of radionuclide chemistry which is relevant for the safety of nuclear waste disposal. This includes solubility of radionuclides, and mobility and retention in repository barriers. Non-radioactive chemical conditions in groundwater, bentonite and concrete are also important in this context and therefore part of the studies.

Solubility data for technetium have been compiled and the spontaneous reduction of technetium (pertechnetate) and neptunium (neptunyl) in a deep repository environment have been demonstrated. A low redox state promotes the retention of these important elements which can otherwise appear as mobile.

Sorption models are being tested and a review of all available information of matrix diffusion have been made. The term matrix diffusion refers to the phenomena where dissolved radionuclides diffuses into a connected system of microfractures into the seemingly intact rock and are sorbed on the mineral surfaces there. This is the most important mechanism for radionuclide retention in the far-field, used in performance assessment calculations of radionuclide dispersal.

Diffusion experiments with cesium, strontium and iodine in compacted sodium bentonite (buffer material) have been performed. The results have not been reported yet.

Column experiments on the transport properties of colloids have been concluded and reported. Dissolved electrolytes and neutral pH in the groundwater tend to destabilise natural mineral colloids, but colloid transport is indeed possible under special conditions. The strongest argument against colloids as potential carriers of radionuclides is the fact that the their concentration in natural groundwater at depth is very low. This has been further manifested in a review of this and related matters on colloids. A remaining issue that needs to be further evaluated is the influence of gas on particle transport. This is being investigated.

The geosphere is not sterile, at least not down to about 1000 m. Microbes are being investigated in the field and in the laboratory. A comprehensive review has been made on subterranean bacteria and their importance for performance assessment. Influences of bacteria activities are not necessarily negative but the existence of sulphide reducing bacteria, SRB, must be considered. Sulphide produced by SRB can, in theory, corrode the copper canister. Special experiments have therefore been performed and it has been demonstrated that SRB do not survive in the bentonite buffer, which surrounds the canister. This adds to the safety margin for corrosion resistance.

In-situ experiments for the CHEMLAB probe are being prepared and another validation exercise, the consequences of concrete in a repository, has been summarised in an interim report from British Geological Survey, BGS. Cement has many applications in underground construction but it may also rise the pH of groundwater in the near-field of the repository. Laboratory tests are being made on the influence of high pH solutions (simulated concrete pore water) on geochemical conditions. The studies are performed by BGS and jointly supported by NAGRA, NIREX and SKB.

7.3.5 Natural analogue studies

Natural analogue studies are used to support performance assessment, justify the assumptions made and validate models. The present SKB natural analogue program comprises three international projects: Jordan, Oklo and Palmottu, and the study of old concrete constructions made by Portland cement. Data obtained from previous participation in international studies, i.e. the projects Poços de Caldas and Cigar Lake, have been revisited in order to promote further model development. A notable result of that is the improvement of calculation models for radiation deposition and radiolysis which can be further applied to the assessment of spent fuel leaching. Explicit use of analogue information in performance assessment studies have been made but, so far, it is not common.

The first phase of the investigations of the 2 billion years ($2 \cdot 10^9$ years) old reactor zones in Oklo, Okelobondo and Bangombé have been concluded and reported. This study was directed by the French CEA and supported by EU. Organisations from other countries including SKB have participated. SKB's involvement was concentrated to the fossil reactor in Bangombé, which is situated about 20 km away from Oklo and Okelobondo. The final report was issued in 1995 and some points of particular interest in relation to spent fuel disposal deserve to be mentioned, such as: evidence of water radiolysis in the form of H_2 and O_2 in fluid inclusions, preserved metallic inclusions of platinum elements which serve to isolate technetium in spent nuclear fuel, remnants of plutonium (in the form of ^{235}U) trapped in the clay (chlorite) surrounding the reactor zones, etc. Models for calculation of hydraulic flow and redox front migration were applied and tested by observations of environmental tracers and geochemical changes. Equipments for sampling of colloids, dissolved organics and microbes were tested in the field. The sampling was successful and the results and interpretations will be reported later. A second phase of the Oklo project has been planned and proposed to the EU for support. It is expected to start in early 1996 and continue for three years.

The 1.7 to 1.8 billion year old uranium-thorium deposit at lake Palmottu in southwestern Finland consists of a 1 to 15 m thick discontinuous subvertical ore zone, which extends from the surface and down to a depth of about 300 m. The uranium mineralisation is situated in a host rock with the same hydrogeological, hydrochemical and geological conditions that are anticipated for a Finnish (and Swedish) spent fuel repository in the Fennoscandian Shield. The uraninite has chemical properties in common with spent fuel and the fact that the mineralisation extends from the ground surface and downwards makes it possible to study and compare the geochemical reaction of uranium under both oxidising (near the surface) and reducing (at depth) conditions. The Palmottu ore deposit has been investigated as a natural analogue to spent fuel in granitic rock since 1988 and the results obtained have been reported. The continuation of the study has been suggested as a project to EU and accepted for their support. The new Palmottu project is managed by the Geological Survey of Finland and SKB is participating together with ENRESA (Spain), BRGM (France), and other organisations in Finland, the UK and Germany. It started in 1995 and is expected to continue until 1999.

Natural hyperalkaline areas found in Jordan are being studied as analogues to underground repositories for low- and intermediate level waste, where concrete is used for construction and waste packaging. The pH of groundwater in Jordan hyperalkaline areas reaches values of about 12 – 13, which has generated typical solid cement phases among the natural minerals in the area. The environment is rich in elements, which also occur in waste as nuclides. Three things can be studied: the development of concrete, the development of a concrete influenced rock environment and the behaviour of radionuclide elements in such an environment. The first phase of the project was jointly funded by NAGRA, NIREX and Ontario Hydro. SKB participated in the second phase together with NAGRA and NIREX. A third phase is now under way, jointly supported by HMIP (Her Majesty's Inspectorate of Pollution), NAGRA, NIREX, and SKB.

Old constructions made of Portland cement are continuously being sampled to investigate the development of the cement paste. Samples have now been taken from the foundation of an old school built at the end of the 19th century in the Swedish town Gävle, from a dam (tunnel) in Älvkarleby in Sweden built in 1917, from a 90 year old water tank from the castle in the Swedish town Uppsala, from the hydropower dam at Sillre/Oxsjön, from Mid-skogsforsen hydropower plant and from Rocksta mill. The results point to the benefits of a tight concrete, saturated conditions and stagnant groundwater. This will improve the quality of the cement by slow hydration reactions. An interim report is available.

7.3.6 Biosphere

The biosphere studies address the transport of nuclides from the aquifers above the bedrock, through natural and domestic ecological systems and into different foodstuffs. Dose to man is calculated as an endpoint and compared to regulative limits. Dose to (or effect on) biota other than man is also considered.

If radionuclides are released from the repository they will enter the biosphere in primary receptors of the deep groundwaters. They will be diluted or accumulate as they are transported in ecological systems, and can finally be consumed causing dose to man or other species. A set of dose factors for the different groundwater recipients and based on typical but reasonably pessimistic ecosystems and on unfavorable assumptions regarding mans use of natural resources is under development for coming safety assessments. The activities during 1995 have continued to have a focus on model comparison and testing, mainly within the international programs BIOMOVS and VAMP. Here the modelling tool, BIOPATH, and the uncertainty tool, PRISM, used by SKB have been tested in several applications.

7.4 OTHER LONG-LIVED WASTE THAN SPENT NUCLEAR FUEL

Long-lived low and intermediate level waste (LLW and ILW) is a third category of waste in addition to spent fuel and short-lived LLW and ILW. The quantities are relatively minor and the main sources are waste from research activities and used components from the power reactors which have been situated inside or near the reactor core (core components and reactor internals). Core components are stored at CLAB and research waste is collected, stored and conditioned at Studsvik. The present concept for disposal of this waste is to build a facility near the repository for spent fuel and at a comparable depth. It will consist of three parts: SFL 3, 4 and 5. SFL 3 and 5 will consist of concrete constructions for final storage inside caverns. SFL 4 is simply the tunnel system remaining after the completed emplacement of in SFL 3 and 5. Waste from Studsvik and operational waste from CLAB and the encapsulation plant will be emplaced in SFL 3. Decommissioning waste from CLAB and the encapsulation plant will finally be placed in SFL 4. Reactor core components and internal parts, all packed in concrete containers, will be disposed of in SFL 5. Concrete, sand and/or bentonite will be used as backfill in the various parts of the repository.

The total volume of waste is estimated to about 25 000 m^3, but strictly taken not all of this falls into the category of long-lived waste. More than half of the total volume consist of waste which could in principle be disposed of in SFR such as operational waste and decommissioning waste from CLAB and the encapsulation plant. However, SFL 3-5 can also receive short-lived LLW and ILW that arises in a post-closure period of SFR.

A prestudy has been made with the aim to make a first preliminary assessment of the near-field barriers to radionuclide dispersal. The prestudy has been concluded and reported, and it was indicated that the barriers of the conceptual design are efficient to protect man and the environment from the waste. The investigations have been continued in a second phase starting in October 94 with the main aim to prepare for a future safety assessment. The second phase of the study of other long-lived waste consists of the following parts:

- Preparation of tables with radionuclide content and waste composition to be used in a safety assessment.
- Preparation of a chemical data base containing information on water chemistry, concrete composition and chemistry, radionuclide sorption, diffusion and solubility, organic complexes and colloids.
- Analysis of alternative scenarios (e.g. ice age), hydraulic influences, the effects of colloids, microbes and gas formation.
- Compilation of barrier properties; waste package, concrete construction, near-field rock, backfill of concrete, bentonite and sand.
- Comparison between different design alternatives.
- Testing and development of transport models.

Disposal of long-lived LLW and ILW is being studied in other countries too. Therefore an informal exchange of experience has been established between SKB and the organisations ANDRA (France), NAGRA (Switzerland) and NIREX (the UK).

7.5 ÄSPÖ HARD ROCK LABORATORY

The Äspö Hard Rock Laboratory has been constructed to provide an opportunity for research, development and demonstration in a realistic and undisturbed underground rock environment down to the depth planned for the future deep repository. The work performed is described in more detail in the Äspö HRL Annual Report 1995 /7-1/ and Chapter 18.

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. The total length of the tunnel is 3,600 m. The last 400 meters were excavated with a tunnel boring machine (TBM) with a diameter of 5 meters. 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. Construction of the facility was completed in the summer of 1995.

Work during the construction phase has focused on verification of pre-investigation methods and development of detailed investigation methodology which has been applied during tunnel construction. The comprehensive work on data collection for detailed characterization of the underground at Äspö was completed during 1995. These results are used for comparison of the predictions made of rock properties and groundwater flow and composition based on surface and borehole data with actual observations in the tunnel. Procedures for management of large quantities of data have now been developed to a point where SKB is in possession of a data production methodology that meets rigorous requirements on quality and overview. Results from investigations made during the construction phase are now reported. The final reporting of the experiences and evaluation of results from the pre-investigation and construction phases is in progress.

To obtain a better understanding of the properties of the disturbed zone around underground opening and the importance of different excavation methods, ANDRA, UK Nirex and SKB have joined efforts to conduct a study of the disturbed zone. The ZEDEX project comprised investigations before, during and after excavation of drifts excavated by tunnel boring machine (TBM) and by drill and blast methods (D&B). The far-field measurements, more than 2 m from the drift perimeter, showed no evidence of damage to the rock for any of the excavation techniques. However, hydraulic changes were observed which may result from opening, closing or minor shearing of existing fractures due to stress redistribution. In the near-field, there was little evidence of damage around the drift excavated by Tunnel Boring Machine (TBM) except within a few centimeters of the drift perimeter. The measured extent of the damaged zone for the Drill & Blast (D&B) drift, reached a maximum depth of about 80 cm in the floor of the tunnel, where higher energy explosives were used and was less well developed in the walls.

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). 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. The first set of tracer tests are currently in progress and will be used for predictive transport modelling by the Task Force on modelling of groundwater flow and transport of solutes.

To identify suitable sites for the experiments to be undertaken during the Operating Phase a drilling and investigation program was completed in the beginning of 1995. The site selected for the first Stage of the Tracer Retention Understanding Experiments (TRUE) was then characterized in greater detail. One of the features identified was then selected for tracer testing. The selected feature is a reactivated mylonite with one major fracture plane and a few sub-parallel minor fractures.

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. A special borehole probe, CHEMLAB, has been designed for different kinds of retention experiments where data can be obtained representative for the *in situ* properties of groundwater at repository depth. The results of experiments in the CHEMLAB probe will be used 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 buffer material, transport out of a damaged canister and transport in an individual fracture. In addition, the influence of naturally reducing conditions on solubility and sorption of radionuclides will be tested. The CHEMLAB probe is currently being manufactured and will be put into use in 1996.

A "Task Force" with representatives of the project's international participants has been formed. The Task Force shall be a forum for the organizations supporting the Äspö Hard Rock Laboratory Project to interact in the area of conceptual and numerical modelling of groundwater flow and solute transport in fractured rock. The evaluation of Task No 1, the LPT2 pumping and tracer tests has been completed. A wide variety of conceptual as well as numerical models have been used to predict water flow and tracer breakthrough in this rather large scale. Eleven modelling groups participated in modelling of the LPT2 test. Evaluation has shown that all models, with respect to groundwater flow, represented the measured LPT2 data well. The hydraulic impact of the tunnel excavation at Äspö HRL was defined as the 3rd Modelling Task. The objective is to evaluate how the monitoring and the study of the hydraulic impact of the tunnel excavation may help for site characterization. This is an exercise in forward as well as inverse modelling which is currently in progress. Task 4 which includes predictive modelling of the TRUE radially converging and dipole tracer tests has recently been initiated. The Task Force has also produced an Issue Evaluation Table listing key issues related to the performance of the geological barrier.

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 this framework, a full-scale prototype of the deep repository will be built to simulate all steps in the deposition sequence. 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.

Preparatory tests of compaction techniques of different backfill materials were performed during the autumn of 1995. The *in situ* compaction tests, which were performed in the final part of the TBM tunnel, have been completed and the compacted back-fill has been excavated and removed from the tunnel. The compaction technique developed, where a vibrating plate is used on inclined layers, was successful and provided higher density than horizontal compaction. Laboratory measurements have been performed to provide data on properties of different mixtures of ballast material and bentonite.

Presently (March 1996) ten organizations (including SKB) from nine countries participate in the work at the Äspö Hard Rock Laboratory and contribute to the results obtained. The results of this work are reported in the Äspö International Cooperation Reports. The Bundesministerium für Bildung, Wissenschaft, Forschung und Technologie (BMFT) in Germany joined the collaboration in July 1995.

7.6 ALTERNATIVE METHODS

The main direction of the SKB RD&D-programme is towards completing the first step with deposition of some 5—0 % of the spent fuel in a repository within about 20 years time. In parallel the work on alternative treatment and disposal methods is followed and supported in a limited scale.

During the last few years the possibility for **partitioning and transmutation (P&T)** has attracted renewed interest. SKB supports some work in this area at the Royal Institute of Technology (KTH) in Stockholm and at Chalmers Institute of Technology (CTH) in Gothenburgh. The work at KTH is emphasized on safety related issues and at CTH on processes for partitioning. A status report on the P&T-work /7-2/ has been issued during 1995. A main conclusion is that at present there seems to be no economical gain and only an insignificant reduction in potential future radiation doses from P&T as compared to current industrial fuel cycles. An assessment of P&T is also included in SKB's RD&D-Programme 95.

SKB has established a separate research project on the "Very Deep Hole Concept" aiming at improving our knowledge about conditions at 1000 – 5000 m depth in the geosphere. So far the activities have focused on state-of-the-art compilations of data bases in terms of tectonics, hydrogeology, geophysics, rock mechanics and geochemistry.

8 COST CALCULATIONS

8.1 COST CALCULATIONS AND BACK-END FEE

According to Swedish law all back-end activities including the decommissioning of the nuclear power plants are the responsibility of the nuclear power plant owners. The costs are covered by a fee on nuclear electricity paid to the State and collected in funds, one for each nuclear power plant. The fee is set annually by the government.

Each year SKB calculates the future electricity production and the future costs for the back-end operations related to this electricity production. The results of the 1995 calculations were presented in PLAN 95 /8-1/. The total future electricity production (from 1995) was estimated to be about 1 080 TWh, if all twelve reactors are operated to the year 2010. Up to the end of 1994 about 930 TWh have been produced making a total of about 2 010 TWh in the Swedish programme. For this production a fuel amount of about 7 840 tonnes of U is required.

The total future back-end costs were estimated to be about GSEK 49.2 (price level of January 1995) 1 GSEK = 10^9 SEK = 0.14 · 10^9 US$. Up to and including 1995 already GSEK 10.1 have been spent. The total cost for the back-end of the nuclear fuel cycle is thus about GSEK 59.3. The breakdown of the costs are roughly (old reprocessing costs excluded):

Transportation of waste	5%
Interim storage of spent fuel	17%
Encapsulation and final disposal of spent fuel and long-lived waste	40%
Final disposal of operational and nuclear power plant decommissioning waste	5%
Decommissioning and dismantling of nuclear power plants	22%
Miscellaneous including R&D, pilot facilities, and siting	11%

Based on SKB's cost calculations and the estimated real interest rate, the government has decided that the fee for 1996 shall be SEK 0.019 per kWh on an average. This is the same fee as for the last thirteen years.

The fee has been paid into funds at the Bank of Sweden. These funds are administrated by The Swedish Nuclear Power Inspectorate (SKI), who in 1992 took over this responsibility from the previous National Board for Spent Nuclear Fuel, SKN. The total sum in the four funds was at the end of 1995 about GSEK 17.9, an increase by GSEK 2.5 during 1995.

In the end of 1995 the Parliament decided to change the financing system and the management of the funds. The purpose was to make the full financial responsibility of the nuclear utilities more transparent and to improve the interest earned on the funds. To this end the following revisions were introduced in the Act on Financing:

- The funds shall be administered by a new organization called "The Nuclear Waste Fund".
- The funds shall be invested in the National Debt Office and earn the same interest as State bonds and similar instruments issued by the National Debt Office, e.g. long term bonds with a real rate of interest.
- The utilities shall provide formal guarantees that all costs for the radioactive waste management will be covered. The guarantees shall cover the case of a premature shutdown of a reactor, as well as unexpected cost increases.
- With the introduction of guarantees, the contingencies normally included in the fees can be removed, so that the fees in the future shall reflect the probable costs.
- The fees shall be based on 25 years of reactor operation and a probable future real rate of interest on the funded money.

These change are effective as of January 1, 1996. In the beginning of 1996 about GSEK 12,5, has been invested in long-term (18 and 8 years) instruments with a guaranteed real rate of return of 4,3 – 4,7%.

8.2 REPROCESSING

The Swedish policy for the management of spent fuel is the once-through strategy without reprocessing of the spent fuel. SKB has therefore transferred the rights to use its contracts with COGEMA to other customers.

A small portion of the Swedish spent nuclear fuel (about 140 tonnes) is planned to be reprocessed at BNFL's facility at Sellafield.

8.3 DECOMMISSIONING OF NUCLEAR POWER PLANTS

In 1994 a comprehensive study of the technology and costs for decommissioning the Swedish nuclear reactors was completed /8-2/. The study was focused on two reference plants, the boiling water reactor (BWR) Oskarshamn 3 and the pressurized water reactor (PWR) Ringhals 2. Subsequently the result from these plants have been translated to the other Swedish plants.

The cost of decommissioning Oskarshamn 3 was estimated to be about MSEK 940 in January 1994 prices. The estimate for Ringhals 2 was MSEK 640. In total for all twelve reactors the cost of dismantling will be MSEK 8 800.

These cost estimates are considerably lower than the cost estimates made in many other countries. During 1995 much work has been devoted, both by SKB and SKI, to understanding why this is the case. Comparisons have been made with German and US cost estimates /8-3, 4/ and most of the differences have been explained.

The most important factor is the well developed Swedish waste management system with sea transports and disposal in SFR. This system allows the removal of large components in fairly simple waste packages. There is thus no need for large scale segmentation of these components during dismantling, except for the pressure vessel and internals with a high dose rate.

Another factor is that much of the planning and preparation can be made with in-house staff and a third factor is the different regulatory environment.

From the decommissioning of all twelve Swedish reactors a total of about 140 000 m^3 of radioactive waste will be generated (the exact amount of waste depends on how much material that will be decontaminated). Most of this will be transported to the SFR final repository at Forsmark.

SKB's engagement in the OECD/NEA international cooperative programme on decommissioning has continued during 1995. SKB is responsible for the programme coordinator function. The programme comprises 29 decommissioning projects in 11 countries. The majority of the projects are small first generation power demonstration reactors.

The projects include all stages of decommissioning from preparation for a long-term rest and surveillance period of the plant to a total dismantling. Examples of the latter are the Shippingport reactor in the USA where the dismantling was completed in 1988, the Japanese JPDR reactor and the German Niederaichbach reactor where the dismantling has been completed.

9 NUCLEAR FUEL SUPPLY

Sweden imports all uranium for its nuclear power plants, and the purchasing is normally handled by the utilities.

SKB is in charge of industry-wide coordination and matters relating to market surveys, strategic stockpiling of uranium and certain purchases of enriched uranium.

9.1 NATURAL URANIUM

The Swedish nuclear power plants have annual requirements of about 1 600 tonnes of natural uranium. These requirements are met by producers from Australia, Canada, Kazakstan, Uzbekistan and the Russian Federation. The uranium is mined mainly in modern mines, mostly open-pit mines. Production from Kazakstan and Uzbekistan is mined by in-situ leaching, ISL, which is a method that gives small impact on the environment.

There are important stocks of uranium both in the west and in CIS-countries. As these stocks are now being sold, both the long term price and the spot price have been lowered in recent years, in spite of the fact that uranium production is declining, see Figure 9-1. A new source is coming to the market as high enriched uranium from nuclear weapons disarmament will be diluted to low enriched uranium in the Russian Federation and sold to the US.

In Sweden there are low-grade uranium resources, however the cost of producing from these resources would be much above world market prices. There was some production in southern Sweden from shales near Ranstad in the late 1960-ies. That area is now being restored by the SKB sister company SVAFO.

9.2 CONVERSION AND ENRICHMENT

Conversion is a chemical process for production of uranium hexafluoride from uranium concentrates. Natural ura-

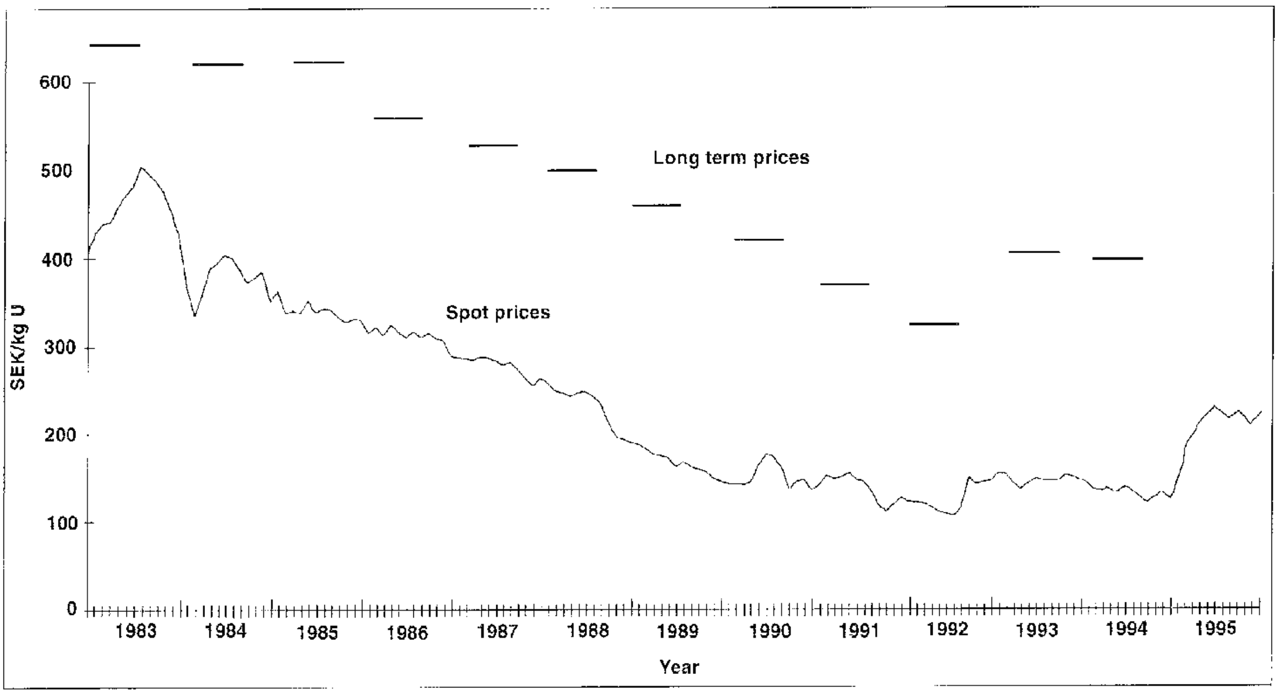

Figure 9-1. Long term and spot prices for uranium.

Long term price = Average price for long term deliveries to the European Community.
Spot price = Average spot price each month for the unrestricted market, published by the German company NUKEM, 1983 – 1994, and spot price for the restricted market from 1995 (Sweden entered the European Union in 1995 which means that uranium purchases are now regulated in accordance with the Euratom Treaty).

Figure 9-2. Gas centrifuges in the Ural Electrochemical Integrated Plant for enrichment of uranium.

nium contains 0.71% of the isotope uranium-235. Enrichment is a process to increase this content up to 3-5% of uranium-235. This low enriched uranium is a suitable fuel for light water reactors which are used in Sweden.

The Swedish utilities have a diversified and reliable supply of conversion services from Canada, France, United Kingdom and the USA. There is also a reliable supply of enrichment services from Eurodif in France, Urenco in the Netherlands, the United Kingdom and Germany, and USEC in the USA.

Techsnabexport Co Ltd in the Russian Federation delivers low enriched uranium to the Swedish utilities, which means that this includes both natural uranium, conversion and enrichment. Deliveries to Sweden come from the Ural Electrochemical Integrated Plant in Novouralsk, see Figure 9-2. SKB transports such low enriched uranium by the ship M/S Sigyn from the port of St Petersburg to the fuel fabrication plant in Västerås, Sweden.

9.3 FABRICATION OF FUEL ASSEMBLIES

The Swedish Utilities are purchasing fuel fabrication services with the objective of lowest fuel cycle cost. This procedure has led to many orders to ABB Atom, but also to French, German, Spanish and US fuel fabricators.

Fabrication of fuel assemblies both for BWRs and for PWRs as well as BWR channels, BWR control rods and other components are performed in Sweden at the ABB Atom Fuel Fabrication Facility in Västerås.

Fabrication of nuclear fuel at the ABB Atom Facility was around 195 tonnes of UO_2 during 1995. Of this volume about 105 tonnes were exported to Belgium, Finland, Germany, Switzerland, United States and South Korea. Furthermore, about 150 tonnes of UO_2 were delivered to other fuel fabricators as UO_2-powder or UO_2-pellets.

The ABB Atom fuel assembly design SVEA 96/100, where the fuel rods are divided into four minibundles with 5 x 5 rods separated by a water cross, is now the dominating BWR fuel design used in Swedish BWRs.

9.4 NUCLEAR FUEL STOCKPILE

SKB is responsible for holding a strategic stockpile of low enriched uranium and zirkaloy, corresponding to an electricity production of 35 TWh. This amount has been decided by the Swedish parliament.

Uranium in the above mentioned stockpile, in fuel under fabrication and at the nuclear power stations is sufficient for about two years of operation of the twelve reactors in Sweden.

Table 9-1. Costs for nuclear fuel in 1995.

	SEK/kWh	Million SEK in 1995
Natural uranium	0.008	530
Conversion	0.001	70
Isotope enrichment	0.008	530
Fuel fabrication	0.009	600
Strategic stockpile	0.001	70
Total nuclear fuel	**0.027**	**1.800**

9.5 COSTS

The costs for front end supply of nuclear fuel in 1995 in Sweden are shown in Table 9-1 (the production of nuclear electricity was 66.7 TWh in 1995).

The costs for nuclear fuel have been stable since 1987 up till and including 1995, see Figure 9-3.

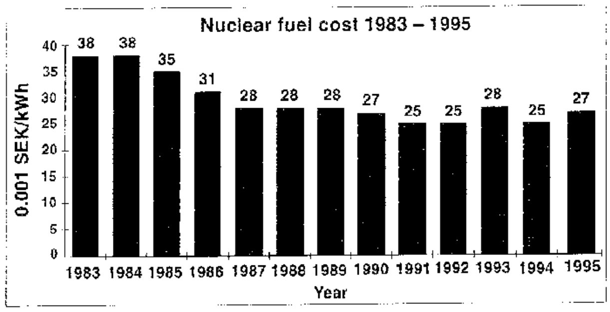

Figure 9-3. *Nuclear fuel costs 1983 – 1995.*

10 CONSULTING SERVICES

10.1 BACKGROUND

The international review of the KBS reports (1978–84) made SKB's activities internationally recognized. Since then SKB has actively participated in international co-operation activities and strengthened its position as an attractive partner. As a consequence foreign organizations have shown an interest in contracting SKB for services in their own programs.

The international interest for SKB has several reasons. Sweden has developed a well functioning system for transports and disposal of radioactive waste. SKB has a facility for interim storage of spent fuel (CLAB) and a repository for low- and intermediate-level waste (SFR). In addition SKB has a comprehensive RD&D program and a broad distribution of technical reports.

Since 1984 there is a special division – NWM (Nuclear Waste Management) – within SKB for marketing and management of external services. For each assignment a tailored project team is organized with due consideration of the competence required. It may be experts from SKB's own staff or from groups contracted for different tasks in the Swedish radioactive waste management program.

SKB's external services shall, of course, carry their own costs with some margin. They are, however, also of value by stimulating the staff, improving their competence and broaden their views.

Since 1984 more than 120 assignments have been accomplished for organizations in Australia, Belarus, Belgium, Canada, Czech Republic, Finland, France, Hungary, Japan, Lithuania, Estonia, Russian Federation, South Korea, Spain, Switzerland, Taiwan, United Kingdom and USA. The assignments have dealt with long-term safety, overall planning, canister and buffer materials, transports, field investigations, site selection, decommissioning and facility design.

10.2 NWM WORK DURING 1995

During 1995 SKB was contracted by organizations in Estonia, Lithuania, Belarus, Finland, Spain, Belgium, France, Taiwan and Japan. In parallel marketing activities have been going on in Southeast Asia, Russian Federation, Republic of Korea and also in some European countries. In all some twenty assignments have been concluded, distributed over nine countries.

The activities in Lithuania continued to have high priority during the whole year. Manufacturing of the first ten intermediate dry spent fuel storage casks started and first delivery to Ignalina NPP is scheduled for May 1996. The project work for the cementation facility for spent resins has been somewhat delayed. A contract is now planned for autumn 1996. The long-term waste management plan for Lithuania has been supplemented with cost estimations which is foreseen to be used in connection to the implementation of a funding system for radioactive waste management in Lithuania.

As a result of the national waste management plan a number of new projects have been given priority. Discussions are under way and project implementations are scheduled for spring 1996.

The decommissioning work of the two shorebased nuclear submarines has continued in Estonia. A preliminary decommission plan has been prepared as well a Site Management Plan. Work is also in progress regarding categorization, sorting and handling of solid waste including twenty control rods.

On behalf of the Ministry of Emergency in Belarus planning work has started regarding a national waste management plan.

Extensive radarreflection (RAMAC) as well as rock stress measurements have been carried out in several deep boreholes in Finland on behalf of TVO, see Figure 10-2.

Continued support has been given to the Spanish geological company ITGE in connection to development of hydrogeological borehole instruments.

Continued assistance has been given to Japan Nuclear Fuel Ltd, JNFL, in connection to gastransport through bentonite based barriers in connection to the phase 2 licensing procedure of the Rokkasho low level waste repository.

A report describing SKB's R&D work on "The Natural Barrier System" has been prepared for Mitsubishi Corporation in Japan.

Figure 10-1. *Aerial photograph of the Paldiski facility in Estonia.*

Figure 10-2. Equipment for rock stress measurements, which have been carried out on behalf of TVO in Finland.

On behalf of Energy and Resources Laboratories, ERL, as well in agreement with Taiwan Power Company, TPC, a course has accomplished in Sweden named "Training of Staff in Site Investigations Methodology and Technique". The course comprised theory as well as practical work in the field, see Figure 10-3.

Figure 10-3. Taiwanese specialists together with Geosigma and SKB personnel.

11 PUBLIC AFFAIRS AND MEDIA RELATIONS

11. 1 GENERAL

According to Swedish law, the nuclear power utilities are obliged to adopt whatever measures are needed to manage and dispose of the nuclear waste in a safe manner. SKB, which has been given responsibility for this in practice, has been conducting research and development towards this end for more than 20 years. A system which deals with all the waste for a long time to come has been in operation since 1988, and plans are now being made for how to dispose of the waste in the longer term.

Information and communication concerning nuclear waste disposal – both the core problems and the principles on which the solutions should be based – is an important part of the nuclear power industry's responsibility. People have a right to know how this waste will be disposed of, the costs of which they are paying via their electric bill. Furthermore, good knowledge is a prerequisite in order for the public at large, local and national authorities and the government to be able to pursue the democratic decision-making process that is required to realize the long-term solutions.

It is important that the information should put the waste issue in a proper perspective. Its importance must not be underplayed, since the waste constitutes a grave risk if it is not properly managed. On the other hand, the risk posed by waste that has been properly managed and disposed of must not be exaggerated either. One difficulty is that there is often a big difference between how qualified experts and other groups evaluate risks. A risk analyst often sees risk as a product of the probability of something occurring and the possible consequences of this happening. Others add factors that have to do with the perception of the risk. The total risk that is ultimately perceived consists of both the "technical" risk and the conceptual risk. For most people, the perceived risk is greater if it is forced on them rather than being undertaken voluntarily (for example, air pollution or passive smoking compared with "voluntary" smoking). In the same way, a "new" risk is perceived as being greater than an old familiar one (for example, AIDS compared with the much greater risk in Sweden of cardiovascular disease). Another important factor is whether the situation is controlled by others rather than under a person's own control (for example, the conceptual risk of travelling by air compared with driving a car). These are some of the factors that influence people's risk perceptions, and which should be appreciated by both experts and others.

The goal of SKB's information is to broaden and deepen the public's knowledge regarding:

- the radioactive waste that exists today, and the fact that it will pose a risk in the future if it isn't handled properly, for example if it goes astray;
- the fundamental ethical and technical principles that guide Swedish waste management policy:
  - the nuclear waste must be dealt with in a responsible fashion with high standards of safety,
  - the planned systems must be designed so that we do not shift any environmental or economic burdens to future generations;
- the fact that the knowledge and the capability to build safe repositories now exists in Sweden and other countries and that SKB is actively participating in international research and development;
- the system we have built up in Sweden and that is already being used to dispose of all radioactive waste for a long time to come;
- the work SKB has now begun of siting a deep repository for spent nuclear fuel. Apart from the disposal method, we aim to have in 20 years' time a site, a facility under construction and funds. This will enable future decision-makers to either continue along the beaten path, or choose other solutions.

11.2 SKB's INFORMATION ACTIVITIES

The best way to bring about a dialogue with people is to meet them face-to-face. That is why SKB holds exhibitions on a large scale, with the participation of the company's own personnel. Visits are made to schools, local communities and trade fairs throughout the year, in some cases with SKB's exhibition trailer. In this way, SKB gets to meet the general public, local political and community leaders and special-interest groups in an open dialogue.

SKB's facilities – CLAB, SFR and the Äspö Hard Rock Laboratory – are open to visitors by appointment and have permanent exhibitions that can be visited year-round. At the localities where SKB initiates feasibility studies, information offices are opened with associated exhibitions. There, interested visitors can come into direct contact with representatives of SKB.

SKB also has a broad selection of information material, such as brochures and reports, video cassettes, overhead

Figure 11-1. *SKB conducts extensive information activities in Sweden's schools.*

Figure 11-2. *SKB's transport vessel M/S Sigyn is used during the summer as a floating exhibition hall. In the cargo hold, visitors meet SKB's personnel, who tell them about Sweden's radioactive waste.*

Figure 11-3. *SKB ran a major series of ads in 1995.*

transparencies with speaker scripts, audio cassettes, mini-exhibitions, touch-screen computers etc.

The basic philosophy is that anyone who wants to should be able to find out the facts, principles and future plans for the radioactive waste.

As in previous years, SKB pursued ambulatory information activities during 1995. With lecturers and mobile exhibitions, SKB visited 42 schools, meeting 6,878 pupils in 315 classes, see Figure 11-1. Increased interest was noted during the year among teachers, and special teacher conferences were arranged in conjunction with the school visits. The school information package "At Depth" was revised after an evaluation in 1995 and is now available at most upper-secondary schools in the country. SKB also took part in eight trade fairs of different kinds.

During the summer the transport ship M/S Sigyn served as a floating exhibition hall, see Figure 11-2. For the seventh year in a row, visitors were able to view equipment used to handle the waste, such as transport casks, as well as models of the planned deep repository and the prospective canister. New for the year was a series of on-board seminars to which citizens' groups and politicians were invited for debate. SKB's exhibitions were visited by a total of 61,000 persons, including members of the public, upper-secondary school pupils, local political and community leaders and special-interest groups. Of these, 51,000 visited the exhibition on board the Sigyn.

SKB's facilities CLAB, SFR and the Äspö Hard Rock Laboratory (HRL) also received a large number of visitors from both Sweden and other countries. Among the visitors were U.S. senators and several groups of teachers, who spent their in-service training days visiting the facilities.

During the year, SKB ran a major series of advertisements in the daily press and trade journals, see Figure 11-3. The ads dealt with four questions relating to the Swedish nuclear waste: How long is it hazardous? Who takes care of it? What do we do with it? and Where will it be disposed of? A fact book could also be ordered from the company free of cost during the ad campaign. In response to the ads, 6,000 people ordered additional information from SKB.

Lagerbladet, SKB's newsletter, was published twice during 1995 and distributed to about 25,000 subscribers.

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Part II

Research, Development and Demonstration during

1995

CONTENTS PART II

12 RD&D-PROGRAMME

12.1 BACKGROUND

According to the 12 § of the Act on Nuclear Activities the owners of the nuclear power plants are responsible for conducting the research, development and other measures necessary for the safe handling and disposal of radioactive wastes arising from the nuclear power production. A programme for conducting the necessary activities must be submitted to the pertinent authority every third year. By the end of September 1992 SKB submitted its third RD&D-Programme /12-1/ to the Swedish Nuclear Power Inspectorate – SKI. The government decision on the programme was given in December 1993. The government decided that the programme should be supplemented by SKB The supplement thus requested by the government was submitted by SKB to the Nuclear Power Inspectorate in August 1994 /12-2/.

A government decision on this supplement was given on May 18 1995. A brief summary of main points in that decision is given in section 12.2.

In late September 1995 SKB submitted its fourth RD&D-programme to SKI /12-3/. The programme covers the period 1996-2001 in some detail. A brief summary is given in section 12.3. The programme was sent by SKI for review to about 60 organizations. The SKI review is expected to be available in the spring of 1996 and the review by KASAM in mid-1996 before the government decides on RD&D-Programme 95.

12.2 GOVERNMENT'S DECISION ON SUPPLEMENT TO RD&D-PROGRAMME 92

The following is a brief account of some of the main points in the government's decision on SKB's supplement to RD&D-Programme 92 /12-4/.

The government decides that SKB in its continued programme shall present plans and programmes for establishment of technical requirements on barriers, systems and components based on performance assessments as well as for investigations of possible repository sites. Furthermore, general siting studies as well as site specific feasibility studies shall in a comprehensive way be presented in future RD&D-programmes.

The government directs SKI to give financial support from the waste funds at the request from municipalities where SKB conducts feasibility studies. The money shall be used to cover costs for the municipality to follow and evaluate the study and to inform the public about matters concerning final disposal of nuclear fuel waste. The financial support shall not exceed 2 million kronor per year.

The government finds that SKB has supplemented its RD&D-Programme 92 in accordance with the requirements set forth in the government's decision Dec 13, 1993.

The government finds that a (future) decision to give a permit to construct an encapsulation plant will mean considerable bindings with respect to future handling and disposal methods. Such a decision should not be taken until a comprehensive safety assessment of the whole management system including the final disposal step has been submitted and the method has been shown to be suitable.

The government states that SKB has in a comprehensive way described its view on criteria and methods to find a suitable site for a deep repository. The siting factors and criteria given by SKB are a suitable starting point for the continued siting work.

The government finds that the planned detailed characterization of a site constitutes a step in the construction of a nuclear facility intended to for final disposal of long-lived radioactive wastes. Thus the application for a permit to start such characterization will be evaluated according to both the Act on Management of Natural Resources and the Act on Nuclear Activities. The applications for such a permit should include an account of feasibility studies for 5-10 municipalities, of geoscientific site investigations of at least two sites and also the reasons for selecting these sites.

12.3 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 was published in a separate report in the autumn of 1995 /12-5/.

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, among other things, dependent on the fact that certain knowledge must be available before the next step is taken. 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 methods for excavation and construction of the repository. 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.

13 TECHNICAL DEVELOPMENT ON CU-STEEL CANISTER FOR SPENT FUEL

13.1 DEVELOPMENT OF DESIGN

13.1.1 General

The work of designing the canister takes 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. A status report on this work was presented in the RD&D-Programme 95.

13.1.2 Requirements on performance and properties

The requirements on the properties of the canister has been 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 requires that the canisters are 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. The canister must thus meet two primary functional requirements in order to provide the necessary isolation in the deep repository:

The canisters must remain **impervious** over a long time, which in turn imposes requirements on

- initial integrity,
- corrosion resistance and
- strength.

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

The canister design must meet the requirements made by both normal and abnormal operating cases in the plants and during transport. 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
- possibly retrieved 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.

13.1.3 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 1995 in a slightly modified design of 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.

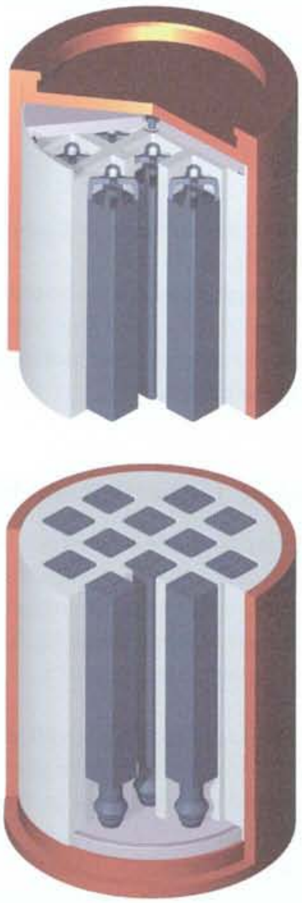

Figure 13-1. Design of copper canister with cast insert.

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 13-1. 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 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 electron beam welding. The canister insert is of cast steel and has a minimum wall thickness of 50 mm.

13.2 MATERIAL STUDIES AND TESTS

13.2.1 General

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 proneness 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.

13.2.2 Investigated materials

Copper canister

The principal strategy has been to select as pure a copper material 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.

A pure copper with the following limits and 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.

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.

13.2.3 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.

External corrosion

Modelling of corrosion processes in copper shows the following corrosion depths after 100,000 years for a probable case and a conservative case /13-1/:

	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

Corrosion due to formation of sulphide "whiskers" has been investigated by means of literature studies and attempts to model whisker formation. 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 departure should perhaps be further investigated.

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. If this is true, microbial corrosion could not have any decisive effect on the life of the canister.

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. More recent results have shown that this is not the case, but that stress corrosion cracking is unlikely under repository conditions. This has been confirmed by subsequent investigations, which show that the tendency towards stress corrosion cracking in the repository environment is small /13-2/.

Internal corrosion

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. Earlier investigations of the sensitivity of the steel canister to stress corrosion cracking caused by radiolysis products show that stress corrosion cracking cannot be ruled out if the residual air content exceeds 10 volume ppm. 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 outer copper shell has been investigated in recent years /13-3/. 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 done.

Under aerobic (oxic) 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 oxygenated period. Under anaerobic (anoxic) 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.

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 in the deep repository. This load deforms the copper canister and presses it against the insert. Maximum strains in the copper wall are calculated to be less than 4%. 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. 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.

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. The grain size in the material and the shape of the grains also affect the resolution in ultrasonic investigations. The ultrasonic signal is spread and attenuated if grains are large or irregular. A fine-grained 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.

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.

13.3 CANISTER FABRICATION

13.3.1 General

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 1994–95, 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 other two methods /13-4/.

The 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 live up to the objective. But there are good prospects for achieving the desired grain size in the material by means of modified process parameters and possibly cooling during extrusion.

Regarding other fabrication methods, HIP and electrodeposition, studies are under way to determine the value of carrying out trial fabrication on a full scale with them as well. 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.

13.3.2 Results of trial fabrication 1994–95

Trial fabrication of four canisters was begun in 1994. The canisters will have an insert fabricated from a steel tube /13-4/. The first full-scale canister with self-supporting inner steel canister is shown in Figure 13-2. It was completed with sealing of the lid at TWI in September 1995.

Figure 13-2. *The first full-scale canister produced during fabrication trials.*

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 and extrusion of whole tubes. In both cases, a bottom is then welded on by means of electron beam welding.

The selected copper alloy 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 fabrication of tube from plate. The most important thing in the forming operation is to fabricate the tube halves with such precision that they can be finish-machined, see Figure 13-3, 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.

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.

Figure 13-3. Assembling of copper cylinder and steel insert.

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. 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 of cast iron, since the alloy has very good castability. If steel or a bronze is chosen, the price is higher due to poorer material yield and higher material prices.

13.3.3 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 /13-5/. 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 one 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.

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.

13.4 SEALING METHOD

13.4.1 General

It must be possible to seal the copper canister to high standards of reliability and leaktightness, as well as inspectability. To fulfil the stringent requirements on sealing of the copper canister, a method is being developed employing electron beam welding of the 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. Methods for nondestructive testing of the weld are being developed in parallel in order to verify that it complies with the stipulated requirements. Formulation of requirements is also under way.

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. Welding with both horizontal and vertical electron beam was done on oxygen-free pure copper and on oxygen-free copper with a low phosphorus content on straight workpieces. The main purpose of the work was to determine the optimum pressure in the welding chamber for lid welding in the pressure range 5 Pa to 100 kPa.

13.4.2 Results of trial welding 1994–95

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, but the scattered pores that were detected in the weld have been deemed to be acceptable. The trial serie was completed in 1995 and some of the conclusions were /13-6/:

- A weld with a penetration of 70 mm, with of 10 mm and a round bootomed profile can be made using 75 kW beam power.
- Root porosity can be eliminated by using suitable beam focus parameters.
- The electron beam needs to be tilted down from the horizontal to maintain good beam to joint alignment.
- A fronting bar is necessary to support the top bed and it must be firmly attached to the lid or the canister wall.

The changes made in lid design during these trails 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 equipment's long-term stability was tested and the high-voltage equipment was modified to reduce the risk of and consequences of discharges.

Results obtained so far 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.

13.4.3 Nondestructive testing

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. That sensitivity that could be achieved corresponded to 0.5 mm side-drilled holes. A total of seven lid welds have been tested with ultrasonics. All were inspected with pulse-echo of compression waves from above the weld, most with manual scanning /13-7/. 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. Detectability for defects was not always directly related to size.
  Gel as a coupling medium, instead of immersion in liquid, sometimes gave inadequate coupling, and water as a contact 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 detection of surface-breaking defects should be further explored.

14 TECHNICAL PLANNING OF SITE INVESTIGATIONS AND CONSTRUCTION OF A DEEP REPOSITORY

14.1 PLANNING FOR SITE INVESTIGATIONS

14.1.1 General

As a preparation for the forthcoming site investigations for candidate repository sites, planning work is going on in the following fields:

- development of the geoscientific investigation programme,
- preparation of techniques and routines for data management,
- preparation of instruments and methods, including development, refinement and investment, etc.

A general base for the planning work is the experiences from earlier site investigations, including the Äspö Hard Rock Laboratory (HRL), conducted by SKB.

14.1.2 Geoscientific investigation programme

A programme for geoscientific site investigations is in preparation. It will be published in good time before site investigations are begun. SKB RD&D-Programme 95 gave the general guidelines for this site investigation programme.

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.

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 14-1. 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.

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 a suitable site in the order of 5 km^2 within a stipulated area where the potential for a deep repository is greatest and thereby where the continued investigations should be concentrated. The choice of site will be confirmed by mapping of fracture zones, rock type boundaries and other geological conditions at depth. Seismic reflection surveys and the first deep exploratory drilling are carried out during this stage with analyses and measurements of groundwater chemistry, hydraulic conductivity and rock stresses as key parameters.

Provided that the initial investigations and analyses indicate suitable conditions, the investigation programme proceeds with complete site investigations. The complete site investigation is carried out in two drilling and measurement programme steps, aiming at verifying and supplementing the picture of rock discontinuities distribution and rock type properties in more detail.

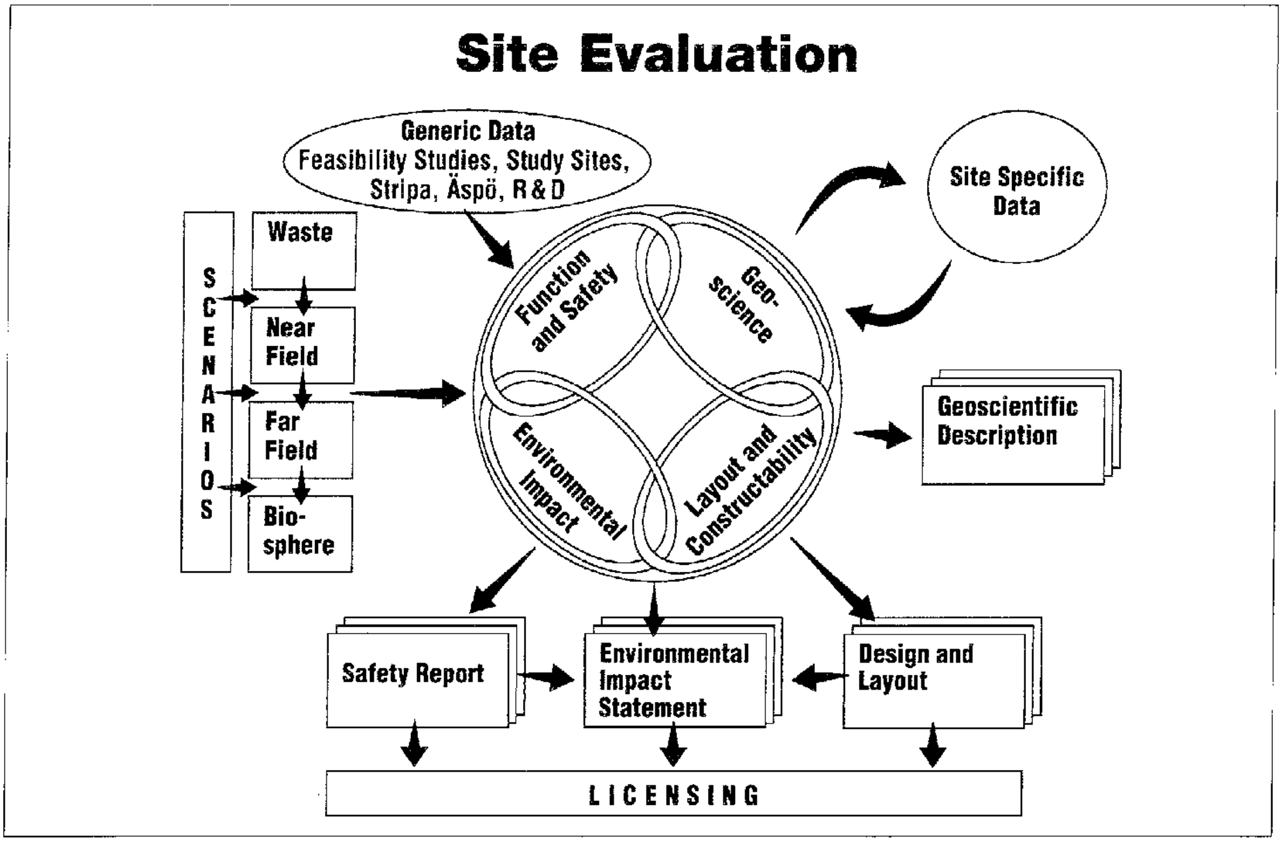

Figure 14-1. Site evaluation activities and their interaction.

A tentative logistic plan for the interaction within the site evaluation is shown in Figure 14-2.

The parameters to be investigated and measured will first of all be defined by the siting factors of importance for siting of a deep repository /14-1/. This is the case in particular for those which are related to the geosphere and which are of relevance for the site investigation stage of the siting process. Specification of these parameters, with reference to which of the siting factors the parameter is related to, is underway.

14.1.3 Data management techniques

Efficiency and correctness in the management of data is of most importance for a site investigation programme. Strict handling of data will be needed for the quality assurance of the investigations, in which the traceability of data of all steps in the data refinement chain, from data collection to final result, is a major task. All investigations will be carried out according to QA-plans which in turn refer to manuals or other specifying documents. Routines for the QA procedures are under preparation and some of them have been used in ongoing field measurements at the Laxemar site.

One prerequisite for correct data management is the existence of an efficient central database. The intention is that all produced data shall be stored in that database as well as the database shall be the central source for further data processing and interpretation work, i.e. the database will serve as a QA-tool in the investigation process. Therefore the database system must be very efficient in data storing as well as in data retrieval. Based on the previous geological database and the experiences gained in the Äspö HRL, a new database (SICADA) has been developed, see further description under Äspö HRL. The new database will record and manage not only the data but also the individual activities which have produced the data.

As a tool for rock modelling and visualization of structures, rock type bodies, etc in the rock volume, a "Rock Visualization System" (RVS) is under development. Also this development is carried out in the Äspö HRL, see Chapter 18.

14.1.4 Instruments and methods

In general, instruments and methods that will be used in the site investigations will depend on what kind of measurements shall be carried out according to the site investigation programme. Experiences from SKB's earlier site studies, existing tools and established methods will be the base for the measurements, but new and modified

MAIN ACTIVITIES DURING SITE INVESTIGATIONS

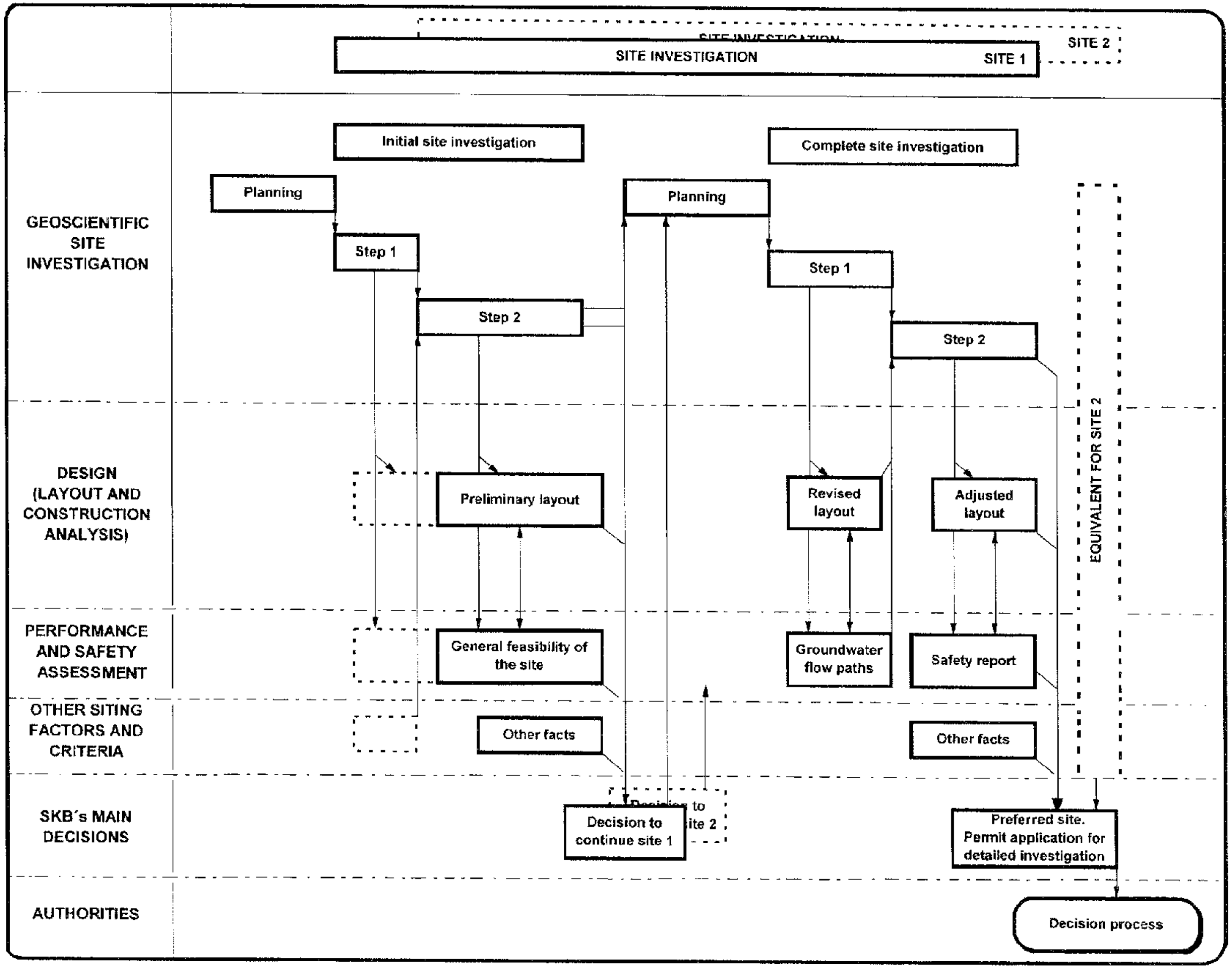

Figure 14-2. Main activities during site investigations.

methods will be used as well. Preparation work carried out during 1995 includes development of new methods, refinement of measurement tools and testing of routines in conjunction with field work.

BIP-1500 borehole-TV system

The new borehole-TV system BIPS 1500 was described in the Annual Report for 1994. The system has been extensively used during 1995, for several SKB project but also in Finland and in France. Time has been spent on building up routines for measurements, data analysis and production of high resolution colour images. The capacity and correctness of the method to identify and describe fracture orientation in relation to core logging and other methods will be evaluated.

Logging has been carried out in several borehole diameters with use of centralizers, one for each diameter, which is very important in order to get the right geometry in the images of the borehole wall. Minor modifications in the software are underway.

Experiences from logging show that the method is not very sensitive to particles in the borehole water, due to the small space between the window and the borehole wall. However, in large diameter holes this can be a problem. In old holes the quality of the image might be bad due to precipitation on the borehole wall. This effect varies between holes depending on the groundwater chemistry. In one borehole (KLX02) for example the precipitation increased between loggings performed during a period of a year while in another approx 8 years old hole only minor precipitation occurred. To be sure, TV logging shall be performed quite soon after drilling has been completed.

The new borehole TV system opens up a refined methodology to make geological documentation of a borehole. The TV-logging will probably be the base for the documentation while the information from the drill core will be used to add measurements of the geological characteristics of the rock, fracture surface and fracture fillings. The aim is to integrate these two activities into one "geological borehole documentation" system, a work which has been initiated.

Depth calibration technique

Comparisons of results from different measurements in boreholes, or integrated analysis of collected data, sometimes fail due to incorrect depth data for the measurements. Even with relatively good accuracy of the depth measurements during logging etc, like 0.1–1%, the absolute error may be as much as 1–10 m for a 1000 m borehole. The incorrectness is different for the different borehole equipment and depends on the inclination of the borehole, groundwater level etc.

In the Annual Report for 1994 a method was described of how to increase the correctness by drilling-in depth markers (diamond impregnated copper rings) at appropriate levels in the borehole. During 1995 a first tool for setting rings in 76 mm holes was finished and secondly another tool for 56 mm holes was made. In 1996 the project will continue with a full scale field test in a borehole but also to adapt sensors to the different borehole measurement tools and to develop efficient routines for correcting the depths during borehole measurements.

Hydraulic testing

SKB is using two different kinds of equipments for hydraulic testing, the Umbilical Hose System and the Pipe String System. The Umbilical Hose System is used for injection tests only, but the tests are automatically performed by means of a computer programme. The hose principle enables fast hoisting up and down the hole. Some prestudies on modifying the technique by integrating down-hole pump and flow meter have been initiated during 1995.

The Pipe String System is used both for injection tests and pumping tests, with higher capacity compared with the Umbilical Hose System. On the other hand the hoisting is more time consuming due to pipe handling. To be able to carry out site investigations at two sites in parallel a second Pipe String System is under manufacturing.

Field tests with the existing Pipe String System have been conducted in the deep Laxemar borehole KLX02 during 1995. Pumping tests from 300 m sections were carried out, the deepest 1100–1400 m down. Malfunctions in the pressure measurement system, partly due to fabrication errors, made that the measurement programme had to stop earlier than planned, see also section 16.3.5. This actualized the need for exchanging some components in the measurement system and the need of redundancy, where possible.

Others

Technical documentation of instruments constructed by SKB is being put together. The level of documentation for instruments and methods must be relatively high in order to fulfil the goals for the quality assurance of the site investigations.

A programme for service and maintenance of SKB equipment is running. Even if not in use at present, the accessibility and functionality of all instruments must be high for field measurements.

Instruments and methods related to the Äspö HRL, i.e. underground investigation methods and instruments for experiments (planned or ongoing) of various kind are reported in /14-2/. In special, the development of CHEMLAB, a system for various radionuclide experiments in boreholes might be of interest for the reader, see Figure 18-6.

14.2 TECHNICAL STUDIES CONCERNING THE CONSTRUCTION OF A DEEP REPOSITORY SYSTEM

14.2.1 Analysis of global thermo-mechanical effects from a repository

A study of the global thermo-mechanical effects caused by a KBS-3 type repository is in progress. The study is divided into three phases:

1. Effects of different thermal loading in an elastic model.
2. Effects of non-linear behaviour in solid rock materials and discontinuities.
3. Influence on stresses and fracturing from the excavated geometry of the repository.

The first two phases have been finalized and are reported in /14-3, 4/.

In the *first phase*, three thermal intensities (6.0, 6.9 and 10.0 W/m^2) and two thermal conductivities of the rock mass (3.0 and 3.7 W/m,K) were used in six analyses, with each analysis carried out for the time of 100, 200, 400, 1 000, 2 500, 10 000 and 60 000 years after deposition. With a canister spacing of 6.0 m in the deposition drifts and a drift spacing of 40 m, the 6.0 W/m^2 is equivalent to 1 440 W per canister. The analyses were conducted with the 3-dimensional program STRESS3D.

Results and conclusions in the first phase were based on three factors:

1. Temperature level and distribution.
2. Presence of tensile stresses.
3. Calculated strength/stress ratios for two sets of material properties.

The maximum temperature of 70.5°C and maximum principal stress of 66.1 MPa were obtained in the center of the repository when applying highest thermal load and lowest conductivity. The maximum temperature and principle stress respectively occurred 400 and 100 years respectively after deposition.

Tensile stresses developed at shallow depths for all combinations of thermal loading and material properties. A maximum tensile stress of 3.4 MPa occurred close to the ground surface at 1 000 years of thermal loading. The stresses developed down to 80 m below ground surface for the 10.0 W/m^2 loading case.

A large amount of shear failures were calculated at shallow depths as a result of the low level of principal stresses. The highest thermal loading also resulted in a minor shear failure inside the repository.

In the *second phase* heat intensities of 6 and 10 W/m^2 were applied on three fracture networks with increasing degree of complexity. Simplifications of the Äspö network were used. The different networks consisted of two, four and nine fracture zones, respectively, all of them cutting through the complete model. The thermal loading was applied over a 1 km^2 large deposition site (as in the first phase) and calculations were done for the of 100, 400 and 1 000 years after deposition. The modelling was conducted with the 3-dimensional distinct element program 3DEC using Mohr-Coulomb material model for intact rock and Coulomb slip criterion for fracture zones.

All comparisons of results between the two phases are based on the same temperature influence. The stress increase at the repository level was generally highest for the continuum elastic cases and became gradually lower as more joints were included in the models. The area with tensile stresses extended somewhat downwords. In general, fracture zones closed near the repository and expanded at shallow depths.

Applying 10 W/m^2 resulted in local yielding of rock material in some of the fracture zones. The more fracture zones that were introduced in the model, the more the yielding area decreased. Some of the results from phase 2 are illustrated in Figures 14-3 to 14-7.

The results from the first two phases of the study indicate that it should be possible to store spent nuclear fuel with an initial thermal load of 10 W/m^2 in a KBS-3 type repository.

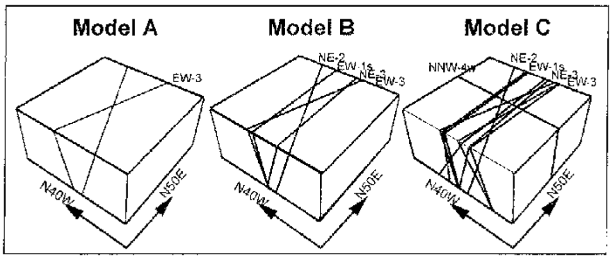

Figure 14-3. *Fracture zone geometries.*

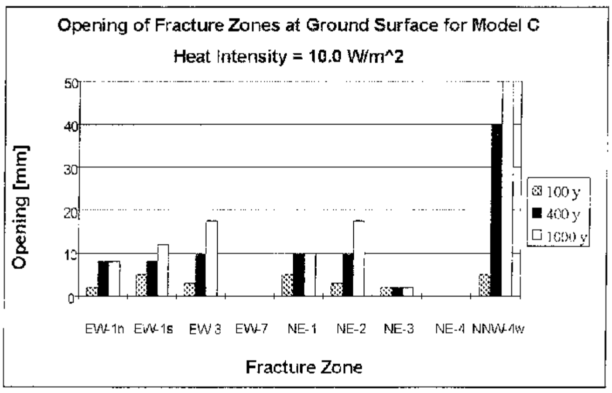

Figure 14-4. *Maximum opening of fracture zones.*

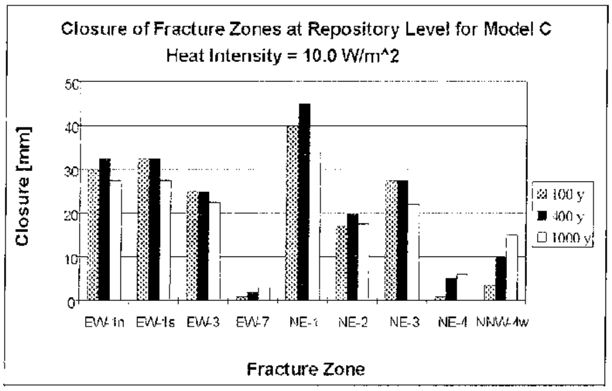

Figure 14-5. Maximum closure of fracture zones.

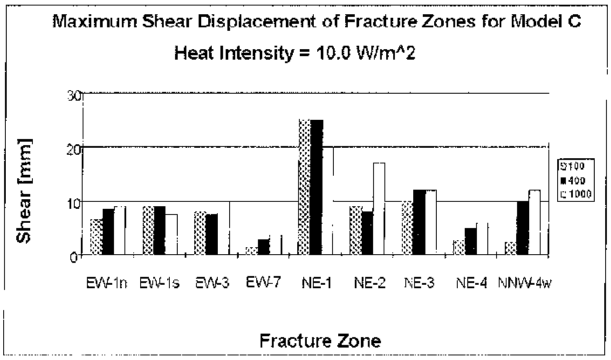

Figure 14-6. Maximum shear displacements.

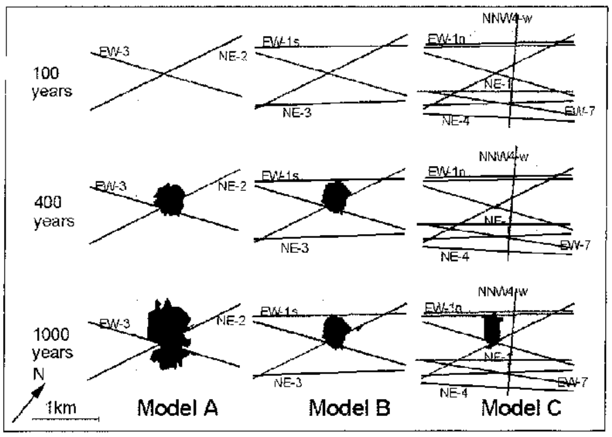

Figure 14-7. Shear failure at ground surface for heat intensity of 10 W/m² (shaded area). The square represents projection of repository boundary.

14.2.2 Calculations of distances between deposition tunnels and deposition holes due to temperature limitations in the bentonite

The shape of an underground repository for spent fuel will be influenced by demand for limited temperature in the bentonite around the canister. The temperature in the bentonite is influenced mainly by the distance between the canisters, size of the canister, the thermal effect of the fuel and thermal properties of the surrounding materials.

Distances between deposition tunnels and canisters that are connected together are calculated in /14-5/ with the finite element program "Solvia". The calculations are based on combinations of different thermal loads, allowed temperatures in the bentonite and conductivities in the bentonite. Four different deposition concepts were studied:

- F1 Vertical deposition holes, diameter 1 580 mm, one canister in each hole.
- F2 As F1, but two canisters in each hole.
- F3 Horizontal deposition holes, diameter 1 580 mm, in both sides of the tunnel.
- F4 As F3, but hole diameter 2 400 mm.

Some of the most important data used in the calculations were:

- Canister length 4 850 mm, diameter 880 mm.
- Conductivity of the bentonite was alternatively 0.75, 1.00 and 1.50 W/m,K.
- Allowed increase of temperature was alternatively 62 K, 72 K and 82 K.
- The initial thermal load at deposition was respectively 1 300 W and 1 600 W.

A diagram showing the maximum increase of temperature in the bentonite as a function of the distance between the canisters and between the deposition tunnels was drawn for each case. The distances that corresponded to 62, 72 and 82 K temperature increase in the bentonite respectively could by that be decided and be shown in diagrams, see example in Figure 14-8.

From those diagrams some conclusions can be drawn:

1. Increased thermal load in the canisters will result in increased repository area. (If the number of canisters will be reduced, the repository area may in some combinations be reduced.)
2. Increased conductivity or increased allowed temperature in the bentonite will result in reduced repository area.
3. Reduced distance between deposition tunnels will result in reduced repository area, but also in increased distance between the canisters.

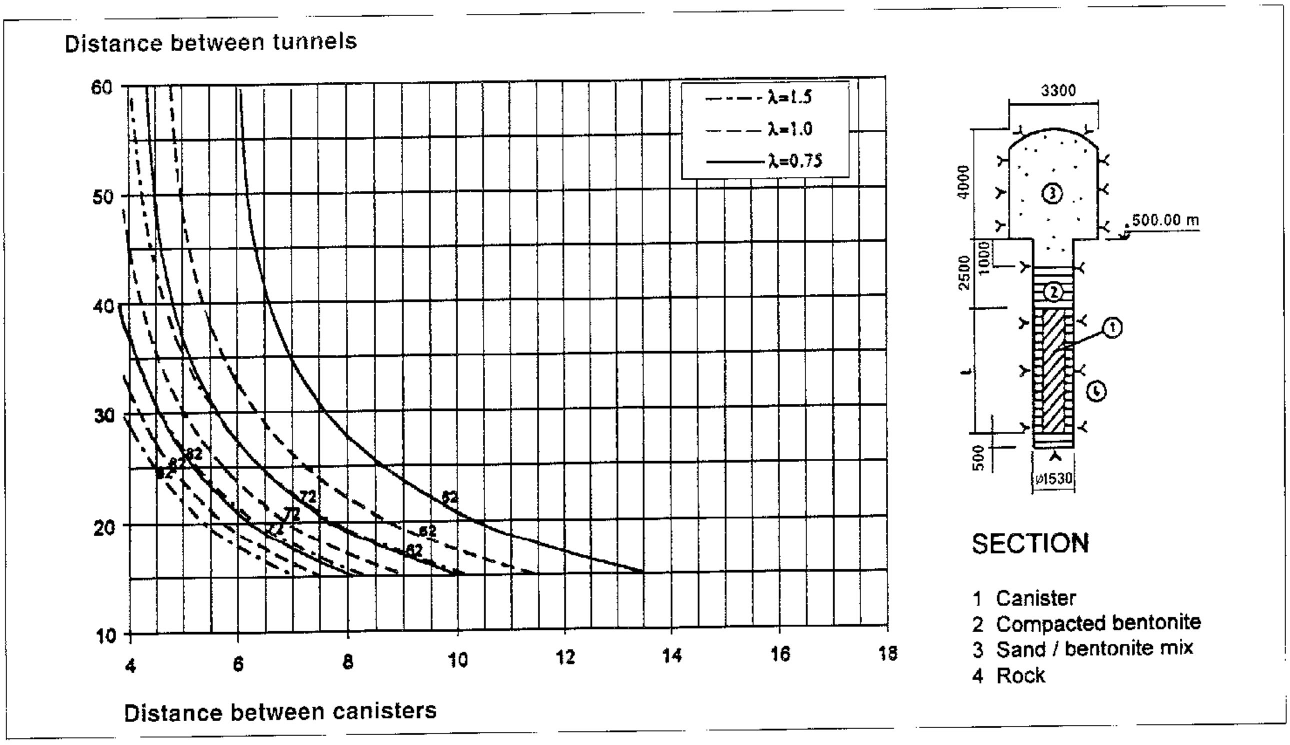

Figure 14-8. Diagram showing maximum temperature increase, dT (K), on the surface of the canister at different thermal conductivities, λ (W/m,K), in the compacted bentonite. Heat load 1 300 W/canister. Deposition concept F1.

4. If the distance between the canisters is minimized, the repository area will be increased.
5. The repository area will be smallest in alternative F3 and largest in F4.
6. The total length of deposition tunnels will in most cases be considerably shorter in alternatives F2 and F3 than in F1 because two canisters will be disposed of in each position, while the distance between the positions will not increase too much.
7. Length reduction of repository tunnels according to conclusions no 2, 4 and 6 above will result in cost reduction.
8. The thermal load will in the different alternatives and parameter combinations vary between 5 and 10 W/m^2 repository area and by that be accepted with regard to global thermo-mechanical effects.

The results indicate that the following items are advantageous for a dense pattern of canisters.

- High thermal conductivity in the bentonite from start.
- High allowed temperature in the bentonite.
- Horizontal deposition and/or vertical deposition of two canisters in each hole.

14.2.3 Constructability analysis

The excavation of the repository introduces building materials into and onto the rock along the tunnels, ramps and shafts. These materials are cement-based grouts, and cement and iron based rock support materials. The amounts are of interest to the performance assessments and safety analyses as the majority of the material will have to be left when the repository is backfilled and sealed after deposition. During 1995 a constructability analysis was conducted covering a whole repository including accesses. In order to be able to base the analysis on realistic data all data were taken from the rock volume that has been modelled in Äspö HRL. Also the experience from the excavation of the ramp was used. The rock volume, however, is not large enough to host all sections of the repository but the data available have been used in the analysis over the whole hypothetical repository /14-6/.

The selected rock volume is situated at 450 m in the Äspö HRL. Both excavation by conventional drill and blast, and by full-face boring (Tunnel Boring Machine) have been considered. The large rock caverns in the section for other Long-lived Waste and in the Central Area underground have, however, been assumed to be developed by conventional drill and blast because of the sizes of the caverns. In both alternatives a ramp provides the main access to the repository level with a layout rather alike the actually excavated ramp in the Äspö HRL.

The constructability analysis has been based on a judgement of rock quality, the properties of existing discontinuities, and their distance of influence in the neighbouring rock. Different quality classes have been identified for both rock support and grouting based on standard solution for these activities that are used in Sweden. The classification schemes also differ between the two different excavation methods. Each tunnel, ramp and shaft have been separated into smaller sections which have been classified. One lack of accuracy in the estimate by this approach is that extraordinary efforts are not possible to consider in standard solutions, but they are estimated to represent a minor part of the consumeables required for the construction of the repository.

The result of the constructability analysis indicated a consumption of around 10 000 tonnes of cement in cement-based materials and around 1 000 tonnes of iron in the whole repository when conventional drill and blast is applied. In the case of TBM the consumption amounted to around half these figures. All figures are valid for Äspö HRL rock conditions only.

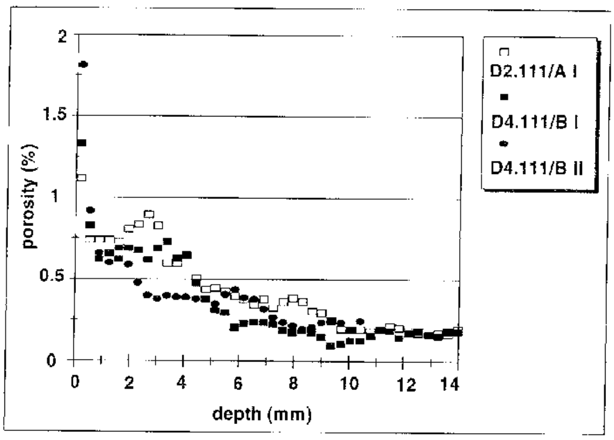

Figure 14-9. Porosity profile of three samples in hole 3 in the Research Tunnel at Olkiluoto.

14.2.4 Properties of excavation disturbed zone caused by boring of deposition holes

Three vertical full-scale deposition holes has been bored in the Finnish Research Tunnel at Olkiluoto with a novel boring technique which was based on rotary crushing and removal of the muck by vacuum flushing, as reported in last year's Annual Report /14-7/. During the boring changes were made in the operating parameters in order to register their effect on the quality of the hole. The studies were focused on the surface roughness of the hole and the properties of the disturbed zone, caused by the boring.

The surface roughness was measured by laser profilometer /14-8/. The combined average value for peaks and valleys is 1.3 mm indicating a roughness with a depth of a couple of mm. The increase in rock surface area due to this roughness is about 15%. The extra volume below a surface touching all peaks corresponds to about 1% of the hole volume. The properties of the disturbed zone were studied by two novel methods, the 14CPMMA method (carbon 14 polymethylmethacrylate) /14-9/ and the He-gas method /14-10/. The disturbance was studied in laboratory from 98 mm core samples taken from the wall at different locations in the holes, representing different boring parameters. With the 14CPMMA method the geometry of microfractures and porosity profiles were determined. The method involves impregnation of the rock with carbon 14 labelled methylmethacrylate, irradiation polymerization, autoradiography and optical densitometry with digital image processing techniques. The result indicated that the disturbed zone occurred as a zone of increased porosity to a depth of around 10 mm from the rock surface. Figure 14-9 shows an example of a porosity profile going radially out from the hole. The average measured porosity in this zone ranged from 0.3 to 1.8 vol-%, while the porosity of the undisturbed rock varied between 0.1 and 0.15 vol-%. The fissure apertures in the disturbed zone ranged from 500 nm to 20 μm.

The effective diffusion coefficients and permeability of the disturbed zone were determined by measuring helium gas diffusion and flow respectively. According to the results there is a zone of higher permeability and diffusivity closest to the surface of the hole. The thickness of the zone and the maximum depth of separate microcracks were estimated on the basis of the measurements. In general the permeability and diffusion coefficients in the disturbed rock samples were found to be one order of magnitude higher than those of the undisturbed rock. The extent of the disturbance was correlated to the geometry of the cutters and boring parameters indicating that the depth and the properties of the disturbed zone can be affected by the design of the cutter head.

14.2.5 Grouting

State of knowledge and development needs

To prevent groundwater inflow to tunnels and caverns and to provide mechanically stable conditions in very fractured rock, grouting operations may play a major role in the construction of the deep repository. When driving the ramp in the Äspö HRL, it was noted that well proven grouting methods were not enough, especially not in transmissive discontinuities having a high piezometric pressure. If grouting has to be carried out while constructing the deep repository, it is of prime interest that the results from the grouting operations can be predicted in the different kinds of discontinuities that has to be grouted.

In 1995 a compilation of the present state of knowledge and development needs was carried out /14-11/. The objectives were to outline specific repository related topics which need to be addressed, i.e.

- to develop theoretical knowledge, suitable grouting materials and technical know-how in tightening and

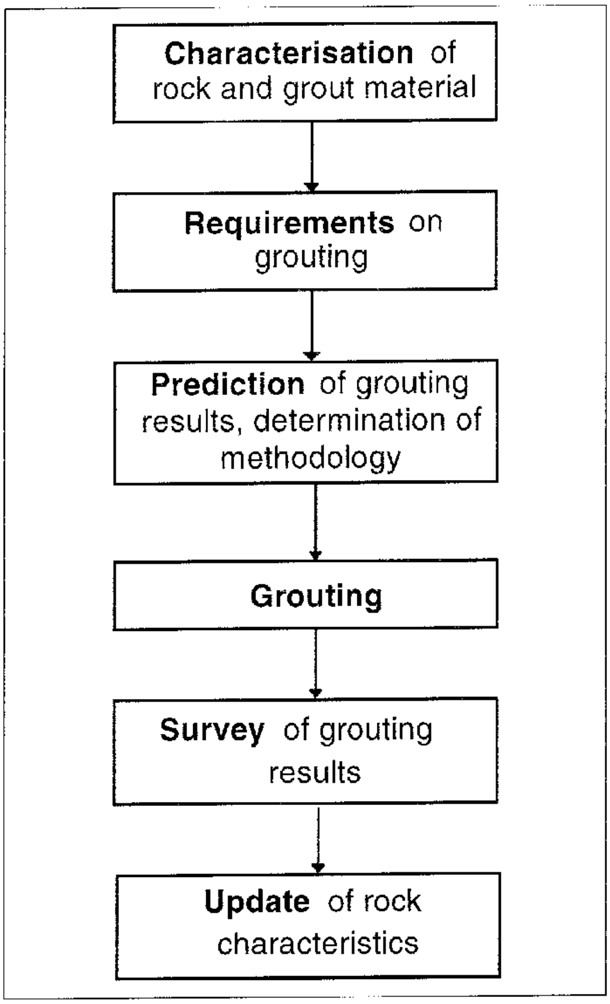

Figure 14-10. The grouting process.

stabilising extensive, water-bearing discontinuities having full piezometric pressure down to the repository depth,

- to define the needs of tightening minor and fine, discrete discontinuities and systems of such discontinuities, and to develop the theoretical knowledge and the technical know-how that are needed in order to meet those needs.

Three types of discontinuities, that might need grouting, are identified.

- Extensive, transmissive discontinuities.
- Strongly water-bearing discontinuities with moderate thickness and extension.
- Fine discontinuities and systems of fine discontinuities.

Grouting of the first two types of discontinuities are to be studied at first, because they are influencing the stability of ramp, shafts and tunnels. If they occur, such discontinuities may have to be crossed during ramping or shaft sinking down to the repository level, or during tunnelling on the repository level. The fine discontinuities have no influence on the stability but they are of significant importance in the repository area. The tightening of such fine discontinuities has been studied earlier in the Stripa project. Further development will come later in time.

The grouting method and grouting material may vary in different types of discontinuities, but the grouting process is always carried out according to the same schedule, see Figure 14-10. An inventory of the current state of knowledge among Swedish contractors and consultants has been initiated as a pilot study. This will also be a basis for further research and development work. In the late 1995, research and development work was initiated for:

- Characterisation of rock from a grouting point of view.
- Extension mechanisms of grouting in rock.
- Cement based grouting material.

14.2.6 Plugs for sealing off axial water flow along tunnels

General

Low permeability tunnel plugs, keyed into slots which cut in the rock walls, have been suggested as a means of cutting off axial flow-paths in the excavation disturbed zone (EDZ). In a previous numerical investigation, which included mechanical, hydraulical and hydromechanical aspects of the performance of tunnel plugs, it was assumed that the tunnel interior has a conductivity equal to that of the surrounding undisturbed rock mass, and that the EDZ is the major contributor to the axial flow /14-12/. These analyses were made by use of local axisymmetric continuum models. The effect achieved by plugging was found to be relatively modest, unless the EDZ conductivity was assumed to exceed the virgin rock conductivity by several orders of magnitude. The mechanical stability of the rock around the plug was found to be better if slots were cut in a triangular rather than in a rectangular shape. During 1995 larger and more general flow models, which take into account the heterogeneity of the rock mass surrounding a plugged tunnel section and the possibility of a permeable tunnel interior, have been analyzed /14-13/. The effects of slot excavation on the system of local discrete fractures have been analyzed by use of 3-dimensional distinct element models. The main objectives were:

- To estimate the axial flow along conductive deposition tunnels and to investigate how the flow rates can be reduced by tunnel plugs.
- To investigate under what conditions the effects of the plug would be significantly reduced, and to see if

the disturbance caused by tunnel and slot excavation may produce such conditions.

- To investigate if the conclusions drawn in the previous continuum investigation regarding slot shape are valid also if fractures are modeled explicitly.

Preliminary estimate

The flow along an infinitely conductive deposition tunnel can be estimated by use of analytical expressions if the surrounding rock mass is described as a porous continuum of infinite extension. The maximum flow appears to be essentially proportional to the tunnel length squared. Construction of plugs can schematically be regarded as a subdivision of a tunnel in shorter tunnel segments, each with a maximum flow that can be estimated by use of a modified version of the analytical expression. These preliminary estimates indicate flow rate reductions on the order of 95% if plugs are constructed every 100 m in a 1000 m tunnel.

FEM flow calculations

Figure 14-11 shows schematically how a conductive tunnel, integrated in a system of fracture zones, causes redirection of groundwater flow in the surrounding rock mass. Figure 14-12 shows flow rates obtained from an axisymmetric FEM model before and after construction of a plug symmetrically between the two zones that intersect the tunnel. For the data assumed in this simulation the plug gives a 99.9% reduction of the tunnel axial flow. This high relative reduction is a result of high flow rates prior to plugging. Less conservative assumptions regarding the conductivity of the tunnel backfill, the fracture zones and the EDZ, give a smaller absolute flow after plugging, but a smaller relative reduction. The conductivity of the rock in the vicinity of the plug is the key factor to the efficiency of the plug.

Reduction of the plug performance

The flow resistance provided by the rock outside the plug is essential to the plug performance. If the conductivity outside the slot is assumed to increase by three orders of magnitude compared with the case shown in Figure 14-12, the plug still reduces the axial flow by about 85% according to the FEM analyses. Neither the previous continuum analyses nor the 3D-distinct element calculations indicate, however, that stress relaxation effects associated with the slot excavation should give cause to permeability increases of this magnitude.

Slot shape

The 3D-distinct element calculations support the conclusions drawn previously regarding slot shape. The differences found between triangular and rectangular slots are, however, not large. In addition, every fracture system represents a unique case, meaning that it would be possible to find fracture geometries that would give a better performance for slots of rectangular shape.

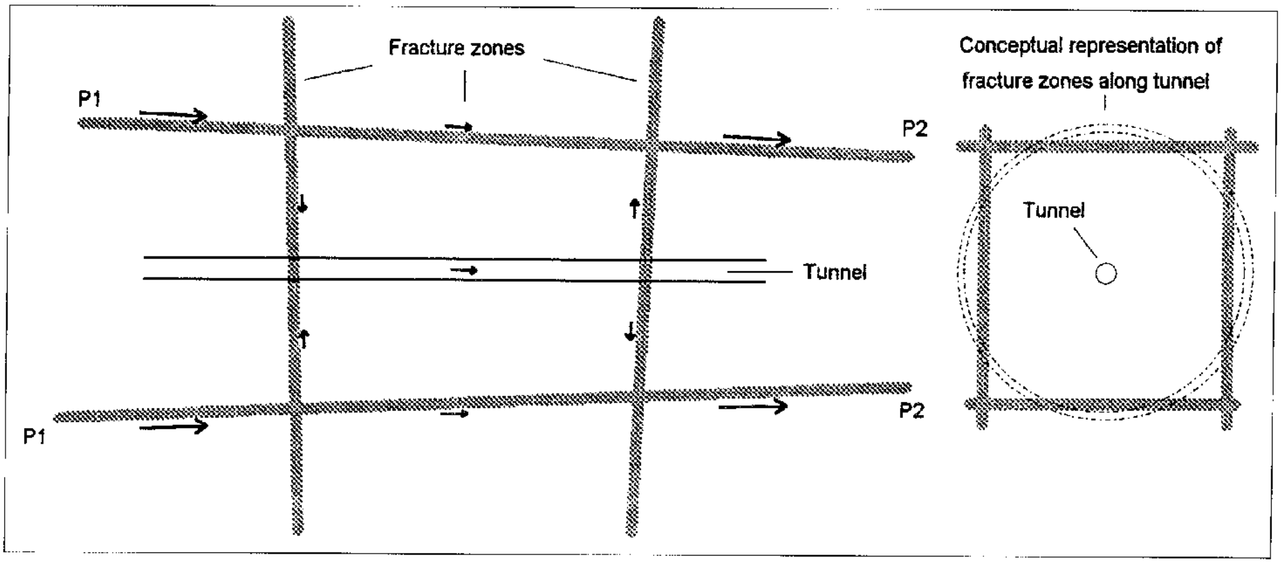

Figure 14-11. Left: Schematic illustration of groundwater flow redirection caused by conductive tunnel. Right: Transversal section of the tunnel and surrounding conductivity structures. The system of conductivity zones aligned with the tunnel may conceptually be represented by a cylindrical structure.

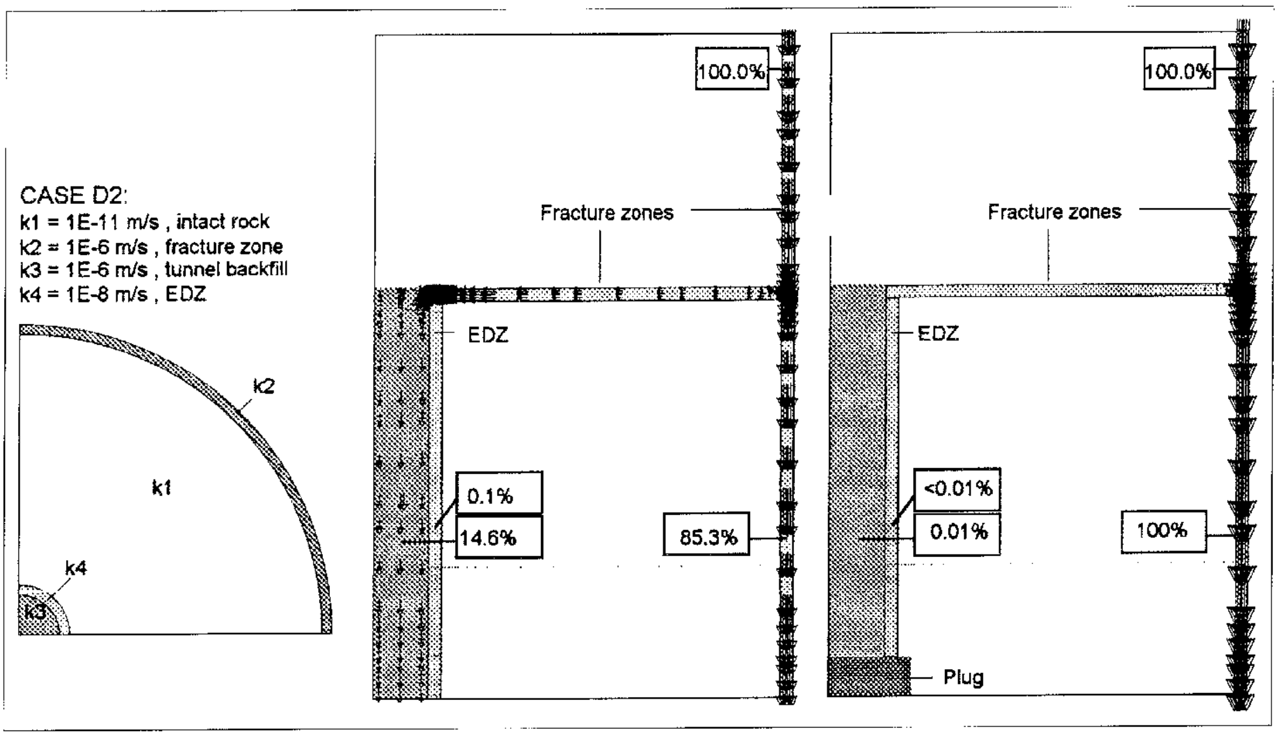

Figure 14-12. Flow rates before and after plugging. The left figure shows hydraulic conductivities assumed for the different parts of the model. The width of the EDZ and the width of the fracture zones are both set to 0.5 m. Numbers in dotted frames denote mean axial conductivity before and after tunnel excavation. Percentages are given in relation to the total axial flow through the model.

14.2.7 Methods for deposition and retrieval

Vertical deposition of canister

The reference deposition method according to KBS-3 is that the bentonite buffer is set whereafter the canister is lowered into the centre hole in the buffer blocks. The canister is handled without any surrounding radiation shield in the deposition tunnel thereby minimizing the hight in the deposition tunnel needed during deposition into the deposition hole. The shielding is instead arranged as a wall behind the deposition machine and against the transport tunnel. During 1995 conceptual studies have been conducted on the alternative with the buffer and the canister deposited together in a package, and how the deposition can be arranged if the canister is to be shielded all the way up to the deposition hole. These studies indicate the technical feasibility of several solutions and the needed increase in dimensions of the deposition tunnel.

Deposition of canister in radiation shield

The transport cask with the canister that comes from CLAB is placed in a reloading station above ground or under ground. There the canister is transferred from the transport cask to a radiation shielded deposition container. On the repository level the container is pulled on a low-built vehicle that is equipped with a revolving cradle. The vehicle is driven to the deposition tunnel where the container is transferred to the deposition vehicle operating in the deposition tunnel. The deposition operation is illustrated in Figure 14-13.

Deposition of canister and bentonite in one package

For this alternative a station is needed for assembling the package, a station that can be placed above or under ground. In case of an under ground location the bentonite blocks and the canister can be transported under ground in different units. The package that is confined in a cylinder, which is placed in a radiation shielding container, is transported to the deposition tunnel where the container with its load is transferred to the deposition vehicle, which moves to the deposition hole, raises the container and lowers the package. Figure 14-14 illustrates the situation when the package has been placed and the cylinder is being lifted up.

Horizontal deposition of canister

Horizontal emplacement of single canisters in holes on both sides of the deposition tunnel has been studied as an alternative that utilizes the tunnel in an efficient way. Several different feasible technical solutions were initial-

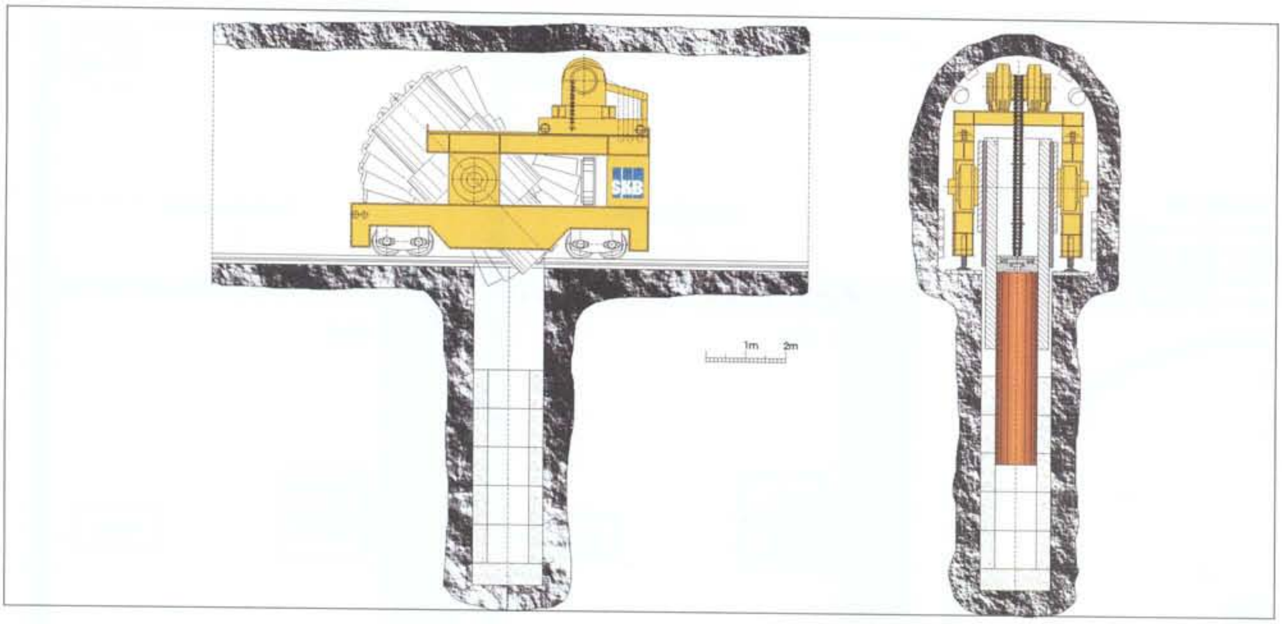

Figure 14-13. *Left: The deposition vehicle is positioned above the deposition hole and the container with the canister is raised and lowered into the upper part of the hole. Right: The canister is lowered into the centre of the bentonite blocks.*

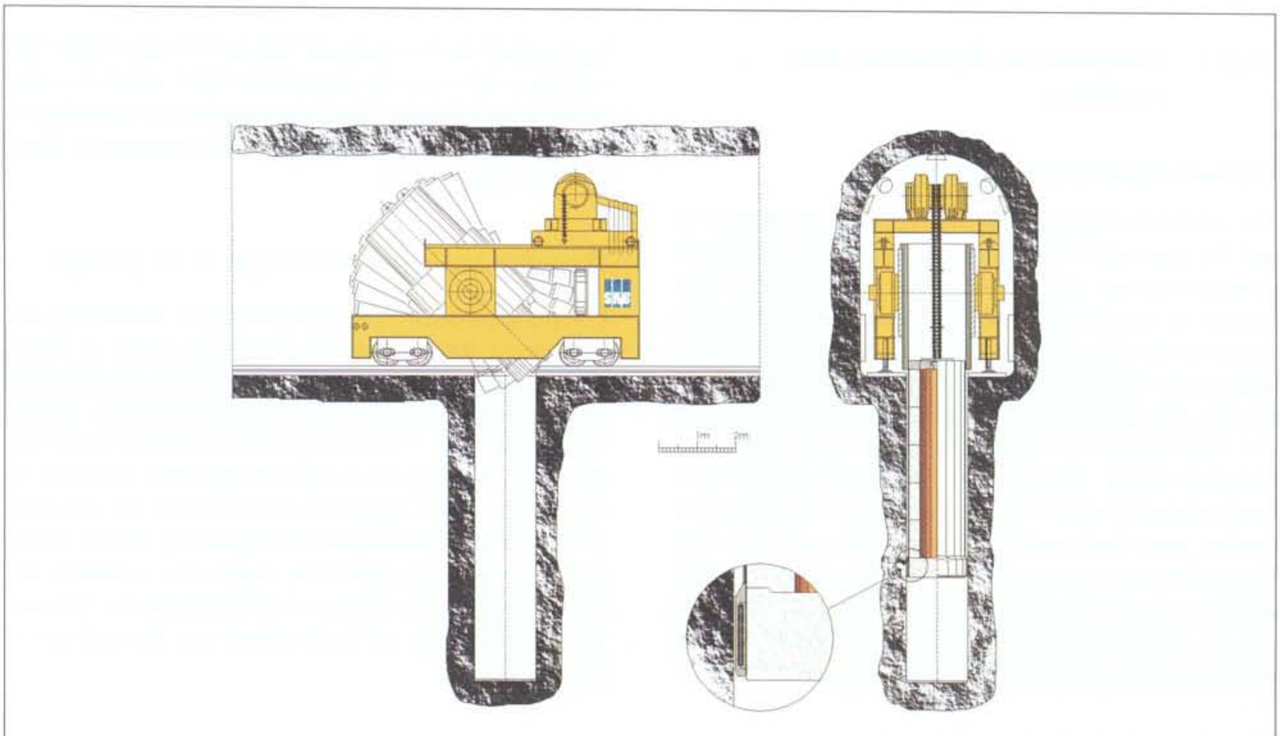

Figure 14-14. *Vehicle for deposition of canister and bentonite blocks in one package. Right: The cylinder is being lifted after having placed the package on the bottom of the hole. The enlarged detail shows the gripping device in the cylinder that consists of a compressed air operated rubber clamping sleeve. The device is attached to the bottom bentonite block with a force that is enough for holding the weight of the whole package during emplacement in the hole.*

ly recognized out of which two were judged to provide the best potential for development /14-14/. One of these alternatives is based on a sequence which starts with inactive emplacement of the bentonite buffer followed by emplacement of the canister. A guiding cylinder is placed inside the hole in the bentonite blocks for the canister to slide in so that the blocks are not damaged. This cylinder is retrieved after emplacement of the canister.

The other alternative is based on the emplacement of canister and bentonite blocks in one package. The difference between this alternative and the alternative regarding vertical emplacement is the tool for emplacement, which in a horizontal hole consists of a half cylinder with spherical wheels that rolls on the walls of the hole. The canister/bentonite and the bentonite and bentonite/ballast plug respectively are emplaced separately. The canister/bentonite package lies during transportation on the half cylinder within a radiation shielding transportation container. When the unit comes to the deposition hole the half cylinder with its load is pushed into the deposition hole. After that it reaches its position inside the hole the half cylinder is rotated 180° around its axial line so that the package comes on the floor of the hole and the cylinder on the top. The half cylinder is then pulled out and the plug package moved into the hole.

Retrieval of canister after backfilling

The emplacement process is considered to be possible to reverse in case of any incident during deposition that would require a restart in the deposition sequence. But after that emplacement has been completed, the top of the deposition hole and the deposition tunnel backfilled, and the bentonite saturated and swollen a retrieval will require that the bentonite is removed and that the canister is gripped before it can be lifted up.

The key process in retrieval of a stuck canister is the removal of the bentonite and studies have started on several alternatives. The application is general and does apply to both vertical and horizontal emplacement. The different methods can be separated into four main groups:

1 Mechanical methods
   - full-face boring,
   - cutting,
   - coring.

2 Hydrodynamic methods
   - high-pressure water jet excavation,
   - low-pressure water erosion,
   - disintegration of the bentonite in conjunction with water flushing.

3 Thermal methods
   - heating,
   - cooling.

4 Electrotechnical methods
   - electric concentration of water.

The analysis of feasibility of each method has started and also studies regarding the necessary technical background for judgement of differences in development potential.

15 SAFETY ANALYSIS

15.1 GENERAL

Methods to be used in the assessments of the long term safety of a deep repository were presented in a supplementary report to the SKB RD&D-Programme 1992 – published in August 1994 /15-1/. During 1995 the assessment methodology has been further discussed and developed. The SKB programme for coming safety assessments and a general discussion of the state of the art with regard to how to do them was presented to the Swedish authorities in the SKB RD&D-Programme 1995. See chapter 12 for a short summary or /15-2/. The section in the RD&D programme presenting the principles for radiation protection and how they are applied in the planned repository is reproduced in section 15.2 .

Permission to site and construct the encapsulation plant for spent nuclear fuel and the deep repository require supportive documentation of the environmental consequences including assessments of both the operating and long term safety of the activities. The coming safety assessments are listed below.

Since a number of consecutive safety evaluations have to be produced in the process of implementing final disposal, it was found expedient to develop a standard template for how to report these assessments of long term safety. The suggested template is presented in a report called SR-95 /15-3/ and a short overview is given in section 15.3. In SR-95 an overview is also given of the repository system given highest priority today, the state of the art of post closure safety assessments is discussed in detail, and the available numerical tools are presented. The template is presently under authority review.

Chapter 15 also gives an overview of recent developments in SKB with regard to scenario methodology (15.4), coupling of numerical models (15.5) and modelling of transport in the geosphere (15.6).

15.2 PRINCIPLES FOR RADIATION PROTECTION AND SAFETY

15.2.1 General

Radioactive waste must be handled in keeping with established principles for protection against ionizing radiation /15-4, 5/.

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

- 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 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 /15-6/:

- 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 /15-7/ 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.

15.2.2 Safety functions of the repository

To achieve the desired safety during the construction of a deep repository, during the operating phase and during 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 15-1 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 multibarrier principle, i.e. safety must not be solely dependent on the

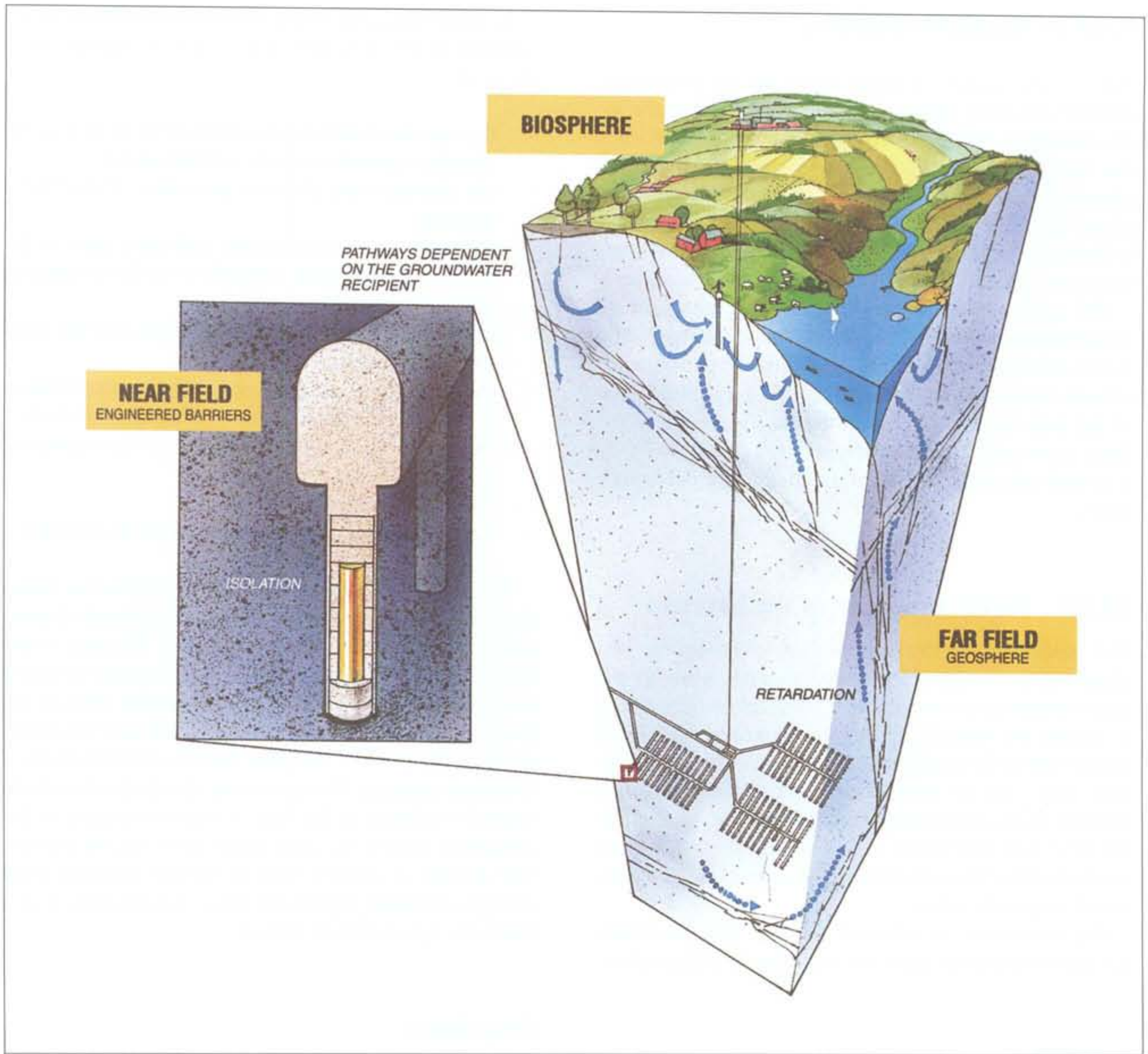

Figure 15-1. Parts of the deep disposal system and their most important safety functions.

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.

15.2.3 Safety functions of the barriers

The guidelines for final disposal of spent nuclear fuel discussed in Sweden /15-6, 7/ 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.

15.3 A TEMPLATE FOR REPORTING LONG-TERM SAFETY - SR-95

Background and intentions

Safety reports serve as a basis for a series of decision steps in the implementation of the deep disposal scheme. The various reports must therefore have a fundamental continuity and uniformity. The reports can, however, present safety judgements based on different underlying facts, and the decisions are sometimes of differing character. This may lead to differences in how the assessments are carried out and how the safety evaluation is reported. For example, the safety report for SR-I will have to examine the safety-related importance of the variation in site characteristics for different sites in Sweden, and the report for SR-D will need to clarify any safety differences between the two sites that have been characterized by means of site investigations. In the subsequent reports, the safety-related importance of a progressively expanding body of geoscientific background data will have to be demonstrated.

Safety reports must always satisfy three fundamental requirements:

- The purpose of the decision that is to be based on the safety report must be clearly defined, and the executed safety evaluation must be relevant to this purpose.
- Assumptions, analyses and results (incl. uncertainties) must be reported to such an extent and in such a way that an independent review can be carried out.
- It must be demonstrated that safety is adequate in relation to given acceptance criteria or that the safety potential is sufficient to permit a transition to the next development phase.

In order to create a continuity in safety reporting and a uniformity between reports, a template has been developed for giving an account of long-term safety, SR-95 /15-3/. The template is also intended to facilitate the preparation and review of safety reports, and to simplify comparisons between how, for example, an expanded body of data influences the assessment of safety potential and uncertainty.

To clarify the structure of the template, a proposed synopsis is exemplified with illustrative text describing the underlying premises and the status of the assessment capability today. One chapter discusses the features and processes important for the primary safety function of the repository, to isolate the waste from the ground water. Another discusses the processes important for the secondary safety function – to limit the solubility of the radionuclides and their transport through the surrounding geosphere in case the isolation is imperfect. The methods and modelling tools that SKB has at its disposal for the upcoming safety assessments are also presented in SR-95.

Structure of the template

Each safety report is organized in main parts as follows:

- Premises and purpose.
- Description of the deep repository system.
- Evolution of the repository system with time.
- Evaluation and conclusions.

A basis for the subdivision of the main parts into chapters is to facilitate the "reuse" of chapters that don't change from one safety report to the next, and to attempt to concentrate the needs for changes to a few chapters when there have been changes in underlying premises, methodology or design specifications.

– Premises and purposes

This main part should clarify what phase of development the work with the repository system is currently in and the specific decision for which the safety report is supposed to provide support. The overall safety goals and acceptance criteria for the activities, as well as any stage requirements, should be set forth.

A retrospect should be provided with reference to previously conducted performance and safety assessments of importance. An account of the safety report´s scope or specific premises and delimitations should be given, along with a description of the assessment methodology employed, particularly if any special simplifications or unfavourable conditions have been introduced.

– System description

The system description includes chapters for

- quantities and characteristics of the radioactive waste,
- design, general layout and dimensions of the deep repository,
- materials and dimensions for the engineered barriers around the waste, and
- site-specific conditions with regard to the geoscientific characteristics of the facility site and the surrounding biosphere.

In addition, there is a chapter describing how the layout and other design parameters of the facility have been adapted to exploit the safety potential of the site. This chapter is separated from the general design and layout, since it will probably be necessary to make progressive site-specific adaptations during both the site investigation and construction phase in order to be able to effectively exploit the potential of the site for good safety.

The safety account is based on the specific system defined by the system description. The report will not describe work carried out in previous stages in order to optimize the design and dimensions of the repository via performance assessments, sensitivity and variation studies and safety assessments.

– Analysis of the evolution of the repository system

This main part provides a detailed description of the methodology employed to assess the safety of the repository, the calculations made and the models and data used for this purpose. The division is divided into chapter-by-chapter accounts of:

- general review of the evolution of the repository system – choice of scenarios,
- analysis of the intended performance of the repository and its limitations,
- analysis of radionuclide transport,
- detailed qualitative and quantitative assessment of selected scenarios.

The first three chapters also contain accounts of methods and tools that are used in the analyses. The account of the analyses that clarify the normal performance of the repository, to isolate and contain the waste, is separated from the account of the analysis of radionuclide solubility and transport in the event of defective canisters. This is done in order to clarify the differences in how the primary safety function (isolation) and the secondary function (low solubility and retardation) are achieved.

It is possible that the amount of data used for certain safety assessments can be so extensive that it should be broken out to a separate chapter to increase readability. This chapter should then include both measurement data, including uncertainties, and methods for translating them to parameters for the safety assessment.

– *Evaluation and conclusions*

The results of quantitative and qualitative assessments of the performance of the repository must be evaluated with regard to relevance and uncertainty and be compared with safety goals and acceptance criteria. The material must also be judged with respect to the decision for which the safety report constitutes supporting material. This material and the summarizing conclusions of the safety judgements is to be presented in the last two chapters of the report.

15.4 SCENARIO METHODOLOGY

The deep repository consists of a chain of barriers – engineered, i.e. man-made, and natural. The main function of the barriers is to isolate the radioactive waste. If the isolation is broken the repository system dilute dissolved radionuclides to acceptable levels via retardation and dispersion.

The purpose of the safety assessment is to determine whether the repository system functions and meets stipulated criteria for all realistic states of the system. This requires tools/models to simulate and analyze the behaviour of the repository. An important part of the safety assessment is to compile information on the characteristics and performance of the repository and to identify the system states that should be simulated and analyzed. The methodology used for this purpose is called scenario methodology.

A scenario is a description of a hypothetical future situation. The concept of a scenario includes both a course of events emanating from a set of specified premises and the future situation this course of events leads to.

The scenarios chosen for analysis in a performance assessment should together provide a reasonably comprehensive picture of the possible evolutionary pathways of the system. Unrealistic scenarios such as the exclusion of a barrier or process can be also be included. This kind of scenarios are often called "What-If", "Worst-Case" or "Bounding" scenarios. The purpose of them is to demonstrate the robustness of the system or its sensitivity to uncertainties in barrier performance and processes. The choice of scenarios should be made on the basis of a systematic description of the system and its evolution with time. An important part of the scenario methodology is to offer a procedure for developing such a systematic description. The scenario methodology also consist of a documentation plan, and a description of how expert judgement is used and handled.

Scenario methodology is still under development. So far development has taken place by trying out a number of methods. During the course of the work, shortcomings and merits have been identified and lessons have been learned from the experience. A need to portray the repository system, its features and processes in a schematic and illustrative fashion and to link them together in cause/effect relationships was identified early. Different types of diagrams were tried for this purpose and the interaction matrices of the RES (Rock Engineering System) method were considered to be best suited to the purpose /15-8/.

In an interaction matrix the main concepts or variables governing the system are identified and listed along the leading diagonal of a matrix. Then, in the off-diagonal terms of the matrix, all the interactions are considered. During 1995 a methodology for identification, structuring and ranking of features, events and processes using interaction matrices has been developed and applied /15-9/. SKB's scenario methodology as it has been developed up to today includes the following steps:

1. Use interaction matrices to identify, structure and rank features, events and processes that can influence the performance of a repository system.
2. A method for documentation of the interaction matrices.
3. A plan for how expert opinions are to be handled in connection with the documentation of the interaction matrices.
4. Selection of premises for scenarios.
5. Description and justification of scenarios.
6. Qualitative analysis of the scenarios using the interaction matrices.
7. Documentation of the scenario specific interaction matrices.

Construction of the interaction matrices has two main purposes:

- to show that no features, events or processes have been overlooked, and
- to identify important interactions and processes.

Furthermore, the process of constructing the interaction matrices is considered to lead to a better understanding of the system and to facilitate communication between different experts.

During 1995 effort has been laid on issue 1-3 of the scenario methodology. Methods to carry out issue 4-7 are still under development.

15.5 MONITOR 2000

During 1995, a new user interface for complex model-chain calculations of radionuclide transport has been developed. The aim has been to improve SKB's model calculations capacity in safety assessments.

Calculations of radionuclide transport from the repository are performed, in SKB's safety assessments, as complex chains of inter-connected mathematical models. A typical model chain may consist of several hundred connected submodels describing transport in different parts of the engineered and natural barriers.

The calculations are administered by a control program called PROPER. PROPER is able to handle also probabilistic executions of model chains, whereby uncertainties in model input are propagated to the calculation results.

During 1995, a new user interface for PROPER, called Monitor 2000, has been developed. Monitor 2000 is a graphic interface which will replace the former text-based interface. The introduction of Monitor 2000 will considerably reduce the operator time required to set-up and further handle complex model calculations. Furthermore, the quality assurance of the calculations will be improved through built-in controls in the new interface. Comprehensive reports of complex calculations will be automatically generated, improving the traceability of performed calculations.

Monitor 2000 will be delivered in the beginning of 1996 and will be used for transport calculations in SKB's coming safety assessments.

15.6 MODELLING OF TRANSPORT IN THE FAR FIELD

Background

Assessment modelling of groundwater flow in fractured, low-permeable rock is complex since the flow is concentrated within fractures. Still, calculations of water movement and transport of radionuclides constitute an important part of the safety analysis. Conceptual, mathematical and numerical models need to be further refined and tested. It is also necessary to show the impact of different conceptual models on the simulated results.

Model development related to radionuclide transport is included in the safety analysis programme. However, complementary activities may be found in the Geoscience programme as well as in the Chemistry programme.

Verification and testing of models are important. Therefore, the modelling of groundwater flow and transport of solutes within the Äspö Hard Rock Laboratory project is of special interest. A Modelling Task Force was initiated in 1992 as a part of the international cooperation at Äspö HRL. The work will increase our understanding of the fundamental processes involved as well as give an opportunity to test our models with site-specific data.

In SR-95 /15-3/ the methodology for far field transport modelling is presented in detail as well as the available assessment models.

Development of models

Interaction matrices are used to give a systematic description of the process system for the deep geological repository according to the KBS-3 method. An interaction matrix for the far field has been produced and documented during 1995 /15-9/.

The *channel network* approach describes the flow paths in fractured rock as a network of connected channels with different lengths, conductances, volumes and widths. The model can simulate the transport of a solute through this network where the solute may diffuse into the rock matrix. It is the water-conductive channels and not the fractures in the rock that constitute the basis for the model. A User's Guide and model description has been produced.

The simulations of the Äspö LPT2 experiment and the data requirements for the Channel network model were summarized in a Licentiate treatise by Björn Gylling /15-10/. In addition some calculations using the concept of self-similarity to improve dispersion properties of the CHAN3D were presented.

The *stochastic continuum* model replaces the network of fractures by a continuum, with hydraulic conductivity adjusted to obtain a volume-averaged flow which correlates well with observations. This model is in wide use and a solid body of experience and know-how has been built around it. The HYDRASTAR/Inferens computer programs use the stochastic continuum model to perform two tasks:

- a geostatistical synthesis of a conductivity field from packer tests,
- the groundwater flow simulations and particle trackings which use the conductivity field.

The SKB HYDRASTAR/Inferens simulation model uses geostatistical techniques for extrapolating packer tests, interpreted as point conductivity measurements, to the 3-D computational continuum. The current work aims at adding other inverse modelling features to make use of heads and flows recorded in long-term and interference pumping tests /15-11/.

The final model will obviously be very demanding in terms of computer resources. Therefore, the feasibility of a 3D inverse modelling project based on HYDRASTAR/Inferens must be assessed with respect to gain in prediction capabilities, development costs, computational resources, etc.

A validity document for the groundwater flow and transport model NAMMU has been produced. This was the first in a series for the SKB assessment models.

A study of colloids and their importance for the performance assessments has been published /15-12/.

Application of models

The modelling of the forthcoming Äspö TRUE experiments has been performed on a conceptual basis. A case study was performed to illustrate transport of non-interacting and interacting solutes. The channel network model was also used to model a more complicated geometry

/15-13/. The importance of the flow rate distribution and the flow wetted surface was emphasized.

The work with coupling the near field model NUC-TRAN to the far field model CHAN3D was started during 1995. A set of illustrating calculations will use Äspö data as in the case of SR-95.

Numerical groundwater flow modelling based on the continuum assumption requires that results from hydraulic field tests are adjusted to the chosen calculation scale. A project has been performed with the overall objective to develop and evaluate a new methodology for assigning hydraulic conductivity values for rock blocks in different scales /15-14/. Hydraulic packer tests at Äspö have been utilized in a somewhat new way since the pressure responses as a function of time have been used. Based on this information, the number of geologically and geometrically acceptable discrete fracture network blocks have been reduced based on this hydraulic information. Hereby, for different calculation scales a number of possible discrete fracture networks models are obtained which all

- are conditioned on geological and geometrical data, and at the same time
- reproduce the real hydraulic test responses.

16 SUPPORTING RESEARCH AND DEVELOPMENT

This chapter presents the activities for the general development of understanding and databases in areas important for repository safety. Also presented are specific research actions that have been initiated to clarify unresolved issues. The chapter is divided into sections for Spent Fuel, Buffer and Backfill, Geoscience, Chemistry, Natural Analogue Studies, and The Biosphere.

The technical development and research on the Copper-Steel Canister is presented in Chapter 13.

Activities related to the development of methodology and tools for performance and safety assessments are discussed in Chapter 15. Investigations regarding long-lived radioactive wastes other than spent nuclear fuel are presented in Chapter 17. All the in-situ experiments and sampling made in the HRL at Äspö are reported in Chapter 18.

16.1 SPENT NUCLEAR FUEL

The results accumulated in the fuel experimental programme in the hot cells at Studsvik Nuclear have given a good understanding of the corrosion of spent nuclear fuel and a review of the status of the Swedish program has been published /16.1-1/. The cooperation with other groups in the world performing similar studies has continued during 1995, through the spent fuel workshop that was held in Überlingen, Germany, and was arranged by the Forschungszentrum Karlsruhe, Institute for the Waste Management Technology.

16.1.1 Fuel characterization

A general description of the properties of spent fuel which are considered important from the point of view of fuel corrosion and which summarizes the work done on fuel characterization during recent years has been published /16.1-2/. The surface area of the spent nuclear fuel available for the corrosive attack by ingressed groundwater is an essential parameter for determining the absolute corrosion rates. Pellet cracking and fragment size distribution have been discussed in relation to the surface area. Recent results on the specific surface area determinations of two reference fuels used in the SKB's experimental program have been obtained using the BET method with Kr sorption. Reproducible values in the range 70-120 cm^2/g have been obtained. These values are larger than the calculated geometrical surface area and estimates of the average Kr penetration depth are conceptually acceptable, showing that the obtained results may be used in corrosion rate calculations. On the other hand the attempts to relate the surface area to fuel burnup are inconclusive.

16.1.2 Corrosion of high level spent nuclear fuel

The corrosion test series using the sequential corrosion scheme with replacement of the leach solution with fresh solution are in progress at Studsvik Nuclear hot cells since 1982. The analyses of the leach solution are performed directly without separations or isotope dilution via mass spectrometry with an inductively coupled plasma source instrument (ICP-MS), in operation since 1992. Considerable effort has been devoted to the improvement of the analytical procedure /16.1-3/ and actually multi elemental analysis with this technique has substituted most of the other methods. An advantage of the ICP-MS method is that low concentrations of other fission products, like the trivalent lanthanides, can be measured. Since their solubilities and speciation in conditions relevant to a deep repository were not available, thermodynamic data for the aqueous species of the Rare Earth Elements (REE) with the most important ligands relevant for granitic groundwater conditions have been selected and validated /16.1-4/. The study indicates that at the initial stages of fuel dissolution, the concentrations of the REE will be determined by the $UO_2(s)$ matrix dissolution. Later on solid phosphates, hydroxycarbonates and carbonates may limit their solubility. A typical predominance diagram for europium in repository conditions calculated with the selected constants is shown in Figure 16.1-1.

The results from spent fuel corrosion tests performed under oxic conditions indicate that the ^{99}Tc release occurs by the preferential dissolution of metallic particles containing Mo, Tc, Ru, Rh and Pd. The size and composition of these metallic particles depends on the burnup and temperature regime in the fuel during irradiation. For this purpose experiments were conducted with dissolution of specimens of reference fuels in concentrated nitric acid and separation of a heavy residue consisting predominantly of these metallic particles. The composition of the residue has been estimated on the basis of examination by SEM (Scanning Electron Microscopy) and EPMA (Electron Probe MicroAnalysis) together with results from the analysis of the filtrates and ORIGEN calculations of the inventories /16.1-5/. In dissolution tests performed on two residue specimens with simulated granitic groundwater under oxic conditions, the measured concentrations of Mo and Tc were of the same order of magnitude as those obtained in corrosion tests with spent fuel.

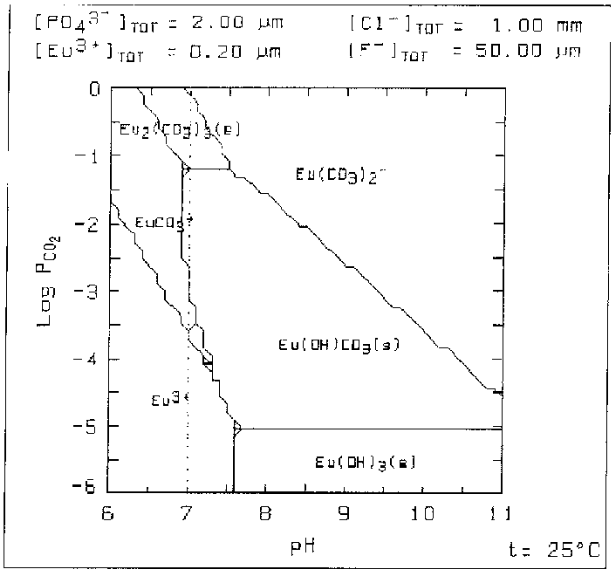

Figure 16.1-1. *Predominance area diagram for the* Eu^{3+}*-*$CO_2(g)$*-*H_2O *system.*

A combined experiment was started in 1985 to study both spent fuel corrosion and radionuclide diffusion in bentonite clay at simulated repository conditions. Pieces of a spent fuel pin, 42 MWd/kgU, together with the Zirkaloy cladding were placed in diffusion cells surrounded with bentonite clay (Wyoming MX-80) of dry density 2100 kg/m^3. The diffusion cells were located in cylinders containing a low saline (ionic strength 0.0085 M, pH = 8.0–8.2) carbonate groundwater that was previously equilibrated with bentonite clay. A total of ten diffusion cells were used. In some of the cells, additives of metallic iron, metallic copper or the Fe(II) mineral vivianite were mixed with the clay. Experiments with total contact times of 101, 197, 386 and 2213 days were performed. The experimental setup and some results have been reported in earlier publications /16.1-6 to -11/. So far the release and the apparent diffusivity of the following elements have been reported: cesium (^{134,137}Cs), strontium (^{90}Sr), technetium (^{99}Tc) and cobalt (^{60}Co).

The extremely low concentrations of the elements that are of most interest in the long term safety perspective have made it necessary to develop suitable analytical procedures. A special procedure to analyze ^{99}Tc using the ICP-MS technique has been developed /16.1-10/ and a radioanalytical procedure to analyse ^{90}Sr in low concentrations is presented /16.1-9/.

The results obtained with ICP-MS show that the analysis of very long-lived radionuclides is possible to achieve at very low detection limits, in some cases several orders of magnitude lower than with radioanalytical procedures. Further improvement of the detection limits of ICP-MS using a desolvating nebulizer shows that in this way the detection limit for most elements may be decreased by about one order of magnitude. There are, however, some elements for which the detection limits are not improved to the same extent (e.g. ^{99}Tc). Another drawback using the desolvating nebulizer is the significant influence of the sample matrix. High concentrations of acid or saline solutions will significantly decrease the sensitivity using the nebulizer.

However, it has been shown that radioanalytical procedures for some of the trivalent actinides (americium and curium, ^{241}Am, ^{244}Cm) probably can reach lower detection limits than with ICP-MS. Besides, radioanalytical analysis of the short-lived plutonium-isotope (^{238}Pu) may be used to analyze low concentrations of plutonium under the circumstance that the isotopic composition of plutonium in spent fuel is well known.

Some preliminary results for other elements, not reported earlier, have recently been presented /16.1-11/. The data on the leaching and diffusion of neptunium (^{237}Np) are shown in Figure 16.1-2. From these results an apparent diffusivity of 10^{-14} m^2/s has been determined. Due to the very low concentration of ^{237}Np, these results should be regarded as preliminary and further improvement of the analysis with the desolvating nebulizer is in progress.

The results of the leaching of uranium from the spent fuel and its transport through compacted bentonite are shown in Figure 16.1-3. The analysis of uranium is based on the uranium isotope ^{236}U, since the concentration of the other uranium isotopes ^{235,238}U is obscured by the natural concentration of uranium in the bentonite clay (about 8 ppm). Based on a calculated isotopic composition of the uranium in the spent fuel, a total released fraction of about 10^{-5} after one year was estimated. This value will,

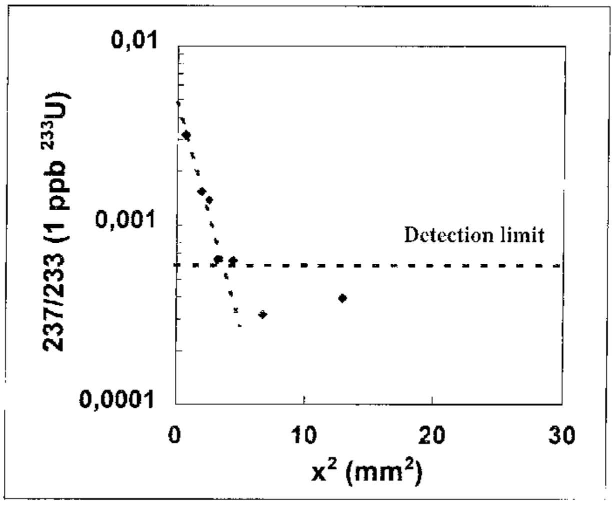

Figure 16.1-2. *Preliminary results of the diffusion of neptunium (*^{237}*Np) from spent nuclear fuel in contact with compacted bentonite.*

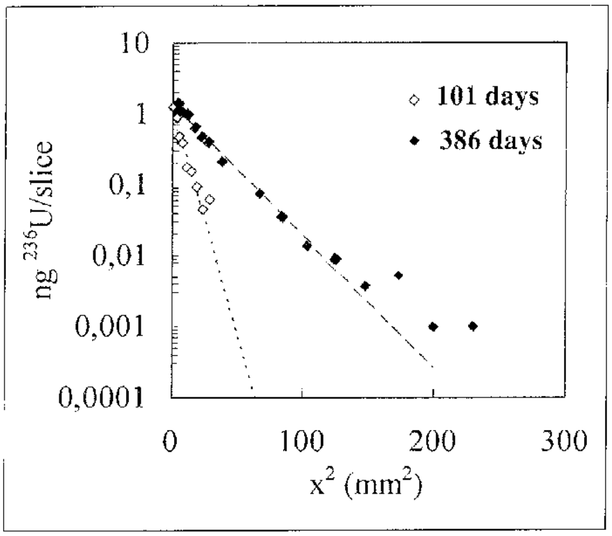

Figure 16.1-3. Preliminary results of the diffusion of uranium (^{236}U) from spent nuclear fuel in contact with bentonite clay.

however, later be corrected with the actual uranium isotopic composition of the fuel used in this experiment. The observed apparent diffusivity of uranium has been determined as 10^{-13} m^2/s.

16.1.3 Alpha radiolysis

The radiolysis of water produces equivalent amounts of oxidizing and reducing species. The consumption of oxidants by interaction with the spent fuel matrix results in a net production of reducing species, primarily as H_2. A mass balance study for radiolytically produced oxidants,

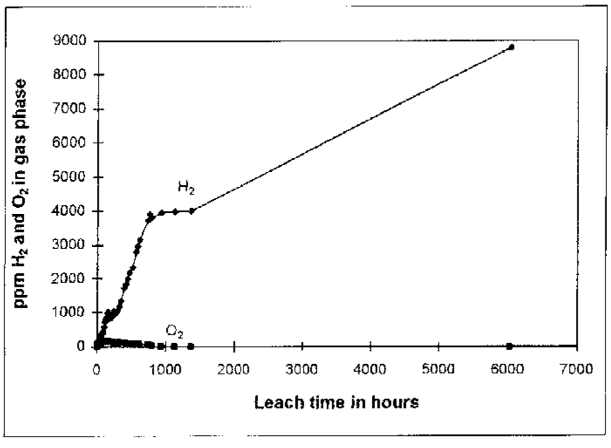

Figure 16.1-4. Oxygen and hydrogen concentrations in the gas phase plotted versus leach time (1.88 g fuel, 14 cm^3 distilled water, 50 cm^3 Ar).

reductants and dissolved uranium has been performed in a closed system initially containing fragments of used fuel and oxygen free distilled water /16.1-12/.

As discussed previously /16.1-13/ a clear deficiency of oxidants was found in the overall redox system.

Long term experiments have confirmed these results. The O_2 concentration increased to a maximum value within about one week, whereafter it decreased with time, see Figure 16.1-4. The hydrogen concentration was found to increase continuously during a more than eight months period, while the concentrations of oxygen remain very low. Additional experiments with controlled chemical conditions and overlapping longer time intervals are in progress. A more realistic modelling of the radiolytic oxidation of spent fuel may be derived from including these data in the kinetic model presently under development.

16.1.4 Models

Spent fuel is disposed of under chemical conditions normally described by two intensive parameters, pH and E_h. The evolution of these parameters is largely dependent on the respective buffer capacities, i. e. alkalinity (acid neutralizing capacity) and reductive capacity (RDC, oxidant neutralizing capacity). The spent fuel matrix constitutes a moderate pool of alkalinity, but a larger pool of redox buffering capacity. The experimental data on dissolution of spent fuel and uranium dioxide under oxidative conditions suggest a kinetic rather than thermodynamic control of the process. A kinetic model accounting for the redox capacity of the spent fuel matrix has been under development recently /16.1-14/.

The global rate of dissolution of the spent fuel matrix is expressed in terms of the following processes:

- Initial fast dissolution of an already oxidized UO_{2+x} surface layer. The rate of this initial dissolution has been determined experimentally /16.1-15/.
- Intrinsic oxidative dissolution of the fuel matrix in bicarbonate solutions. The rate of this process depends on the concentration of the radiolytically produced oxidants and on the bicarbonate concentration.
- Precipitation of secondary phases. The rate of this process depends on the U(VI) concentration, surface of fuel to volume of solution ratio and on the groundwater composition.

Preliminary calculations have indicated that it is possible to develop a quantitative kinetic model which accounts for all the above processes. The basics of the model has been included in the EQ 3/6 code package making possible the successful modelling of early release data from fuel leaching experiments. Experimental and modelling efforts are in progress to further improve this kinetic model.

16.1.5 Natural analogues

The experimental and modelling studies in uranium minerals as natural analogues for the stability of the $UO_2(s)$ spent fuel matrix may be used to make estimations on leaching of spent nuclear fuel, taking into account both its analogies and differences with uraninite.

The thermodynamic and kinetic information collected has been summarised in a unified model which reconciles dissolution data for uraninites from different locations. This model has proven that the original redox state of the UO_{2+x} phase dominates the redox evolution of the contacting water and therefore the solubility of the uranium oxide phase /16.1-16/. This effect has been observed also in the fuel leaching experiments performed in Studsvik and it has been rationalised in terms of the reducing capacity (RDC) of the uranium oxide matrix.

The long term stability of spent nuclear fuel is strongly related to the pathways of the oxidative alteration of the uranium matrix. Thermodynamic studies on secondary alteration products have resulted in a value for the solubility product of becquerelite /16.1-17/ of log Kso = 30.83±0.05. This indicates that becquerelite has a larger stability field than previously estimated, in agreement with the observations from drop-test data obtained by Wronckiewicz et al. /16.1-18/. Recent mineralogical data indicate that these intermediate U(VI)-oxide phases are able to incorporate trace elements in their structure. This could result in a beneficial effect on the release of radionuclides through radiolytic alteration of the fuel matrix, taking into account the increased stability of becquerelite.

The stability of U(VI) silicate phases soddyite and uranophane as end products for the oxidative alteration of the spent fuel matrix has also been determined experimentally. In order to avoid some of the experimental uncertainties observed with natural samples, synthetic phases prepared according to previously established procedures have been used. This has resulted on an estimation of the stability of soddyite, in agreement with previous determinations from Nguyen et al. /16.1-19/ of log Kso = 5.74 ± 0.21. The determination of the equilibrium solubility of uranophane is more problematic. Even with synthetic phases and using solution compositions, which would guarantee the thermodynamic stability of the solid phase, it was difficult to obtain reproducible solubilities. However, by using upper stability limits for this phase, equilibrium calculations indicate that the favoured alteration phases are becquerelite and uranophane. This last phase is the most stable end product of the oxidative alteration of the spent fuel matrix. Anyhow, the formation of both schoepite and to a lesser extent soddyite has been documented in some leaching experiments as well as in studies of natural system. This would indicate that kinetic constraints due to preferential formation and dissolution controlled by structural patterns of the solid phase play a major role in the oxidative alteration pathways of the spent fuel matrix.

16.2 BUFFER AND BACKFILL

16.2.1 Modelling of THM behaviour of buffer materials

Two reports that deal with modelling of thermo-hydro-mechanical processes in buffer materials have been published during 1995. They concern the present state of knowledge at water-saturated and unsaturated conditions and represent different levels of knowledge.

Water-unsaturated buffer materials

This report /16.2-1/ contains a description of a preliminary material model for unsaturated buffer materials. The model is based on:

- General knowledge of unsaturated soils. There is today a good basis for understanding and modelling the behaviour although there is still a lack of some knowledge.
- Results from laboratory tests of important processes. A large number of fairly simple but important laboratory tests have been performed on sodium bentonite MX-80, which is the reference buffer material. These tests were necessary for supporting the proposed model and for calibrating parameters for the model.
- THM model for water saturated buffer material, with a natural transition when the material is transferred from an unsaturated to a saturated state. The model at saturation hence represents a special case of the more general model for the unsaturated state.
- Finite element code ABAQUS with its wide range of possibilities. Some changes and improvements have been made, especially concerning temperature induced water vapour flow, which is a process that has been implemented and added to the code.
- Some calibration and validation calculations, which are shown in the report.

The most important conclusions are that the preliminary model works and is useful in some cases, but also that earlier indications remain that the model does not apply with accuracy to all cases, particularly not to mechanical regimes.

Water-saturated buffer materials

This report /16.2-2/ describes a thermo-hydro-mechanical model of water saturated buffer materials and how it can be used for calculation of THM processes with the finite element code ABAQUS.

Laboratory tests have been performed on samples with different bentonite composition at different void ratios and temperatures and with different pore water composition. The following tests are accounted for in the report:

- Triaxial tests.
- Swelling/compression tests.
- Swelling pressure tests.
- Hydraulic conductivity tests.
- Creep tests.

These tests and several other test types shown in other reports, as shear tests, thermal conductivity tests, and thermal expansion tests, have all been used for the material model. E.g. the triaxial tests have been used for evaluating the stress-strain-strength behaviour at a change in deviatoric stress and the swelling/compression tests have been used for evaluating the volume change caused by a change in average stress.

The material model, which consists of many submodels, can be considered to exist in two variants. One is a so called tentative model which describes a defined process in a preliminary way with e.g. a formula. The tentative model is called the CLAYTECH/S/T model where S stands for saturated and T stands for tentative. The other variant is a model completed and adjusted to or implemented for the finite element code ABAQUS. This model is called the CLAYTECH/S/A where A stands for ABAQUS. The report describes both these models. They are very similar with a few exceptions, the submodel for volumetric creep being the most important one.

The hydraulic and thermal submodels are available in the standard version of ABAQUS, while a new mechanical model has been developed and implemented for the code. The mechanical model consists of the following main components:

- A curved critical state line (CSL).
- A curved failure envelope.
- A cap that defines the limit between elastic and plastic volumetric strain.
- A porous elastic region with a variable Poisson's ratio.
- A plastic region with contractancy on the cap and dilatancy outside the CSL.

The model is based on the effective stress theory and Darcy´s law with a variable hydraulic conductivity. The heat flow is modelled as resulting from thermal conduction with a variable heat conductivity. The thermo-mechanical response is linked to the thermal expansion of the water and particles.

The mechanical model has been checked with some calculations, which simulate laboratory tests that have been carried out. Two different types of swelling/compression tests and two types of triaxial tests have been simulated and compared to measured results. These calculations show that the model generally can be used for mechanical calculations of the behaviour of water saturated buffer materials, but also that there is still a lack of some understanding. It is concluded that the available model is relevant for the required predictions of the THM behaviour.

16.2.2 Cement pore-water impact on bentonite

The pore-water of fresh unaltered cement is basically a strong alkali hydroxide solution saturated with calcium hydroxide. In aged cement the easily dissolved alkali hydroxides will be removed from the system and the pH will drop to a value of about 12. Laboratory test have been performed in order to determine the impact of high pH cement pore-water on compacted bentonite. Two types of artificial cement pore water solutions were used, one was dominated by alkali hydroxides, which lead to a very high pH (13.5), and the other was dominated by calcium hydroxide and therefore with a more moderate pH (12.5). The investigation comprised 1, 4 and 16 months series with hydrothermal cell tests, percolation tests and diffusion tests. The swelling pressure and the hydraulic conductivity were measured during the whole test period in the percolation tests. After termination, the clay was analyzed with respect to element distribution, mineralogy and physical properties. The water solutions were analyzed with respect to ion content and pH. The following observations were made

- ion exchange in the bentonite to cement pore-water ions,
- increase in the bentonite cation exchange capacity,
- dissolution of cristobalite in the bentonite,
- minor formation of illite in the bentonite,
- minor formation of chlorite in the bentonite,
- formation of CSH (II) in the bentonite,
- wash away of CSH-gel into surrounding water.

The major effect on the bentonite was the replacement of the original charge balancing cation by cement pore-water cations. This cation exchange reaction is well known and not considered as a major problem in a buffer with highly compacted bentonite, since the buffer functions are similar for bentonite saturated with the involved cations if the buffer density is above 2.0 g/cm^3. The mineralogical alteration has in general only concerned a minor part of the bentonite mass and has not affected the buffer functions in a significant way. The analyses have demonstrated that the rate of illite and chlorite formation have been retarding with time. The identification of CSH (calcium silicate hydrate) substances is important since it explains the lack of expected zeolites, see Figure 16.2-1. This finding will improve the modelling of the system.

16.2.3 Precompaction of bentonite blocks

Blocks of highly compacted bentonite for buffer materials should either be made in a size that is small enough to be manageable by hand or as large as possible for machine handling. During 1995 the testing of technique for pro-

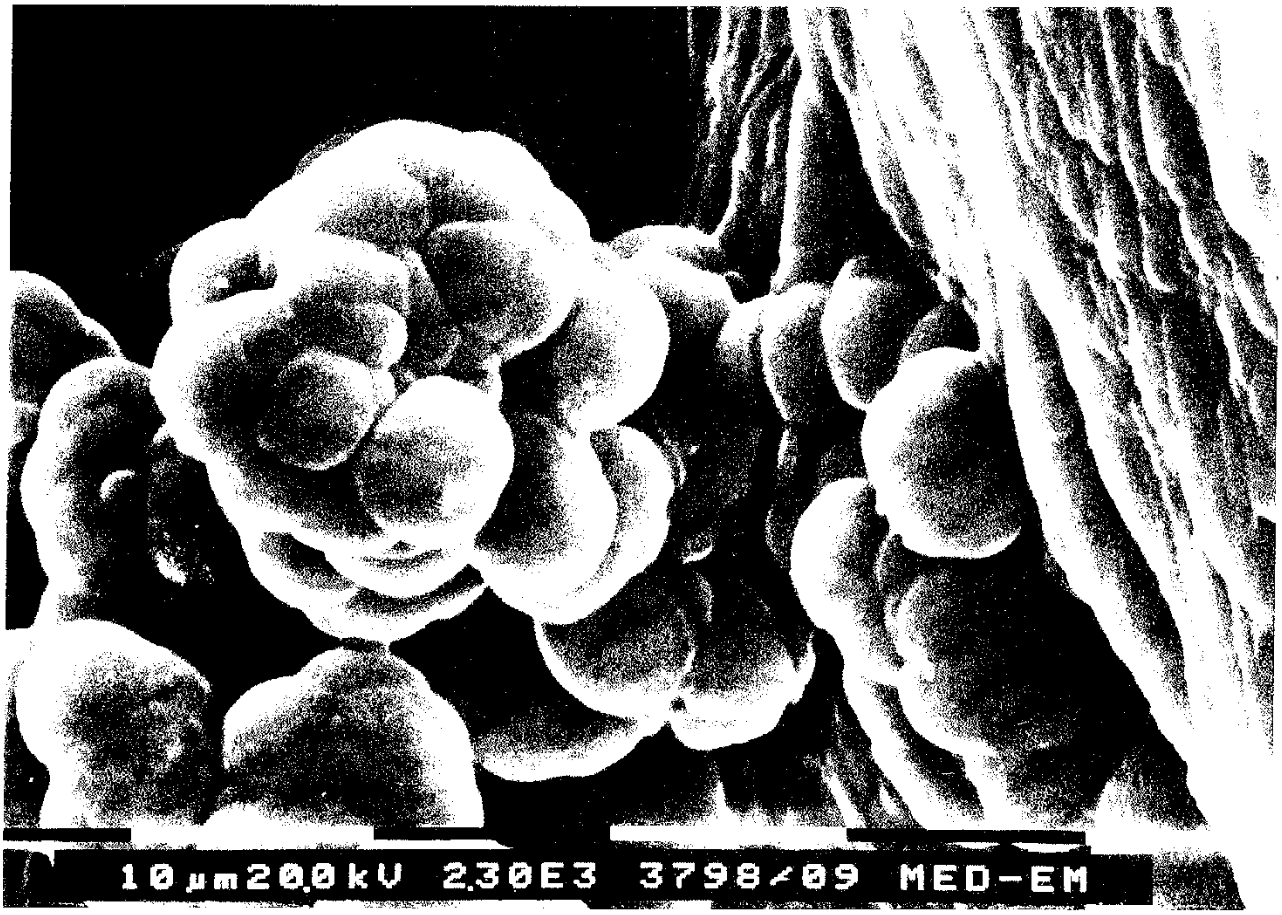

Figure 16.2-1. CSH substances from the 16 month percolation test with a solution pH of 13.5. The material is concentrated by evaporation of the bentonite percolating water, and examined by use of scanning electron microscopy. The silica-calcium ratio in this sample is around 3. The magnification is 2300 times.

duction of small blocks has been finished and test compaction of very large blocks has started.

The tests of compaction of small blocks, which were made in the brick factory of Höganäs Bjuf AB in Bjuv in southern Sweden, have resulted in a proposal for a technique for industrial production of small blocks. In order to minimise the air entrapped in the blocks during compaction several measures need to be taken in order to achieve good quality of the blocks.

- The raw material should consist of fairly coarse granules.
- The compaction should be made in several steps.
- The gaps between the piston and the form in the press should not be too narrow.
- The ratio of the height to the length should not exceed 0.4.

To prevent damages on the blocks due to friction between the bentonite and the form, the form should be lubricated with oil before the bentonite is poured into it. The lubricating oil also prevents damages on the blocks due to sticking of bentonite to the form and pistons.

In order to prevent damages caused by the expansion during removal of the block from the form, the block should be compacted to a high degree of saturation. In the case when a compaction pressure of 100 MPa is used the water ratio of the bentonite should not be below 18% (water ratio is weight of water divided by weight of solid material). The damage can also be prevented by making the form slightly conical.

The blocks must be wrapped in plastic sheeting after compaction in order to prevent desiccation. The water ratio of the bentonite should not exceed 20% in order to prevent the blocks from going mouldy.

Two production series were made in Bjuv in order to test and demonstrate the capacity for producing large amounts of blocks in an industrial way. In the first series 70 blocks were compacted from coarsely ground bentonite with water added so that the water ratio became 18%. The second series comprised production of 70 blocks of 30 weight-% MX-80 bentonite and 70 weight-% crushed rock for backfilling. The entire production line was simulated from the bentonite/ballast/water mixing procedure to the plastic wrapping of the block piles. The quality of the blocks was very high with the small exception of the edges which had a tendency to crumble after some time of storage.

During 1995 a large form for compacting blocks with a diameter of 1.0 meter and a height of 0.4 meter was constructed and tested. Two test blocks were made by compaction to 100 MPa in a large press in Ystad. These tests will continue during 1996.

16.2.4 Backfilling of tunnels by use of crushed rock as ballast

The concequences of replacing sand material of quartz type as ballast by chrushed rock in tunnel backfills have been investigated /16.2-3/. A basic question has been whether such replacement alters the hydraulic conductivity and compressibility as well as expandability, and also if the physical and chemical stabilities are altered.

The key factor is the microstructural constitution of the bentonite/ballast mixtures, which is primarily controlled by the grain size distribution of the ballast. It is preferably of Fuller's type with a low D_5 value, i.e. at least 5 weight-% finer than 0.05 mm, in order to get both high densities and low conductivities. A number of examples show that the gradation of suitably composed artificial mixtures of glacial material can in fact be obtained by crushing crystalline rock without any processing.

The compactability of backfills with quartz sand is higher than that of backfills with crushed rock as ballast. This means that for a given compaction energy, the density will be lower for crushed rock based backfills. This means that the hydraulic conductivity and compressibility of this sort of backfills will be somewhat higher than of quartz sand based backfills and that their expandability will be somewhat lower. Furthermore, the physical stability of crushed rock based backfills in terms of piping and erosion resistance will be somewhat lower than that of quartz sand based backfills. The chemical stability, which is determined by the conversion of the smectite component in the bentonite to illite, is practically independent of whether the ballast is pure quartz or rock with potassium bearing minerals because the temperature in the backfill will be too low to yield significant conversion of smectite to illite in the heating period.

In order to reach the same densities of quartz sand based and crushed rock based backfills, which turn out to give fairly similar physical properties, the latter backfills need more effective compaction or, alternatively, a higher bentonite content. Laboratory tests indicate that if the bentonite content in crushed rock based backfills is not increased but the density is enhanced, these backfills will serve equally well as quartz sand based backfills with the densities implied by the basic KBS-3 concept.

Field tests at Äspö with crushed rock based backfills in the 5 m diameter TBM tunnel have shown that these mixtures can be effectively compacted on site to the required density.

16.2.5 Buffer and backfill handbook

In 1995 the second part of the handbook has been prepared. It contains chapters on buffer materials, ballast materials, artificially produced clay/ballast mixtures and their chemical and physical properties. It also compiles descriptions of common and new techniques for preparation of buffers and ballast materials and application of clay/ballast mixtures.

The purpose of the handbook is to work out a basis for standardizing methods for determining characteristic properties of buffer and backfill materials and for preparing and applying them in repositories. A major issue is the access to suitable commercial smectite-rich clays for production of compacted blocks and clay-based materials for backfilling of tunnels and shafts. The results of an inventory of such materials is given in Part 2, also with respect to Swedish resources. They represent Mesozoic strata in Skåne but so far no systematic field investigation for identifying sufficiently smectite-rich areas has been made. Other considered smectite-rich materials are from Europe, Japan and USA.

The outcome of the comprehensive work to determine material criteria for producing blocks and bentonite/ballast mixtures, and the implications for producing manually manageable highly compacted blocks of bentonite and bentonitic materials as well as techniques for preparing, applying and in situ compacting tunnel backfills, is compiled in Part 2.

16.2.6 Gas migration experiments in bentonite

The copper/steel canisters are expected to have a very substantial life in the repository environment. It can nevertheless not be completely ruled out that there is a defect in the copper and eventually, corrosion may penetrate the canister walls. Assuming that available oxygen has been entirely flushed out of the repository or consumed in chemical reactions, corrosion of the steel component of each canister under anoxic conditions will lead to formation of hydrogen. Depending on the hydrogen production rate and on the rate of diffusion of this gas in the pore water of the clay buffer, it is possible that a gaseous phase will form in proximity to the exposed surfaces of the steel.

In April 1995 a programme of gas migration experiments in bentonite started. The key questions for the programme to answer are:

- At what minimum gas pressure will the hydrogen enter the buffer clay?
- How much of the buffer clay pore water is displaced by the gas?
- Is gas flow dispersed or focused along "preferential pathways"?
- What controls the direction of the gas flux?
- What is the maximum gas pressure that can be developed within the buffer?

Scope of the experimental studies

The objective of the work is to carry out a programme of carefully controlled laboratory experiments on samples of MX-80 buffer clay so as to

- identify the main variables and material-dependent parameter in the gas migration problem,
- define the main characteristics of the process and
- quantify the parameters and functional relationships between variables.

Methodology and approach

Mass transport experiments on compact clays are always very time-consuming and, recognising that there are strict limits on the number of tests that can be performed over any reasonable time-scale, it is not possible to examine the sensitivity of the gas flow parameters to all the possible variables of the problem. Given our current level of understanding, there are strong arguments in favour of focusing on a simplified system. This does not preclude the possible introduction of additional experimental variables at some future date, should the need arise. Simplifying assumptions relating to the current experimental programme are:

- Isothermal problem (no spatial or temporal variations in temperature).
- Fixed buffer clay composition (clay-type, exchangeable cation, etc.).
- Clay is fully-hydrated and initially 100% water-saturated.

It was demonstrated in earlier studies that the effective stress, σ_{eff}, acting on the material is a fundamental parameter in gas migration. Experiments in which stress is not controlled are difficult, if not impossible, to interpret. A key feature of the tests is that the sample is subjected to an independently-controlled total stress, σ which is usually maintained constant during gas injection.

The experiments are designed to reveal as much about the material responses of the clay as possible. Each test comprises a complex experimental history in which the independent variables of the problem (i.e. flow-rate and stress) are deliberately varied throughout the course of the test. The differential pressure between the injected phase

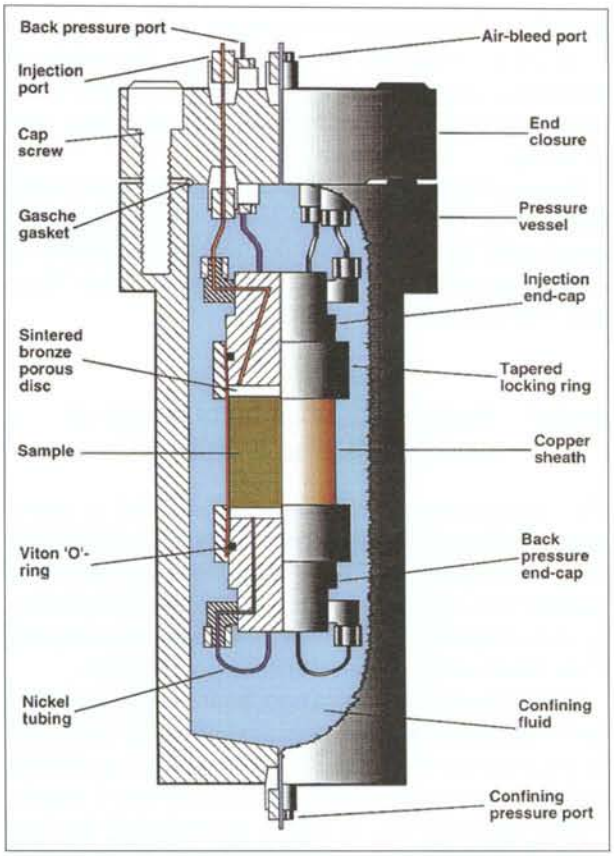

Figure 16.2-2. *Schematic of pressure vessel and sample assembly.*

(gas) and the downstream phase (water) is regarded as the main dependent-variable.

Experimental details

The axial flow (AF-geometry) apparatus comprises 5 main components: (a) a sample assembly, (b) a pressure vessel together with its associated pressure control equipment, (c) a fluid injection system, (d) a back pressure system and (e) a microcomputer-based data acquisition system.

Control of the injection flux is achieved using a microprocessor-controlled pump. The gas is water-saturated helium which is used as a safe replacement for hydrogen. The apparatus is assembled in an specially-constructed constant temperature chamber held at 20±1°C. Precise temperature control very is important in gas transport experiments.

The sample assembly, see Figure 16.2-2, consists of a cylindrical clay sample, two 0.50 cm thick "high air-entry" sintered stainless steel porous discs, two tapered end-caps, a thin copper sheath (polymer coated), two locking-rings, and the tubing connections to the vessel end-closure. The clay samples are 4.90 cm in diameter and 4.90 cm long. Copper sheaths are used to avoid the leakage of helium by diffusion during very low flow-rate experiments.

Brief description of results

One sample in a trial resaturation test exhibited very pronounced swelling which ruptured the copper sleeve. Resaturation of the second sample showed that the swelling pressure of the supplied buffer clay, under the test conditions, was close to 15 MPa. This was much higher than anticipated. Gas breakthrough occurred at a gas pumping rate of 375 $\mu L.hr^{-1}$ and an excess gas pressure of 14.9 MPa, which is very close to the swelling pressure. The test exhibited a primary peak followed by a spontaneous negative transient. A second negative transient was noted during a period of shut-in. Extrapolation of the transient gave an excess gas pressure at shut-in (= capillary pressure, P_{co}) of somewhat less than 11.5 MPa.

The next two tests both exhibited gas entry pressures which were significantly greater than total stress. During the initial gas pressurisation phase of these two tests, leaks developed between the gas injection circuit and the confining pressure circuit, leading to gradual and uncontrollable increases in the confining pressure.

Despite the earlier difficulties, the fifth test proved to be very successful. Gas breakthrough occurred at the primary peak at an excess pressure of 15.3 MPa and was followed by a very well-defined and spontaneous negative transient. This was followed, in turn, by an extended period of shut-in (zero flow- rate) which yielded another well-defined negative transient. The extrapolated excess gas pressure at shut-in (= capillary pressure, P_{co}) was, once again, somewhat less than 11.5 MPa. Given the high quality of the data emerging from this test, it was decided to follow a complex test history. Gas injection was reinstated at a pumping rate of 370 μL.hrl. The secondary peak was very small (excess pressure of circa 14.2 MPa), suggesting that the gas pathway(s) had not resealed during the shut-in period. This might be explained by the presence of residual gas bubbles along the pathway(s). The sample was then subjected to a descending history (i.e. a step function) of pumping rates (90, 45 and 0 μL.hrl) to evaluate the functional-dependence of gas flow on pressure gradient. In a final phase of this test, which is ongoing, an attempt will be made to replicate the primary peak by raising the confining pressure.

Processing of the data from MX-80-4-G gives the differential pressure against elapsed time relationships shown in Figure 16.2-3 and Figure 16.2-4 and the gas permeability versus net mean effective stress plot shown in Figure 16.2-5. The analysis assumes continuum gas flow behaviour.

It was concluded from the first phase of testing that highly compacted MX-80 bentonite is an exceedingly good barrier to gas-phase transport. Entry of a gas-phase into the clay is only possible if the gas pressure exceeds the sum of the external water pressure and the swelling pressure. The high swelling pressure therefore leads to a high gas entry pressure.

16.3 GEOSCIENCE

16.3.1 Overview

The geoscientific research at SKB is related to the crystalline bedrock and to the repository design. The research work is guided primarily by the need for input data for the long-term safety assessments that are being done. Furthermore, the geoscientific R&D work is supposed to be of

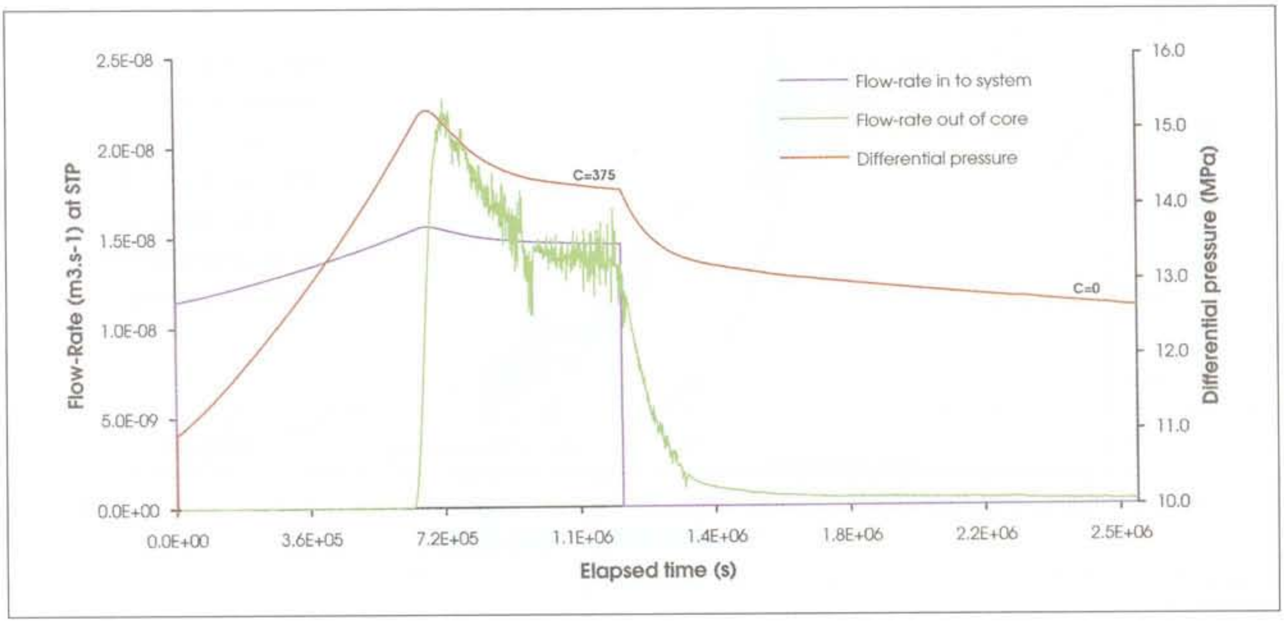

Figure 16.2-3. Experimental history MX-80-4-G (part 1).

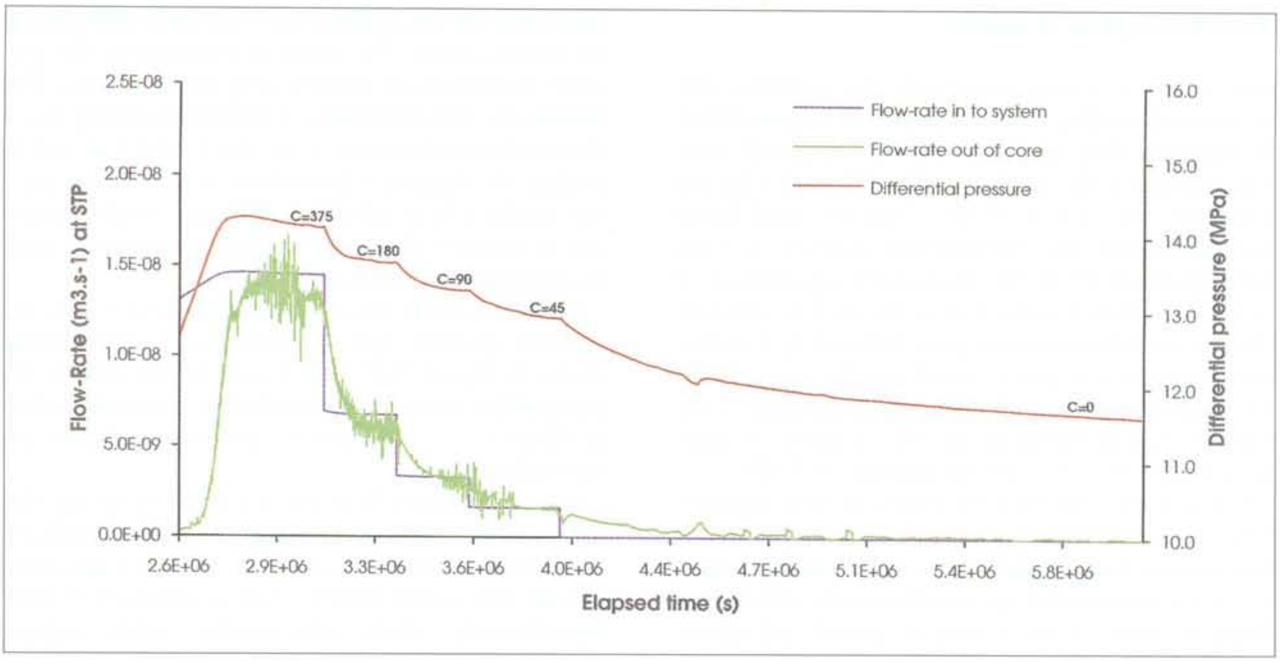

Figure 16.2-4. *Experimental history MX-80-4-G (part 2).*

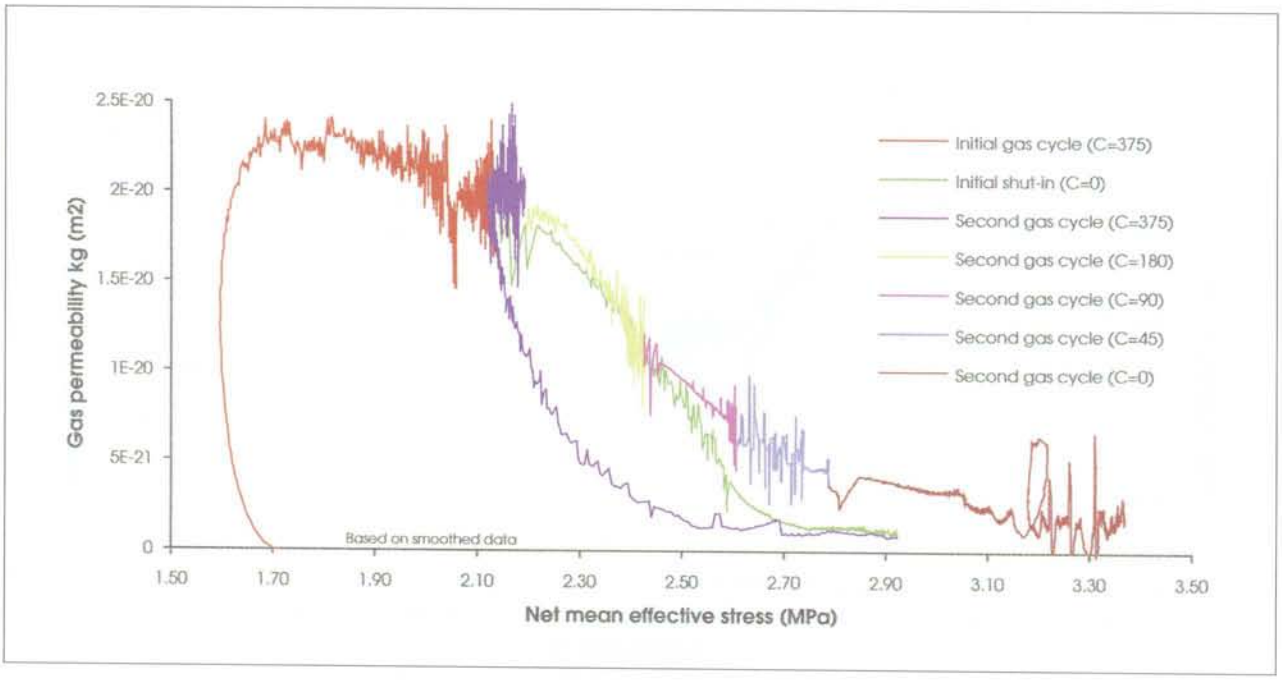

Figure 16.2-5. *Gas permeability against net mean effective stress MX-80-4-G.*

benefit in solving the civil engineering problems that are associated with the construction of a deep repository.

The rock has a number of fundamental properties that are being exploited for the long-term performance and safety of the repository. These are:

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

These properties can be more or less coupled to each other through physical or chemical processes. The application of essential safety related factors and properties for the siting process and in overviewing scales are discussed in a separate report /16.3-22/.

The rock provides long-lasting mechanical protection against external forces. A final repository in rock also provides good protection against changes in climate. Climatic changes can result in a changed biosphere with a considerably higher sea level, or alternatively can give rise to permafrost and formation of glaciers, with a lowering of the sea level as a result. The impact of such changes is minimized if a repository is placed in deep geological formations.

It is of fundamental importance for the safety of the repository that the chemical environment is stable. Reducing conditions of the groundwater are of great importance for the life of the canister and for the slow dissolution of the fuel matrix. Groundwater chemistry is determined for the most part by interaction between the minerals of the rock and the groundwater and is consequently stable over long spans of time. The chemical environment of the rock is also important for how radionuclides can be transported. Here the interaction between the different nuclides and rock is of importance.

A low groundwater flux in the rock is of importance both for the durability of the barriers and for the slow transport of none or weakly sorbing nuclides in the rock. The water flux is generally determined by the topography of the ground surface and by the hydraulic conductivity of the rock, which is in turn dependent on its fracture content.

The geoscientific programme at SKB embraces broad knowledge build-up within geology, geophysics, rock mechanics and geohydrology. The programme also includes method development and development of numerical computer models. A strong link exists to SKB's programme for instrument development.

The activities and the projects within the geoscientific programme are often coordinated with other special areas, such as geochemistry and hydrochemistry. Furthermore, the work is integrated with the research activities that are being conducted within:

- The Äspö Hard Rock Laboratory.
- Safety assessments.
- Natural analogues.
- The siting programme.

The overall objectives and main activities of the 1995 geoscience programme are expressed in the SKB RD&D-Programme that was released in September 1992. During 1995 the geoscience programme has involved the following main tasks:

- Groundwater Movements in Rock.
- Bedrock Stability.
- Groundwater and Rock-Mechanical Numerical Modelling.
- The Laxemar Deep Drilling Project.
- Geochemistry.
- Thermal conditions.

The following chapters briefly present summaries from these general geoscientific activities which were published during 1995.

16.3.2 Groundwater movements in rock

An understanding of groundwater movements is essential for a detailed safety analysis of a repository. The groundwater flow affects the degradation of engineered barriers, the dissolution of the waste and the transport of solubles in the water.

The relative importance of the parameters that describe flow in the bedrock can be treated in performance assessments and safety analyses. One of the factors that has importance for assessment of radionuclide transport of non-sorbing and sorbing species is the flow-rate of water. The flow rate of water in the bedrock is dependent on conductivity, connectivity of fractures, the boundary conditions and the driving forces.

The conceptualization of the groundwater flow distribution is important for the overall assessment of radionuclide transport, both nonsorbing and sorbing.

Estimation of hydrogeological properties in vulnerability and risk assessments

The main objective of this doctoral thesis, partly funded by SKB was to evaluate and develop approaches for hydrogeological vulnerability and risk assessments, /16.3-1/

The first project was aimed at application and evaluation of properties of standardized groundwater vulnerability classification systems.

The second project was-aimed at development of a guidance framework for monetary risk assessments before collection of new data.

The third project was aimed at the development of approaches and methods for preparing geological and hydrogeological decision bases in the siting and construction phases of a nuclear waste repository. The concept of a Positioning Index was developed for describing the conditions for fuel canister positioning with respect to specific compliance levels. A Bayesian Markov Geostatistical Model (BayMar) was developed for probability estimations with respect to set compliance levels. Application of the methodology showed that BayMar is capable of predicting conditions with respect to lithology, hydraulic conductivity, and rock quality designation index (RQD) reasonably well for both site and repository tunnel scales. A comparison between BayMar and indicator kriging showed that the two methods are analogous in describing the spatial correlation structure since the autocovariance function of BayMar is the reciprocal of the variogram. Main advantages of BayMar are that:

- it is capable of handling all mutually exclusive states simultaneously;
- no fit to any standard variogram is required; and
- it provides a formal way for handling both previously existing information and professional judgements.

When comparing predictions at Äspö, BayMar gave results closer to observed conditions than did indicator kriging /16.3-2/.

Overview of in-situ hydraulic testing

During 1994 an ad hoc-group on the knowledge of hydraulic testing and interpretation was created. The overall objective of the group (HYDRIS) was to formulate state of the art in general terms and identify future R&D-activities in the subject for SKB on the basis of a well structured overview /16.3-3/.

Geometry and hydraulic characteristics of rock fractures

The geometrical features of the intersections between joints and fractures in rocks have an influence on the groundwater flow and transport. Enlarged apertures along the intersection are supposed to form channels with higher conductivity compared to the average conductive properties of the individual joints.

A three year programme was initialized 1992 in order to develop an investigation method to obtain more information on the void geometry inside joints and their intersections. During 1994 laboratory flow experiments were accomplished by means of a biaxial cell and during 1995 a compilation of the aperture measurements was published in a doctoral thesis /16.3-4/.

The void geometry has a major influence on the hydromechanical properties of fractures, and the thesis concerns the properties of the fracture void geometry of single rock fractures. It is suggested that the parameter aperture should be used to describe the fracture void geometry and a definition of the aperture is proposed. The relation between void geometry and other fracture properties such as roughness, conductivity, stiffness and channelling are discussed.

The spatial correlation of the aperture distribution over a fracture surface influences both the mechanical and the hydraulic properties of the fracture. Therefore, a parameter defining the spatial correlation should be included in the description of the aperture distribution. It is proposed that the geostatistical parameters range and sill be used for this purpose. Aperture measurement methods may be divided into three groups depending on the basic principle of the method: surface topography measurements, grout or resin injection and casting techniques. Different experimental techniques have been developed within the thesis work. The methods are applicable to fractures of different nature and size.

A compilation of measurement results indicates that the spatial correlation (range) of fracture apertures increases with increasing mean aperture and that the range is correlated with the coefficient of variation.

16.3.3 Bedrock stability

An in-depth analysis of the possible effects of geological processes on a final repository is under way. Essential questions are whether recent movements can lead to new fracturing and whether load changes or rock block movements can decisively alter the geohydrological situation around a final repository. The objectives are to:

- quantify or set limits on the consequences of earthquakes, glaciation and land uplifts of importance in analysing the safety of a final repository for spent nuclear fuel,
- process, evaluate and increase knowledge concerning the geodynamic processes in the Baltic Shield.

Maps of the seismicity in Fennoscandia

SKB continuously follows the seismic events that occurs in the Baltic Shield. During 1995 a complete list of located Norwegian earthquakes for the years 1612-1993 were compiled /16.3-5/. On behalf of SKB the Seismological Department, Uppsala University, provided an updated database which comprises the epicenters of earthquakes in Fennoscandia during the years 1375-1993, /16.3-6/.

Special interest has been devoted to shallow seismic events. The objective of the study was to catalogue and examine shallow seismic events, i.e., with focal depths down to 2-3 km, during the years 1980-1992.

Shallow seismic events often generate short-period Rayleigh waves, Rg. Rg-waves are surface waves with a retrograde elliptical particle motion and are most clearly seen on vertical-component seismograms. They are easy to identify in seismograms since they display the phenomenon known as normal dispersion, i.e., the velocity of

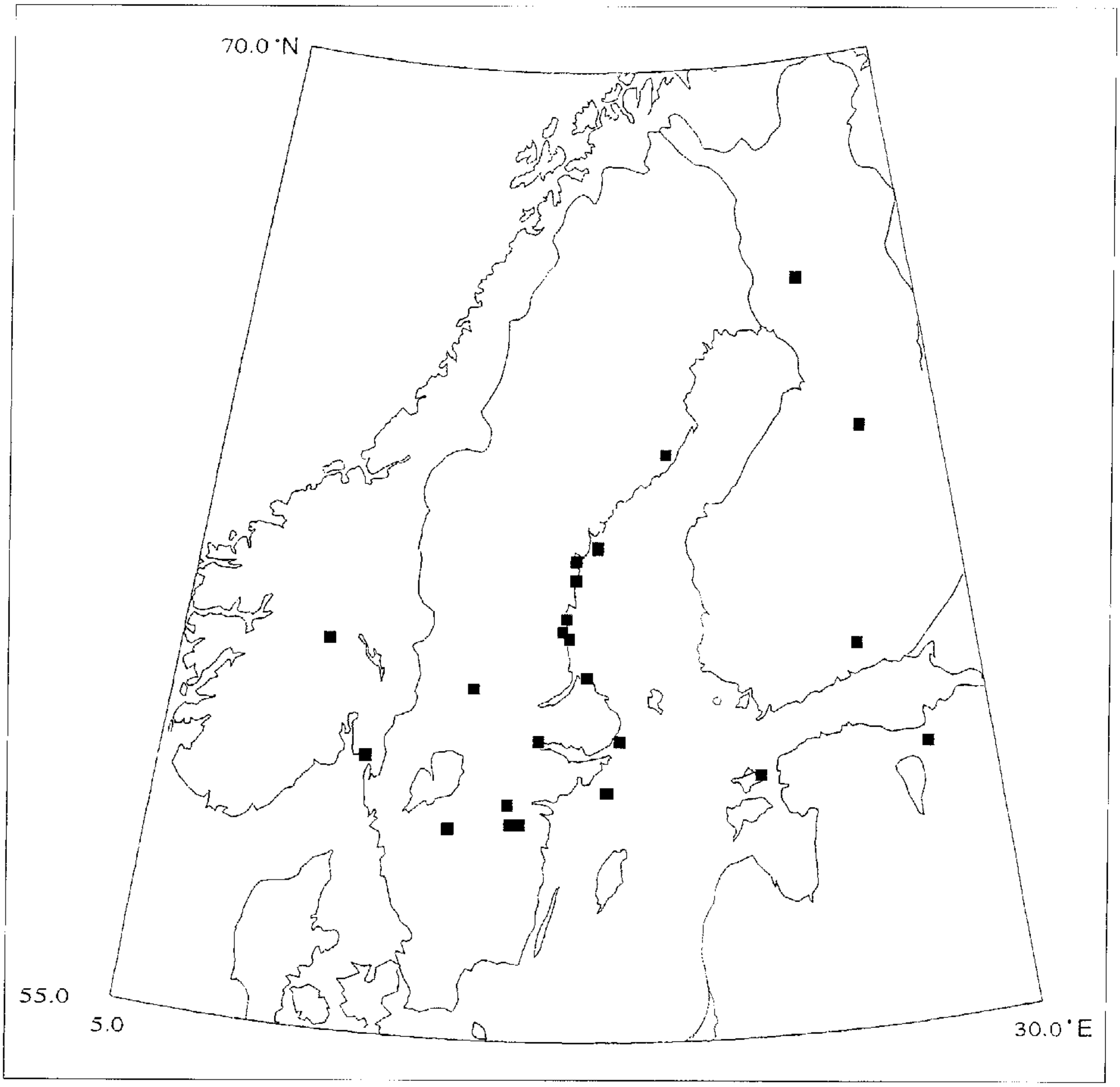

Figure 16.3-1. Near-surface earthquakes in the Baltic Shield during the years 1967 – 1992 (from /16.3-7/).

propagation increases with increasing period. The Rg-wave velocity is approximately 0.92 Vs, where Vs is the S-wave velocity, and thus arrives as a clearly identifiable phase after the S-wave. Rg-waves are rapidly attenuated and consequently recorded only by the nearest stations (in Sweden normally at distances less than 200 km, occasionally as much as 600 km). Due to the sparse number of stations in the Swedish Seismograph Station Network (SSSN), the number of Rg-phases observed for any event is small.

Another way to determine the focal depth is from macroseismic data, i.e., reports of how people have perceived an earthquake and what effect it has had at the surface (e.g., cracks in the ground or buildings). A shallow earthquake is felt with higher intensity but over a smaller area than a deeper earthquake of the same magnitude. Empirical relationships give estimates of the focal depth as a function of intensity and area of perceptibility.

SSSN seismograms for 260 Swedish earthquakes were systematically scrutinized in search of Rg-waves, in order to identify shallow earthquakes. In a previous study for the years 1967–1979, nine events were identified as shallow earthquakes in Sweden. The locations of these events showed a clear concentration to the east coast of central Sweden. Out of the nine identified shallow Swedish earthquakes in the previous study, the new compilation now instead found a grouping of earthquakes in central Sweden and only two events along the coast of the Gulf of Bothnia. Figure 16.3-1 shows the identified shallow earthquakes for the years 1967–1992 (from /16.3-7/).

Coupling between earthquakes and groundwater pressure, chemistry changes

In Japan the Power Reactor and Nuclear Fuel Development Corporation (PNC) is carrying out research into the coupling between earthquakes and changes in groundwater pressure and chemistry. SKB has an agreement with PNC for exchange of information on this subject. PNC is conducting in-situ measurements at the Kamaishi research mine, /16.3-8/.

It is too early to draw any far reaching conclusions on these phenomena. More data need to be collected temporally and spatially and existing data need more processing and interpretation. However the following preliminary conclusions can be drawn after approximately five years:

- Ground motion decreases with depth
- Earthquakes can cause changes in groundwater pressures up to 3 m of head.
- These pressure changes last for a few days to a 1 week after which the monitored pressure returns to the baseline trend.
- Not all earthquakes cause changes.
- It is not clear if earthquakes cause changes in groundwater chemistry.

Thermal evidence of a Caledonide foreland

During 1995 SKB initiated studies on thermal evidences of an extended Devonian molasse sedimentation in Fennoscandia. A separate project on mainly apatite fission track analyses was set up. The general objective is to increase the understanding of the vertical loadings on the crystalline basement in a geological time perspective.

So far this sedimentation is discussed in a report which is a compilation of different indicators, /16.3-9/. Thermal indicators include $\delta^{18}O/\delta^{13}C$ and conodont alteration indices. (Conodonts are extincted marine animals represented in the fossil record. During heating, organic material within the lamellar structure of the conodont elements breaks down and carbon fixing produces a continous colour change which in turn can be calibrated with temperature.) Furthermore illite/smectite ratios and illite crystallinity, oil maturation in Lower Palaeozoic sediments, as well as apatite fission tracks and lead mobility in basement rocks all suggest raised temperatures during the Late Palaeozoic, as high as 125°C at the present level of erosion. These temperatures persisted for at least 100 Ma in south central Sweden. Calculations indicate a sedimentary cover thickness of 3+/-1 km, a thickness which probably varied geographically.

The authors of the report propose that this sedimentary cover was largely composed of molasse, which was eroded from the Caledonides, and deposited into a Caledonian foreland basin. In southern Sweden this basin probably would have interfered with a basin developed to the north of the Danish-Gemman arm of the Caledonides. Regional variations in sedimentary thicknesses may be explained by flexing of the lithosphere and the temporary existence of a foreland bulge. A dynamic interaction between the load/erosion of the Caledonian nappe stack and coupled changing geometry of the foreland basin and bulge is compatible with the stratigraphic, petrographic and fluvial directions of Devonian sediments remaining in the Baltic Sea and Baltic States. In southern Scandinavia the Permian to Triassic uplift and erosion reduced the cover significantly, see Figure 16.3-2.

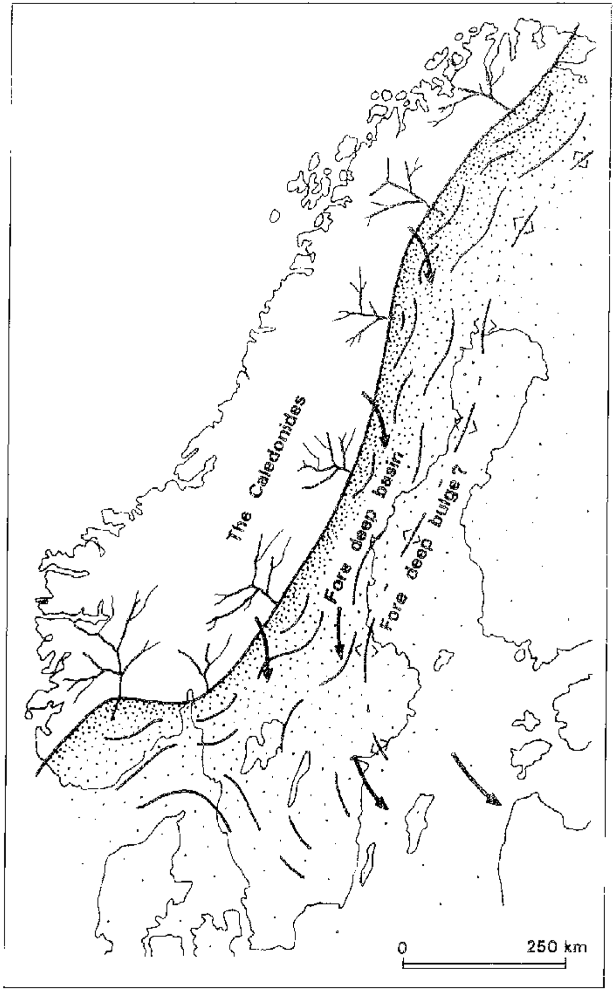

Figure 16.3-2. Tentative sketch of a Caledonian foredeep basin. The sedimentation was not contemporaneous over the entire basin. During the Late Silurian terrestrial facies prograded towards SE /Basset, 1985/, a direction which was confirmed also during the Devonian in the Baltic Sea area /Kurss, 1992/.

Reconstructing the tectonic history of Fennoscandia from its margins, the past 100 million years

Special interest has been devoted to tectonic indicating markers from the margins of the Baltic shield. In the absence of onland late Mesozoic and Cenozoic geological formations the tectonic history of the Baltic Shield over the past 100 million years can most readily be reconstructed from the thick sedimentary basins that surround Fennoscandia on three sides. Changing patterns of sediment thickness accompanying active tectonics, as observed on high resolution multichannel seismic reflection lines, record the boundary conditions of deformation internal to the Baltic Shield. Tectonic activity around Fennoscandia through this period has been diverse but can be divided into four main periods. The highest levels of deformation on the margins of Fennoscandia were achieved around 85Ma, 60-55Ma and 15-10Ma, with strain-rates around 10^{-9}/year. Within the Baltic Shield long term strainrates have been around 10^{-11}/year, with little evidence for significant deformations passing into the shield from the margins, /16.3-10/

Rock Stress Database of Sweden

In order to retrieve and compile the measurements conducted in Sweden so far, a database which includes all primary stress measurements has been developed on behalf of SKB. Some of the early measurements have been discarded, mainly due to uncertainties in the quality of the results. A separate report describes shortly the subject of rock stress measurements, and the structure and content of the database, /16.3-11/.

The present version of the database contains 574 individual results obtained from measurements in 79 boreholes. The results have either been obtained by measurements with a three dimensional overcoring method or with the hydraulic fracturing technique. The results cover a depth range from approximately 10 m to almost 1 000 m below surface. Close to 145 measuring points are located at a depth larger than 400 m below surface. Geographically, the database includes data from Kiruna in the north to Stidsvig in the south of Sweden.

The database has been developed in a personal computer environment but has, since completion, also been converted to function on workstations using the operative system of UNIX.

16.3.4 Groundwater and rock mechanical modelling

Numerical models are primarily refined within the framework of the activities at the Äspö Hard Rock Laboratory. However, some supplementary efforts emphasizing coupled processes are pursued within the SKB general R&D programme.

Thermal-Hydro-Mechanical modelling

Development and verification of coupled thermo-hydromechanical models has been taking place in the DECOVALEX project, (international cooperative project for de DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation). DECOVALEX was initiated by SKI and SKB has been a member of the Steering Committee for the project. Within the DECOVALEX project SKB has emphasized the analytical approaches for a better understanding of the calculation results and their dependence on boundary conditions and dimensionality.

The first three year programme (DECOVALEX I) ended during 1995 and the SKB involvement has been summarized in four Technical Reports and two scientific papers /16.3-12, -13, -14, -15, -16, -17/.

Palaeohydrogeological programme

Europe has been subject to major changes in surface environment and climate over the last million years. At their most extremes, these have involved loading by up to 3 km of glacier ice and deep freezing by permafrost. A large quantity of detailed data has been collected over the past 20 years that support the Milankovitch theory, in which global climatic change is ultimately driven by changes in incident solar radiation because of long-term changes in the Earth's orbit around the Sun. There are several climate models based on Milankovitch cycles, usually calibrated with known climatic data from previous glaciation periods. These models also allow forecasts to be made of the future climate. During 1990-1991, SKB and Teollisuuden Voima OY (TVO) in Finland carried out a joint inventory of the international state of knowledge regarding ice ages. The inventory resulted in a choice of two different models, the ACLIN and Imbrie & Imbrie models in order to describe a future ice age scenario. Both these models show that conditions similar to those of the present warm period will recur in 120 000 years, although a period with a relatively warm climate can, however, be expected in 75 000 years.

SKB has initiated a palaeohydrogeological programme, /16.3-18/, with the following general objectives:

- to identify the principal climatically-driven processes that, over a time scale of the order of 100 000 years, could affect the integrity of a deep waste disposal site and influence the dispersal of radionuclides from it,
- to develop models of these processes that can be first constrained by and then tested against the geological and hydrogeochemical record of these processes in the past.

Through these two objectives confidence can be built in that we are able to understand and predict the groundwater conditions in possible repository areas based on known long-term climatic conditions.

Furthermore, through an additional objective, a basis will be formed for performance assessment for a nuclear waste repository in the long-term perspective and that is:

- to develop a future climate function that can be used to drive the process models and produce a probabilistic estimation of the future operation of these processes and potential impacts on a specific repository site.

During 1995 sensitivity testing has been done for a time dependant glaciation model which covers Scandinavia and has been developed by means of SKB funding /16.3-19, -20/. Regional palaeohydrogeological modelling has commenced in a regional perspective and with databases exemplified for the Äspö-Laxemar area /16.3-21/.

16.3.5 The Laxemar deep drilling project

The natural groundwater flux at repository level is not necessarily controlled by the local flow gradients, but is more likely governed by regional topographic conditions. It is judged essential to further refine regional flow models that shed light on long-term transient changes. This is especially true for coastal repositories, where the transient flow changes can be affected by glaciation, deglaciation, land uplift and the salt/fresh water boundary, which in turn alter the boundary conditions of the calculation models. To obtain a better understanding of the water flux in a regional perspective, surrounding Äspö HRL, and at depths exceeding 1000 m, a hole was drilled in autumn 1992. The coredrilling (wire line drilling technique) was carried out in the Laxemar area near the Simpevarp peninsula in the municipality of Oskarshamn. The hole was drilled from 200.8 m to a total depth of 1700.5 m with standard NQ dimension (diam. 75.7 mm).

During 1993 the investigation phase commenced and the following activities were accomplished during 1995:

- Additional caliper logging and TV-monitoring.
- Compilation of geological data.
- Comparative analysis of fractures in KLX02 and some cored boreholes at the Äspö HRL.
- Finalization of hydraulic testings.

In 1996 the hydraulic testing programme will be analysed and successively the rock-stress measurements will be performed. An integrated analysis and interpretation is now foreseen to late 1996 and 1997. /16.3-23, -24, -25/

16.3.6 Hydrogeochemistry

Hydrogeochemistry involves the chemical processes and interactions taking place in the bedrock and which are of importance for assessment of the long-term safety of a repository. In this context it is mainly the chemistry of the groundwater which is considered. The chemistry of the minerals is of interest only through its potential effects on the hydrochemistry and on the retardation of radionuclides transported by the groundwater. In favourable and stable hydrochemical conditions the copper canisters are likely to remain intact for millions of years.

The groundwater chemical composition is a result of chemical processes and mixing. A good knowledge of the chemical processes makes it possible to differentiate between mixing and reaction, and to delineate the end-members which are mixed to the samples collected. The end-members can be considered as tracers added to the groundwater at different occasions in the geological past. When and if these occasions can be defined it is to some extent possible to track the evolution of the groundwater system, sometimes very far back into the past. When the conditions of the past are known the knowledge can be used to predict the conditions of the future.

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% /16.3-26/. 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.

Äspö

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 /16.3-27/.

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 /16.3-28/. To further clarify this relationship, more dating methods will be applied, including Cl-36 analysis and noble gas analyses. The results from both

Table 16.3-1. Chemical composition of groundwaters in Äspö diorite and greenstone. The concentrations are given in mg/l,*µg/l.

	ÄSPÖ DIORITE		GREENSTONE
Substance	High conductivity	Low conductivity	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

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 /16.3-29/. 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.

Redox conditions

Bacterial oxygen reduction has been found to be most important when it comes to consuming dissolved oxygen in infiltrated surface water /16.3-30/. 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 that this process has occurred on a large scale /16.3-31/. 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.

Water-mineral reactions

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 /16.3-32/, 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 /16.3-33/. Table 16.3-1 shows a comparison between samples taken in low- and high-conductivity boreholes in different rock types /16.3-34/.

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 /16.3-27, -35, -32, -36/. The kinetics of the ion exchange reactions are such that the reactions can be studied both in the laboratory and in the field and are affected by changes in groundwater conditions during tunnel construction.

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 /16.3-37/.

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 /16.3-36/.
- Reduction of iron(III) minerals and accompanying increase in carbonate and iron concentration /16.3-36/.
- Reduction of sulphate and increase in carbonate and sulphide concentrations /16.3-31/.

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 /16.3-35/.

The *fracture-filling mineral composition* bears traces of former hydrochemical conditions. This can be utilized in two different ways:

- to trace previous groundwater flow patterns,
- to evaluate the transport of radionuclide-like substances.

Research aimed at clarifying former flow situations is being conducted within the hydrogeological programme.

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 /16.3-35/. 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 /16.3-32/. 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.

16.3.7 Thermal conditions

It is reasonable to assume that the so called SKB Study Sites constitute a representative range of bedrock temperature conditions likely to be encountered at the repository site. This is because of the geographical spread of the study sites, as well as the spread in rock types encountered in the boreholes. Table 16.3-2 shows a summary of temperature conditions at all sites /16.3-38/. The temperatures at 500 m depth below the ground level varies between 5.5 –14.4°C, while the temperature gradient varies between 9.5 –15.5°C/km. If also those granitic areas of anomalous high radiogenic heat production are included the plausible range of temperature gradients might be extended up to say 18°C/km. As noted above the values given in Table 16.3-2 differ slightly (<1°C and <1.5°C/km) from values presented in reports summarizing results from the study sites. The values given in Table 16.3-2 are believed to be more representative of the conditions 500 m below ground than previously presented values.

Regarding the thermal conditions of the crystalline rock types found in Sweden, separate maps have been compiled for thermal conductivity, geothermal heat-flow and temperature at 500 m depth /16.3-39/. The presentation of the thermal conductivity is based on processing of earlier published material. Calculations of the thermal conductivity from the mineral content was made for about 4000 samples, including a number of samples with measured thermal conductivity, see Figure 16.3-3. The surface heat flow and the temperature at 500 m depth is based on a work in progress, Atlas of Geothermal Resources in Europe.

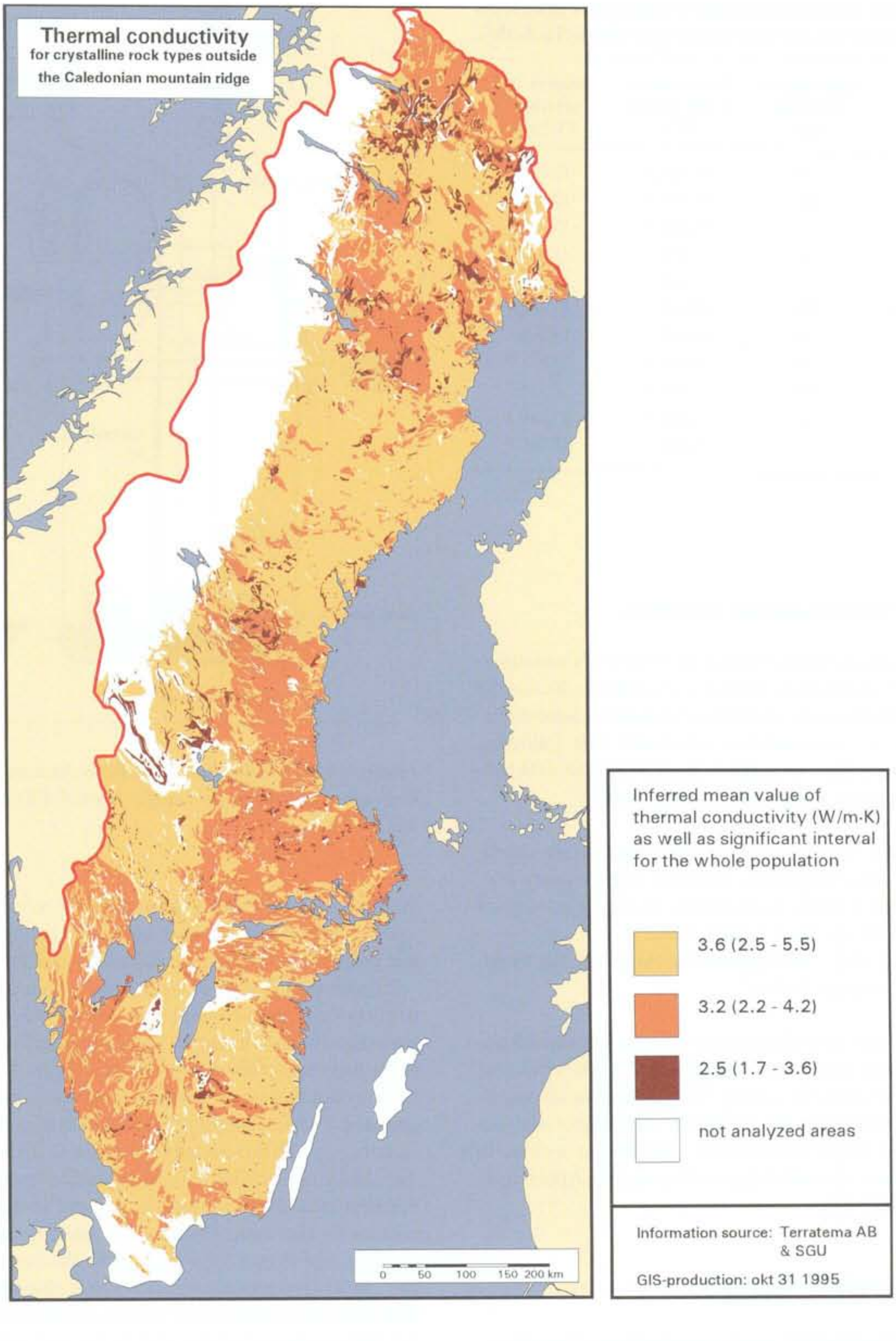

Figure 16.3-3. *Thermal conductivity of crystalline bedrock types in Sweden, (from /16.3-39/).*

Table 16.3-2. Temperatures and temperature gradients at 500 m depth for the SKB study sites (from /16.3-38/).

Study Site	Approximate elevation (m)	Temperature at 500 m level (°C)	Temperature gradient (°C/km)
Sternö	23	13.4±0.3	12.5±0.5
Klipperås	200	13.4±0.5	13.7±0.5
Kråkemåla	15	14.4±0.3	13.9±0.5
Fjällveden	60	13.2±0.5	15.0±0.5
Finnsjön	33	11.6±0.3	12.7±0.3
Svartboberget	290	10.0±1.0	13.2±0.6
Gideå	115	10.9±0.5	15.5±0.6
Kamlunge	150	8.7±0.5	11.6±1.0
Taaviunnanen	675	5.5±0.5	9.5±1.0
Laxemar	18	14.4±0.7	15.2±0.5
Äspö	7	14.6±0.3	15.0±0.3

16.3.8 Miscellaneous activities

The SKB geoscientific programme often deals with interdisciplinary approaches. Thus it is essential to discuss the obtained R&D results in informal manners where different points of view could be ventilated. The following seminars have been arranged with participation of different experts in the broad field of geoscience:

- "Impacts of longterm change on subsurface conditions – time sequences, scenarios and boundary conditions for safety assessments of a deep geological repository for spent nuclear waste".
- "GIS at SKB – information exchange with the Swedish Geological Survey".

Beside these open discussions it is of great importance to present and assess the ongoing R&D work within the international scientific society. The SKB Geoscience programme encourages the involved consultants and researchers to attend international meetings as well as to publish papers in scientific journals (see also Appendices 2 and 7 in Part III).

16.4 CHEMISTRY

16.4.1 Solubility and speciation of radionuclides

Results from the measurements of Tc(IV)-solubility in neutral to alkaline carbonate solutions /16.4-1 or 16.4-2/ have been included together with other published or assessed values in a thermodynamic data base for technetium /16.4-3/. This data base will be used to calculate equilibrium solubilities of technetium at both ambient and

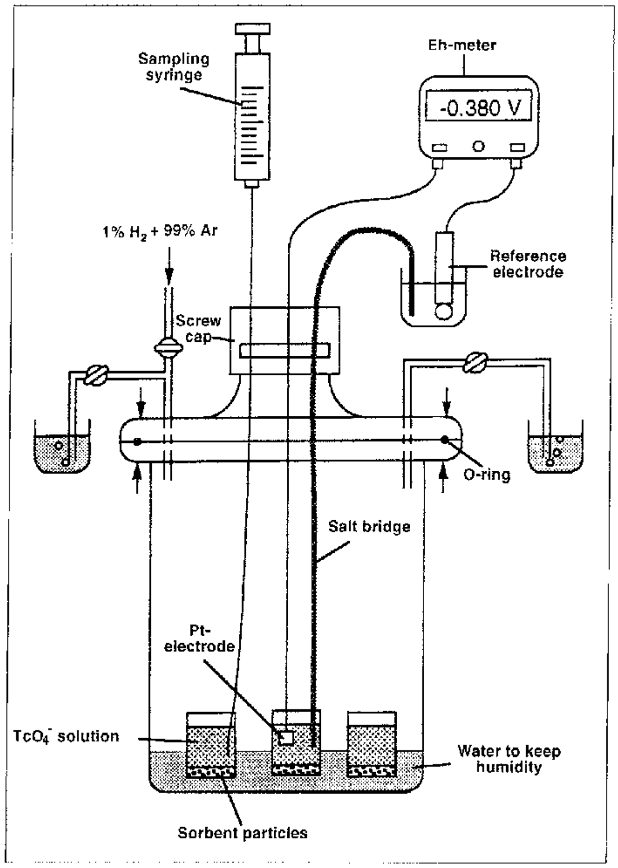

Figure 16.4-1. Equipment used for reduction of TcO_4^- by heterogeneous electron transfer from Fe(II) bearing solids.

increased temperatures. The data base is valid up to a maximum of 300°C which is more than enough to calculate technetium solubility at repository conditions.

Experiments have been performed which confirm that pertechnetate and neptunyl ions will be reduced to their tetravalent oxidation states (Tc(IV) and Np(IV)) in a deep repository environment /16.4-4/, see Figure 16.4-1. This is important because tetravalent technetium and neptunium have a lower solubility /16.4-2/ and a much stronger tendency to sorb on rock mineral surfaces. The kinetics of the reduction reactions have been studied and it has been demonstrated that sorption on mineral surfaces which contain ferrous iron, Fe(II), are needed to mediate the reaction, see Figure 16.4-2 /16.4-5/. Divalent iron ions, Fe^{2+}, in a homogeneous solution is not efficient /16.4-6/. However, the conditions for effective reduction are fulfilled both in the repository and in the rock-groundwater environment where solid surfaces with Fe(II) are plentiful. For example, there will be an inner steel canister, a bentonite overpack containing Fe(II) and the surrounding granitic rock which contains plenty of minerals with ferrous iron.

Swedish experts, supported by SKB, are participating in the OECD/NEA project TDB with the aim to produce critical reviews of the chemical thermodynamics of those

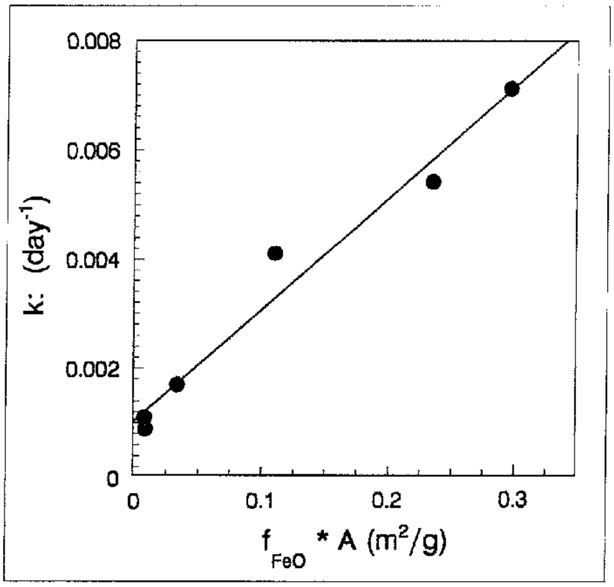

Figure 16.4-2. Pseudo first order rate constants for the disappearance of technetium from solution by reduction of TcO_4^- on electromagnetically separated fractions of fissure filling, plotted against f_{FeO} * A, *where* A = *BET surface area (m^2/g) and* f_{FeO} = *iron content given as the mass ratio of* FeO.

elements that are of particular relevance to the safety assessment of waste disposal. Two important milestones have been reached by the publication of two completed reviews: Chemical thermodynamics of uranium /16.4-7/ and Chemical thermodynamics of americium /16.4-8/.

16.4.2 Retention by sorption and diffusion

Sorption of radionuclides on mineral surfaces are described by ion exchange and surface complexation models. Fundamental studies of this kind are being supported by SKB /16.4-9/. Surface charge titrations as well as batch sorption experiments are carried out on pure mineral phases of quartz, hematite and goethite and on metal oxides. The primary aim is not to replace the use of sorption coefficients, K_d-values, but to investigate their reliability. The use of sorption coefficients, K_d, is still the best approach in performance assessment /16.4-10/.

Diffusion of radionuclides into the connected system of microfractures in seemingly intact rock is an important retention mechanism, referred to as "matrix diffusion". In fact sorption of radionuclides on the mineral surfaces of the microfractures is more important for the calculated radionuclide retention than sorption on the surfaces of open fractures with flowing water. The properties of the bulk rock adjacent to open fractures are therefore important. The chemical parameters governing the chemical retention of radionuclides are diffusivity D_e, porosity ε (diffusion porosity) and sorption on the bulk rock K_d. Other important parameters in this context are the "flow wetted surface" and "penetration depth" by diffusion. Throughout the years much investigative work has been carried out in the area of matrix diffusion. The laboratory work has produced vast amounts of data, which has contributed to the understanding of matrix diffusion and connected phenomena. The data available should make a good data base for use in calculations in the safety assessment. However, it is often difficult to extract data from the literature with sufficient information about experimental conditions etc to be useful.

Matrix diffusion laboratory studies have covered various phenomena and effects that could be of importance in the safety of the repository. Some of them are pore sizes, pore connectivity, anion exclusion, surface diffusion, the impact of large rock overburden, the question if weathering hinders or enhances diffusion, and the effect of dead-end pores to diffusion in small rock samples. Therefore a literature review covering the investigations in the area has been carried out /16.4-11/. Matrix diffusion data available in the literature has been collected and theory and experimental procedures have been studied. Also natural analogues connected to matrix diffusion has been reviewed in the report.

Diffusion of cesium, strontium and iodine through a bentonite clay buffer is important to consider in performance assessment. The release is delayed and slowed down by this process. Considering its importance as identified in SKB 91 and later confirmed in the SR 95 performance assessment studies, new experiments have been performed during 1995. The aim was to measure directly the diffusivities (D_a and D_e) and the sorption in the compacted clay (K_d). The results remains to be reported.

16.4.3 Influence of colloids and microbes

Column experiments with colloids and radionuclides have been concluded and reported /16.4-12, -13, -14/. The importance of ionic strength and pH were demonstrated. A high enough ionic strength or a pH close to the point of zero charge destabilises a colloidal suspension of minerals. However, it was also demonstrated that mineral colloids are indeed mobile in a rock-groundwater environment under special conditions and also capable of carrying radionuclides. The strongest argument against colloids as a significant carrier of radionuclides from a deep repository is the fact that the concentration of natural colloids is very limited. This has been further manifested in a recent review prior to the safety assessment study SR 95 and used as a reference there /16.4-15/. It was concluded that the expected concentration of colloids is in the range of 20-45 µg/l which is considered to be a very low value, certainly low enough to make colloids unimportant.

In addition to natural colloids it is also necessary to evaluate if colloidal particles can be generated by the

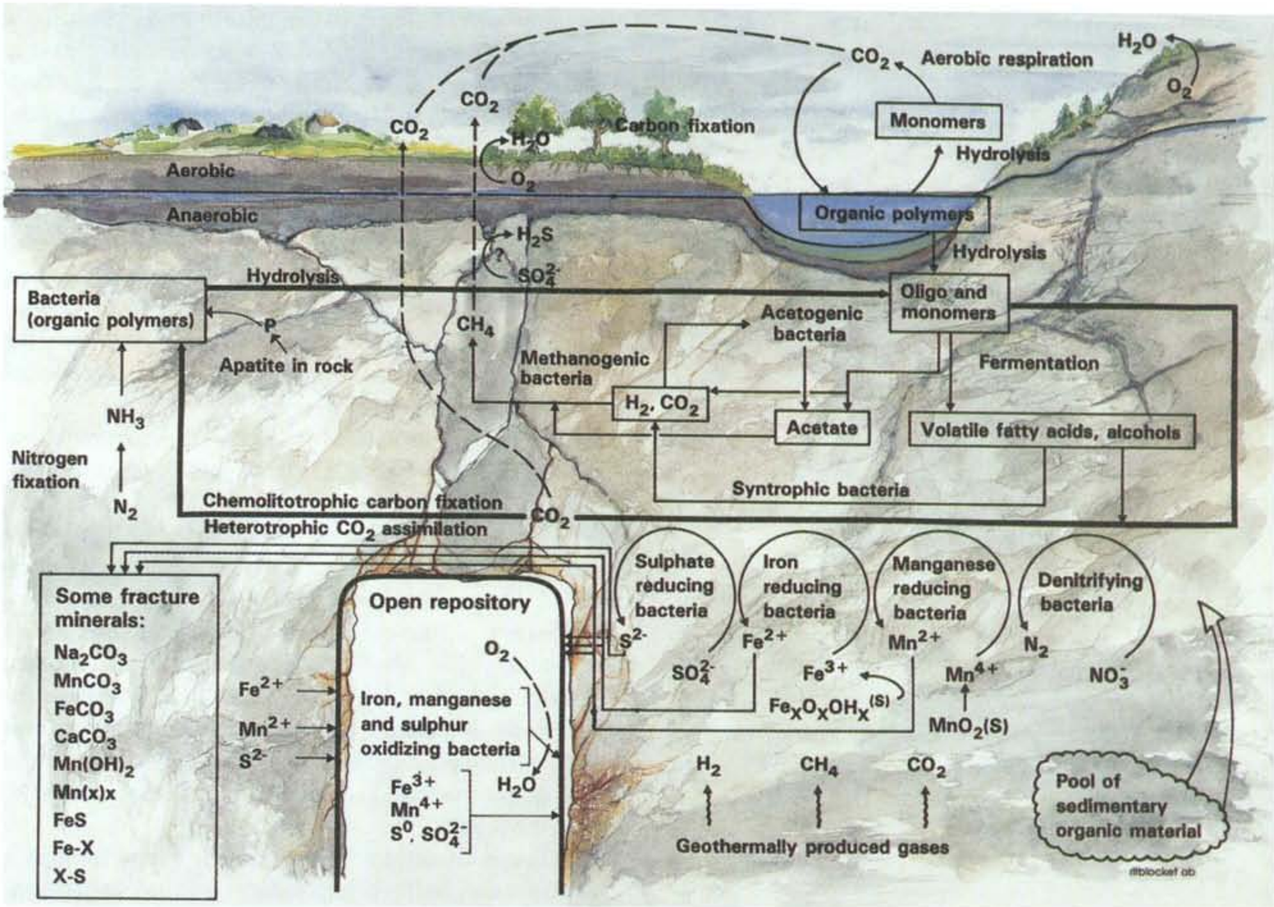

Figure 16.4-3. Summary of possible subterranean bacterial processes.

materials used or any other disturbance. For example a bentonite clay can in principle generate colloids and strong hydraulic changes due to glaciation could also influence the stability of colloids due to dilution of groundwater etc. This is discussed in the previously mentioned review report and it is concluded that none of these effects should be of any significant importance /16.4-15/.

Transport of colloidal particles by gas, for example hydrogen gas released as a result of steel corrosion, is another issue that has been considered. The transport of mineral particles by gas bubbles has been demonstrated by experiments /16.4-16 and 16.4-17/. Preliminary studies indicate that this is not of any particular concern but the field remains of interest and more careful investigations are therefore supported.

The geosphere is not sterile, at least not from the surface and down to a depth of 1 km, and probably much deeper than that. Microbes can in principle influence the isolation of radioactive waste, for example solubility and transport of radionuclides, and gas generation. However, of most importance is their ability to change the chemical environment with which the canister and the waste can come into contact, see Figure 16.4-3. Studies of subterranean microbes are being performed at Äspö and within the frames of natural analogue studies, for example at Oklo, Palmottu and in Jordan. The state of the art of investigation results and possible implications of microorganisms on the performance assessment of radioactive waste disposal have been summarised and reported /16.4-18/. As a result of this study the following tasks were identified to be of relevance for the performance assessment of radioactive waste disposal.

Existence of subterranean bacteria:
Energy sources and fluxes of energy available for bacteria at repository conditions.

Dissolution and transport of radionuclides:
Production of complexing agents by bacteria.

Redox processes catalysed by bacteria:
Bacterial consumption of oxygen and production of reducing compounds such as sulphide and ferrous iron.

Bacterial recombination of radiolysis products:
Removal of oxidising components.

Bacterial gas production and consumption:
Production and consumption of carbon dioxide, hydrogen and methane.

Bacterial influence on corrosion:
Sulphide corrosion of copper.

Influences of microbial activities are not necessarily negative, for example bacterial mediation of redox processes are helpful to maintain a low redox potential in the repository and its surroundings. Gas consumption and decomposition of organic materials are other examples of processes with potentially positive implications. Example of negative influences are gas generation, formation of complexing agents and, most important, generation of sulphide by sulphate reducing bacteria. Sulphide produced by bacteria can corrode copper and this has been discussed as early as 1978 /16.4-19/, when the copper canister alternative was first presented. Therefore, we are pleased to have obtained results that demonstrate that sulphate reducing bacteria cannot survive in a bentonite clay buffer due to the lower chemical activity (vapour pressure) of water in there /16.4-20/. Sulphide production outside the buffer is less problematic due to the transport resistance of the clay.

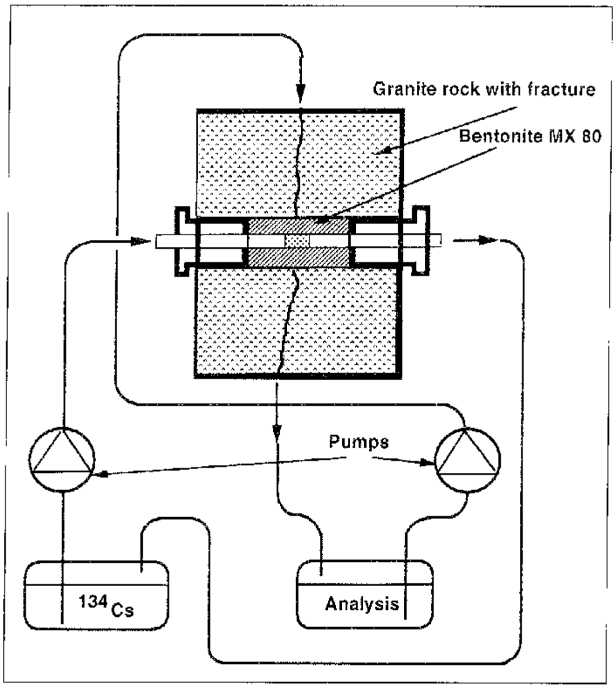

Figure 16.4-4. Near-field release simulation experiment to be implemented in the CHEMLAB-probe.

16.4.4 Validation experiments

In-situ experiments with radionuclides are planned for the Äspö Hard Rock Laboratory to be performed in the CHEMLAB-probe. Parallel with the development of the probe, the experiments are being prepared in the laboratory. Diffusion cells have been constructed and experimental equipment for working under in-situ hydraulic pressure in the borehole. An example of experiments being planned is the simulation of near-field release of radionuclides from the waste to an open rock fracture and the retention of that caused by the bentonite buffer. One of the experimental setups being tested is shown in Figure 16.4-4.

Cement is frequently used in underground construction for several reasons, for example concrete structures and pavement, cement grouting of fractures and shotcrete on tunnel walls. Concrete made of Ordinary Portland Cement has a pore water with a high pH due to alkali hydroxides (NaOH and KOH) and portlandite ($Ca(OH)_2$). There are models to calculate the interaction between concrete pore water and the rock, but they need to be tested. Therefore experiments have been performed where synthetic cement pore fluids are percolated through columns filled with crushed rock minerals. British Geological Survey is performing the experiments, jointly supported by NAGRA, NIREX and SKB. A first set of experiments, their results and interpretations have been reported /16.4-21/. The results indicate that reactions of high pH-water with granitic rock minerals are slow. The tendency is to form CSH (Calcium Silicate Hydrate) phases which diminish the flow porosity of the columns. It has been decided to continue the experiments and the project with a second phase from September 95 to August 97.

16.5 NATURAL ANALOGUE STUDIES

16.5.1 Natural analogues and performance assessment

SKB has, since 1981, been involved in natural analogue investigations. This has become increasingly popular over the last ten years or so and such studies form an integral part of many national programmes for nuclear waste disposal. It has been found useful as a means of identifying and understanding processes which could occur in the vicinity of a repository and to derive data to test laboratory based models /16.5-1/. However, so far, explicit use in performance assessment studies are more sporadic and most of the applications relate to conditions in the near-field; see for example Table 16.5-1 /16.5-2/.

An example of a near-field related application is the test of models used in performance assessment to simulate the development of a redox front due to radiolysis. It has been postulated that oxidants produced by a radiolytic process in a damaged canister escape to the surroundings, which are normally reducing. The transport of oxidants would

Table 16.5-1. Processes and analogues studied within SKB's programme and the use of analogue information in safety assessment.

Processes in the near field	Natural analogues	Safety assessment
Canister corrosion	7	KBS-3, SR-95
Bentonite stability	8 (3,4,6)	KBS-3, SR-95
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	8.	Bentonite deposits
2.	Alligator Rivers	9.	Uraninite samples
3.	Cigar Lake	10.	N Sweden (drill cores)
4.	Oklo	11.	Porjus (old concrete)
5.	Palmottu	12.	Uppsala (old concrete)
6.	Maqarin	13.	Äspö (fracture filling)
7.	Copper objects		

be by diffusion in the clay surrounding the canister, and further by a combination of flow and diffusion in the rock fractures surrounding the deposition hole. The advance of the redox front has been calculated with models and it was concluded that it is unlikely that the front will ever move past the bentonite clay buffer /16.5-3 and 16.5-4/. Several of the modelled processes were compared with data from the uranium mine Osamu Utsumi in Poços de Caldas, Brazil /16.5-3, 16.5-4 and 16.5-5/.

In the previous years of the natural analogue study of the Cigar Lake uranium deposit in Canada, several models were developed which were based on calculation models for performance assessment. One of these tailored models, the steady state, near-field transport model, is conceptually based on solubility limited dissolution of spent fuel, diffusive and advective mass transport, a porous medium approximation of the geological formations and the typical characteristics of the geometry of the Cigar Lake uranium deposit. The calculations using these models show that the release of uranium from the ore body is negligibly low; the release of sulphate equals roughly the release of dissolved hydrogen, indicating possible water radiolysis /16.5-6 and 16.5-7/.

Another model applied to Cigar Lake was the model of radiation energy deposition. The problem of water radiolysis was addressed directly by calculating the proportion of total radiation energy deposited into the pore water in the ore body. The calculation results show that only a small fraction of the total radiation energy has been deposited into the pore water /16.5-8/. The observed production rate of oxidants in the Cigar ore is in agreement with what is calculated by the mass transport model /16.5-9/.

Models that couple mass transport and chemical reactions have also been applied to the Cigar Lake case. Attention was focused on the water-rock interaction within the clay halo surrounding the ore body, especially on the hematisation of the clay adjacent to the ore. It was assumed that water radiolysis in the ore body is the source of oxidant production. The results of the computer simulation show that, with a certain rate of oxidant production, pyrite in the ore zone can be oxidised and it is also possible for hematite to precipitate in the clay near the ore. Only a slight amount of uraninite in the ore will be oxidised. However, there is a threshold value for the oxidant production rate. Below this value, pyrite in the ore is not oxidised and only trace amounts of hematite are expected to form in the clay halo. The threshold value coincides with the value obtained by the mass transport calculations and by the model of energy deposition /16.5-6/.

Results from the Cigar Lake exercise have been revisited and further evaluated but this has not yet been reported.

16.5.2 Oklo

The first fossil reactor zone in the uranium ore of Oklo, Gabon, was discovered in 1972. Since then no fewer than 15 different spots have been found with traces of nuclear reactions (i. e. reactor zones) in Oklo, Okelobondo and Bangombé. Okelobondo is situated near Oklo but Bangombé is about 20 km to the south. The fossil reactors are all in a uranium ore belonging to the same geological formations consisting of sandstone, organic rich shales, mudstone and dolomite. The uranium deposits are concentrated in a sandstone layer about 3-6 m thick. The reactor zones in Oklo were active 2 billion years ago and it was the same type of nuclear fission as for example in a light water reactor with uranium fuel. The radioactive isotopes have since decayed, but traces of the reactions remain, as do many of the materials that were involved.

The project "Oklo, Natural Analogue for a Radioactive Waste Repository" started in 1991, managed by CEA and with financial support from the EU. Active collaboration was established with SKB and other organisations outside the European Communities at that time. SKB's interest has mainly been focused on the reactor zone in Bangombé which is suitable to study because it lies relatively close to the ground surface (about 15 m). The program of the Oklo project was divided into the following tasks:

- In situ sampling.
- Study of the source term.
- Studies of the geochemical system (ancient and recent migration).
- Integrated flow and transport modelling.

The project was finished and concluded in 1995, and the final report consists of 6 parts:

- Volume 1. Acquirement of the natural analogue program /16.5-10/.
- Volume 2. Part 1. The fission reactors and the geochemical systems. Source term: isotopic characterisation, nuclear parameters and modelling /16.5-11/.
- Volume 2. Part 2. The fission reactors and the geochemical systems. Geology of the reactor zones, study of the walls and of the ancient transfers /16.5-12/.
- Volume 3. Part 1. Characterisation and modelling of the far-field migrations from the reactor zones on the Okelobondo and Bangombé sites. Synthesis of the hydrogeological investigations /16.5-13/.
- Volume 3. Part 2. Characterisation and modelling of the far-field migrations from the reactor zones on the Okelobondo and Bangombé sites. Synthesis of the hydrogeochemical investigations /16.5-14/.
- Volume 3. Part 3. Characterisation and modelling of the far-field migrations from the reactor zones on the Okelobondo and Bangombé sites. Modelling of the flow and solute transport /16.5-15/.

Volume 1 is a summery of the project, written by the project manager (P-L Blanc). Volume 2, which consists of two separate parts, describes the reactors, their geological characteristics, mineralogical and isotopical composition. The study concentrated on reactor zones 10 and 13 at the Oklo mine and the near surface reactor zone at Bangombé. Volume 3 summarises the hydrogeological and hydrogeochemical investigations, and the modelling of hydraulic flow and solute transport. Some points of particular interest in relation to spent fuel disposal can be mentioned:

- Evidence of water radiolysis have been found in the form of free H_2 and O_2 in fluid inclusions. Despite the fact that oxygen was generated the overall environment remained reducing as shown by the presence of metallic lead, galena and coffinite.
- Metallic inclusions, consisting of platinum elements and containing all of the technetium, are generated in spent nuclear fuel. The metallic inclusions in Oklo are still there which indicate an excellent isolating capability.
- Dissolved plutonium has been trapped in the clay (chlorite) surrounding the reactor zones.
- The hydraulic flow has been characterised by a combination of hydraulic measurements and hydrochemical sampling and analysis. This worked particularly well at the relatively undisturbed Bangombé site. Environmental tracers such as tritium and oxygen-18 proved to be very valuable.
- Coupled model calculations of geochemistry/mass-transport were applied and tested on the observed redox front at Bangombé (the STELE-code).
- Colloids, dissolved organic matter and microbes were successfully sampled in the groundwater and analyzed.

A second phase has been planned and proposed to the EU for support. The second phase will start in early 1996 and continue for three years. CEA will also manage the second phase and SKB has applied to participate, now as a full member together with other interested parties.

16.5.3 Palmottu

The uranium mineralization at Palmottu Lake in Finland 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 in the late 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 was used as an analogue to a deep repository for spent fuel in granitic rock /16.5-16/. The geology, hydrogeology, groundwater chemistry, climate etc are similar to that expected at the sites chosen for deep disposal of in Finland and Sweden. SKB has so far participated in the project as "an active observer". However, the continuation of the Palmottu project has been proposed to EU and accepted for their support. The new Palmottu project is managed by GTK and SKB participates as full member together with ENRESA and BRGM. It started at the end of 1995 and will continue until 1999. The goals of the new Palmottu project are as follows:

- Provide 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.

The new Palmottu project is divided up in two parts. The first part is a careful characterisation of the site including the drilling of an additional hole, much deeper than any of the existing prospection holes and with a large enough diameter for the instruments required. The continuation of the project will depend on the outcome of the first part.

16.5.4 Jordan

Water from the hyperalkaline springs in Maqarin can have pH-values as high as up to 13. These alkaline springs in Maqarin are still active but there are also remnants from the previously existing hyperalkaline springs in central Jordan. The cause of hyperalkalinity is a series of reactions involving the spontaneous combustion of bituminous marlstone. Events causing air to intrude into the layers of marlstone starts an oxidation of pyrite. The temperature increases as a result of that, ignites the organic matter and a combustion zone is created, which generates portlandite and reactive calcium silicates in the marl. When water seeps in it obtains a pH in the range 12-13 and a whole series of typical cement minerals are formed by the reaction between water and minerals; see Figure 16.5-1. The reaction zones, their surroundings and the water are being studied as an analogue to the use of concrete in a repository. Several long time aspects of cement have been addressed such as longevity of concrete, influence of high pH-water, colloid formation, microbial activity at high pH, solubility of radionuclides in hyperalkaline water etc; see Table 16.5-2.

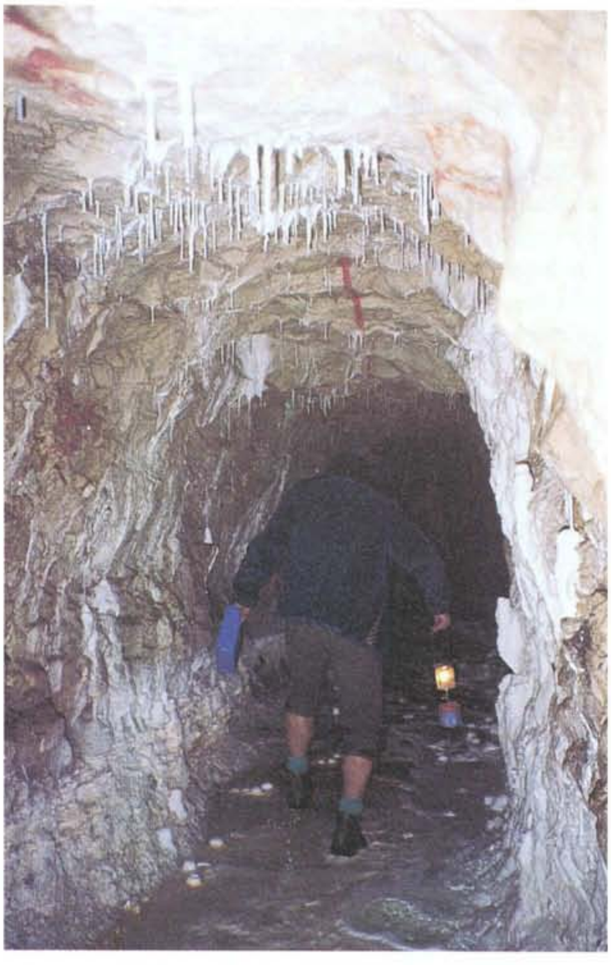

Figure 16.5-1. *The Maqarin site in Jordan. Adit A-6, a tunnel used for sampling of hyperalkaline groundwater and minerals. The secondary deposits is a mixture of portlandite and carbonate.*

Table 16.5-2. Reactions of interest in the cement analogues of Jordan.

REACTANTS
Bituminous marl and pyrite

SPONTANEOUS IGNITION
$FeS_2 + O_2 \rightarrow$ Ignition

BURNING OF BITUMEN (a kerogenic organic matter)
Bitumen + $O_2 \rightarrow CO_2$ + Heat (ca 1000°C)

FORMATION OF "CEMENT CLINKER" CS_x (*C*alcium *S*ilica)
Heat + Marl (limestone with clay) $\rightarrow CS_x + CaO$

HYDRATION OF "CEMENT CLINKER" (CS_x) AND BURNT LIME (CaO)
$CS_x + H_2O \rightarrow$ *CSH*-phases (*C*alcium *S*ilica *H*ydrate)
$CaO + H_2O \rightarrow$ Portlandite ($Ca(OH)_2$)

FORMATION OF HYPERALKALINE WATER
$Ca(OH)_2 \rightarrow Ca^{2+} + 2OH^-$ (pH = 12.5)

HYDROLYSIS REACTIONS
Minerals + $OH^- \rightarrow$ Secondary minerals (e.g. *CSH*-phases)
Organics + $OH^- \rightarrow$?

FORMATION OF COLLOIDS?
CSH-phases $\rightarrow$ Colloids?

SOLUBILITY AT HIGH pH
High $[OH^-]$ and low $[CO_3^{2-}]$ (compared to normal g. w.)
Sn, Se, Ni, Pb, Ra, Th and U

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 involved testing (validation) of chemical data and codes for calculating solubility of radionuclides. This is not easy due to the high pH of the water. There are plenty of minerals in Maqarin, and one can analyze how much of the elements Sn, Se, Ni, Pb, Ra, Th and U has dissolved in the groundwater /16.5-17 and 16.5-18/. 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.

16.5.5 Old concrete

Portland cement was introduced at the start of this century. In other words it is not very old, but it should be possible to observe trends of change in the samples of old concrete structures. It should also be possible to connect these observations with the study of the much older mineral samples from Maqarin and central Jordan. Therefore, whenever possible, an effort has been made to find old concrete samples that have been in a water-saturated environment. If the samples have been in the air, carbonatization dominates, caused by reactions with carbon dioxide from the air. The investigation of old concrete structures has not been finished yet but six different locations in Sweden have been sampled:

- Sillre/Oxsjön hydropower dam.
- School building in Gävle.
- Midskogsforsen hydropower plant.
- Rocksta mill.

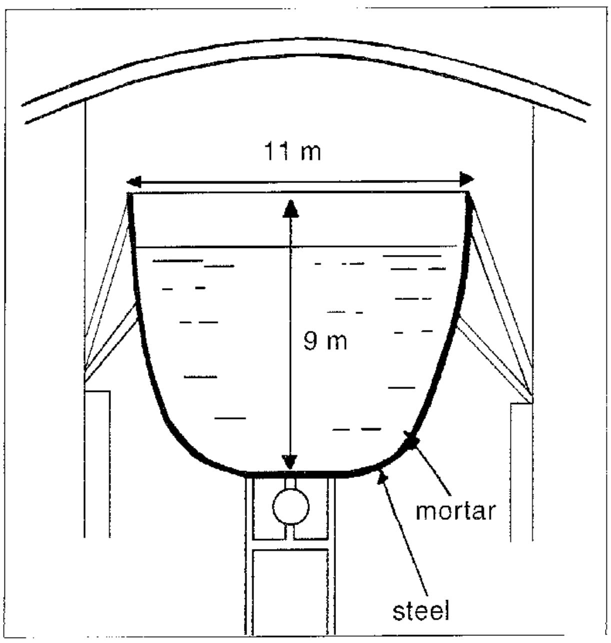

Figure 16.5-2. Sketch of a water tank built 1906 inside one of the towers of Uppsala castle. The water tank is covered inside with a 20 mm thick layer of mortar.

- Älvkarleby hydropower plant.
- Water tank in Uppsala castle; see Figure 16.5-2.

The results so far have been summarised in an interim report /16.5-19/. Investigations of old concrete constructions have also been reported in a review of concrete properties /16.5-20/.

16.6 BIOSPHERE

This subject treats the transport of nuclides from the aquifers above the bedrock, through natural and domestic ecological systems and into different foodstuffs. As an normal endpoint, the dose to man is calculated and commonly compared to regulative limits. Dose to (or effect on) biota other than man is also considered.

In short, the contaminants reaching the biosphere are considered to

- enter primary receptors,
- be transported in ecological systems, possibly causing external dose,
- accumulate in foodstuffs like plants animals and fish,
- be consumed and cause internal dose.

The modelling of processes in receptors and ecosystems starts with the outflow of dissolved radionuclides from the bedrock to an aquifer. This aquifer feeds contaminated water into different receptors (well, river, lake) and the contaminants can be transported through different physical compartments like soils, sediments, waters and air. The processes considered include chemical/physical activity such as sedimentation, biological activity such as bioturbation, and human activity such as farming.

This part of the modelling is done with compartment models for which volumes and transfer factors between compartments have to be calculated. Solving the (linear) differential equations produces time dependant concentrations for the different compartments.

After this the accumulation in foodstuffs, intake and dose calculations are pure multiplication for each scenario.

16.6.1 Validation of models

The vast number of transport processes involved, can be rationally treated with compartment models where several processes are put together into one transfer rate. Such models have been extensively used in this area since the 70-ties. General validation of such complex models is really not possible, but a "validation document" has been prepared, describing conceptual (processes) and numerical modelling /16.6-1/ and some attempts to determine a justified area of application for some models have been made in BIOMOVS /16.6-2/, VAMP /16.6-3/. The numerical performance of a code is easier to verify, and has been done in PSAC /16.6-4/.

An overview of the uncertainties in biosphere modelling /16.6-5/ demonstrated that the sources of uncertainty are generally many and much depending on the scenario. The confidence intervals normally span several orders of magnitude.

BIOMOVS

The BIOspheric MOdel Validation Study is an international cooperative study initiated in 1985 to test models designed to calculate the environmental transfer and bioaccumulation of radionuclides and other trace substances. To SKB this has been an opportunity to test the widely used modelling tool BIOPATH and the uncertainty tool PRISM in several applications. The first study was run for five years and ended in 1990. BIOMOVS I forcefully demonstrated the shortcomings of our present capabilities for biosphere modelling /16.6-2/.

In 1991 the second phase, BIOMOVS II, was started, jointly managed by the five organisations AECB, AECL, CIEMAT, ENRESA and SSI. SKB has put emphasis on the theme "Reference Biospheres", as it is of great value to get an international consensus how to deal with the modelling and conceptual uncertainties arising with time. A general methodology and a list of FEPs (features events and processes) have been published. The RES method has proven to be a useful tool in demonstrating process interactions for different scenarios.

The proposed "reference" methodology and FEP list, is also used and tested in a related theme "Complementary studies" with emphasis on the processes involved in modelling. In a validation scenario for ^{14}C within the "Validation and Uncertainties" theme, we are stochastically testing the model used for the SFR safety assessment with almost all parameters as PDFs. Within the last theme SKB also takes part in a "Modeler interpretation" test, where both models and modelers are compared for a number of scenarios.

VAMP

SKB has been participating in an IAEA/CEC program "Validation of Models on the transfer of Radionuclides in Terrestrial, Urban and Aquatic Environment and Acquisition of Data for that Purpose" (VAMP). In the aquatic part of this programme, modelling of ^{137}Cs in lakes and uncertainty analysis has been intercompared between several working groups from several countries. The implications are that simpler models with site specific parameters reflecting retention time, give the best estimation /16.6-3/.

16.6.2 Dose factors

Land rise and gravel zones

As more interest is focused on the10 000 year time frame, land rise is one of the more predictable changes that will occur. Land rise will cause changes in the studied biosphere, as turning sediments into land. Connected to this is the role of gravel zones in the sediments. When the sea level drops, these gravel zones can play an important role in transporting ground water laterally. A literature survey has been initiated on these questions.

Forest ecosystems

As most of the land area of Sweden is forest, a study has been started to evaluate the potential dose-factors connected to pathways in a forest ecosystem and society.

Dose factors in the Äspö area

More realistic dose factors for the Äspö environment were calculated for seven nuclides; ^{14}C, ^{99}Tc, ^{129}I, ^{135}Cs ^{237}Np ^{240}PU and ^{241}Am. An approximately 100 km^2 big area west of Äspö was studied and six types of recipients could be identified. Using pathways identified at the site and current habits a set of dose factors with uncertainty intervals was calculated /16.6-6/. The volumes, currents and residence times in the straits around Äspö have also been estimated /16.6-7/. A reevaluation of these data are currently under way with the assumption that the release enters the biosphere via a major crush zone, thereby distributing the release to the different recipients.

16.6.3 Radionuclide transport

NATAN

One way of understanding long time transport processes in the biosphere is to study transport of natural occurring elements. In particular, sorption and migration of radionuclides in the interface between biosphere and geosphere is of special interest. An inventory of good candidate sites was produced 1993 /16.6-8/. A specific scenario was submitted for a validation exercise in BIOMOVS but not enough knowledge seams to be present to predict nuclide transfers in soils and sediments for this limited time span. The data have been compiled into a report to be reviewed within BIOMOVS.

The distribution of radionuclides in soils and sediments

The modelling of transport in soils and sediments have been heavily relying on the sorption assumption expressed as a single K_D-value. To deepen the knowledge about the theoretical background to K_D-values, available theoretical models for ion-exchange and surface-complexation have been adapted for biospheric conditions The results show that the work with surface complexion model for actinides increases the understanding of both laboratory measurements as well as studies of natural systems. The triple layer surface complexation model could estimate the dependence of K_D as a function of important chemical parameters such as pH and E_H.

The power of the surface complexation model is that equilibrium constants obtained under well laboratory conditions on well determined minerals easily can be used to estimate sorbtion under a much wider variety of conditions. K_D-value for Ra could be more precisely determined if the Ca concentration in the environment was known. The elements handled were Cs, Ra, Np, U and Pu /16.6-9/.

16.6.4 Effects on biota other than man

In the Radiation Protection Act from 1988 it is stated that man and nature should be protected from harmful effects of radiation. The need for consideration of protection of nature within the EIA process has been pointed out by both SSI and SKI. The effects on plants and animals can be summarised as

- Change in species diversity or number of individuals.
- Reduction of number of individuals of rare and threatened species.
- Introduction of new species or prevention normal regrowth.
- Reduction of agriculture or otherwise productive area.
- Degradation of habitat of existing species.

These effects are not likely to occur at acute doses below 0.1 Gy or doserates below 1 mGy/d for animals or 10 mGy/d for plants /16.6-10/.

A literature survey was completed during 1993 /16.6-11/ and is now be followed by an attempt to estimate the natural and seminatural levels of radionuclides in nature /16.6-12/. Estimating the doses that some species normally get and looking for effects may add in understanding the possible effect on ecological health.

17 OTHER LONG-LIVED WASTE THAN SPENT NUCLEAR FUEL

17.1 GENERAL

Long-lived LLW and ILW will, according to present plans, be disposed of in a separate part of the deep underground repository SFL. The waste consists of waste from research, medicine and industry, presently being handled and stored at Studsvik, and used reactor internal parts, including core components from the nuclear reactors to be stored at CLAB. Aside from a higher content of long-lived isotopes, typical long-lived LLW and ILW has a great deal in common with the SFR waste. In fact, some quantities of typical short-lived SFR-waste, like for example the operational waste from CLAB and the planned encapsulation plant (EP), may also be disposed of in the deep repository; that is in case SFR is closed before end of operation of CLAB/EP.

SKB has started investigations of long-lived LLW and ILW in Sweden. The goal 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 gathered on both existing waste and projected waste. The information required is about quantity and composition of the waste and how it has been packaged, or will be packaged in case it has not been produced yet.

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 repository design, research and safety assessment. We are following this international development work and informal cooperation has been established with organizations in France (ANDRA), Switzerland (NAGRA) and the UK (NIREX). In Sweden the preparation of the final safety assessment report for SFR and its complementary addition has contributed considerably to the knowledge of barrier performance and safety of an underground repository for low and intermediate level waste. However, relying solely on international development work in the field and on experience from SFR is not enough to prepare for a safety assessment in this case. It is necessary to carry out experiments that are not being done elsewhere and there are models that need to be adapted to deep repository conditions. In particular the experiments can take a considerable time to complete and need to be started in good time. This was a reason for the prestudy that was carried out in 1993-94 and reported early 1995 /17-1/. The prestudy included a performance assessment of the barriers to radionuclide dispersal based on the conceptual design presented in Plan 93 /17-2/. The results have guided the planning of further experiments and studies, and initiated a second phase. The goals of the second phase is to have basic data and models prepared for a future safety assessment.

17.2 THE WASTE

17.2.1 LLW and ILW from Studsvik

The waste from Studsvik consists of waste from research carried out at Studsvik and also waste collected from other users of radioactive materials in Sweden e.g. industry, universities and hospitals. The raw waste materials consist of activated and contaminated scrap metals, precipitates, ashes, ion exchange resins, glove boxes, radiation sources, laboratory outfit and radiation protection equipments. Concrete is used for conditioning, when necessary, and 200 l steel drums or concrete containers (1.2 m x 1.2 m x 1.2 m) are mainly used for packagings. Waste from Studsvik is divided up in two categories: waste suitable for disposal in SFR and waste for later disposal in the deep underground repository (SFL 3). All long-lived LLW and ILW is destined for SFL 3. Produced waste packages are stored in Studsvik and some of the short-lived waste has been sent to SFR in Forsmark. An inventory of the expected waste from Studsvik to SFL 3 was made within the frame of a prestudy /17-1/. Volumes, radionuclide content and the content of other materials (metals, plastics etc) were estimated /17-3/. A total volume of 1500 m^3 of waste packages from Studsvik to SFL 3 was calculated; see Table 17-1. Work is presently being made on an updated version of that inventory.

17.2.2 Operational waste from CLAB and EP

Operational waste from CLAB and later also from the encapsulation plant will be disposed of in SFR in Forsmark. The waste consists of ion exchange resins, filter aid and solid waste from service and maintenance. The waste is suitable for disposal in SFR and this will continue for as long as SFR is in operation. However, in case SFR is closed before the end of operation of CLAB and the encapsulation plant, this kind of waste will also be sent to the deep underground repository (SFL 5), although it is characterised as short-lived LLW and ILW. The ion exchange resins will be conditioned in concrete and the

Table 17-1. Characteristics of waste for SFL 3-5.

Characteristics	SFL 3	SFL 4	SFL 5
Total waste volume	**5000 m^3**	**10 000 m^3**	**10 000 m^3**
Waste origin and volume (outer volume of packages)	Studsvik LLW and ILW 1500 m^3	CLAB Cassettes 6600 m^3	BWR Core (reactor) components Internal parts (of reactor) 8000 m^3
	CLAB, Encapsulation Plant Operational waste 3500 m^3	CLAB, Encapsulation Plant Decommissioning waste 2200 m^3 Transport containers 1200 m^3	PWR Core (reactor) components Internal parts (of reactor) 2000 m^3
Packaging	Concrete mould (1.2x1.2x1.2 m) and steel drum (200 l)	Steel container (2.4x2.4x2.4 m)	Concrete container (1.2x1.2x4.8 m)
Activity content year 2040	$3 \cdot 10^{15}$ Bq	$2 \cdot 10^{14}$ Bq	$1 \cdot 10^{17}$ Bq
Dominating radionuclides	^{3}H ^{63}Ni ^{60}Co ^{137}Cs ^{239}Pu ^{241}Pu ^{241}Am	^{55}Fe ^{63}Ni ^{60}Co ^{125}Sb ^{137}Cs	^{63}Ni ^{60}Co ^{3}H ^{59}Ni ^{55}Fe ^{14}C

concrete moulds will have the same outer dimensions as the concrete containers for Studsvik waste (1.2 m x 1.2 m x 1.2 m).

Operation of CLAB includes receiving of spent fuel and core components from the nuclear power plants, repackaging and storage in water filled pools. Radioactive waste consists mainly of crud from the fuel receiving operations for spent fuel (cooling) and ion exchange resins from the cleaning of the storage pool water. Minor quantities of metal scrap and trash from maintenance and replacement of equipments are generated. It is anticipated that the encapsulation operation in the encapsulation plant and, later, the treatment of core components will give rise to the same type and amount of waste as the receiving operation in CLAB. An inventory of the expected waste production for SFL was made within the frame of a prestudy /17-1/. Volumes, radionuclide content and the content of other materials (metals, cellulose etc.) were estimated /17-3/. A total volume of 3500 m^3 of waste packages from CLAB and the encapsulation plant was calculated (SFL 3); see Table 17-1. Work is being made on an updated version of that inventory.

17.2.3 Core components and internal parts

Neutron irradiation induces a relatively high specific activity in the different structural materials in the central part of the reactor, close to the core or inside the core. The induced activity decreases with the distance from the core and at a few meters distance from the core the crud activity dominates. Waste, in form of disused metallic reactor core components and internal parts, is produced both during operation and decommissioning of nuclear power units. Most of it is composed of stainless steel, but there are also additional materials such as boron steel, boron carbide, hafnium, zircaloy, inconel and boron glass. Before the deep underground repository (SFL 5) is built, used reactor components are kept in interim storage at the power plants and in CLAB. Concrete containers have been suggested for packaging of the waste (1.2 m x 1.2 m x 4.8 m). An inventory of the expected waste production was made within the frame of a prestudy /17-1/. Volumes, radionuclide content and the content of different materials were estimated /17-3/. A total volume of 10 000 m^3 of waste packages containing core components and internal parts was calculated for disposal in SFL 5; see Table 17-1. Work is being made on an updated version of that inventory.

17.2.4 Decommissioning waste from CLAB and EP

Short-lived decommissioning waste from the power reactors will be disposed of in an extension of SFR (SFR 3). However, CLAB and the encapsulation plant will be decommissioned very late in the program, and hence the waste is allocated to the deep underground repository (SFL 4) although it consists of short-lived waste suitable for disposal in SFR. Decommissioning waste from CLAB and the encapsulation plant will consist of metals and concrete. The steel components are mainly from the cooling and clean-up system in the storage pool (tubes, pumps and tanks). The concrete is from the buildings, since concrete has to be chipped off from rooms potentially classified as contaminated. The waste packaging has not been finally designed yet but a provisional "unit vessel" (2.4 m x 2.4 m x 2.4 m) was anticipated and a total volume of waste packages of 2200 m^3 was estimated; see Table 17-1. Radionuclide content and the content of concrete and steel has been estimated in the prestudy /17-1 and 17-3/. The deep repository (SFL 4) will also receive some decommissioning waste from Studsvik.

17.2.5 Miscellaneous waste

The storage canisters from CLAB (stainless steel) will give rise to a total volume of 6600 m^3 of waste packages if they are to be disposed of in "unit vessels" with a side length of 2.4 m (SFL 4); see Table 17-1. The radionuclides in the storage canisters will mainly be associated with crud. An estimate of radionuclide content has been made in the prestudy /17-1 and 17-3/.

Transport casks and transport containers, used for transportation of high- and intermediate-level waste between the power plant, the interim storage and the final disposal site, may eventually have to be disposed of as waste (SFL 4). This would correspond to a total volume of 1200 m^3; see Table 17-1.

17.3 THE REPOSITORY

The repository parts for long-lived LLW and ILW, SFL 3-5, is planned to be situated about 1 km away from SFL 2 for spent nuclear fuel and at about the same depth. Transport tunnels will connect the two repository areas. The total excavated rock volume for SFL 3-5 will be about 110 000 m^3 and the total volume of waste is estimated to about 25 000 m^3, measured as total volume of waste packages /17-3/; see Table 17-1. SFL 3 and SFL 5 are situated in parallel rock caverns. Transport tunnels are connecting the vaults and will also be used for disposal (SFL 4); see for example Plan 93 /17-2/.

17.3.1 SFL 3

SFL 3 is mainly intended for long-lived waste from Studsvik (from research, medicine and industry). It shall also receive operational waste from CLAB and from the encapsulation plant. According to the conceptual design of Plan 93 /17-2/, the SFL 3 is an underground vault divided into a number of concrete compartments. The waste packages are emplaced in the concrete compartments and it has been discussed to backfill the voids with porous concrete. The outside of the concrete structure will be backfilled with bentonite clay and/or sand (backfill material has not been finally decided yet). The waste volume to be hosted in SFL 3 has been estimated to about 5000 m^3; see Table 17-1.

17.3.2 SFL 4

SFL 4 is intended for the decommissioning waste from CLAB and the encapsulation plant, the storage canisters from CLAB and the transport casks and transport containers. The intentions are to use the tunnel system remaining after the completed emplacement in SFL 3 and SFL 5. The tunnels used for disposal will be designated SFL 4. The walls are covered with shotcrete, the floor paved with concrete, sand backfill will be used and the total waste volume is estimated to 10 000 m^3; see Table 17-1.

17.3.3 SFL 5

SFL 5 shall receive core components and internal parts from the nuclear power reactors. Three repository vaults were foreseen in the conceptual design of Plan 93 /17-2/. The rock walls will be covered with shotcrete (ca 0.1 m) and floor covered with a layer of gravel (ca 0.3 m). The repository vaults are divided into compartments with concrete walls. The waste packages are to be disposed of in the compartments on a floor of concrete. A lid of concrete elements is placed on the top of the compartments after they have been filled with waste packages and the voids between the concrete structure and the rock will finally be backfilled with sand. The total volume of waste packages foreseen to be emplaced in SFL 5 is about 10 000 m^3; see Table 17-1.

17.3.4 Design

The design is not left at Plan 93 but further developed on the basis of experience gained by the assessment of barrier performance. A new design will presumably be ready and presented next year.

17.4 BASIC DATA AND MODELS FOR PA

17.4.1 Concrete and hydrochemistry

The state of knowledge in the area of concrete stability under repository conditions has been compiled /17-4/. Although modern concrete of the Portland type has not existed for very long there are good reasons to be more optimistic of its properties in performance assessment than before. There is in fact a tendency for old concrete to become better with age and, provided near stagnant groundwater conditions can be warranted in a repository, more faith can be put in the integrity of the concrete constructions. Further studies of old Portland concrete constructions have underlined this general conclusion /17-5/.

Leaching tests have been performed with crushed cement paste and water with a simulated groundwater composition /17-6/. Both normal granitic groundwater (Na-Ca-HCO_3) and deep saline groundwater (Na-Ca-Cl-SO_4) were used in the leach tests. The pH of the water was followed and the ions Na^+, K^+, Ca^{2+}, Mg^{2+}, OH^-, SO_4^{2-} and Cl^- analyzed together with Al_{tot} and Si_{tot}. Minerals were examined in the cement paste after leaching to see if secondary phases had formed. The results of the experiments are explained by model calculations and used to predict the chemical stability of concrete itself and the chemical environment created by concrete in a repository /Appendix C of the reference 17-1/. Similar studies have been carried out in, for example, the UK and Switzerland. A general conclusion is that concrete will control the near-field chemistry for as long as the waste is of any relevance to safety, provided that the groundwater flow is low.

17.4.2 Radionuclide chemistry

Laboratory investigations are carried out in order to produce the necessary data to be used in calculations of radionuclide retention. The work is long-term and mainly focused on radionuclide chemistry in the repository (solubility, sorption, diffusion, etc.) and the influence of organic substances. The following investigations are being conducted:

- Solubility measurements of Ni, Pu, Eu and Th in cement environment.
- Measurements of radionuclide sorption on concrete (Kd-values). The elements studied are: Th, Eu, Cm, Pm, Co, Ra and Ni (IV-, III- and II-valent metal ions).
- Measurements of radionuclide diffusion in cement paste and cement mortar. The elements studied are: Ni, Cs and tritium.
- Measurements of radionuclide diffusion through a mixture sand/bentonite (85% sand and 15% bentonite, Wyoming MX-80). The elements studied are: Cs, Tc and Ni. The influence of cement pore water on the diffusion is also being tested.
- Degradation of cellulose (both pure crystalline cellulose and industrial products like wood, paper, cotton, cement additives, etc.) at high pH of 10-13.5 and anoxic conditions. The degradation products are identified, for example the complex forming agent isosaccharinic acid, ISA.
- Complexation capacity of ISA, especially with III-valent metal ions.
- Measurements of the influence of ISA on sorption of radionuclides in cement. The elements to be studied are: Th, Pm and Ni (IV-, III- and II-valent metal ions).
- Measurements of the influence of ISA on diffusion of radionuclides in cement. The elements to be studied are: Pm, Ni and Cs.

Similar work is being performed in other countries. Experience has been gained, and time and resources saved by cooperation between ANDRA, NAGRA, NIREX and SKB on this and related subjects.

17.4.3 Hydraulic conditions

Hydraulic flow in the near-field of the repository is of importance for performance of the barriers and for the retention of radionuclides. Water will eventually reach the waste packages after closure and sealing of the repository. This can not be prevented, but the strategy is to prevent water flow in the central parts so that any mass-transport will have to occur by diffusion, which is a very slow process as compared to transport by advection. A combination of low conductive barriers (concrete constructions and backfill, bentonite layers, plugs, etc.) and draining backfill (sand and gravel) can direct the flow around, instead of through, the inner parts containing the waste. This has been demonstrated to work for SFL 3-5 in a recent study based on the design according to Plan 3 /17-7/. Implications of variations in the original design concept was investigated and the experience gained will be used to guide the outline of new versions.

According to the deep repository concept there is a minimum distance of 1 km between the repository for spent fuel, SFL 2, and the part SFL 3-5. The groundwater flow between the repository parts as a result of emplacement in relation to different topographies of the ground surface above the repository have been briefly evaluated. A general flow direction from SFL 2 towards SFL 3-5 is preferred and it should be possible to arrange it that way once a definite site have been selected /17-8/.

17.4.4 Near-field mass transport

The computer code NUCTRAN /17-9/ is used in performance assessment to calculate near-field release of radionuclides from an anticipated defect canister in SFL 2. In that scenario the nuclides escape from the canister through a small hole. Thereafter the nuclides migrate through the backfill material, through various pathways into flowing water of open rock fractures in the near-field. The transport mechanism of importance is diffusion; advective transport is limited to the rock surrounding the repository. In the model the different parts of the near-field are represented by compartments, such as the inner volume of the canister, the hole in the canister, the bentonite buffer, the tunnel backfilling, the water conducting fractures in the rock surrounding the deposition hole, etc. This is similar to a finite difference model for three dimensional problems, but considerable less cells or compartments are necessary and computing time is saved as a result of that. The code NUCTRAN has been further developed to include calculations of release from SFL 3-5. The capability of NUCTRAN to predict radionuclide release from SFL 3 and 4 (as defined in Plan 93) was tested by comparing with calculations by the code TRUMP, which uses the integral finite difference method /17-10/. Good agreement was obtained which demonstrates the capability of NUCTRAN to handle the release calculations, not only for the spent fuel repository SFL 2 but also for SFL 3-5 /17-11/; see Figures 17-1 and 17-2.

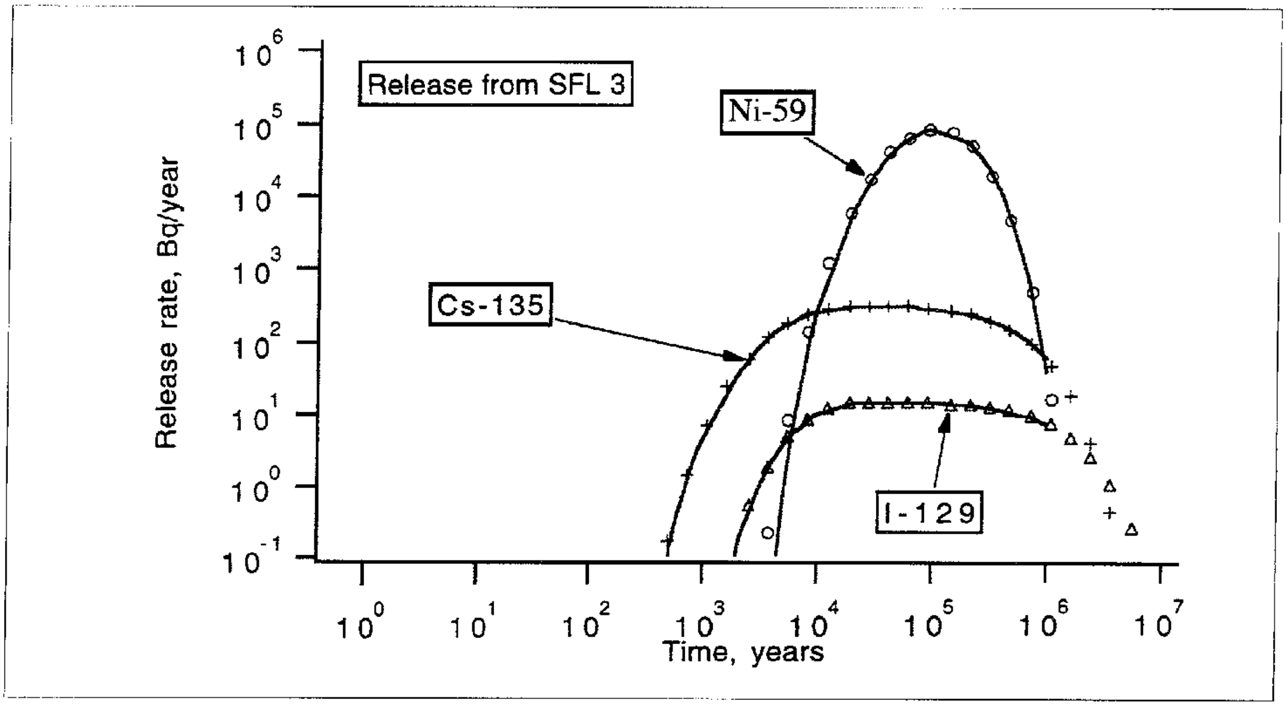

Figure 17-1. Calculated release from the near-field of SFL 3. The markers show the results obtained with NUCTRAN and the solid lines show the results obtained with TRUMP.

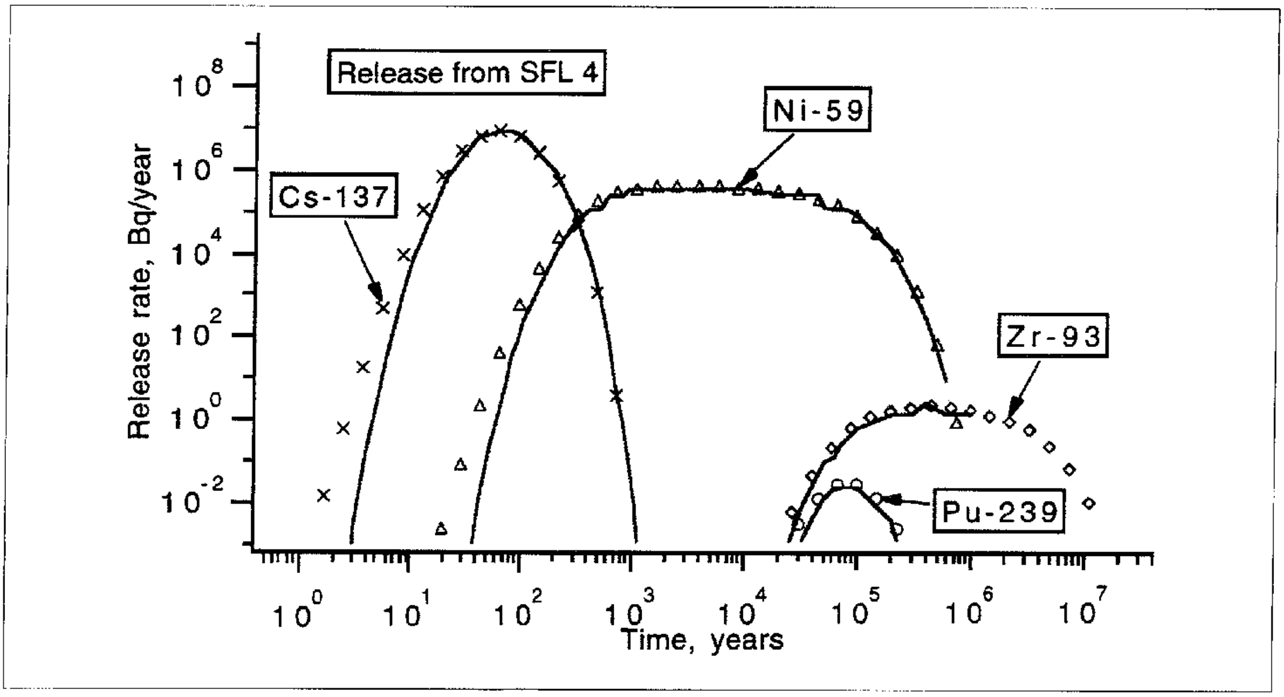

Figure 17-2. Calculated release from the near-field of SFL 4. The markers show the results obtained with NUCTRAN and the solid lines show the results obtained with TRUMP.

18 THE ÄSPÖ HARD ROCK LABORATORY

The Äspö Annual Report /18-1/ provides a detailed description of the achievements for 1995 and the reader is referred to this publication for further information.

18.1 BACKGROUND

The Äspö Hard Rock Laboratory constitutes an important part of SKB's work to design and construct a deep geological repository for spent fuel and to develop and test methods for characterization of a suitable site. In the R&D Programme of 1986 SKB proposed to construct an underground laboratory. A proposal that was positively received by the reviewing bodies. In the autumn of 1986, SKB initiated field work for the siting of an underground laboratory in the Simpevarp area in the municipality of Oskarshamn. At the end of 1988, SKB decided in principle to site the laboratory on southern Äspö about 2 km north of the Oskarshamn power station, see Figure 18-1. Construction of the Äspö Hard Rock Laboratory started on October 1st, 1990 after approval had been obtained from the authorities concerned. Excavation work was completed in February 1995.

The work within the Äspö Hard Rock Laboratory, HRL, has been divided into three phases; the pre-investigation, the construction, and the operating phase.

During the **Pre-investigation phase, 1986–1990**, studies were made to provide background material for the decision to locate the laboratory to a suitable site. The natural conditions of the bedrock were described and predictions made of geological, hydrogeological, geochemical etc. conditions to be observed during excavation of the laboratory. The investigations have been summarized in six Technical Reports /18-2 – 18-7/. This phase also included planning for the construction and operating phases.

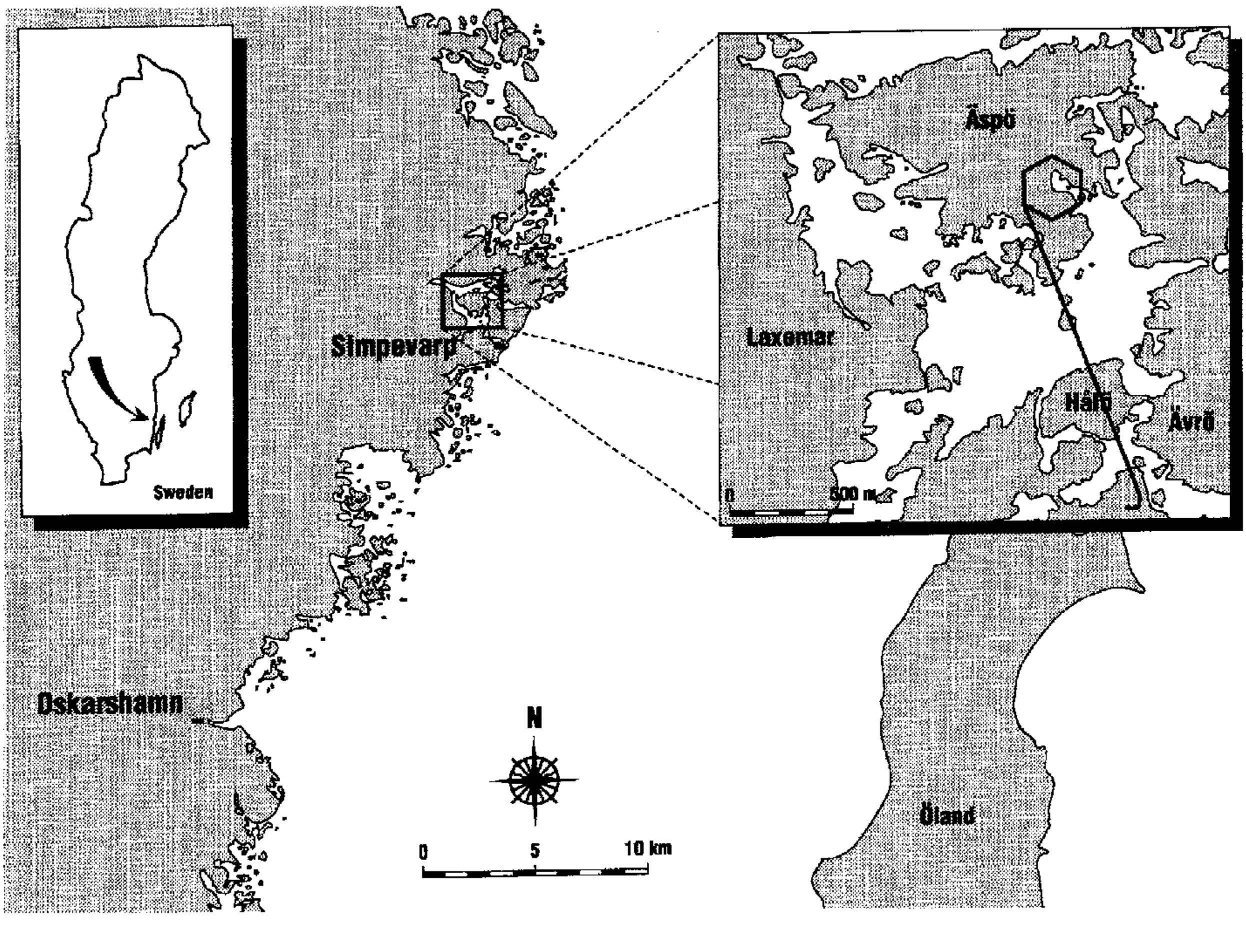

Figure 18-1. Location of the Äspö HRL.

During the **Construction phase, 1990–1995**, comprehensive investigations and experiments were performed in parallel with construction of the laboratory. The excavation of the main access tunnel to a depth of 450 m and the construction of the Äspö Research Village were completed.

The **Operating phase** began in 1995. A preliminary outline of the program for the Operating phase was given in SKB's Research, Development and Demonstration (RD&D) Programme 1992. Since then the program has been revised and the basis for the current program is described in SKB's RD&D Programme 1995 /18-8/.

18.2 ENGINEERING AND CONSTRUCTION WORK

The Äspö Hard Rock Laboratory has been designed to meet the needs of the research, development, and demonstration projects that are planned for the Operating Phase. The underground part of the laboratory consists of a tunnel from the Simpevarp peninsula to the southern part of Äspö where the tunnel continues in a spiral down to a depth of 450 m, see Figure 18-2. The total length of the tunnel is 3600 m where the last 400 m have been excavated by a tunnel boring machine (TBM) with a diameter of 5 m. The first part of the tunnel has been excavated by conventional drill and blast techniques. The underground tunnel is connected to the ground surface through a hoist shaft and two ventilation shafts. Äspö Research Village is located at the surface on the Äspö Island and it comprises office facilities, storage facilities, and machinery for hoist and ventilation, see Figure 18-3.

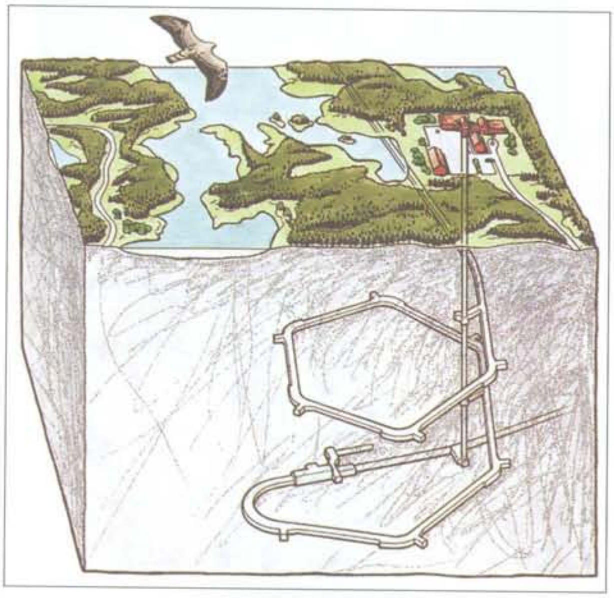

Figure 18-2. Schematic design of the Äspö HRL. The lower part of the facility has been excavated by a 5 m diameter Tunnel Boring Machine.

The rock excavation work on level -450/-460 m was completed by the contractor in the beginning of the year. During the spring the construction of pumping sumps and fire protection walls by the shafts was finished on levels -220 m, -340 m and -450/-460 m. The final inspection of the contractual work in construction phase 2 was carried out in the beginning of July. The work was accepted with a list items to be corrected. All items on this list were taken care of by the contractor in September.

The installation work and commissioning of the elevator have continued through the autumn 1995. The contractor reported the contractual work ready for final inspection in December. The inspector did not approve the installations and the inspection work will continue when the contractor has improved the functions and corrosion protection of installations.

The equipment for ventilation of the tunnels and caverns was earlier installed in the surface building. During the spring 1995 the installations in shafts and tunnels were carried out. The ventilation system was commissioned by the contractor during the summer and final inspection was carried out in August.

At level -220 m, -340 m and -450 m the permanent drainage system has been installed and a functional test has been carried out. The total system was commissioned in June and handed over to site organisation in July. In the beginning there were problems with the pumps and the tunnel was flooded. The control system for the pumps has been modified in order to prevent the same type of failure to happen again.

18.3 INVESTIGATIONS AND EXPERIMENTS – NEW RESULTS 1995

Stage goal 1 – Verification of pre-investigation methods

In order to get more detailed information of different kinds of fracture zones supplementary investigations have been performed in the Äspö tunnel after excavation during 1994-95. The predicted sub-vertical "NNW-structures" are in some cases possible to identify as "minor fracture zones" at depth. In other cases they are built up as a more or less complex system of conductive "fractures" trending NW to NS. A special study of gently dipping structures reveals only two narrow fracture zones. Gently dipping fractures occur as fracture swarms rather than zones.

Rock stress measurements have been undertaken at several occasions during the site investigation- and construction phases of the Äspö Hard Rock Laboratory. All stress data produced at Äspö to date have been summarized in /18-9/. Measurements reported comprise hydraulic fracturing and overcoring tests in three deep surface boreholes, and overcoring tests conducted in several short boreholes drilled from the ramp. Results show a rather

Figure 18-3. *Aerial view of the Äspö Research Village.*

consistent, NW-SE and sub-horizontal orientation of the maximum principal stress. In this direction, stress magnitudes are relatively high, reaching values of about 30 MPa at 499 meters depth. The intermediate- and minimum principal stresses are of considerably lower magnitudes, resulting in an anisotropic stress field and large shear stresses.

Borehole radar measurements using directional antenna have been performed in the Äspö HRL. The radar results have been compared to results from core and tunnel mapping /18-10/. The comparison shows that the dominating vertical WNW- to NW-striking fractures and waterbearing fractures have been detected by the borehole radar. The frequent sub-horizontal fractures and waterbearing fractures have also been detected by the radar as well as the vertical NNE- to NE-striking fractures. The result from the radar measurement also shows two subgroups of gently dipping WNW- and ENE-striking reflectors, which can not be considered as prominent fracture orientations. Correlation between reflectors in boreholes and structures in the tunnel were acceptable. Fractures, fracture zones and waterbearing fractures from tunnel mapping were correlated with radar reflectors. Correlation between radar reflectors and core mapping was considered good. The investigation shows that geological phenomena in the core were mainly indicated by radar reflectors and vice versa.

To estimate the groundwater recharge under natural and disturbed conditions is very difficult. During the 1995 a new method have been developed and tested on the Äspö data in order to see if it is possible to define a more robust boundary condition, than for example constant rate, that can handle both natural conditions and large disturbances due to for example inflow to a tunnel system. The method has been found to be simple and efficient and yields reasonable infiltration rates /18-11/.

Evaluation of obtained results for the final reporting of the Stage Goal "Verification of pre-investigation methods" has started. Three Technical Reports which summarize the results from the pre-investigation and construction phases will be published in 1996.

Stage goal 2 – Finalize detailed investigation methodology

The detailed characterization of a repository will encompass investigations during construction of shafts/tunnels to repository depth. Finalizing the detailed investigation methodology is Stage goal 2 of the Äspö HRL project.

Data collected in the cored test borehole KA3191F along the first 200 m extension of the TBM tunnel have been compared to data from the TBM tunnel /18-12/. The results show that it is possible to make a good prediction of the major rock types in a tunnel by use of a cored borehole like KA3191F. Minor dikes with an orientation more or less parallel to the borehole may of course be misjudged. Fracture data (frequency, fracture fill) is possible to predict while fracture orientation by use of TV-inspection is much dependent on the orientation of the borehole in relation to the orientation of the main fracture sets. There is normally a good correlation between geophysical logging data in the borehole and sections with increased fracturing/alteration of rock in the tunnel. It was also possible to identify the major conductive sections in the tunnel. However, minor conductive structures may not be seen as inflows in the cored borehole if they appear as single fractures which happen to be more or less closed at the intersection of the borehole. Radar reflectors and low RQD in the core do to some extent correspond to conductive parts of the rock. However, neither radar nor low RQD capture all waterbearing structures mapped in the tunnel or in the borehole.

To obtain a better understanding of the properties of the disturbed zone and its dependence on the method of excavation ANDRA, UK Nirex, and SKB have decided to perform a joint study of disturbed zone effects. The project is named ZEDEX (Zone of Excavation Disturbance EXperiment). The objectives of ZEDEX are:

- to understand the mechanical behaviour of the Excavation Disturbed Zone (EDZ) with respect to its origin, character, magnitude of property change, extent, and dependence on excavation method,
- to perform supporting studies to increase understanding of the hydraulic significance of the EDZ, and
- to test equipment and methodology for quantifying the EDZ.

The experiment is performed in two test drifts near the TBM Assembly hall at an approximate depth of 420 m below the ground surface. The TBM test drift constitutes part of the main access tunnel of the Äspö HRL, the test section is 35 m long and located directly after the TBM assembly hall. The first four test rounds in the D&B test drift were used for testing the "smooth blasting technique" based on low-shock explosives and the remaining five rounds were used for testing the effects of "normal blasting". The shape of the blasted drift was designed to be circular with a flat floor and the diameter (5 m) was designed to be about the same as the TBM drift. A number of boreholes were drilled axially and radially relative to the test drifts to assess the properties and extent of the EDZ.

The initial conditions in the rock mass have been characterized by several techniques including; tunnel mapping and core logging to determine geotechnical classification factors (Q, RMR), *in situ* seismic (P- and S-wave) velocity measurements and radar measurements. The rock mass response to excavation has been observed by mapping of induced fractures, multi point borehole extensometers (MPBX) and convergence measurements, laboratory testing on core samples, and by acceleration, vibration, seismic velocity, permeability and acoustic emission (AE) measurements.

The two test drifts were located in grey medium grained Äspö diorite with irregular sheets of red fine grained granite cutting the drifts at various locations. Borehole radar, seismic reflection and geotechnical mapping were all useful in delineating the features. The main fracture sets are all steeply dipping, striking NW and NE. Seismic tomographic imaging showed variability in velocity and attenuation interpreted to be associated with fracturing. The average P-wave velocity of about 6 km/s indicated a very good quality rock mass. Seismic anisotropy measurements showed good correlation with fracture orientations.

The magnitude of the main principal stress (σ_1) is estimated to be approximately 32 MPa at the ZEDEX site. The direction of σ is approximately NW and horizontal. The magnitudes of σ_2 and σ_3 are estimated to be 17 and 10 MPa, respectively.

Acoustic Emission (AE) monitoring was performed when TBM excavation was stopped overnight at 9 m, 15 m, 22 m and 25 m measured from the start of the TBM tunnel. In the D&B drift AE monitoring was undertaken after each blast round. The spatial distribution of AE events was similar for both drifts but the number of events recorded for the D&B drift excavation was about 10 times greater than for the TBM drift, see Figure 18-4. For rounds where blasting failed most AE events were located within the blasted yet in place rock, but there was some evidence that the damage also extended further into the walls than for successful blasts. Estimates of crack initiation stress based on the maximum differential stress ($\sigma_1 - \sigma_3$) at the AE event locations gave an average value of about 25 MPa. This low stress value suggests that the cracking may have been occurring in rock already damaged directly by the excavation process.

Generally, the measurements in the far-field (more than 2 m from the drift perimeter) showed no evidence of damage to the rock for any of the excavation techniques in this generally good quality rock mass at Äspö. MPBX displacement measurements showed that deformation in the far-field was predominantly elastic. Seismic measurements showed no measurable change in the far-field. The AE source locations showed that there was very sparse activity beyond about 2 m from the tunnel perimeter, but the distribution of events was similar for both TBM and D&B drifts at that distance.

Hydraulic pressure build-up tests performed before and after excavation in two of the boreholes parallel to the TBM drift showed a general decrease in permeability after the excavation. Hydraulic tests in the short radial boreholes in the TBM drift did not show any notable induced

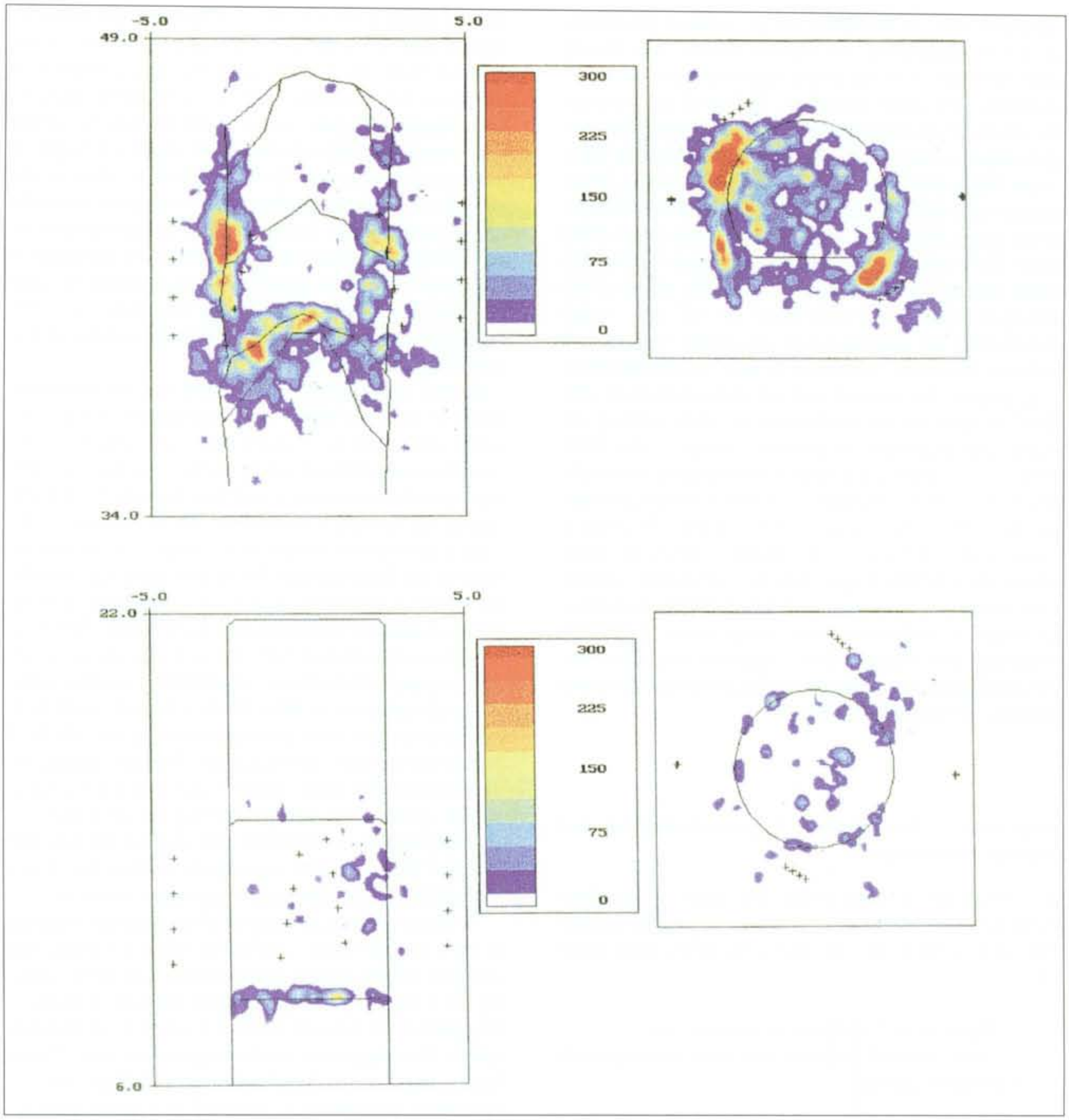

Figure 18-4. *Comparison of acoustic emission source location event density between the blasting and the TBM excavation. The upper figures show event density for monitoring after three successive blast rounds. Events recorded are displayed in both plan view and cross section looking toward the drift face. The lower figures show event density of events recorded over 3 successive nights of monitoring during excavation of the TBM drift.*

effect by excavation on matrix permeability. Around the D&B drift hydraulic properties in the boreholes parallel and radial to the drift generally show increased permeability after excavation even though both large increases and decreases were observed locally. Hydraulic tests in the short radial holes from the D&B drift showed clear evidence of increased matrix permeability in a zone in close proximity to the drift wall.

The results of the measurements made in the short radial boreholes to assess the near-field damage (permeability, seismic velocity, acoustic impedance and fracture mapping) showed a strong correlation between the different measurement types. From the measurements performed it is possible to conclude that the damaged zone formed by the excavation of the D&B drift was greater in degree and extent than that caused by excavating the TBM drift. This damaged zone is characterized by induced fracturing, increased permeability and reduced seismic velocities. It must, however, be borne in mind that relatively few measurements were made around the TBM drift to determine the extent of the damaged zone. The maximum measurable damage zone was approximately 80 cm in the floor of the D&B drift is probably a consequence of the larger charges used in the lifter holes for the rounds and the effect of the larger displacement caused by the flat shape of the floor. The extent of the damaged zone in the walls of the D&B drift was approximately 30 cm. The extent of the damaged zone was corroborated by AE data which showed that the majority of the AE events were located within a "thin skin", less than 1 m thick, around the drifts.

In general, the suite of methods employed in the ZEDEX Project has been effective in characterizing the extent and magnitude of property change in the EDZ. However, a significant problem in evaluating the results which have been obtained in a heterogeneous geologic setting is the relative scarcity of data points. To obtain a better statistical base to substantiate conclusions made above, the ZEDEX Project will be extended by another year. The project extension will include drilling and measurements in additional short radial holes, predictive modelling and extended data evaluations focused on understanding the crack initiation mechanisms and observed changes in hydraulic properties.

Stage goal 3 – Tests of models for groundwater flow and radionuclide migration

It is necessary to demonstrate the safety of the deep repository over long spans of time. Important phenomena that must be taken into account in the safety assessment are:

- transport of corrodants to the canister, and
- possible transport of radioactive materials away from a defective canister.

These phenomena are in turn highly dependent on groundwater flow and chemistry.

There are today several fundamentally different models for describing groundwater flow and radionuclide transport. Uncertainty exists regarding the accuracy, precision, and reliability of the models. This uncertainty includes conceptualization of heterogeneity, the ability to collect realistic data over an entire repository area, and the ability to carry out realistic calculations. It is important to test and demonstrate different approaches in practice in preparation for the licensing process.

The purpose of the Block Scale Redox Experiment was to investigate the chemical changes when oxidizing water is penetrating previously reducing fracture systems and to evaluate if complete flow paths can be oxidized from the surface to the repository. The experiment is now completed and reported /18-13/. It was found that the inflow of surface derived freshwater was reducing already at the depth of 70 m. The explanation to the rapid consumption of oxygen was the redox reactions mediated by bacteria. The content of organic matter in the infiltrating surface water was higher than the amount of dissolved oxygen. It took about twenty days for the bacteria to adjust to the changes of groundwater flow which took place when the tunnel penetrated the fracture zone. The bacterial processes continue even when the oxygen has been consumed in case there are organic matter left to be oxidized. In stead of reducing oxygen some bacteria can reduce iron (III) minerals as electron acceptors. Thus the content of dissolved ferrous iron increases.

Several experiments are planned for the Operating Phase of the Äspö HRL. These experiments require sites which meet specific requirements with respect to rock conditions and groundwater properties. A separate project was carried out which provided base data and recommendations for locating experiments /18-14/. Based on this work a provisional allocation was made of experimental sites for the Radionuclide Retention Experiment (RNR), the Redox Experiment on a local scale (REX) and the Tracer Retention Understanding Experiment (TRUE) at the experimental level (340–460 m level). Based on defined experimental criteria, experimental volumes tentatively allocated for the REX, RNR (Chemlab) and TRUE experiments have been investigated within the SELECT Project using eight cored boreholes. The rock sampled by the boreholes has been characterized with different geological, geophysical and hydrogeological techniques. As a complement to conventional core logging, the new BIP borehole TV system was successfully utilized. The results of the SELECT programme are presented in /18-15/.

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). 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.

The general objectives of the TRUE experiments /18-16/ are to

- develop the understanding of radionuclide migration and retention in fractured rock,
- evaluate to what extent concepts used in models are based on realistic descriptions of rock and if adequate data can be collected in site characterization,
- evaluate the usefulness and feasibility of different approaches to model radionuclide migration and retention.

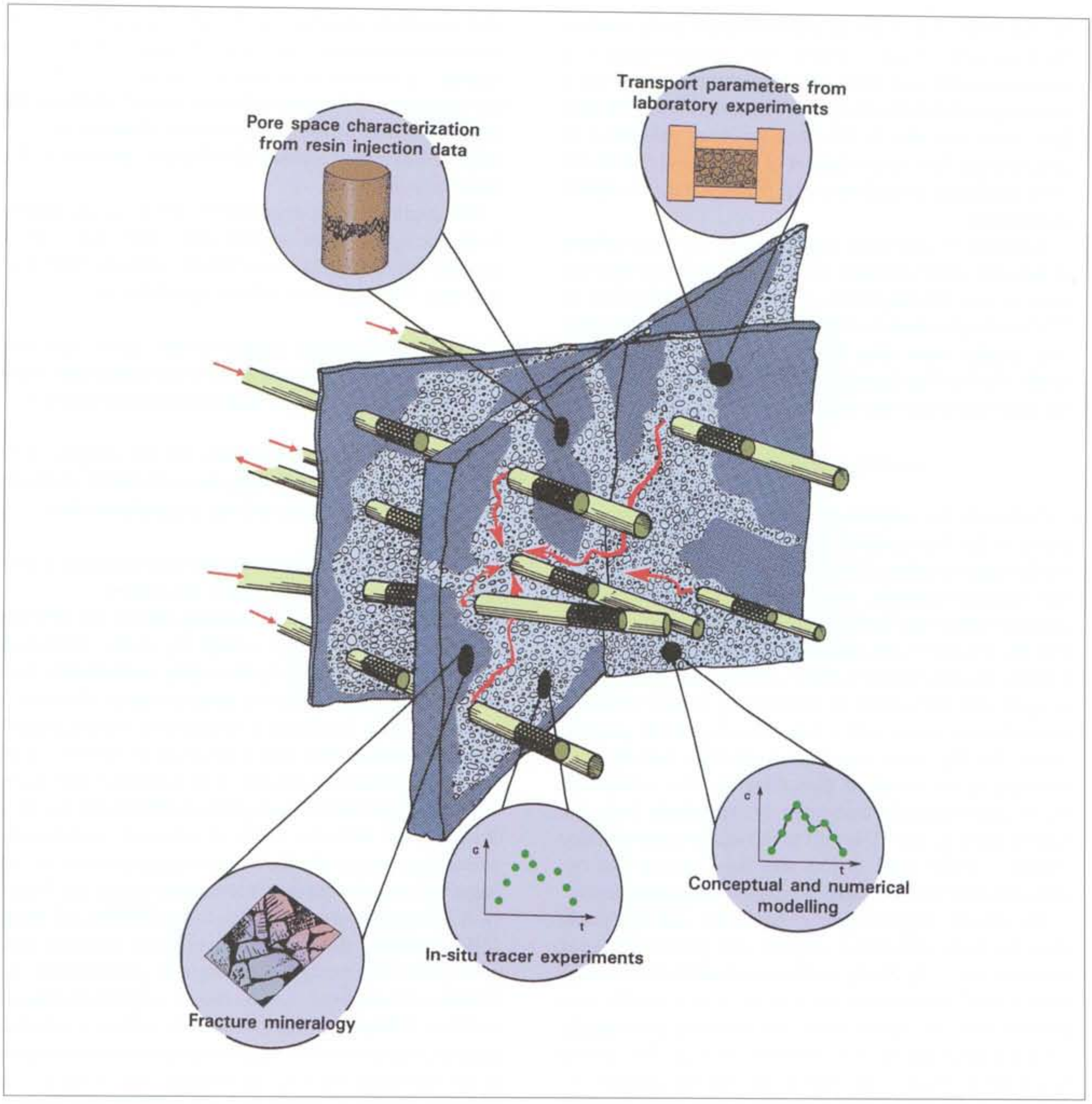

Figure 18-5. *Principal outline and components of the TRUE-1 experiment.*

The first tracer test cycle (TRUE-1) constitutes a training exercise for tracer testing technology on a detailed scale using non-reactive tracers in a simple test geometry, c.f. Figure 18-5. In addition, supporting technology development is performed for sampling and analysis techniques for matrix diffusion, and for understanding of tracer transport through detailed aperture distributions obtained from resin injection. The TRUE-1 cycle is expected to contribute data and experience which will constitute the necessary platform for subsequent more elaborate experiments within TRUE.

During 1995 work within the TRUE experiment has mainly been devoted to characterization of the site where the tracer experiments during the First TRUE Stage will be conducted, and development of resin injection technology.

The integrated analysis of the results of the characterization has been used to construct a descriptive structural-hydraulic model of the TRUE-1 Block on a Block scale (L~50 m) and on a Detailed scale (L~5–10 m). The developed models are the basis for predictions of the radially converging and dipole tracer tests planned for the

Spring 1996. One of the identified features at the site has been selected for tracer testing. The selected feature is a reactivated mylonite with one major fracture plane and a few subparallel minor fractures. The characterization data have been provided to the Task Force on modelling of groundwater flow and transport of solutes which will use it for predictive modelling of the planned tracer transport experiments.

A project of supporting laboratory test has been defined to develop and test weakly sorbing tracers before they are used *in situ*. Investigations of the sorption behaviour of ^{137}Cs to Äspö diorite have been finalized during the year. The results show that the K_d for Cs varies between 0.001–0.07 m^3/kg depending on particle size. The K_d for the different tracers are increasing in the order

$$Na < Ca \approx Sr < Rb \approx Ba < Cs.$$

Generally, the sorption is higher on Äspö diorite compared to the Fine-grained granite. This can be explained by the higher content of biotite in the Äspö diorite. Among the common naturally occurring minerals, biotite is the mineral which has the highest cation exchange capacity. For Na, Ca and Sr, the sorbed part can be easily desorbed. For Rb, Ba and especially for Cs, it has been observed that a significant part cannot be desorbed by simply replacing the old water phase with a non-spiked synthetic groundwater. The K_d-values vary with the particle size; decreasing particle size results in increasing K_d. The variation in K_d is approximately one order of magnitude from the largest particle size (2-4 mm) to the smallest particle size (0.045 – 0.090 mm). It has also been shown that the non-desorbed part increases with decreasing particle size.

The project Degassing of groundwater and two phase flow has been initiated to improve our understanding of observations of hydraulic conditions made in drifts, interpretation of experiments performed close to drifts, and performance of buffer mass and backfill, particularly during emplacement and repository closure. The in-situ test program began with a pilot test with the objective to get data on the magnitude of degassing effects on permeability, time scales required for resaturation, and requirements on equipment for subsequent tests. The test showed no two-phase flow effects /18-17/. Subsequent measurements indicated that the gas contents of KA2512A (0.5% v/v) was probably too low to cause two-phase flow effects. The results show no evidence of other processes that might reduce transmissivity at low borehole pressures, such as calcite precipitation, increase in effective stress, or turbulence.

Laboratory experiments have been carried out at Lawrence Berkeley National Laboratory (LBNL) during 1995 to investigate the effect of gas contents of similar low range as observed in the pilot hole field test on flow reductions due to degassing. Greater flowrate reductions were observed for greater gas contents, however, for relatively high gas contents, no further decrease in steady state flowrates and increase in gas saturations were observed for increasing gas contents. In addition to water gas contents, a number of parameters are of importance for the magnitude of flowrate reduction. Orders of magnitude differences in flowrate reduction were observed for fractures with different aperture distribution, aperture widths and inlet conditions.

The detailed scale experiment (REX) is planned to focus the question of oxygen and other redox active material that is trapped in the tunnels when the repository is closed. The objectives of the experiment are:

- How does oxygen trapped in the closed repository react with the rock minerals in the tunnel and deposition holes and in the water conducting fractures?
- How long time will it take for the oxygen to be consumed and how far into the rock matrix and water conducting fractures will the oxygen penetrate?

A test plan for the experiment has been prepared and the laboratory part of the experiment has started.

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. A special borehole probe, CHEMLAB, has been designed for different kinds of retention experiments where data can be obtained representative for the *in situ* properties of groundwater at repository depth, see Figure 18-6. The results of experiments in the CHEMLAB probe will be used 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 buffer material, transport out of a damaged canister and transport in an individual fracture. In addition, the influence of naturally reducing conditions on solubility and sorption of radionuclides will be tested. The CHEMLAB probe is currently being manufactured and will be put into use in 1996.

A "Task Force" with representatives of the project's international participants has been formed. The Task Force is a forum for the organizations supporting the Äspö Hard Rock Laboratory Project to interact in the area of conceptual and numerical modelling of groundwater flow and solute transport in fractured rock. The long term strategy of the Äspö TF has been discussed within the Äspö International Joint Committee (IJC). It was concluded that the work in the TF should be tied to the experimental work performed at the Äspö HRL. Furthermore, the work should be performed within the framework of well defined and focused Modelling Tasks. The TF group should attempt to evaluate different concepts and model-

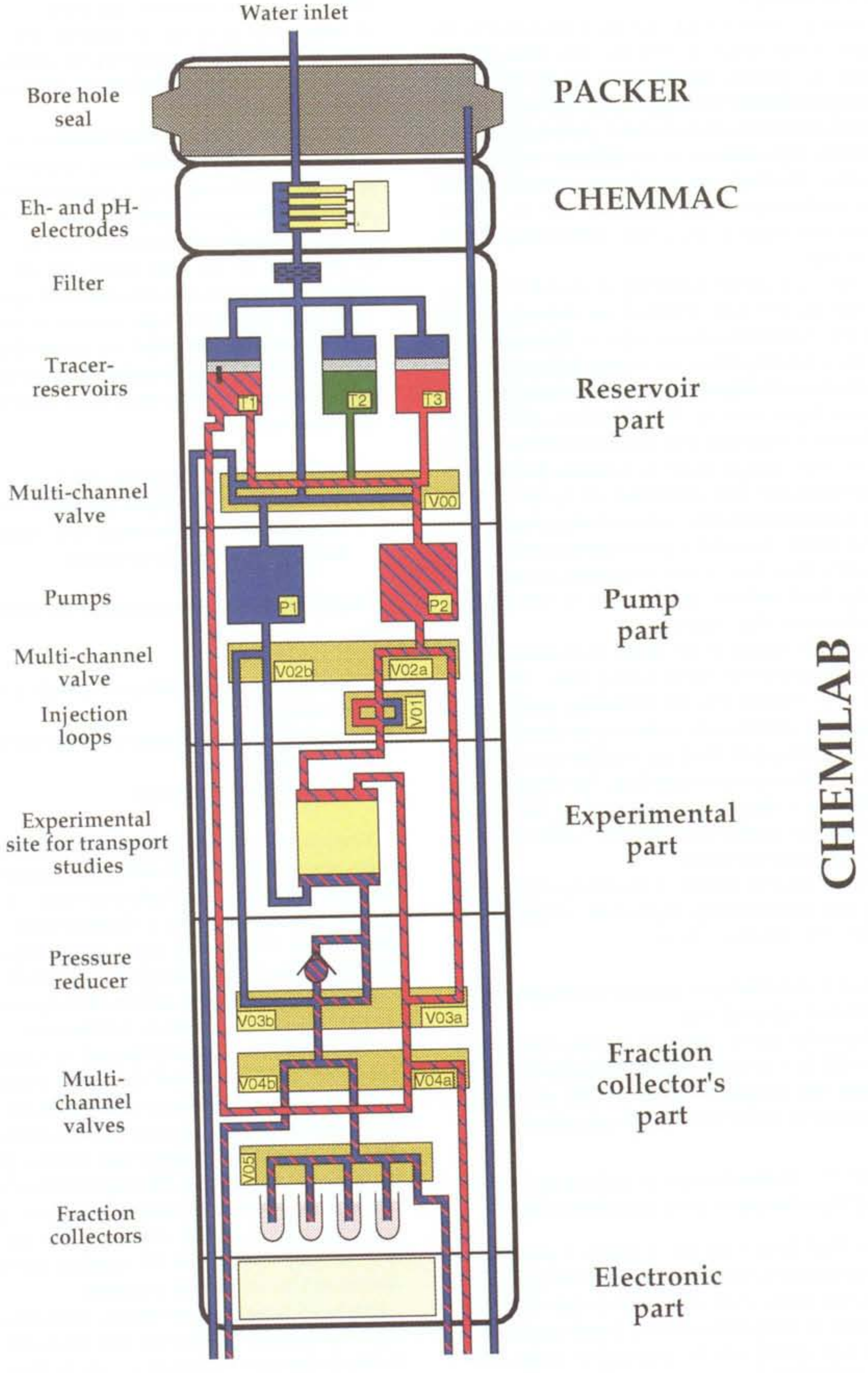

Figure 18-6. *Schematic illustration of the CHEMLAB probe.*

ling approaches. Finally, the TF should provide advice on experimental design to the Project Teams, responsible for different experiments.

An Issue Evaluation Table has been compiled by the delegates of the Äspö TF, /18-18/. The table aims at reflecting our current understanding of the key issues related to performance of the geological barrier, availability of reliable data, how they can be or are being addressed by different organizations or at different underground laboratories. The Table will also provide valuable help in relating performance assessment as well as characterisation key issues to the actual, forthcoming experiments at Äspö.

Task No 1 concerned modelling of the LPT2 experiments and has now been finalized. An evaluation report on Task No 1 has been written, /18-19/. Eleven different groups have modelled Task No 1 using different conceptual and numerical methodologies for simulating flow and transport in fractured rocks. With respect to groundwater flow, all models represented the measured LPT2 data well. Therefore, the capacity exists to perform three-dimensional groundwater flow modelling on a site scale. In general, the data supplied for Äspö, including the geologic structural model, provided a good representation of the real system. However, a few consistent errors in the modelling work indicate minor errors in the geologic structural model of the Äspö site.

The hydraulic impact of the tunnel excavation at Äspö HRL was defined as the 3rd Modelling Task. The objective will be to evaluate how the monitoring and the study of the hydraulic impact of the tunnel excavation may help for site characterisation. Task No 3 will be an exercise in forward as well as inverse modelling. The first part may be regarded as a direct continuation using the existing groundwater flow models from Task No 1. The modelling work is in the reporting phase.

Task No 4 concerns predictive modelling of the tracer retention and understanding experiment (TRUE-1) at Äspö HRL. The objectives are to:

- develop understanding of radionuclide migration and retention in fractured rock,
- evaluate what can be achieved with the existing data set of TRUE-1 in terms of transport predictions, and
- evaluate the usefulness and feasibility of different approaches to model radionuclide migration.

Stage goal 4 – Demonstration of technology for and function of important parts of the repository system

The Äspö Hard Rock Laboratory makes it possible to demonstrate and perform full scale tests of the function of different components of the repository system which 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 repository. Within this framework, a full-scale prototype of the deep repository will be built to simulate all steps in the deposition sequence. 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.

The Back Fill and Plug Test includes a series of tests of alternative backfill materials and emplacement methods and a test of a full scale plug. It is a test of different backfill materials and emplacement techniques. It will also be a test of the hydraulic and mechanic functions of a plug and be the basis for the plug design for the plugs in the prototype test. A test plan was made during 1995 /18-20/.

The field tests were made in order to test and further develop available technique for backfilling deposition tunnels. The field tests comprised the three main engineering activities required for backfilling tunnels in a repository. These activities are:

- crushing of rock for ballast material,
- mixing of ballast material, bentonite, and water
- transportation, emplacement, and compaction of the backfill in the deposition tunnels.

Furthermore, the tests comprised the following activities:

- development and testing of compaction tools and techniques,
- continuous measurements of results from the mixing and compaction,
- excavation of the backfill.

Emplacement and compaction of 5 different backfill types were made. Several different machines were used for the compaction. The horizontal layers in the bottom bed were compacted using a vibrating tamp roller with 4 tons weight. The inclined layers were compacted with a vibrating plate with the weight 700 kg, which was rebuilt for electrical power drive and for allowing compaction of inclined layers close to the roof of the tunnel.

About 1050 tons of uncompacted or compacted TBM muck and about 1000 ton of compacted mixtures of bentonite and crushed TBM muck were placed and compacted in the tunnel. The final layout of the test is shown in Figure 18-7. The layout was changed due to water problems in the inner part of the tunnel. The large inflow of water made it impossible to compact horizontal layers, since water was accumulated on the surface of the top layer and destroyed the layer due to the resulting increased water content and the subsequent loosening.

Extensive sampling and density measurements were made both during compaction and excavation. The following preliminary conclusions could be drawn from the field tests:

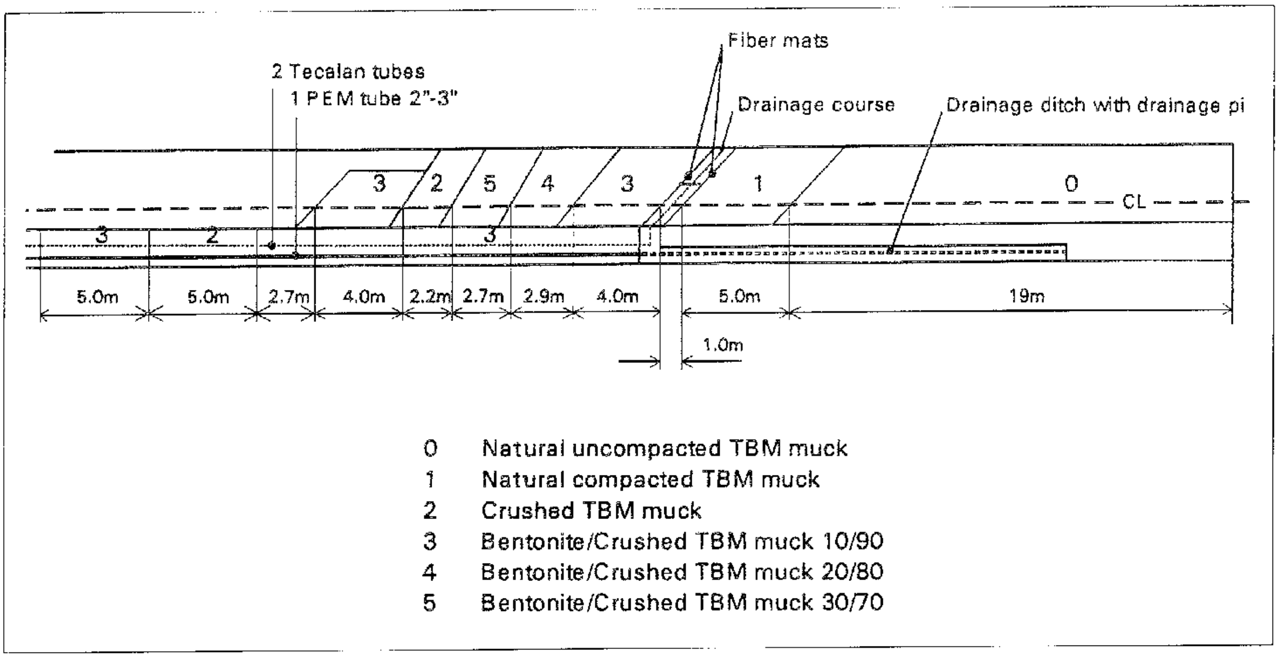

Figure 18-7. *Final layout of the compaction test.*

- The procedures to crush TBM muck to a desired grain size distribution and to mix bentonite, crushed rock, and water were successful.
- The technique to compact horizontal layers yielded unsolvable problems in wet areas.
- The technique to compact inclined layers was successful except at the zone close to the rock.
- Very wet areas may require special treatments.
- The excavation was successful

Data management

One of the main objectives with the Äspö Hard Rock Laboratory is to test, develop and improve techniques before they are applied at the candidate sites. In this context efficient techniques are required to handle, interpret and archive the huge amount of data collected during a site characterization process.

The first investigation database, GEOTAB, was already set up by SKB during the eighties.The aim of setting up this database was to preserve all data from the KBS-3 investigations and the pre-investigations at Äspö. During the construction phase of the Äspö Hard Rock Laboratory new needs led to the development of the SADB (Site Activity Database) which included a complete event list describing all performed measurements and engineering activities. Directly after the introduction of SADB a discussion started concerning the possibilities to combine the concepts of GEOTAB and SADB. This discussion resulted in a decision to develop a new database by combining the concepts of GEOTAB and SADB. A project was defined and the work started in June 1994. The first version of SICADA was put into operation during 1995. The name has been composed by in sequence put together the bold characters in the descriptive text "The SKB **Si**te **C**haracterization **Da**tabase System".

A three dimensional rock model is built by successive collection, processing and interpretation of site data. All site data will be stored in SICADA. Furthermore all geological and geophysical maps will be available in SKB's GIS database. Some of these maps need to be transferred to the visualization system. The experiences obtained from the investigations at Äspö Hard Rock Laboratory have shown that it is very important to have the possibility to test interactively in 3D different possible connections between observations in boreholes and on the ground surface. By effectively visualizing the rock model, based on available site data, it is possible to optimize new investigation efforts. During the design of the Deep Repository the rock model will be the basis for optimization of the tunnel layout. As several research groups will work with data from one site it is important to have only one "certified" visualization system. To meet these needs SKB decided to develop a the Rock Visualization System (RVS) in November 1994. A detailed system specification was completed in December 1995. Realization of the system will be made during 1996.

18.4 INTERNATIONAL PARTICIPATION

The construction of the Äspö Hard Rock Laboratory (HRL) has attracted significant international attention. The experience being gained at Äspö concerning, for instance, site investigation methodology, rock excavation, measurement techniques, and collection of data of importance to safety assessments, will be of interest to most countries that have their own plans for deep geological disposal of nuclear waste. SKB is open to and welcomes international participation in the Äspö HRL. Presently (March 1996) nine organizations from eight countries participate. They are:

- Atomic Energy of Canada Limited (AECL), Canada.
- Teollisuuden Voima Oy (TVO), Finland.
- Agence National pour la Gestion des Dechets Radioactifs (ANDRA), France.
- The Power Reactor and Nuclear Fuel Development Co. (PNC), Japan.
- The Central Research Institute of the Electric Power Industry (CRIEPI), Japan.
- United Kingdom Nirex Limited (Nirex), United Kingdom.
- United States Department of Energy (USDOE), USA.
- National Cooperative for the Disposal of Radioactive Waste (NAGRA), Switzerland.
- Bundesministerium für Bildung, Wissenschaft, Forschung und Technologie (BMBF), Germany.

19 ALTERNATIVE METHODS

The main direction of the SKB RD&D-programme is towards completing the first step with deposition of some 10% of the spent fuel in a repository within about 20 years time. In parallel the work on alternative treatment and disposal methods is followed in a limited scale.

During the early 1990s the possibility for partitioning and transmutation has attracted renewed interest. SKB supports work in this area at the Royal Institute of Technology (KTH) in Stockholm and at the Chalmers Institute of Technology (CTH) at Göteborg. The work at KTH is emphasized on safety related issues and at CTH on processes for partitioning.

SKB is also carrying out further research work related to the disposal in very deep boreholes. A brief account of the work conducted on the alternative methods during 1995 is given in the following sections.

19.1 KTH WORK ON TRANSMUTATION

The research activities 1995 were concentrated on the analysis of different acceleratordriven systems from the Swedish nuclear energy perspective. A very condensed review of different conceptual designs of accelerator-driven systems, their most important disadvantages and the feasibility of these systems from the Swedish perspective is presented in Table 19-1. It was concluded that the thermal neutron system with a liquid fuel is the most interesting.

Thus the neutronics and performance of a molten-salt graphite moderated accelerator- driven system was investigated in some detail. The system is characterized as follows:

> A graphite-moderated, thermal neutron system with the molten-salt fuel. Capability to burn plutonium, to transmute minor actinides, to transmute some of the fission products and finally to produce net energy based on U-Pu or Th-U fuel cycles – so called Los Alamos system.

The detailed 3-dimensional, reactor-physics model of such systems was implemented and a number of computer simulations were performed based on the Monte-Carlo technique. The simulations were focused on investigations of some important parameters of the systems such as: reactivity margins, temperature reactivity coefficients, power distribution etc.

19.2 CTH WORK ON PARTITIONING

19.2.1 Introduction

In 1993 the R&D project, Partitioning and Transmutation (P&T) was initiated at the Department of Nuclear Chemistry, CTH, with the objective to investigate separation processes in connection with transmutation of long-lived radionuclides in nuclear waste. Partitioning is a necessary stage when transmutation is considered. Furthermore, to make transmutation a successful operation, high separation efficiencies are needed to avoid losses of untransmuted material to different waste streams.

Different ways to achieve a sufficient separation yield are considered, namely aqueous based, molten salt based, pyroprocessing and ultra centrifuge separation techniques. Aqueous based separation is by far the most developed technique and a lot of experience is available in this field. In reprocessing of spent nuclear fuel solvent extraction has been utilized for more than 50 years. It is realistic to believe that the aqueous based separation technique is able to achieve the desired separation efficiency and therefore the research at the department is focused on aqueous based separation systems.

19.2.2 Solvent extraction research

Extraction reagents intended for partitioning must show a selective extraction, have a high loading capacity and possess a high stability towards radiolysis. A common demand on all new extraction reagents is that they must obey the CHON principle, i.e. they must contain only carbon, hydrogen, oxygen and nitrogen since the extractants should be completely incinerable and not produce any secondary waste. One group of extraction reagents that fulfils these criteria is the long chain quaternary ammonium salts, for example Aliquat-336. This is a liquid anion exchanger which has a high selectivity for tetravalent elements e.g. neptunium and plutonium and also for originally negatively charged species like pertechnetate ions.

The minor actinides, mainly americium and curium, are also intended for transmutation and must be separated from the other elements. One of the problems in partitioning is the separation of trivalent actinides from the lanthanides since their chemical behavior is almost iden-

Table 19-1. Review of different conceptual designs of accelerator-driven systems (by KTH group).

ADS SYSTEM	TECHNICAL DISADVANTAGES	FEASIBILITY FROM "SWEDISH" PERSPECTIVE
Solid fuel and thermal neutrons	Flux/power density distribution make system unfeasible. Serious safety concerns with "beam-on" power transients (e.g. unprotected loss of flow accidents)	Similar to LWR. Subcriticality advantages important but technically unfeasible.
Aqueous system with liquid/quasi-liquid fuel (suspensions slurries)	Pressurized system can not meet the "blow-down" criterium. Low pressure drives economy to the bottom. Radiolysis of water creates serious problems in the case of solutions. Erosion problems in the case of slurries.	Too risky, no technology developed, very difficult defense in depth - safety barriers.
Solid fuel and fast neutrons	Very high inventory/burnup ratio (e.g. 3160/250 for the Japanese minor actinide burner). Liquid sodium cooling – with all the odds. Very powerful accelerator needed (BNL Phoenix) if electrical energy is to be produced. Huge heavy metal (Pb) reservoir (Rubbia's Energy Amplifier), very difficult maintenance problems. Radiation damage problems.	Low acceptance for fast neutron systems – but subcriticality can eventually balance it. No experience in fast reactor technology, sodium cooling questionable. Too large inventory. Is heavy metal (in large volume!) cooling acceptable environmentally?
Liquid fuel and fast neutrons. Fuel in chloride salt solution (Japanese concept) or solved in liquid Pb (preconceptual German, Russia and other ideas).	Even worse inventory/burnup ratio 4.5% (e.g. 5460/250 for Japanese minor actinide burner). Problems with chloride salts (corrosion). In liquid Pb system – liquid Pb technology not trivial.	Inventory equal almost to the total amount of MA in Swedish spent fuel after 30 year program. Other concerns as above.
Liquid fuel and thermal neutron spectrum (molten fluoride system)	Only limited experiences from Oak Ridge MSRE but technology exist. High neutron fluxes required to break actinides build-up. Kind of on-line reprocessing required.	Low inventory, low pressure system. Possibility for passive safety solutions. Possibility for physical separation. But – technology still unproven.

tical. Different strategies are considered for the separation; i.) coextraction of the actinides and the lanthanides in a first stage and then separation of the trivalent actinides and lanthanides in a second stage, ii.) oxidation of the trivalent actinides to a higher oxidation state whereafter the actinides are selectively separated or iii.) highly specific extraction reagents that preferentially extract all the actinides over the lanthanides. The optimal extractant should separate all actinides from the other elements in a few stages. Picolinamides are believed to have this ability but more studies are needed to verify their usefulness as selective actinide extractants. The main research is thus performed according to the first strategy named above and malonamides are possible extractants for this separation route /19-1/ For the separation of the actinides from the lanthanides in a second stage, TPTZ (2,4,6-tri-(2-pyridyl)-1,3,5-triazine) has been suggested.

Long chain quaternary amines

In 1993 a PhD student started to work with the objective to investigate the extraction of elements with Aliquat-336 from nitric acid media. This extraction reagent was earlier proposed by the LANL (Los Alamos National Laboratory) in their suggested aqueous based separation process. Extraction data for uranium, technetium, zirconium and niobium have been determined. An investigation of the behavior of neptunium in the Aliquat-336 - nitric acid system has started. Neptunium possesses several oxidation states in solution and therefore careful control of the oxidation state must be performed in order to achieve the correct distribution data. In the beginning neptunium is present in its pentavalent state and show a very poor extraction. For the reduction of Np(V) to Np(IV) hydrogen and a platinum black catalyst are used to avoid the use of reduction agents that may affect the distribution of neptunium in the extraction experiments.The oxidation state is spectrophotometrically controlled and therefore macro amounts of neptunium are used. The experiments are done in a glove box in the alpha laboratory. Preliminary results show that it is possible to extract neptunium in its tetravalent state. The experimental distribution data, /19-2/ and /19-3/, are then parameterized and inserted in a code for process calculations /19-4/. The number of steps and volume flow ratios needed in a counter current extraction battery to achieve a certain separation efficiency are then calculated /19-3/. The results from this process calculation will be presented at ISEC (International Solvent Extraction Conference) which will be held in Melbourne, Australia, March 1996.

Malonamides and TPTZ

The second project at our department involves another PhD student since 1994 and concerns nitrogen containing extractants such as malonamides and pyridine compounds. The extractants are considered to have a great potential in a partitioning and transmutation process. These extractants follow the CHON principle and are therefore completely incinerable and do not contribute to the secondary waste.

Malonamides

The malonamides are efficient coextractants for actinides and lanthanides leaving the remaining fission products in the aqueous phase. It has been shown that the molecular structure of the malonamides has a great influence on the metal extraction /19-5/.

By changing the substituents on the nitrogens and on the central carbon atom the metal extraction is changed. A lengthening of the carbon chain on the central carbon increases the extraction because of the increased lipophilicity of the molecule. To minimize the steric hindrance in the molecule one of the nitrogen substituents (R') has to be a methyl group. The other nitrogen substituent is suggested to be a quite short carbon chain but is not yet optimized. One of the objectives with this project was to study how two different R groups (a phenyl group and a cyclohexyl group) with the same number of carbon atoms, effect the metal extraction. Within the collaboration with the University of Reading these two new malonamides have been synthesised and the influence of the different structures on the metal extraction has been investigated.

Extraction data for some trivalent, tetravalent and hexavalent elements have been determined and the dependence of the metal extraction on nitric acid and malonamide concentration was studied /19-6/. To evaluate the extraction mechanism in this rather complex system, the nitric acid extraction is important and was therefore thoroughly investigated. A first suggestion of an extraction mechanism was proposed containing both an ordinary coordination mechanism and an anion-exchange mechanism. The results from this work will also be presented at ISEC (International Solvent Extraction Conference) in Melbourne March 1996.

TPTZ

TPTZ is a polydentate pyridine compound and has the ability to separate the trivalent actinides from the trivalent lanthanides. The greater extraction for the actinides compared to the lanthanides with TPTZ is ascribed to a more covalent character of the actinide-extractant bond.

TPTZ is quite hydrophilic and to achieve a sufficient extraction a lipophilic acid has to be added to the organic phase. Today, α-bromocapric acid is used as the lipophilic ligand but to fulfil the CHON principle other compounds has to be found. To make the extracted complex more hydrophobic it has also been suggested to attach carbon chains to the pyridyl groups in TPTZ.

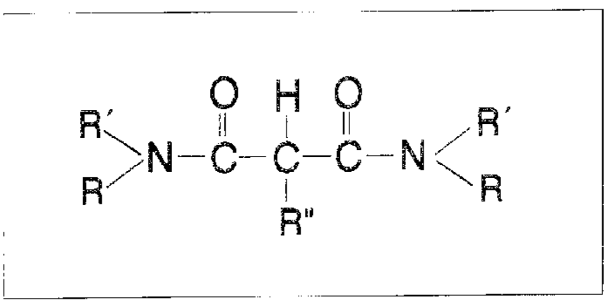

Figure 19-1. The general structure of a malonamide.

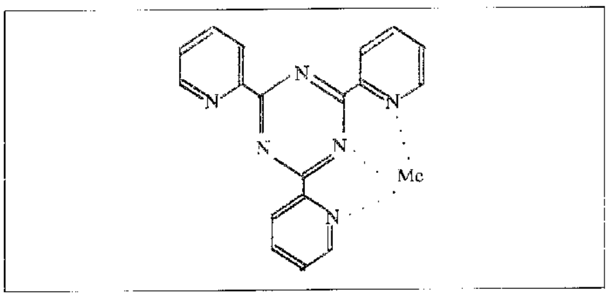

Figure 19-2. 2,4,6-tri-(2-pyridyl)-1,3,5-triazine (TPTZ).

One graduate student has finished her diploma work, concerning the extraction TPTZ in December 1995 /19-7/.

19.3 INTERNATIONAL COLLABORATIONS ON PARTITIONING AND TRANSMUTATION

England

A close collaboration has been initiated with the University of Reading concerning synthesis and investigations of new nitrogen containing extractants. Jan-Olov Liljenzin, CTH visited the University of Reading twice in 1995.

France

In 1994 a collaboration was initiated with Claude Musikas (former research director at CEA). It was agreed that Musikas will follow the research work with amide extractants at our department and review our research programme. C. Madic and C. Charbonnel (CEA) visited our department in May 1995. The future NEWPART project was discussed. Experiences with the AKUFVE system were also exchanged.

USA

The cooperation with LANL and KTH has been continued. From July 1994 to August 1995 Erik Möller from the Department of Neutron and Reactor Physics at KTH conducted his research and studies at Los Alamos Laboratory. Exchange of scientific information, data and results were proceeding. W. Gudowski was invited to take part in a peer review procedure of the ADTT-office in March 1995.

An informal collaboration was initiated in 1992 between Los Alamos National Laboratory (LANL) and the Department of Nuclear Chemistry, CTH. The collaboration involves exchange of information and results within aqueous based partitioning processes.

Japan

A collaboration was initiated in 1993 between the Department of Nuclear Chemistry, CTH and the Department of Fuel Cycle Safety Research at Japan Atomic Energy Institute (JAERI). It was agreed to exchange information, results and personnel between CTH and JAERI. In 1995 Shinichi Suzuki from JAERI visited CTH where molecular modelling with the computer code "CAChe System" was discussed for some specific extractants.

Germany

Discussions about the future EC work has been held by the CTH-group with the European Transuranium Institute, Karlsruhe. Within the collaboration with Technische Universität in Dresden possible transmutation reactions are evaluated and a closer collaboration is expected during 1996.

European Community

In 1995 the KTH-group became a coordinator of the EC-funded project: *Impact of the Accelerator Based Technologies on the Nuclear Fission Safety*. The project will last 3-years, 11 institutes from 8 countries are participating in this project.

Russia

The contacts with Russia scientists working in the ISTC project: "Feasibility Study of Principal Technologies in Accelerator Based Conversion of Military Pu and Long-Lived Radioactive Waste" were also active. W Gudowski, KTH participated in their workshop in October 1995.

19.4 GEOSCIENTIFIC APPRAISAL OF CONDITIONS AT LARGE DEPTHS

In 1992, SKB completed a comparative evaluation of several alternative concepts for geological disposal of high-level nuclear waste. The host medium was presumed to be crystalline rock. Different repository systems were evaluated and compared with respect to long-term repository performance, construction feasibility and cost. The concepts studied included three alternatives of disposal in mined excavations at moderate depth (tentatively about 500 m), including the so called KBS-3 method. As a fourth alternative, disposal in very deep boreholes (2000-4000 m) was considered. The study concluded that disposal according to the KBS-3 method was the most favourable alternative. As a complement to this main alternative, however, it has been decided to also pursue background research related to other options of disposal in crystalline rock. Such complementary work is motivated by the ambition to maintain flexibility for major system changes, if so desired in later stages of the disposal program.

The overall objective of the present work is to contribute to a better understanding of parameters and processes at large depths in the geosphere, concerning waste disposal. We define "large depths" as the interval between 1000 m and 5000 m, i.e. deeper than most conventional rock engineering efforts, comparable to the depths typically aimed at in deep drilling projects and occasionally in mining, but much too shallow to involve subcrustal considerations. The term "near-surface" here refers to the uppermost 1000 m. The scope is limited to the study of important characteristics of the geological medium at depth only; implications with respect to possible disposal concepts are not considered. Furthermore, the project is a deskstudy in that it involves only collection, compilation and interpretation of existing material. No field investigations are planned for the moment.

In terms of geoscientific disciplines, the work embraces geology, geophysics, hydrogeology, hydrochemistry, and geomechanics. In a first stage, data from the literature on fundamental parameters will be compiled and reviewed. The compilation will be limited to data that are judged to be relevant with respect to conditions in Swedish crystalline bedrock. Table 19-2 presents a tentative and condensed list of parameters of interest, ordered by discipline.

In a second stage, the information collected will be brought together and an integrated interpretation will be attempted. The outcome of this process will depend very much on the reliability and degree of generality that can be assigned to the various sets of data obtained. The long term aim is to develop a valid, descriptive model of the important processes and parameters of the geosphere at depth and to put this in relation to nearsurface conditions.

Table 19-2. Tentative list of fundamental geoscientific parameters for which available information will be compiled.

Disciplines	Geoscientific parameters
Geology-lithology	Rock types, metamorphic grade, structure
Geology-tectonics	Frequency, orientation etc of fractures and fracture zones, displacements, mineralization
Hydrogeology	Pore pressure, conductivity, frequency of conducting fractures etc.
Hydrochemistry	Fluid composition, water/rock interactions, isotopes, origin
Geophysics	Porosity, density, other standard petro physical parameters, geothermal conditions, seismic parameters
Rock Mechanics	Stress conditions, strength, stability

20 INTERNATIONAL COOPERATION

An important part of SKBs programme is to follow the corresponding research and development work conducted in other countries and to participate in international projects within the field of nuclear waste management.

These efforts give positive results in many ways e.g.:

- contributions to method and model development,
- broadened and strengthened databases,
- exploration of alternatives for repository and barrier design, material selection etc.,
- insights in programmes to broaden the public confidence in waste management systems.

The international work gives a perspective to the domestic programme and is an aid to the SKB strive for maintaining state-of-the art in relevant scientific areas of nuclear waste management.

20.1 SKB's BILATERAL AGREEMENTS WITH FOREIGN ORGANIZATIONS

SKB has signed formal bilateral agreements with the following organizations in other countries:

- USA – US DOE (Department of Energy),
- Canada – AECL (Atomic Energy of Canada Ltd) and ONTARIO HYDRO,
- Switzerland – Nagra (Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle),
- EC – EUROATOM,
- Finland – TVO and IVO (from January 1, 1996: POSIVA),
- Japan – JNFL (Japan Nuclear Fuel Ltd.).

Discussions started during 1995 with CEA in France to renew the previous agreement from 1994.

The formal agreements are similar in their construction and cover information exchange and cooperation within handling, treatment, storage and final disposal of radioactive waste. Exchange of up-to-date information (reports), as well as results and methods obtained from research and development, are main points in the agreements. Arranging joint seminars and short visits of specialists to other signatories' facilities are other examples of what is included within the framework of the agreements. General reviews of the signatories' waste programmes and activity planning are held at approximately one to two years intervals.

In the case of exchanges of personnel of long duration or extensive direct project cooperation, special agreements are generally concluded within the framework of the general agreement.

SKB also has information exchange without formal agreements with organizations in the other Nordic countries, Germany, Belgium and Great Britain.

20.2 COOPERATION WITH DOE, USA

During 1995 a SKB/USDOE bilateral information exchange meeting was held in the US.

The bilateral information exchange agreement between USDOE and SKB was renewed during 1995.

The cooperation between USDOE and SKB concerning the Äspö HRL continued during 1995. Due to budget reductions USDOE will be terminating the participation within the Äspö HRL Agreement during 1996.

20.3 COOPERATION WITH AECL AND ONTARIO HYDRO, CANADA

The cooperation has mainly been concentrated to the following issues:

- Natural analogues. Concerning the joint AECL/SKB work at Cigar Lake see section 16.5.
- Buffer and Backfill. SKB has together with ANDRA in France supported the AECL investigation of the bentonite backfill from the buffer mass heater test in URL, see section 16.2.
- Spent fuel. Cooperation work has during 1995 been performed with AECL Research and through the spent fuel workshop held in Canada, see section 16.1.

During 1995 SKB hosted a Benchmarking meeting for Ontario Hydro on cost estimates for nuclear fuel waste disposal.

20.4 COOPERATION WITH Nagra, SWITZERLAND

SKB has during 1995 had an extensive cooperation with Nagra. Some of the items involved have been

- safety analysis and performance assessment,
- natural analogue studies, see section 16.5,
- underground construction material performance, see section 14.2,
- other long-lived waste than spent nuclear fuel, see section 17,
- fracture characterization at Äspö (see Äspö Hard Rock Laboratory Annual Report 1995 /20-1/).

20.5 COOPERATION WITH CEA, ANDRA, DCC AND IPSN, FRANCE

The cooperation with organizations in France have mainly concerned the following issues:

- Natural analogues. SKB is engaged in the EC sponsored natural analogue project in Oklo which CEA is managing, see section 16.5.
- Instruments. IPSN/CEA in Cadarache, France, has performed development work on a borehole probe (CHEMLAB)
- Buffer and Backfill. As mentioned in section 16.2 above, ANDRA has participated in the work on bentonite buffer from the Canadian URL facility, see section 16.2.
- Other long-lived waste than spent nuclear fuel. Informal exchange of experience has been established, see section 17.

A CEA/SKB information exchange meeting was held in 1995, in Sweden.

20.6 COOPERATION WITH TVO AND IVO, FINLAND

SKB has a very close cooperation with TVO in many fields of the research on nuclear waste management. The following areas have during 1995 been the most active cooperation items:

- Safety analysis.
- Radionuclide retention research.
- Exchange of experience and technology for site investigation.
- Characterization of full-scale deposition holes, see section 14.2.
- Documentation work on relevant information on buffer and backfill materials. standardized and recommended laboratory and field test methods etc, see section 16.2.
- SKB is participating in the investigations of a uranium mineralization in Palmottu, a natural analogue to a repository with spent fuel, see section 16.5.

During 1995 a bilateral information exchange meeting between SKB, TVO and IVO was held in Sweden. SKB and TVO scientists have during 1995 had numerous meetings where information and experience exchange have been carried out.

20.7 COOPERATION WITH JNFL, JAPAN

During 1995 the cooperation has been carried out through study visits at SKB facilities and through informal information exchange meetings.

20.8 COOPERATION WITH NIREX, UK

Though there is no formal agreement on general information exchange with NIREX, a lot of general cooperation work has been performed outside the Äspö HRL work in which NIREX is participating. The areas where this cooperation has been made are mainly:

- Natural analogue studies, see section 16.5.
- Other long-lived waste than spent nuclear fuel, see section 17.

20.9 COOPERATION WITH EURATOM, EC

20.9.1 STC

SKB is represented by Sten Bjurström in the EURATOM Scientific and Technical Committee, STC. STC is attached to the Commission and has an advisory status in the fields of nuclear research and nuclear applications.

20.9.2 CHEMVAL

The first phase of the EC project CHEMVAL for verification and validation of chemical equilibrium calculation programs and coupled models for geochemistry transport was finalized and reported during 1990. A new phase of the CHEMVAL project called CHEMVAL2 started up during 1991 with participants from the EC countries, Sweden, Finland and Switzerland. The project has run from 1991-1994 and comprised temperature effects, ion strength effects, organic complexes, sorption, coprecipitation and coupled geochemical transport, see section 16.4

20.9.3 The Oklo Natural Analogue Project

SKB has participated in the first phase of the Oklo Natural Analogue Project, managed by the French organisation IPSN and supported by EC. A second phase is being planned (managed by CEA) and SKB is going to join as a full participant, see section 16.5.

20.9.4 The Palmottu Natural Analogue Project

SKB has participated in the original project as an observer. A new phase is being planned which is to be supported by EC. SKB will be a full participant of the new phase, see section 16.5.

20.9.5 Natural Analogue Working Group

Natural Analogue Working Group (NAWG) is an international group working with natural analogues and their use in the safety assessment modelling. It's organized by EC.

SKB has been represented in this group since its start in 1985. Presently one of SKB consultants, Dr John Smellie, is the chairman of the group.

20.10 COOPERATION WITHIN OECD NUCLEAR ENERGY AGENCY

20.10.1 RWMC

One of OECD/NEA's principal areas of cooperation is radioactive waste management in the member countries. These questions are dealt with by the **Radioactive Waste Management Committee (RWMC)**, where SKB is represented through Per-Eric Ahlström. Some work is carried out in joint international projects, and working groups are formed to facilitate information exchange or prepare material as a basis for joint opinions or coordination.

Seminars and workshops are arranged within important areas to document and discuss the state of development and the direction of future work.

The groups and projects within the area of radioactive waste management where SKB during 1995 was providing personnel or funding are listed below.

PAAG (Performance Assessment Advisory Group) functions in an advisory capacity to RWMC in matters pertaining to cooperation on means and methods for performance and safety analyses of final disposal systems.
Member from SKB: Tönis Papp

SEDE (Site evaluation and design of Experiments for Radioactive Waste Disposal) functions in an advisory capacity to RWMC in matters pertaining to the activities of experimental work in the member countries.
Member from SKB: Lars Olof Ericsson

IPAG (Integrated Performance Assessment of Deep Repositories) is a forum for the exchange of experience and information on recently completed performance assessment studies.
Member from SKB: Allan Hedin.

GEOTRAP
A PAAG/SEDE workshop was arranged in April 1995, hosted by GRS in Germany. The issue was successor to the INTRAVAL project as an international forum dealing with the prediction of radionuclide migration in heterogeneous media. The acronym GEOTRAP has been used. The workshop was titled "The prediction of radionuclide migration in geological media; practical approaches to the resolution of relevant issues." A number of key issues were identified and discussed. It was decided upon to recommend a series of PAAG/SEDE arranged workshops:

1. Rationale, implementation, modelling and interpretation of field tests.
2. The development and testing of transport concepts and codes on the light of the information from the field.
3. Models and data abstraction for site-specific synthesis, PA and confidence building.

Member from SKB: Anders Ström

Cooperative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects is a forum for information exchange and cooperation on various decommissioning projects all over the world.
Member from SKB: Hans Forsström. SKB is also sponsoring a programme coordinator, Shankar Menon, Studsvik Energiteknik AB.

Expert Group on Geochemical Modelling and Data deals with matters of common interest within geochemistry, including the buildup of a common thermodynamic database, TDB, and augmentation of the database for sorption data, SDB.
Member from SKB: Fred Karlsson

Working Group on the Regulatory Aspects of Future Human Actions at Radioactive Waste Disposal Sites deals with different aspects on human intrusion into waste repositories. The group was initiated in 1990. The group decided during its second meeting in 1995 to be dissolved. The results of the work performed in the group have been presented in a report and to PAAG. The group has pro-

posed to PAAG to consider the development of reference scenarios for future human actions.
Member from SKB: Lena Morén.

20.10.2 TDB

The TDB Project (Thermochemical Data Base) is under the direction of OECD/NEA. The goal is to develop a chemical thermodynamic database for a number of elements that are of importance for the safety assessment of the final disposal of radioactive waste. The development of the database entails not only collecting and storing published data, but also critical review. Review is carried out by a group of international experts selected for each element. At present the work is concentrated on neptunium, plutonium and technetium. The uranium and americium database has to been completed.

The TDB Project is a very important effort to develop a well documented, reviewed and internationally accepted database. SKB is supporting the activity and Swedish experts are participating in the review work.

20.11 COOPERATION WITHIN IAEA

Cooperation has during 1995 also been conducted within the International Atomic Energy Agency, IAEA, concerning the management of radioactive waste.

The cooperation is conducted in different ways, including the publication of reports consisting of:

- proceedings from international symposia,
- guidelines and standards within established areas of activity,
- status reports and methodology descriptions within important areas undergoing rapid development.

IAEA has an expert advisory group for its waste management programme (the International Waste Management Advisory Committee, INWAG) and arranges for information exchange within different special areas through Joint Research Programmes. IAEA publishes an annual catalogue on current research projects within the waste management field in the member countries.

An important new IAEA initiative is the RADWASS programme to work out international safety standards and guidelines. SKB will participate in the Standing Technical Committee for Disposal within the RADWASS programme.

20.11.1 VAMP

SKB is, through Studsvik EcoSafe, participating in an IAEA/CEC program on "Validation of Models on the Transfer of Radionuclides in Terrestial, Urban and Aquatic Environment and Acquisition of Data for that Purpose" (VAMP), see section 16.6.

20.12 COOPERATION WITHIN NORDIC NUCLEAR SAFETY RESEARCH

NKS, Nordic Nuclear Safety Research, is a voluntary cooperation body financed by relevant national authorities, nuclear companies and other organisations within the five Nordic countries: Denmark, Finland, Iceland, Norway and Sweden). The cooperation is in the fields of nuclear safety, radiation protection and emergency preparedness. Research projects are initiated and supported in these fields and one example is AFA-1, which deals with long-lived low and intermediate level waste. Waste of this kind is, more or less, present in all five countries and therefore of common interest. SKB is involved in AFA-1 and contributes to the study. AFA-1 is divided into three areas:

- Characterisation of long-lived and intermediate level radioactive waste (Subproject AFA-1.1).
- Function analyses for the near-field of repositories for long-lived waste (Subproject AFA-1.2).
- Environmental impact statement for repositories for long-lived waste (Subproject AFA-1.3).

The program in its present form started in 1994 and is expected to continue until 1997.

20.13 OTHER INTERNATIONAL COOPERATION

20.13.1 BIOMOVS

As indicated in section 16.6 SKB is participating in an international cooperative study BIOMOVS II (BIOspheric MOdel Validation Study) to test models for calculation of environmental transfer and accumulation of radionuclides in the biosphere. SKB has during 1995 taken active part in the scenario definition work where the RES methodology work, see section 15.2, has been used. The BIOMOVS study is expected to produce a final report during 1996.
Member from SKB: Sverker Nilsson

20.13.2 DECOVALEX I AND DECOVALEX II

Interest in developing coupled models has increased in recent years. The purpose is to be able to describe conditions in the near field of a repository in particular with greater realism. Within the framework of the DECOVALEX project (international cooperative project for the DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation), development and verification of coupled thermo-hydro-mechanical models is being conducted. SKI initiated the project during 1992 and is also the organization in charge of its execution. Nine countries are participating in the project. The first phase of the project was finalized in 1995, see section 16.4. A second phase, DECOVALEX II, was initiated in November 1995 with the same objectives. Four tasks for the project have been defined:

- Predictive modelling of coupled processes associated with the NIREX RCF shaft excavation at Sellafield, UK.

- THM modelling of the Engineered Barrier Experiment at Kamaishi Mine, Japan.

- Discussion and peer review of development of constitutive relationships of rock joints.

- Preparation of statements on the state-of-science on coupled THM-processes related to nuclear waste repository performance assessment.

Member from SKB: Lars Olof Ericsson

20.14 INTERNATIONAL COOPERATION IN THE ÄSPÖ HARD ROCK LABORATORY

As is mentioned in Chapter 18 the Äspö HRL has gained great international interest. The following organizations have up to the end of 1995 signed agreements to cooperate in joint work at the Äspö HRL:

- AECL, Canada,
- PNC, Japan,
- CRIEPI, Japan,
- ANDRA, France,
- TVO, Finland,
- UK NIREX, UK,
- USDOE, USA,
- Nagra, Switzerland,
- BMBF, Germany.

Most of the participating organisations have one or several groups working on models for groundwater flow and radionuclide migration. To coordinate this work a special Task Force has been formed.

For further information, see the Äspö Hard Rock Laboratory Annual Report 1995 /20-1/.

21 DOCUMENTATION

The scientific work in the SKB programme is documented at different levels:

- in reports requested by law and submitted to the Swedish Government or its authorities such as KBS-3, RD&D-Programme 95 and PLAN 95,
- in the series of SKB Technical Reports, in contributions to scientific journals, symposia and conferences in different subject areas, see Appendix 2,
- in SKB Arbetsrapporter,
- in SKB HRL International Cooperation Reports,
- in SKB HRL Progress Reports,
- in SKB Djupförvar Projektrapporter,
- in SKB Inkapsling Projektrapporter,
- in internal SKB memos,
- in technical memos and notes.

Further, the bulk of basic data from geological site characterization activities, spent fuel studies etc. are collected and stored in the electronic data base systems at SKB.

21.1 TECHNICAL REPORTS

SKB Technical Reports and many main reports, like for instance the KBS-3 report, are written in or translated to English. They are given a broad distribution to the scientific community in the nuclear waste management field in order to get feedback to the program by the comments, discussions and contacts between specialists that they may give rise to. SKB Technical Reports are filed as microfiche at IAEA in Vienna and are available through them. Abstracts of the 1995 Technical Reports are included in part IV of this Annual Report.

21.2 CONTRIBUTIONS TO PUBLICATIONS, SEMINARS ETC.

The contributions to conferences, symposia and scientific journals have been extensive during 1995, see Appendix 2.

Both SKB own staff as well as the contractors of SKB have been involved in this work.

21.3 SKB BIBLIOGRAPHICAL DATABASE

SKB has built up a database containing bibliographical data and abstracts on all reports currently available in the SKB library. The database, called BIBAS, contained by the end of 1995 about 11 400 references. The software used to manage the database is AskSam which has a powerful free-text search capability.

21.4 THE GEOLOGICAL DATABASE SYSTEM – GEOTAB

See section 18 on the Äspö Hard Rock Laboratory – Data management, p. 141.

21.5 COMPUTER SYSTEM AT SKB

An overview of the network and the computers at SKB is shown in Figure 21-1.

21.5.1 Computer network – LAN and WAN

The SKB network covers more than 1000 km from Malå in the north to Äspö in the south, not considering the internet connection. The network is very open in the sense that a user at any node can log into any other node (except into PCs), depending on his rights. Most filesystems are shared throughout the whole network.

The current physical locations are: the office at Brahegatan (with a special computer room) in Stockholm, the office of Äspö Hard Rock Laboratory north of Oskarshamn, the offices in Nyköping, Östhammar and Malå and two consultants in Stockholm and Luleå.

The computers at all sites are connected to local area networks (LAN type "thin wire ethernet"). The LANs in Stockholm and Äspö are connected via ethernet bridges, operating over leased 64 kbps lines. SUN workstations and Cisco routers are routing the traffic via ISDN connections to Nyköping, Malå, Östhammar, Luleå and KTH, making the segments appear as one. The whole network is directly connected to Internet via a 64 kbps leased line.

Only one standard protocol is used in the network – TCP/IP. TCP/IP is used by all connected computers

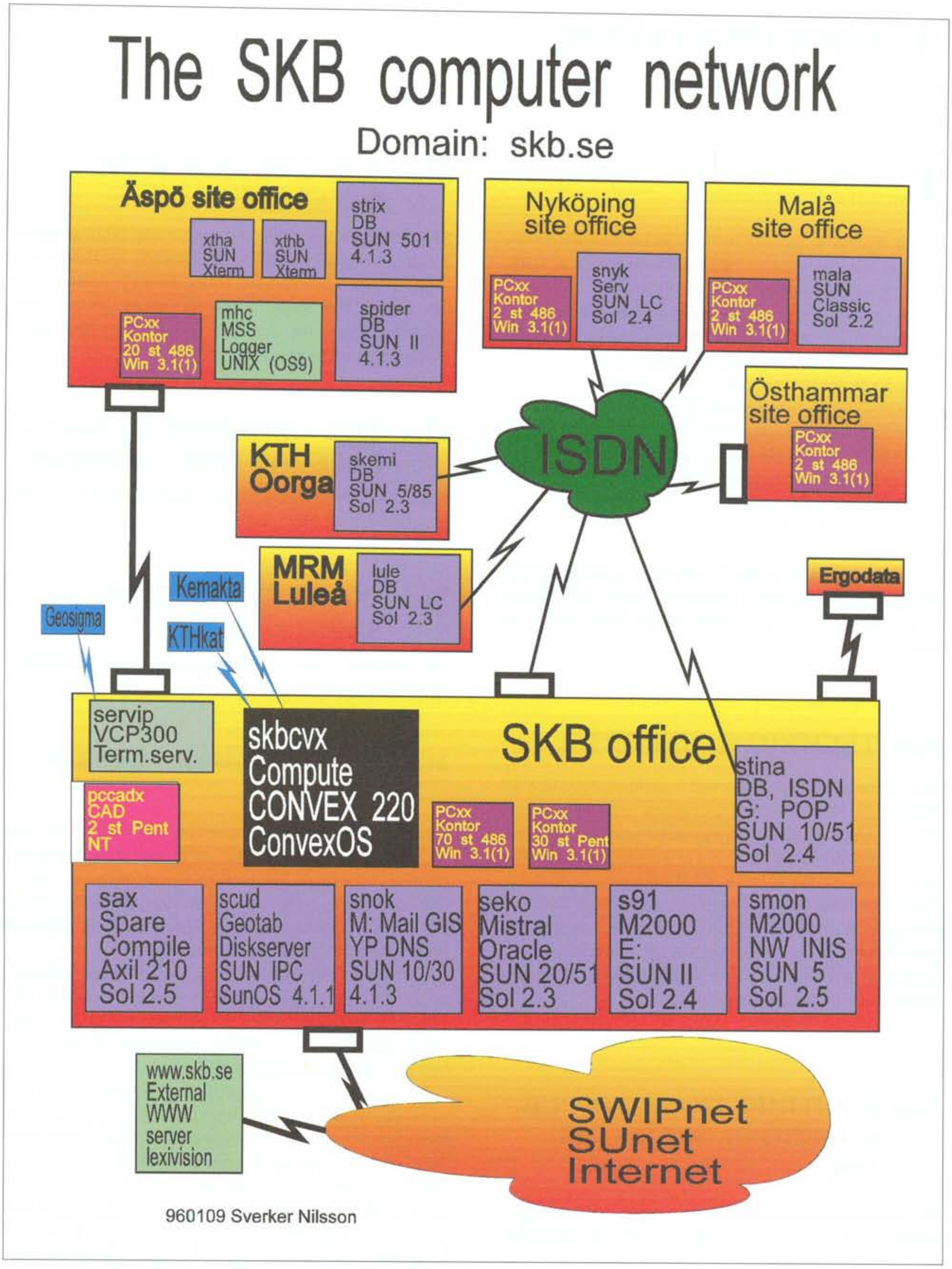

Figure 21-1. The SKB computer network.

(nodes) and also used for PC networking, terminal sessions, NIS, DNS, mail and file transfer. This facilitates implementation of new communication applications.

The connection to the Internet (64 kbps) is mostly used for WWW and incoming terminal and file transfer sessions, making it possible for contractors at the Swedish universities as well as in UK and US, to work interactively with our computers and browse our databases. This incoming access is limited to a few identified hosts and to certain types of traffic but outgoing access is unlimited. All workstations and PCs have direct connection to the internet and the use of common tools as WWW is increasing.

As SKB is contracting several companies for different work in the computer system a wide area network (WAN) with serial communication lines still exist. Currently 20 lines are connected to the computers in the computer room. Of these, 9 are used as dialup lines (2 in Gothenburg) and the rest connected via multiplexors and leased lines to 3 different sites in Stockholm and Uppsala.

21.5.2 Electronic mail

The mail systems used in all PCs and UNIXhosts is Zmail. This tool integrates smoothly into the E-mail international mailing system, covering 90% of all UNIX machines worldwide. All users has a worldwide E-mail address on the form skbsn@skb.se where "sn" is the person's initials, the preceding "skb" identifies the organization and "skb.se" identifies the SKB network domain. Furthermore all employees at SKB has the address firstname.lastname@skb.se.

21.5.3 Minisupercomputer

The CONVEX C220 is a 2-processor vector computer. Despite it's age (1989), it can still outperform most current workstations – especially when it comes to memory intensive floating point calculations on long vectors. It has been very easy to operate, running 24 hours a day with no major problems and with the expected vector capacity of about 24 Mflops (floating point operations per second). The operating system is a BSD UNIX 4.3 system with system V extensions. The current hardware configuration is 256 Mbyte main memory, 6 Gbyte on 6 disks, a 6250 bpi tape drive, 2 ethernet transceivers and 16 asyncronous ports.

21.5.4 Workstations and measuring system

Currently 14 SUN workstations are mainly used as PC network servers and communication servers, but they are of course also used as personal workstations and for presentation purposes (GIS and CAD). One workstation will run the legal accounting system in 1995.

The different data media coped with are Exabyte tapes (2, 5 and 7 Gb), QIC tapes (0.15 and 0.3 Gb), CDs, 1/2-inch tapes and diskettes.

The main machine in the automatic measuring system at Äspö is also a UNIX-like system, connected to the network, sharing disk and backup device with a SUN workstation and accessible from the all other nodes in the WAN.

21.5.5 PC network

The networking software used for PC networking is PCNFS from SUN Microsystems. The main use is to keep a common secure file system, making document transfer very easy and the common software and standards consistent throughout the company. A PC in this LAN is served by several file servers simultaneously, to improve performance. At least one server has been sited at each site. A typical PC is nowadays a 486 or Pentium with at least 200 Mb disk running DOS 5 Windows 3.11, MS-Office programs and others.

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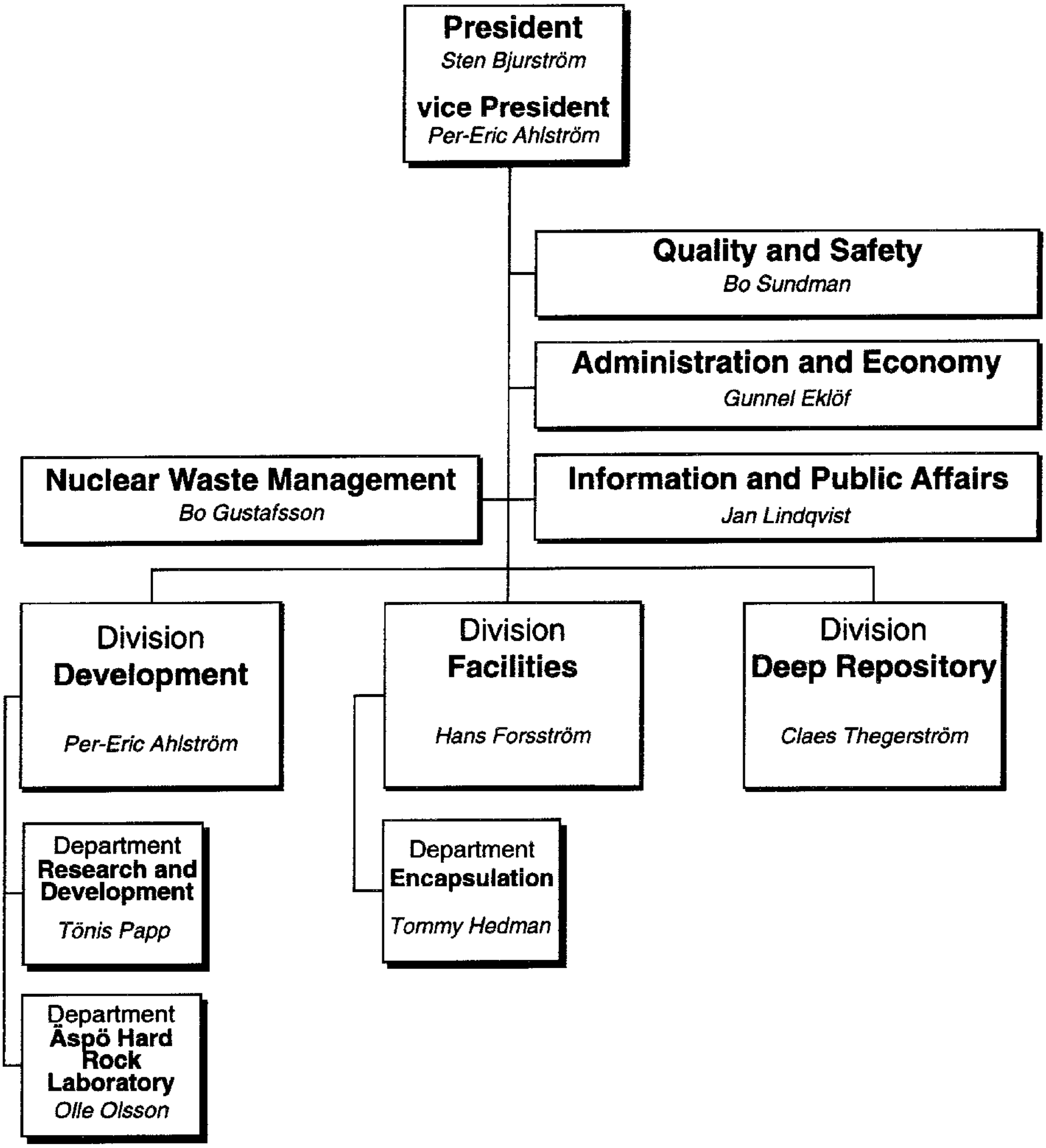
Part III

Appendices

CONTENTS PART III

ORGANIZATION CHART FOR SKB
APRIL 1996



SKB TECHNICAL STAFF

Note: Several persons of the staff work in more than one function; sometimes these functions belong to different divisions or departments.

DEVELOPMENT

Per-Eric Ahlström	Vice President, Director Development
Monica Hammarström	International cooperation
Fred Karlsson	Chemistry, Natural analogues, Other waste types

Research and Development:

Tönis Papp	Research Director
Lars O Ericsson	Geoscience
Allan Hedin	Uncertainties – Safety analysis
Lena Morén	Scenario – Safety analysis
Sverker Nilsson	Biosphere – Safety analysis
Patrik Sellin	Near-field – Safety analysis
Kastriot Spahiu	Nuclear fuel
Anders Ström	Far-field – Safety analysis
Peter Wikberg	Geochemistry, Groundwater chemistry

Äspö Hard Rock Laboratory:

Olle Olsson	Director of Äspö Hard Rock Laboratory
Olle Zellman	Operations Manager Äspö
Lars Andersson	Service Engineer
Ebbe Eriksson	Database systems
Johannes Heikillä	Service Engineer
Leif Jirhem	Project administrator, Manager Experimental services
Tomas Karlsson	Electronic Engineer
Katinka Klingberg	Project coordinator, Chemistry
Mats Ohlsson	Manager Data systems
Gunnar Ramqvist	Project coordinator
Leif Stenberg	Project coordinator, Geology

Computer Services:

Sverker Nilsson	Manager, System Administrations, Databases
Robert Perhat	PC support
Vacant	PC support
Vacant	System administration and GIS

DEEP REPOSITORY

Claes Thegerström	Director Deep Repository
Kaj Ahlbom	Feasibility study Östhammar
Karl-Erik Almén	Site investigations
Anders Appelgren	Site office Nyköping
Göran Bäckblom	Overview studies and Technical coordination
Gunnar Bäckström	Public affairs
Torsten Eng	Feasibility study Nyköping, EIA studies
Torbjörn Hugo-Persson	Site office Malå
Bengt Leijon	Feasibility study Malå
Gerd Nirvin	Site office Östhammar
Christer Svemar	Design studies, Buffer and Backfill material
Jerker Tengman	Project administration
Erik Thurner	Field measurements, Instruments

FACILITIES

Hans Forsström	Technical Director Facilities
Bo Gustafsson	Deputy Director
Swen Berger	Transportation
Jan Carlsson	Quality assurance, SFR – Safety and waste, Other waste types
Peter Dybeck	SFR – Operation, Transportation
Marie Skogsberg	SFR – Safety and waste
Stig Pettersson	Engineering & Costs, Design studies, Layout
Per Riggare	SFR – Safety and waste
Bo Sundman	SVAFO
Maria Wikström	Engineering & Costs
Jan Vogt	CLAB, Encapsulation process system

Encapsulation:

Tommy Hedman	Project Manager
Claes Göran Andersson	Canister manufacturing
Olle Broman	Project administrator
Stig Ericsson	Canister laboratory
Göran Fröman	Process systems
Kristina Gillin	Process systems
Kjell Mårtensson	Fuel storage
Olle Sanner	Layout
Lars Werme	Canister, Nuclear fuel

QUALITY AND SAFETY

Bo Sundman	Director

NUCLEAR FUEL

Ingemar Lindholm	Director Nuclear Fuel Supply
Eva Backlöf	Transport manager
Göran Schultz	Contracts

NUCLEAR WASTE MANAGEMENT

Bo Gustafsson	Director Nuclear Waste Management
Lars B. Nilsson	Senior consultant

LECTURES AND PUBLICATIONS 1995

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TR 95-18
Thermal evidence of Caledonide foreland, molasse sedimentation in Fennoscandia
Tullborg, Eva-Lena 1); Larsson, Sven Åke 1); Björklund, Lennart 1); Samuelsson, Lennart 2), Stigh, Jimmy 1)
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November 1995

TR 95-19
Compaction of bentonite blocks. Development of technique for industrial production of blocks which are manageable by man
Johannesson, Lars-Erik; Börgesson, Lennart; Sandén, Torbjörn
Clay Technology AB, Lund, Sweden
April 1995

TR 95-20
Modelling of the physical behaviour of water saturated clay barriers. Laboratory tests, material models and finite element application
Börgesson, Lennart 1); Johannesson, Lars-Eric 1); Sandén, Torbjörn 1); Hernelind, Jan 2)
Clay Technology AB, Lund, Sweden 1); FEM-Tech AB, Västerås, Sweden 2)
September 1995

TR 95-21
Conceptual model for concrete long time degradation in a deep nuclear waste repository
Lagerblad, Björn; Trägårdh, Jan
Swedish Cement and Concrete Research Institute
February 1994

TR 95-22
The use of interaction matrices for identification, structuring and ranking of FEPs in a repository system. Application on the far-field of a deep geological repository for spent fuel
Skagius, Kristina 1); Ström, Anders 2); Wiborgh, Marie 1)
Kemakta, Stockholm, Sweden 1); Swedish Nuclear Fuel and Waste Management Co, Stockholm, Sweden 2)
November 1995

TR 95-23
Spent nuclear fuel. A review of properties of possible relevance to corrosion processes
Forsyth, Roy
Caledon Consult AB
April 1995

TR 95-24
Studies of colloids and their importance for repository performance assessment
Laaksoharju, Marcus 1); Degueldre, Claude 2); Skårman, Christina 1)
GeoPoint AB, Sollentuna, Sweden 1); University of Geneva, Switzerland 2)
December 1995

TR 95-25
Sulphate reduction in the Äspö HRL tunnel
Laaksoharju, Marcus (ed.)
GeoPoint AB, Sollentuna, Sweden
December 1995

TR 95-26
The Äspö redox investigations in block scale. Project summary and implications for repository performance assessment
Banwart, Steven (ed.)
Dept. of Civil and Environmental Engineering, University of Bradford, UK
November 1995

TR 95-27
Survival of bacteria in nuclear waste buffer materials. The influence of nutrients, temperature and water activity
Pedersen, Karsten 1); Motamedi, Mehrdad 1); Karnland, Ola 2)
Department of General and Marine Microbiology, The Lundberg Institute, Göteborg University, Göteborg Sweden 1); Clay Technology AB, Lund, Sweden 2)
December 1995

TR 95-28
DECOVALEX I – Test Case 2: Calculation of the Fanay-Augères THM Test – Thermomechanical modelling of a fractured rock volume
Börgesson, Lennart 1); Hernelind, Jan 2)
Clay Technology AB, Lund, Sweden 1); Fem-Tech AB, Västerås, Sweden 2)
December 1995

TR 95-29
DECOVALEX I – Test Case 3: Calculation of the Big Ben Experiment – Coupled modelling of the thermal, mechanical and hydraulic behaviour of water-unsaturated buffer material in a simulated deposition hole
Börgesson, Lennart 1); Hernelind, Jan 2)
Clay Technology AB, Lund, Sweden 1); Fem-Tech AB, Västerås, Sweden 2)
December 1995

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DECOVALEX I – Bench-Mark Test 3: Thermo-hydro-mechanical modelling
Israelsson, Jan
Itasca Geomekanik AB
December 1995

TR 95-31
DECOVALEX I – Test Case 1: Coupled stress-flow model
Rosengren, Lars 1); Christianson, Mark 2)
Itasca Geomekanik AB 1); Itasca Consulting Group Inc. 2)
December 1995

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Partitioning and transmutation (P&T) 1995. A review of the current state of the art
Skålberg, Mats 1); Landgren, Anders 1); Spjuth, Lena 1); Liljenzin, Jan-Olov 1); Gudowski, Waclaw 2)
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December 1995

TR 95-33
Geohydrological simulation of a deep coastal repository
Follin, Sven
Golder Associates AB, Stockholm, Sweden
December 1995

TR 95-34
General Siting Study 95 – Siting of a deep repository for spent nuclear fuel
Swedish Nuclear Fuel and Waste Management Co
October 1995

TR 95-35
A selected thermodynamic database for REE to be used in HLNW performance assessment exercises
Spahiu, Kastriot; Bruno, Jordi
MBT Tecnologia Ambiental, Cerdanyola, Spain
January 1995

TR 95-36
Reconstructing the tectonic history of Fennoscandia from its margins: The past 100 million years
Muir Wood, Robert
EQE International Limited
December 1995

Appendix 5

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POST-GRADUATE THESES SUPPORTED BY SKB 1995

Developments of some in situ tracer techniques applied in groundwater research
Byegård, J
Doctoral thesis, Dept. of Nuclear Chemistry, Chalmers University of Technology, Göteborg, Sweden, 1995

Transport of radionuclides and colloid through quartz sand columns
Christiansen-Sätmark, B
Doctoral thesis, Dept. of Nuclear Chemistry, Chalmers University of Technology, Göteborg, Sweden, 1995

Simulation of fluid flow and transport of solutes in fractured rock using a channel network model – Application to field experiments
Gylling, B
Licentiate treatise, Dept. of Chemical Engineering and Technology, Royal Institute of Technology, Stockholm, Sweden, 1995

Aperture distribution of rock fractures
Hakami, E
Doctoral thesis, Div. of Engineering Geology, Dept. of Civil and Environmental Engineering, Royal Institute of Technology, Stockholm, Sweden, 1995

Development and test of models in the natural analogue studies of the Cigar Lake uranium deposit
Liu, J
Doctoral thesis, Dept. of Chemical Engineering and Technology, Royal Institute of Technology, Stockholm, Sweden, 1995

The near-field transport in a repository for high-level nuclear waste
Romero Aranguiz, L
Doctoral thesis, Dept. of Chemical Engineering and Technology, Royal Institute of Technology, Stockholm, Sweden, 1995

Estimation of hydrogeological properties in vulnerability and risk assessments
Rosén, L
Doctoral thesis, Dept. of Geology, Chalmers University of Technology, Göteborg, Sweden, 1995

Geostatistical methods for prediction of mass transport in groundwater
Wen, X-H
Doctoral thesis, Div. of Water Resources Engineering, Dept. of Civil and Environmental Engineering, Royal Institute of Technology, Stockholm, Sweden, 1995

SKB ANNUAL REPORT 1995

Part IV

Summaries of Technical Reports Issued during 1995

SKB Technical Report 95-01

Biotite and chlorite weathering at 25°C: The dependence of pH and (bi)carbonate on weathering kinetics, dissolution stoichiometry, and solubility; and the relation to redox conditions in qranitic aquifers

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January 1995

ABSTRACT

We have studied the kinetics and thermodynamics of biotite and chlorite weathering in the pH range $2 < pH < 10$ at 25°C. The dissolution is highly non-stoichiometric and pH dependent in the whole pH region. By XPD we have identified a clay mineral to be the main weathering product formed. A model of biotite dissolution and the formation of secondary solubility controlling minerals, such as Fe(III)-hydroxide, Na-clay, quartz and gibbsite is used to explain experimental equilibrium concentrations of silicon, iron, aluminium and magnesium. The model predicts redox potentials in the range of -200-400 mV at neutral pH and qualitatively agrees with field data reported in the literature. We use observed iron release rates to make conservative estimates of timescales of 1) the depletion of molecular oxygen from deep aquifers (10^1-10^2 years); and 2) the development of characteristic Fe(II) concentrations (10^{-5} M in 10^{-1} years).

The Fe(II)-bearing clay minerals formed during these experiments are similar to the fracture-filling-material observed at the Äspö Hard Rock Laboratory. Such clays can provide reducing capacity to a repository. They can help maintain anoxic conditions by consuming oxygen that enters the repository during the construction and operation phases thereby helping maintain the redox stability of the repository regarding canister corrosion. The half-life of oxygen trapped in the repository at the time of closure depends on the rate of oxygen uptake by Fe(II) minerals, sulfide minerals and organic carbon. Fe(II)-caly minerals are important to the redox stability of a repository, as well as providing a sorption barrier to radionuclide migration.

SKB Technical Report 95-02

Copper canister with cast inner component. Amendment to project on Alternative Systems Study (PASS), SKB TR 93-04

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March 1995

ABSTRACT

The Project on Alternative Systems Study, PASS, was described in a report dated october 1992. In the report, the reference repository concept KBS-3 is described together with three other alternatives. In the report several designs for fuel storage canister are presented. This report describes a recently developed design for the inner component of the composite, steel and copper, canister which is the main alternative in the KBS-3 model. The new design will be manufactured by casting. A cast insert with inner walls eliminates the need for a stabilizing filler in the canister and guarantees that the fuel remains sub-critical during sufficient time in the repository. The cast insert is judged, to, in comparison with the steel tube alternative, lead to a considerably simplified process in the encapsulation plant and lower development and investment cost. Positive side effects of the design are that the mechanical strength is improved by a factor 2-3 and that the difficult filling operation is avoided in the encapsulation process. The drawbacks are higher weight and probably higher unit price for the empty canister.

SKB Technical Report 95-03

Prestudy of final disposal of long-lived low and intermediate level waste

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January 1995

ABSTRACT

The repository for long-lived low and intermediate level waste, SFL 3-5, is foreseen to be located adjacent to the

deep repository for spent encapsulated fuel, SFL 2. The SFL 3-5 repository comprises of three repository parts which will be used for the different categories of waste. In this report the work performed within a prestudy of the SFL 3-5 repository concept is summarised. The aim was to make a first preliminary and simplified assessment of the near-field as a barrier to radionuclide dispersion.

A major task has been to compile information on the waste foreseen to be disposed of in SFL 3-5. The waste comprises of; low and intermediate level waste from Studsvik, operational waste from the central interim storage for spent fuel, CLAB, and the encapsulation plant, decommissioning waste from these facilities, and core components and internal parts from the reactors. The total waste volume has been estimated to about 25 000 m^3. The total activity content at repository closure is estimated to be about $1 \cdot 10^{17}$ Bq in SFL 3-5. At repository closure the short-lived radionuclides, for example Co-60 and Fe-55, have decayed considerably and the activity is dominated by nickel isotopes in the metallic waste from the reactors, to be disposed of in SFL 5. However, other radionuclides may be more or equally important from a safety point of view, e.g cesium-isotopes and actinides which are found in largest amounts in the SFL 3 waste.

A first evaluation of the long term performance of the SFL 3-5 repository has been made. A systematic methodology for scenario formulation was tested. It was possible to carry through the different steps in the methodology, to formulate a Reference Scenario for SFL 3-5 and to define a Reference Case to be quantitatively analysed. An important result from this part of study is the documentation. The documentation comprises of the Influence diagram with the linked data base with documents, protocols and decisions behind the development of the Reference Scenario. This will facilitate future reevaluations.

During the prestudy, investigations and experimental works have started up at different organisations to support the need for data in the evaluations of the SFL 3-5 repository performance.

There is a number of processes and mechanisms that may be of potential importance in the evaluation of long-term repository performance. In the prestudy, only s' np-lified calculations have been performed on average hydraulic conditions in the repository, potential increase in temperature and potential gas formation in the waste packaging. The chemical conditions in the repository and factors influencing availability and retardation of elements have been considered in the selection of data for the Reference Case calculations.

The near-field release of contaminants was calculated for a selected number of radionuclides and chemotoxic elements. The radionuclide release calculations revealed that Cs-137 and Ni-63 would dominate the annual release from all repository parts during the first 1000 years after repository closure and that Ni-59 would dominate at longer times. The highest release rates arises from SFL 4, despite the fact that the total content of radionuclides are more than one order of magnitude less compared to the other repository parts. The main reason is that in the calculations retention of radionuclides is only considered for the sand backfill in SFL 4. In order to get a measure of the radiotoxicity, the near-field releases were converted to intermediate doses by assuming that the entire release from the repository was captured in a drinking-water well. The intermediate doses were for all the studied radionuclides below 0.1 mSv/year for the simplified well scenario. Lead and beryllium were chosen as model substances for chemotoxic elements and included as an example in the release calculations for SFL 3. For the same simplified well scenario as for the radionuclide releases, it was found that the near-field releases of lead and beryllium would result in concentrations in the water in the well which are below the guideline values used for drinking water.

Finally, it must be remembered that the near-field release calculations have been performed on a preliminary waste inventory and for subset of elements, and the defined Reference Case does not consider all processes and mechanisms assessed to be of potential importance in the Reference Scenario. The importance of these processes and mechanisms have to be evaluated in future performance assessments. In addition, other scenarios for example describing the potential influence of initial conditions in the repository and changes of environmental conditions have to be studied.

SKB Technical Report 95-04

Spent nuclear fuel corrosion: The application of ICP-MS to direct actinide analysis

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January 1995

ABSTRACT

The ICP-MS technique has been applied to the analysis of the actinide contents of corrodant solutions from experiments performed to study the corrosion of spent nuclear fuel in simulated groundwaters. Analysis was performed directly on the solutions, without employing separation or isotope dilution techniques. The results from two analytical campaigns using natural indium and thorium internal standards are compared.

Under both oxic and anoxic conditions, the U contents can be determined with good accuracy and precision. The same applies to Np and Pu under oxic conditions, where

the solution concentrations range down to about 0.1 ppb. Under anoxic conditions, where solution concentrations are lower by one or two orders of magnitude, reasonable results for these two actinides can be obtained, but with much lower precision. Direct analysis of Am and Cm, however, gave unsatisfactory results, since the technique is limited by poor measurement statistics and background uncertainty.

SKB Technical Report 95-05

Groundwater sampling and chemical characterisation of the Laxemar deep borehole KLX02

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February 1995

ABSTRACT

The Laxemar deep borehole, KLX02 (1705 m depth), located close to the Äspö Hard Rock Laboratory (HRL), has been investigated. Groundwater sampling was conducted on two occasions and using different methods. The first sampling was taken in the open borehole using the so-called Tube sampler; the second sampling carried out using the SKB-packer equipment to isolate pre-determined borehole sections. Groundwater compositions consist of two distinct groupings; one shallow to intermediate Sodium-Bicarbonate type [Na(Ca,K):HCO_3-Cl(SO_4)) to a depth of 1000 m, and the other of deep origin, a Calcium-Chloride type [Ca-Na(K):Cl-SO_4(Br)], occurring below 1000 m. The deep brines contain up to 46000 mg of Cl per litre. The influence of borehole activities are seen in the tritium data which record significant tritium down to 1000 m, and even to 1420 m. Mixing modelling shows that water from the 1960's is the main source for this tritium. The high tritium values in the 1090–1096.2 m section are due to contamination of 1% shallow water from 1960 and 2% of modern shallow water. The upper 800 m of bedrock at Laxemar lies within a groundwater recharge area; the sub-vertical to moderate angled fracture zones facilitate groundwater circulation to considerable depths, at least to 800 m, thus accounting for some of the low saline brackish groundwaters in these conducting fracture zones. Below 1000 m the system is hydraulically and geochemically "closed" such that highly saline brines exist in a near-stagnant environment.

SKB Technical Report 95-06

Palaeohydrological implications in the Baltic area and its relation to the groundwater at Äspö, south-eastern Sweden – A literature study

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March 1995

ABSTRACT

A literature study of different groundwaters in the circum Baltic region is presented in this work. The study is mainly focused on the isotopic signatures observed in different groundwaters in Sweden and Finland. Several saline groundwaters in the Baltic region at depth of 150 to 500 m depth show stable (δD, $\delta^{13}C$, $\delta^{18}O$) and radiogenic ($\delta^{87}Sr$) isotope assembly which is suggestive of a marine origin. However, a discrepancy is sometimes observed between the stable as well as radiogenic isotopes of the intermediate groundwater, which suggest a mixture of fossil marine water and a post-glacial runoff of meltwater. In order to explain this phenomenon, the initial setting in $\delta^{18}O$ may have been depleted due to large input of high latitude marine water or cold meltwaters. A solution to the contradiction between the strontium ($\delta^{87}Sr$) and stable isotope (δD, $\delta^{13}C$, $\delta^{18}O$) signatures of the groundwater and of the calcite fracture fillings at Äspö and other places is attained, if it is assumed that the strontium in Baltic Sea water has undergone a significant decrease in $\delta^{87}Sr$ since the last glaciation. A scenario can be constructed to suggest that the Baltic Sea during the initial stage of the Litorina sea (8000 to 5000 Y B.P.) contained strontium with much larger $\delta^{87}Sr$ values. Another explanation for the positive $\delta^{87}Sr$ values may be due to water/rock interaction between the groundwater and the abundant fracture clay minerals, which are observed at Äspö. Typically most of the saline groundwaters occur both in Sweden and Finland below the highest marine shore line during the Holocene. Almost all inland groundwaters show a totally different pattern which is typically non marine, meteoric in origin. This study also summarises the stable isotope (δD, $\delta^{13}C$, $\delta^{18}O$) geochemistry dependence of the important global and regional environmental changes which may have influenced of the palaeohydrology as well as groundwater formation and in the Baltic region and especially at Äspö.

SKB Technical Report 95-07

Äspö Hard Rock Laboratory. Annual report 1994

SKB

April 1995

ABSTRACT

The Äspö Hard Rock Laboratory is being constructed as part of the preparations for the deep geological repository of spent nuclear fuel in Sweden. The Annual Report 1994 for the Äspö Hard Rock Laboratory contains an overview of the work conducted.

Present work is focused on verification of pre-investigation methods and development of detailed investigation methodology which is applied during tunnel construction. Construction of the facility and detailed characterization of the bedrock are performed in parallel. Excavation of the main access tunnel was completed during 1994 and at the end of the year only minor excavation work remained. The last 400 m of the main tunnel, which has a total length of 3600 m, was excavated by a 5 m diameter tunnel boring machine (TBM). The tunnel reaches a depth of 450 m below ground.

Preparations for the Operating Phase have started and detailed plans have been prepared for several experiments.

Nine organizations, including SKB, from eight countries are now participating in the work at the Äspö Hard Rock Laboratory and are contributing to the work performed and results achieved.

SKB Technical Report 95-08

Feasibility study for siting of a deep repository within the Storuman municipality

Swedish Nuclear Fuel and Waste Management Co (SKB)

January 1995

SKB Technical Report 95-09

A thermodynamic data base for Tc to calculate equilibrium solubilities at temperatures up to 300°C

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April 1995

ABSTRACT

Thermodynamic data has been selected for solids and aqueous species of technetium. Equilibrium constants have been calculated in the temperature range 0 to 300°C at a pressure of 1 bar for T <100°C and at the steam saturated pressure at higher temperatures. For aqueous species, the revised Helgeson-Kirkham-Flowers model is used for temperature extrapolations.

The data base contains a large amount of estimated data, and the methods used for these estimations are described in detail. A new equation is presented that allows the estimation of $\Delta_r C^0_{pm}$ values (correct formula see document) for mononuclear hydrolysis reactions. The formation constants for chloro complexes of Tc(V) and Tc(IV), whose existence is well established, have been estimated. The majority of entropy and heat capacity values in the data base have also been estimated, and therefore temperature extrapolations are largely based on estimations. The uncertainties derived from these calculations are described.

Using the data base developed in this work, technetium solubilities have been calculated as a function of temperature for different chemical conditions. The implications for the mobility of Tc under nuclear repository conditions are discussed.

SKB Technical Report 95-10

Investigations of subterranean microorganisms. Their importance for performance assessment of radioactive waste disposal

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June 1995

ABSTRACT

The main actors in this report are the microorganisms (microbes is a synonymous designation). Let us therefore first define this concept. The following description can be read in the Swedish National Encyclopaedia: A microorganism is an organism that is invisible for the naked eye (smaller than some tenths of a millimetre), and requires adapted techniques for a successful study, such as sterilisation, sterile techniques, methods in molecular biology, microscope, and particular culturing devices and chemicals. Bacteria, unicellular algae, yeast, microscopic fungi, and protozoa (unicellular animals) are all microorganisms; sometimes, also viruses are improperly included (they are not true living entities). A unifying characteristic is that all microorganisms have the ability to grow as unicellular organisms in nature. The intention with this report is, with present knowledge as a starting-point, to give a broad and thorough description of how microorganisms may influence performance safety assessment of repositories for radioactive waste. Potential positive, neutral and negative effects have been identified. To the best of our knowledge, bacteria are the totally dominating microorganisms in the deep subterranean environment. Therefore, the report mainly deals with this group of microorganisms. Most of the microbiological concepts discussed in the report are discussed in relation to the disposal concepts for radioactive waste in specific concluding sections in the text and also in summaries ending each chapter.

The first chapter gives an overview of the Swedish concepts for disposal of low-, intermediate- and highlevel radioactive waste. Further, the geological, chemical and hydrological conditions in repositories, that can be related to microorganisms are discussed. The following two chapters are intended to give the reader knowledge of the requirements for life of microorganisms and possible limiting or stimulating growth factors. Their relations to oxygen, temperature, pH, radiation, pressure, availability of water and nutrients, and their need for energy sources are discussed in detail. More, a basic knowledge about bacterial processes is presented. The participation of bacteria in the cycling of carbon, nitrogen, sulphur, iron, manganese and hydrogen is dealt with.

Chapter four is mainly a literature review of the knowledge about subterranean bacteria. Chiefly, we aim at a current description of the far-field environment in which repositories are or will be placed. The problems connected with sampling without disturbing the samples is reviewed. Also, a large number of techniques are presented for the determination of the numbers of bacteria in natural environments, their spatial distribution and species diversity, and methods for determination of microbial activity. Thereafter follows a summary of results about the microbiology in various subterranean environments such as hydrothermal groundwater, deep sediments (mainly in North America), the hard rock laboratories of Stripa and Äspö, and from a number of different natural analogues, among them are Cigar Lake, Canada; Oklo, Gabon; Maqarin, Jordan; Poços de Caldas, Brazil and Palmottu in Finland. Chapter five treats investigations concerning microorganisms in repository(like) environments, the near-field. At first, a literature overview is given and then the conditions for life in the near-field are discussed. Particularly, the low water availability in bentonite at the intended swelling pressure is hypothesised to be to low for an active life of most microorganisms – they will simply be desiccated. Finally, the chapter five treats microbial corrosion and redox processes as they are of significance for the performance of construction materials such as canisters and concrete, and for the mobility of radionuclides.

Chapter six brings up the possibilities to predict presence and activity of microorganisms in the far-field as well as in the near-field by mathematical models. An important conclusion drawn is that successful modelling demands a multi-disciplinary approach. Microbiology is an integrated part of the whole, dealt with also in other disciplines such as geology, chemistry, hydrology, and additionally in the near-field, technology and engineering. At last, chapter seven summarises the conclusions made in the report and identifies research needs and how microorganisms may influence performance safety assessment of radioactive waste disposal.

The report consists of 43 figures, 36 tables, it has approximately 200 pages of text and refers to 293 articles, book titles and reports. Separate lists of figures and tables are included after table of contents. Further, an index is included at the end of the report to facilitate for the reader to where in the report central concepts, names etc. can be found.

SKB Technical Report 95-11

Solute transport in fractured media – The important mechanisms for performance assessment

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June 1995

ABSTRACT

The most important mechanisms that control the release of contaminants from a repository for nuclear or chemical waste have been studied. For the time scale of interest for the disposal of nuclear and even chemical waste, diffusion into the rock matrix is an important factor which retards and dilutes the contaminants. It is found that the water flow-rate distribution and the flow-wetted surface are the entities that primarily determine the solute transport. If the diffusion in to the rock matrix is negligible, the solute transport is determined by the water flow rate and the flow porosity. This is shown by simulations using analytical solutions obtained for simple geometries, such as the flow in a fracture or a channel. Similar results are obtained for more complex systems, such as flow in a fracture with variable aperture and through a network of channels. It is also found that the use of a retardation factor relating the travel times of interacting and noninteracting solutes is inappropriate and may be misleading.

SKB Technical Report 95-12

Literature survey of matrix diffusion theory and of experiments and data including natural analogues

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August 1995

ABSTRACT

Diffusion theory in general and matrix diffusion theory in particular has been outlined, and experimental work has been reviewed. Literature diffusion data has been systematised in the form of tables and data has then been compared and discussed. Also natural analogues supporting matrix diffusion theory and pore connectivity, have been briefly reviewed and discussed. Strong indications of surface diffusion and anion exclusion have been found, and natural analogue studies and "in-situ"-experiments suggest pore connectivity in the scale of metres. Matrix diffusion, however, mostly seem to be confined to zones of higher porosity extending only a few centimetres into the rock. Surface coating material do not seem to hinder sorption or diffusion into the rock.

SKB Technical Report 95-13

Interactions of trace elements with fracture filling minerals from the Äspö Hard Rock Laboratory

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June 1995

ABSTRACT

This report focuses on the distribution of stable elements and natural radionuclides (for example REEs, Th, U, Ra, Sr and Cs) in natural fracture systems. These elements are chemical analogues to or identical with artificial radionuclides occurring in a nuclear waste repository. They have been redistributed by natural processes in the past; mobilisation, transport and deposition of which the latter is mani-fested as "enrichments" of the elements in fracture fillings. Fracture fillings dominated by Fe-oxyhydroxide, calcite and clay minerals show the highest concentrations. These minerals are thus important in retarding radionuclides and have been studied specifically; sequential extraction has for example been applied to clayish gouge material and fresh Fe-oxyhydroxide/calcite precipitates.

REEs, Sc, T, U, Ra, Sr and Ba are scavenged by the mixed Fe-oxyhydroxide/calcite precipitates which are deposited at fracture exposures in the galleries of the Äspö Hard Rock Laboratory. Precipitates from different fractures show large variations in concentration levels of trace elements, REE patterns, and activity and activity ratios of natural radionuclides, reflecting variations in physical, chemical and hydrological properties of the fractures. Formation of Fe-oxyhydroxide will occur e.g. in connection to the construction of a repository and at the geo-

sphere/biosphere interface. In connection with the precipitation of Fe-oxyhydroxide radionuclides can be retarded by co-precipitation and/or sorption.

The incorporation of REEs, Sr, T and U in calcite is significant. The precipitation rate influences (increases) the amount of Sr incorporated and probably other elements as well. Significant precipitation/dissolution of calcite can be expected in the vicinity of a disposal due to the temperature anomaly introduced by the radioactive waste. Uptake/release of radionuclides in connection to these processes should be considered in predictive modelling.

Clay minerals have high sorption capacity and are important in the retention of Cs and Sr (ion exchange) as well as of REEs, T, U and Ra. The sorption of Cs is characterized by at least to different processes; rapid surface adsorption and slower interlayer sorption. The size of the former, determined in the sequential extraction, is in agreement with that calculated from the Cs concentration in the groundwater and Kd-values determined in batch experiments. REEs and T seem to be strongly sorbed onto the clay mineral surfaces. T in such positions will effectively release radium to the groundwater through α-recoil.

The importance of clay minerals in radionuclide retention is emphazised by the results from this study; also small amounts of clay minerals in fractures and fracture zones can significantly influence the radionuclide migration. Accurate determination of quantities and types of clay minerals is therefore very important for radionuclide migration modelling.

SKB Technical Report 95-14

Consequences of using crushed crystalline rock as ballast in KBS-3 tunnels instead of rounded quartz particles

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February 1995

ABSTRACT

Backfills in KBS-3 tunnels with crushed crystalline rock as ballast (RB) can be prepared with the same gradation as "ideally" composed backfills with glacial sand as ballast (SB). RB backfills need more compaction energy to obtain the same density as SB backfills, but the physical properties at any density are fairly similar and RB backfills may therefore well be used although they have to be more effectively compacted on site than SB backfills. Such compaction can, however, be achieved in different ways. The chemical stability of both types is similar.

SKB Technical Report 95-15

Estimation of effective block conductivities based on discrete network analyses using data from the Äspö site

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September 1995

ABSTRACT

Numerical continuum codes may be used for assessing the role of regional groundwater flow in far-field safety analyses of a nuclear waste repository at depth. Such codes are sensitive to a number of factors, among them, the values of conductivity or permeability assigned to the grid cells or blocks in the model. The focus of this project is to develop and evaluate one alternative method based on Discrete Fracture Network (DFN) models to estimate block-scale permeability values for continuum codes such as HYDRASTAR, NAMMU and PHOENICS. Data from the Äspö HRL and surrounding area are used.

DFN models consist of discrete fractures in a three-dimensional volume of rock. The models constructed in this study are based upon measurements of fractures in the HRL and from the KAS series of boreholes. The model is stochastic, so it is possible to generate multiple fracture network realizations. A 30-m packer test is simulated in each realization, and the result is compared to the 30-m transient well tests carried out in KAS02 and KAS03. Only those realizations similar to the actual transient tests are retained. These realizations are then partitioned into a series of blocks ranging in size from 10 m to 50 m. The fractures in each block are converted into finite elements, and the effective block permeability between opposing faces is computed using the MAFIC code. These results are analyzed to determine population statistics and spatial correlation models for each block size, and for different DFN model fracture size assumptions. To evaluate this approach further, a 50-m NAMMU model was created out of 125 10-m blocks. The results of a simple flow experiment on this 50-m NAMMU model were compared to the results for the same boundary conditions on the parent 50-m MAFIC DFN model from which the 10-m blocks were derived. The comparison illustrates that there are a

number of important factors relating both to the selected continuum code and to the geometry of the fracture network that are important to consider if a stochastic continuum approach is used. The work also demonstrated that revision of the fracture mapping and well test protocols used in the HRL and elsewhere at Äspö could reduce the uncertainty for portions of the DFN models.

SKB Technical Report 95-16

Temperature conditions in the SKB study sites

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June 1995

ABSTRACT

This report is intended to serve as a basis for estimation of the range of ambient temperatures likely to be encountered in the final repository for spent fuel. In this report the borehole measurements from the SKB study sites, total 11 sites including Äspö Hard Rock Laboratory, are presented.

It is reasonable to assume that the SKB study sites constitute a representative range of bedrock temperature conditions likely to be encountered at the repository site. This is because of the geographical spread of the study sites, as well as the spread in rock types. The temperature at 500 m depth below the ground levels varies between 5.5 to 14.5°C between the various sites, while the temperature gradient varies between 9.5–15.5°C/km. If also those granite areas of anomalous high radiogenic heat production are included the plausible range of temperature gradient might be extended up to say 18°C/km.

SKB Technical Report 95-17

Measurements of colloid concentrations in the fracture zone, Äspö Hard Rock Laboratory, Sweden

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June 1995

ABSTRACT

The applicability of light scattering in combination with photon correlation spectroscopy (PCS) for determination of concentration and size distribution of colloidal matter in a deep groundwater was tested in situ and on-line at Äspö Hard Rock Laboratory, Oskarshamn, Sweden.

Well-defined reference colloids (sizes predominately in the range 50–500 nm) of Fe_2O_3, $Al(OH)_3$, SiO_2, kaolinite, illite and a high molecular humic acid (Aldrich) in aqueous media were used as model substances for calibration of the PCS instrument (signal vs. concentration). The intensity of scattered light was found to be dependent on the composition of the colloids. The concentration ranges where a quantitative determination of the colloids could be achieved were Fe_2O_3 0.03–2 mg/l, $Al(OH)_3$ 0.1–2 mg/l, SiO_2 0.1–7 mg/l, kaolinite 0.5–10 mg/l, illite 0.5–50 mg/l and humic acid 0.5–75 mg/l.

The colloid concentration in the rather saline groundwater (electrical conductivity of 660 mS/m corresponding to a total concentration of dissolved salts of 3900 mg/l) was below the detection limit for the used PCS equipment which corresponds to a colloid concentration not higher than 0.5 mg/l and probably below 0.1 mg/l according to the measurements on-line and in situ at Äspö and in comparison to the calibrations performed with reference colloids.

The results clearly demonstrated that the stability, concentration and composition of a colloidal-size suspended phase in an anoxic groundwater with high content of Fe(II) (0.3 mg/l), like the one in Äspö, is extremely sensitive to exposure to atmospheric conditions during sample handling and preparation. Diffusion of air into the closed measuring cuvette was enough to alter the colloid content significantly within 6 hr. A particle fraction with a size distribution in the range 170–700 nm was formed within 45 minutes when air was allowed to diffuse into the aqueous phase from the air filled upper part of the cuvette. The corresponding time to generate a significant colloid precipitate was less than 1 minute (size distribution range 100–600 nm) when a stream of air (1.5 ml) was bubbled through the water samples. The precipitated colloidal phase consisted of a mixture of ferric (hydr)oxide and calcium carbonate in all three cases.

SKB Technical Report 95-18

Thermal evidence of Caledonide foreland, molasse sedimentation in Fennoscandia

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November 1995

ABSTRACT

The Phanerozoic rocks present on the Fennoscandian Shield are dominantly of Cambrian to Silurian age. They represent a relatively thin sedimentary cover (less than 500 metres). The question is: why do we not see any remnants of younger sedimentary rocks? Did they ever exist, or have they been eroded, transported and redeposited elsewhere?

$\delta^{18}O$ and $\delta^{13}C$ analyses of Ordovician limestones from different places in Sweden and from the Oslo region in Norway show modification of their original marine signature according to the $\delta^{18}O$-values, while the $\delta^{13}C$-values generally are typical for marine limestones. In some cases the modifications can be explained by intrusions of dykes or by metamorphic events, but in most areas the redistribution of the oxygen isotopes indicates burial diagenesis. From a number of published investigations, raised temperatures at the present surface during the Late Palaeozoic, are indicated by different temperature indicators. We suggest that these increased temperatures were due to a sedimentary cover of mainly Devonian sediments deposited on top of the Cambrian-Silurian sequence. This palaeo-cover caused raised temperatures at the present rock surface, as shown by $\delta^{18}O$ values, as well as vitrinite reflectance, conodont colour alteration index (CAI) and by illite crystallinity in the Cambro-Silurian rocks.

In the Proterozoic basement, annealing of fission tracks in apatite and mobility of radiogenic lead also give evidence of increased temperatures. A model where the thickness of the Upper Palaeozoic cover of the Caledonian foreland is 2-4 kilometres thick is suggested. This cover mainly consisted of Late Silurian-Devonian erosional products from the Caledonides the latter which formed during a Silurian continent-continent collision. A major Permian to Triassic uplift and erosion reduced the cover significantly.

SKB Technical Report 95-19

Compaction of bentonite blocks. Development of technique for industrial production of blocks which are manageable by man

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April 1995

ABSTRACT

In this report a useful technique for producing compacted blocks of bentonite is described. The report only deals with the technique to produce uniaxially compacted blocks (weight of the blocks: 10-15 kg) which are manageable by man.

Tests for producing blocks with a weight of approximately 10 kg were carried out at Höganäs Bjuf AB in Bjuv. This industry is normally producing refractory bricks and other refractory products. The plant has facilities for handling large volumes of clay. Furthermore there are machines suitable for producing uniaxially compacted blocks. Performed tests at the plant show that it is possible to compact blocks with good quality. Best quality was reached with a coarsely ground bentonite at a water ratio of 20%. The compaction was performed with lubricated form and stepwise loading.

The tests at Höganäs Bjuf AB were preceded by tests in the laboratory. In these tests smaller samples were compacted for studying how different factors affect the quality of the samples (density, water ratio, homogeneity et cetera). The influence of following factors was studied:

- water ratio of bentonite,
- bentonite type and granulometry,
- compaction pressure,
- compaction rate,
- form geometry,
- form lubrication,
- form heating.

The results from these tests were used to modify and optimize the technique in the factory.

SKB Technical Report 95-20

Modelling of the physical behaviour of water saturated clay barriers. Laboratory tests, material models and finite element application

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September 1995

ABSTRACT

This report deals with laboratory testing and modelling of the thermo-hydro-mechanical (THM) properties of water saturated bentonite based buffer materials. A number of different laboratory tests have been performed and the results are accounted for. These test results have lead to a tentative material model, consisting of several sub-models, which is described in the report. The tentative model has partly been adapted to the material models available in the finite element code ABAQUS and partly been implemented and incorporated in the code. The model that can be used for ABAQUS calculations agrees with the tentative model with a few exceptions.

The model has been used in a number of verification calculations, simulating different laboratory tests, and the results have been compared with actual measurements. These calculations show that the model generally can be used for THM calculations of the behaviour of water saturated buffer materials, but also that there is still a lack of some understanding. It is concluded that the available model is relevant for the required predictions of the THM behaviour but that a further improvement of the model is desirable.

SKB Technical Report 95-21

Conceptual model for concrete long time degradation in a deep nuclear waste repository

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February 1994

ABSTRACT

This report is mainly a state-of-the-art report of concrete long time durability in the environment expected in a deep site underground nuclear waste repository in Swedish crystalline bedrock. Some of the results referred to have been produced during research at CBI, Stockholm.

The report treats how the concrete and the surrounding groundwater will interact and how they will be affected by cement chemistry, type of aggregate etc. The different mechanisms for concrete alteration treated in the report include sulphate attack, carbonation, chloride attack, alkali-silica reaction and leaching phenomena.

In a long time perspective, the chemical alterations in concrete is mainly governed by the surrounding groundwater composition. After closure the composition of the groundwater will change character from a modified meteoric to a saline composition. Therefore two different simulated groundwater compositions (ALLARD and NASK water) have been used in modelling the chemical interaction between concrete and groundwater.

The report also includes a study of old and historical concrete which show observations concerning recrystallization phenomena in concrete.

A conceptual model for concrete degradation in a KBS-3 repository is presented.

SKB Technical Report 95-22

The use of interaction matrices for identification, structuring and ranking of FEPs in a repository system. Application on the far-field of a deep geological repository for spent fuel

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November 1995

ABSTRACT

The basic devise in the Rock Engineering Systems (RES) approach, the interaction matrix, has been used to identify, structure and rank Features, Events, and Processes (FEPs) describing barrier performance and radionuclide behaviour in the far-field of a deep geological repository for spent fuel. The result is a first version of the Process System, PS, for the far-field of a deep repository, structured in an interaction matrix with supporting documen-

tation. The documentation is compiled in databases, one containing matrix specific information and one containing general FEP descriptions.

The study has shown that an interaction matrix is feasible to use both for the structuring of the PS and for visualisation of the PS. The developed documentation system increases the transparency of the system description and makes it possible to trace back the judgements made during the construction of the matrix. This will facilitate review work and future revisions as well as consistent treatment of different issues in the system.

This study is a first step in the application of a systematic method to establish a structured description of the PS for a deep repository for spent fuel. The work could be seen as a part of the preparation for the forthcoming performance and safety analyses. The next step would be to develop the PS for the remaining parts of the repository system to the same level as has been done for the far-field system. Before the PS is evaluated for different selected system premises, a scientific review of the contents of the PS for the whole repository system would be beneficial.

SKB Technical Report 95-23

Spent nuclear fuel. A review of properties of possible relevance to corrosion processes

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April 1995

ABSTRACT

The report reviews the properties of spent fuel which are considered to be of most importance in determining the corrosion behaviour in groundwaters.

Pellet cracking and fragment size distribution are discussed, together with the available results of specific surface area measurements on spent fuel. With respect to the importance of fuel microstructure, emphasis is placed on recent work on the so-called structural rim effect, which consists of the formation of a zone of high porosity, and the polygonization of fuel grains to form many subgrains, at the pellet rim, and appears to be initiated when the average pellet burnup exceeds a threshold of about 40 MWd/kgU.

Due to neutron spectrum effects, the pellet rim is also associated with the buildup of plutonium and other actinides, which results in an enhanced local burnup and specific activity of both beta-gamma and alpha radiation, thus representing a greater potential for radiolysis effects in ingressed groundwater. The report presents and discusses the results of quantitative determination of the radial profiles of burnup and alpha activity on spent fuel with average burnups from 21.2 to 49.0 MWd/kgU.

In addition to the radial variation of fission product and actinide inventories due to the effects mentioned above, migration, redistribution and release of some fission products can occur during reactor irradiation, and the report concludes with a short review of these processes.

SKB Technical Report 95-24

Studies of colloids and their importance for repository performance assessment

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December 1995

ABSTRACT

The processes, parameters and data used to evaluate the potential of nuclide transport by a colloid facilitated mechanism are reviewed and discussed in this report. Both steady-state (present situation) and possible future non-steady-state (changing situation) hydrogeochemistry in the geosphere are covered.

In the steady-state scenario, the colloid (clay, silica, iron(III) hydroxide) concentration is around 20-45 $\mu g \cdot l^{-1}$ which is considered to be a low value. The low colloid concentration is justified by the large attachment factor to the rock which reduces the stability of the colloids in the aquifer. Both reversible and irreversible sorption processes are reviewed.

In the non-steady-state scenario, changes of hydrogeochemical properties may induce larger colloid concentrations. This increase of concentration is however limited and relaxation is always observed after any change. Emphasis is placed on the glaciation-deglaciation scenario.

SKB Technical Report 95-25

Sulphate reduction in the Äspö HRL tunnel

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December 1995

ABSTRACT

Sulphate reduction is a well studied process in sea bed sediments, but is less known in deep groundwaters. Sulphate-reducing bacteria can bring about the reduction of sulphate to sulphide using for example organic substances present in natural groundwater as reducing agents. Sulphide production is of particular interest for disposal of spent fuel in copper canisters because sulphide is in fact the only substance present in deep groundwater that will cause the corrosion of copper. Oxygen, another copper-corrodant is not present in deep groundwater and sulphate will not react with copper unless microbes reduce it to sulphide.

Evidence and indications of sulphate reduction based on geological, hydrogeological, groundwater, isotope and microbial data gathered in and around the Äspö Hard Rock Laboratory (HRL) tunnel have been evaluated.

This integrated investigation showed that sulphate reduction had taken place in the past but is most likely also an ongoing process. Anaerobic sulphate-reducing bacteria can live in marine sediments, in the tunnel sections under the sea and in deep groundwaters, since there is no access to oxygen. The sulphate-reducing bacteria seem to thrive when the Cl^- concentration of the groundwater is 4000–6000 mg/l. Sulphate reduction is an in situ process but the resulting hydrogen-sulphide-rich water can be transported to other locations. A more vigorous sulphate reduction takes place when the organic content in the groundwater is high (10 mg/l, DOC) which is the case in the sediments and in the groundwaters under the sea. Some bacteria use hydrogen (geogas) as an electron donor instead of organic carbon and can therefore live in deep environments where the access to organic material is limited. The sulphate-reducing bacteria seem to adapt to changing flow situations caused by the tunnel construction relatively fast. Sulphate reduction seems to have occurred and will probably occur where conditions are favourable for the sulphate-reducing bacteria such as anaerobic brackish groundwater with dissolved sulphate and organic carbon or hydrogen.

SKB Technical Report 95-26

The Äspö redox investigations in block scale. Project summary and implications for repository performance assessment

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November 1995

ABSTRACT

On March 13, 1991, tunnel construction at the Äspö Hard Rock Laboratory opened a vertical fracture zone at a depth of 70 meters. Three weeks later, a sharp dilution front was observed in the originally saline groundwater. This signaled the arrival of shallow surface water at the 70 m depth. There was a brief oxidizing disturbance at that time as indicated by the disappearance of dissolved iron from the inflowing groundwater. These results were expected based on earlier predictions of surface water and oxygen breakthrough times. After this initial period, however, the fracture zone remained persistently anoxic.

Intensive anaerobic respiration resulted from the increased recharge. Rather than input of molecular oxygen, there was an inflow of reducing capacity in the form of young organic carbon. Microbial processes were extremely effective in using the carbon to scavenge oxygen and other oxidants from the shallow bedrock environment. Because the thin soils at this site were apparently so effective as a source of organic carbon, these processes may be even more pronounced at sites with more extensive soil cover.

SKB Technical Report 95-27

Survival of bacteria in nuclear waste buffer materials. The influence of nutrients, temperature and water activity

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ABSTRACT

The concept of deep geological disposal of spent fuel is common to many national nuclear waste programs. Long-lived radioactive waste will be encapsulated in canisters made of corrosion resistant materials e.g. copper and buried several hundred meters below ground in a geological formation. Different types of compacted bentonite clay, or mixtures with sand, will be placed as a buffer around the waste canisters. A major concern for the performance of the canisters is that sulphate-reducing bacteria (SRB) may be present in the clay and induce corrosion by production of hydrogen sulphide. This report presents data on viable counts of SRB in the bedrock of Äspö hard rock laboratory. A theoretical background on the concept water activity is given, together with basic information about SRB. Some results on microbial populations from a full scale buffer test in Canada is presented. These results suggested water activity to be a strong limiting factor for survival of bacteria in compacted bentonite. As a consequence, experiments were set up to investigate the effect from water activity on survival of SERB in bentonite. Here we show that survival of SERB in bentonite depends on the availability of water and that compacting a high quality bentonite to a density of 2.0 g/cm^3, corresponding to a water activity (a_w) of 0.96, prevented SERB from surviving in the clay.

SKB Technical Report 95-28

DECOVALEX I – Test Case 2: Calculation of the Fanay-Augères THM Test – Thermomechanical modelling of a fractured rock volume

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ABSTRACT

A large scale thermo-mechanical test in the French uranium mine Fanay Augères at the depth 100 m has been simulated with the finite element code ABAQUS. The calculations have been compared with measurements of rock displacements.

The calculations included excavation of the test room and the heating and cooling periods of the test. The rock has been modelled both as linear elastic and as fractured with elastic-plastic properties of the fractures.

A sensitivity analysis of factors influencing the results were made with elastic calculations. It showed that the boundary conditions and the properties of the surrounding rock had no influence on the effect of the heating but a substantial influence on the effect of the excavation.

Comparisons between calculated and measured results of the heating and cooling test showed that the introduction of fractures did not yield more accurate results than those obtained with the fracture free linear elastic model.

SKB Technical Report 95-29

DECOVALEX I – Test Case 3: Calculation of the Big Ben Experiment – Coupled modelling of the thermal, mechanical and hydraulic behaviour of water-unsaturated buffer material in a simulated deposition hole

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ABSTRACT

PNC:s large scale laboratory test with an artificial deposition hole, named BIG-BEN, has been simulated with finite element calculations with the code ABAQUS. The test comprised water uptake from an artificial rock and heating of a canister in a deposition hole with the diameter 1 m during 5 months. Water content, pore pressure, and total pressure in the buffer was measured during the test.

The given data of the material properties were supplemented with results from own laboratory tests in order to determine parameters required for the calculation. The vapour flow process, which is not included in ABAQUS, was implemented and added to the code. After calibration of the properties of the buffer material, a completely coupled thermo-hydro-mechanical calculation of the test was done. The calculated thermal and hydraulic results were in good agreement with measured values while the prediction of the mechanical response was less good.

SKB Technical Report 95-30

DECOVALEX I – Bench-Mark Test 3: Thermo-hydro-mechanical modelling

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ABSTRACT

The second phase of DECOVALEX (DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation) involved continuation of the problem(s) initiated in phase I and one new problem, Bench-Mark Test 3 (BMT3). Bench-Mark Test 3 was formulated by the French research teams supported by ANDRA and CEC.

The bench-mark test concerns the excavation of a tunnel, located 500 m below the ground surface, and the establishment of mechanical equilibrium and steady-state fluid flow. Following this, a thermal heating due to nuclear waste, stored in a borehole below the tunnel, was simulated. The results are reported at (1) 30 days after tunnel excavation, (2) steady state, (3) one year after thermal heating, and (4) at the time of maximum temperature.

The problem specification included the excavation and waste geometry, material properties for intact rock and joints, location of more than 6500 joints observed in the 50 by 50 m area, and calculated hydraulic conductivities. However, due to the large number of joints and the lack of dominating orientations, it was decided to treat the problem as a continuum using the computer code FLAC /Itasca, 1992a/.

The problem was modeled using a vertical symmetry plane through the tunnel and the borehole. The rock mass was modeled as an elastic – ideally plastic material with a Mohr-Coulomb failure criterion. Equivalent continuum material properties were calculated using Rock Mass Rating by Bieniawski, (1979) and Rock Mass Strength by Stille et al., (1982).

Flow equilibrium was obtained approximately 40 days after the opening of the tunnel. Since the hydraulic conductivity was set to be stress dependent, a noticeable difference in the horizontal and vertical conductivity and flow was observed. After 40 days, an oedometer-type consolidation of the model was observed.

Approximately four years after the initiation of the heat source, a maximum temperature of 171°C was obtained. The stress-dependent hydraulic conductivity and the temperature-dependent dynamic viscosity caused minor changes to the flow pattern.

The specified mechanical boundary conditions imply that the tunnel is part of a system of parallel tunnels (which is the case for many storage concepts). However, the fixed temperature at the top boundary maintains the temperature below the temperature anticipated for an equivalent repository. The combination of mechanical and hydraulic boundary conditions cause the model to behave like an oedometer test in which the consolidation rate goes asymptotically to zero.

SKB Technical Report 95-31

DECOVALEX I – Test Case 1: Coupled stress-flow model

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ABSTRACT

This report presents the results of the Coupled Stress-Flow Model, TC1 (Test Case 1) of Decovalex. The model simulates the fourth loading cycle of a Coupled Stress-Flow Test (CSFT) and subsequent shearing up to and beyond peak shear resistance.

The first loading sequence, termed “Sequence A”, consists of seven normal loading steps: 0, 5, 15, 25, 15, 5 and 0 MPa. The second loading sequence termed “Sequence B”, consists of the following eight steps: unstressed state, normal boundary loading of 25 MPa (no shearing), and then shearing of 0.5, 0.8, 2.0, 4.0, 2.0 and 0 mm.

According to the problem definition, Decovalex (1991), the shearing was to be made by changing the boundary stresses while keeping the joint normal stress at a constant value of 25 MPa. It was found, however, that an applied boundary stress of 25 MPa produces an average joint normal stress, at five monitoring points along the joint, that is significantly greater. Further, it is not possible to specify stress boundary conditions to produce a specific amount of shear displacement after the joint reaches peak shear strength. Therefore, displacement boundary conditions were applied so that the average joint normal stress from the 25 MPa normal boundary loading step was kept constant during shearing.

The normal stress along the joint is not constant, but higher at the joint ends due to bending in the model. A higher normal stress results in a smaller aperture at the

ends of the joint. The ends of the joint will control the flow rate for the entire joint. If the flow rate is used to determine an average joint aperture, this will lead to errors.

Two different options regarding the rock joint behavior were modeled in accordance with the problem definition. In Option 1 a linear elastic joint model with a Coulomb slip criterion was used. In Option 2 a non-linear empirical (i.e. Barton-Bandis) joint model was used.

The hydraulic condition during both Loading Sequence A and B was a constant head of 5 m at the inlet point and 0 m at the outlet point.

All model runs presented in this report were performed using the two-dimensional distinct element computer code UDEC, version 1.8.

SKB Technical Report 95-32

Partitioning and transmutation (P&T) 1995. A review of the current state of the art

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ABSTRACT

The recent development in the field of partitioning and transmutation (P&T) is reviewed and evaluated. Current national and international R&D efforts are summarized. It is concluded that P&T is scientifically feasible and the recent R&D in the field of P&T makes it more technical mature than a few years ago. Nuclear transmutation with energy production is feasible in nuclear reactors where fast and thermal breeder reactors are the most efficient for transmutation purposes. The operation of subcritical nuclear reactors by high current proton accelerators that generate neutrons in a spallation target is also an interesting option for transmutation and energy production, that has to be more carefully evaluated. These accelerator-driven systems (ADS) are probably the only solution for the transmutation of long-lived fission products with small neutron capture cross sections and actinide isotopes with rather small fission cross sections.

The requirements on the separation chemistry in the partitioning process depends on the transmutation strategy chosen. Recent developments in aqueous based separation chemistry opens some interesting possibilities to meet some of the requirements, such as separation of different actinides and some fission products and reduction of secondary waste streams. In the advanced accelerator-driven transmutation systems proposed, liquid fuels such as molten salt are considered. The partitioning processes that can be used for these types of fuels will, however, require a long term research program. The possibility to use centrifuge separation is an interesting partitioning option that recently has been proposed.

P&T is a complex issue with regard to environmental, safety, technological, economical, political and public acceptance aspects. In some countries (Japan, France and the Russian Federation) it has, however, received political or institutional backing as a complementary future nuclear fuel strategy. At present there seems to be no economical gain and only an insignificant reduction in future radiation doses from P&T as compared to the reference concept of the closed fuel cycle and direct disposal of spent nuclear fuel. However, some of these conclusions can be changed by future long-term research and/or by a change in the economical situation.

SKB Technical Report 95-33

Geohydrological simulation of a deep coastal repository

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ABSTRACT

This conceptual-numerical study treats the dewatering and resaturation phases associated with the construction, use and closure of a coastal nuclear waste repository located at depth in sparsely fractured Baltic Shield rocks. The main objective is to simulate the extent and duration of saline intrusion for a reasonable set of geohydrological assumptions. Long-term changes in the chemical environment associated with saline intrusion may affect the properties of the buffer zone material (bentonite). The first part of the study deals with history matching of a simple model geometry and the second part treats the dewatering and resaturation phases of the simulated repository. The history matching supports the standpoint that the occurrence of saline ground water reflects an ongoing but incomplete Holocene flushing of the Baltic Shield. The drawdown after fifty years of dewatering is highly dependent on the permeability of the excavated damaged zone. If the permeability close the repository is unaltered

the entire region between the top side of the model and the repository is more or less partially saturated at the end of the simulation period. The simulations of a fifty year long recovery period suggest that the distribution between fresh and saline ground waters may be quite close to the conditions prior to the dewatering phase already after fifty years of closure despite an incomplete pressure recovery, which is an interesting result considering the objective of the study.

SKB Technical Report 95-34

General Siting Study 95. Siting of a deep repository for spent nuclear fuel

SKB

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ABSTRACT

One of SKB's tasks is to compile a basis for the siting of a deep repository for Swedish spent nuclear fuel and other long-lived nuclear waste. The aim of the siting work is to compile all of the information which is necessary for a site to be selected and to obtain a siting permit in order to start construction of the deep repository through the detailed characterization of the selected site.

In the government decision of May 18, 1995, concerning SKB's programme, the government made the following statement with regard to 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, contains SKB's overall report on general siting studies carried out on a national scale in accordance with the government decision.

This report is mainly based on the extensive background material that SKB prepared as a part of the research and development work which has been conducted since the late 1970's.

The target group for this report is assumed to comprise experts within the different related areas as well as decision-makers and members of the general public who are particularly interested in these issues. For this reason, the report must contain information that is scientifically cogent and concise. At the same time the report must provide ample background information and easy-to-read background information. These requirements are not always compatible. Consequently, the reader, depending upon his/her specific background, may have to consult background technical reports or general material in order to obtain maximum benefit of the report.

In General Siting Study 95, important siting factors have been described and applied on a national scale. For each such siting factor, separate conclusions have been reported in detail.

All of the factors, of which the long-term radiological safety is the central factor, must be taken into account in the siting of the deep repository. However, to make an overall evaluation of this factor, site-specific data on the bedrock must be available. Such data can only be obtained by carrying out extensive investigations at sites which can only be selected on the basis of partially incomplete information. This is a characteristic which distinguishes the siting of underground facilities in general, and a deep repository in particular, from other types of industrial siting, where information on all of the vital factors is relatively easy to obtain.

An important part of the work on preparing General Siting Study 95 has been to explore the possibilities and limitations which are related to a general siting study on a national scale.

A general siting study on a national scale cannot provide information on scientific, technical and societal factors with the necessary level of detail in order to identify sites which are suitable for the siting of a deep repository. The information presented in this study in general also covers conditions at the surface and not at the depths envisaged for the deep repository, 400–700 m below the surface. Thus, the suitability of a site is best evaluated through feasibility studies and site investigations and evaluations which must be carried out in connection with pertinent permitting of regulatory authorities.

However, General Siting Study 95 makes it possible to identify areas which are less suitable, or of less interest. However, areas which are of less interest, on a national scale, cannot be excluded since, on the local scale of a particular area, there may be many sites of interest which may be omitted when generalizations are made on the national scale. The general siting study also examines several scientific, technical and societal conditions in different parts of Sweden. These conditions provide a basis for assessing interest in, and for carrying out more detailed siting studies (feasibility studies).

A number of national databases which, in one way or another are, or may be significant for the siting factor of long-term radiological **Safety** have been described and evaluated. This includes rock types, topography, well data and the Highest Shoreline. Furthermore, geological deformation zones, lineaments, future ice ages and earthquakes have been included. The possibility of unintentional intrusion and the selection of discharge areas are also discussed.

The siting factor, **Technology**, describes how the feasibility may be affected by different conditions. After the waste is encapsulated, it must be transported to the deep repository. Various alternatives exist, depending on the location of the deep repository in relation to the encapsulation plant. There is no actual restriction on how the deep repository can be located in terms of the means of transportation. The preferred means of transportation would require railroads or harbours. It is easier to carry out investigations of the bedrock, facility design and safety assessment if the geoscientific conditions of the site are easy to understand and interpret. A number of attempts have been made to, on a national scale, assess possible regional differences in interpretability. Even if it is suitable to primarily seek out sites which are easy to interpret, a similar reliability in results might be obtained at a more complex site, although more extensive investigation of the site would be required. This is thus a question of optimization, among other issues. The deep repository, as an engineering and construction project, benefit from the same factors which benefit the long-term radiological safety. Thus, there is no conflict between safety and simplicity of implementation with regard to the actual construction work.

The factor, **Land and Environment,** must also be taken into account in the siting of the repository, bearing in mind the stipulations of the Act concerning the Management of Natural Resources etc. Furthermore, there are areas which are protected against exploitation by law, e.g. national parks, nature reserves etc.

With regard to the siting factor, **Society**, is concerned there are a large number of conditions which are treated in the report which can largely be evaluated on a more detailed scale in connection with feasibility studies and site investigations.

The goals of General Siting Study 95 have been described in the RD&D Programme 92 Supplement:

- *"In a general fashion (on a national scale), shed light on conditions of interest for determining which parts of the country are unsuitable, interesting or suitable for siting a deep repository.*

- *Yield data for determining SKB's interest in feasibility studies in different regions or municipalities.*

- *Provide indications of what must be particularly taken into account and studied in connection with continued more detailed studies within suitable or particularly interesting areas.*

- *Yield data for putting coming site selection in its national and regional context."*

The conclusions from this work is now related to the established goals.

"In a general fashion (on a national scale), shed light on conditions of interest for determining which parts of the country are unsuitable, interesting or suitable for siting a deep repository."

An overall evaluation of the applicable siting factors shows that it is unsuitable to locate the deep repository in the Scandinavian mountain range, Skåne and Gotland, primarily for geological reasons. Furthermore, the Scandinavian mountain range is an area of national interest with regard to nature conservation and outdoor activities. Siting of the repository in the bedrock below Öland is considered to be technically possible, although unsuitable in terms of the management regulations of the Act concerning the Management of Natural Resources etc.

Siting of the repository in areas which are directly protected by law is neither necessary or desirable and must be avoided.

Areas which are of national interest in other contexts cannot be simply excluded. However, work should not focus on siting the deep repository in such areas or should at least focus on siting and designing the deep repository so as not to negatively affect the aim of protecting areas which are of national interest.

The unnecessary use, or blocking of natural resources must, if possible, be avoided. Areas where unusual rock types occur or where there is a possibility of ore, especially bedrock consisting of acid volcanic rock types, are therefore of less interest. By avoiding these rock types, there is less possibility of future, unintentional intrusion into the deep repository as well.

The conclusion which was previously drawn by SKB that there are many areas in Sweden which appear to be suitable for the siting of a deep repository has not been altered by General Siting Study 95. In the future, siting should also focus on bedrock commonly found in Sweden, preferably more or less transformed granite-like rock types, or old, highly metamorphosised sedimentary bedrock. This type of "interesting" bedrock exists in large parts of Sweden.

It is not necessary to exclude areas containing gabbro, or areas where the bedrock is covered by sedimentary rock, in connection with siting. These areas can assessed especially if feasibility studies will be carried out in municipalities with this type of bedrock.

It is technically possible to locate the repository beneath large lakes or beneath the sea. However, there is no particular reason to seek out or avoid such a siting.

The long-term radiological safety is not the only factor taken into consideration in the siting of the deep repository but also the practical implementation of the work. Consequently, areas with bedrock that is easy to interpret as well as areas where the rock is relatively low-conductive are of interest. In general, there is no conflict between the factors which are favourable to the long-term safety and the simple, technical implementation of work. It is most suitable to evaluate the question of the variation of the bedrock and other properties in connection with more detailed studies.

As far as transportation and communication are concerned, the availability of harbours, railroads or airports is good. Thus, on the national scale, there is no real limitation of possible areas in terms of these factors.

"Yield data for determining SKB's interest in feasibility studies in different regions or municipalities."

The siting factors described in this report have already been largely applied in the feasibility study which was carried out in Storuman and in the ongoing (October 1995) feasibility study in Malå. These factors have also been applied to municipalities which have evinced interest in initiating feasibility studies but which have been considered to be less interesting by SKB.

In General Siting Study 95, it is emphasized that many of the siting factors should, above all, be applied on a local scale in connection with feasibility studies and site investigations. This study is one of the bases which can be used in connection with the planning of work to find a suitable site for the repository.

For general technical and societal reasons, it is of interest to conduct feasibility studies in municipalities with existing nuclear activities: Nyköping, Oskarshamn, Varberg and Östhammar, which has been previously reported.

Besides the already completed, ongoing or planned feasibility studies, it is suitable to carry out one additional or a few additional feasibility studies. It should be an advantage to further work and with regard to discussions with different communities that an overall report on the general siting studies which have been carried out now exists. The report makes it easier than before for all concerned to understand the background and general conditions that exist for siting work in different parts of the country.

As a basis for concrete discussions with different communities concerning feasibility studies, SKB intends to use regional general siting studies which are based on this study. In such regional general siting studies, it is of particular interest to preliminarily identify areas within one region which are expected to have suitable conditions with regard to industrial experience, availability of industrial land and proximity to harbours or railroads as well as where it is expected that there is bedrock with a good potential for fulfilling technical and safety-related requirements.

"Provide indications of what must be particularly taken into account and studied in connection with continued more detailed studies within suitable or particularly interesting areas."

General Siting Study 95 provides such indications, in several respects. Such factors include possible conflicts of interest in connection with land use in expansive areas of Sweden, the occurrence of ore mineralizations etc.

"Yield data for putting coming site selection in its national and regional context."

General Siting Study 95 contains background material and a description of general conditions in different parts of Sweden which make it possible to, in connection with site licensing (based on 5–10 feasibility studies and 2 site investigations), evaluate whether the selected site is acceptable from different aspects and whether it is based on a sufficiently broad basis of information.

The background material often contains large variations on a local scale, which cannot be specified on a national scale. Thus, in all probability, suitable areas can be found both in regions which, on a national scale, appear to be of greater interest, as well as in regions which, on a national scale, appear to be of lesser interest. This general approach can be applied to several of the databases or situations which are described in this report.

Regional and local conditions will have to be more closely investigated in feasibility studies and site investigations in each relevant case.

SKB Technical Report 95-35

A selected thermodynamic database for REE to be used in HLNW performance assessment exercises

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ABSTRACT

A selected thermodynamic database for the Rare Earth Elements (REE) to be used in the safety assessment of high-level nuclear waste has been compiled. Thermodynamic data for the aqueous species of the REE with the most important ligands relevant for granitic groundwater conditions have been selected and validated. The dominant soluble species under repository conditions are the carbonate complexes of REE. The solubilities of the oxides, hydroxides, carbonates, hydroxycarbonates, phosphates and other important solids have been selected and validated. Solubilities and solubility limiting solids in repository conditions have been estimated with the selected database. At the initial stages of fuel dissolution, the $UO_2(s)$ matrix dissolution will determine the concentra-

tions of REE. Later on, solid phosphates, hydroxycarbonates and carbonates may limit their solubility. Recommendations for further studies on important systems in repository conditions have been presented.

SKB Technical Report 95-36

Reconstructing the tectonic history of Fennoscandia from its margins: The past 100 million years

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ABSTRACT

In the absence of onland late Mesozoic and Cenozoic geological formations the tectonic history of the Baltic Shield over the past 100 million years can most readily be reconstructed from the thick sedimentary basins that surround Fennoscandia on three sides. Changing patterns of sediment thickness accompanying active tectonics, as observed on high resolution commercial multichannel seismic reflection lines, record the boundary conditions of deformation internal to the Baltic Shield. Tectonic activity around Fennoscandia through this period has been diverse but can be divided into four main periods: a) pre North Atlantic spreading ridge (100–60Ma) when transpressional deformation on the southern margins of Fennoscandia and transtensional activity to the west was associated with a NNE-SSW maximum compressive stress direction: b) the creation of the spreading ridge (60–45Ma) when there was rifting along the western margin; c) the re-arrangement of spreading axes (45–25Ma) when there was a radial compression around Fennoscandia, and d) the re-emergence of the Iceland hot-spot (25–0Ma) when the stress-field has come to accord with ridge or plume 'push'. Since 60Ma the Alpine plate boundary has had little influence on Fennoscandia. The highest levels of deformation on the margins of Fennoscandia were achieved around 85Ma, 60–55Ma and 15–10Ma, with strain-rates around 10^{-9}/year. Within the Baltic Shield long term strainrates have been around 10^{-11}/year, with little evidence for significant deformations passing into the shield from the margins. Fennoscandian Border Zone activity, which was prominent from 90–60Ma, was largely abandoned following the creation of the Norwegian Sea spreading ridge, and with the exception of the Lofoten margin, there is subsequently very little evidence for deformation passing into Fennoscandia. Renewal of modest compressional deformation in the Voering Basin suggests that the 'Current Tectonic Regime' is of Quaternary age although the orientation of the major stress axis has remained approximately consistent since around 10Ma. The past pattern of changes suggest that in the geological near-future variations are to be anticipated in the magnitude rather than the orientation of stresses.