SKB

TECHNICAL REPORT

96-25

SKB ANNUAL REPORT 1996

Including Summaries of Technical Reports
Issued during 1996

Stockholm, May 1997

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FOREWORD

The Annual Report on SKB's activities during 1996 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 – an encapsulation plant and a 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 within 12-15 years. Work is undertaken for the development of encapsulation technology on an industrial scale and design of an encapsulation plant. A Canister Laboratory is being built for this purpose. The siting process for the final repository for spent fuel has started with feasibility studies in a few Swedish municipalities to evaluate the potential technical conditions and requirements and influence of the region.

International co-operation and exchange of information in all fields of the back-end of the nuclear fuel cycle is 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 May 1997

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 1996.

Lectures and publications during 1996 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 postgraduate theses based on work financially supported by SKB.

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

SKB is jointly owned by the four Swedish utilities that own and operate nuclear power plants. The task of SKB is to transport, store and dispose of the spent nuclear fuel and radioactive wastes from the nuclear power plants and to perform the research and development and other measures necessary for this work.

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 1996 in total around 2 500 tonnes of spent fuel (uranium weight) have been received.

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. SFR has a current capacity of about 60 000 m^3 of waste. At the end of 1996 a total of 21 000 m^3 of waste have been deposited in SFR.

Transportation from the nuclear sites to CLAB and SFR is made by a specially designed ship, M/S Sigyn.

SKB is in charge of a comprehensive research, development and demonstration programme on geological disposal of nuclear waste. The total cost for RD&D during 1996 was 124.0 MSEK.

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 1997 there are nine 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 1996 the basic design work was continuing for the facility. Also development work for the manufacturing and sealing of the copper canister was performed. A building was procured in Oskarshamn for the installation of a canister laboratory for further development and testing of the canister welding and non-destructive examination procedures. Most of the design work for the laboratory was also completed.

Siting activities for a deep repository included feasibility studies in the municipalities of Östhammar and Nyköping. Technical, geoscientific and socioeconomic studies were performed. The municipality of Malå conducted an independent evaluation of the SKB feasibility study which was completed during 1995. The municipal council of Oskarshamn decided to participate in a feasibility study for that community. Extensive local involvement and discussions has been and is an important part of the siting activities. As a background for siting evaluations and as a supplement to the previous General Siting Study '95 SKB started some regional siting studies.

Cost calculations for the total nuclear waste management system, including decommissioning of all reactors, are updated annually. The total cost is estimated to 53 billion SEK for 25 years of operation of each of the nuclear power reactors; the calculational basis prescribed by the revised legislation.

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 1996 more than 140 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 1996 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 1996

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. The main data of the 12 units are shown below and their locations are shown in Figure 1-1. The nuclear power plants generated about 52% of the total Swedish electric energy produced in 1996.

Swedish reactors

Reactor	Type	Power MW_e	Commercial operation	Energy availability in 1996 %
Barsebäck 1	BWR	600	1975	82,0
Barsebäck 2	BWR	600	1977	73,2
Forsmark 1	BWR	970	1980	94,8
Forsmark 2	BWR	970	1981	91,4
Forsmark 3	BWR	1160	1985	89,1
Oskarshamn 1	BWR	445	1972	61,4
Oskarshamn 2	BWR	605	1975	72,9
Oskarshamn 3	BWR	1160	1985	85,1
Ringhals 1	BWR	830	1976	90,3
Ringhals 2	PWR	875	1975	85,4
Ringhals 3	PWR	915	1981	92,8
Ringhals 4	PWR	915	1983	91,0

1.2 LEGAL AND ORGANIZATIONAL FRAMEWORK

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

1. Vattenfall AB is the largest electricity producer in Sweden and owns the Ringhals plant.
2. Barsebäck Kraft AB (subsidiary of Sydkraft AB) is the owner of the Barsebäck plant.
3. OKG AB is the owner of the Oskarshamn plant. Sydkraft is the major shareholder of OKG.
4. 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.

The total central staff of SKB is about 100 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.

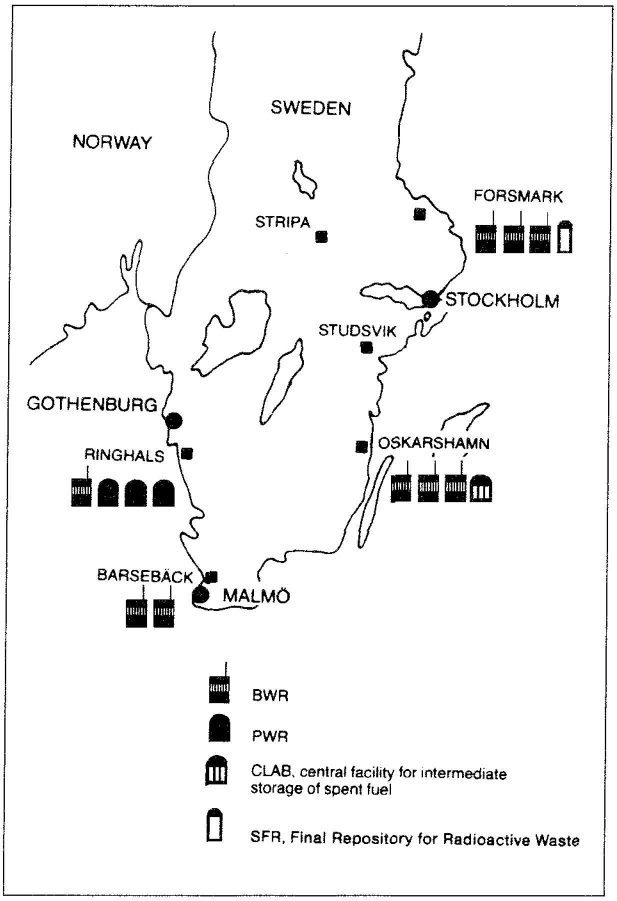

Figure 1-1. The Swedish nuclear power programme.

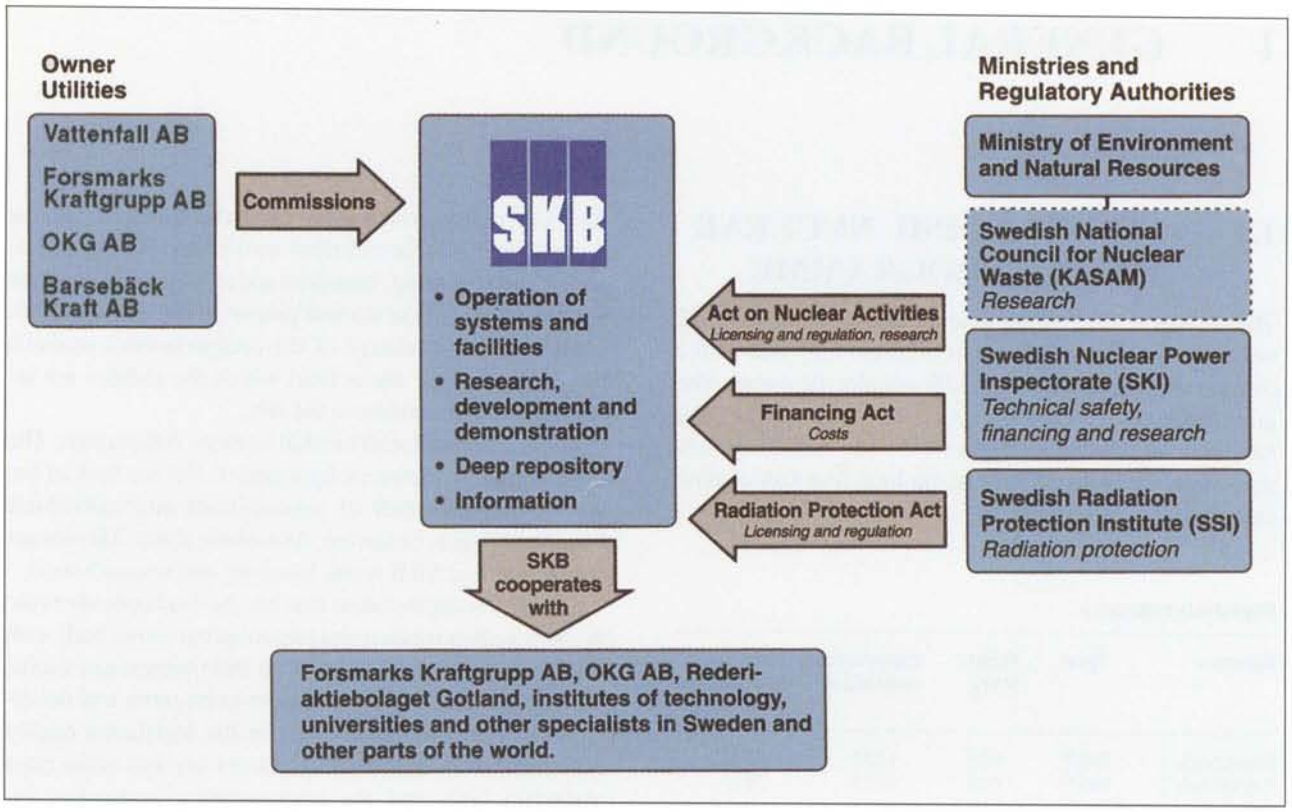

Figure 1-2. *Legal framework for activities of SKB.*

There are three important laws which regulate the nuclear activities.

1. The Act on Nuclear Activities.
2. The Act on the Financing of Future Expenses for Spent Nuclear Fuel etc.
3. 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 through 1996 been 0.019 SEK per kWh. A revision of the act has been taken by the Swedish parliament and will enter into force from 1997. The changes are explained in Chapter 8.

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.

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.

Directly under the ministry of environment works the Swedish National Council for Nuclear Waste (KASAM), which consists of a group of scientists appointed by the government.

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 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 arise 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 amount of waste disposed was by the end of 1996 about 21 000 m^3 and 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 had by the end of 1996 about 2500 tonnes of spent fuel in storage and a current capacity of 5 000 tonnes. An expansion to 8 000 tonnes capacity is planned for 2004.

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-Programmes 92 and 95 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 at the earliest. The next stage for a full size 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.

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	Transuranium-bearing waste	Waste from the Studsvik research facilities	Solidified in concrete	Low to medium level. Contains some long-lived nuclides	c. 2 000 m^3
3	Core components and internals	Scrap metal from inside reactors 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. 50 000 m^3
5	Decommissioning waste	From dismantling of nuclear facilities	Untreated for the most part	Low to medium level. Short-lived	100 000 – 150 000 m^3

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

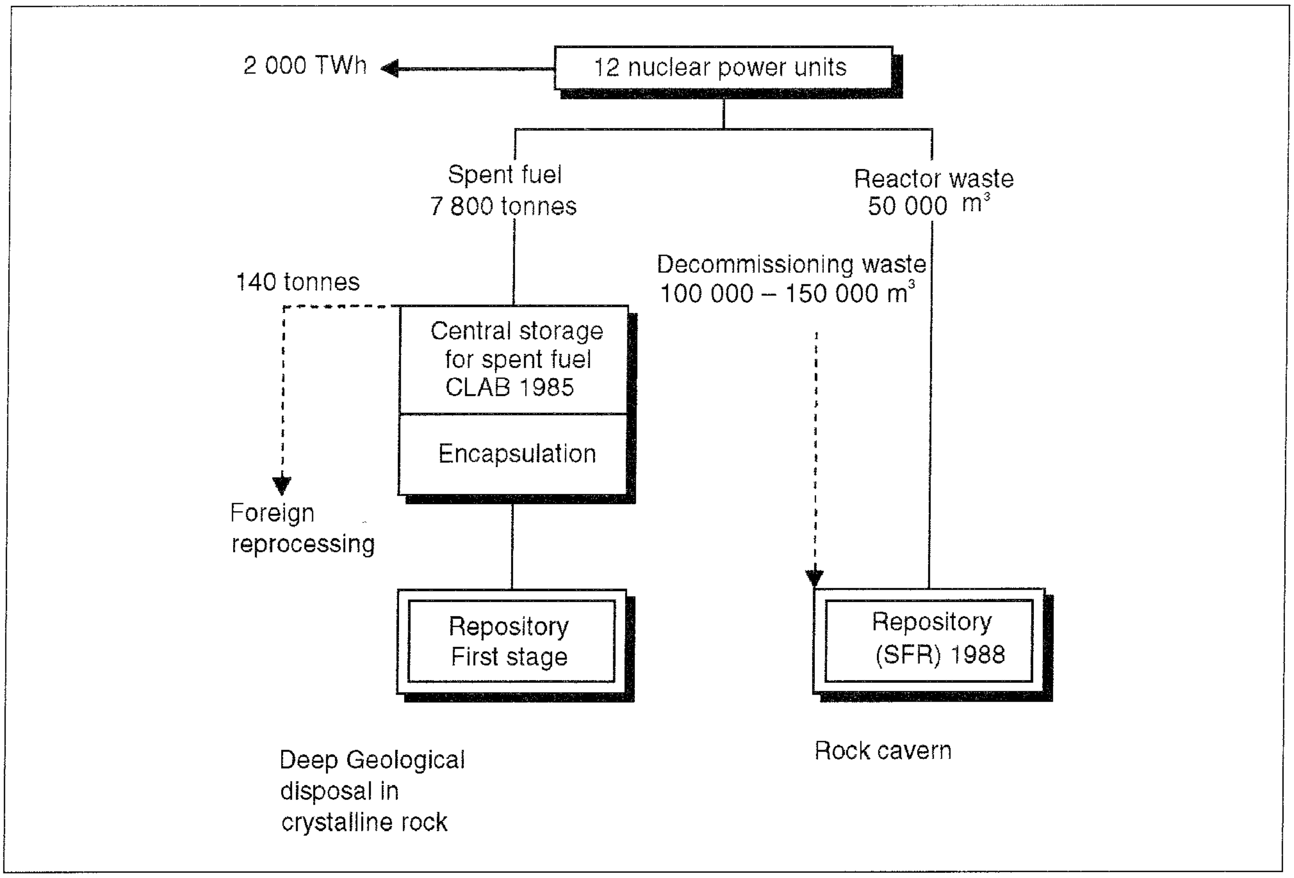

Figure 1-3. *Main system for management of radioactive waste in Sweden.*

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 1996 CLAB had been in operation for 11.5 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 2540 tU from the 12 Swedish reactors have been shipped to the facility and placed in storage.

In 1996, 79 casks containing spent fuel assemblies from the Swedish reactors were received together with 5 casks containing core components. The total fuel quantity shipped to CLAB during the year amounted to 242 tU. In parallel to the fuel receiving activities 69 fuel assemblies have been transferred from old canisters to new compact

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

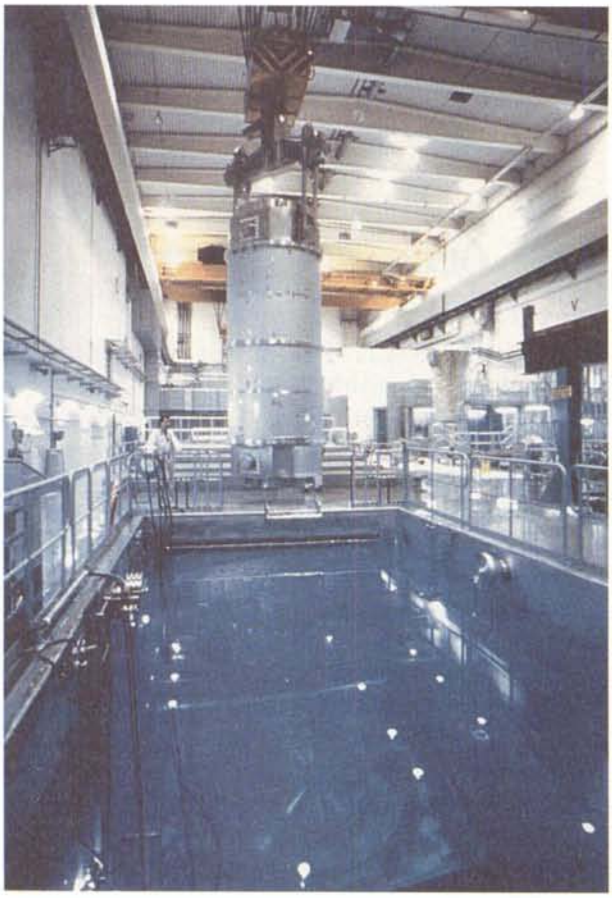

Figure 2-2. *Fuel cask with protective shirt being moved from the cooling cell to the unloading pool.*

storage canisters, see section 2.3. By the end of the year about 3/4 of the PWR fuel and 2/5 of the BWR fuel was stored in the new canisters.

In spite of the comparatively great amount of fuel received the total occupational dose in 1996 was 50 mmanSv, which is the lowest figure recorded since start of operation in 1985. The dose corresponds to less than 20% of the value calculated in the safety assessment made during the design phase.

The release of radioactivity to the environment during 1996 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 eleven 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 continuos plant modernisation programme.

The public interest in CLAB has increased and in 1996 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 utilisation 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. The design work on this expansion of CLAB continued in 1996 and construction is expected to start in late 1998.

An Environmental Impact Assessment Forum (EIA-forum) has been established for the CLAB expansion and the planned Encapsulation Plant. A preliminary EIS report concerning the CLAB expansion has been prepared and has been sent to the participants of the EIA-forum.

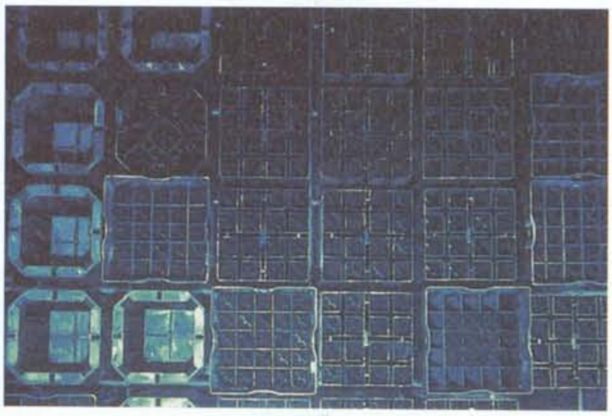

Figure 2-3. *BWR-fuel canister of the old and new types containing 16 respectively 25 fuel assemblies. The octagonal canisters are used for highly radioative reactor core components.*

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.

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

3.2 OPERATING EXPERIENCES

In 1996 the ship M/S Sigyn sailed around 32 000 n.m. 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 79 transport casks with spent fuel, 5 transport casks with core components, 68 IP-2 containers (ATB) with reactor waste and 70 ISO containers with low level waste have been transported with the transportation system during the year, see Figure 3-1. Like earlier years, no measurable dose rates have been registered to the ship's crew.

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 1996 eleven different transports with heavy equipment have been made 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 17 harbours, including the capital Stockholm.

Figure 3-1. Loading of ATB-container on board M/S Sigyn.

4 FINAL REPOSITORY FOR RADIOACTIVE WASTE, SFR

4.1 GENERAL

The Swedish Final repository for Radioactive 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. 50 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 1996 a total of 21 200 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.

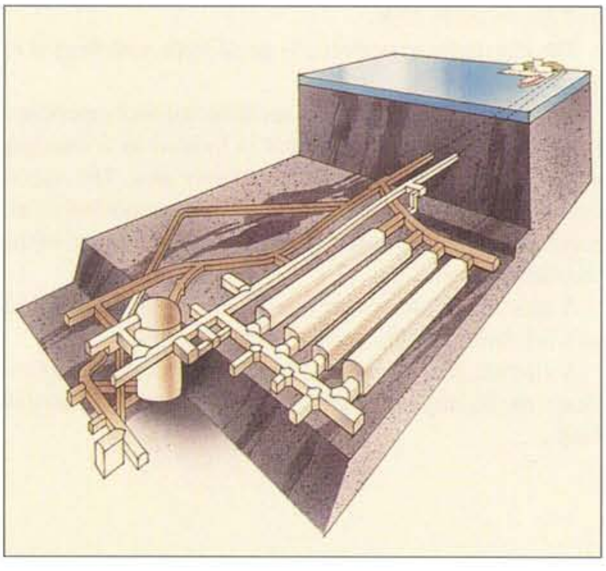

Figure 4-1. Overview of tunnels and storage chambers in the first construction stage of SFR.

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 is 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 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 1996, 39 waste types (of a total of about 50) were accepted for disposal. In 1996 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.

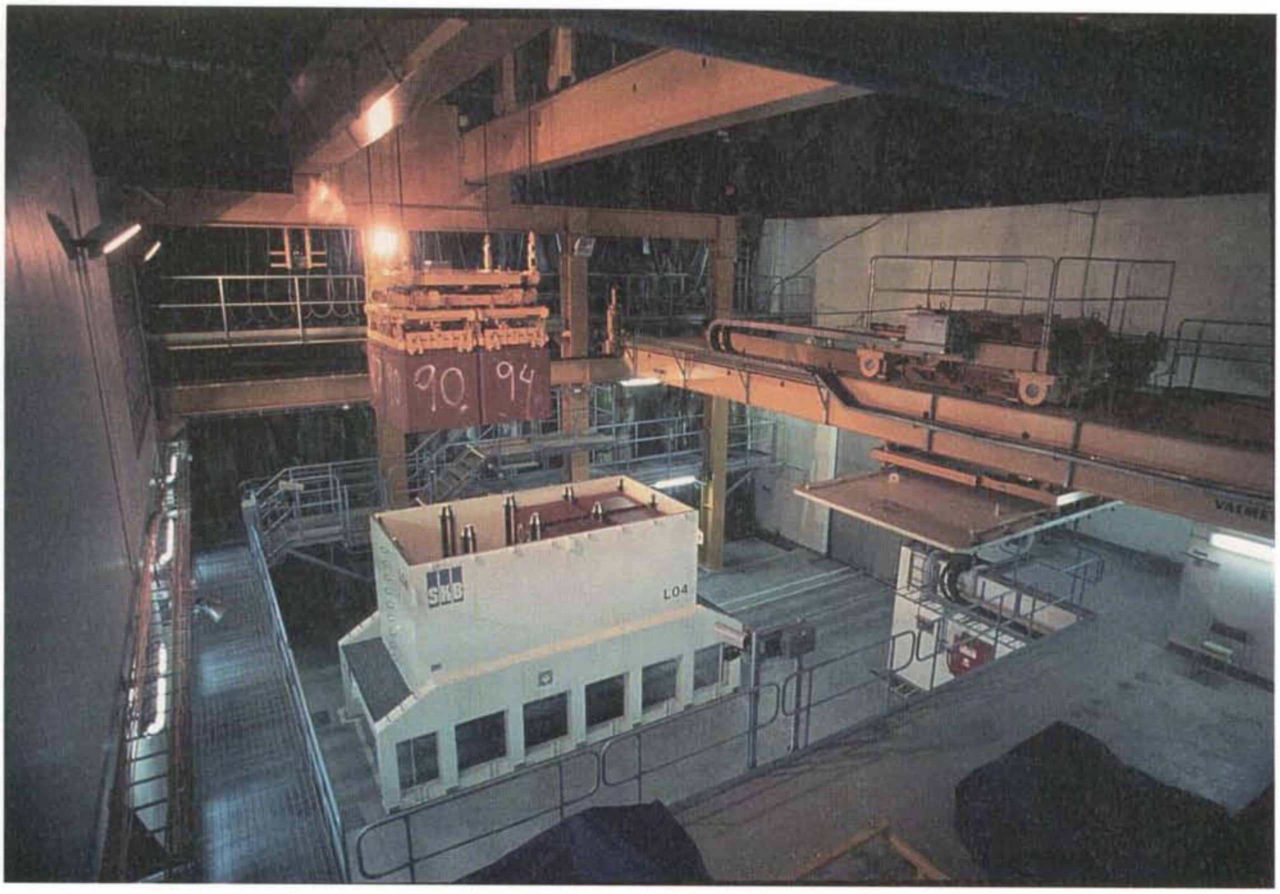

Figure 4-2. The operational waste is transported in special transport containers. In SFR the waste packages are unloaded with remote-controlled handling equipment.

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 organisation. The staff for operation and maintenance of SFR consists of about 15 people.

In full operation the facility has an annual disposal capacity of about 6 000 m^3. Up till the end of 1996 a total of 21 200 m^3 of waste has been deposited. During 1996, 2 600 m^3 was deposited in SFR compared to planned 2 500 m^3 according to a firm price agreement which have been established between FKA and SKB.

For 1997, 1 500 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 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 final operation in 1997.

A 10-year programme for corrosion preventing activities in the facility was started up in 1994 and continued in 1996.

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 95 /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 1996 the government appointed a National Coordinator for the siting process, see section 5.6. During 1996 the activities of SKB have been focused on:

- 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.
- Further development and description of siting criteria and the siting process.
- Background and national overview studies concerning different aspects of siting a deep repository.

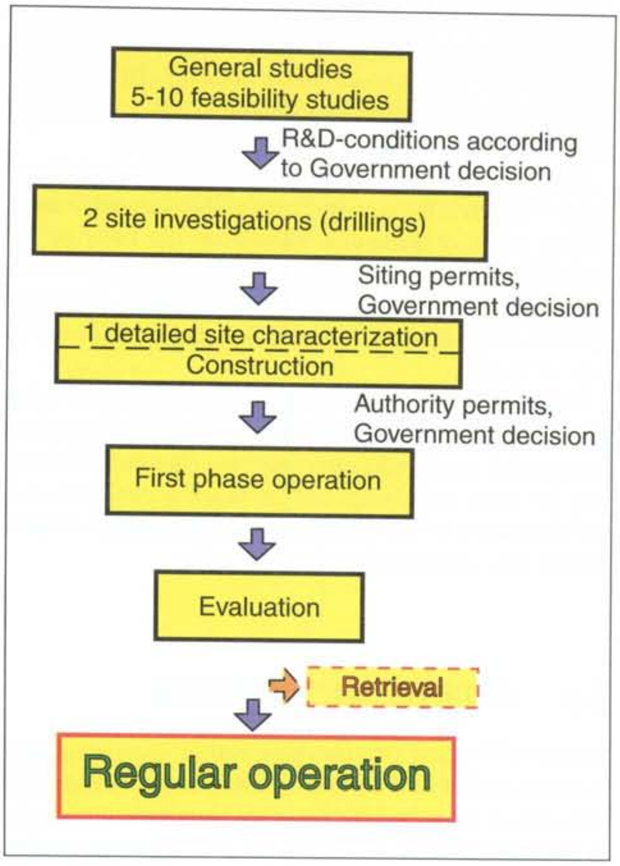

Figure 5-1. Main steps to implement a deep repository in Sweden.

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. The main steps to implement a deep repository can be seen in Figure 5-1.

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 performing a review of the results before having a referendum in September 1997. Feasibility studies are now also underway for two municipalities (Nyköping, Östhammar) in the southern part of Sweden. In the autumn of 1996 the municipality of Oskarshamn decided to participate in a feasibility study. The study will start during 1997. 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 construction of a repository (shafts-tunnel) by a detailed site-characterisation.

5.3 SITING ACTIVITIES

During 1996 siting activities have mainly involved general siting studies of the whole of Sweden, the information activities after the completion of the Malå feasibility study in northern Sweden and the start of two feasibility 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. 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 feasibility 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 feasibility studies can receive 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 feasibility studies and their neighbour municipalities as well as safety authorities and other concerned parties.

In December 1996 the government stated that before SKB enters into the site investigation phase the following material should be reported:

- A system analysis discussing the complete system (transports, handling, encapsulation etc) where alternatives to the KBS-3 system shall be assessed as well as a "zero-alternative" which means that no repository is constructed and that all waste will be kept in the CLAB facility.
- A long-term safety assessment of the deep repository for long lived wastes.
- A comprehensive summary of all to date performed geological studies. This will include all study site investigations made all over Sweden, feasibility studies, comparison with studies abroad (e.g. in Finland) etc.

The government statement in December 1996 will imply some delays in the siting programme compared to what was intended when the RD&D-Programme 95 was distributed in September 1995.

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.

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

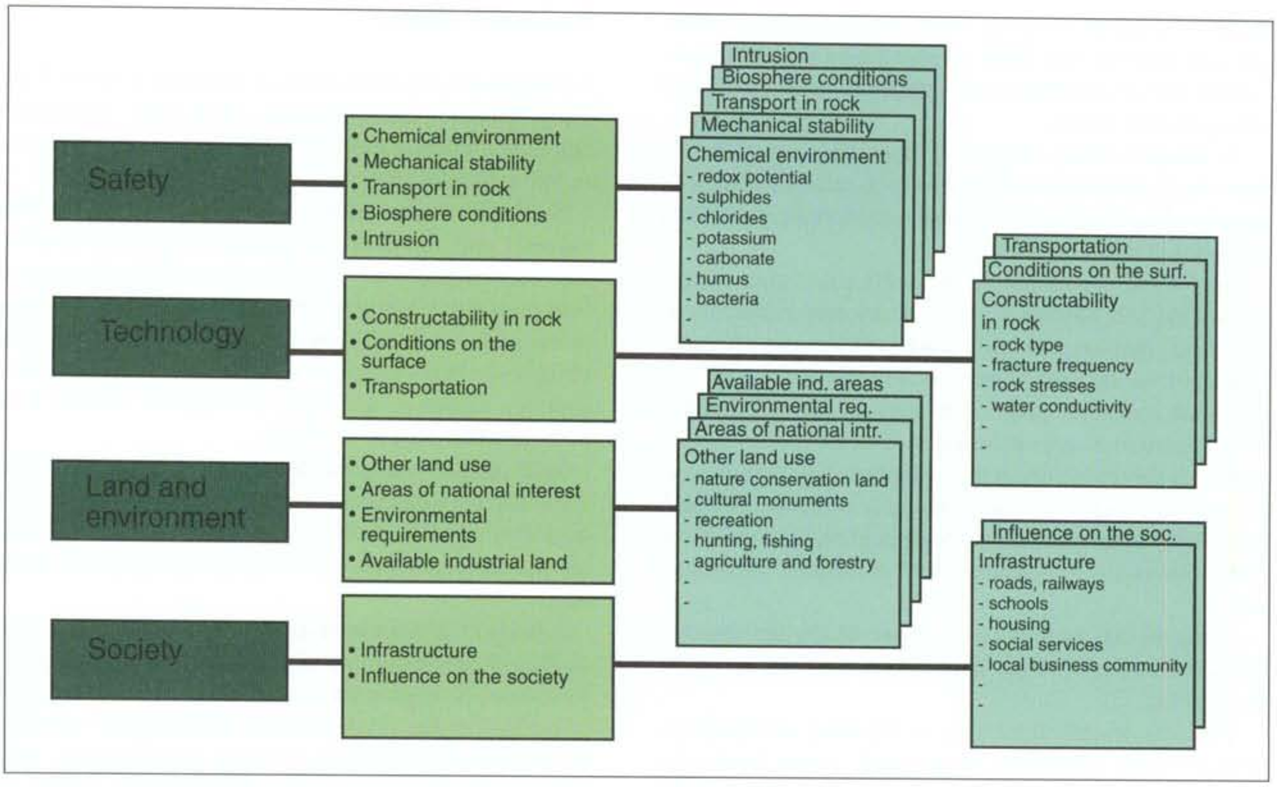

Figure 5-2. *Factors that determine the suitability of a site for a deep repository.*

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 geographical information system, GIS, 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 various 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 transhipment 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.

General Siting Study 95 /5-4/, contains SKB's overall report on general siting studies carried out on a national scale. 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.

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 utilization, 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. The future siting should 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.

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 1996 SKB has worked with feasibility studies at three municipalities, Malå, Nyköping and Östhammar.

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.

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.

Feasibility studies describe, in as much detail as possible, the prospects for siting a deep repository in the studied municipality and to shed light on the possible positive and negative consequences of such a siting.

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 feasibility studies have reflected this standpoint, and allowed insight by the municipalities into planning and performance of the work. Local information activities have formed an important part of the studies. Offices have been established when beginning the feasibility studies, and have been open to the public on a daily basis.

The feasibility studies have entailed broad investigation programmes, 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 have been done in the form of studies, based on existing information.

Compilation and analysis of geoscientific data have 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 results produced. The data available

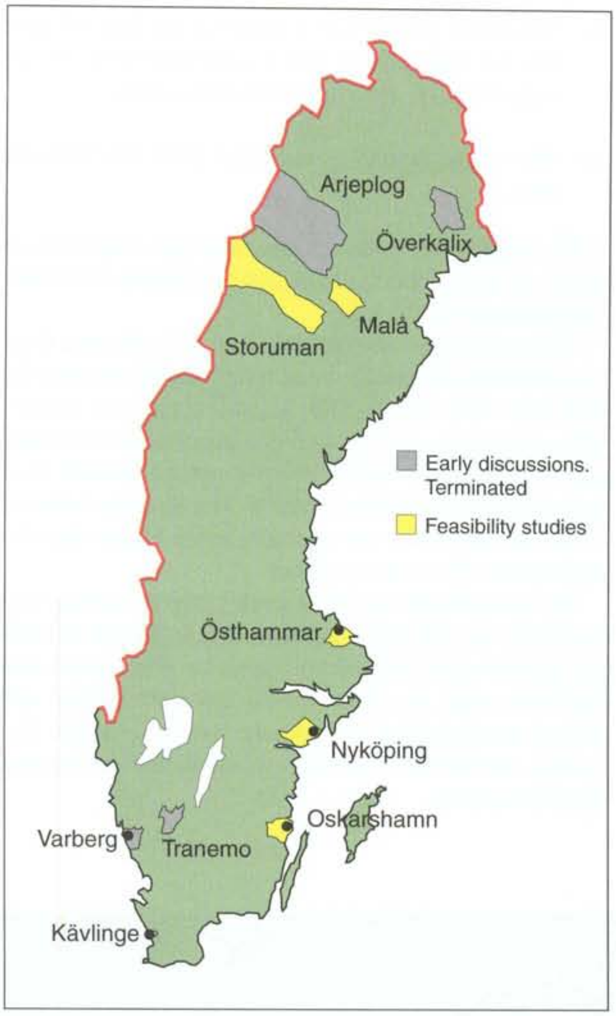

Figure 5-3. *Municipalities where feasibility studies are concluded, in progress or have been discontinued.*

are superficial and largely refer to conditions observable at surface.

Operating a deep repository 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 any municipality.

To be an interesting candidate a municipality should also offer good siting premises with respect to current and planned land use, as well as a variety of land preservation interests.

Besides evaluating technical and environmental siting prospects, the feasibility studies have 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 studies have aimed at providing background information rather that attempting any kind of ready-made answers. The current situations in terms of population, business and industry, labor market, services and municipal finances have been compiled. Possible future developments have been outlined and the societal impacts of implementing a deep repository project have been estimated. Generally out of the total, approximately SEK 15 billion investment related to the deep repository, it is found that some 30%, or SEK 4 billion, can be absorbed locally. It is anticipated that a deep repository, when fully operational, will provide approximately 200 direct jobs and at least 100 indirect jobs within a municipality.

Feasibility study of the Malå municipality

The feasibility study in Malå commenced in early 1994 and was concluded in March 1996, when a final report summarising investigations and results was published. An important conclusion was that there are large areas within the municipality that – as far as can be judged from existing information – would offer good siting premises with respect to bedrock conditions. The areas of interest are located within vast regions covered with predominantly granitic rocks.

Malå municipality is located in the inland region of northern Sweden, and transportation of spent fuel and backfill material to a repository within the municipality would therefore involve both sea- and overland components. The feasibility study concluded that both railway and road are possible alternatives for the land transport part. Improvements of present infrastructure would, however, be required for both alternatives. The municipality was furthermore found to offer good siting premises with respect to current and planned land use, as well as a number of land preservation interests.

On the basis of these and other results, SKB concluded that site investigations in Malå municipality may become of interest. Decisions in that respect can, however, not be made at this stage of the siting process.

Malå municipality has decided to conduct a local referendum in order to decide upon their further participation in the siting process. The date for the referendum has been set to September 21, 1997 and the question posed will be "shall SKB be allowed to pursue further siting studies within Malå municipality".

As a preparation for the referendum, the municipality has also organised 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. The work is lead by a local committee and independent expertise within fields of concern is being consulted for review contributions. The review process is currently underway and results are expected to become public in April 1997.

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/.

Although there are no nuclear power plants in Nyköping, the municipality is the site of the former nuclear research establishment, Studsvik – now privatized.

An SKB information office was inaugurated in Nyköping in the end of October 1995.

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 etc). During fall 1996 some of the opposition groups left this third reviewing group due to disagreement with how available funds were allocated, working procedures etc.

The scientific work was completed and reported during 1996. Three different siting cases have been studied:

a) The whole facility, both the above ground part and the underground part are placed in the Studsvik area

b) The above ground part is placed in the Studsvik area but the underground part is separated up to 10 km from Studsvik and connected with a ramp.

c) The whole facility is separated from the Studsvik area.

The geological studies shows that the conditions at Studsvik are complicated but that favourable conditions might exist in the vicinity.

From a general geological standpoint it is the area from the coast between Studsvik and Nyköping and towards the NW part of the municipality that offers the most favourable conditions, see Figure 5-4. This area has no larger deformation zones and fracture zones are present in a repetitive and predictable pattern. The distance between larger fracture zones are normally much longer than the dimensions of a deep repository.

The permeability of the bedrock in the Nyköping municipality as well as the geochemistry is judged to fulfil the general needs for a deep repository. This means that sufficient areas are present with low permeability and normal geochemistry without any dissolved oxygen (reducing conditions) which is prerequisite for the planned disposal system.

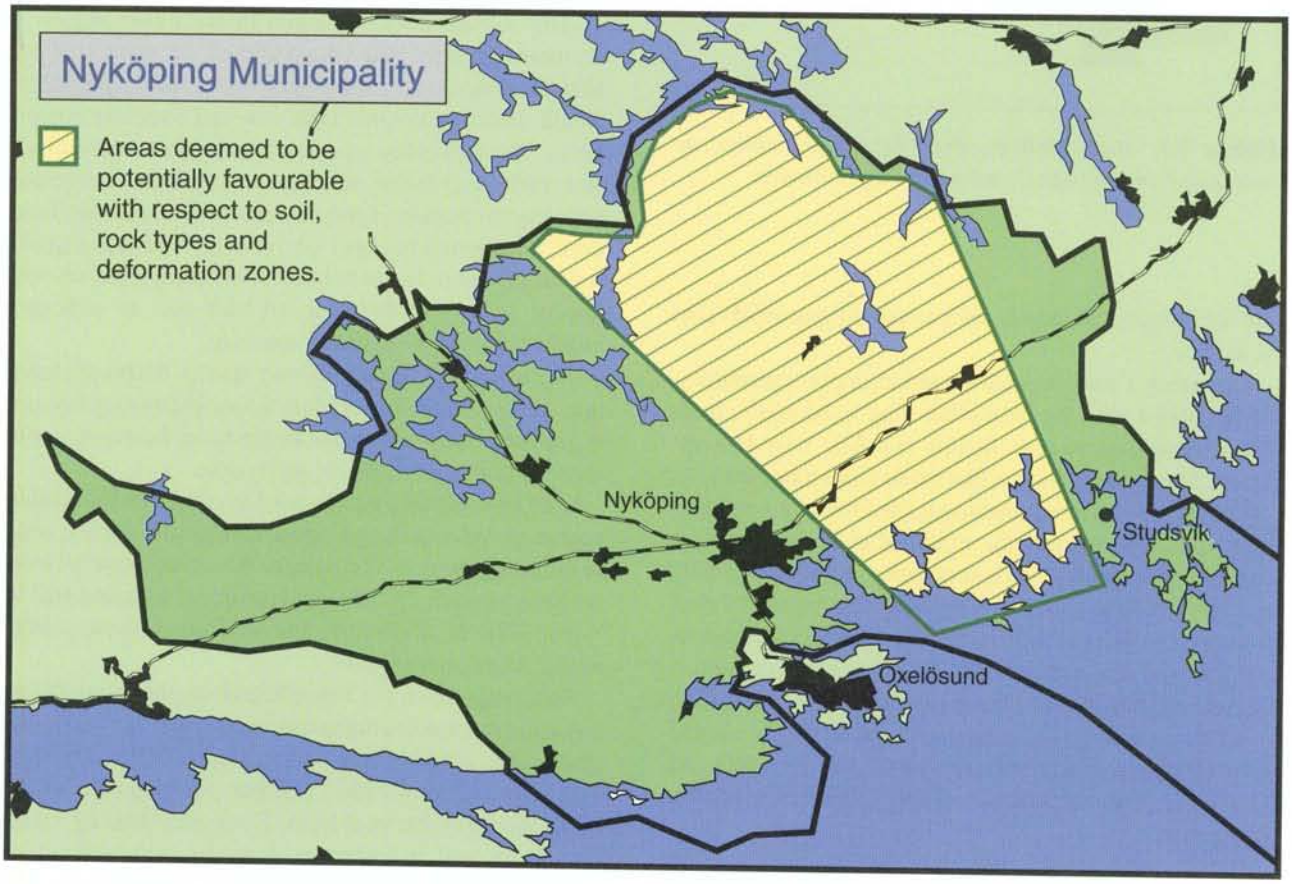

***Figure 5-4.** From a geological standpoint most favourable part of the Nyköping municipality.*

The technical conditions to site the above ground part of the repository in the present industrial area of Studsvik seems favourable. Such a solution would involve the use of present infrastructure in form of security functions, heating, water supply etc. Other advantages would be that new roads or railways for heavy transports in this case do not have to be constructed since the cargo can be received at the Studsvik harbour.

Canisters with spent fuel will probably be transported on ships similar to the M/S Sigyn and other types of materials needed in the repository with conventional ships. If Studsvik is chosen as the site for the surface facilities of the deep repository, materials as sand, bentonite etc must be taken in to the Studsvik harbour with relatively small ships. An alternative for these transports is the harbour in Oxelösund.

The technical conditions for siting the above ground part of the repository in other locations than Studsvik seems generally positive. The roads in the municipalities have a good traffic capacity. The capacity to deal with loads up to 100 tonnes remains to be investigated. The railroad net is not very dense but has a high technical standard.

The region "East Södermanland", where Nyköping is a part, will have a estimated number of work opportunities at about 31,000 to 35,000 for a long period to come. The construction of a deep repository in the region would not have any drastic effects on the employment rate but give a substantial economic stimulation effect. The overall development and growth of the region are not judged to be substantially changed due to a siting of a deep repository.

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 preceded 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 started. An information office, similar to the one in Nyköping, was established.

The results of the geological studies indicated nine areas of interest within the municipality. Several of these are, however, located in nature-protected areas, which limits the possibilities for siting.

Two harbours are judged feasible for transportation of the spent fuel and the bentonite clay. One of them is the Forsmark harbour, where the SFR facility is located. If the bedrock conditions are suitable in this area, the surface facilities of the deep repository might be located close to the SFR facility. An artist view of such a siting is presented in Figure 5-5. Other possible locations of the surface facilities within the municipality have been presented.

5.4 TECHNICAL STUDIES CONCERNING THE CONSTRUCTION OF A DEEP REPOSITORY SYSTEM

In parallel with siting activities SKB is conducting a programme for technical studies concerning the design and construction of a deep repository. These studies provide input to safety assessment work, feasibility studies and cost calculations.

The design of the repository for other long-lived waste than spent fuel has been reviewed as part of the preparation work for the Safety Report 97. The improvements consist of a standardised cavern sizes as well as adjusted utilization of the caverns leaving a minimum of space between the cavern wall and roof and the waste packages.

The analysis of the stresses induced by the thermal load of the repository in the rock mass at ground surface has continued with sensitivity analysis of the impact of rock material properties. The results do not contradict the conclusions from earlier studies that the design-determining factor is the temperature allowed in the bentonite and not the thermally induced stresses.

Thermo-mechanical analysis of the stability of the rock around deposition hole and deposition tunnel, that take into consideration the effect of the fracture system in the rock, has started. The target is to eventually use discontinuum models based on discrete fracture network models with data from Äspö HRL. So far the study has focused on continuum models and analytical solutions for establishing references for subsequent study phases.

The work on grouting technology during 1996 has been literature studies on characterisation of the rock from grouting point of view, extension mechanism for grouts in rock and suitable cement-based grouting materials.

Investigations of the temporary plugging of backfilled deposition tunnels have resulted in a plug design that features an extension out into a slot in the surrounding rock in order to allow the groundwater head to recover around all deposition holes soon after deposition, which is estimated to facilitate fast saturation of the bentonite around the canisters.

Studies of retrieval methods after the bentonite buffer has saturated and swollen have continued with bench and pilot scale tests of disintegration of the bentonite with a brine for removal of the bentonite slurry by pumping. The tests suggest that a sodium chloride content of between 4 and 6% by weight is most effective.

A more detailed presentation of the studies mentioned above are given in Chapter 13.

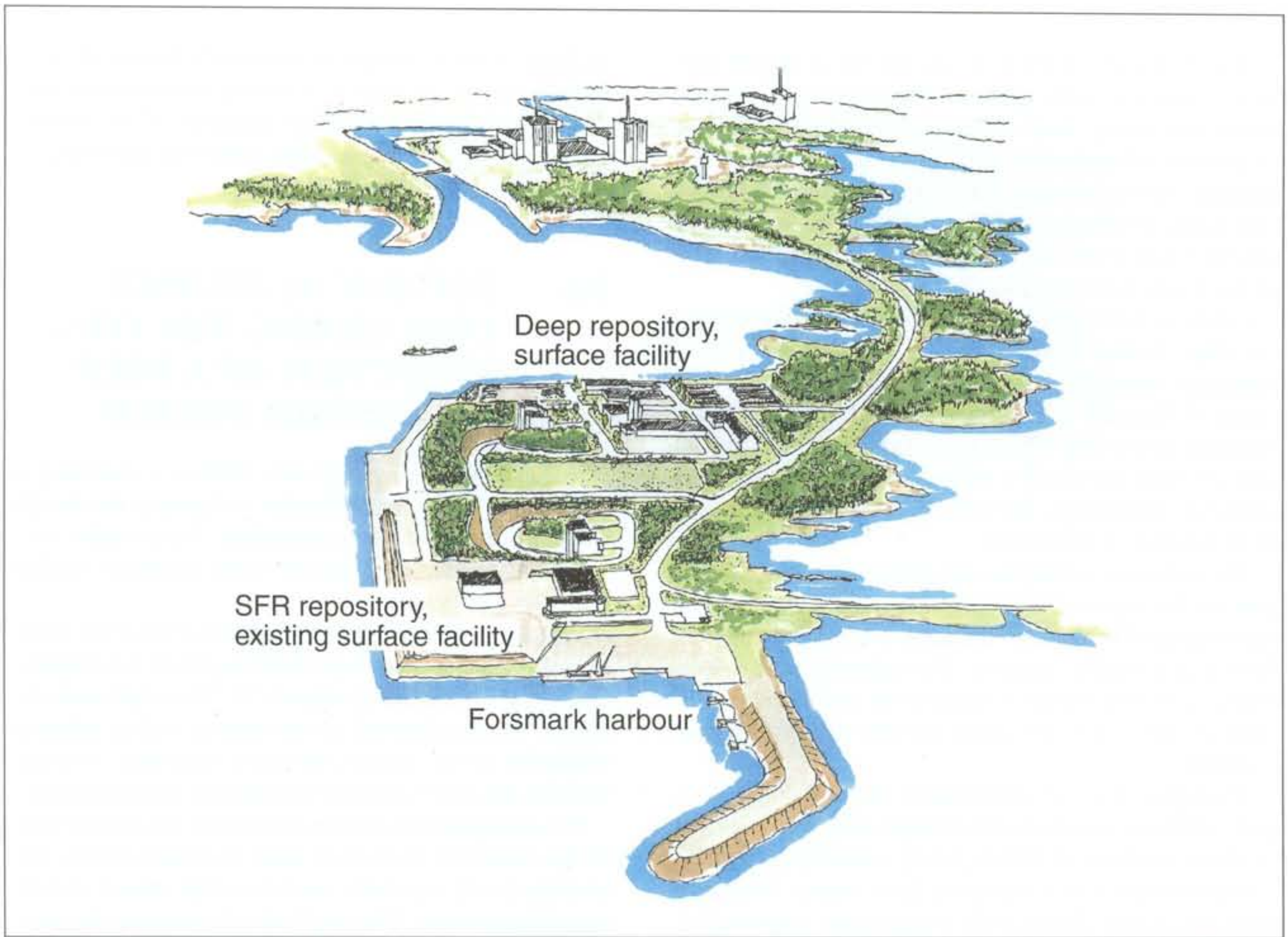

Figure 5-5. *An artist view of how the surface facilities of the deep repository siting might be located besides the existing surface facility of SFR facility (final repository for radioactive operational waste). This option is feasible if the underground part of deep repository is located within 10 km from the Forsmark nuclear power plant.*

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 instruments and methods, including development, refinement and investment, etc
- preparation of techniques and routines for data management,
- development of a site evaluation programme.

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

The geoscientific investigation programme for the deep repository has the following main goals:

- The investigations should for every investigated site provide a geoscientific understanding of the site and its regional environs with respect to present-day situation and natural ongoing processes.
- The investigations should for every investigated site 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 site evaluation programme has the following main goals:

- The site evaluation should for every investigated site evaluate whether the fundamental safety demands and other siting criteria are fulfilled, and see to it that the repository in the best way will be adopted to the local environmental and rock conditions.
- The site evaluation should compare qualified sites, first of all with regard to long-term performance and radiological safety but also with regard to the other siting criteria.

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 steps, 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.

One important step in the development of a geoscientific investigation programme has been the compiling of a report which systematically presents all geoscientific parameters which, in one way or another, will be treated in the process of site evaluation in general, and in the performance assessment in special.

Work in the area of investigation methods and instruments have involved method tests of surface geophysical methods for potential use in the initial site investigations, aiming at surveying relatively large areas with regard to relevant site parameters in order to be able to concentrating the deep investigations on a smaller site. The development of a system for Geological Borehole Documentation (GBD) in progress. The system is based on the borehole TV system BIPS and core logging aiming at integrating those methods to an efficient characterisation tool.

Clear and efficient routines for data management will be an important tool for fulfilling QA requirements of the site investigations. Work on technical documentation, manuals and QA-routines for all methods and equipment which will be used in the site investigations is going on. During method tests performed quality plans are used for testing also the feasibility of these QA routines.

Further information on the work conducted in the area of site investigation programme are presented in Chapter 13.

5.6 ENVIRONMENTAL IMPACT ASSESSMENT

SKB plans for the deep repository include the preparation of an Environmental Impact Statement (EIS), see Figure 5-6. 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 EIS for the deep repository has to be presented when SKB applies for a license to start detailed site characterisation. The content of an EIS 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 EIS. As mentioned in section 5.3 the County Administration has a responsibility to facilitate that necessary contacts are taken to ensure that all parties have been involved in the Environmental Impact Assessment (EIA) process on the regional level.

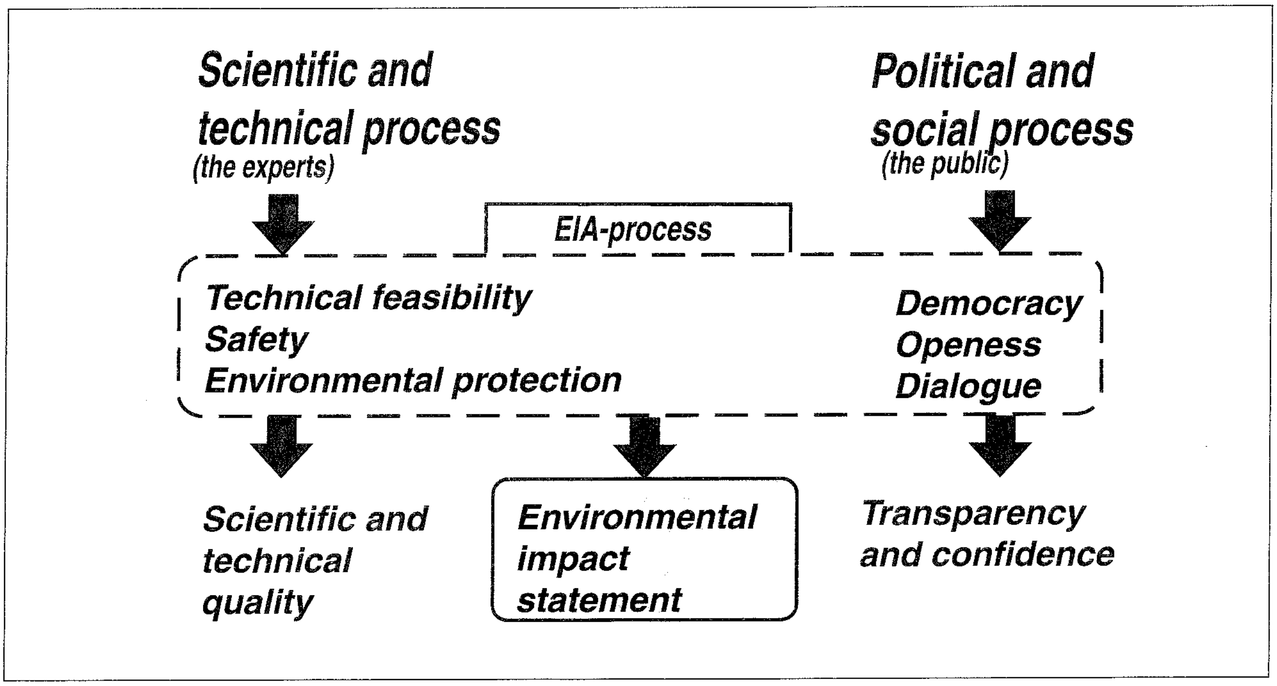

Figure 5-6. Siting and implementation of a deep repository system is a scientific and technical process as well as a political and social process. All parties concerned should be involved in an open dialogue to develop a transparent EIS of high quality.

During 1996 there were discussions on having an EIS 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 coordinator was 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. In May 1996 a National Coordinator for Nuclear Waste Disposal was appointed by the government. His main task will be to promote co-ordination of information and investigation inputs found necessary by municipalities affected by SKB studies concerning siting of facilitities for spent nuclear fuel and nuclear waste. The national co-ordinator is to propose forms for the intercharge and also to be prepared to co-ordinate liaison between the municipalities and county administrative boards affected by the studies.

The county boards in the counties affected by feasibility studies have and formed discussions groups (EIA-groups) during 1996. The participants in these groups are the municipality in question, some of the authorities and SKB. The work has so far been concentrated on information on present work in the feasibility studies.

6 ENCAPSULATION PLANT PROJECT

6.1 GENERAL

SKB's development and testing of methods for the encapsulation of spent nuclear fuel has continued as presented in the RD&D-Programme 95. A Canister Laboratory is now being built for the further development and testing of the primary parts of the encapsulation process. The Swedish concept is to encapsulate the fuel in durable copper canisters to be emplaced in a deep repository. 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. The reference copper canister has a cast insert of cast iron with channels for the fuel.

Other designs of canisters have earlier been studied, such as a homogeneous copper canister, made by hot isostatic pressing 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 considered for the encapsulation plant. Advantages are e.g. logistics for the handling, use of existing resources and minimal impact on the environment. The canisters will be manufactured at workshops on other sites and delivered to the encapsulation plant after control.

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 probably remain intact for millions of years, which provides a considerable margin of safety.

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 made by both normal and abnormal operating cases in the plant. It must also be able to withstand handling accidents that could give the personnel and the plant to unacceptable exposures 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 canister 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. 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 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 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 also 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 surrounding bentonite 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.

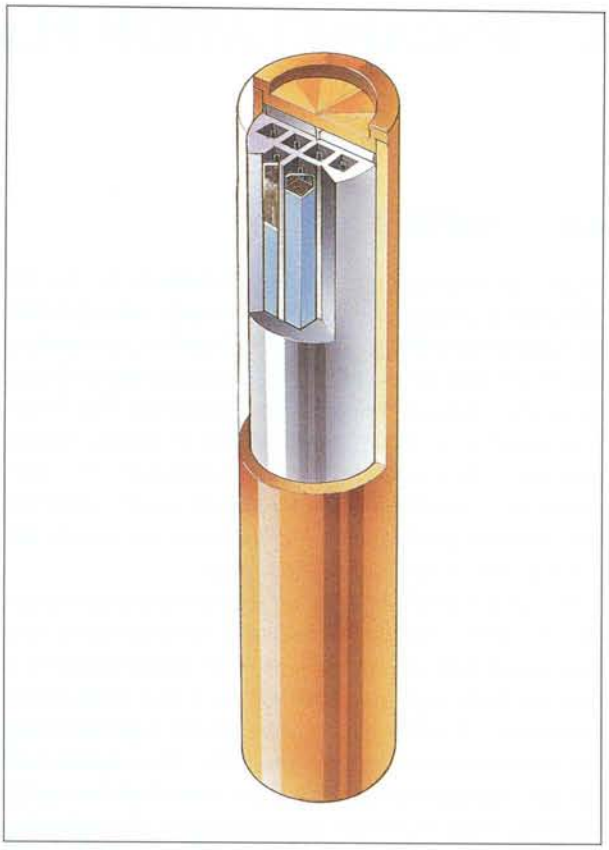

Figure 6-1. *Copper canister with cast insert.*

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 risk 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. The reference canister is shown in Figure 6-1.

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 choice of material grades will be studied in the continued work and be chosen with a view towards the criteria described above.

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.

As an alternative for the insert, a cast inner component of spheroidal graphite iron is considered. 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, lenght 4850 mm and weight 25000 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.

Figure 6-2. *An fullsize insert from cast iron.*

6.3 CANISTER FABRICATION AND SEALING METHODS

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, 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.

For the canister insert trial castings in steel and nodular iron have been made with promising results. Nodular iron has a very good castability. Bronze is also beeing studied as an alternative material but no test castings have so far been made. If steel or bronze is chosen the price will be considerably higher due to poorer material yield and higher material prices. In parallel with the test production during 1996 a Quality Manual according to the demands in ISO 9001 and IAEA 50-C-QA has been developed. The target is to have a quality assurance of the complete canister manufacturing process. Casting of an iron insert is shown in Figure 6-2.

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.

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 was completed. This method is feasible but requires a 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 inclined weld should be tested.

These tests started at TWI in 1996 and will continue in the canister laboratory.

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. Pieces from lids were also inspected by digital radiography, as a reference for ultrasonic inspection. The number and distribution of detected defects varied considerably for the various welds. This proved to be useful for testing the sensitivity of the different methods used.

Canister Laboratory

A laboratory for encapsulation technique, Canister Laboratory, is now being built to be in operation 1998. In this laboratory the primary parts of the encapsulation process will be tested. The Canister Laboratory is situated in the town of Oskarshamn, where part of an old shipyard has been purchased for that purpose, see Figure 6-3. The main parts of the laboratory are the electron beam welding and non-destructive testing operations. There are, however, plans to also test and demonstrate other parts of the encapsulation process at a later stage, e.g. transfer of fuel to the disposal canister and sealing of the cast insert.

The electron beam welding (EBW) equipment has been purchased from TWI and shall be delivered in the beginning of 1998. The EBW-equipment will be able to perform welds both horizontally and at a variable angle. X-ray and ultrasonic equipments are planned to be used for the inspection of welds. The first trial series are planned to be finished and results reported in beginning of 1999. The results from the first trials will be submitted to support the design work for the Encapsulation Plant as well as the application for construction of the facility.

***Figure 6.3.** Canister Laboratory at the shipyard in the city of Oskarshamn.*

6.4 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. This location provides possibilities to extend several existing functions into the Encapsulation Plant. These functions include the fuel elevator, cooling systems, water purification systems and electrical power supply. At a first stage only spent fuel will be encapsulated but preparations are made for the later addition of equipment for treating core components.

The plant shall be designed and constructed mainly for the encapsulation process. Further shall it be possible to

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

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.

In 1996 the design work resulted in a Basic Design, which will form the basis for the application for construction of the plant. BNFL Engineering Ltd. did all work involving the encapsulation process and ABB Atom AB designed the auxiliary systems. The layout of the plant and co-ordination of the design work was performed by SKB.

The plant is designed for an annual output of approximately 210 disposal canisters per year, i.e. on the average one canister per workday. The operating staff shall be able to work both in the Encapsulation Plant and in CLAB. During normal operation, approximately 30 people will be working daytime with encapsulation and maintenance.

The Encapsulation Plant will be approximately 65 x 80 meters in size and about 25 meters high, which is equivalent to the height of the existing receiving building in CLAB.

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 is 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 assemblies which are stored in CLAB have very different values of burn-up and residual power. These properties give the heat output of the canisters, which is a restricting factor in the Deep Repository. To minimize the total number of disposal canisters, the combination of fuel assemblies in a canister is optimized. Based on the fuel data in CLAB, storage canisters with suitable fuel for encapsulation are chosen. For the transfer of fuel from the storage pools underground in CLAB to the Encapsulation Plant, an existing fuel elevator is used.

In a handling pool in the encapsulation plant the storage canisters are placed in a rack. In order to simplify visual inspection of operations, the handling machine is operated from a platform which is situated directly above the pool. After identification, one fuel assembly at a time is lifted out and transferred to a fuel monitoring station where gamma measurements are made in order to calculate, for example, burn-up and residual power. The results of these measurements are compared with the historical fuel data. The assembly is then placed in a transfer canister which is similar to a storage canister but can only hold 12 BWR or 4 PWR fuel assemblies, which is the same as the number of fuel channels in a disposal canister.

When a transfer canister is filled, it is brought to a transfer bogie and lifted up out of the water by an inclined elevator. It is then lifted, by a handling cell crane, to the handling cell where it is placed in one of two drying stations. There, the fuel is dried with recirculating air with a temperature of about 120°C. A shielded frame with an empty disposal canister is docked from below to another part of the cell, see Figure 6-5. The connection between the disposal canister and the handling cell is tight so that the outside of the canister is not contaminated and the air in the cell does not escape and cause airborne activity in other parts of the plant.

When the fuel assemblies are dry they are transferred, one by one, to the insert of the disposal canister. The top of the disposal canister is protected so that the surface will not be damaged during handling. When the canister is filled, the disposal canister is moved to the next station.

In an inerting and lidding station, the air in the cast insert is removed and replaced with argon. The connection between the canister and the inerting and lidding station

***Figure 6-5.** Shielded Frame for transport and docking of work stations.*

is similar to the connection at the handling cell. When the atmosphere in the insert is of the required quality, a steel lid is bolted to the cast insert. The lid is tested for tightness before the canister is transferred to a welding station.

At the welding station the disposal canister is docked to a welding chamber within the station. When the canister is connected, the chamber is vacuumed down and a copper lid is placed on the copper canister. The lid is then sealed to the canister using electron beam welding. During welding the canister is rotated and the welding equipment is fixed. When the weld is completed, the disposal canister is transferred to the next station for testing.

When the canister is docked to the NDT and machining station, a visual control of the weld is made before the weld area is machined. Non-destructive testing is then performed using both ultra sonic and X-ray techniques. If the weld contains defects which are repairable, the canister is brought back to the welding station for re-welding. In case a weld has failed in a way that it can not be repaired, the copper lid is removed, the steel lid unbolted in the inerting and lidding station and the fuel, finally, unloaded in the handling cell.

When a canister has passed the non-destructive testing it is lifted out of the shielded frame, using a remote controlled shielded handling machine, and is transferred to a monitoring and decontamination station. Smear tests are taken on the entire outer surface to monitor that it is clean. The station is equipped with high pressure water which can be used if there is need for decontamination, after which new smear tests are taken. The surface dose rate is also measured before the canister is transferred to the buffer store.

The buffer store for filled canisters is situated under a radiation shielded floor with openings over the storage positions. The buffer store is cooled with air and has approximately 50 storage positions.

When a canister is to be delivered to the Deep Repository it is transferred, from the buffer store to a loading position. Underneath this position a transport cask for disposal canisters is docked. The cask is then lifted, with an overhead crane, to a transport frame which the cask is lowered onto. The same type of handling is already used, routinely, in CLAB. With a transport vehicle, the cask with the disposal canister is moved out of the Encapsulation Plant for further transportation to the Deep Repository.

6.5 ENVIRONMENTAL IMPACT ASSESSMENT

In 1994 a forum for Environmental Impact Assessments forum (EIA-forum) was established. It is chaired by the county authorities and has representatives from the Oskarshamn municipal council, the nuclear power inspectorate (SKI), the radiation protection institute (SSI) and SKB. The various aspects of construction and operation of an encapsulation plant at CLAB have been discussed in the forum.

In the beginning of 1996 the results were documented in a planning report which was sent out for comments by concerned local parties. An exhibition in the community hall was also arranged to display the plans for the encapsulation plant and the result of the EIA work. The comments which were received were discussed in the EIA-forum and a addendum to the planning report was prepared. In January 1997 the planning report with the addendum was approved by the municipal council as a base for an EIS report.

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. In late September 1995 SKB submitted its fourth RD&D-programme to the Nuclear Power Inspectorate – SKI. The programme covers the period 1996-2001 in some detail.

The programme was sent by SKI for review to about 60 organizations and SKI received comments from about 35 of these organizations. Based on these comments and on their internal evaluation SKI issued its evaluation report to the government in May 1996. The programme was also reviewed by KASAM, the governments scientific advisory committee on nuclear waste, which sent its report to the government by the end of June 1996. Section 12.2 gives the summary of SKI's evaluations and proposals, section 12.3 the summary from KASAM's report. The government decision on the programme was taken on December 19, 1996. It is given in unofficial translation in section 12.4.

The decision can be summarized as follows:

- The programme fulfils the stipulations of the law.
- SKB shall carry out a system analysis of the entire final disposal system (encapsulation plant, transportation system and repository). This analysis shall allow for an overall integrated safety assessment of the entire disposal system. Alternatives to the KBS-3 method should be described as well as variations of the method. The zero alternative (continued storage in CLAB) should also be analysed and ongoing work on transmutation should be presented.
- SKB shall carry out a safety assessment of the long-term safety of the repository and describe needs for supporting research, how research results should be transferred and how uncertainties should be treated.
- SKB shall supplement the "General Siting Study 95" by specifying the factors determining the selection of suitable site.

The importance of a well-defined and transparent site selection process is emphasized. SKB should be able to specify criteria for evaluation of candidate sites and specify factors disqualifying a site from further investigations

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

The RD&D-work during 1996 has followed the plan set forth in RD&D-Programme 95. The expenditures were 124 MSEK in total of which 51 MSEK were for the Äspö HRL, 56 MSEK for other supporting R&D and 17 MSEK for international cooperation.

7.2 SAFETY ANALYSIS

During 1996 the SKB RD&D-Programme 1995 was reviewed by the regulatory authorities and the Government. With regard to safety issues the governments pointed out in its decision /7-1/ that there is a need for a comprehensive assessment of the long term safety of a deep repository to be presented before two sites are selected for site investigations. It was furthermore required that the relationship between priorities in the R&D activities and the safety analysis should be clarified, how results from the R&D activities are utilised, and how the basic uncertainties are to be handled in the coming work.

SR 97

The present planning of the work on safety assessments and the methods and tools used therein has thus been adjusted. The goal is to make during 1997 a full set of evaluations for a generic safety report (SR 97) for a repository based on the present prioritised design (largely a KBS-3 like design with a copper steel canister) utilising site data from the Äspö area and two earlier investigated study sites in Sweden. The report is planned to be available during 1998.

Safety principles

The safety principles underlying the repository design have earlier been discussed in the supplement to the RD&D programme 1992. I.e. a multi barrier concept to utilise three levels of safety functions:

- the isolating capacity of the canister,
- the retarding effect of slow dissolution and sorbtion in the engineered barriers and the geosphere, and

- the siting and layout of the repository with due regard to the importance of the first biosphere recipient of the deep ground waters.

The materials used in the repository are 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.

Technical barriers

New calculations of radionuclide inventory and decay heat for the typical fuel have been performed for PWR fuel with a burnup of 42 MWd/kgU and two high burnup fuels: a BWR fuel with a burnup of 55 MWd/kgU and a PWR fuel with a burnup of 60 MWd/kgU.

In case the canister integrity is lost the fuel dissolution has to be quantified. A new model for the matrix dissolution has been developed for SR 97. It is based on a set reactions and rate constants describing water radiolysis and rate constants for the reactions between the oxidants O_2, H_2O_2 and the UO_2 surface. The number of moles of consumed oxidants (O_2, H_2O_2) are equivalent to the number of moles oxidised uranium.

Owing to the very low water flux inside a canister, many of the radioelements will precipitate as secondary minerals if they are released from the fuel matrix. A new study of solubility's, for critical radionuclides as well as comparisons with trace element concentrations in natural systems, has been done.

Sorbtion and diffusion coefficients for radionuclides in compacted bentonite have been reviewed and the data set has been updated for the use in SR 97.

The geosphere

In SR 97, three hypothetical sites will be used, arbitrarily named Aberg, Beberg and Ceberg. The data for these sites is taken from previous investigations conducted by SKB at sites in Sweden. These are:

- Aberg, which is based on data from the Äspö Hard Rock Laboratory, in southern Sweden;
- Beberg, based on investigations at Finnsjön, in central Sweden; and
- Ceberg, based on investigations at Gideå, in northern Sweden.

The main objective of the analysis will not be to compare the sites. Instead focus in the far field will be on how different conceptual models for describing flow and transport in fractured rocks affects the performance of the geosphere barrier illustrated by three different geographical locations in Sweden. Furthermore, it must be transparent how the data of each site have been used in the assessment models utilised.

The initial activities within SR 97 have been devoted to compilation of data and geoscientific understanding of the sites. Even though no new data will be collected there are differences in methodology affecting the interpretation of the sites which need to be overcome before going into further analyses.

For the transport calculations a new compilation of radionuclide sorption coefficients for transport in fractured rocks is being performed. For every element, there is a recommendation of a realistic K_d value with an uncertainty limit. The selection is based on experimental investigations.

Biosphere

The radionuclides that may be released from a deep repository will be mainly transported by the groundwater and come into contact with the biosphere through running water, lakes, wells or the sea. The groundwater can also penetrate sediments, peat bogs or soils in the discharge areas. The various recipients are represented by modules of typical ecosystems where dose-factors for critical groups are calculated. The developed modules are bog, lake, running water and Baltic sea coast. Modules under preparation are cultivated and natural grasslands, while forest will be developed later.

7.3 SUPPORTING RESEARCH AND DEVELOPMENT

7.3.1 General

Chapter 15 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 12.

7.3.2 Spent nuclear fuel

The studies on the behaviour of spent nuclear fuel in repository conditions have given new results on fuel dissolution, on fuel corrosion modeling, on alfa radiolysis, and on natural analogues for the nuclear fuel. Studies have been initiated on spent fuel dissolution under more real anoxic or reducing conditions which will require more involved experimental setups. New separation and analysis methods have been developed to analyse 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 americium, curium, europium and uranium have been obtained.

Radiolysis experiments in carbonated solutions 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. The kinetic model for the dissolution of the spent fuel matrix has been under development and has been tested against spent fuel dissolution data.

From experimental and modelling studies on uranium minerals as natural analogues for the stability of the spent fuel matrix, the stability of the U(VI) silicate phases soddyite as an end product for the oxidative alteration of the spent fuel matrix has also been determined experimentally. Data on the kinetics of dissolution of this phase have been obtained.

7.3.3 Geoscience

The general supporting 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. Furthermore models for evaluating coupled thermo-hydro-mechanical processes are developed.

During 1996 the geoscience programme has included research aiming at increasing the understanding of the ongoing shore level displacement in Fennoscandia.

The shore level displacement is mainly due to two co-operative vertical movements, the glacio-isostatic uplift and the eustatic sea level rise. The shore level displacement is estimated by subtracting the eustatic from the isostatic component. The cause of the difficulty of modelling the shore level displacement has been the lack of empirical data of the glacio-isostatic uplift as well as reliable data of the eustatic rise. However, due to investigations of the so-called lake-tilting phenomenon the size of the glacio-isostatic uplift has been discernible. By this knowledge it has been possible to start an iteration process for detailed estimations of the glacio-isostatic uplift and the eustatic rise using empirical data of the shore level displacement. The displacement is empirically known from 63 shore level curves in the area affected by the Scandinavian ice during the Late Weichselian.

Another project has focused on describing different space geodetic measurements of relative point positioning over distances ranging from tens to thousands of kilometres. By means of these new techniques it is possible to trace plate tectonic motions and also detect strain patterns within continents. The SWEPOS system consists of permanently operating GPS (Global Positioning System) stations in Sweden. The system has been designed, devised and furnished as a joint effort between the National Land Survey of Sweden and the geoscience group at Onsala Space Observatory, Chalmers University of Technology. In the SKB report the operations within SWEPOS are described emphasizing the possibilities to detect crustal motions in Fennoscandia. A separate project named BIFROST has been created at Onsala Space Observatory. BIFROST stands for Baseline Inferences for Fennoscandian Rebound Observations, Sea-level and Tectonics. It combines the efforts of a number of investigators at different sites and contributes to a number of international research programs in geophysics and geodesy. The project group intends to run the BIFROST project of at least ten years if deformation rates of 0.1 mm/yr are to be concluded at a 95 percent confidence level. First results (2.5 years) indicate movements which generally support the notion of a dominating displacement pattern due to the postglacial rebound of Fennoscandia. Relative horizontal movements along the baselines are also indicated.

In the KBS-3 concept 4 500 canisters are buried in solid rock at a depth 400 – 700 m below the ground surface. The canisters are emplaced in parallel tunnels at a spacing of about 6 m. The distance between the tunnels is about 25 m. The canisters emit heat due to radioactive decay in the nuclear waste. The heat sources from all canisters create a complex three-dimensional, time-dependent temperature field in the ground in and around the repository. The heat emission decreases with time. However, the emitted heat warms the rock and induces a thermoelastic stress field. The stress and strain fields are of interest, since they may influence the conditions for fracture closure, opening or propagation. A theoretical study was initiated at Lund University, to analyse the thermoelastic process in the rock caused by heat sources in a site specific perspective as a function of time. A particular aim for the analytical approach was to gain a physical understanding to quantify particular processes and their interactions. Exact analytical solutions for the time-dependent, three-dimensional responses have been derived for the repository and for a finite line source. The solutions may be used as boundary conditions in numerical modelling of the local processes around a canister. The presented solution could also be used to verify numerical models.

7.3.4 Chemistry

The chemistry program investigates radionuclide chemistry, needed for the safe disposal of nuclear waste, i.e. solubility of radionuclides, mobility and retention of radionuclides in repository barriers (i. e. buffer, backfill and rock), and non-radioactive chemical conditions in groundwater, bentonite and concrete.

Solvent extraction technique has been critically evaluated as a method to determine the chemical constants for actinides. The evaluation is presented in a technical report. The kinetics of reduction of technetium and neptunium in groundwater has been thoroughly investigated by Daqing Cui and presented in his Doctoral Thesis at the Royal

Institute of Technology, Dept. of Nuclear Chemistry, Stockholm. This investigation is important because in the low redox state these elements are much less soluble and mobile.

Fundamental studies of sorption are supported by SKB and an investigation of uranium sorption, in co-operation with Los Alamos National Laboratory, has been completed and presented. Experiments have been performed with sorption of strontium, cobalt and caesium on fracture minerals. These minerals have a high capacity for sorption and if it can be demonstrated that the fracture fillings are thick enough this could be an important "radionuclide trap", for example, for caesium.

Electric conductivity is being tested as a method to measure rock matrix diffusion (diffusion in the connected micropores of seemingly intact crystalline rock). The aim is to develop a fast and reliable method to measure diffusion which can also handle relatively thick samples. The latter is necessary in order to avoid disturbances from fractures extending through the samples. Experimental data on matrix diffusion in general have been compiled and reviewed with the aim to provide an input to the far-field transport calculations for SR 97. A new and very promising approach to define the diffusivity values was tested.

Diffusion of caesium, strontium and iodine in compacted bentonite clay has been accurately measured and reported. The experimental observations support the view that negative ions are hindered by anion exclusion and that positively charged caesium and strontium, on the other hand, get their mobility enhanced by surface diffusion (mass-transport in the concentrated layer of cations at the negative clay mineral surfaces). Diffusion data are being compiled for SR 97.

Subsurface micro-organisms from Stripa and Äspö have been studied and the results have been summarised by Susanne Ekendahl in her Doctoral Thesis, presented at University of Göteborg, Dept. of General and Marine Microbiology. A functional ecosystem in the deep granitic groundwater environment was indicated and it was concluded that the biosphere extends down into the geosphere. Microbiological investigations were introduced in our investigations in 1987 and the work is presently concentrated on methanogenes which seems to play a key role in the deep subsurface geosphere. Efforts are therefore made to improve the quality of sampling and analysis of gas in groundwater.

Survival of sulphate reducing bacteria in compacted bentonite has been studied. The low water activity (chemical activity) in compacted bentonite is lethal for sulphate reducing bacteria. This was first observed in the microbial analysis of sand/bentonite backfill used in the Canadian Buffer/Container Experiment at AECL's Underground Research Laboratory. The international microbial study of the Buffer/Container Experiment was co-ordinated by AECL and supported by AECL, ANDRA and SKB.

Cement is very useful in underground construction and has many applications but it may increase the pH of groundwater in the near-field. Laboratory experiments are therefor made to test the influence of high pH solutions (simulated concrete pore water) on geochemical conditions. The experiments are carried out at British Geological Survey and jointly supported by Nagra, Nirex and SKB. This is referred to as a "laboratory validation experiment". Three other laboratory validation experiments are in progress: 1) Radionuclide migration in over-cored fractures; 2) simulation of near-field release from a clay buffer to a rock fracture and; 3) colloid migration in a clay buffer.

7.3.5 Natural analogue studies

Natural analogues are studied to support performance assessment. Analogues can be used to justify the assumptions made and validate models. Many of the analogue projects are performed as international projects to share the costs and the large amount of work. Participation from organisations in different countries add different views to the study and promote a critical review of the results which is of great value. SKB is presently involved in three such international analogue projects: Oklo, Palmottu and Jordan. Data gathered in previous projects that has already been terminated, such as Poços de Caldas and Cigar Lake has been revisited. The data from trace element analysis of groundwater in Poços de Caldas were utilized to test a model for solubility that included co-precipitation and co-dissolution (previous modelling assumed pure solid phases). Cigar Lake data was further evaluated to test, among other things, a new model for radiolysis of spent fuel and to compare the physical properties of compacted bentonite with Cigar Lake clay. The radiolysis model was a large step forward and the clay comparison demonstrated that bentonite could deteriorate considerably and still be superior to Cigar Lake clay, which, despite its low quality, functions very well as a barrier to groundwater flow and radionuclide dispersal. A geochemical computing package was improved and successfully applied to the Cigar Lake deposit. The results were used to distinguish the more recent low temperature processes and it was shown that the hematite concentrations adjacent to the orebody/clay interface is likely to have been formed by hydrothermal alteration and not, as previously anticipated, by radiolysis oxidation.

The first Oklo natural reactor was discovered in 1972 during routine quality control analyses in a uranium processing plant in France. Further investigations showed undeniably that the ore deposit, which had formed about 2 billion years (2 109 years) ago, had undergone a period of sustained fission reaction. In 1991, a project "Oklo, Natural Analogue for Radioactive waste Repository" was started by CEA, with additional support by the European Commission. Organisations from other countries, including SKB, have participated and this study was finished and reported in 1995. A continuation in a second phase of that project has been decided and SKB has joined the organisations CEA and ENRESA as a full member of the new EC supported project. The Oklo project, Phase 2 began officially in June 1996 and will continue for 3 years. A field expedition was mounted in late 1996 to ensure water samples and hydraulic measurements from the Bangombé site which, at that time, was likely to be mined out. The field mission was successful and studies of the samples have been initiated at laboratories in Denmark, France, Spain, Sweden, etc. Previous analyses of colloids, microbes and dissolved organic matter in groundwater from Bangombé has recently been reported and the results are remarkably consistent with what is found at investigation sites in Sweden and elsewhere. The conditions and development of criticality in Oklo was carefully investigated by R Naudet at CEA and published in his book in 1991 (see section 15.5.2) and Oklo is the only place on earth with evidence of natural criticality. SKB has, with permission, translated this part of the book from French to English and used the information to compare Oklo with the conditions in a deep repository for spent fuel.

The new Palmottu project started in November 1995 and is expected to continue until 1999. It is managed by the Geological Survey of Finland, GTK, and SKB participates together with ENRESA and BRGM. The site is a uranium mineralisation at Lake Palmottu in Finland where the ore forms 1 – 15 m thick subvertical zone that extends to a depth of about 300 m. The orebody and surrounding rock is investigated as an analogue to spent nuclear fuel in a granitic rock repository. A new deep borehole have been drilled in Palmottu and used for sampling and investigations of the rock and groundwater in order to up-date the structural and hydraulic model of the site. The mobile groundwater chemistry laboratory, developed by SKB, was used for groundwater sampling. The first part of the Palmottu project will be finished and reported in spring 1997, and evaluated by EU before any continuation is decided.

A natural analogue to concrete has been found in Maqarin, Jordan. Bituminous marl, with an organic content of 15 – 20%, has been burnt in-situ to a cement-like material after spontaneous ignition, possibly caused by pyrite oxidation. The metamorphic zone of cement-like material lies within the bituminous marl. The zone extends through the area with strongly varying thickness of about 2 – 3 m to a maximum of 60 m. When groundwater comes in contact with the metamorphic zone it develops a composition close to that of cement pore water; pH is 12 – 13 and the water is rich in calcium, sodium, potassium and sulphate. The water and the minerals in the hyperalkaline areas in Maqarin has been studied as a natural analogue to the use of cement in a repository. Minerals normally found only in cement and concrete are abundant in the area and the environment in Maqarin is also rich in elements which occur in the waste as nuclides. The Maqarin project started in 1990 with funding from Nagra, Nirex and Ontario Hydro. SKB has been participating since 1991. EA (Environmental Agency, UK), Nagra, Nirex and SKB are jointly supporting the present third phase which is co-ordinated and administered by SKB. The investigations are concentrated, but not limited, to Maqarin. Areas in Central Jordan have also been visited. The project is now in its third phase; Phase 1 has been reported, and Phase 2 and 3 are going to be reported.

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 1996 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 is produced in relatively minor quantities in Sweden. The main sources are waste from research and used reactor components which have been situated inside or near the reactor core, i. e. core components and reactor internals. Core components are stored at CLAB and research waste is collected, conditioned and stored at Studsvik. The concept is to finally dispose of this waste in an annex to the deep repository for spent fuel. The annex will be separated from the facility by about 1 km. A new design has been made, based on the experiences from SFR. The annex consists of two caverns with an inner concrete construction similar to SFR-BMA and a system of interconnected tunnels, see Figure 13-4. Waste from Studsvik and operational waste from CLAB and the encapsulation plant will be placed in one of the caverns. Reactor core components and internal parts will be disposed of in the other. Decommissioning waste from CLAB and the encapsulation plant will finally be placed in the tunnels after the caverns have been filled. Concrete will be used as backfill in the inner parts and crushed rock in the space between the inner constructions and the rock walls. The various parts are to be sealed with plugs. The guiding principle is to direct any flowing groundwater past the inner constructions and waste packages.

The total estimated waste volume is about 25 000 m^3. All of the waste does not belong to the category long-lived waste. About half of the total volume is waste which, if produced today, would go to SFR such as operational waste and decommissioning waste from CLAB and the encapsulation plant. The reason for allocating short-lived ILW and LLW to the deep repository is that it arises in the post-closure period of SFR, according to present plans.

A prestudy of the annex repository, including an assessment of its near-field barriers to radionuclide dispersal, was concluded and reported in 1995. In the beginning of this year, 1996, it was decided to include 'other long lived waste than spent nuclear fuel' in the performance assessment report SR 97. The report SR 97 is focused on spent fuel disposal but it will also cover the performance of the annex. It will be high-lighted in the main report SR 97 and further information added in an Appendix. Aims and timeplans have been set up in accordance with the 96 decision and contain the following points:

- Support development of a new design.
- Revise the waste characterisation report with updated tables of waste composition and radionuclide content.
- Compile data on water chemistry, concrete chemistry, radionuclide sorption, diffusion, solubility, organic complexes and colloids.
- Calculate hydraulic flow in the near-field of the repository.
- Develop and apply near-field transport models.
- Analyse scenarios and processes for the performance of repository design.

Disposal of long-lived LLW and ILW is being studied in other countries too, and informal information exchange has been established between SKB and the organisations ANDRA (France), Nagra (Switzerland) and Nirex (the UK). This has been very valuable, not least for the efforts to compile data on radionuclide chemistry. Investigations in this field are being actively pursued by all four organisations.

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 1996 /7-2/ and Chapter 17.

Geoscientific investigations on Äspö and nearby islands began in 1986. Since then, bedrock conditions have been investigated by several deep boreholes, the Äspö Research Village has been built and extensive underground construction work has been undertaken in parallel with comprehensive research. This has resulted in a thorough test of methods for investigation and evaluation of bedrock conditions for construction of a deep repository. The construction of the Äspö Hard Rock Laboratory was finished in 1995. To mark this important milestone in the history of the Äspö Hard Rock Laboratory SKB organized "Äspö 96" as a combined 10-year anniversary and inauguration of the Äspö Hard Rock Laboratory facilities.

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 loops 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.

Work on the final reports for the Stage Goal "Verification of pre-investigation methods" is in progress. There will be five final reports. One providing an overview of the investigations performed at Äspö and the surrounding region during the first 10 years. Three reports on the comparison of predictions based on pre investigations and outcome. Finally, there will be a report presenting the current model of Äspö based on investigations performed to date.

The objective of the ZEDEX project is to compare the mechanical disturbance to the rock for excavation by tunnel boring and blasting. The results from ZEDEX indicate that the role of the EDZ as a preferential pathway to radionuclide transport is limited to the damaged zone. The extent of the damaged zone, which is the hydraulically significant part, can be limited through application of appropriate excavation methods.

The rock surrounding the repository constitutes a natural barrier to release of radionuclides from a deep repository. The most important function of the natural barrier is to provide protection for the engineered barriers through stable chemical and mechanical conditions and to limit transport of corrodants and radionuclides through slow and stable groundwater flux through the repository and reactions of radionuclides with the host rock.

The objectives of the Fracture Classification and Characterization project are to develop methodology for characterization of fractures with respect to tectonic evolution, infillings and wallrock alteration and to use this information for classification of fractures in terms of their importance for radionuclide transport. The only striking difference found between individual water-conducting features is the internal fault geometry. No other distinguishing criteria (such as lithologic domains, mineralogy of fracture infills, transmissivity etc) were identified. On the basis of the geometric arrangement of master faults and splay cracks five types of water-conducting features could be distinguished.

The Tracer Retention Understanding Experiments are made to gain a better understanding of radionuclide retention in the rock and create confidence in the radionuclide transport models that are intended to be used in the licensing of a deep repository for spent fuel. During 1996 a series of tracer experiments in radially converging and dipole flow configuration have been performed. These tests have been subject to blind predictions by the Äspö Task Force on groundwater flow and transports of solutes. A preliminary comparison between model predictions made by the Äspö Task Force and experimental results, shows that most modeling teams predicted breakthrough from all four injections, although some teams predicted distinctly lower mass recoveries from the two injections which in-situ did not produce breakthrough. The breakthrough times predicted by the modeling teams are also in accord with those observed in the experimental results.

The objective of the Resin technology development is to establish a technique by which a description of the pore space of a feature investigated with tracer tests can be mapped by epoxy resin injected into the feature. The pore volume is measured in a number of sections /slices of the fracture using a combination of photographic and microscopic techniques and subsequent image processing. The obtained data is planned to be used to reduce uncertainties in the description of the heterogeneity of the studied feature. Three 200 mm cores were drilled to sample the fracture that had been injected with resin. There were problems obtaining intact core samples and a refined drilling technique using a smaller diameter had to be developed.

The main objective of the TRUE Block Scale Experiment is to increase understanding and our ability to predict tracer transport in a fracture network over spatial scales of 10 to 100 m. The TRUE Block Scale Experiment has been initiated as a joint project between ANDRA, Nirex, Posiva, and SKB. The total duration of the project is approximately four years from the start in July 1996.

The REX project focuses on the reduction of oxygen in a repository after closure due to reactions with rock minerals and microbial activity. A field experiment will be performed where the consumption of oxygen in contact with a fracture surface will be studied. Laboratory measurements are in progress and preparations have been made for the field experiment which will start in 1997. Preliminary measurements of bacteriological oxygen consumption and dissolved methane and hydrogen in Äspö groundwaters have been performed. These results show that oxygen may be consumed by methanotrophic bacteria in a closed nuclear waste repository.

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 probe has been delivered and the first tests performed at Äspö.

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. A degassing and two-phase flow test was conducted at the TRUE resin site. In the test water with a gas contents of about 17% was injected into the fracture. A flow reduction of 50% was observed when the pressure in the withdrawal hole was reduced to atmospheric. Degassing is considered to be the most likely explanation for this behavior.

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.

The Prototype Repository Test is focused on testing and demonstrating repository system function. The Backfill and Plug Test includes tests of backfill materials and emplacement methods and a test of a full scale plug. Planning and preparations for these experiments has continued during 1996. Demonstration of methods for deposition and retrieval of canisters will be made in a new tunnel excavated at the 420 m level.

The Long Term Tests of Buffer Material aim to validate models of buffer performance at standard KBS-3 repository conditions, and at quantifying clay buffer alteration processes at adverse conditions. Two tests holes have been instrumented and the temperature raised to 90 and 120°C, respectively.

Eight organizations from seven countries are currently participating in the Äspö Hard Rock Laboratory in addition to SKB. The results of this work is reported in the Äspö International Cooperation Reports.

7.6 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.

The possibility for partitioning and transmutation is still attracting considerable interest. SKB has since the early 1990s supported work in this area at the Royal Institute of Technology (KTH) in Stockholm and at the Chalmers Institute of Technology (CTH) in Göteborg. The support from SKB is increased from 1997 and is coordinated with other work by the various groups in Sweden active in this field. The work supported by SKB at KTH is emphasized on systems and safety studies and at CTH on studies of processes for partitioning. Both groups have a broad international cooperation.

SKB runs a separate research project on the "Very Deep Hole Concept" aiming at improving our knowledge about conditions of 1000 – 5000 m in the geosphere. During 1996 the project has continued its state-of-the-art compilations of data bases in terms of tectonics, hydrogeology, geophysics, rock mechanics and geochemistry. Among other activities, Russian experiences has been reported on petrophysical measurements, lithological and reflection seismic data in the vicinity of a super-deep drilling project in Middle Ural.

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 that is set annually by the government is based on 25 years operation of each power plant.

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 1996 calculations were presented in PLAN 96 /8-1/. The total electricity production from the start of the reactors was estimated to be about 1 650 TWh, if all twelve reactors are operated for 25 years. Up to the end of 1996 about 1070 TWh had been produced. The amount of fuel required for 25 years production will be about 6 400 tonnes of uranium. If the reactors are operated for 40 years, 2 700 TWh will be produced and 9 500 tonnes of U will be required.

In PLAN 96 a new cost calculation method has been applied, based on statistical principles. The costs are calculated based on a reference scenario for the measures, facilities etc that are needed to manage and dispose of the spent nuclear fuel and to decommission and dismantle the nuclear power plants. This scenario necessarily contains uncertainties. In order to take these uncertainties into consideration the cost components are regarded as stochastic variables. Due regard is taken to variations in the assumptions made on the technical system, the time schedule and the cost level of different items.. The uncertainties are then dealt with by means of a statistical weighing-together of their influences on the costs. The result is presented as a distribution function, which gives the probability that the cost will be below a certain cost.

The total future back-end costs for 25 years of operation were estimated to be about GSEK 42.2 (price level of January 1996) 1 GSEK = 10^9 SEK = 0.13 $*$ 10^9 US$. Up to and including 1996 already GSEK 10.6 have been spent. The total cost for the back-end of the nuclear fuel cycle is thus about GSEK 52.8. The breakdown of the costs are roughly (old reprocessing costs excluded):

Transportation of waste	6%
Interim storage of spent fuel	20%
Encapsulation and final disposal of spent fuel and long-lived waste	34%
Final disposal of operational and nuclear power plant decommissioning waste	7%
Decommissioning and dismantling of nuclear power plants	23%
Miscellaneous including R&D, pilot facilities, and siting	10%

The cost figures given are the best estimate of the future costs, i.e. the cost that have 50% probability to be exceeded or to be kept, which should be the basis for the fees. To cover contingencies etc the utilities should provide guarantees. This is in accordance with the revisions of the Act on Financing that was decided by Parliament at the end of 1995. The most important changes were:

- 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.
- 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 funds shall be administered by a new organization called "The Nuclear Waste Fund".

Based on SKB's cost calculations and the estimated future real interest rate, the government has decided that the fee for 1997 shall be SEK 0.011 per kWh on an average. This is substantially lower than last year. The main reason for this is the more favourable interest earned on the funds as from 1996, with a guaranteed real rate of return of more than 4% on a large share of the funds. At the same time the utilities were required to put up guarantees corresponding to GSEK 7.6.

The total amount in the funds was at the end of 1996 about GSEK 20.8, an increase by GSEK 2.9 during 1996.

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

During 1996 only limited studies have been done in the field of decommissioning of nuclear power plants. These have in particular been concerned with collecting experiences applicable to decommissioning from the important refurbishment work being performed at the Swedish reactors.

SKB's engagement in the OECD/NEA international cooperative programme on decommissioning has continued during 1996. SKB is responsible for the programme coordinator function. The programme comprises 30 decommissioning projects in 11 countries. The majority of the projects are small first generation power demonstration reactors /8-2/.

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 CONSULTING SERVICES

9.1 BACKGROUND

The international review of the KBS reports (1978-84) made SKB's activities internationally recognised. Since then SKB has actively participated in international co-operation activities and strengthened its position as an attractive partner. As a consequence foreign organisations 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 in operation 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 organised 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 140 assignments have been accomplished for organisations 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.

9.2 NWM WORK DURING 1996

During 1996 SKB was contracted by organisations in Belgium, Estonia, Finland, France, Japan, Lithuania, Spain, Taiwan and Belarus. In parallel marketing activities have been going on in Southeast Asia, South Africa, Russian Federation and EC TACIS program. In all some twenty assignments have been concluded, distributed over ten countries during the year.

The realization of the intermediate dry spent fuel storage in Lithuania has continued to be of high priority. Some ten storage casks have been delivered from the German company GNB during the year. Construction of the storage area is scheduled for autumn 1997 and commission of the storage facility during spring 1998. In parallel a new tendering round has been carried out in which high priority has been given to the possibility of local manufacturing in Lithuania.

As a result of the earlier developed national waste management strategy plan, safety analysis have been initiated for three existing repositories for low and medium level waste in Lithuania.

In Estonia preparatory work has been going regarding sorting and conditioning of low and intermediate level waste at the Paldiski site where two shore based nuclear submarines are under decommissioning.

Radar reflection measurements as well as TV-logging has been carried out on behalf of Posiva OY in Finland in several deep boreholes on different candidate sites for a final repository for spent fuel.

"Peer reviews" have been carried out on two spent fuel storage facilities on behalf of BNFL in England.

On behalf of Instituto Tecnologico GeoMinero (ITGE) in Spain, continued advice and design review has been going on in connection to the development of hydrogeological borehole instruments.

A complete set of a radarreflection equipment (RAMAC) has been delivered to Taiwan Power Co in Taiwan.

The decision by the Swedish government to support the Russian Federation in the "clean up" of the Kola peninsula from primarily spent fuel has resulted in a rather important engagement of SKB. SKB has taken the initiative to form an Industrial group consisting of Kvaerner Maritime (Norway), SGN (France), BNFL (England) and SKB. Negotiations are in progress with the Russian counterpart Minatom regarding a number of projects considered to be of very high priority.

Discussions with ESKOM in South Africa is in progress regarding a site investigation program for a planned repository for spent fuel at Vaalput.

The Hungarian NPP PAKS has consulted SKB regarding the preparatory work for a planned underground repository for low- and intermediate level waste.

10 PUBLIC AFFAIRS AND MEDIA RELATIONS

10. 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 electricity 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.

10.2 NOTABLE INFORMATION-RELATED EVENTS

Every year SKB pursues various types of ambulatory information activities. Activities in schools are an important example. With lecturers and mobile exhibitions, SKB visited 7 710 pupils in 369 classes during 1996. SKB has noticed a steady increase in interest among teachers over the years, and 13 teacher conferences, in which 285 teachers participated, were therefore also arranged in conjunction with the school visits. The school information package "At Depth" was most recently revised after an evaluation in 1995 and is now available at most upper-secondary schools in the country. SKB also took part in six trade fairs of various kinds.

During the summer the transport ship M/S Sigyn served as a floating exhibition hall. For the eight year in a row, visitors (45 000 in 1996) were able to view equipment that is used to handle the waste, such as transport casks. They

Figure 10-1. On board the Sigyn, visitors get to learn more about a future deep repository.

Figure 10-2. In every municipality where a feasibility study is conducted, SKB has a local office to which the municipal residents are welcome to come with their questions.

also had a chance to enter a model fashioned like a tunnel, in which a genuine copper canister was emplaced just like in a future deep repository. In addition to the exhibition, 12 seminars were held on board, to which various non-government organizations, politicians and the general public were invited to debate ethical and environmental issues and various future-oriented topics.

SKB's facilities CLAB, SFR and the Äspö Hard Rock Laboratory also received a large number of both Swedish and foreign visitors. Among the visitors were delegates from the European Nuclear Society conference in Stockholm and several groups from the Swedish municipalities in which feasibility studies have been conducted or are planned.

SKB's exhibitions were visited by a total of 85 000 persons including members of the public, upper-secondary school pupils, community leaders and special-interest groups.

During the year, SKB ran a series of advertisements in the daily press and trade journals dealing with the nuclear waste issue from an ethical perspective. The ads addressed the questions: "Should we wait until the next generation to come up with better methods to deal with our nuclear waste?" and "Should we pass on responsibility for dealing with our nuclear waste to coming generations?" As a follow-up to the ad campaign, a publication entitled *Conversations on the Sigyn* could also be ordered from the company, free of charge. The publication describes the discussions concerning nuclear waste disposal that occurred during the seminars that were held on board the Sigyn during the summer information tour.

Lagerbladet, SKB's newsletter, was published four times during 1996 and distributes to about 25 000 subscribers.

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Part II

Research, Development and Demonstration during

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CONTENTS PART II

11 RD&D-PROGRAMME

11.1 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. In late September 1995 SKB submitted its fourth RD&D-programme to the Nuclear Power Inspectorate – SKI /11-1/. The programme covers the period 1996-2001 in some detail.

The programme was sent by SKI for review to about 60 organizations and SKI received comments from about 35 of these organizations. Based on these comments and on their internal evaluation and a few consultants review SKI issued its evaluation reports /11-2, 11-3/ to the government in May 1996. The programme was also reviewed by KASAM, the government's scientific advisory committee on nuclear waste, which sent its report to the government by the end of June 1996 /11-4/. Section 11.2 gives the summary of SKI's evaluations and proposals, section 11.3 the summary from KASAM's report. The government decision on the programme was taken on December 19, 1996. It is given in unofficial translation in section 11.4.

11.2 SUMMARY OF SKI's EVALUATION AND PROPOSALS ON RD&D-PROGRAMME 95

(Reproduced from /11-2/)

Summary of SKI's Evaluation and Proposals

In SKI's opinion, the Swedish Nuclear Fuel and Waste Management Co (SKB) has presented a research and development programme which fulfils the basic requirements stipulated in 12 § of the Act on Nuclear Activities. On the whole, the programme is appropriate with regard to the development and evaluation of a method for the final disposal of spent nuclear fuel and nuclear waste in the Swedish bedrock. The quality of the supporting research programme is high. However, in SKI's view, the General Siting Study should be supplemented with regard to certain points and the siting factors should be further specified.

For some years, SKB's programme has been in a transitional phase, moving from research and method development work to the planning, design and siting of facilities, initially on a pilot scale. Thus, in SKI's view, it is important that SKB should more precisely focus the future RD&D programme on gathering the information that the Government, regulatory authorities and municipalities concerned need for the licensing of these facilities in accordance with different legal acts, particularly, the Act on Nuclear Activities (with simultaneous licensing in accordance with the Radiation Protection Act) and the Act concerning the Management of Natural Resources etc. Such licensing actions include an overall evaluation of the entire final disposal system proposed by SKB. SKB must describe alternative solutions, including a zero alternative, in accordance with the current requirements on environmental impact statements, as well as take into account the additional guidelines issued by the Government in its decision of May 18, 1995. To provide guidelines for such a more detailed direction of the RD&D Programme, SKI describes in this evaluation and the attached Appendix 1 *(to SKI's report /11-2/)* what SKI considers, as the authority drafting proposals to the Government in accordance with the Act on Nuclear Activities, should be reported in connection with licensing under various legal acts. In Appendix 1, SKI describes proposals for various stages in the licensing process in order to ensure that the safety link between different facilities in SKB's proposed main system alternative are satisfactorily taken into consideration. The proposals are mainly intended to be used as points of departure for a continued dialogue on these issues with the various parties concerned. More formalised guidelines for the various stages in the licensing process can and should be provided by SKI and other regulatory authorities in the form of regulations and general recommendations as well as by the Government in connection with the first licensing decisions.

In SKI's opinion, it is particularly important that SKB should perform and present an in-depth and comprehensive safety assessment of SKB's proposed main system alternative (KBS-3), before pursuing work and making further decisions which would entail increasingly stronger commitments to the system as a whole as well as to the detailed design of all of the parts of the system. In SKI's view, there are several sound reasons why such a safety assessment should be presented and subjected to evaluation by national and international experts. The reasons are described in detail further on in this evaluation report. Thus, SKI proposes that the Government, with the support of 12 § of the Act on Nuclear Activities, should impose the condition that before SKB initiates site investigations, SKB should present an in-depth and comprehensive safety assessment of SKB's proposed main system alternative as well as commission and present an independent evaluation of the safety assessment by national and international experts. In connection with such a condition, SKI proposes that the Government commission SKI to, in consultation with SSI, issue guidelines for the structure of such a safety assessment including a subsequent evaluation by independent experts.

In its review statement and accompanying background reports, SKI presents, on the basis of its own evaluation and the comments of external reviewing bodies, a number of views on the direction and structure of the continued RD&D programme which SKB should take into account.

1. Background to SKI's Review Statement

On September 30, 1995, SKB submitted to the Swedish Nuclear Power Inspectorate (SKI) the programme for research and development concerning the handling and final disposal of nuclear waste and the decommissioning and dismantling of nuclear facilities as stipulated in 12 § of the Act on Nuclear Activities. The programme is called *Treatment and Final Disposal of Nuclear Waste; Programme for Encapsulation, Deep Geological Disposal and Research, Development and Demonstration* (RD&D Programme 95). SKI has also evaluated two other SKB reports, *Siting of a Deep Repository for Spent Nuclear Fuel* (General Siting Study 95) as well as *Template for Safety Reports with Descriptive Example.*

In accordance with the Act on Nuclear Activities, SKI must examine and evaluate the programme as well as submit the documents on the matter to the Government, together with its own review statement, within six months. SKI has requested and been granted permission by the Government for an extension of the evaluation period by two months.

SKI has submitted RD&D Programme 95 to a large number of reviewing bodies for comment. SKI has used both Swedish and foreign experts in its evaluation.

SKI's entire evaluation consists of this *Statement* and Appendix 1, which has been decided upon by SKI's Board, as well as the underlying *Review Report*, the contents of which are summarised in Summary and Conclusions. Furthermore, a special report has been prepared in which the *Comments of the Reviewing Bodies* have been compiled.

2. Premises of SKI's Evaluation

In its statement of March 1993 concerning the previous RD&D Programme 92, SKI proposed that a complete safety assessment should be submitted to SKI no later than at the time when SKB starts its detailed characterisations of possible deep repository sites, that SKB should submit a plan for describing the design basis of the canister and that the technical criteria for the site selection should be reported.

In the Government's decision of December 1993 concerning SKB's RD&D Programme 92, SKI's proposal was adopted and the Government required that SKB should submit supplementary information on its programme. SKB submitted such a supplement in August 1994 and the report was evaluated by SKI in the same way as RD&D Programme 92. In SKI's evaluation of the supplement, SKI emphasised, in particular, that factors and criteria for further siting work must be specified and quantified as well as that the design basis of the canister must be described. SKI emphasised that the feasibility studies should be completed before the sites for investigation are selected and that a site investigation plan should be presented before these investigations were initiated.

In accordance with the Government's decision of May 1995 concerning SKB's supplement to RD&D Programme 92, SKB must describe its plans for the specification of technical requirements for the repository including the canister, on the basis of safety assessments, as well as for the investigation of possible repository sites. Furthermore, general siting studies and feasibility studies must be compiled and presented in future R&D programmes.

In its decision, the Government declared that the implementation of detailed characterisations must be considered to be a stage in the construction of a nuclear installation as well as that, for the licensing of the first facility in the final disposal system (the encapsulation plant, in accordance with SKB's plans), SKB must present data and other information which would allow the final disposal system to be evaluated, as a whole. Furthermore, the Government stated that the Government intends to give SKI the responsibility of, in accordance with the Ordinance on the Act concerning the Management of National Resources, etc., after consulting with the Swedish Radiation Protection Institute (SSI), the Swedish Environmental Protection Agency (SNV) and the county administrative boards concerned, submitting information to the county administrative boards on areas which are of national interest for the final disposal of nuclear waste. The Government also emphasised the fact that SKI, in accordance with the Ordinance on Nuclear Activities, may stipulate that an Environmental Impact Statement (EIS) which enables an overall evaluation of a planned activity to be made should be prepared for licensing actions under the Act on Nuclear Activities.

In January 1996, the National Council on Nuclear Waste (KASAM), SKI, SSI, the Municipality of Oskarshamn and the County Administrative Board of Kalmar submitted to the Government a joint document concerning the need for the co-ordination of the siting process for the repository on a national level.

The comments of the reviewing bodies mainly deal with issues concerning the decision-making process and the basis for decision-making which is needed in different contexts. The KBS-3 disposal method is discussed in relation to the current state of knowledge and the need for safety assessments. The Municipality of Oskarshamn emphasises the link between the encapsulation plant and the deep repository. The comments of the municipality must be accorded special importance since SKB intends to locate the encapsulation plant in connection with the central interim storage facility for spent nuclear fuel (CLAB) and the municipality is considering whether a feasibility study must be carried out in the municipality.

In the light of the contents of the Government's decision and the joint document submitted to the Government as well as the comments of the reviewing bodies, SKI has largely focused its evaluation on decision-making processes and the need for a basis for decision-making at different stages in order to develop and implement the programme proposed by SKB. In this connection, SKI has focused, in particular, on ensuring that the safety links between different facilities in SKB's proposed main system alternative are satisfactorily taken into consideration. In SKI's view, it is important that SKB's future RD&D programme should be structured so that it is focused on obtaining the information that the Government, regulatory authorities and municipalities concerned need for the licensing of the facilities concerned in accordance with different legal acts, particularly, the Act on Nuclear Activities (with simultaneous licensing in accordance with the Radiation Protection Act) and the Act concerning the Management of Natural Resources etc. The comments of the reviewing bodies also focus on such issues.

3. SKI's General Conclusions and Evaluations

The conclusions and evaluations of more purely technical issues relating to the encapsulation plant, the canister and the deep repository are treated in the attached Summary and Conclusions and are reported in detail in the attached review report. Some general conclusions and evaluations are presented below. The subsequent analysis of the decision-making process and need for a basis of decision-making are based on these premises.

System Studies and Alternative Methods

The KBS-3 method has been accepted by the regulatory authorities and the Government as the main line for further development work. However, ultimate approval of the method has not yet been given. An up-to-date, comprehensive and in-depth safety assessment is necessary for the evaluation of the entire proposed system which must be carried out no later than in connection with the licensing of the first facility in the system as well as for the systematic focusing and prioritisation of further RD&D work. Alternatives to the KBS-3 method must be described and must be further investigated, especially the zero alternative, as a basis for further decision-making concerning the ultimate choice of a disposal system. In SKI's opinion, with regard to alternatives which are being discussed on an international level, transmutation is currently not a realistic disposal alternative for Sweden. In order to develop this method, considerable investment would be required in terms of time and money. Even if it is successful, in a realistic industrial process, a far from negligible quantity of long-lived radioactive residual products will remain after transmutation. It is questionable whether the volume reduction which is achieved counterbalances the increased complexity of the waste management process, which would also entail reprocessing.

Siting of the Encapsulation Plant and the Deep Repository

Even if alternative encapsulation plant sites are possible, it is natural that the plant should be located at CLAB. The process of siting the deep repository may be considerably delayed in relation to SKB's plans. SKB is experiencing obvious difficulties in attaining 5-10 feasibility studies and is also experiencing obvious difficulties in obtaining the acceptance of the communities for further studies in the municipalities. Currently, feasibility studies have been carried out in two municipalities – Storuman and Malå – and are currently in progress in two – Nyköping and Östhammar. The Municipality of Oskarshamn is considering whether a feasibility study should be conducted within the municipality. Following a local referendum, the Municipality of Storuman rejected further studies in the municipality. The siting process must be improved in several respects. Among other things, the municipalities would like more transparent information and clearer decisions to be made with regard to the system selected and the siting criteria. Therefore, for example, the Municipality of Oskarshamn states that, before any decision is made by a municipality to participate in site investigations, all of the feasibility studies must be completed and evaluated by SKI and SSI. Furthermore, a site investigation programme must exist. In SKI's view, SKB's general study on the siting of the deep repository for spent nuclear fuel must be supplemented in several respects, e.g. the consequence of siting such a repository on the coast or inland as well as the advantages and disadvantages of siting the repository in southern or northern Sweden. Furthermore, there are a number of questions which must be resolved, including, biosphere-related issues and recharge and discharge areas.

The Municipality of Oskarshamn further states that the municipality cannot take a decision on an application for a permit to construct the encapsulation plant before an application has been submitted for a permit to carry out detailed characterisation of a deep repository site. SKI observes that this requirement from the Municipality of Oskarshamn means that the question of a permit for the siting and construction of the encapsulation plant in the municipality cannot be evaluated until the site investigations have been carried out if the municipal veto is to be respected.

Supporting R&D

SKB's supporting research and development (R&D) is of a satisfactory quality. However, SKB needs to more clearly identify and exploit the links to the needs of the safety assessment for a technical and scientific basis of information in order to better prioritise its R&D activities. For example, the links between safety assessments and R&D with regard to issues relating to a deep repository and the R&D programme being carried out at the Äspö Hard Rock Laboratory should be better established. Supporting R&D will be needed for a long time in the future even after the encapsulation plant and Stage 1 of the deep repository are completed.

Canister Manufacturing and Encapsulation

SKI observes with satisfaction that SKB is constructing a facility for inactive testing of the sealing of the canisters and non-destructive testing. Considerable work remains to be carried out by SKB before there is a basis for an application for a permit to construct the encapsulation plant. SKB plans to submit such an application within about two years. The realism of SKB's time schedule is questionable since considerable work is necessary within the areas of canister manufacturing, sealing and control.

Safety Assessments

Considerable progress has been made since RD&D Programme 92. Newly developed methodology and models must now be applied and evaluated. Previous evaluations of important safety factors must be reconciled against new knowledge and modifications of the disposal system.

In SKI's opinion, it is particularly important that SKB should perform and present an in-depth and comprehensive safety assessment of SKB's proposed main system alternative (KBS-3), before pursuing work and making further decisions which entail increasingly stronger commitments to the system as a whole as well as the detailed design of all of the parts of the system. In SKI's view, there are several reasons why such a safety assessment should be presented and subjected to evaluation by national and international experts, such as:

- a comprehensive, up-to-date safety assessment is necessary for the evaluation of the entire proposed system which must be carried out no later than in connection with the licensing of the first facility in the system as well as for the systematic focusing and prioritisation of further RD&D work,
- design and quality requirements for the canister must be in agreement with an up-to-date safety assessment of the entire system,
- the design of measurement programmes for site investigations and the evaluation of measurement data from these investigations must be in agreement with an up-to-date safety assessment of the entire system,
- SKB's safety assessment methods and models must be thoroughly evaluated in the light of the developments which have taken place since SKB presented its first detailed KBS-3 analysis 13 years ago.

In this context, SKB should describe, in detail, how safety assessments will be used in order to satisfy the above-mentioned needs. From the above, it is obvious that it is desirable that such a safety assessment should be presented before SKB starts its site investigations.

4. General Conclusions concerning the Decision-making Process and Safety Assessments

There are a number of questions of strategic importance with regard to the process for the evaluation of applications and decision-making concerning permits to site, construct and operate the various facilities in the final disposal system. Safety assessments must be available when reporting and decisions are to be made. Links/relationships exist between different issues which means that an overall evaluation must be made. More formalised guidelines for the structure of the various stages in the licensing process can and should be provided by SKI and other regulatory authorities in the form of regulations and general recommendations as well as by the Government in connection with the first licensing decisions

Examples of questions which must be dealt with are time-schedules for applications and the scope of the basis for decision-making, especially issues relating to long-term safety and safety assessments. The conditions which must be specified in various permits; whether certain conditions can be connected to RD&D evaluations and what can be regulated in SKI's regulations concerning safety assessments and EIA's must be dealt with.

The nuclear facilities concerned are an encapsulation plant and a deep repository. The facilities must be evaluated in accordance with both the Act concerning the Management of Natural Resources etc. and the Act on Nuclear Activities. Licensing under the Act concerning the Management of Natural Resources etc. concerns overall issues such as the siting of the facilities, their nature and scope as well as issues relating to land use etc. Licensing in accordance with the Act on Nuclear Activities focuses on safety-related issues where an overall evaluation must be made of the impact of the activity on human health and the environment. From a practical standpoint, it is possible to co-ordinate licensing within the framework of the applicable legislation. In its decision of May 1995, the Government specified that a reasonable condition is that the municipal council in the municipalities concerned should have access to SKI's statement to the Government concerning licensing under the Act on Nuclear Activities before they make a decision in accordance with the Act concerning the Management of Natural Resources etc. (Section 4). Furthermore, the Government stated that licensing under both acts will be co-ordinated by the Cabinet Office. Licensing in accordance with the Radiation Protection Act will be carried out in connection with licensing in accordance with the Act on Nuclear Activities.

SKI's opinion on requirements concerning technical reporting in connection with an application for a permit to construct the encapsulation plant, as well as in connection with an application for a permit to carry out detailed characterisations and the extension of the deep repository, in stages is provided in Appendix 1. SKI considers technical reports of this type to be necessary for SKI to be able to draft proposals in accordance with the Ordinance on Nuclear Activities. SKI also describes the information which it is desirable to present prior to the start of site investigations. Furthermore, the structure and co-ordination of different stages in the licensing process is discussed. At this stage, the proposed guidelines for applications and licensing processes which SKI presents in Appendix 1 must be considered as points of departure for a continued dialogue on these issues with the parties concerned. Before SKI makes a formal decision on what technical reports must be submitted in connection with various permit applications, a formal review process (including an evaluation by independent bodies) will take place. The Appendix also describes which decisions in connection with the licensing process must be made by the Government.

11.3 SUMMARY OF KASAM's REVIEW OF RD&D-PROGRAMME 95

(Reproduced from /11-4/)

Main Features of the Programme

SKB's programme can be seen as two parts which must be developed in parallel with each other. One part comprises measures that SKB has the authority to implement. The second part comprises measures which require the permission of external parties before they can be implemented.

The first part comprises research, investigations, assessments of repository performance and safety, design, manufacturing and testing of components as well as the design and planning of final disposal facilities. RD&D Programme 95 shows that valuable progress has been made in this area. The second part comprises the siting of the planned facilities and the investigations which must be carried out before applications are submitted for permission to construct these facilities. SKB has not made similar progress in this part of the programme.

There may be reasons for the difficulties experienced by SKB in the siting work over which SKB has no control. One of these reasons may be a lack of transparency with regard to the veto issue. KASAM discusses the veto issue in Section 3.1 *(of KASAM's report /11-4/)*. At the same time, in KASAM's view, there are possibilities for SKB to improve the credibility of its own work and its technical solution to the final disposal issue. A few such possibilities are discussed in the various chapters of this Review Report.

Radiation Protection Principles and Safety Analysis

The safety assessment is a central part of the evaluation of the safety of nuclear installations. The results of a safety analysis are compared with the basic principles for radiation protection (radiation protection standards). Even if these principles are established by the regulatory authorities in accordance with international and Nordic recommendations, continued discussion about their practical application is necessary. KASAM considers it to be important that SKB should continuously describe how the principles can be applied to spent fuel disposal, even if the detailed evaluation can only be made at the time that the licence application is prepared.

System-related Issues

In accordance with the Government's decision of May 18, 1995, SKB must present an integrated safety analysis of the final disposal system as a whole. KASAM considers this to be an important requirement. This is especially important with regard to how risks connected with individual parts of the system can be compared and weighed against each other and with regard to establishing the commitments made to the system as a whole on through decisions concerning individual parts of the system, such as which requirements must be made on the design and construction of the deep repository as a consequence of the selection of a particular canister design.

RD&D 95 does not specify how SKB intends to meet the Government's requirements. For example, SKB's Template for Safety Reports with Descriptive Example (SR 95), only deals with the final repository, and not the encapsulation plant. Furthermore, the transportation system is not discussed. As stated under the heading "Siting", KASAM is of the opinion that SKB should submit an application for a permit to carry out detailed characterisation of a possible site of a deep repository at the same time, or before an application is submitted for permission to construct the encapsulation plant. KASAM emphasises that this must not result in SKB postponing its account of how the overall safety analysis is to be carried out. Instead, it is important that this should be done as soon as possible. Such a report should primarily describe methodology for how principles for safety and radiation protection can be applied to the entire system so that different types of risks can be compared and weighed against each other. In KASAM's view, it is very important that this overall safety analysis should be prepared and subsequently evaluated by the regulatory authorities. This will provide a basis for the further design of the programme.

Transparency and Comprehensibility

The safety analysis must be developed to improve transparency and comprehensibility. KASAM wishes to emphasise, in particular, two approaches which can contribute to this aim:

- Facts, best estimates and opinions must be systematically presented. The latter will mainly be introduced into the analysis through the selection of scenarios.
- The safety analysis may appear to be very complicated, but often, the results are determined by a few basic physical and chemical principles, solubility limitations and dilution. A systematic presentation of how the results from a safety analysis are dependent upon such factors should considerably improve the possibility of communicating the results to others, besides experts.

In KASAM's view, it is important that the scenario selection work should be prioritised in the programme. Scenarios can be selected even if the sites for the encapsulation plant and the repository have not yet been identified. The scenario analyses can contribute to the basis for decisions on the design of the repository and the siting of the repository.

Questions relating to safety analysis are also dealt with under the heading "Siting".

Siting

Municipal Veto and the Government's Right to Override a Municipal Veto

The veto issue is important for the municipalities which are, on a voluntary basis, currently participating or considering participating in feasibility studies for a deep repository. The issue concerns how the feasibility studies relate to the Government's subsequent formal possibility of overriding a veto, i.e. the Government's possibility of granting permission for detailed site characterisation in spite of a municipal veto.

In Section 3.1, KASAM attempts to investigate the municipal veto right and the formal possibilities that the Government has of overriding a municipal veto, especially with regard to the selection of a site for detailed characterisation. KASAM believes that the uncertainty surrounding this question has a considerable impact on municipal decisions and that this counteracts the efforts being made to locate a suitable site for a deep repository.

In KASAM's view, there are reasons which are strongly in favour of keeping the Government's formal possibility of overriding a municipal veto. However, because of the uncertainty that exists, KASAM recommends that the Government should clearly state the circumstances under which its possibility of overriding a municipal veto can be used. This is an important prerequisite for a transparent decision-making process.

Decision-making Process – Time-schedule and Co-ordination

In its review of RD&D 92, KASAM emphasised the importance of ensuring that the decision-making process is open and transparent and that it is also perceived as such by the general public. After the evaluation of RD&D 92, parts of the decision-making process concerning licensing were clarified by the Government in its decision. The decision stated that licensing in accordance with the Act concerning the Management of Natural Resources etc. and the Act on Nuclear Activities is to be carried out simultaneously. However, in KASAM's view, there are still some aspects of the decision-making process and the handling of the issues relating to the system which are unclear.

SKB's time-schedule shows that SKB intends to submit an application for permission to site and construct an encapsulation plant at around year-end 1997, whereas SKB does not intend to submit an application for permission to site and conduct detailed characterisations until the year 2002. On the whole, SKB treats the three main parts of the final disposal system (encapsulation, deep repository and transportation system) separately and the discussion of system-related issues is limited.

According to SKB's plan, when SKB submits an application for permission to site and construct an encapsulation plant, the site investigations will not have been completed. Consequently, KASAM envisages that it will be difficult to compile a complete basis for decision-making by that time. There are mainly two deficiencies which can reduce the credibility of the process:

- It will not be possible to make an evaluation of the alternative siting of the encapsulation plant next to the deep repository site since that site will not be known.
- There will be no data available from the actual deep repository candidate sites.

KASAM recommends that the application for a permit to site the encapsulation plant and the application for a permit to conduct detailed characterisations of a candidate site for a deep repository should be submitted at the same time. This procedure would also mean that a realistic description of the proposed transportation system can be included.

In KASAM's view, all of the stages in the decision-making process for the entire final disposal system must be described in an integrated manner in order to establish the basis upon which the different decisions will be made. It should be possible for the co-ordinator for nuclear waste, recently appointed by the Government, to participate in compiling such a description.

Site Selection Factors

The siting of a deep repository is an issue comprising scientific as well as political aspects. A site must be identified which is sufficiently safe for the final disposal of nuclear waste and which can be evaluated, in geological terms, as being a satisfactory site. SKB has specified a number of site selection factors within the areas of safety, technology, land and environment as well as societal aspects which, according to the Government's decision of May 18, 1995, should be a starting point for further siting work.

In its decision, the Government also requested that SKB submit a general siting study. SKB has now published General Siting Study 95 with general information concerning the Swedish bedrock which, in KASAM's view, should be supplemented. KASAM realises the difficulties of a gradual and systematic site selection programme which is only based on geological and other safety-related factors. Thus, for example, the feasibility studies can only provide very limited information concerning the properties of the bedrock at repository depth at the sites which have been studied. On the other hand, it should be possible for studies on a national and regional level to provide better material for comparison than that provided by SKB's General Siting Study 95.

The site selection factors are generally specified by SKB. The range of values for the factors which can be accepted in order for a site to be considered suitable is not always specified. For a process to be credible, the factors must be more clearly defined than they have been so far. SKB must also specify what knowledge it expects to acquire about the factors at various stages in the site selection process, i.e. prior to the selection of sites for site investigations and prior to the selection of a site for detailed characterisation. KASAM also finds that SKB's programme lacks a discussion of different main siting alternatives, such as the advantages and disadvantages of siting a repository in southern and northern Sweden, or of siting a repository on the coast or inland.

A detailed definition of the site selection factors which are important to safety should be achieved with the help of the safety analysis. General Siting Study 95 states that this shall only be done when the safety assessment for the encapsulation plant is prepared. In KASAM's view, this is too late if the site selection process is to be credible. The factors should be defined before the site investigations are started.

Furthermore, according to General Siting Study 95, SKB intends to present a site-specific safety assessment for a deep repository at the candidate site which is recommended by SKB for detailed characterisation. However, in order for comparisons to be made, KASAM believes that site-specific safety assessments must be carried out for both of the sites where the site investigations are to be carried out.

Basis for Decision-Making at the Local and Regional Level

At present, SKB has reached the stage where it has carried out feasibility studies at two municipalities, Storuman and Malå. A referendum was held out at Storuman municipality. Because of the results obtained, the municipality is no longer eligible for further study. SKB has now started feasibility studies in the municipalities of Nyköping and Östhammar. The municipality of Oskarshamn is also considering whether to participate in a feasibility study.

SKB states that feasibility studies will be carried out in 5 – 10 municipalities in order to obtain a basis for selecting sites for site investigations. KASAM finds that the process which is underway may provide an adequate basis for selecting sites for site investigation. KASAM also believes that it is important that SKB should continue to describe, in a national perspective, the Swedish bedrock on a general and regional scale. The description can gradually focus on those regions which are of interest and, thereby, become increasingly detailed. This work is important for two reasons:

- The feasibility studies which are carried out must be put in a context so that the sites which are selected are found to have good geological conditions, seen from a national perspective.
- It cannot be guaranteed that the feasibility studies which have now been initiated will lead to acceptable sites or a sufficiently large range of sites. Consequently, more data may be necessary in order to identify additional suitable areas.

In KASAM's view, this part of SKB's work should be viewed as a natural continuation of the general siting studies which have been carried out and as a complement to the feasibility studies.

Environmental Impact Statements (EIS)

The Government considers the EIS to be very important and, in its decision of May 18, 1995, it emphasises the importance of establishing a transparent process, or Environmental Impact Assessment (EIA) for the preparation of the Environmental Impact Statement (EIS) at an early stage. The county administrative boards will be given the responsibility for co-ordinating the EIA. However, no further guidance is provided on how "a transparent process" is to be established. Furthermore, SKB's RD&D Programme 95 does not provide any guidance on this subject.

As before, KASAM would like to emphasise the importance of the EIA. It should be possible for the National

Co-ordinator for Nuclear Waste Disposal to provide assistance during the EIA. At the same time, KASAM would like to emphasise that it is the actual functions of the EIA that are important and not the formal framework.

Issues relating to the final disposal system are highly complex. Therefore, various parties involved will find it necessary to develop their knowledge of the subject before making the necessary decisions. KASAM proposes that a systematic programme should be established to achieve this. This should be included in the tasks of the National Co-ordinator for Nuclear Waste Disposal. This can be achieved in parallel to the investigations carried out by SKB which will result in an EIS and licence application. With such an arrangement, the parties involved can investigate, in various forms, individual issues which are considered to be of particular importance and difficult. This should contribute to an efficient development of competence within, e.g. the municipalities concerned. It must be emphasised that the aim is to make preparations for the decision-making process, not to initiate it by making evaluations, e.g. as regards to whether final disposal at a particular site will be safe.

Engineered Barriers

SKB has changed its canister design in three stages without providing a detailed motivation for the changes. The fuel canister is a prototype design and, at the same time, one of the most important barriers against the dispersion of radioactivity. Even if many aspects of the properties of the canister have now been studied by SKB, KASAM recommends that SKB should use the entire length of time at its disposal for development and further study and not commit itself exclusively to one alternative.

In KASAM's view, it is important that SKB should build confidence in the ultimately selected canister design as being a result of a process of maturity which has been carried sufficiently far. Thus, SKB should describe, in detail, the development process for the canister, the advantages and disadvantages of the alternatives studied and the reasons why SKB believes that the final design is sufficiently mature to be a basis for decision-making on the construction of the encapsulation plant and the manufacturing of canisters.

KASAM considers SKB's plans to establish a pilot facility for testing the sealing of the canisters and control of full-size canisters to be of value. This facility will prove valuable in focusing the verifying research on the specific properties of manufactured canisters. It will also enable Swedish researchers, to a greater extent than at present, to participate in research concerning manufacturing. This is important in order to develop the same high level of expertise with regard to the manufacturing of the canisters as there is with regard to the canister properties.

KASAM recommends that SKB should use the production capacity which must be developed by sub-contractors and the resources of the pilot facility to manufacture a relatively large number of sample canisters. This will be of great value in establishing the range of variations of the canister properties and in developing quality control methods. An extensive manufacturing of canisters on a pilot scale would allow for a more extensive trial deposition of inactive canisters in the Äspö Hard Rock Laboratory than SKB has so far intended.

Supporting R&D

General Comments

The nature of SKB's programme has successively changed from research to implementation in project form. At the same time there is a continuing need for supporting R&D. It is extremely important for credibility that SKB's research should be subjected to the same degree of peer review as that found at universities and institutes of technology. This cannot be achieved exclusively via SKB's normal international contacts through joint projects and in international organisations. SKB has compiled a large body of valuable knowledge in its reports. In order to improve the availability of such knowledge, SKB should also publish its research results in scientific publications to an increasing extent.

A critical stage of SKB's activities is when the research results are transferred to SKB's project work, especially when factors which may have a negative impact on safety are dismissed as insignificant. In this context, KASAM would like to mention the action of bacteria in promoting copper corrosion as an example of an area where greater knowledge is needed before SKB can dismiss microbial corrosion as insignificant.

Regardless of how much research is done, there will always be a degree of uncertainty. This is the case, for example, with regard to the hydrological description, where different models are possible. In KASAM's view, SKB must develop its approach to how such basic uncertainties should be handled.

Äspö Hard Rock Laboratory

KASAM recommends that SKB should expand the planned trial deposition of inactive canisters in the Äspö Hard Rock Laboratory. The methods and technology for the manufacturing and control of the engineered barriers as well as those for deposition must be verified. The integral performance of the canister and the buffer must be studied and analysed. So far, SKB has only been able to describe the planned repository by using drawings and calculation data. A considerably more extensive trial deposition than that planned by SKB, which involves four canisters, should contribute to the early detection of any deficiencies in methods and technology and should contribute to the increased confidence and insight of those outside the group of experts into SKB's final disposal work.

European Union

A comprehensive research programme (Nuclear Fission Safety) is underway within the EU. A significant portion of the programme consists of nuclear waste management research. The current programme covers the period from 1994 to 1998. After that time, a new research programme is expected to be launched.

As a member of the EU, Sweden contributes to the funding of this research. The results of the research will have an effect on SKB's programme. In SKB's RD&D Programme 95, no strategy has been developed for how EU's research will be optimally utilised from a Swedish perspective. Sweden has to now become actively involved in the determining the content and structure of the programme for the next four-year period. In KASAM's view, it is especially important that the EU's nuclear waste management programme should also provide scope for work concerning EIA and public participation.

KASAM's recommendations

In summary, KASAM recommends that SKB should:

- continuously describe, how it intends to apply the principles for radiation protection;
- as soon as possible, prepare an integrated safety analysis for the entire final disposal system;
- develop the safety analysis in order to improve transparency and comprehensibility and incorporate a systematic presentation of facts, best estimates and opinion;
- define its site selection factors and specify how they can be used at different stages of the siting work;
- carry out general siting studies on a regional scale to provide a clearer basis for selecting municipalities for feasibility studies;
- modify its time-schedule so that the applications for a permit to site and construct an encapsulation plant and to carry out the detailed characterisation of a candidate site for the deep repository are submitted at the same time;
- increase peer review of its research and investigation work.

Furthermore, KASAM recommends that the Government:

- clarify, as soon as possible, the conditions under which the Government can override the municipal veto;
- emphasise the importance of SKB, the regulatory authorities and Swedish researchers on the whole, actively participating in the EU's work on nuclear waste management and of Sweden participating to ensure that issues relating to democracy and public participation as well as environmental impact assessments are taken into account within such work.

Finally, KASAM recommends that the recently appointed National Co-ordinator for Nuclear Waste Disposal should organise a systematic programme for the preparation of those participating in the EIA prior to the evaluation of licence applications and EIS.

11.4 THE SWEDISH GOVERNMENT's DECISION ON SKB's RD&D-PROGRAMME 95

(unofficial translation)

Decision of the Government

On the basis of 12 § of the Act (1983:4) on Nuclear Activities. the Swedish Government has decided the following.

The Swedish Nuclear Fuel and Waste Management Co (SKB) shall, in its further research and investigatory work, carry out a system analysis of the entire final disposal system (encapsulation plant. transportation system and repository). This system analysis shall allow for an overall, integrated safety assessment of the entire final disposal system including how principles for safety and radiation protection are to be applied, in practice, in the safety assessment work. Furthermore. the system analysis shall include an account of the alternative solutions to the KBS-3 method described by SKB in previous research programmes or which have been described in international studies. Different variations on the KBS-3 method should also be described. In addition. the consequences which would arise if the planned repository should not be constructed (zero alternative) as well as ongoing international work on transmutation must be presented.

Furthermore, SKB shall carry out a safety assessment of the long-term safety of the repository. Moreover, SKB shall describe how the need for further supporting research and development work is linked to the safety assessment and how the research results are to be transferred to the final disposal project as well as how basic uncertainties are to be treated in the further work.

No later than in connection with the presentation of the next research and development programme, SKB shall supplement "General Siting Study 95" by specifying, in greater depth than before, the factors which should determine the selection of a suitable site for a repository for spent nuclear fuel and long-lived radioactive waste.

The Issue

On September 30, 1995, the Swedish Nuclear Fuel and Waste Management Co (SKB) submitted to the Swedish Nuclear Power Inspectorate (SKI), a programme for research and development concerning the treatment and final disposal of nuclear waste in accordance with the stipulations of 12 § of the Act (1984:3) on Nuclear Activities. The programme is called the "Treatment and Final Disposal of Nuclear Waste. Programme for Encapsulation, Deep Geological Disposal and Research, Development and Demonstration" (RD&D Programme 95). Two reports were attached to the programme "Siting of a Deep Repository for Spent Nuclear Fuel" (General Siting Study 95) as well as "Template for Safety Reports with Descriptive Example" (SR 95). The programme was prepared by SKB on behalf of those who hold a license in accordance with the Act on Nuclear Activities to own and operate a nuclear reactor .

RD&D Programme 95 is the fourth programme which was compiled in accordance with 12 § of the Act on Nuclear Activities. The Government made a decision on the first programme (R&D Programme 86) on November 26, 1987 (cf ME1087/87), the second programme (R&D Programme 89) on December 20, 1990 (cf M90/1165/6), the third programme (RD&D Programme 92) on December 16, 1993 (cf *M931252516)* as well as on the RD&D Programme 92 Supplement on May 18, 1995 (cf M931122815).

In § 26 of the Ordinance (1984:14) on Nuclear Activities, it is stipulated that SKI should submit, no later than on March 31, 1996, its own review statement on RD&D Programme 95. In a letter dated February 13, 1996, SKI requested an extension of the review period by two months. On March 28, 1996, the Government granted SKI permission to submit its review statement no later than by May 31, 1996.

On May 21, 1996, SKI submitted to the Government a statement on RD&D Programme 95. SKI attached a "Review Report" (English translation. SKI Report 96:57) to the statement as well as a "Summary and Conclusions" (English translation, SKI Report 96:56). SKI distributed RD&D Programme 95 to various reviewing bodies. The review statements are compiled in SKI Rapport 96:41 (no English translation available).

In accordance with the Government's decision of May 27, 1992, concerning the terms of reference of the Swedish National Council for Nuclear Waste – KASAM (Dir. 1992:72), KASAM presented its review of RD&D Programme 95 in a report entitled "Nuclear Waste. Disposal Technology and Site Selection – KASAM's Review of the Swedish Nuclear Fuel and Waste Management Co's (SKB's) RD&D Programme 95" (English translation, SOU 1996:101).

On June 27, 1996, SKB was given the opportunity to submit a statement on SKI's and KASAM's statements. On September 27, 1996, SKB submitted such a statement.

Reasons for the Government's Decision

Like SKI, the Government is of the opinion that RD&D Programme 95 fulfils the stipulations of 12 § of the Act on Nuclear Activities.

The Government has observed that SKB, in RD&D Programme 95, describes the facilities which according to SKB are necessary for a safe treatment and final disposal of spent nuclear fuel and nuclear waste. These facilities comprise a plant for the encapsulation of spent nuclear fuel and a deep repository for the final disposal of the fuel. The intention is to construct the deep repository in stages. The first stage is a demonstration stage with the possibility of retrieving the deposited spent fuel during this stage. According to SKB's plans, the ultimate decision on final disposal of the waste should not be made until after the demonstration deposition has been concluded and the results have been evaluated as well as not until after other alternatives have been considered.

As stated by the Government in its decision of December 16, 1993, on RD&D Programme 92, the Government would like to once again emphasize, in particular, that even if the KBS-3 method should be a reasonable choice of a demonstration deposition method, SKB should not make a commitment to any specific treatment and disposal method before an overall and in-depth assessment of the safety-related and radiation protection issues is presented. It is the opinion of the Government that SKB must compile and describe, in a more detailed manner the alternative solutions to the KBS-3 method which have been presented in previous research programmes. Different variations on the KBS-3 method should also be described. In particular, the consequences which would arise if the planned repository should not be constructed (zero alternative) should be presented in greater depth than it has been so far. The further work on transmutation should also be described.

In its decision of May 18, 1995, on the RD&D Programme 92 Supplement, the Government observed that the decisions made in accordance with Chapter 4 of the Act (1987:12) on the Management of Natural Resources etc. and 5 § of the Act on Nuclear Activities on the construction of the planned encapsulation plant may involve major commitments with regard further treatment and final disposal methods. In the opinion of the Government these decisions should not be made before an assessment of the final disposal system as a whole has been described and the planned final disposal method shown to be suitable. In accordance with the Government's opinion, an overall integrated assessment of the radiation protection and safety-related issues concerning the planned final disposal method as well as concerning other possible alternatives should be made available to the municipalities concerned.

The Government shares SKI's opinion that SKB, in SR 95, has presented a template which is an adequate and flexible framework for future safety reporting. However, the template should be further developed and presented in greater detail in the way described by SKI in its statement to the Government. SKB should also take into consideration the opinions put forward by KASAM in this context. An assessment of the repository's long-term safety should, in the opinion of the Government, be completed before an application is submitted to the authorities for a licence to construct the planned encapsulation plant, as well as before site investigations are initiated at two or more sites.

The Government shares SKI's opinion that SKB's research work is mainly of a high standard viewed from an international perspective. However, in the opinion of the Government, it is important that further research should, to an adequate extent, take into consideration the requirements which may be made on the future evaluations of the safety assessments by the competent authorities. Furthermore it is important that SKB should be able to clearly describe how it intends to treat basic uncertainties. Thus, in the opinion of the Government, SKB should, in its next research and development programme in particular describe how the need for further supporting research and development work is linked to the safety assessments and how the research results are to be transferred to the final disposal project as well as how basic uncertainties are to be treated in the further work.

On May 18, 1995, in the government decision on the RD&D Programme 92 Supplement, the Government stated that SKB should present an overall report of its general studies and site-specific feasibility studies, after their completion, in forthcoming research and development programmes. Results so far achieved as well as a specification of the additional knowledge necessary for further work must also be reported. SKB has presented an overall report of its general studies on a national scale in "General Siting Study 95".

In this context the Government would like to emphasize the importance of a well-defined and transparent site selection process. The Government shares the opinion expressed by SKI and KASAM concerning the information presented in "General Siting Study 95" and the view which has clearly emerged from the comments of the reviewing bodies on this report, that the ongoing site selection process must be clarified in several respects. In the Government's opinion in the light of the experience gained from the siting work, SKB's overall report on general studies, feasibility studies and any other background and comparative information which, after consultation with the government appointed National Co-ordinator for Nuclear Waste Disposal, SKB may wish to present, must be made available to the municipalities concerned before the site selection process can proceed to the stage of site investigations at no less than two sites. Furthermore, for the planned final disposal method, SKB should be able to specify criteria for the evaluation of candidate sites and specify which factors will determine whether a site will be excluded from further investigations.

Moreover, before site investigations at no less than two sites are initiated, SKB should consult with SKI and SSI on the premises which should apply in the investigation work.

Like the views expressed by SKI in its statement, the Government is of the opinion that SKB should supplement "General Siting Study 95" by defining, in greater depth than it has been so far, the factors which should determine the selection of a site which is suitable for a repository for spent nuclear fuel and long-lived radioactive waste. SKB should also describe the consequences of a coastal siting and those of an inland siting of the repository as well as the consequences of siting the repository in southern as well as in northern Sweden. SKB should take into account the other opinions that SKI and KASAM have expressed in their statements concerning "General Siting Study 95".

In the Government's decision of May 18, 1995, the Government stated that the applications for licenses in accordance with Chapter 4 of the Act on the Management of Natural Resources etc. and 5 § of the Act on Nuclear Activities to construct a repository for spent nuclear fuel and nuclear waste should contain information for comparative assessments which show that site-specific feasibility studies, in accordance with SKB's description, have been carried out at 5 – 10 sites in the country and that site-specific studies have been carried out at no less than two sites as well as the reasons for selecting these sites. The Government takes it for granted that SKB, in consultation with the municipalities concerned, will be given the opportunity to carry out site-specific feasibility studies in such a way that an adequate basis for decision is available prior to SKB's consultation with SKI and SSI regarding the site investigations. SKB should make strong efforts to ensure that the municipalities concerned are given as adequate information as possible before different decisions are made in the siting work.

12 TECHNICAL DEVELOPMENT ON CU-STEEL CANISTER FOR SPENT FUEL

12.1 DEVELOPMENT OF DESIGN

12.1.1 General

The work of designing the canister proceeds in step through the compilation of basic premises, requirements on function and properties, and criteria for sizing and design. This work is currently in progress and is scheduled to be finished by the end of 1997. A report of the status of the work will be presented here, but it should be understood that changes may occur before the final report is presented.

The basic premises for the design are that the spent fuel from the Swedish nuclear power programme must be encapsulated for disposal in a deep repository in such a way that the safety is maintained at a high level during:

- encapsulation,
- transport,
- disposal,
- final storage.

12.1.2 Requirements on function and properties

The requirements on the function and properties of the canister are that: all existing fuel types can be encapsulated, long-term safety can be achieved in the repository and all normal and abnormal operating cases during handling and encapsulation must be met with maintained safety.

Fuel types

The fuel properties of importance are presented in Table 12.1-1. The measurements given in the table limit the overall dimensions of the canister.

Table 12.1-1. Fuel properties and dimensions.

FUEL TYPE	BWR	PWR
Total length	4.398 m	4.243 m
Cross section area	140x140 mm^2	214x214 mm^2
No. of fuel pins	63 – 100	15x15 or 17x17
Enrichment % U-235)	max. 3.6% (with Gd 4,2%)	max. 4.2%
Burnup (max.)	60 MWd/kg U	60 MWd/kg U
Burnup (ave.)	38 MWd/kg U	38 MWd/kg U
Decay time (min.)	30 years	30 years
Residual power/bundle	100 – 150 W	300 – 450 W

Long-term safety

Release of radioactive substances from the deep repository is prevented by a multibarrier system consisting of the fuel itself, a corrosion resistant canister, the bentonite buffer and the host rock. The canister prohibits all dispersion of radioactivity as long as it remains intact. The other barriers can delay and reduce the dispersion to acceptable levels, should the canister be penetrated. The requirement on the canister is that it should completely isolate the waste and remain intact for at least 100.000 years. This requirement leads to requirements on:

- initial integrity,
- chemical durability in the environment expected in the repository,
- mechanical strength during the conditions expected in the repository.

In order to fulfil the requirement that the other barriers shall delay and reduce the dispersion to acceptable levels, should the canister begin to leak, additional requirements must be put on the canister. These requirements are:

- choice of a material that does not adversely affect the buffer and rock,
- limitation of heat and radiation dose in the near field,
- a design of the canister, which remains subcritical even if water enters the canister,
- limitation of the bottom pressure on the bentonite.

Industrial safety

The overall property requirements with respect to fabrication and handling are that the canister be designed to be able to be:

- fabricated and produced in serial production corresponding to 200 canisters per year with adherence to specified quality requirements,
- meet all normal and abnormal operating cases during handling and encapsulation without exposing the personnel and the facility to unacceptable radiation doses or release of radioactivity,

- transported to and disposed of in deep repository in a safe way,
- retrieved, if so is required, from the deep repository in a safe way.

12.1.3 Design criteria

Impermeability

The requirements on impermeability can be subdivided into initial impermeability, chemical durability and mechanical strength.

Initial impermeability

The canisters shall be impermeable to water when they leave the encapsulation plant. This means that the canisters must be fabricated, sealed and controlled with methods that guarantee that no more than 0.1% of the canisters can have defects that can allow water to penetrate after disposal.

Chemical durability

The canisters shall have a service life of 100 000 years in the repository. The fission products and americium dominate the toxicity of the fuel waste during the first thousand years. Thereafter, plutonium dominates for tens of thousands of years up to about 100 000 years at which time it has decayed with about an order of magnitude. After the decay of the shorter lived fission products, the radiotoxicity is dominated by americium and plutonium, both of which are elements with extremely low solubilities and mobility in both buffer and bedrock.

Mechanical strength

The design load for the canister is 15 MPa outer pressure. This pressure is composed of a hydrostatic pressure of 7 MPa, corresponding to disposal at maximum 700 m depth and 8 MPa bentonite swelling pressure, corresponding to a bentonite density of 2000 kg/m^3 or slightly higher. These loads are additive and isostatic. The design for these loads shall be made using customary safety margins.

The strength of the canister shall also be calculated to withstand increased hydrostatic pressure in conjunction with glaciation. At the maximum of a glaciation this might result in an additional hydrostatic pressure of 30 MPa and a total pressure of 45 MPa. This may only occur far in the future, when radiotoxicity has decreased considerably. It may then be considered as an offset condition, for which no additional safety margins are required.

Uneven pressure build-up and uneven loads can occur both during the saturation phase and after saturation. The load cases which are to be considered as design load cases are presently being defined through interaction with repository design.

Considering the dimensions of tunnels and deposition holes any possible rock creep will not be observed as an additional load for the canister. Lithostatic pressure is, thus, not a design basis case.

Interaction with other barriers

Materials interaction with buffer and rock

The canister material and its corrosion products must not in a substantial way affect the function of the buffer.

Dissolved substances from the canister material or its corrosion products must not chemically alter the buffer so that its swelling properties, hydraulic conductivity and diffusion resistance are substantially deteriorated.

Heat load on buffer and rock

The surface temperature of the canister must not exceed 90°C.

Increased temperature in conjunction with unfavourable water chemistry can in a negative way affect the chemical stability of the buffer. Substantial evaporation of water from the canister surface may lead to salt enrichment in the vicinity of the canister and, thereby, create a more aggressive water chemistry than otherwise anticipated.

Radiation dose to buffer and rock

The surface dose rate at the canister must not exceed 500 mGy/h.

The radiation shielding of the canister must be such that the radiation will not substantially change the water chemistry in the near field through radiolysis. The contribution from radiolysis products to the canister corrosion must negligible compared to the corrosion caused by residual oxygen in the repository.

Criticality

The canister must be designed so that the fuel remains subcritical, should water intrude into the canister.

The bottom pressure against the bentonite

The canister must be designed so that the bentonite layer under it is capable of supporting the canister for very long times.

12.1.4 Design

Choice of material

Chemical durability

The deep groundwaters in the granitic bedrock in Sweden are oxygen free and reducing from a depth of 100 to 200 m. The redox potential below these depths is -200 to – 300 mV on the hydrogen scale and the pH values are neutral or mildly alkaline. The chemical environment in the immediate vicinity of the canister is controlled by the bentonite pore water. Table 12.1-2 shows typical values for carbonate containing groundwater with low chloride content equilibrated with bentonite.

Table 12.1-2. Typical values in mmol/dm^3 for carbonate containing groundwater with low chloride content equilibrated with bentonite

Na^+	Ca^{2+}	Cl^-	SO_4^{2-}	HCO_3^-/CO_3^{2-}	pH
90	9.2	1.8	33	3	9.3

The chloride concentrations are not affected by the bentonite and will range from 0.15 mmol/dm^3 to 1.5 mol/dm^3, depending on the composition of the groundwater. With the exception of chlorides, the concentrations of other anions will, for a foreseeable future, be controlled by the interaction with bentonite.

Several candidate canister materials have been studied both in Sweden and abroad. These materials include immune, or partially immune, materials (copper), corroding materials (carbon steels) passive materials (titanium and stainless steels), and ceramic materials (alumina and titania).

For Swedish conditions, the corrosion evaluation gives copper the highest ranking. This is based on the thermodynamic stability of copper over a wide range of water compositions under reducing conditions and on the very low availability of species that can affect the thermodynamic stability of copper under these conditions. There are also both archaeological and natural material available, which can be used to validate models for long term copper corrosion. From the corrosion point of view, pure copper is to be preferred. However, considering other requirements than corrosion resistance, such as availability, weldability and mechanical properties, the copper material is specified as UNS C10100 (Cu-OFE) with the additional requirements P: 40-60 ppm, H: < 0.6 ppm, S: < 6 ppm and the grain size 180 – 360 μm (as measured with ASTM E 112-95).

Mechanical strength

The mechanical strength of the canister is to a large extent dependent on the design. From fabrication reasons as well as from strength reasons, it is assumed that the canister will be designed as a cylinder with flat bottom and lid.

The material in a cylindrical shell design must, in order to bear the external load with desired safety factor, have a yield strength of at least 250 MPa. From corrosion point of view, copper has been judged to be the best material. The chosen copper quality, however, has a yield strength of only about 50 MPa. Although the strength of copper can be increased by alloying, this has been considered unsuitable for corrosion reasons. Thus, if copper is chosen, the canister must be designed so that the necessary strength if given to the canister by a supporting inner structure. An insert of a material of sufficient strength has been judged to be the most advantageous alternative.

Effect on other barriers in the deep repository

The effects of heat, radiation, criticality as well as the consequences of bottom pressure on the bentonite are primarily related to the canister design and not to the choice of canister materials. The canister materials can affect the performance of the buffer chemically through dissolution of the material itself or of its corrosion products. It can also affect the buffer mechanically through the build-up of a thick corrosion product layer. For passive materials, ceramics and copper, the general corrosion is very small and these effects can be considered to be negligible. Corroding materials, such as iron and steel, have the potential for detrimental interaction with the buffer chemically, through release of dissolved iron and mechanically, through a faster build-up of lower density corrosion products.

Canister design

Principle design

The canister is designed so that it has an outer copper canister as a corrosion barrier. In order to give the necessary mechanical strength this copper canister is internally supported by a load bearing insert. A basis for the design has been that it shall be possible to encapsulate 12 BWR fuel elements or 4 PWR fuel elements in each canister.

Copper canister

The wall thickness of the copper canister shall be at least 30 mm.

The copper corrosion under expected reducing repository conditions has very conservatively been estimated to less than 5 mm, but will most probably not exceed a few tenths of a millimetre. It has been proposed that during glaciations, oxygen containing water may episodically penetrate down to the repository horizon. Assuming an increase in water flow rate of a factor of 10 during a total of 5 000 years and an oxygen content of 45 mg/dm^3, the result would be a total corrosion depth of 2 mm. Assuming a pitting factor of 5, as in KBS-3, the deepest corrosion pit would be 10 mm. This is assuming no transport resistance in the bentonite. SKB 91 has shown that the rate of transport of corrodant to the canister will be limited to an equivalent flow of 10 litre per year, regardless of the water flow rate in the repository. This corresponds to a decrease by a factor of ten compared to the case described above.

These estimates have been conservative or very conservative and contain, therefore, large implicit safety factors. With that background, a safety factor of 3 is reasonable, i.e. a wall thickness of 30 mm would be sufficient from corrosion point of view. For the more realistic corrosion estimates, this corresponds to a safety factor of 30.

It should be pointed out that although 30 mm is a suitable minimum wall thickness from corrosion standpoint, other factors such as fabrication, handling and sealing etc also have an influence on the choice of wall thickness.

Load bearing insert

The load bearing component of the canister shall be designed for an outer overpressure of 15 MPa, with a safety factor of 3 against plastic collapse. The choice of safety factor is based on building design standards. Low alloyed steel has a yield strength of 200 – 330 MPa at 100°C. Necessary strength for these materials cannot be obtained by a self supporting cylinder with reasonable wall without limiting the canister dimensions.

As an alternative to a self supporting insert, a cast insert can be used as shown in Figure 12.1-1. The insert is fabricated so that each fuel element can be placed in an individual channel. This insert can be fabricated either in ductile cast iron or cast steel. The design of this insert is controlled more by casting consideration than by design against overpressure. The wall thickness and the thickness of the inner dividing walls are determined by the casting properties of the alloy used.

A control calculation for a canister of this design shows a collapse pressure of over 80 MPa, i.e. a safety factor of over 5 and a safety margin of more than 30 MPa for a possible increased load during glaciation.

Design of canister with insert

In addition to corrosion and mechanical strength, also radiation shielding, residual heat generation and criticality must be considered.

In order to achieve a maximum dose rate of 500 mGy/h, a minimum total (copper canister plus insert) wall thickness of about 100 mm is required.

The requirement of a maximum surface temperature of 90°C limits the maximum allowed residual heat generated by the fuel in the canister. At present burnups and intermediate storage times indicate a limit of 12 BWR and 4 PWR fuel elements, respectively.

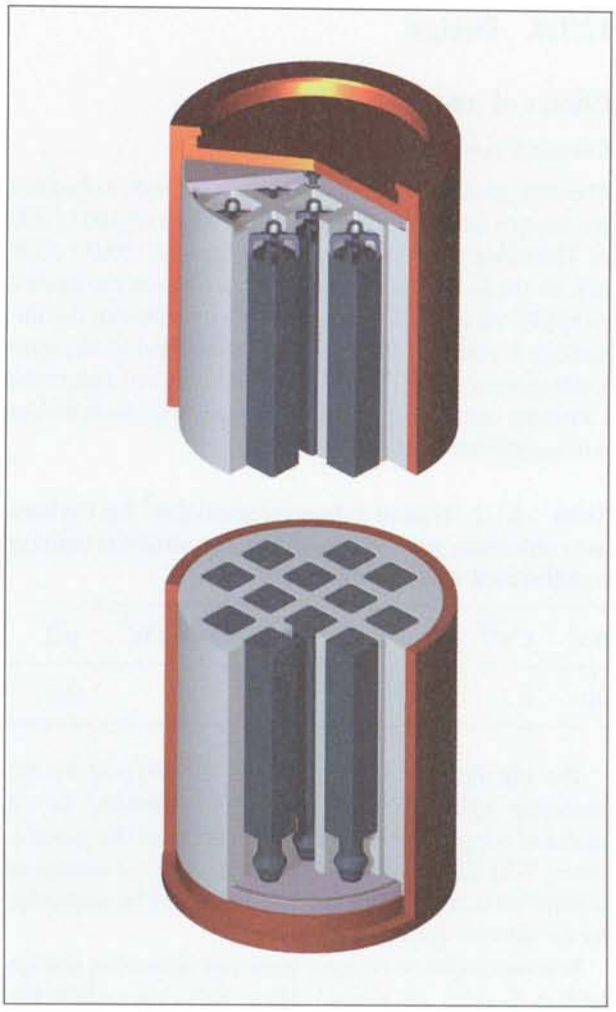
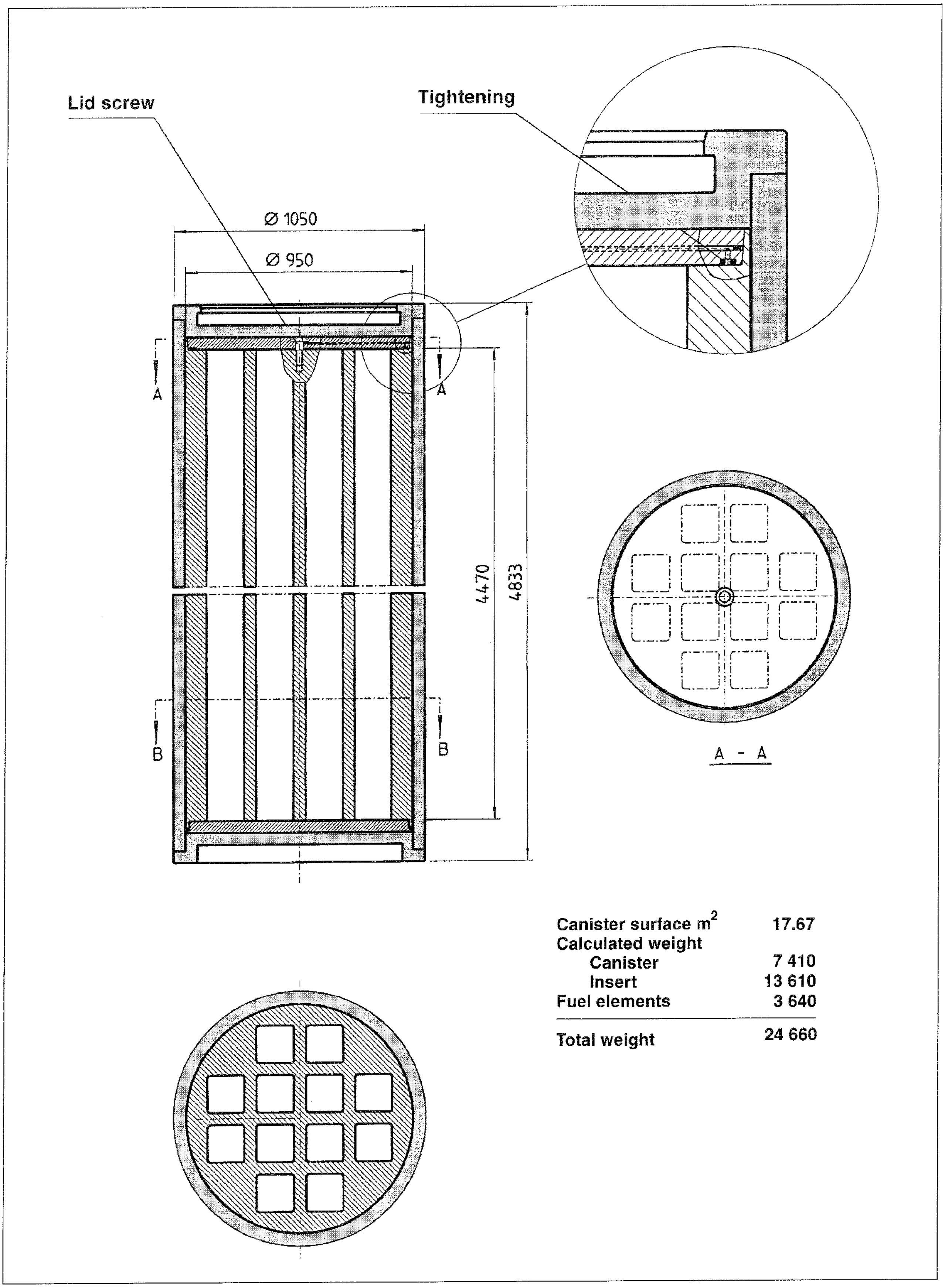


Figure 12.1-1. *Design of copper canister with cast insert.*

For a self supporting cylinder, there is a risk for criticality, should water penetrate into the insert. This can be counteracted by filling the void space in the canister with a suitable material. This complicates the handling in the encapsulation plant. With a cast insert, the internal dividing walls will suppress criticality even if the insert is water filled, if some credit is taken for burnup.

12.1.5 Reference canister

The design of the reference canister is shown in Figure 12.1-2. It consists of a cast insert of steel or ductile cast iron with 50 mm separations between the fuel channels and a minimum 50 mm materials coverage to the periphery. The function of the insert is to bear to external load in the deep repository, but also to guarantee that no leakage from the insert occurs during the sealing of the copper canister. This is achieved by welding a bottom to the insert (steel insert) or having an cast integral bottom (ductile cast iron) and a lid sealed with either a central bolt or eight bolts along the periphery of the insert. The interior is



Canister surface m^2	17.67
Calculated weight	
Canister	7 410
Insert	13 610
Fuel elements	3 640
Total weight	24 660

Figure 12.1-2. Drawing showing the design and principal dimensions of the reference canister, BWR version.

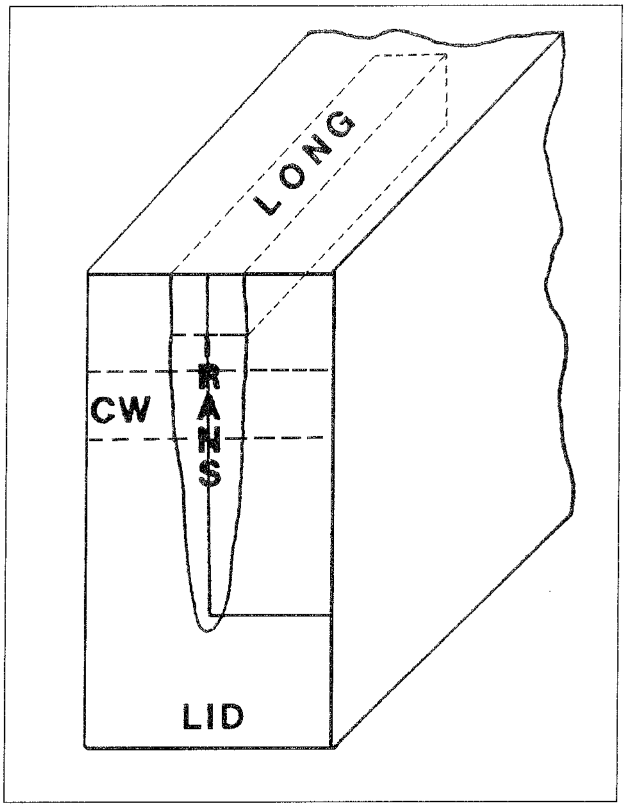


Figure 12.2-1. Sketch, which simulates the weld between the lid and the cylinder of the canister, showing how cross-weld (CW), longitudinal and transverse specimens were removed for creep testing.

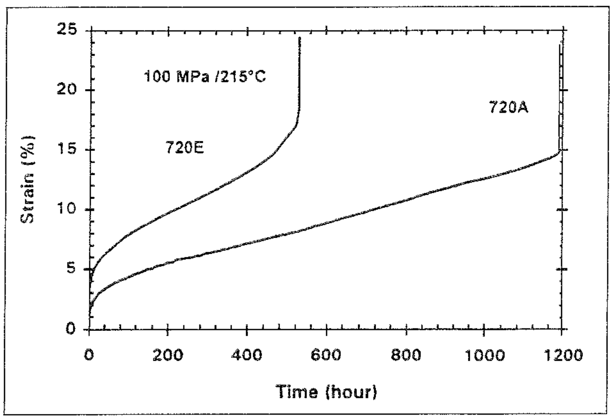


Figure 12.2-2. Creep curves for longitudinal all-weld specimens. 720A is from the top of the weld near the surface, 720E from the bottom.

sealed with a elastomer gasket. The insert is placed inside a 50 mm thick copper canister. The radial gap between the two components is 2 mm. This ensures a maximum strain of 4% in the copper when it is deformed towards the insert.

The design of the copper lid and bottom has been chosen so that it will be possible to inspect the weld both with ultrasonic testing and with digital radiography.

12.2 MATERIALS STUDIES AND TESTS

12.2.1 General

As was discussed in section 12.1, the materials chosen for the reference canister design are oxygen free copper (UNS C10100) with an addition of 40 to 60 ppm of phosphorous. For the load bearing insert, two materials have been selected as reference materials, ductile cast iron (Swedish Standard 140717-00) and cast steel (Swedish Standard 141306-02). During 1996, mechanical testing has continued on the copper material. This work includes creep testing, development of constitutive models for creep in copper and investigations of grain growth in cold worked copper. For the insert materials, work has been performed on establishing the corrosion rate under anaerobic conditions. A programme for additional corrosion testing of copper will be implemented in 1997 and will continue also during the first half of 1998.

12.2.2 Results of materials investigations

Creep testing

Uniaxial creep tests have been performed at 215 – 250°C on welded joints in the reference material /12.2-1/. Weld specimens were tested in the longitudinal and transverse directions and cross-weld specimens containing both parent metal and weld metal were also tested. The specimens were taken from a lid weld performed at TWI (Cambridge, UK), during 1995. The lid had a diameter of 880 mm, corresponding to the dimensions of a canister with self supported steel cylinder. Radiography and metallography showed the weld to be essentially sound and representative for lid welds. It contained about 8 minor flaws, randomly distributed over the welded zone. Figure 12.2-1 shows how the creep specimens were cut from the welds.

The grain size was found to 300 µm the lid and 280 µm in the copper cylinder. The welds are very coarse grained with columnar grains several millimetre long. All tests failed at elongations greater than 15%, most of them at 20 – 30%, apart from one cross-weld specimen, which ruptured at 11%. All cross-weld specimens failed in the weld metal. The longest test time was 6 000 hours. Figure 12.2-2 shows creep curves for longitudinal all-weld specimens. Specimen 720A is from the top of the weld near the surface, while 720E is from the bottom of the weld.

In conclusion, it can be said that the welded structures showed good ductility. Furthermore, the phosphorous has been found not to be lost during the welding process neither at reduced pressure nor in high vacuum.

Recrystallization in copper

The recrystallization of copper with about 50 ppm of phosphorous has been studied with the object of determining if any significant grain growth would occur during long term storage of a nuclear waste canister made of such copper /12.2-2/. Grain boundary migration rates were determined in specimens recrystallized at 300 and 350°C. These data in combination with literature data on migration rates were used to estimate possible migration rates at the storage temperature of the waste canister, 80 – 100°C. Since the measured migration rates were subject to solute drag, the extrapolation to low temperatures could not a priori be expected to be straightforward. Therefore, the extrapolation was made with the help of a specific formulation of the solute drag theory and a few additional assumptions. This led to predictions of migration rates at low temperatures, which turned out to be roughly the same as those obtained from a straightforward extrapolation of the experimental data. Calculations were also made of the drag at migration rates typical of pure copper. These calculations indicated that the drag would be larger than the driving force and it is, thus, impossible for the boundary to break loose from the solute.

The conclusions were that based on the recrystallization data from 300 and 350°C and literature data, the grain growth in the waste canister will be negligible. However, even if no large scale grain growth occurs it is possible that quite small amounts of grain growth could lead to substantial phosphorous segregation to the grain boundaries. This point needs further consideration.

Constitutive models for copper

In order to calculate residual stresses due to welding, experimentally determined material parameters are needed. Therefore, an optimization method for determination of constitutive parameters using experimental results has been developed and applied. The material model is based on previously performed creep and creep relaxation measurement on copper and the parameters are accurately adjusted to the experimental data. The model has also been implemented into a finite element code /12.2-3 /.

Anaerobic corrosion of iron and steel

In earlier work performed by AEA Technology, the anaerobic corrosion rate of mild steel has been determined in pH range 8.0 to 10.5 for different water compositions /12.2-4, -5/. The corrosion rates were generally found to be in the range 0.1 to 1 μm/year at room temperature and 50°C. Supplementary experimental work was performed during 1996 to establish whether replacing the mild stel insert with a ductile cast iron insert would have an effect on the hydrogen evolution rate /12.2-6/. The temperature range was also extended for both materials to 85°C.

The results showed that the corrosion rate at 50°C in saline groundwater at pH 7-8 after 5 000 hours were for ductile cast iron (0.1 μm/year) approximately five times lower than that of carbon steel (0.5 μm/year), see Figure 12.2-3. In water with higher pH (10.5), the corrosion rates

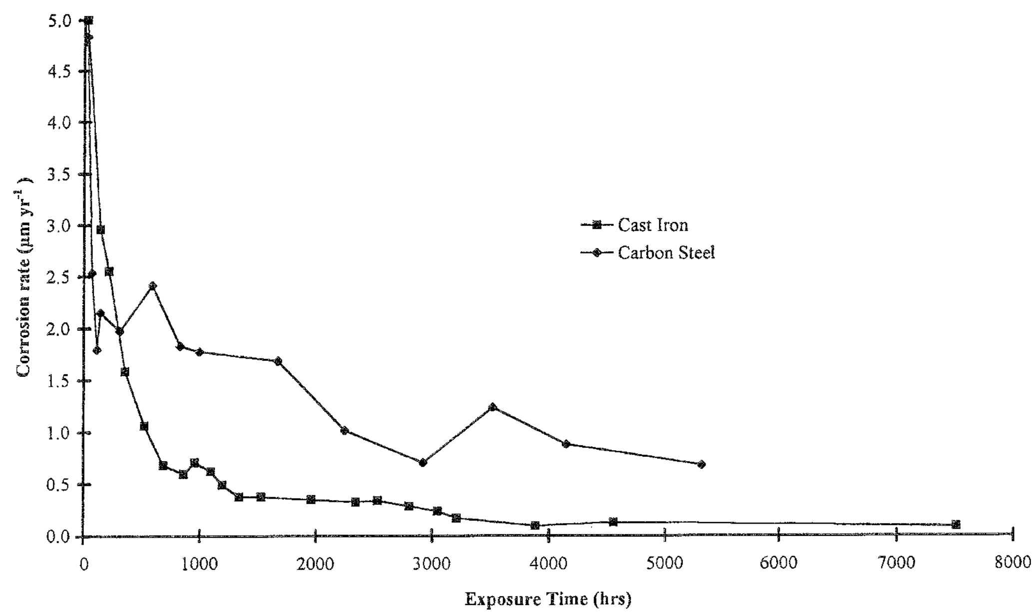


Figure 12.2-3. Anaerobic corrosion rates for cast iron and carbon steel in artificial saline groundwater, at 50°C.

were about a factor of ten lower. At 85°C the corrosion rates were initially higher than at 50°C, but after 5 000 hours to 0.1 μm/year. In contrast to the behaviour at 50°C, ductile cast iron and mild steel behaved almost identically.

12.3 CANISTER FABRICATION

12.3.1 General

Production methods

The reference canister can be produced using different methods. The following methods have been identified for the **copper cylinder**:

- Rolling of copper plates to half cylinders to be joined by electron beam welding.
- Tube extrusion.
- Hot Isostatic Pressing, HIP.
- Electrodeposition.
- Spray forming.

During 1995 – 96 continued trial fabrication and evaluation has been carried out with the first two methods, which are commercially available in full size /12.3-1/. The trials have 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, new tests during 1996 have confirmed that it is possible to satisfy the requirements on the microstructure to an essential degree. In the case of extrusion, the tests during 1995 showed promising results 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.

HIP and electrodeposition are regarded as possible competitive future fabrication methods but a longer evaluation and development period will be needed for this application. Regarding HIP and electrodeposition no equipment exist with which full scale canisters can be produced. Lab-scale experimental production has to be made to study material properties and the evaluation must also include estimations of the costs of full size equipment and the costs of serial production.

The earlier design of the **canister insert** based on a steel tube has been replaced by a cast insert with channels for the fuel elements. Trial castings in steel and nodular iron have been made with promising results. Bronze is being studied as an alternative material but no test castings have so far been made. Nodular iron has advantages concerning fabrication and price. Cast iron has a very good castability. If steel or bronze is chosen the price will be considerably higher due to poorer material yield and higher material prices.

Quality assurance

In parallel with the test production during 1996 a Quality Manual according to the demands in ISO 9001 and IAEA 50-C-QA has been developed. The target is to have a quality assurance of the complete canister manufacturing process up to and including the delivery to the encapsulation plant.

12.3.2 Results of trial fabrication 1995 – 1996

Two complete full-scale canisters with inserts fabricated from steel tubes were finished during 1995, Figure 12.3-1. After this the design of the canister has changed. A cast insert with channels for the fuel elements has resulted in a diameter increase. The outer diameter of the copper canister has increased from 891 mm to 1050 mm.

During 1996 the manufacturing of 6 complete canisters with the new dimension with cast inserts have started. The inserts are being made both in cast steel and in nodular iron. All 6 copper tubes are produced by roll forming of copper plates to half cylinders followed by longitudinal welding, Figures 12.3-2, 12.3-3 and 12.3-4. The lids and bottoms in copper are performed by pressforging followed by finish machining.

Figure 12.3-1. *The first full-scale canister produced during fabrication trials.*

Figure 12.3-2. Rolling of cylinder halves for the copper tube.

Copper canister

For the first two complete canisters (outer diameter 891 mm) two different methods were tested for fabrication of the copper tubes. These methods were roll forming of tube halves with subsequent joining of the two halves by electron beam welding and extrusion of seamless tubes. In both cases, a bottom was welded on by means of electron beam welding.

Rolling mills with a capacity to produce sufficiently large plates for **roll forming** exist at a number of locations in Europe and USA. In order to obtain the specified grain size in the range of 180–360 µm during rolling a reduction in the order of 5 is necessary. 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 however shown that it is possible to obtain the desired grain size with less reduction as well.

Roll forming is a conventional method for fabrication of tube from plate, Figure 12.3-2. The most important thing is to fabricate the tube halves with such precision that they can be finished machined without excessive machining allowance and can be joined together by electron beam welding, Figures 12.3-3 and 12.3-4. The trials have shown that tube halves can be formed in this way with straightness and roundness within the tolerances over the length of the plate.

Before these trials **extrusion of copper tubes** of the size in question had never been done before. One complication was that the necessary ingot dimension was not available as a standard. For this reason a special continuos casting mould had to be built.

Also with extrusion 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 pressure would be required. It turned out that only 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 quite coarse. 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.

It is likely that a lower extrusion temperature will result in a reduced grain size. Furthermore, it is possible to cool the material during extrusion. To find and understand the optimum parameters for the extrusion of copper a computer model is now being developed in combination with laboratory-scale experiments.

Figure 12.3-3. Dimensional check of half-cylinder before final machining.

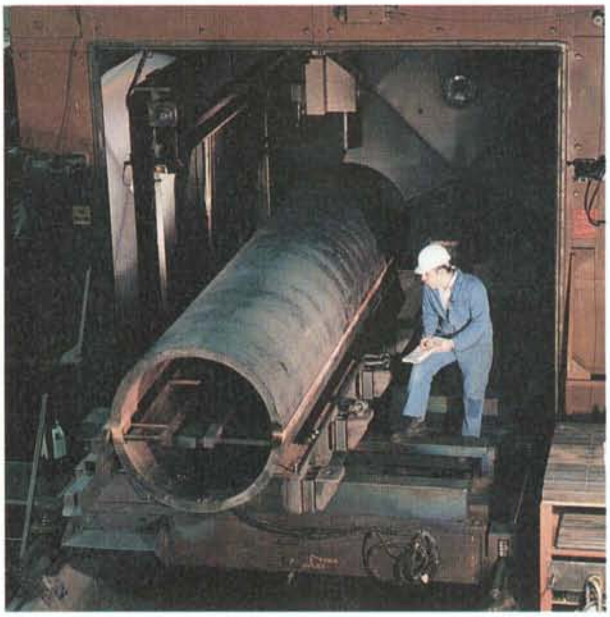


Figure 12.3-4. Longitudinal electron beam welding of half-cylinders.

Figure 12.3-5. Preparation before casting of insert. The cassette of steel-profiles is being placed in the mould.

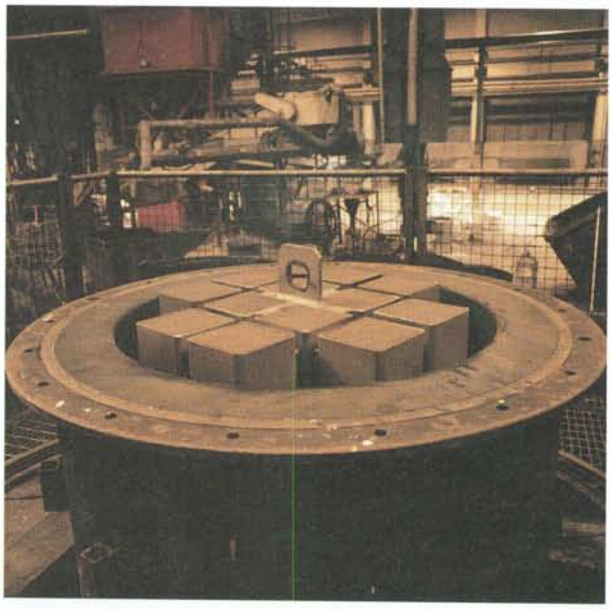


Figure 12.3-6. Steel-cassette in the mould. The channels are filled with foundry sand in order to withstand the pressure from the liquid metal during casting.

Canister insert

The first two canisters finished during 1995 had inserts produced as steel tubes. These were fabricated both by pressing/rolling and by extrusion. Conventional methods were used in both cases and the trials were carried out without problems.

In the current design the insert is produced as a casting with channels for the fuel elements. Figure 12.3-8. Test castings during 1995 and 1996 have been made in steel (one half length and one full sized) and in nodular iron (one full size with integrated bottom) and for both materials with promising results. The steel grade which has been used is SS 1206-02 and the nodular iron is SS 0717. Figures 12.3-5 to 12.3-8 show different steps in the foundry production of inserts.

Nodular iron has some advantages compared with cast steel. The castability of iron is such that it is likely that it will be possible to cast the insert with integrated bottom. The first test casting made in this way showed a positive result. With the cast steel insert it will be necessary to weld a bottom plate to the insert. One other difference is the need of heat treatment cycles after casting. Nodular iron can be used in the as-cast condition while cast steel will need heat treatment to achieve the right hardness and structure. Cast steel will also show a poorer material yield. In total inserts both in cast steel and nodular cast iron are probably technically feasible but inserts in steel will be quite more expensive than inserts in nodular iron.

The first full-size test castings of inserts in steel and nodular iron will be evaluated during 1997 and followed by a number of further test castings.

The possibilities to cast the inserts using a bronze alloy has been studied. So far it has not been possible to find a foundry where a casting of this size can be made. Of course the cost of a bronze insert will be considerably higher than the cost of an insert in steel or iron.

12.3.3 Studies of other methods

Hot isostatic pressing, electrodeposition and spray forming are also being studied as alternative fabrication methods for the copper canister. None of these methods is at present developed to a stage where it is likely that full sized copper canisters can be produced with the specified material properties. Regarding HIP and electrodeposition no equipment exist with which full scale canisters can be produced.

These techniques are regarded as possible competitive future fabrication methods but a longer evaluation and development period will be needed for this application. Lab scale experimental production has to be made to study material properties and the evaluation must also include estimations of the costs of full size equipment and the costs of serial production.

Figure 12.3-7. *Casting of an insert.*

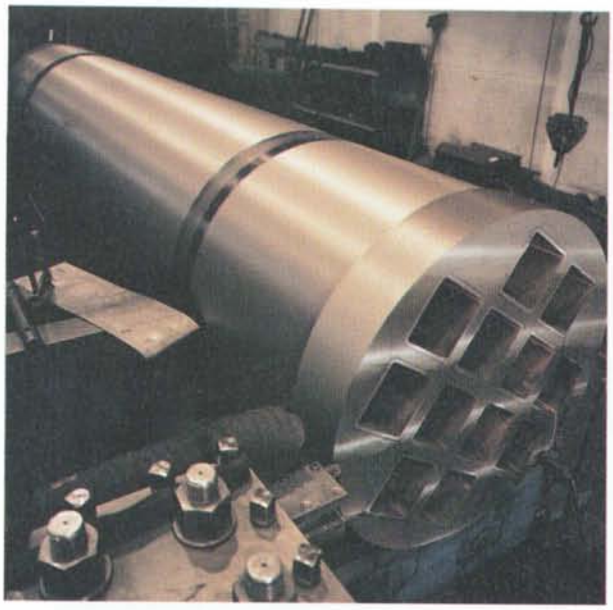


Figure 12.3-8. Machining of insert.

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 /12.3-2/. No practical tests have been conducted by SKB since 1982 – 83. The investigations indicate however 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.

In order to obtain good mechanical properties with HIP, surface oxides on the powder particles must be removed. This can be done with hydrogen gas at about 350°C. Water formation by the reduction reaction may however occur. The practical problems concerning this are likely to be quite serious. An other problem is that no equipment does exist with which full scale canisters can be produced.

Electrodeposition

In electrodeposition, copper is precipitated directly on the insert by means of electrolysis. Promising results were obtained in initial tests at the model level. Copper was deposited on several miniature steel canisters.

The structure of electrodeposited copper can be controlled to a great extent by the process parameters and subsequent heat treatment. Initial creep tests showed a poor creep ductility compared with the reference material. The creep tests were, however, conducted on material whose structure was not representative of the material which is foreseen for the canisters.

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.

Further evaluation and development of this method over a long period will be needed first in laboratory scale and later, if the results are continuously promising, in full scale. It has however to be observed that no equipment for full scale trials exist today.

Spray forming

Spray forming is a powdermetallurgical method with which molten metal is directly transferred to the surface of a solid body. Nitrogen gas is blown through the melt in such a way that small droplets are formed and quickly cools down. The solidifying droplets hits the recipient and a solid body with high density is formed. The fast solidification results in a fine grained equiaxial structure with no segregations. The nitrogen gas gives a protection against oxidation which is a common problem in many powder metallurgical applications. On the other hand nitrogen gas can be encapsulated in the solidified metal creating porosities if the alloy does not have elements that form nitrides.

A preliminary study was made by SKB during 1996 to investigate the possibilities with sprayforming for fabrication of copper cylinders. One available commercial application of spray forming is the so called Osprey Process. This is used in industrial applications and at least one equipment exists in which a full scale canister could be fabricated.

The copper specified for the SKB canisters has however no solubility for nitrogen nor does it have any nitride-forming element. The experiences also show that pure copper can not be made by spray forming without porosities. If, however, the copper is being alloyed with a strong nitride-forming element like Zr, a material free from porosities can be made.

Like HIP and electrodeposition a long period of lab-scale experiments including studies of the resulting material properties will be necessary before a full scale copper cylinder can be produced. No such studies or experiments are presently conducted by SKB.

12.3.4 Quality assurance

In parallel with the test production of copper cylinders and inserts during 1996 the first version of a Quality Manual for Canister Manufacturing has been developed. This Quality Manual is made according to the demands in ISO 9001 and IAEA 50-C-QA and relevant demands in ISO 14001 shall be followed. It is organized under SKB's general policies for management, quality, nuclear safety, purchasing and environmental issues. The target is to have a quality assurance of the complete canister manufacturing process up to and including the delivery to the encapsulation plant.

In summary the content of the Quality Manual for Canister Manufacturing is as follows:

- Policies and targets.
- Organisation, responsibilities and authorities.
- Description of quality system. Quality audits.
- Development and management of canister manufacturing. Qualification of processes. Identification and traceability.
- Management of support processes (purchasing, external quality audits etc).
- Flowcharts of material and product.
- Document control.
- Handling of nonconforming products.
- Corrective and preventive actions. Neverending improvements.
- Training and competence.
- References. Glossary etc.

Of course the Quality Manual and the total Quality System will be subject to continuos development and improvement. To the Quality System for Canister Manufacturing belong also Procedures, Technical Specifications for Material and Testing etc.

12.4 SEALING METHOD

12.4.1 General

The copper canister must be sealed with high requirements on reliability and integrity of weld, as well as inspectability. To fulfil this, a method for electron beam welding at reduced pressure (typically 1 kPa) has been developed employing a electron beam gun originally developed for non-vacuum electron beam welding within a EUREKA project 1986 – 1992. After the project had been concluded, the equipment was applied to non-vacuum welding of copper. However, this technique only allowed successful welding of copper sections up to about 30 mm thick. A development programme for welding of copper under reduced pressure was, therefore, conducted during 1992 and 1993 and successfully concluded. Electron beam welds with sufficient penetration and without root porosity could be achieved.

During 1994 and 1995, this technique was applied to welding full size lids and later also to welding lids and bottom onto full size canisters. These tests were successful, although no perfect, flawless, lid welds were achieved. It was decided that further development work to perfect the lid and bottom welds was to take place in SKB's canister laboratory, which is currently being established in Oskarshamn and which is scheduled to be operative during the first half of 1998.

The welding of the lid with a horizontal electron beam require the use of a fronting bar in order to support the top bead during welding. This fronting bar was made as an integral part of outside of the lid. Successful welding of lids and bottoms require narrow tolerances between the

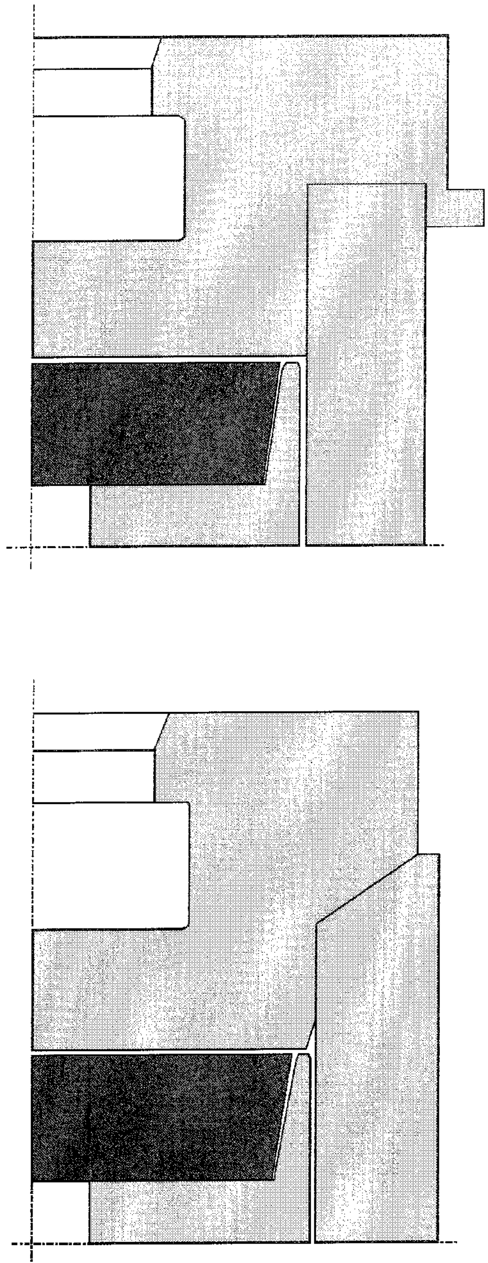


Figure 12.4-1. *Sketches showing the two alternative lid designs for horizontal and inclined beam welding.*

copper tube and the lid/bottom. Although this could be managed in the laboratory environment it caused some concern for the handling in the encapsulation plant, where all operations will have to be performed by remote control. It was, therefore, considered valuable if an alternative lid design could be found, where the positioning of the lid would be simpler and, also, where the fronting bar could be realised in a simpler way. This could be achieved by welding with an inclined electron beam. Figure 12-4-1 shows the two alternative lid designs.

12.4.2 Results from trial welds 1996

During 1996 the electron beam welding equipment has been modified to allow welding with beam that could be inclined from horizontal to 45°. Preliminary tests on smaller copper blocks at an angle of about 37°. The aim of these tests was to establish the feasibility of inclined beam welding. Previous tests at 90° had shown that if the beam angle is too steep, the weld pool becomes unstable and molten copper tends to be expelled during welding resulting in unacceptable defects in the weld. After proper welding parameter had been developed, the tests were promising and a series of three full scale lid welds were performed. These tests confirmed the conclusions from the smaller test blocks, but also showed that the problem with unstable weld pool could not be completely eliminated. Although most parts of the welds had a near perfect appearance, occasional expulsions of copper could still occur. Further development work, if judged necessary, with inclined beam welding will be performed at the Canister Laboratory.

12.4.3 Non-destructive testing

The application of array technique for ultrasonic inspection of the copper canisters has been investigated at Uppsala University /12.4-1 /. The objective of the project is to develop ultrasonic array techniques for assessing the integrity of the circumferential electron beam weld between the lid and the wall of the copper canisters. Due to the radiation emitted by the nuclear waste, the inspection must be entirely automated. In order to achieve the required extremely high level of confidence in the results, the ultrasonic system should be capable of inspecting 100% of the circumferential weld zone in thick section copper. Copper is regarded as a difficult material to inspect ultrasonically, mainly due to its coarse grain structure and high attenuation.

Taking into account the above issues an ultrasonic array system consisting of a linear ultrasonic array, made of piezoelectric composite material, and a computer controlled multi-channel electronic system was proposed as a solution to the problem. The array system chosen for this application enables electronic focusing and rapid electronic scanning eliminating the use of a complicated mechanical scanner. It is also capable of reducing the influence of noise by using a combination of multi-element transducer array (spatial diversity), focused sound field and split spectrum processing. Furthermore, it is characterised by inherent flexibility enabling 100% inspection of the weld zone even if the weld is not parallel to the surface.

The research activity addressed two issues: the development of software for calculation of sound fields generated by a linear array in copper immersed in water, and a laboratory verification of the array technique using an ultrasonic system specially designed for this application.

The results obtained to date are reported in /12.4-1/ for test blocks containing artificial defects (drilled holes) as well as natural defects. The results show that the system efficiency is sufficient to detect 1.5 mm diameter flat bottomed holes located in the weld zone in a copper block at the depth of 60 mm from the surface.

The report also presents a numerical method and a software package developed for modelling elastic fields in solids immersed in water and inspected using a linear array. The method is based on angular spectrum approach (ASA). The computational intensity of the ASA application has been optimised by a proper selection of parameters. The method has been implemented using a Pentium PC and is capable of calculating transient elastic fields in an immersed solid.

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

13.1 PLANNING FOR SITE INVESTIGATIONS

13.1.1 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 instruments and methods, including development, refinement and investment, etc,
- preparation of techniques and routines for data management,
- development of a site evaluation programme.

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

13.1.2 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 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. The geoscientific investigations 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 prioritised 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 investigations are used in the site evaluation process, which is a collective term for the interaction with performance and safety assessments, design work and "environment and society" aspects for the evaluation of the feasibility of a deep repository at the investigated site, see Figure 13-1 and further discussion in section 13.1.5.

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.

The development of a detailed site investigation programme can be illustrated as in Figure 13-2. The requested information shall be collected and determined with use of methods from a "tool-box". The programme is also affected by a number of technical and administrative prerequisites and limitations. During 1996 efforts was made to gather and organise the requested information, for the different purposes, in a report. All type of geoscientific information (parameters) which in one way or another will be used during the site investigations are described in the report. The systematic description also discusses in what aspect the parameter are of importance with respect to the safety functions (isolation, retention and recipient function), design and layout, environmental issues and general geoscientific understanding. The parameters are grouped

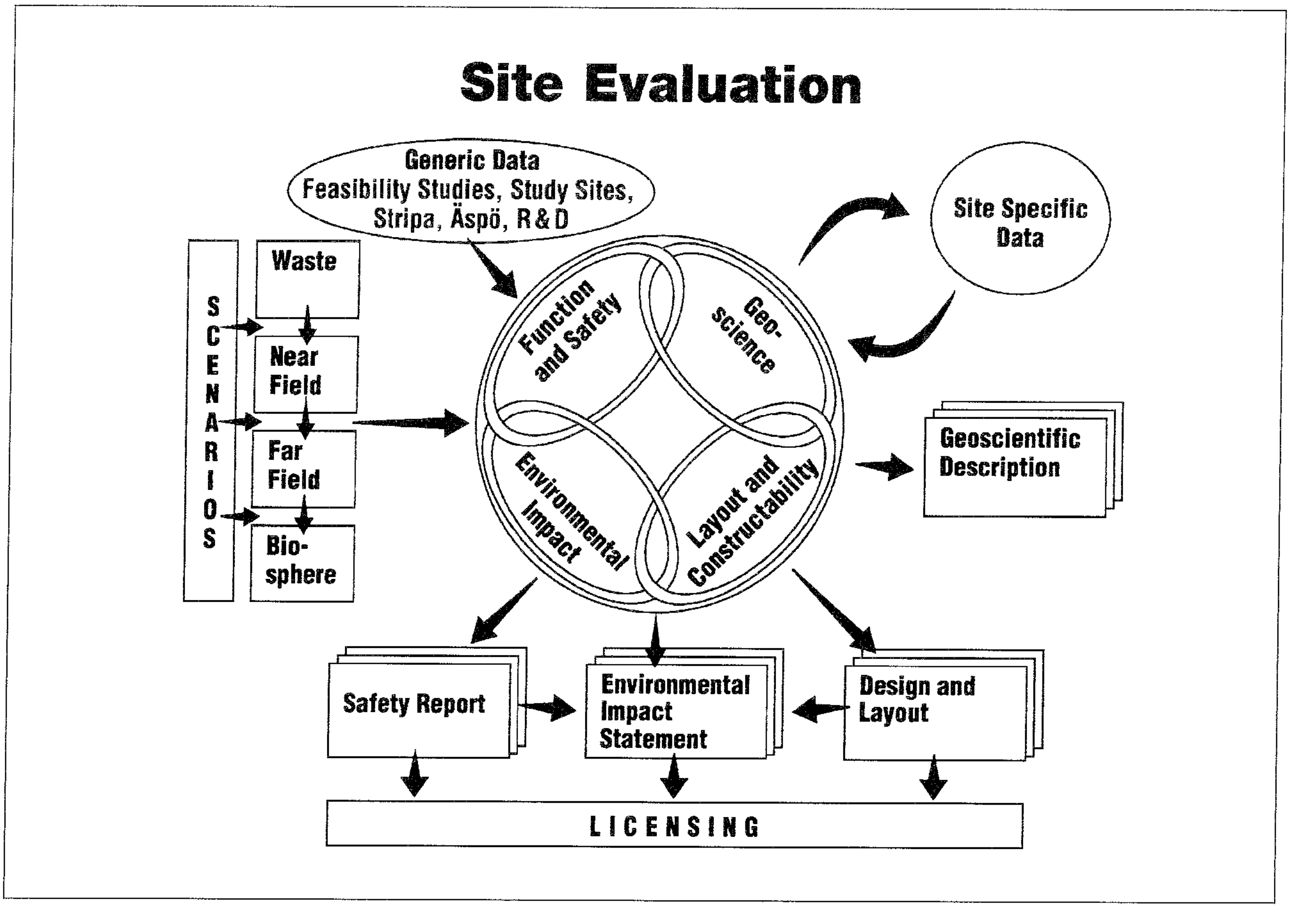


Figure 13-1. Site investigation and evaluation activities and their interaction.

according to the subject areas geology, rock mechanics, thermal properties, hydrogeology, chemistry and transport characteristics. The report is in Swedish /13-1/.

13.1.3 Instruments and methods

In general, choice of methods and instruments for rock characterisation during site investigations will predominantly depend on type of parameter to be investigated but also to some degree on site conditions, accuracy, etc. As illustrated in Figure 13-2 SKB has an existing "tool-box" of methods and instruments available, developed and used during earlier site investigations, not less during the Äspö Hard Rock Laboratory. However, the toolbox can not be kept static but has frequently to be revised due to several reasons like, new parameters to be determined, new technique available, methods/routines/instruments modified or developed for better accuracy or efficiency, etc. Work carried out during 1996 in this field will be discussed in this section.

Surface geophysical studies

During the autumn 1996 a surface geophysical study was started aiming at testing potential geophysical methods to confirm preliminary feasibility of a site or to compare two or more sites in the initial stages of a site investigation, before drilling. Conditions of interest in an initial site investigation is to get an overall idea of the homogeneity of the rock volume and to detect eventual subhorisontal fracture zones of major importance, see Figure 13-3 step 1 and 2 respectively. The methods tested in the field were semi-regional resistivity, vertical electrical sounding and reflection seismics.

The first two methods were run at two areas of which one was a relative young granite pluton showing magnetic homogeneity while the other was older granitic rocks surrounding the pluton. Preliminary results show very homogeneous and higher resistivity values in the pluton compared with the second area. In spring 1997 transient electromagnetic sounding will also be performed before an evaluation of the methods will be made.

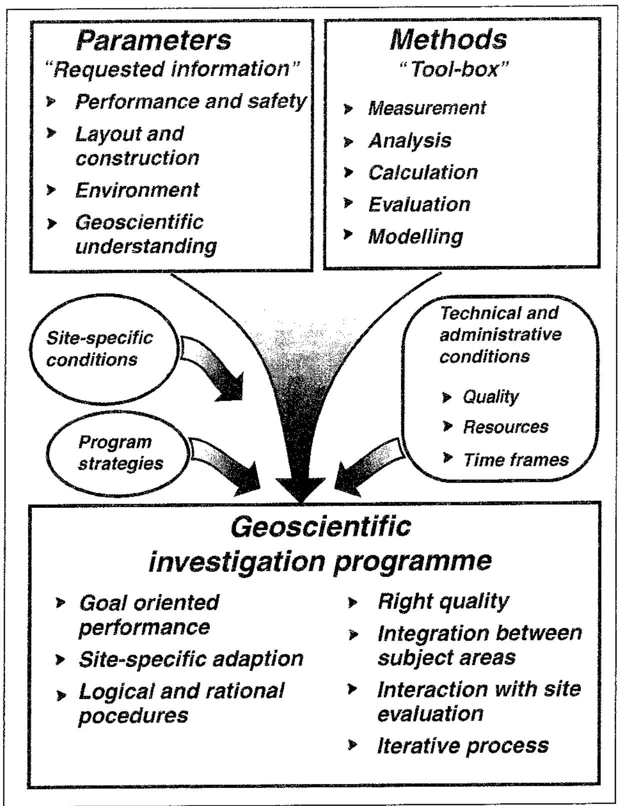


Figure 13-2. *The development of a geoscientific site investigation programme.*

Reflection seismics were conducted at a third site with a probable subhorisontal fracture zone, with unknown strike and dip. Two perpendicular seismic profiles, each of them 1 kilometre long, were run with 5 or 10 m between shot points and geophone separations. An existing cored borehole is situated in the cross. Preliminary seismograms shows very nice data and clear reflectors can be seen. The evaluation is still in progress and the results will be reported in 1997.

Geological Borehole Documentation

The borehole-TV system BIPS 1500, initially discussed in the Annual Report for 1995, has been extensively used also during 1995, for several SKB project as well as in Finland and Switzerland. Minor modifications in measurement and analysis software have been made.

Based on the good experiences with the BIPS system SKB has started the development of Geological Borehole Documentation (GBD) system. The aim is to integrate the use of BIPS logging and core logging to an efficient tool for geological documentation of a borehole. The old core logging system will be replaced and the new will be based on the BIPS image data file into which additional geological data from the core inspection will be registered. In a second stage a complot system will be stream-line adopted for presentation of GBD results.

Depth calibration technique

In the Annual Report for 1995 the copper ring method for length/depth calibration of measurements in boreholes was discussed. 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 1996, the first full scale field test of the copper ring method was carried out in an old 750 m deep and 56 mm diameter wide borehole. With use of a drill machine the diamond impregnated copper-rings were quite easily reamed-out into the borehole wall at certain depths (determined from drill pipe length). Nine rings were set at depths from 20 m down to 700 m. The copper-ring positions were then located with sensors. However, six of the rings were quite soon destroyed when other tools were run in the hole. The reason was probably an over-sized borehole diameter, which was not known due to bad documentation from the drilling of the old borehole. The method will therefore be slightly modified before field test will be continued, now also in a 76 mm borehole.

Water sampling during drilling

Water sampling during drilling was used in the Äspö Hard Rock Laboratory, pre-investigation phase to collect "first strike" water. The sampling were carried out in combination with simple hydraulic tests. The procedure were normally made every 100 m during interruptions in the drilling. The aim was to collect undisturbed water but the results were questionable, mainly due to simple equipment used. However, the method itself was evaluated as promising if the equipment could be improved.

A major drawback with the method and equipment used was that the water sample had to be pumped out from the borehole through the drill pipe. Especially from low permeable sections the water sample was not representative. Therefore a down-hole water sampler were mounted to the packer. The existing drill pipe operated switch valve were still used but in addition to operating the packer inflation now also two hydraulic valves surrounding the down-hole sampler were operated (opened/closed) in the same procedure. Also a memory pressure logger was mounted in order to record the section pressure and the packer pressure. This is to check the equipment function during the sampling period and to be used for preliminary determination of hydraulic parameters.

The new equipment was first used in the Palmottu project (EU natural analogue project in Finland), however, without real success. The fine grained drill debris

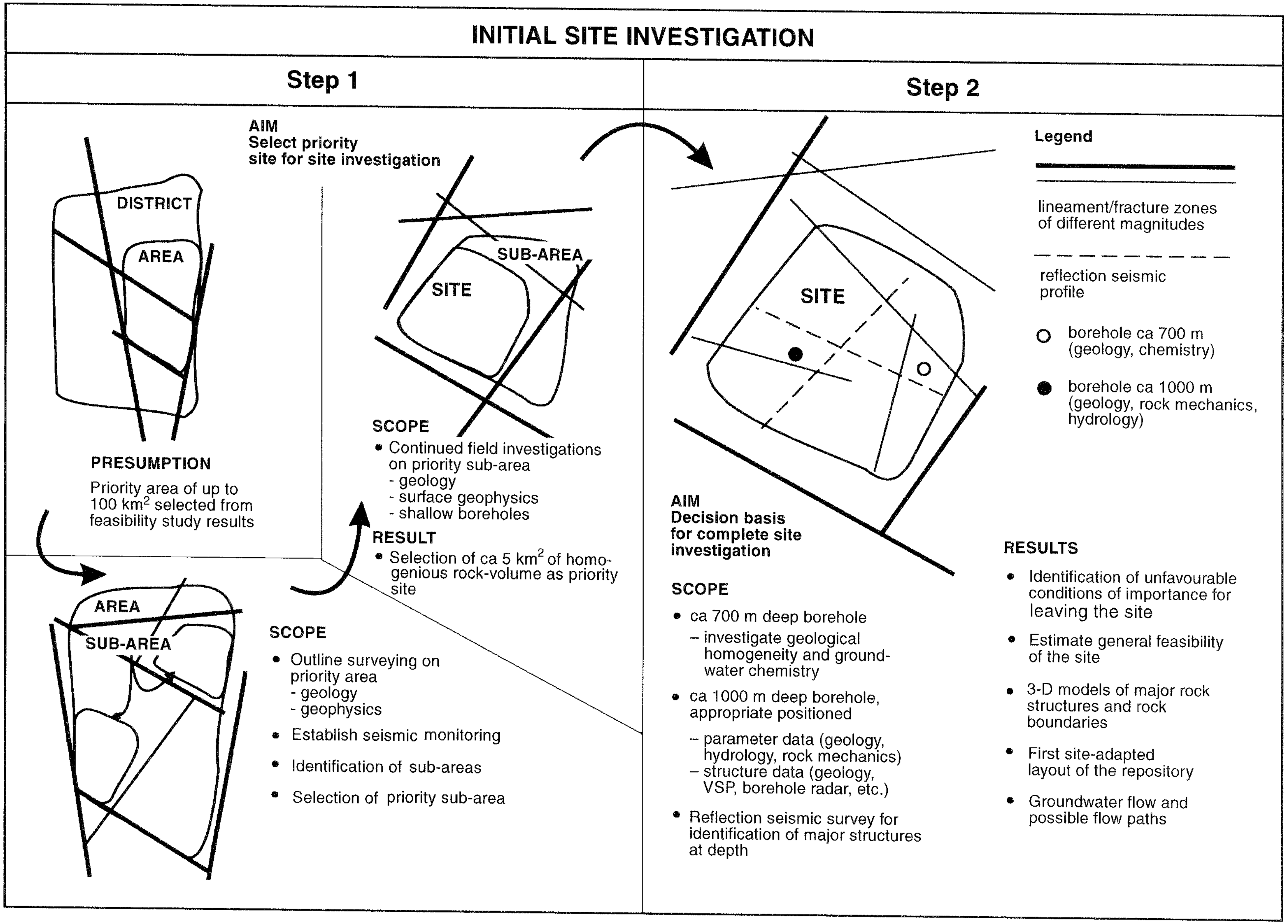


Figure 13-3. *Illustration of goals, activities and type of results during an initial site investigation (to be followed by complete site investigation).*

were clogging the valves. A second test (the equipment now with fine meshed filters) was made after the depth calibration project in the 56 mm borehole. After some initial problem the prototype now functioned well and samples were collected from 500 m and 600 m depth.

Others

A new dilution probe system for in-situ groundwater flow measurement for use in ≥56 mm and 1000 m deep holes were subject to field tests during 1996. The tests were interrupted due to electronic failure and could then not be continued because the depth calibration tests were conducted in the same hole. The field test will therefore be restarted in 1997.

Regarding hydraulic testing SKB uses the two system Pipe String System and the Umbilical Hose System. A second Pipe String Equipment is under manufacturing. A similar Umbilical Hose System are also used with the Mobile Field Laboratories. One of those umbilicals were stuck in a borehole during 1996, and was destroyed during the fishing operation. The umbilical therefore must be replaced.

Technical documentation of instruments constructed by SKB are currently being subject to improvements. 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.

13.1.4 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 currently under preparation and used when field measurements are being conducted.

One prerequisite for correct data management is the existence of a central database with efficient handling procedures with regard to data storage as well as data retrieval. The new database tool SICADA developed at Äspö will fulfil these requirements.

As a tool for rock modelling and visualisation of structures, rock type bodies, etc in the investigated rock volume, the Rock Visualisation System (RVS) also developed at Äspö will be used.

13.1.5 Site evaluation

The main goals for the site evaluation programme are as follows:

- The site evaluation should for every investigated site evaluate whether the fundamental safety demands and other siting criteria are fulfilled, and see to it that the repository in the best way will be adopted to the local environmental and rock conditions.

- The site evaluation should compare qualified sites, first of all with regard to long-term performance and radiological safety but also with regard to the other siting criteria.

The site investigations and site evaluations will be conducted as an iterative process with close interactions between geoscientific investigations, performance assessment and design work, see Figure 13-1. An environmental impact assessment is supposed to provide an overall picture of the planned deep repository.

A programme for the site evaluation will be developed and present procedures and routines for the interaction between the involved, above mentioned activities. The programme development will start during 1997 and will be closely related to a further definition and discussion of siting factors and criterias for feasibility and safety.

13.2 TECHNICAL STUDIES CONCERNING THE CONSTRUCTION OF A DEEP REPOSITORY SYSTEM

13.2.1 Repository for other long-lived waste than spent fuel

The repository area for long-lived LLW and ILW, here marked as SFL 3-5, is planned to be situated about one kilometer away from the repository area for spent nuclear fuel and at a sufficient depth below the ground surface. Transport tunnels will connect the two repository areas. An improved design of the area has been developed during 1996 being a basis for the presently conducted analysis Safety Report SR 97, see Figure 13-4 /13-2/. The waste inventory is described in /13-3/.

SFL 3 is mainly intended for long-lived waste from Studsvik. It will also receive operational waste from CLAB and the encapsulation plant. SFL 4 is intended for decommissioning waste from CLAB and the encapsulation plant, the storage canisters from CLAB and the transport casks and transport containers. SFL 5 will re-

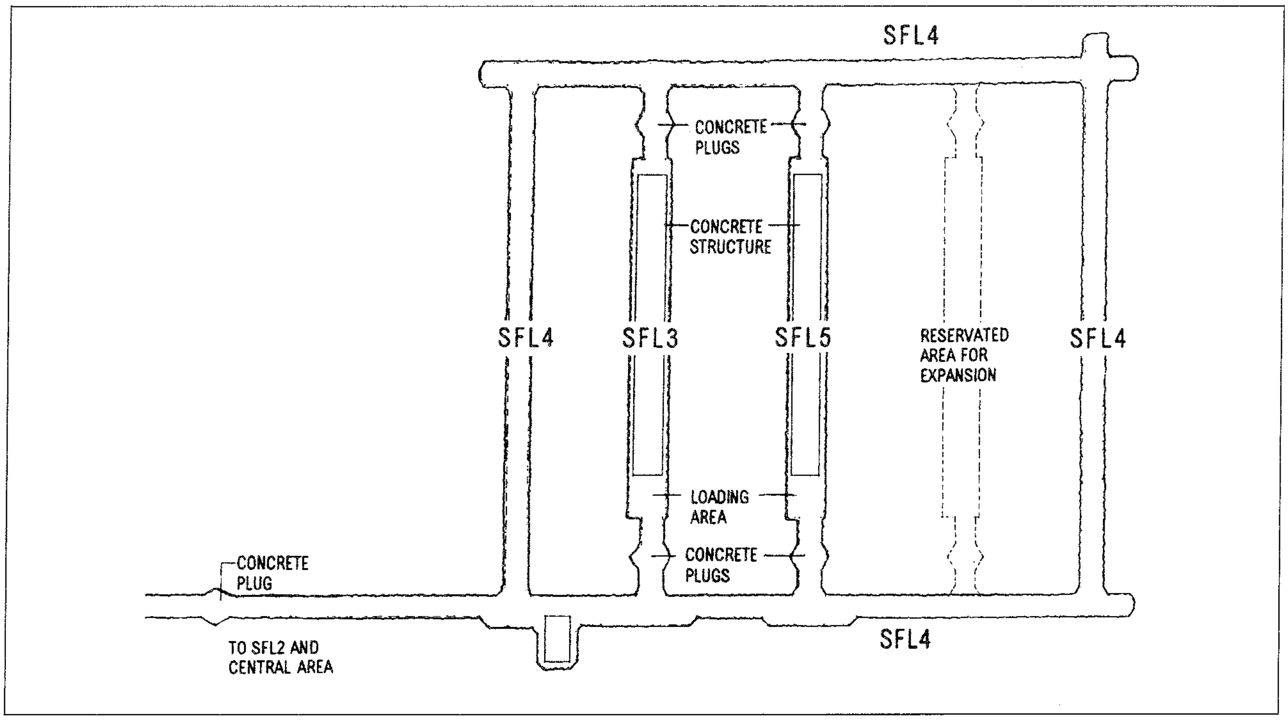


Figure 13-4. *Overview of SFL 3-5*

SFL 3: Repository for operational waste,
SFL 4: Repository for decommissioning waste,
SFL 5: Repository for core components and internal parts from nuclear reactors.

ceive core components and internal parts from the nuclear power reactors.

SFL 3 and 5 are composed of two identical rock caverns with a width of fifteen meters, a height of nineteen meters and a length of approximately 140 meters. The caverns will be furnished with concrete structures which separate the caverns into a number of compartments. In these compartments the waste packages will be emplaced by an overhead crane, whereafter the voids in the compartments, if needed, will be backfilled with porous concrete and the compartments then covered with a concrete lid. The outside of the concrete structure will be backfilled with crushed rock. The entrance tunnels to the rock caverns will finally be plugged with concrete structures.

The tunnel system surrounding SFL 3 and 5 will be used for SFL 4. The required length of tunnels is 700 to 900 meters, depending on tunnel size and waste volumes. The containers and waste packages will be emplaced on a concrete floor. Backfilling will be made with crushed rock.

13.2.2 Analysis of global thermo-mechanical effects from a repository

A study on the global thermo-mechanical effects caused by a KBS-3 type repository has been in progress since 1995. 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 /13-4, -5, -6/. The result suggests that with respect to global thermo-mechanics it would probably be possible to store nuclear waste with even an initial heat load of 10 W/m^2 without causing significant large-scale failure of the rock mass. The 10 W/m^2 is much more than can be tolerated by the criterion of not exceeding a temperature of 90°C in the bentonite.

In 1996 the sensitivity of the results in the 3DEC model in the phase 2 analysis was investigated concerning variations in rock material properties and boundary conditions. The effect of stress boundaries, a lower Young's modulus to represent rock mass deformability, rock mass with zero tensile strength, and the zero-cohesive discontinuity strength were investigated. The results from the study show that the largest influences resulting from changing model parameters were found at the ground surface. The two most sensitive parameters involved in the study were the Young's modulus and the tensile strength of the rock mass.

In the previous calculations a high Young's modulus – 60 GPa – was chosen in order to obtain conservative results. Reduced Young's modulus gave a more favourable stress condition, both at the ground surface and at the repository horizon, which results in less yield of rock at the ground surface and less fracture slip and displacements in discontinuities. For some discontinuities, when the Äspö HRL discontinuity system was applied in the model, the slip diminished completely. The zero tensile strength of the rock mass resulted in the largest effects at the ground surface. The tensile yield reached down to a depth of approximately 140 meters below the ground surface at 1 000 years after disposal (for Äspö HRL discontinuity data). Discontinuity cohesion and stress boundaries had the least influence on overall behaviour.

The new results do not contradict the conclusions from the earlier studies.

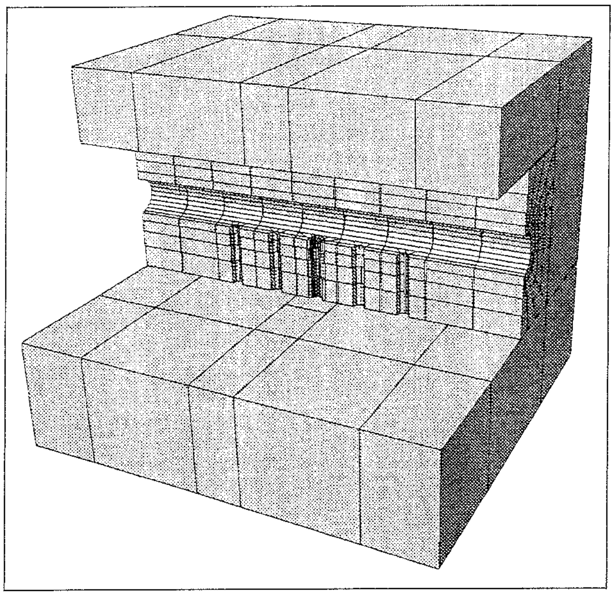


Figure 13-5. Continuum model of TBM tunnel with five deposition holes. Outer blocks are hidden to display the interior.

13.2.3 Thermo-mechanical nearfield rock analysis

General

The fraction of intended emplacement hole positions along a deposition tunnel, that can actually be used for deposition of canisters, is one central issue during different stages of repository site selection, design and construction. Some criteria for rejecting positions deal with long-time performance aspects, e.g. expected post-closure groundwater flow in fractures that intersect the deposition hole and connect to the tunnel or to nearby major fracture zones. Other criteria have to do with construction aspects, e.g. flow into the open deposition hole during construction and emplacement, or the mechanical stability of the deposition hole walls. Acceptance or rejection of individual deposition hole positions thus involve the expected consequences of a number of processes: excavation of tunnel and emplacement hole, interaction with the swelling bentonite buffer, heating and subsequent cooling. Additional, hypothetical processes include effects of glaciation and seismic events. The consequences of these processes will depend on the arrangement of discontinuities in the nearfield, on the mechanical properties of the nearfield and farfield rock, and on the initial stress field. The mechanical aspects of the acceptance/rejection procedure are being addressed in a numerical study in which the TBM tunnel in the Äspö HRL has been selected to represent a deposition tunnel in the central part of a hypothetical KBS-3 type repository. The numerical tool being used for the study is the three-dimensional distinct element code 3DEC, which is developed for analyzing the behavior of rock containing multiple, intersecting discontinuites and has a thermal logic specifically oriented towards solving problems associated with nuclear waste storage. The study is divided in three phases:

1. Continuum models.
2. Generic discontinuum models with a small number of fractures in geometrical arrangements expected to be relevant to acceptance/rejection criteria.
3. Discontinuum models based on discrete fracture network models, generated statistically from data contained in the Äspö HRL database.

The first phase has been completed and reported /13-7/ while phases 2 and 3 are in progress. The results obtained in phase 1 are going to be used as reference in subsequent phases and have been used for verifying the thermal and thermo-elastic parts of the calculations by comparison with analytical solutions. Large continuum models have been analyzed in order to determine temperature-induced displacements at the boundary of the local detailed model. These displacements have been used for prescribing boundary conditions that account for how the volume of the detailed model changes over time. The local detailed continuum model analyzed in phase 1 is shown in Figure13-5.

Comparison with analytical solutions

Figure 13-6 shows three points within the 3DEC model, selected for comparison of 3DEC temperatures with analytically determined temperatures /13-8/. The results are found to agree within less than 0.5°C in all positions, which is exemplified in Figure 13-7 for one of the points. Figure 13-8 shows horizontal displacements derived from a large global 3DEC model and used for specifying

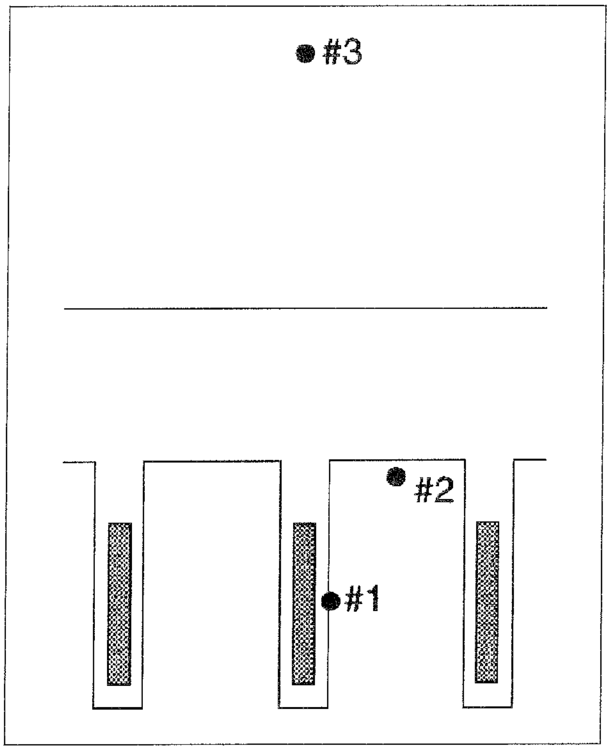


Figure 13-6. Points selected for comparison with analytical results.

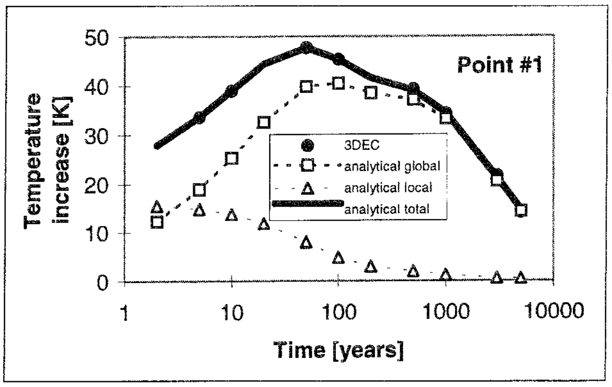


Figure 13-7. Analytically and numerically determined temperature at the wall of the emplacement hole. Analytical solutions are presented for global, local and the sum of them (total) while the 3DEC solution is given only for the total temperature increase.

boundary conditions for the local detailed model. The 3DEC displacements are compared with corresponding results obtained from analytical solutions /13-9/.

Stresses in the tunnel/borehole intersection.

The continuum analysis shows that the maximum principal stress in the tunnel/borehole intersection after excava-

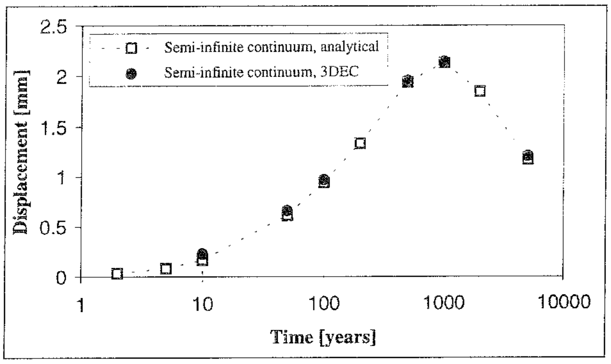


Figure 13-8. Comparison of analytically and numerically calculated thermo-elastic horizontal displacements of the boundaries of the local continuum model.

tion of tunnel and boreholes amounts to about 85 MPa, given the stress field orientation and magnitude prevailing at the depth of the TBM tunnel in Äspö HRL /13-10/.The maximum stress is, however, sensitive to the stress field orientation and would be 160 MPa, had the major horizontal principal stress been normal to the tunnel axis. If the major stress were parallel with the axis, the maximum stress in the intersection would be only 60 MPa. For a 7 W/m^2 initial heat production and a canister spacing of 6 meters by 25 meters, the thermal pulse creates an additional stress of 40 MPa in the intersection. This value applies if the modelled region is integrated in a semi-infinite isotropic homogeneous and linearly elastic continuum with the same properties as those of the modelled region, i.e. without taking scale effects or the influence of discontinuities outside the volume occupied by the repository into account.

13.2.4 Grouting

To prevent groundwater flow into 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 excavating the ramp in the Äspö HRL, it was noted that existing grouting methods are not enough, especially not in transmissive discontinuities having a high groundwater 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 tightened. In 1996 a revision of the compilation of the present state of knowledge and development needs was made /13-11/. The objective of the compilation was 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 grouting and stabilising extensive, water-bearing discontinuities having full piezometric pressure down to the repository depth,

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

Three types of discontinuities, that might need grouting, were 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 these types have an influence on the stability of ramp, shafts and tunnels. If they occur,

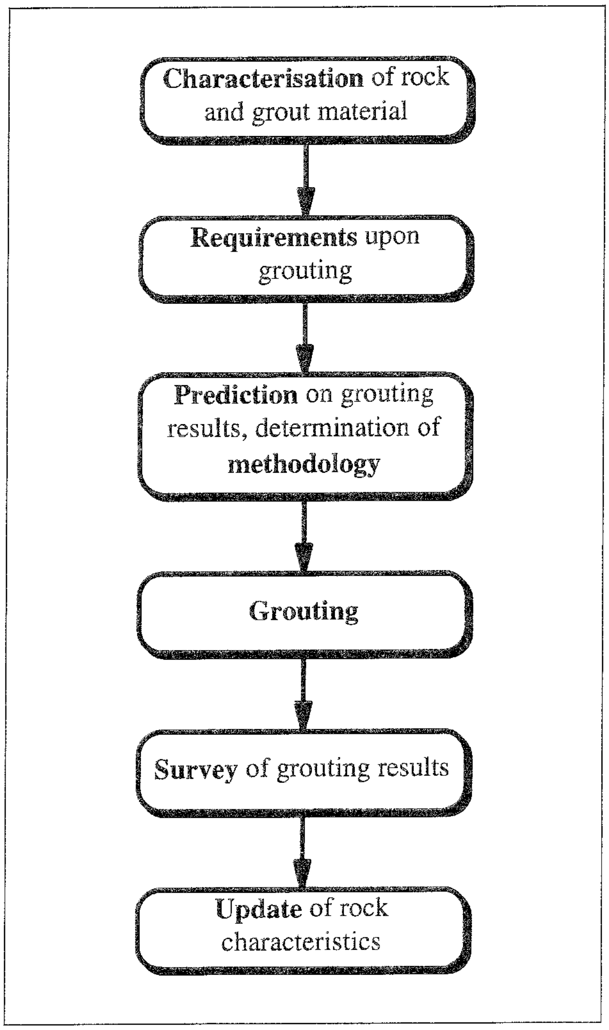


Figure 13-9. *The grouting process.*

such discontinuities may have to be crossed during ramping or shaft sinking down to 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 is scheduled to 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 13-9.

In 1996 an inventory of the current state of knowledge among Swedish contractors and consultants was carried out as a pilot study. The results are reported in /13-12/ and have been considered in the revised compilation of the state of knowledge and development needs.

Research and development work have begun in the following areas

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

The initial stage of the research has been focused on literature surveys. In 1997 the first results of the actual research work will be presented.

13.2.5 Plugs for temporary sealing off axial water flow along tunnels

General

During the operation of a repository each deposition tunnel may need a temporary plug. Individual deposition tunnels in which canisters already have been emplaced and the backfilling operation has been completed, are sealed off in parallel with emplacement of canisters in other tunnels. Two main type of functions are required by such temporary plugs:

1. Hydraulic: Seal against leakage at high water pressures. The ground water pressure should be restored as soon as possible in the backfilled tunnel in order to promote fast water saturation of the bentonite barrier in the deposition holes.
2. Mechanical: Support of the backfill material. The backfill is presumed to be compacted sufficiently to act as support for the tunnel periphery. The degree of compaction should be high also in the roof region close to the plug.

Design principles for a concrete test plug, intended to be constructed within the Backfill and Plug Test in the

ZEDEX drift in the Äspö HRL, have been formulated based on two fundamental assumptions:

1. Concrete/rock interfaces can not be made sufficiently tight by themselves. Considering the high target water pressures and the slow supply of water from the rock to the backfilled tunnel, even minor leaks will jeopardize the hydraulic performance of the plug. There is need for additional sealing that warrants tight and continuous contact between plug and rock also if small relative plug to rock displacements take place as a result of variations in groundwater pressure in the backfilled tunnel.
2. A zone of increased permeability, an EDZ (excavated disturbed zone), with a depth of about 1 meter has been created during tunnel excavation. Flowpaths within the EDZ may short-circuit the plug and need to be cut off.

Design

Figure 13-10 shows a schematic view of the plug design. A 1.5 meter slot that intercepts EDZ flowpaths is cut into the wall. The outermost part is filled with compacted bentonite, which after swelling and homogenization will serve as a tight and flexible seal. The bentonite will be put in place as square-shaped 250 millimeters by 250 millimeters by 100 millimeters blocks. The other parts of the plug are made of concrete. A retaining wall of prefabricated concrete beams is erected beam by beam towards the end of the backfilling operation, and will serve as support for the backfill during construction of the plug itself. The concrete part of the plug is cast in two stages. The first stage will serve only as support for the bentonite blocks, which will be put in place and locked immediately before the reinforced main body, stage 2, is cast. The shape, dimensions and reinforcement of the main body are based on results from ABAQUS FEM calculations, performed assuming rotational symmetry about the tunnel axis and a 3.2 MPa groundwater pressure, at maximum, in the backfilled tunnel. Figure 13-11 shows the ABAQUS model. The groundwater pressure estimate is based on µmFLOW FEM flow calculations, performed assuming typical conductivity data for concrete, tunnel backfill and bentonite seal.

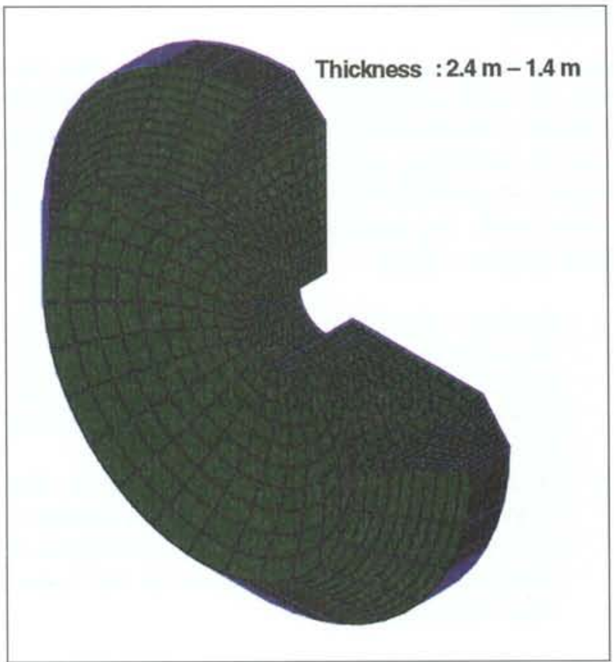


***Figure 13-10.** A 3D view of the ABAQUS finite element mesh for the concrete test plug in the ZEDEX drift.*

13.2.6 Retrieval of canister after backfilling

The deposition process is designed to be reversable so that the canister in every phase of the process can be retrieved in case any incident would occur that require a restart of the deposition sequence. Once the deposition has been completed and the bentonite buffer has saturated and swollen the canister is confined in such a way that it can not be gripped and pulled out by force. First the pressure caused by the bentonite on the canister has to be released.

The analysis that started in 1995 has been concentrating on comparison of different alternatives to remove the bentonite around the canister in the deposition hole. However, potential alternatives to release the canisters without removing the bentonite around the canister have also been considered but not addressed in detail yet.

The different investigated methods have been separated into four main groups:

- Mechanical methods.
- Hydrodynamic methods.
- Thermal methods.
- Electrotechnical methods.

The mechanical methods which are based on cutting or coring are used in different kinds of application in rock. Tools are available with heads suitable for dry excavation of materials like saturated bentonite. Also different hydrodynamic methods are applied in the industry, like water jet cutting. One different hydrodynamic alternative is to disintegrate the bentonite by the addition of salt to the water and to use batches of such a solution, that in the deposition hole mix with disintegrated bentonite to a slurry before being replaced by a new batch. The thermal method of interest is to cool the bentonite around the canister whereby the shrinkage provides a slot between the bentonite and the canister. The electrotechnical method is based on electrical concentration of water closest to the canister. Such a method has been applied in releasing e.g. steel props from non-swelling clay.

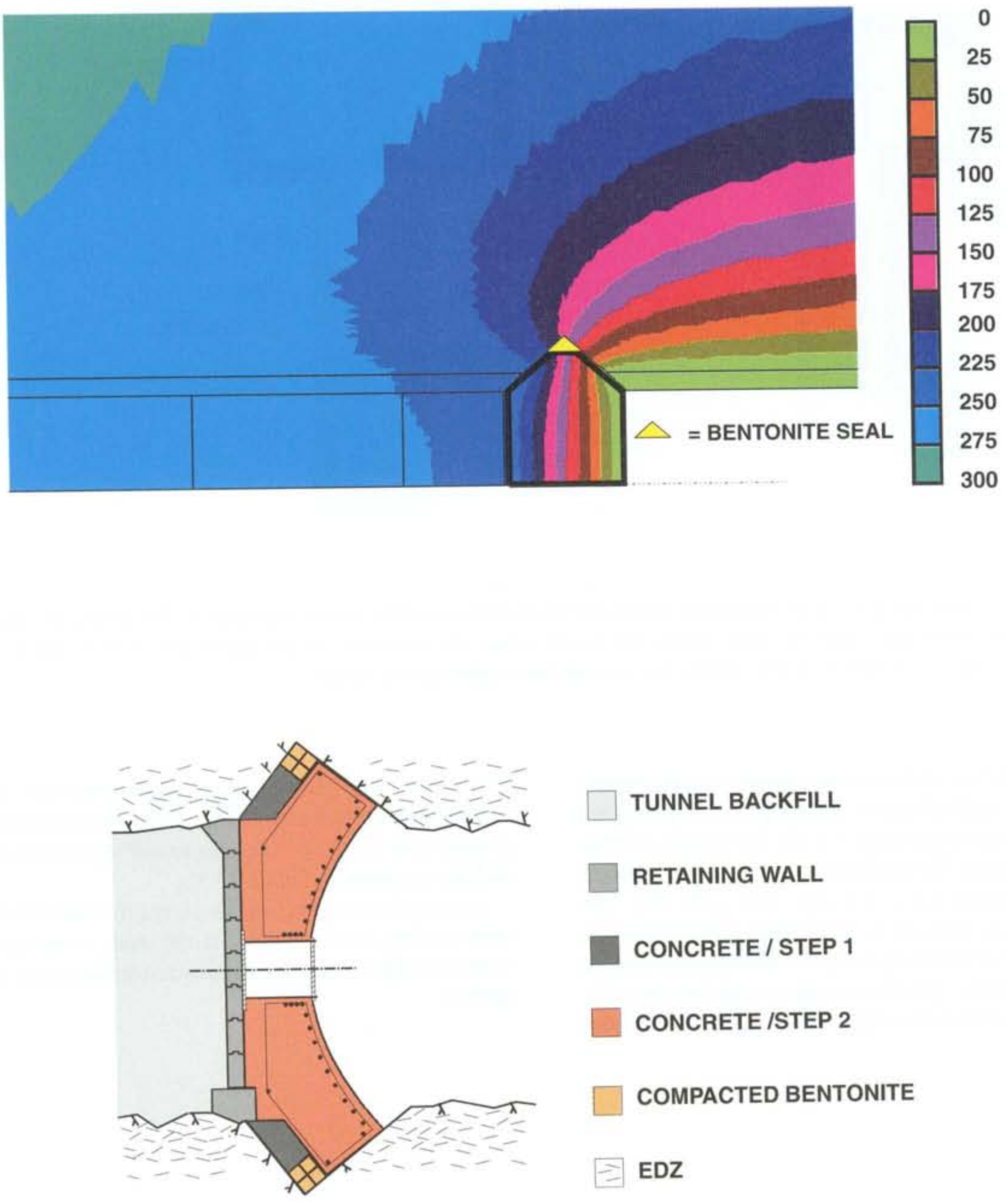


Figure 13-11

Upper — *Central part of axisymmetric FEM model of the ZEDEX tunnel with schematic representation of the plug. The contours show the potential distribution, with labels denoting hydraulic head in meters. The bold lines indicate the concrete part of the construction. The transmissive concrete/rock interfaces create a considerable pressure drop across the outermost bentonite-filled part of the slot.*

Bottom — *Proposed design of plug to be constructed in the ZEDEX tunnel. The black dots and lines within the concrete main body indicate the steel reinforcement. The parabolic shape, the angles of rock/concrete interfaces and the dimensions are based on ABAQUS FEM calculations.*

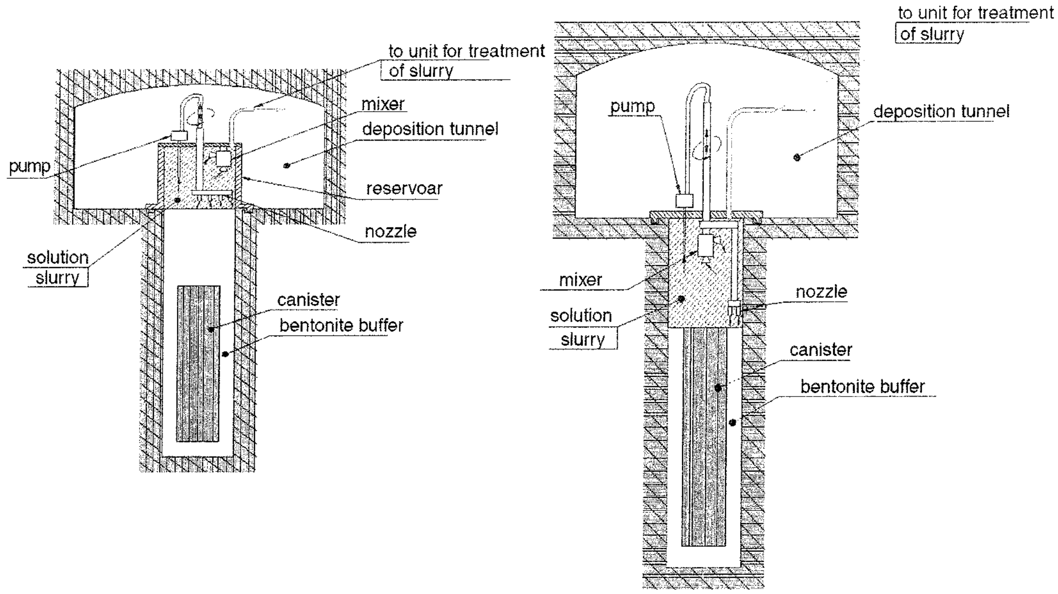


Figure 13-12. *Disintegration of the bentonite buffer and flash removal of the disintegrated skin. The left figure shows the set-up from the beginning and the right figure the set-up when the bentonite in the deposition hole is worked. The pretreatment of the recovered bentonite slurry has not yet been addressed in detail.*

The work during 1996 has been focused on the investigation of the hydrodynamic disintegration method and batches of water containing 2 – 8% by weight sodium-chloride. When the film of disintegrated bentonite is taken away the disintegration of a new skin goes fast. The removal of each skin can be efficiently made with a low flashing speed of the batch slurry on the bentonite surface. The investigations carried out indicate that the most efficient removal of the bentonite is obtained for a salt content of 4 - 6%. This method can be used from the start of bentonite removal and until the canister is free and can be gripped and lifted up. The conceptual application of the method is shown in Figure 13-12.

An equally promising method is the thermal alternative with cooling of the buffer and the study continues also with investigation of the potential and applicability of this method.

14 SAFETY ANALYSIS

14.1 GENERAL

This chapter addresses the present work

- on methods for assessment of long term post closure safety of geologic repositories for spent nuclear fuel,
- on models for use in these assessments, and
- on actual ongoing assessments.

First the general safety principles for the disposal are presented, whereafter an overview is given of the goal and scope and general planning for the coming safety report, SR 97. In the following sections ongoing work will be presented on assessment methodology and assessments of technical barriers, the geosphere and the biosphere for the SR 97 and for future use.

14.2 SAFETY PRINCIPLES

Repository safety depends on the toxicity and accessibility of the waste. The assessment of the repository's safety is influenced by time in that the quantity of toxic radionuclides declines, and in that the quantification of safety functions decreases in precision with time.

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

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

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

The safety functions can be divided into three levels /14-1/:

Level 1 – Isolation

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

Level 2 – Retardation

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

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

Level 3 – Recipient conditions

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

The safety functions at levels 1 and 2 are the most important, and are achieved by 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.

A number of major decisions and permits are required in order to site, design and build the facilities and systems required to dispose of radioactive waste. The bases for these decisions include accounts of both radiological safety during operation of the facilities or systems, as well as of safety during the long-term passive deep disposal period that begins when the repository has been sealed.

The evaluation of the long-term safety of a deep repository basically follows the same principles as those established for operating safety. Since a deep repository has to function in close interaction with the natural geological environment, however, it has been necessary to develop specific methods to carry out these assessments /14-2/.

14.3 SR 97

In SKB's RD&D-Programme for 1992 it was observed that available knowledge is now sufficient in order to select a prioritised design for the deep geologic repository, to designate candidate sites and characterise them, and to carry out the necessary safety assessments. A siting program with feasibility studies to investigate how a repository would affect the community with regard to technical, economic, societal and safety aspects is running. After 5 – 10 feasibility studies two communities will be selected for site investigations. After the sites have been characterised the law requires an application for a siting license that will require a detailed and comprehensive reporting on the safety issues involved.

In a recent decision on the SKB RD&D-Programme 95 the government observed, however, that an overall understanding of the system needed for safe disposal of the waste is a prerequisite for getting an acceptance for site investigations. It is thus required that

- a total systems analysis of the waste management system, and an
- assessment of the long-term safety of the planned system

are presented before the siting process proceeds to site investigations.

The required work to evaluate the long-term safety has now been planned and the report (SR 97) is expected to be available during 1998. The system definition and assessments actually started already during 1994 with the intention to use it as a part of the planned siting license. The assessment methodology and report structure have been discussed in SR 95 /14-2/.

14.3.1 Goal and scope of SR 97

The safety report is to provide a comprehensive evaluation of the long term safety of the planned deep repository and to demonstrate the assessment methods available. It will utilise the present prioritised design as presented in SR 95, supplemented with the new canister design as presented in chapter 13. It will discuss the Swedish geology in a generic way but utilise the site characteristics from three earlier investigated study areas as examples. The report will address both nuclear fuel and other types of long-lived wastes that have to be taken care of.

The format of the SR 97 will with minor changes follow the standard format presented in SR 95.

14.3.2 General methodology

A safety assessment of a deep repository for spent nuclear fuel should evaluate the safety for all reasonable future evolution's of the repository system. The analysis may be seen as consisting of four parts, each with its own chapter in the safety report:

- System Description.
- Choice of Scenarios.
- Analysis of chosen scenarios.
- Summation of Results, Evaluation & Conclusions.

System description

The system description includes descriptions of all processes relevant to the future evolution of the repository system. The system is divided into a number of sub-systems such as fuel, steel insert, copper canister, buffer and geosphere. For each of these, all relevant thermal, hydrological, mechanical and chemical processes are described. The processes as well as the couplings between them have previously been identified and ranked using so called interaction matrices (see scenario methodology below).

Conceptual uncertainties and validity are discussed for each process as well as the issue of the completeness of the description as a whole.

Choice of scenarios

The choice of scenarios may be seen as the choice of sets of initial and boundary conditions for the future evolution of the system. The methodology for choosing scenarios is further presented below (scenario methodology).

Analysis of chosen scenarios

The chosen scenarios are analysed based on the system description and the initial and boundary conditions for the evolution. The methods for the analyses vary widely between scenarios. The expected, normal evolution of the system is largely a compilation and integration of the descriptions of the different processes involved. Scenarios involving radionuclide transport are normally analysed using a chain of coupled computer models etc.

Summation of results, evaluation and conclusions

The results of the analyses of the different scenarios are compiled and evaluated. This is done in view of the uncertainties that inevitably accompany any long term safety assessment. The uncertainties include the issue of completeness regarding both the system description and the choice of scenarios. For each process, conceptual uncertainties and validity issues as well as numerical and data uncertainties need to be taken into account.

Finally, conclusions concerning the long term safety of the repository system are drawn.

Scenario methodology

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 analysed. 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. Instructions of how the description shall be used in the choice and formulation of scenarios, plans for documentation and handling of expert judgement are also parts of the scenario methodology.

Scenario methodology is still under development. To collect and evaluate needed information and to organise it in an illustrative fashion, is as mentioned an important part of the scenario methodology, and the safety assessment as a whole. To achieve this, a set of so called interaction matrices, inspired by the RES (Rock Engineering System) approach /14-3, -4/, are being developed for the repository system. 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. The interaction matrices have proved to be very powerful tools to compile information, and to evaluate the importance of features and processes in different contexts. To provide a clear picture of the function and evolution of the repository system they have, however, shown to be less useful. The development in this area will be continued.

During 1996 much effort has been laid on the development of interaction matrices for the different parts of the repository system. A scenario methodology to be practised in the long term safety assessment SR 97 has also been outlined. The interaction matrices contain information about the repository system. To chose and formulate scenarios statements about for instance tectonics, climate and human society are also needed. During 1996 information concerning future climate and certain human activities and their effects on the repository system has been put together.

Monitor 2000

In 1996 a new, graphic handling system for chains of coupled numerical models for radionuclide transport, has been taken into operation. The system, called Monitor 2000, allows input of the large amounts of data needed in the analyses, using designed menu's. A number of functions in the system assist in assuring that the set of input data is complete and consistent before calculations are executed. A report function facilitates overview of the structure of the calculation case as well as of the results produced.

During 1996, the system has been extensively tested and is now an important tool for quality assurance in the calculation of radionuclide transport in safety assessments.

14.4 ANALYSIS OF TECHNICAL BARRIERS

14.4.1 General

The main focus of the work during the year has been collection and refinement of data and models for the upcoming safety analysis, SR 97. Some of the areas where new results have been obtained are presented in this section.

14.4.2 Inventories

Spent nuclear fuel comprises most of the waste that is to be disposed of in a deep repository. Data on the quantities and composition of the radionuclides in the waste are needed in order to carry out a safety assessment. A typical fuel is chosen for the calculations of radionuclide inventory and decay heat in a safety assessment.

The typical fuel for SR 97 consists of twelve BWR assemblies of type Svea 96 with a burnup of 38 MWd/kgU. The radionuclide inventory and decay heat for the fuel in the repository are calculated by means of a computer program. The properties of the typical fuel as regards quantities, burnup and cladding material etc are used as input data in the calculations.

New calculations of radionuclide inventory and decay heat for the typical fuel as defined above have been performed using the established computer SCALE4.3/ORIGEN-S and CASMO-4. Calculations have also been done for a PWR fuel with a burnup of 42 MWd/kgU and two high burnup fuels: a BWR with a burnup of 55 MWd/kgU and a PWR with a burnup of 60 MWd/kgU.

14.4.3 Fuel dissolution model

A model for fuel dissolution must describe both the release of nuclides due to matrix dissolution and the release from the fuel-clad gap and from cracks in the fuel. It is also desirable to describe the dissolution of any radionuclide content in grain boundaries.

A new model for the matrix dissolution has been developed for use in SR 97.

The model is based on a set reactions and rate constants describing water radiolysis and rate constants for the reactions between the oxidants O_2, H_2O_2 and the UO_2 surface. Figure 14-1 shows the results from the calculations. The number of moles of consumed oxidants (O_2, H_2O_2) are equivalent to the number of moles oxidised uranium. The model calculates that $2.5 \cdot 10^{-7}$ moles of uranium per pellet will be oxidised during the first 3 000 years of water contact, which is about equivalent of an average fractional dissolution rate of $2 \cdot 10^{-7}$/year.

14.4.4 Radionuclide solubility's

Owing to the very low water flux inside a canister, many of the radioelements will precipitate as secondary minerals if they are released from the fuel matrix. Solubility limitations are a very important barrier function, since the release of radionuclides from the near field is often directly proportional to solubility.

A new study of solubility's, that incorporates recently selected data for critical radionuclides as well as comparisons with trace element concentrations in natural systems, has been done for the use in SR 97. This has been done according to the following procedure:

- Ranking of the different radionuclide – natural water ligand interactions by comparison of their relative strength, in order to study the effect of the main chemical parameters on radionuclide solubility and speciation.
- A study of the occurrence and concentration of radioelements (or stable isotopes) in natural systems to be used as a reference to the calculations.
- A summary of radionuclide concentrations in spent fuel dissolution experiments. This information provides another reference level to the calculations.
- Finally, radionuclide solubility limits have been calculated using the EQ3/6 code package. Three different groundwater's have been studied together with a hypothetical bentonite porewater.
- The calculated solubility values have been discussed and compared with both measured values in natural systems and radionuclide concentrations in fuel dissolution experiments in order to check the calculations performed and to provide realistically conservative values for performance assessment. Uncertainty ranges of the values have also been established.

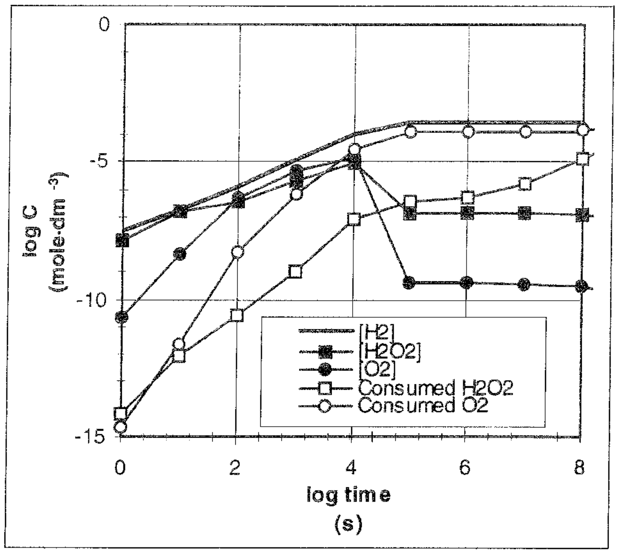


Figure 14-1. Evolution of reductants and oxidants generated by water radiolysis and consumed oxygen and hydrogen peroxide by the oxidation of the UO_2 surface.

14.4.5 Diffusion and sorbtion of radionuclides in bentonite

Sorbtion and diffusion coefficients for radionuclides in compacted bentonite have been reviewed and the data set has been updated for the use in SR 97. The variability and uncertainty of the diffusivity data spans somewhat more than an order of magnitude up and down. Most of the nuclides have an effective diffusivity in around 10^{-10} m²/s. Ion-exclusion effects are observed for C, Cl and Tc in oxidising waters. Effective diffusivities are nearly two orders of magnitude lower for these elements.

Contradictions between measured apparent diffusion coefficient for cations and traditional diffusion sorption theory can be explained by surface diffusion. Surface diffusion effects are found for Cs, Ni, Pa, Pb, Sn, Sr and Zr. Effective diffusivities for these elements are in the order of 10^{-8} m²/s. The surface diffusion effect should decrease in saline waters which is seen for Cs and Sr where there are data available. It is also deemed that Ra will experience this effect because of its similarity to Sr. The other nuclides should also show this decrease but no data are available.

In highly compacted bentonite, sorption and hence its distribution coefficient is not very well defined, and a pore diffusion coefficient or a surface diffusion coefficient is not very well defined either. Therefore an apparent diffusion coefficient and a total concentration gradient should be more relevant in describing the diffusion process in compacted bentonite.

14.5 ANALYSIS OF THE GEOLOGIC BARRIER

14.5.1 Development for SR 97

The safety report SR 95 /14-2/ presented a template for future safety reports to be presented by SKB. Furthermore, the methodology for far field transport modelling was reviewed in detail as well as the available assessment models. During 1996 the safety analysis project SR 97 was launched and most of the work has been devoted to detailed planning of all analyses necessary. Considering the time constraints, SR 95 here constitutes a necessary foundation providing the structure and the tool box.

The interaction matrices are used to give a systematic description of the process system for the deep geological repository according to the KBS-3 method. As discussed above the safety analysis to come will strive for completeness. A preferred method for the SR 97 planning is to use the interaction matrix for the far field as developed by SKB /14-4/ as the starting point. The documented priority assignment within this matrix is especially important in this respect.

In SR 97, three hypothetical sites will be used, arbitrarily named Aberg, Beberg and Ceberg. The data for these sites is taken from previous investigations conducted by SKB at sites in Sweden. These are:

- Aberg, which is based on data from the Äspö Hard Rock Laboratory, in southern Sweden;
- Beberg, based on investigations at Finnsjön, in central Sweden; and
- Ceberg, based on investigations at Gideå, in northern Sweden.

The best available information/understanding for each site will be strived for without performing any new field campaigns. It has to be born in mind that the site characterisation studies and analyses of data have been conducted at different times for different goals.

The main objective of the analysis will not be to compare the sites. Instead focus in the far field will be on how different conceptual models for describing flow and transport in fractured rocks affects the performance of the geosphere barrier *illustrated by three different geographical locations* in Sweden. Furthermore, it must be transparent how the data of each site have been used in the assessment models utilised.

The initial activities within SR 97 have been devoted to compilation of data and geoscientific understanding of the sites. Even though no new data will be collected there are differences in methodology affecting the interpretation of the sites which need to be overcome before going into further analyses.

Among the on-going activities it may be mentioned that a compilation of radionuclide sorption coefficients for transport in fractured rocks is being performed. For every element, there is a recommendation of a realistic K_d value with an uncertainty limit. The selection is based on experimental investigations.

14.5.2 General development of models

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. Coupling of the near field transport compartment model NUCTRAN to the channel network model CHAN3D was reported during 1996. A set of illustrating calculations used Äspö data /14-5/. The resolution of the simulated release from the near field is high in that sense that NUCTRAN considers release through several pathways. It was concluded that this resolution has an impact on the far field results. Depending on where the nuclides are released they may travel in very different paths.

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. 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. These features have been tested using site specific data from Äspö, the LPT2 long term pumping test /14-6/. Unlike previous modelling studies of LPT2, this study use the inverse modelling capabilities of HYDRASTAR to condition the model results to observed hydraulic heads. The purpose of this conditioning via inverse modelling is to improve the reliability of input hydraulic conductivity fields and thus to minimise the uncertainty of the model predictions. The calibrated model simulations were successful in representing the response of the rock mass to the LPT2 pumping test. Specifically, the mean of the Monte Carlo realisations of simulated drawdowns generally reproduced the magnitude, timing and shape of the observed drawdowns.

14.6 ANALYSIS OF THE BIOSPHERE

To assess the radiological consequences of a deep repository, it is necessary to describe the biosphere around the repository site. The description must contain information on the recipients for deep groundwater in the biosphere and on the local ecosystems. Man's exploitation of the natural environment should also be included in the description.

Since the biosphere is expected to undergo changes during the time covered by the assessment, it is necessary to analyse not only the present-day biosphere on the repository site but also possible future variations.

The radionuclides that may be released from a deep repository will be mainly transported by the groundwater and come into contact with the biosphere through running water, lakes, wells or the sea. The groundwater can also penetrate sediments, peat bogs or soils if these are discharge areas. In rare cases, radionuclides can enter the biosphere as gas.

Normally a well is regarded as the most unfavourable scenario, which has been described in earlier safety reports /14-2/. In the current safety assessment a novel approach has been introduced to model the biosphere. The different recipients of the deep groundwaters released into the biosphere are subdivided in functional modules of major ecosystems where dose-factors for critical groups are calculated /14-7/. The developed modules are bog, lake, well, running water and Baltic sea coast. Modules under preparation are cultivated and natural grasslands, while forest will be developed later.

The advantage with the modules are that the biosphere is broken down in handable pieces which more realistically can be described with current knowledge. This gives higher confidence in the appropriateness of selected parameters. The choice of parameters that describe consumption, irrigation, livestock farming etc are based on the present-day biosphere. Even though there are likely to be considerable changes from today's situation at the time the radionuclides reach the biosphere, this is the only system that can be described with reasonable precision. But with good knowledge of the conditions on a specific site in combination with the modularised approach, it likely to make relatively far-reaching predictions regarding time-dependent changes in the biosphere and to analyse their consequences.

15 SUPPORTING RESEARCH AND DEVELOPMENT

15.1 SPENT NUCLEAR FUEL

The results accumulated in SKB's fuel leaching experimental program in the hot cells at Studsvik Nuclear have given a good understanding of the corrosion of spent nuclear fuel. The studies on the durability of spent fuel in groundwater were started in Sweden in 1977, and the present programme of studies was defined for the first time in 1982. A summary of the results and data obtained within this programme in recent years is provided in /15.1-1/, together with a comparison with the database collected in the past ten years.

15.1.1 Fuel characterization

The surface area of the fuel available for the corrosive attack by ingressed groundwater is an essential parameter for determining the absolute corrosion rates. This is especially the case in treating data obtained in flow-through leaching experiments, which are to be started soon in Studsvik. 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 /15.1-2/. Reproducible values of fuel surface area in the range 70 – 120 cm^2/g have been obtained. Fuel samples to be used in the forthcoming flow-through leaching reactors have been prepared during this year. Several PWR spent fuel fragments have been ground and sieved and then the surface areas of two fractions (0.125 – 0.25 mm and 0.25 – 0.5 mm) has been determined using the BET method with argon sorption. Five samples have been prepared in this way and are conserved under Ar atmosphere.

15.1.2 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. Archive data from experiments designed to study the influence of linear power/burnup in the leaching behavior of spent fuel, carried out during the period 1990 – 1993, have been reported /15.1-3/.

The analyses of the leach solutions are performed directly without separations or isotope dilution via mass spectrometry with an inductively coupled plasma source instrument (ICP-MS), in operation since 1992. During this period substantial efforts have been devoted to the improvement of the analytical methods necessary for the analysis of the very low concentrations of actinides and fission products expected from the flow-through experiments, as well as the analysis in high salt media, where a preliminary separation step from the salt rich solution before introduction in the ICP-MS instrument is included in the analytical scheme. A Dionex separation column has also been installed and the work with the separation of the various isobars is in progress.

In the future, the emphasis will be shifted from large series of leach experiments of BWR and PWR in oxidizing conditions to experiments where anaerobic and reducing conditions in the bulk solution will be maintained through inert gas atmosphere or reductants e.g. Fe(II) sources. This will require more involved experimental setups in the hot cell. Potentiometric measurements, both in a non active laboratory and in the hot cell have been performed during the whole year. Blank experiments using unirradiated $UO_2(s)$ have been performed to test the leaching in the presence of various Fe(II) sources, like magnetite, before performing them in the hot cell with spent fuel. A method for in-situ measurements of pH and E_h in the hot cell during the leaching process, using a computer controlled system for potentiometric measurements, is now in operation in the hot cell laboratory. Silver – silver chloride reference electrodes prepared in Studsvik were found to be stable during several months in the chloride concentration of the ground water used as leach solution. The stability of the measuring system is now confirmed for more than one year periods: despite the strong radiation field, pH values stable within 0.05 units have been measured. The use of these data in fuel dissolution modeling in anoxic conditions is in progress.

An Access based database, including all the results obtained in SKB's Spent Fuel Leaching Program, has been under compilation during this year. The structure frame of the database is constructed and the majority of data has been stored. The storage of all the available data, the compilation of an accompanying handbook, as well as improvements towards a more user-friendly structure are some of the tasks where work is still in progress.

A combined experiment to study both the spent fuel corrosion and the radionuclide diffusion in bentonite clay, in conditions which simulate those in the repository, was started ten years ago. Pieces of a spent fuel pin, together with cladding, were placed in diffusion cells surrounded by bentonite clay. The diffusion cells were located in cylinders containing a low saline carbonatic groundwater that was previously equilibrated with bentonite clay. A total of ten diffusion cells were loaded. In some of the diffusion cells, additives of metallic iron, metallic copper or the Fe(II) mineral vivianite were mixed with the clay. Experiments with variable total contact times were performed /15.1-4/. The experimental setup and some results have been reported in earlier publications /15.1-5 – 15.1-8/. 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), cobalt (^{60}Co) and (^{237}Np).

The extremely low concentrations of the elements that are of most interest in the long time safety perspective have made necessary the development of suitable analytical procedures. A method for the separation and analysis of the actinides, based on solvent extraction of americium, curium, neptunium, plutonium, and uranium from dilute nitric acid solutions, followed by selective back extraction is being developed.

Preliminary results for some other elements, not reported earlier, have been obtained. The mobility of trivalent elements (^{154}Eu, ^{241}Am and ^{244}Cm) is shown in Figure 15.1-1. The mobility shows an apparent diffusivity of about 10^{-15} m^2/s in bentonite clay. The analysis of these elements is still in progress.

The results of the leaching of uranium from the spent fuel and its transport through compacted bentonite are shown in Figure 15.1-2. The analysis of uranium is based on the uranium-isotope ^{236}U, since the concentration of the other uranium isotopes $^{235,238}U$ is concealed by the natural concentration of uranium in the bentonite clay (some µg/g). Based on a calculated isotopic composition of the uranium in the spent fuel, a total released fraction of about 10^{-5} was estimated after one year. From these results an apparent diffusivity of 10^{-14} m^2/s has been evaluated. The observed apparent diffusivity of uranium has been evaluated as 10^{-13} m^2/s.

Data from the analysis of ^{236}U in bentonite slices from the six years experiments have also been obtained. Preliminary results from leaching-diffusion experiments in the presence of additives in the clay show that metallic iron reduces by approximately one order of magnitude the release of uranium.

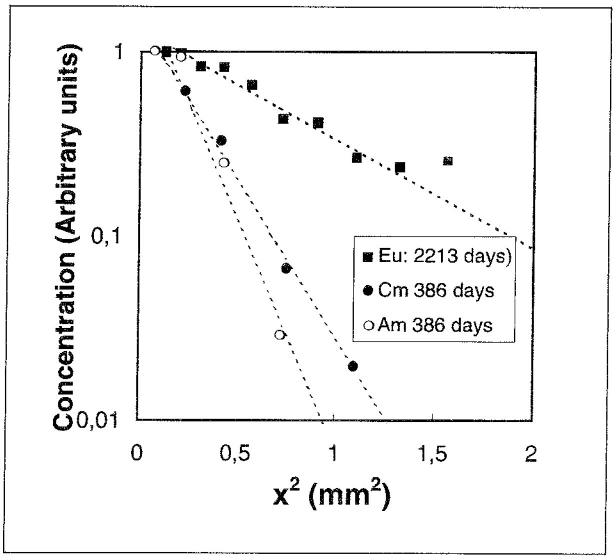


Figure 15.1-1. Preliminary results of the diffusion of europium, americium and curium (^{154}Eu, ^{241}Am and ^{244}Cm) from spent nuclear fuel into compacted bentonite.

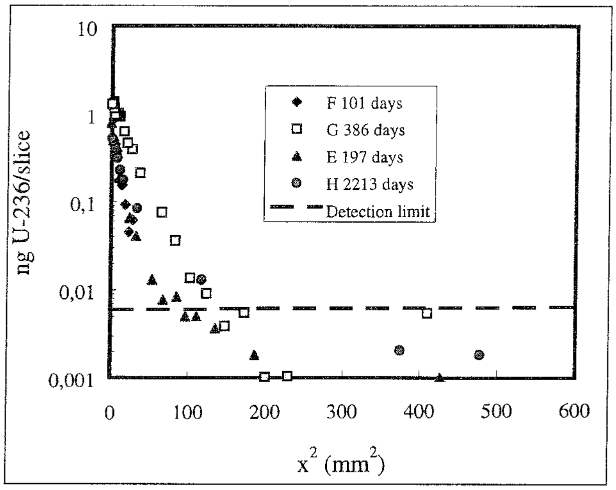


Figure 15.1-2. Experimental results on the diffusion of uranium (^{236}U) from spent nuclear fuel in contact with bentonite clay.

15.1.3 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, reductants and dissolved uranium has earlier been performed in a closed system initially containing fragments of used fuel and oxygen free distilled water, see Figure 15.1-3.

In these experiments the hydrogen concentration was found to increase continuously during a more than eight months period, while the concentrations of oxygen remain very low. A clear deficiency of oxidants was found in the overall system /15.1-9/.

In experiments with initially oxygen free millimolar carbonate solutions the concentrations of hydrogen and oxygen increased continuously, see Figure 15.1-3. A preliminary analysis of the experimental data indicates mass balance for oxidants and reductants when H_2, O_2, H_2O_2 and dissolved uranium are taken into account. A kinetic model for the radiolytic oxidation of spent fuel based on surface redox and dissolution reactions is under development.

The steady – state concentrations and times to reach them in a solution undergoing gamma radiolysis have been analyzed for important radiolysis products, both in the presence and in the absence of UO_2 fuel /15.1-10/. An

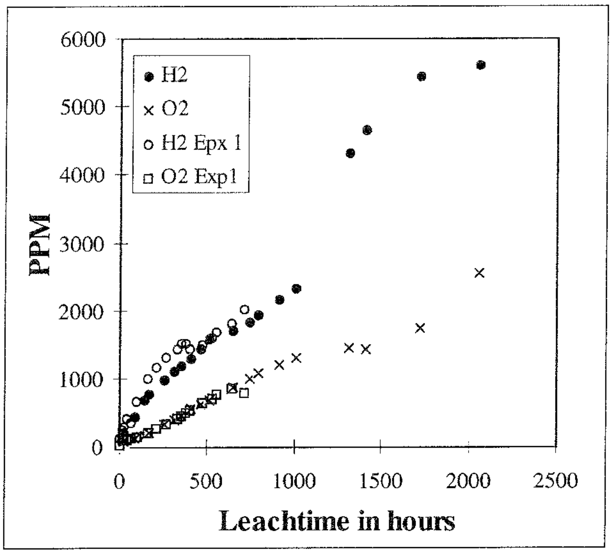


Figure 15.1-3. *Oxygen and hydrogen concentrations in the gas phase as function of the leach time. Results in carbonated solutions compared to distilled water ones.*

evaluation of the water layer thickness, which affects the radiolytic oxidation of used nuclear fuel shows that it is equal to the diffusion range of the radicals.

15.1.4 Models

The development of a spent fuel dissolution model under oxidizing conditions has continued during the whole year in 1996. A special effort was devoted to the development of a simplified, but robust version of the fuel dissolution model to be used in SR 97.

The model under development is a kinetic model, instead of an equilibrium one, since the spent fuel matrix constitutes a dynamic redox system by itself, due to a continuous generation of oxidants and reductants at the fuel – water interface through radiolysis. The main processes considered to progress in the system are:

- The production of oxidants through radiolysis of water. A considerable effort has been devoted to the calculation of the production rate of radiolytic oxidants, able to reproduce the experimental observations on oxidant evolution in spent fuel/water systems.

- The oxidation of UO_2 surface coordination sites, equivalent to the simultaneous reduction of oxidants in the system. Extensive literature search and data treatment has been performed to estimate the most probable rates of oxidation of the spent fuel matrix. these rates are difficult to estimate since usually combined oxidation-dissolution rates are reported in most experiments.

- The dissolution of the oxidized sites with subsequent release of uranium to the solution. The species formed by uranyl in solution depend on the aqueous chemistry of the system.

- The possibility of the formation of secondary U(VI) phases and their influence on the alteration of the $UO_2(s)$ matrix has been also considered. This was done by performing calculations in their absence, considering their formation assuming solid-solution equilibrium and as a third possibility considering also the kinetics of dissolution-precipitation of these secondary phases, while approaching the solubility equilibrium.

All these processes have then been incorporated in a ReDucing Capacity (RDC) kinetic model /15.1-11/ by developing mass balance equations for each component: oxidant, uranium, oxidized and non-oxidized sites in the spent fuel surface, see Figure 15.1-4.

Finally the model has been tested and calibrated with experimental data from leaching experiments with unirradiated uranium oxide, with spent fuel data as well as experimental data from oxidant evolution in spent fuel/water systems. In all cases acceptable explanation of the experimental data on evolution of uranium concentrations in these systems has been achieved /15.1-12/.

The evolution of the plutonium concentrations with time in spent fuel leaching experiments has also lately been achieved with this model. In most of the fuel leaching experiments in oxidizing conditions, the plutonium concentrations increase during the first week to more than ten times the solubility of the solubility limiting solid, then slowly level down to this value (about 2 10^{-9} M) after some months. This behavior has been modelled assuming the following:

- The rate of dissolution of the plutonium is expressed as the fraction of the dissolution rate of the $UO_2(s)$ matrix proportional to the fraction of plutonium in the spent fuel. Plutonium dissolves mainly as Pu(IV) hydroxide, which is a compound known for its strong tendency to form colloidal particles. This explains the increased initial plutonium concentrations. The subsequent aggregation of these colloidal particles to form the amorphous plutonium hydroxide explains the levelling out of the plutonium concentrations with time. Another outcome of the modelling of spent fuel dissolution data is the estimation of a solubility product for the amorphous plutonium hydroxide of the same order of magnitude as the value reported in the literature, which increases the reliability of the model /15.1-12/.

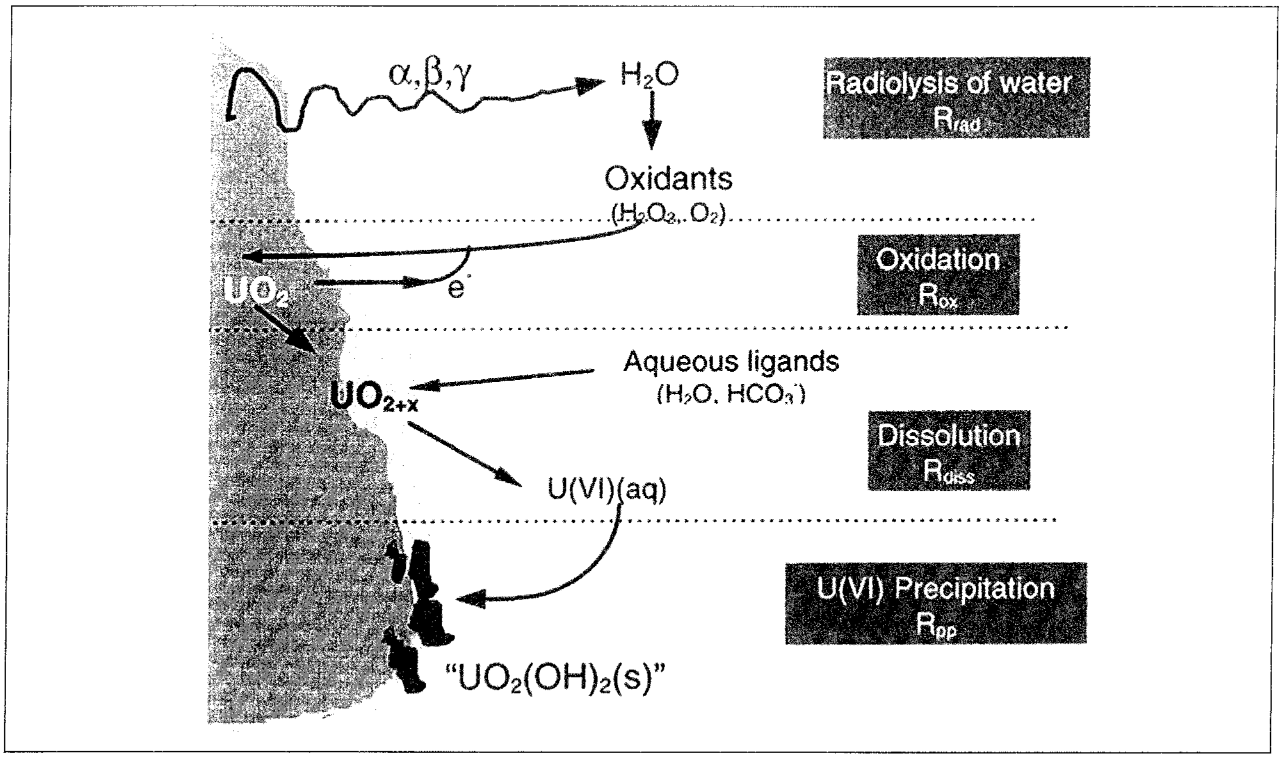


Figure 15.1-4. *A schematic representation of the main processes taking place at the spent fuel/water interface.*

15.1.5 Natural analogues

The experimental and modelling studies on 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. A Ph. D. thesis on this subject was presented this year by Esther Cera at the Autonomous University of Barcelona, Spain /15.1-13/.

Preliminary dissolution results on soddyite indicated an initial increase of the uranium concentration with time before reaching steady-state. In uranophane dissolution experiments an initial increase of the uranium content in solution was followed by a decrease of its content with time. The experimental results obtained have been reported previously /15.1-14/.

Experimental studies on the thermodynamic and kinetic properties of these two solid phases have continued during the period. Synthesis and characterization (XPD, FTIR, SEM, BET) of samples of the two solid phases was followed by dissolution experiments, in order to study the influence of both pH and carbonate on their kinetic and thermodynamic properties. For this purpose, two series of experiments were started in the absence and in the presence of carbonate and a solution composition was selected for each solid phase in order to avoid the precipitation of secondary uranium solid phases during the dissolution tests.

Characterization (XPD) of the leached solid phases after the dissolution period was carried out and was compared with the initial characterization, in order to identify possible phase changes during the dissolution process. In uranophane dissolution experiments carried out in carbonate free solutions the precipitation of a secondary phase, identified by XPD as soddyite, was observed. This result agrees with previous results obtained from dissolution experiments performed with natural samples /15.1-13/.

Thermodynamic modelling of the data obtained from soddyite dissolution experiments performed in the presence of carbonates resulted in a solubility product of soddyite $\log K_{s0} = 3.9 \pm 0.7$. This solubility constant agrees fairly well with the value $\log K_{s0} = 3.0 \pm 0.3$, determined from a natural sample /15.1-13/.

The general trend of the total uranium in solution as a function of time has been modelled using a kinetic equation obtained from the principle of detailed balancing of the dissolution reaction /15.1-15/. In addition, the EQ3/6 code has also been used to model the uranium concentrations as a function of time. Comparable results have been obtained from both modeling exercises.

The initial dissolution rate, normalized to the total surface area used in the experiments as measured with the BET method, gives an average value of $(6.8 \pm 4.4)\ 10^{-14}$ mol $cm^{-2}\ s^{-1}$. As seen in Figure 15.1-5, the kinetic model

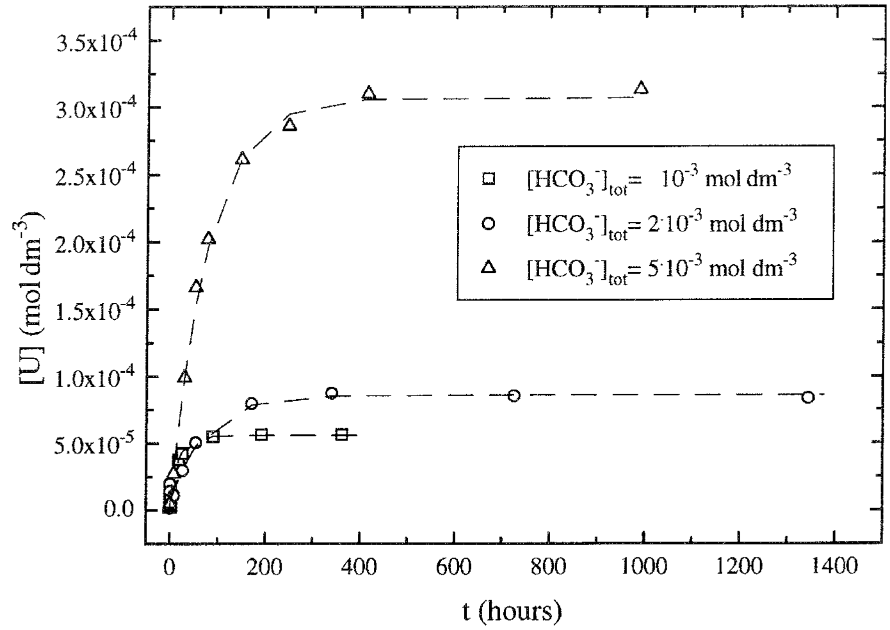


Figure 15.1-5. *Kinetic modelling of the variation of the total uranium concentrations with time during soddyite dissolution in carbonated solutions.*

developed explains quite well the experimental data /15.1-16/.

15.2 BUFFER AND BACKFILL

15.2.1 Microstructural model of bentonite buffer

Dense smectite clay has the capability of providing a practically impermeable embedment of waste canisters, which limits the means of transport of radionuclides from a defect canister to diffusion only. The mechanisms involved in such transport and the diffusive transport capacity are being investigated in order to be able to accurately understand and assess the barrier potential of smectitic clay. The key factors are the physico/chemical properties of smectite minerals and their spatial organisation with respect to the physical state and composition of the pore water. These factors combine to yield the microstructural constitution of the clay which is known to be a function of the bulk density, dominant cation and temperature and which controls the retention and penetration of radionuclides. This means that the groundwater chemistry of the surroundings, i.e. the host rock as well as the radionuclides and various species released from the smectite minerals, have an effect on the tightness of the clay barrier. These effects on the microstructure and its impacts on radionuclide transport are studied.

In 1996 experimental work has contributed to yield a solid basis for developing a microstructural model of bentonite that takes density and pore water chemistry into consideration /15.2-1/. The relevance of initially proposed clay embedding methods for electron microscopy has been verified and the technique is now available for preparing samples that have been used in laboratory experiments performed within the project. The fundamental question of how accurately samples prepared for transmission electron microscopy (TEM) represent the true microstructure has been tested by measuring the swelling pressure and determining X-ray diffraction (XRD) spectra of monomer-saturated clay samples with the dry densities 0.95, 1.27 and 1.59 Mg/m^3. The swelling pressures were found to be on the same order of magnitude as for water-saturated samples, and the XRD spectra showed the same typical MX-80 patterns as of hydrated samples and also that interlamellar adsorption of the monomer had taken place. TEM gave micrographs of the type shown in Figure 15.2-1. They resemble those obtained by use of the ordinary more complex preparation technique that comprises stepwise replacement of pore water by monomer before polymerisation is made.

Numerical modelling for determining microstructural changes by simulated compression has started in 2D and yielded reasonable results. The evolution of the microstructural constitution depends on the impact of the pressure by which the bentonite grains are compacted to form blocks. Numerical modelling in 2D has shown that initially densely layered grains are compressed to yield a

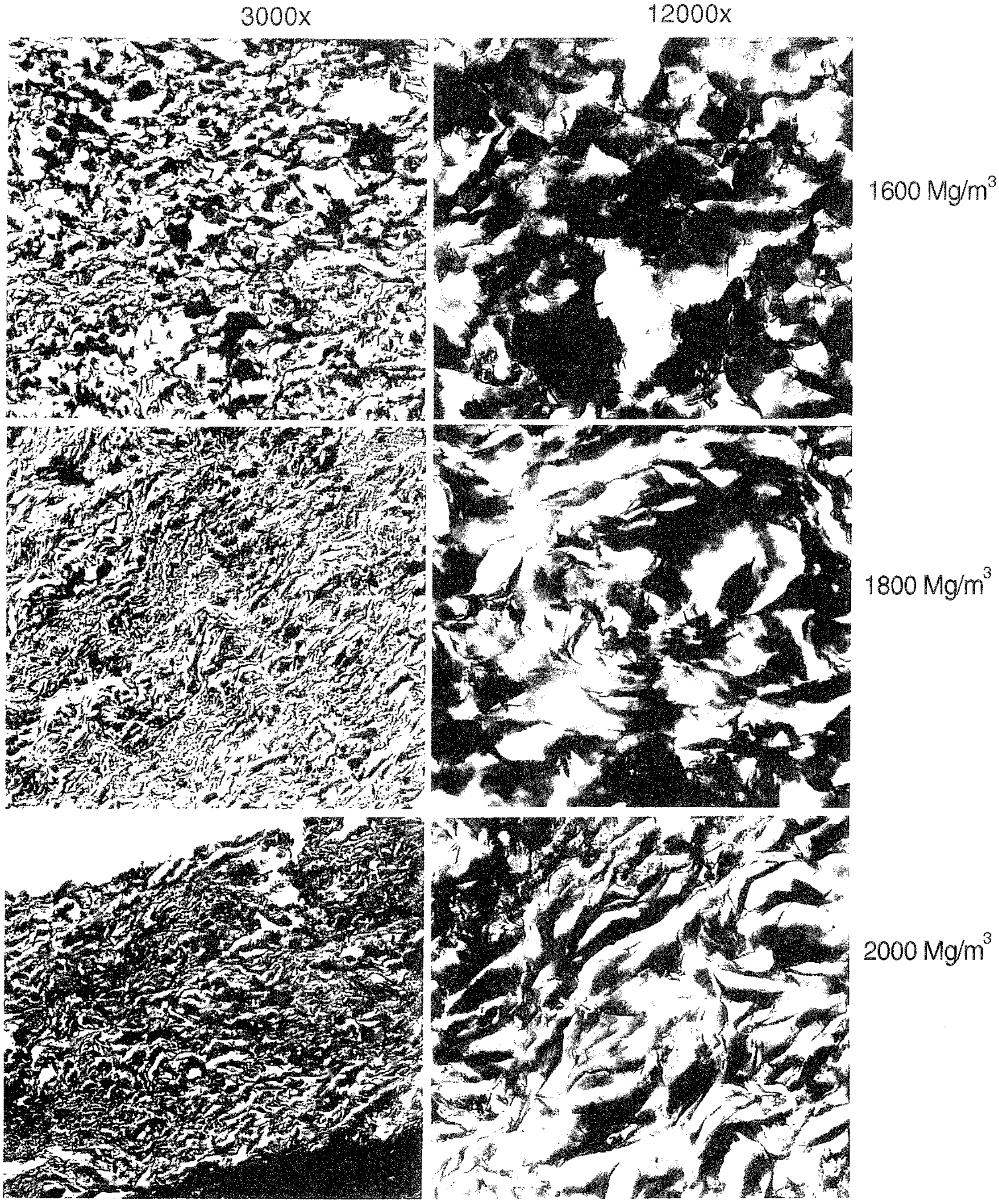


Figure 15.2-1. *Transmission electron microscopy micrographs of MX-80 at different densities. Black parts represent bentonite while light parts represent pore space.*

bulk dry density of about 1.60 Mg/m^3 at 100 MPa pressure, which is on the same order of magnitude as recorded in experiments. The voids between the deformed grains are approximately equivalent to cylindrical tubes with a diameter of 10-40 μm, see Figure 15.2-2, which will be occupied by clay that moves in from the successively expanding and softening grains.

The work is jointly carried out by VTT (Finland), Univ. of Hannover (Germany), KTH and Clay Technology (Sweden) within the frame of a EC-project. The project head is Prof. Roland Pusch at Clay Technology. One specific approach, out of SKB's scope of work, is the investigation of organophilic bentonites, i.e. smectite clays that are pre-treated with certain organic substances, which have been found to sorb cations and high amounts of anions (iodine) as well. The preparation of organo-bentonites is being optimised towards retention of cations and anions according to the chemical spectrum of the relevant radionuclides contained in the waste material. Key data of the properties are being processed for incorporation into nuclide diffusion models.

15.2.2 Modelling of water-unsaturated buffer materials

AECL, Canada, PNC, Japan and SKB started in 1996 a project termed VALUCLAY, which aims at validating Thermo-Hydro-Mechanical (THM) models for unsaturated bentonite-based barriers (VALidation of codes/ models that describe the Unsaturated behaviour of engineered CLAY barriers). In the first phase of the project, with the duration of one year out of five totally, a preliminary version of a quality-assured database, containing values for buffer material properties, has been outlined. A workshop was organised in Tokyo in October to discuss and settle the principles that should be applied for data selection, and to agree on the procedure for data authorisation and quality assurance. Modifications to the original database structure proposed by the VALUCLAY secretariat at AECL in Pinawa, Manitoba, were suggested and discussed. SKB data, compiled in accordance with the decisions taken in Tokyo, have been submitted to the VALUCLAY secretariat for further processing, i.e. merging with corresponding data compiled by the other participating organisations. The files are in MS ACCESS database format and contain basic material properties as well as experimental results and values for parameters associated with specific material models used in THM modelling. In all, 53 properties are included, each corresponding to a database table. The majority of the data concern the bentonite buffer, but for a few properties also data for backfill materials and for the excavation damaged zone have been included.

In the EC-project CATSIUS CLAY the objective is to evaluate different calculation methods for buffer materials (Calculation and Testing of behaviour of Unsaturated Clay). The calculations are separated into five tasks. The first and second were addressed during 1996. The first concerned infiltration in a column with an unsaturated stable medium as well as thermal convection in a saturated stable medium. These examples had for some conditions analytical solutions. The second concerned five laboratory experiments with compacted and water-unsaturated Boom clay, which had a constant overload the whole time while the underpressure in the pore water was increased in steps. The volume was measured until water-saturation was reached. The application of the ABAQUS code /15.2-2/ gave correct results in the first task when analytical solutions were available. In the second task the measured values were only partly possible to simulate with the preliminary THM model used in the ABAQUS code. The result in CATSIUS CLAY thus underlined the statement earlier given that the preliminary model is not fully equipped for accurate predicting of unsaturated mechanical processes. A supplement to the code was, however, developed in the project, which increased the accuracy in the specific problem given in the second task. The guiding relationships used will be further studied for a more general improvement of the preliminary model.

The project is headed by Prof. Eduardo Alonso, Barcelona. Participating groups are CIMNE and UPC (Spain), ANDRA (France), ISMES (Italy), SCK-CEN and Ulg (Belgium), UWCC (England) and Clay Technology (Sweden).

15.2.3 Precompaction of bentonite blocks

Blocks of highly compacted bentonite for construction of the buffer around the canisters should either be made in a size that is small enough to be manageable by hand or as large as possible for machine handling. By man each block can be easily adjusted to an exact position around the canister, while a machine is more complex to maneuvre for adjustment of block positions out of the predestined location. Consequently emplacement of blocks with a machine favours large blocks. During 1996 the development and testing of uniaxial compaction of bentonite blocks have been focused on large blocks with the objective of developing technique for compaction of cylindrical blocks with a full-size diameter for the deposition hole, i.e. 1.65 m.

The tests have been made in a form for bentonite blocks with a diameter of 1.0 meter and a height of 0.4 meter, which was used in a large press in Ystad with a capacity of 30 000 metric tons. The water ratio of the bentonite was varied between 10% and 23% and the compaction pressure between 50 MPa and 150 MPa, the major part being made with a pressure of 100 MPa. After compaction the blocks were released from the form, cut in pieces, and the density and the homogeneity determined.

The results showed that the homogeneity was very good with a standard deviation in density that was less than 0.5%. The only imperfection was a small crack that occurred in the upper corner in some of the blocks. The crack

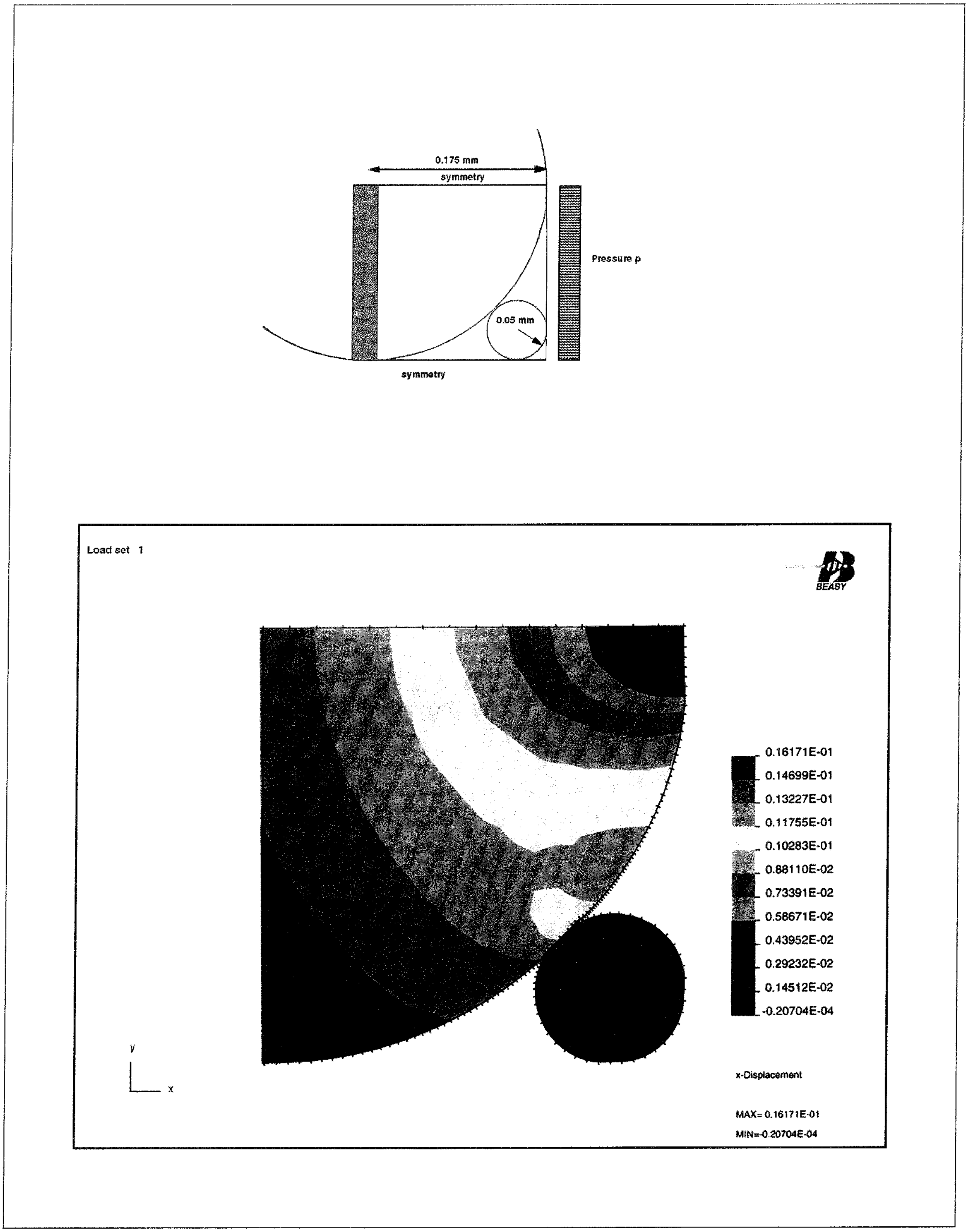


Figure 15.2-2. *Deformation of grains by uniaxial compression (2D) according to a boundary element calculation (BEASY). Upper: Load conditions. Lower: Strain at 100 MPa pressure assuming E=5 000 MPa.*

was caused by the elastic swelling of the block that took place at unloading. The crack, which is too small to affect the function of the block, was largest in the dry blocks and non-existent in the wettest blocks, a direct function of the amount of elastic swelling. Figure 15.2-3 shows two photos, one taken during compaction and the other one taken after compaction. The block was lifted in two hooks which were screwed into the block.

Cylindrical blocks with a central hole are required around the canisters. A form for compaction of such blocks in the scale about 1:6 was also built and tested. The tests showed that such blocks can be made with a very good precision and quality.

The experience gained in the semi-scale tests have been used for scaling up the technique to full scale and a form for blocks with a diameter of 1.65 m and a height of 0.5 m has been designed. Both blocks with and without a central hole can be made in the form. This design is adapted to the needs of Äspö so that compacted bentonite blocks in full scale can be supplied to the field tests. The press in Ystad has sufficient capacity also for the large form.

An alternative way of compacting bentonite blocks for emplacement around the canister is cold isostatic compaction. This technique was used in early 1980-ies when blocks were manufactured for field tests in the Stripa project. During 1996 studies have started on evaluating the possible qualities that can be obtained with isostatic compaction in order to compare it with the quality obtained by uniaxial compaction. A first attempt was to analyse samples from still remaining blocks made to the Stripa tests. The result showed that the block was homogeneous and had a high mechanical strength. No discontinuities or defects were observed.

The study further focused on the properties of the bentonite material used and how the bentonite powder could be pre-treated before being compacted, as this may be of importance to the quality of the blocks. The compaction to the Stripa tests was using the MX-80 quality in the condition it was received from the supplier. Tests at Eirich application laboratory in Germany in 1996 have showed that fine bentonite can be converted into a granulated product with round particles and small size range. However, water has to be added and the granules, therefore, have to be dried before compaction. Such material is free floating and has a low proportion of fine material, which is favourable for the evacuation of air during compaction. The granulated product has been used in test with both isostatic and uniaxial compaction with good results. The character of the blocks, however, differ somewhat between the two compaction methods. The uniaxial compaction seems to crush the granules more effectively than isostatic compaction, which implies that uniaxial compaction provides a higher mechanical strength. The study on pre-treatment processes continues.

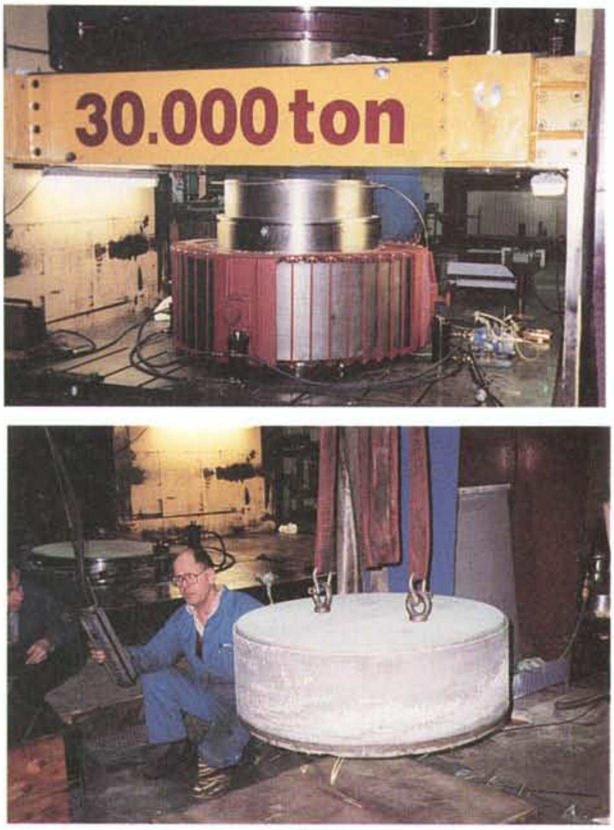


Figure 15.2-3. *Compaction of cylindrical bentonite blocks with the diameter 1 meter in the press in Ystad. The upper picture shows the form and the press and the lower picture shows the compacted block.*

15.2.4 Buffer and backfill handbook

The work with Part 3 of the handbook has progressed during 1996 with the objective of presenting the handbook for international reviewing during 1997. The Part 3 comprises mathematical models, specifically for

- Saturation, swelling and homogenisation.
- Reology.
- Ion diffusion.
- Water transport.
- Gas transport.
- Heat transport.
- Termo-mechanics.
- Mineral alteration.

15.3 GEOSCIENCE

15.3.1 Overview

The geoscientific research at SKB is related to the crystalline bedrock and to the projected 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 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 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. Unoxic (= 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 by processes 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.

The 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 investigation 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.
- Performance of barriers for other waste.

The overall objectives and main activities of the 1996 geoscience programme are expressed in the SKB RD&D-Programme 95. During 1996 the geoscience programme has involved the following main tasks:

- Structural geology and mechanical stability.
- Groundwater chemistry.
- Ability of the rock to limit radionuclide transport.
- Modelling tools and model development.
- The Laxemar Deep Drilling Project.

The following chapters briefly present summaries from these general geoscientific activities some of which were published during 1996.

15.3.2 Structural geology and mechanical 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.

Geological and structural conditions in the bedrock

During 1995 SKB initiated studies on thermal evidences of an extended Devonian molasse sedimentation in Fennoscandia. Mainly apatite fission track analyses was set up and the activities has continued during 1996. The general objective is to increase the understanding of the vertical loadings on the crystalline basement in a geological time perspective. So far fission tracks in basement rocks suggest raised temperatures during the Late Palaeozoic, as high as 125°C at the present level of erosion. Calculations indicate a sedimentary cover thickness of 3±1 km, a thickness which probably varied geographically. It has been proposed that this sedimentary cover was largely composed of molasse, which was eroded from the Caledonides, and deposited into a Caledonian foreland basin.

Mechanical properties of the crystalline basement rock

In collaboration with the Swedish Rock Engineering Research Foundation(SveBeFo) a project was initiated which has focused on the statistical behavior of rock stress measurements. Common statistical tools were used, such as multiple linear regression and variogram analyses in order to describe spatial correlations. Vertical stress and strain data from the Äspö Hard Rock Laboratory were treated. The results show that the variances of vertical stresses are great and that this variance only partly can be explained as a depth dependency. The measured vertical stresses indicate a spatial correlation. The explanation for the great variability is still an open question regarding rock structural differences and/or error in measurements, /15.3-1/. The project will now continue considering the discrepances between results from over-coring versus hydraulic fracturing methods.

The mechanics of rock joints determine to a large extent the behaviour of a jointed rock mass. It is herefore of importance to understand the mechanical behaviour of rock joints to be able to analyse the stability of rock slopes and underground excavations. The mechanics of rock joints was studied in a comprehensive experimental project, /15.3-2/. Shear tests were performed on a number of joint samples with the same topography and this is achieved by using concrete replicas of a natural joint. In these shear tests the normal load conditions and load paths (one way shearing and cyclic) were varied. Also two different kinds of compressive tests were carried out to investigate the normal stress-normal displacement characteristics. In the analysis of these tests, shear stress, dilation and surface degradation were determined. The Coulomb and the empirical version of Ladanyi and Archambault shear strength criteria seem best to fit the shear strength behaviour in this case. From these shear tests it was found that the shear resistance and damage to the joint surface depend of four different mechanisms: dilation and climbing over asperities, breakage of asperities, transportation of gouge material and reattachment of gouge material. It is also indicated from the recorded acoustic emission measurements that different failure modes occurred during the shear tests.

Geodynamic and mechanical processes

In Japan the Power Reactor and Nuclear Fuel Development Corporation (PNC) is carrying out research into the coupling between earthquakes and mechanical impacts on underground facilities. Furthermore changes in groundwater pressure and chemistry are studied according to seismic events. SKB has an agreement with PNC on exchange of information on these subjects. PNC is conducting in-situ measurements at the Kamaishi research mine in the northern part of the main island, Honshu. The project is still in a data collecting phase and the preliminary conclusions which were presented in SKB Annual Report 1995 have not changed. In summary the investigations say that ground motion decreases with depth and that 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.

The shore level displacement in Fennoscandia is mainly due to two co-operative vertical movements, the glacio-isostatic uplift and the eustatic sea level rise. The cause of the difficulty of modelling the shore level displacement has been the lack of empirical data of the glacio-isostatic uplift but also the lack of reliable data of the eustatic rise. However, by an investigation of the so called lake-tilting phenomenon the course of the glacio-isostatic uplift has been discernible. By this knowledge it has been possible to start an iteration process for detailed estimations of the glacio-isostatic uplift and the eustatic rise using empirical data of the shore level displacement /15.3-3, -4/. The modelling also includes information regarding the recent relative uplift and rough information concerning the eustasy. The shore level displacement is empirically known from 63 shore level curves in the area affected by the Scandinavian ice during the Late Weichselian. The model indicates that there are two mechanisms involved in the glacio-isostatic uplift, one slow and the other fast. The main uplift, still in progress, acts slowly. The shore level modelling shows that the slow uplift is in progress with a declining course in the whole area earlier covered by the Scandinavian ice. The peripheral parts of this area, which today seem to submerge, are still affected by a slow uplift. The reason for the submergence is an ongoing eustatic rise. The fast mechanism gave rise to a crustal subsidence during the Younger Dryas restored by a fast uplift during the Preboreal. The subsidence is apparent in shore level curves in a peripheral zone outside the Younger Dryas ice margin. The fast uplift is apparent in the same area but also in central Fennoscandia i.e. the area which were deglaciated rapidly during Preboreal time. The future development regarding the glacio-isostatic uplift, the eustasy and the shore level displacement can be predicted in Fennoscandia using the results from the modelling.

Another project has focused on describing different space geodetic measurements of relative point positioning over distances ranging from tens to thousands of kilometres, /15.3-5/. By means of these new techniques it is possible to trace plate tectonic motions and also detect strain patterns within continents. The SWEPOS system consists of permanently operating GPS (Global Positioning System) stations in Sweden. The system has been designed, devised and furnished as a joint effort between the National Land Survey of Sweden and the geoscience group at Onsala Space Observatory, Chalmers University of Technology. In the SKB report the operations within SWEPOS are described emphasizing the possibilities to detect crustal motions in Fennoscandia. A separate project named BIFROST has been created at Onsala Space Observatory. BIFROST stands for Baseline Inferences for

Fennoscandian Rebound Observations, Sea-level and Tectonics. It combines the efforts of a number of investigators at different sites and contributes to a number of international research programs in geophysics and geodesy. The project group intend to run the BIFROST project of at least ten years if deformation rates of 0.1 mm/yr are to be concluded at a 95 percent confidence level. First results (2,5 years) indicate movements which generally support the notion of a dominating displacement pattern due to the postglacial rebound of Fennoscandia. Relative horizontal movements along the baselines are also indicated.

15.3.3 Groundwater chemistry

Geochemistry 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.

Much of the work related to hydrochemistry is done within the different on-going experiments in the Äspö Laboratory. It is described in its context in Chapter 17. Some general conclusions are summarized here:

The evaluation of the hydrochemistry of Äspö has resulted in a completely new modelling concept called M3, Multivariate Mixing and Mass-balance calculations. The calculations start with a multivariate principal component analyses. Through this mathematical operation a new variable is computed, which is a linear combination of all parameters. This is the first principal component. The second principal component is the best combination of the variation which is not described by the first principal component etc. By using the first two principal components 72% of the variability of the Äspö data is described. The next step is to present the datapoints in the plane of the first and second principal component. Extreme points are defined as reference waters and their origin is related to an end-member. The end-member is the original source for that reference water. Through the mixing calculation all the other observation points are described as proportions of the different reference waters. The distance from an observation point to the reference waters defines uniquely the proportion of each reference water. When the mixing proportions are defined a theoretical composition is calculated. The difference between the calculated and the observed composition is resulting from reactions which have either increased or decreased the concentration of one or several constituents. Figure 15.3-1 illustrates the M3 modelling concept. The M3 modelling has been used for evaluation of hydrochemistry at Äspö and Laxemar /16.3.26 and 16.3.28 i Annual 95/. An extensive use of the modelling concept is made in the reporting of the Äspö pre-construction and construction phase investigations.

Modelling of groundwater flow during the melting of a glacier indicate that meltwater could be driven down to a depth of more than 1000 m due to the driving force of a hydraulic head defined by the thickness of the ice. These indications are apparently in accordance with hydrochemical observations of cold climate signatures at depth of 800 m at Laxemar. However, this might not be comparable. There are signs of oxidation caused by infiltrating groundwater to depths of 200 m at the most. Hydrochemical data also point out that the main proportion of meltwater from the latest glaciation is located between 100 and 300 m depth. Therefore is seems more likely that the cold climate water found at depths of more than 500 m was recharged at some earlier time than the latest glaciation. These conditions seem to be the same also at the Finnish investigation sites.

15.3.4 Ability of the rock to limit radionuclide transport

A thorough 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.

Groundwater flow and advection

The flow of groundwater at great depth in the crystalline bedrock, is mainly governed by the large scale properties. It is generally accepted that the topography represents the large scale gradients for the present day flow in these low-conductive geological formations. The topography of Sweden is characterized by the Caledonian mountain chain, which runs through a large part of the Scandinavian peninsula, and by regular large ridges and vallies that are almost perpendicular to the mountain chain. The conductivity of the bedrock decreases with the depth, and the characteristic length of this variation is much less than the length scale of the topography. A separate project was set

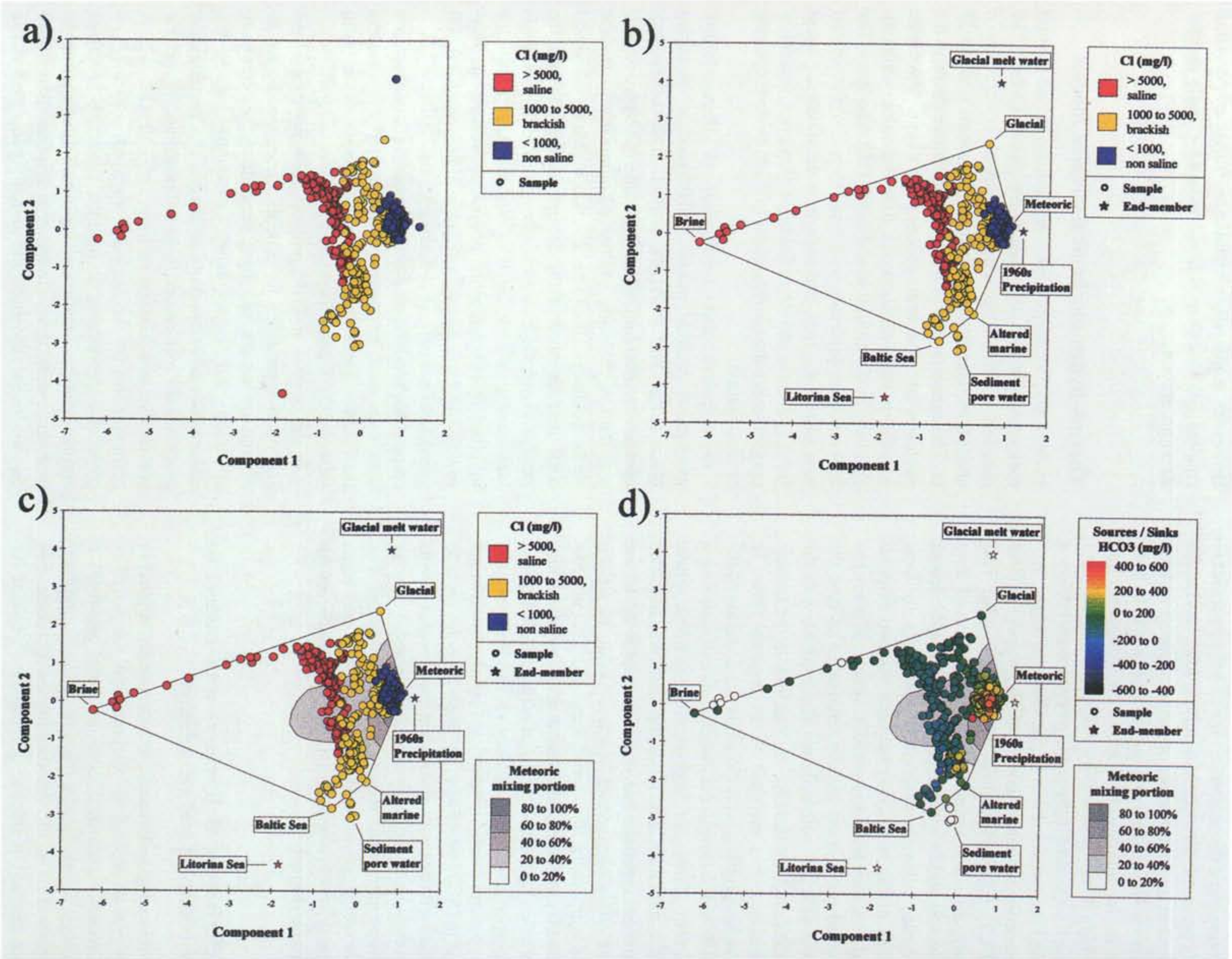


Figure 15.3-1. *Stepwise evaluation of hydrochemical data by means of the M3 modelling concept, a) principal component analysis is used to obtain the maximum resolution of the data set, b) selection of end-members and reference waters, c) mixing calculations and d) mass balance calculations.*

up studying the large scale groundwater flow in the context of topographical governed gradients /15.3-6/. The analysis, given exclusively in closed form, shows that the flow takes place in the uppermost 500 m of the bedrock, which is about the depth of a future repository. The analysis also shows that the residence time of a fluid particle is inversely proportional to the hydraulic gradient and to the ratio between the length scale of the topography and the conductivity. The residence time also depends on the ratio between the lengths that characterize the topography and the variation of the conductivity with depth. The shortest residence time occurs if the repository is located below a valley, and the longest if it is located under a ridge.

In connection with the final repository for spent nuclear fuel, (SFL 2), another repository will be constructed. This repository, SFL 3-5, is for the final deposition of long lived, low and middle active nuclear waste. For the design and for the safety analysis it is necessary to have knowledge of the interaction between the two disposal sites and how they form an hydraulic system. A study has been carried out to gain generic knowledge of how the repository SFL 3-5 will hydraulically interact with the surrounding rock mass, /15.3-7/. The flow in the tunnels was calculated as a multiple of the regional flow. To overcome the problem with the unknown direction of the regional flow, the flow in the tunnels was calculated for a large number of directions representing the whole sphere of possible directions. The modeling study was based on the continuum approach and was implemented by the use of a finite difference method. The flow inside the repository will vary along the tunnels and the size of flow depend on lay-out, direction, length and size of the tunnels, direction and size of regional groundwater flow, as well as properties of filling and barriers inside the tunnels. From a general point of view, and as the tunnels are located in the horizontal plane, the largest average specific flow inside a tunnel will occur when the regional flow is in the horizontal plane and directed in an angle close to the direction of the longest part of the studied tunnel. The smallest average specific flow inside a tunnel occurs when the regional flow is directed at a right angle to the length of the studied tunnel.

Characterization of fractures with respect to nuclide transport and retardation

Image processing techniques are increasingly applied in different type of geological interpretations. On behalf of SKB a recently published doctoral thesis presented a methodology for evaluating fracture apertures and their morphology /15.3-8, -9/. Another follow-up project has now been initiated in order to develop this technique for images recorded by a borehole TV-logger, the so called BIP system. It would be of great advantage if fractures could be assessed from boreholes in-situ in terms of aperture, roughness, infillings, etc.

15.3.5 Modelling tools and model development

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.

Thermo-hydro-mechanical coupled models

In the KBS-3 concept 4 500 canisters are buried in solid rock at a depth 400-700 m below the ground surface. The canisters are emplaced in parallel tunnels at a spacing of about 6 m. The distance between the tunnels is about 25 m. The canisters emit heat due to radioactive decay in the nuclear waste. The heat sources from all canisters create a complex three-dimensional, time-dependent temperature field in the ground in and around the repository. The heat emission decreases with time. However, the emitted heat warms the rock and induces a thermoelastic stress field. The stress and strain fields are of interest, since they may influence the conditions for fracture closure, opening or propagation.

Development and verification of thermo-hydro-mechanical coupled models (THM-models) is taking place in the DECOVALEX project, (international co-operative project for de DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation). DECOVALEX was initiated by SKI and SKB is a member of the Steering Committee for the project. A second three-year period of DECOVALEX is now running. Within the DECOVALEX II project SKB is involved in evaluating the coupled effects on the Borrowdale Volcanic Group in response to the so called RCF3 pump test (test case 1) at Sellafield, UK. NIREX, UK is responsible for defining this test case and also delivers the data bases used. Furthermore SKB is deeply involved in the coupled THM experiment at Kamaishi mine, Japan (test case 2), accomplished by PNC.

In the context of THM projects SKB emphasizes the analytical approaches for a better understanding of the calculation results and their dependence on boundary conditions and dimensionality. A theoretical study was initiated at Lund University, to analyse the thermoelastic process in the rock caused by heat sources in a site specific perspective as a function of time, /15.3-10, -11, -12/. A particular aim for the analytical approach was to gain a physical understanding to quantify particular processes and their interactions. Exact analytical solutions for the time-dependent, three-dimensional responses have been derived for the repository and for a finite line source. The solutions may be used as boundary conditions in numerical modelling of the local processes around a canister. The presented solution could also be used to verify numerical models.

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 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.

The palaeohydrogeological progamme's implications for safety assessment has been compiled in a progress report /15.3-13/. The report presents in a qualitative manner time sequences, scenarios and boundary conditions due to long-term climate change on subsurface conditions.

During 1996 scooping calculations has been done in order to predict the infiltration depth and mixing of subglacial melt water and to develop methods for determining age and mixing of waters from different origin /15.3-14, -15/. There has also been some evaluation of the technique of embedded grids to couple a regional model (length scale ~10 km) to a site scale model.

All results from the scooping calculations ought to be regarded as tentative, as a number of simplifying assumptions were made. It is however clear that the flow and salinity distributions are sensitive to a number of influences like shape of ice front, initial salinity distribution and the conductivity field.

15.3.6 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 was carried out in the Laxemar area near the Simpevarp peninsula in the municipality of Oskarshamn.

In 1996 the hydraulic testing programme was finalized and rock-mechanical aspects was emphasized. Rock stress measurements were performed down to 1400 m depth with the hydraulic fracturing method. The indents in the rock walls are at the moment being oriented by means of packers and the BIP system. In-situ measurements of normal stiffness was also done /15.3-16/.

15.3.7 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 the authorities and different experts in the broad field of geoscience:

- Thermo-elastic analyses of the KBS-3 concept.
- Ongoing investigations of land uplift – results and methods.
- Regional groundwater flow in Swedish basement rock – an analytical solution.
- Glacial induced stresses and fractures.

During 1996 SKB Geoscience programme was involved in the organisation of two international conferences:

- Workshop on the Impact of Climate Change & Glaciations on Rock stresses, Groundwater flow and Hydrochemistry – past, present and future /15.3-17/.
- SWIM 96 – 14th Salt Water Intrusion Meeting /15.3-18/.

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).

15.4 CHEMISTRY

15.4.1 Solubility and speciation of radionuclides

Solvent extraction is one technique among others to determine the constants for speciation of radionuclides in solution. An advantage with this method is the possibility to work with low concentrations which are typical for radionuclide dissolution in groundwater (trace concentrations <10^{-7} moles/l). An evaluation of solvent extraction for measurements on actinides has been made /15.4-1/. It was concluded that a careful application of the extraction technique yields results as reliable as with any other method.

The kinetics of reduction of technetium and neptunium in groundwater has been thoroughly investigated and presented by Daqing Cui in his Doctoral Thesis, presented at the Royal Institute of Technology, Dept. of Nuclear Chemistry, Stockholm /15.4-2/. It was convincingly demonstrated that sorption on solid surfaces are needed to mediate the redox reactions. Pertechnetate and neptunyl ions are reduced to Tc(IV) and Np(IV) by divalent iron on mineral surfaces /15.4-3/. Ferrous iron in solution, Fe^{2+}, can also act as a reductant but it has to be sorbed on solid surfaces to be effective. The reaction is important to performance assessment because the reduced tetravalent forms of technetium and neptunium are effectively retained by sorption in contrast to pertechnetate and neptunyl which are much more mobile. The difference in element solubility due to redox state is about six orders of magnitude which, in practise, is the difference between "soluble" and "insoluble". This is important for performance assessment of waste containing radionuclides of these elements.

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 elements which are relevant to performance assessment. In particular, support has been given to the production of a comprehensive OECD/NEA report on thermodynamic modelling in the field of radioactive waste management.

15.4.2 Retention by sorption and diffusion

Fundamental studies of sorption are supported by SKB, for example, sorption of Pm, Th, Cs, Co, Y and Na on titanium dioxide surfaces. A study of uranyl sorption on aluminium oxide, in co-operation with Los Alamos National Laboratory in USA, has been completed and presented /15.4-4/. Uranium was found to form a so called inner sphere bidentate complex with the aluminium oxide surface. The same sorption reactions are expected on iron oxide surfaces. Surface complexation is the generally used model to describe sorption in detail; the corresponding surface constants are determined by titration, see Table 15.4-1. Surface complexation is valuable as a method to demonstrate our understanding of the processes which govern sorption of radionuclides on mineral. It can also be used to find sources of uncertainties. However, as illustrated by this example, it is not practical to use surface complexation data as a substitute for Kd-values in the calculations of radionuclide transport in performance assessment, see Table 15.4-1.

In addition to sorption studies of the redox sensitive elements Tc and Np, Daquing Cui, in his Doctoral Thesis /15.4-2/, also made experiments with redox stable elements Sr, Co and Cs on fracture filling minerals from granitic rock (Stripa granite). It was concluded that sorption of Sr^{2+} and Co^{2+} are reversible; Sr^{2+} is sorbed by cation exchange and Co^{2+} by inner sphere surface complexation. Sorption of Cs^{+} is partially irreversible as it substitutes potassium in the mineral structure.

Various experimental activities have been started with the aim to improve methods to measure ion diffusion in the connected micropores of crystalline rock, i.e. matrix diffusion. Special emphasis is laid on electrical conductivity measurements. A development and verification of this method would supply faster means for diffusion measurements in low porosity rock, where, in controlling the composition of pore solutions, diffusion of selected ionic species can be studied. This would also give the possibility to study diffusion under different conditions but in the same sample. Conventional through-diffusion experiments with Sr^{2+} and Na^{+} have also been started. Some of these are made on the same samples as have been used for electrical conductivity measurements in order to verify that method. Comparisons are also made with the gas diffusion method, developed at the University of Jyväskylä in Finland, which is another fast and efficient method to measure matrix diffusion properties.

Table 15.4-1. Reactions and constants for surface complexation of uranyl ions on alumina (Al_2O_3) /15.4-4/.

Reaction	Constant		Ref
Acid dissolution of alumina surface			
$SOH + H^+ \Leftrightarrow SOH_2^+$	$pK_{a1}^{int} = 7.4$		15.4-4
$SOH \Leftrightarrow SO^- + H^+$	$pK_{a2}^{int} = 11.3$		15.4-4
$SOH_2^+ + A^- \Leftrightarrow SOH_2^+ A^-$	$pK_{A-}^{int} = -1$		
$SO^- + Na^+ \Leftrightarrow SO^- Na^+$	$pK_{Na+}^{int} = -1.2$		15.4-5
Aqueous uranium speciation	*Original*	*Adjusted*	
$UO_2^{2+} + H_2O \Leftrightarrow UO_2(OH)^+ + H^+$	-4.9	-4.9	15.4-6
$UO_2^{2+} + 2H_2O \Leftrightarrow UO_2(OH)_2 + 2H^+$	-9.7	-11.4	15.4-6
$UO_2^{2+} + 3H_2O \Leftrightarrow UO_2(OH)_3^- + 3H^+$	-18.3	-21.5	15.4-6
$UO_2^{2+} + 4H_2O \Leftrightarrow UO_2(OH)_4^{2-} + 4H^+$	-31.8	-34.5	15.4-6
$2UO_2^{2+} + H_2O \Leftrightarrow (UO_2)_2(OH)^{3+} + H^+$	-2.4	-2.4	15.4-6
$2UO_2^{2+} + 2H_2O \Leftrightarrow (UO_2)_2(OH)_2^{2+} + 2H^+$	-5.02	-5.02	15.4-6
$3UO_2^{2+} + 4H_2O \Leftrightarrow (UO_2)_3(OH)_4^{2+} + 4H^+$	-10.7	-10.7	15.4-6
$3UO_2^{2+} + 5H_2O \Leftrightarrow (UO_2)_3(OH)_5^{2+} + 5H^+$	-14.05	-14.05	15.4-6
$3UO_2^{2+} + 5H_2O \Leftrightarrow (UO_2)_3(OH)_5^{2+} + 5H^+$	-14.05	-14.05	15.4-6
$3UO_2^{2+} + 7H_2O \Leftrightarrow (UO_2)_3(OH)_7^- + 7H^+$	-28.9	-28.9	15.4-6
$4UO_2^{2+} + 7H_2O \Leftrightarrow (UO_2)_4(OH)_7^+ + 7H^+$	-19.8	-19.8	15.4-6
Surface complexation			
$SOH + UO_2^{2+} \Leftrightarrow SOUO_2^+ + H^+$	0		15.4-4
$2SOH + UO_2^{2+} \Leftrightarrow (SO)_2UO_2 + H^+$	-2.1		15.4-4

Constants used			
Inner layer capacitance	1.1 C/m^2	Site density	10 sites/nm^2
Outer layer capacitance	0.2 C/m^2	pKw	13.7
Surface area (BET)	142.9 m^2/g	Ionic strength	0.1 M

Table 15.4-2. Diffusivities of Cs^+, Sr^{2+} and I^- in compacted bentonite clay (MX-80, Wyoming, dry density 1.8 g/cm^3), and simulated normal (Allard) and saline (NASK) groundwater /15.4-8/.

Ion	Allard water, I^a= 0.018 M		NASK water, I^a= 0.218 M	
	D_a m^2/s	D_e m^2/s	D_a m^2/s	D_e m^2/s
Sr^{2+}	$1.2\ 10^{-11}$	$6.7\ 10^{-9}$	$1.0\ 10^{-11}$	$6.0\ 10^{-10}$
Cs^+	$1.0\ 10^{-12}$	$6.4\ 10^{-10}$	$2.3\ 10^{-12}$	$3.5\ 10^{-10}$
I^-	$3.5\ 10^{-11}$	$2.1\ 10^{-13}$	$9.2\ 10^{-11}$	$7.0\ 10^{-12}$

[a] Ionic strength in moles/l (M).

Last year (1995) experimental diffusion data were compiled and reviewed /16.4-7/. An attempt is now made to extract representative data that can be recommended for use in the safety assessment study SR-97. However, the wide variations in values due to, for example, inhomogeneities and not fully known experimental conditions often leads to contradictions when trying to make comparisons between results. Another approach has instead been used which is based on a reference diffusing species, unaffected by the chemistry and electrical charge of the surroundings. Other species can then be related to the reference species from knowledge of their behaviour in a specific surrounding. This supplies a measure which qualitatively, and realistically, predicts the diffusive behaviour of a species.

Diffusion of Cs^+, Sr^{2+} and I^- in compacted bentonite clay has been measured /16.4-8/, see Table 15.4-2. It was found that I^- diffusion is hindered by anion exclusion by

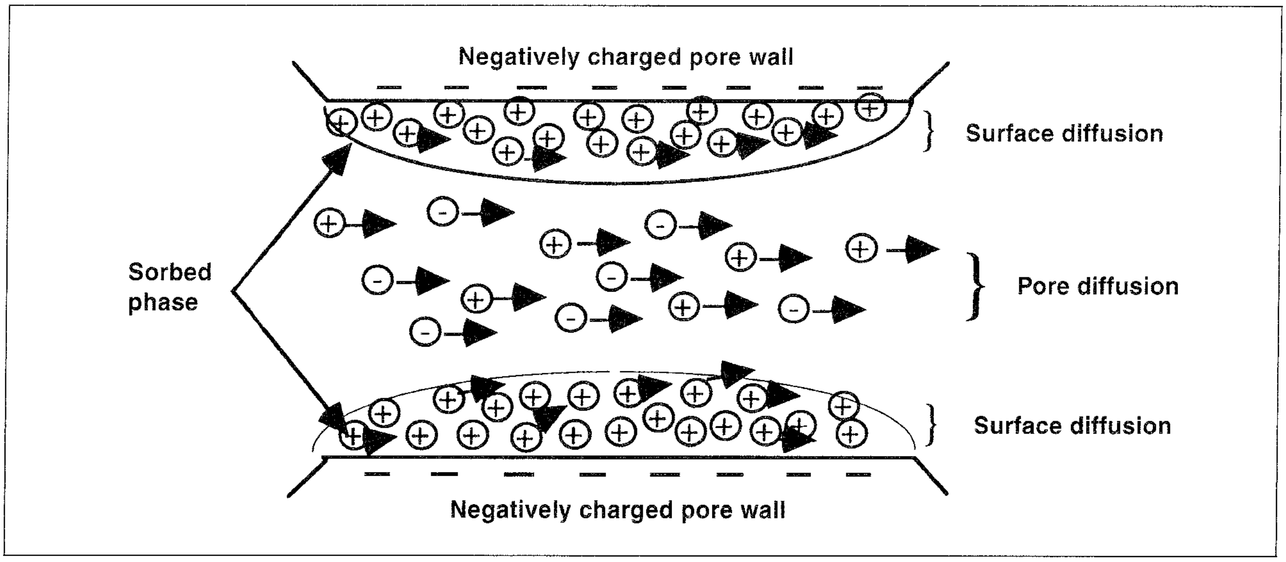


Figure 15.4-1. *Schematic description of ions diffusing in a pore with negatively charged pore surfaces. Anions are restricted to a smaller pore volume due to electrostatic repulsion. Cation exchanging cations concentrate in the sorbed, but mobile phase. (Illustration according to Neretnieks and Ohlsson, Dept. of Nuclear Engineering, Royal Institute of Technology).*

the negatively charged montmorillonite mineral particles. Diffusion of Cs^+ and Sr^{2+}, on the other hand, was enhanced by their mobility in the concentrated layer of cations at the negative surfaces, see Figure 15.4-1. This phenomenon is referred to as surface diffusion. A new cell for diffusion measurements was developed which turned out to be very convenient and will be used in future investigations.

Diffusion and sorption properties of radionuclides in bentonite are compiled and relevant values are selected for the safety assessment study SR-97. This will presumably be reported next year.

15.4.3 Influence of humic substances, colloids and microbes

Capillary zone electrophoresis has been developed as a method to analyse humic substances /15.4-9/. Potentiometric titration has been used as a method to characterise natural organic acids in different groundwater environments /15.4-10/.

Investigation of colloids and microbes have been concentrated to Äspö and the analogue sites. For example, bacteria, colloids and organic carbon have been sampled at the Bangombé site in the Oklo area /15.4-11/. Dissolved organic material and colloids are present in low concentrations and not much different to what can be found at any other location at similar depths. It is difficult so see how these aggregates could be of any major importance to radionuclide transport in the deep subsurface environment. The cations calcium, magnesium and sodium in groundwater tend to destabilise colloids. The situation is of course different in the dilute aquifers at the surface where the concentrations of colloids and humic substances are high and, therefore, important for the solubility and transport of trace metals.

Bacteria in Bangombé were found to vary considerably between boreholes; a result that probably reflects differences in the geochemical environment /15.4-11 and 15.4-12/. Similarities could, possibly, be used to indicate strong hydraulic connections between separate boreholes.

Investigations of subsurface micro-organisms at Stripa and Äspö have been summarised by Susanne Ekendahl in her Doctoral Thesis, presented at Göteborg University, Dept. of General and Marine Microbiology /15.4-13/. It was concluded that *bacteria* dominates with signs of *archaea* and some observations of *fungi*. Deep granitic groundwater environments seems to contain truly functional ecosystems and the biosphere apparently extends down into the geosphere. Much of the sampling was made in boreholes at Äspö, especially along the entrance tunnel, see Figure 15.4-2 and Table 15.4-3. Laminar flow reactors were connected to some of the boreholes in order to catch and grow viable bacteria /15.4-14/, see Table 15.4-3. Methods were also developed to use nucleic acid probes to detect specific microbes on mineral surfaces /15.4-15/.

Indications of fossil bacteria have been found by electron microscopy in calcite minerals, precipitated in water conducting fractures sampled at the 200 m level under Äspö. This has not been reported yet.

Microbiological investigations became a part of our investigations of the deep subterranean environment in 1987. It was found that bacteria exist at deep levels and that they can influence the geochemical environment. The state-of-the-art of these investigations in relation to performance assessment were reported last year. A mini-review of this report has been published this year (1996)

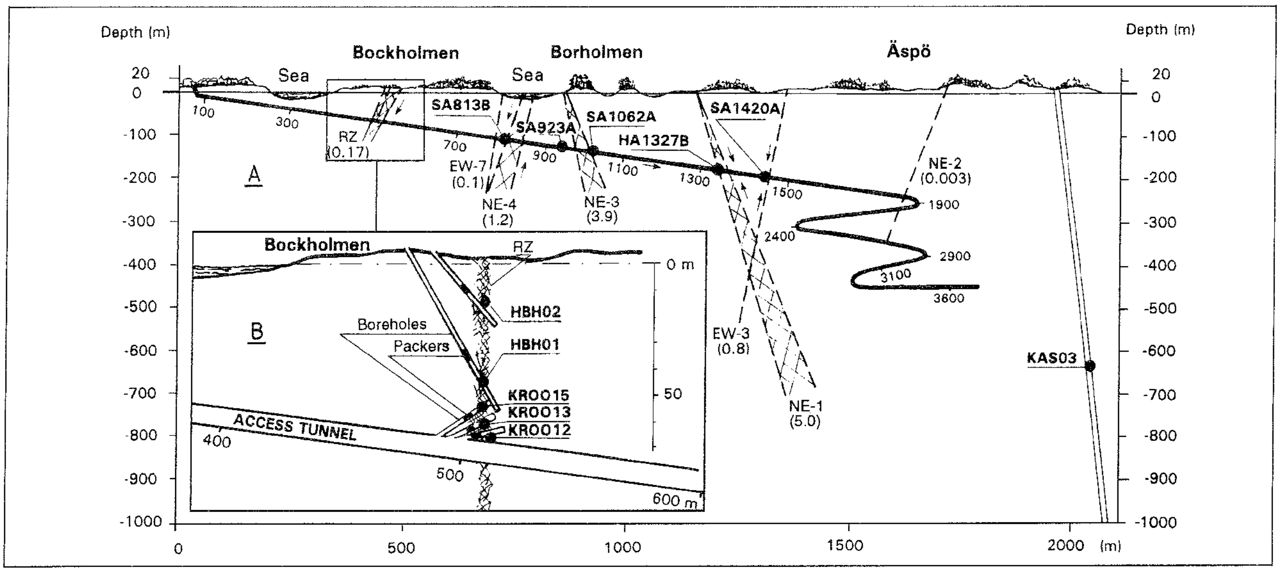


Figure 15.4-2. Points • in Äspö HRL where bacteria have been sampled in boreholes /15.4-14/. The result is presented in Table 15.4-3.

Table 15.4-3. Borehole information with groundwater chemistry data, the total number of attached and unattached bacteria (2 February 1993) and the number of cultivable SRB in groundwater from the Äspö Hard Rock Laboratory tunnel /15.4-14/.

Borehole	Depth (m)	Clone letter [a]	Bacteria ($ml^{-1} \times 10^5$)	Bacteria ($cm^{-2} \times 10^5$)	SRB (ml^{-1})	pH	HCO_3^- (mM)	Cl^- (mM)	SO_4^{2-} (mM)	DOC ($mg\ l^{-1}$)	Drilling water (%)
HBH02	10	l	9.3	– [b]	< 5 [c]	6.4	1.04	0.260	0.15	21	–
HBH01	40	m	3.4	–	< 5	7.5	4.70	18.4	1.34	15	–
KR0012	68	n	1.3	–	< 5	7.7	4.98	23.7	1.42	15	< 0.2 [c]
KR0013	68	o/g	5.1	4.7	< 5	7.5	4.90	50.5	1.45	15	< 0.2
KR0015	68	p	1.5	–	< 5	7.5	4.96	22.3	1.38	22	< 0.2
SA813B	112	q/h	1.4	12.0	2420	7.0	6.64	94.7	2.36	11	< 0.2
SA923A	134	–/i	–	–	< 5	6.8	10.5	121	1.33	9.3	< 0.2
SA1062A	143	r	–	–	–	7.3	6.46	123	1.97	9.1	< 0.2
HA1327B	179	s/j	0.88	5.6	130	7.1	4.48	130	2.12	6.1	< 0.2
SA1420A	192	t/k	0.44	0.15	< 5	7.3	3.61	97.3	3.49	7.5	< 0.2
KAS03	626	u	0.84	–	1390	7.1	0.78	99.5	1.83	1.0	0.3

[a] Letter(s) given to the clone group names to show in which borehole they appeared, water / surface.
[b] No data.
[c] Detection limit.

in a scientific paper /15.4-16/. Investigations now are directed towards methanogenes which seems to play a key role in the deep subsurface geosphere. Efforts are therefore made to improve the quality of sampling and analysis of gas in groundwater.

Survival of sulphate reducing bacteria, SRB, in compacted bentonite has been studied. It was demonstrated that the low chemical activity of water in highly compacted bentonite is lethal for sulphate reducing bacteria /15.4-17/. For example, pure water has the activity of 1 and water in the sample of confined compacted bentonite, (wet density 2 g/cm^3), had the activity 0.96. Two strains of SRB, isolated from deep Äspö groundwater, had been added and both were found to be 100 % non-viable after 1 day.

15.4.4 Validation experiments

Microbial analysis was made of the sand/bentonite backfill in the Canadian Buffer/Container Experiment, BCE, at AECL's Underground Research Laboratory. The BCE had been in operation for 2.5 years when it was decommissioned in May 1994. It had been designed as an engineering test with essentially non-microbial objectives and no microbial analyses were performed at the start of experiments. However, it became apparent that BCE could be used to provide valuable data on microbial survival in buffer materials under realistic conditions. An international microbial study of BCE was initiated by AECL and supported by AECL, ANDRA (France) and SKB. The study was co-ordinated by AECL and the results have been reported through the three participating organisations /15.4-18/. An important observation was that viable micro-organisms had disappeared near the heater surface, where the moisture content was low . This was an early indication that microbes are killed by the low water activity in backfill material which has been important as an inspiration for further experiments in that field. The backfill of BCE mainly consists of a 50/50 mixture of sand and bentonite with a 5 cm layer of pure sand next to the heaters. The in-situ temperatures at the sampling positions varied from 45 to 65°C (with some exceptional samples on the heater clothing at 85°C). The experiment was conducted at the 240 m level in URL where the ambient temperature is about 17°C.

Cement is used in underground construction for concrete structures, pavements, grouting and shotcrete on the tunnel walls. Standard Portland cement concrete has a high pH pore water due to portlandite ($Ca(OH)_2$) and alkali hydroxides (NaOH and KOH). The pH is high enough to weather, for example, feldspars and silica. Tests are therefore made on the rate of high pH weathering reactions. Columns, containing crushed samples of potential host rock, are percolated with simulated cement pore solutions and the products formed are analysed. The experiments are carried out by British Geological Survey, jointly supported by Nagra, Nirex and SKB. A first phase of experiments with pure minerals has been accomplished and the second phase with rock samples is planned to run until August 97.

In addition to the above mentioned studies, the following three laboratory validation experiments were continued during 1996: 1) Radionuclide migration in overcored rock fractures; 2) simulation of near-field release from a clay buffer to a rock fracture and; 3) colloid migration in a clay buffer.

15.5 NATURAL ANALOGUE STUDIES

15.5.1 Natural analogues and performance assessment

Most of the analogue studies are performed as jointly supported projects to share the costs and the large amount of work that has to be performed. Numerous measurements have to be done in the field, a large number of different samples have to be taken and subsequently analysed in the laboratories, and different branches of science have to be involved in the investigations and in the evaluation of the results. Support and participation from organisations in different countries add different views to the study and promote a critical review of the results which is of great value. The *project* is an efficient way to organise an analogue investigation but it necessarily invokes time limits. Sampling and analyses have to be done with great care so, not seldom, the subsequent evaluation phase has to be forced and kept very strict to the original aims. It could therefore be worthwhile to revisit some of the accomplished studies. An example of that is Poços de Caldas where previously obtained trace element data have been evaluated with new solubility models /15.5-1/. Traditionally solubility limits are calculated assuming the radionuclides in equilibrium with pure solid phases but analysis of trace elements in nature demonstrates that incorporation in other solid phases frequently lowers the solubility levels. Revisiting Poços de Caldas data on trace elements demonstrated that better results were achieved when it was assumed that U, Zn and rare earth elements occurred as co-precipitates of ferrihydrite, and that Sr is controlled by co-dissolution of strontium rich fluorites. The calculated concentrations are 2-4 orders of magnitude lower than conservative solubility limits based on pure solid phases of the trace constituents. The same should apply to radionuclides of the same elements.

Another example of a revisited analogue study is Cigar Lake. The AECL/SKB project was carried out from 1989 to 1993. We have since then allocated time and resources to reapraise available data and the main results were /15.5-2/:

- A more realistic radiolysis model was developed and tested.
- Considerable progress was made in understanding and modelling the hydrothermal formation of the deposit.
- The physical properties of clay as a potential buffer to groundwater flow and radionuclide migration were addressed.

The models developed to calculate water radiolysis were realistic to conservative and provide a step forward as compared to earlier overconservative calculations used in performance assessment /15.5-2 and 15.5-3/. An important observation is that conditions can remain reducing if radiolysis rate is as low as demonstrated in Cigar Lake. In this case oxidants are being consumed close to the ore and the groundwater remains reducing. A similar situation at exposed spent fuel would prevent the build-up of a redox front and thereby simplify the chemical processes to be considered for radionuclide release.

The CHEMSAGE computing package /15.5-4/ had been modified to the geochemical version CHEMGEO /15.5-5/, designed to model hydrothermal processes. It

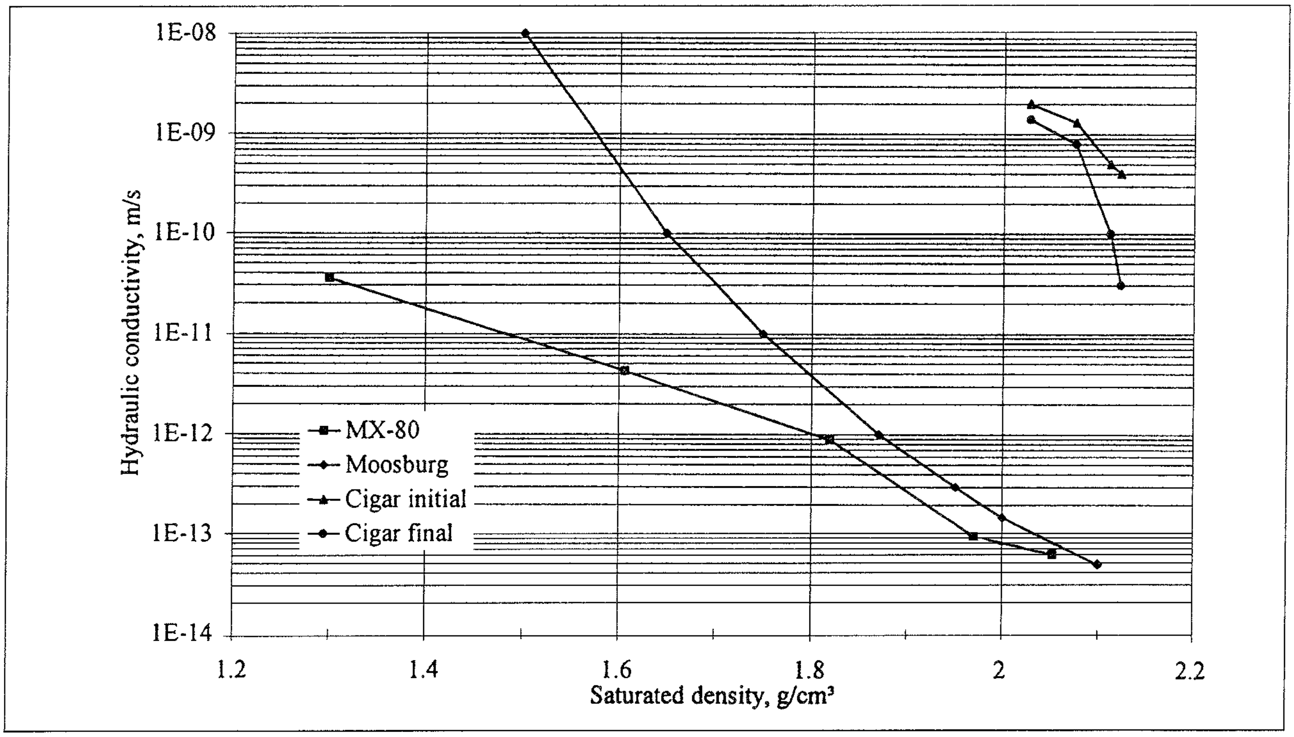


Figure 15.5-1. *Hydraulic conductivity of the Cigar Lake material and two commercial bentonites versus clay density at saturation /15.5-6/.*

was successfully applied to the Cigar Lake deposit to simulate the following conditions there.

1. The composition of the basement hydrothermal fluids.
2. The alteration mineralogy resulting from hydrothermal interaction of the basement rock units.
3. The composition of the sandstone brines.
4. The formation of the uranium orebody by mixing hydrothermal basement fluids with the sandstone brines.
5. The alteration mineralogy around the orebody resulting from the mixing of hydrothermal fluids and the sandstone brines.
6. The controlling factor of the clay phases in the formation of the orebody and the surrounding alteration haloes.

The results were used to distinguish the more recent low temperature processes and it was shown that the hematite concentrations adjacent to the orebody/clay interface is likely to have been formed by hydrothermal reactions, and not, as previously anticipated, by radiolysis oxidation. Graphite has been an important component in origin of and evolution of the Cigar Lake uranium deposit. This may explain why no uranium ore concentrations were found at the nearby Close Lake which has an otherwise similar hydrothermal development.

Cigar Lake clay was compared to commercial bentonites and it was concluded that, whilst exhibiting inferior physical qualities, see Figure 15.5-1, it has provided an efficient sealant to groundwater flow and to radionuclide dissolution and migration over repository timescales. Physically, this is demonstrated by the large differences in hydraulic conductivity and the calculations and observations of radionuclide transport in the near-field of the orebody. Chemically, the redox buffering capacity of Cigar Lake clay has prevented dissolution and migration by oxidation.

15.5.2 Oklo

Until 25 years ago, it was thought that all uranium found in natural occurrences had the same isotopic ratio of $^{235}U/^{238}U$. Then, in 1972, during routine quality control analyses in a uranium processing plant, uranium with a lower than normal isotopic ratio was discovered. The sample came from a mine at Oklo, Gabon, which contained several pockets of extremely high grade ore with 20% or more U. More detailed examination of the uranium samples showed that they also contained other elements that were unusual and that these elements were present in the proportions expected to result from fission of ^{235}U. The evidence was undeniable that the ore deposit, which had formed about 2000 million years ago, had undergone a period of naturally sustained fission reactions.

Early investigations at Oklo centred on what is now called Reactor Zone 2. This was the ore region located in the open pit mine under exploitation in the early 1970s. Fortunately, this zone turned out to be one that had preserved its history well /15.5-7/. Even at the early stages of investigation, suggestions were made to evaluate Oklo as an example of radioactive waste isolation.

Oklo project, Phase 1

Over the years, mining at Oklo exhausted the ore in the open pit and was continued underground. New pockets of ore with evidence of ancient fission reactions were found, in this case well away from surface weathering conditions, and possibly more similar to a repository situation than that of the originally discovered reactor zones. In 1991, a project "Oklo, Natural Analogue for Radioactive Waste Repository" was started by the French CEA, with financial support from the EU. Active collaboration was established with SKB and other organisations that were not members of EU. The objectives were to:

- Investigate and describe the most recently discovered zones at Oklo.
- Study the migration of fission products from the reactor zones both due to ancient events and to recent phenomena.
- Carry out mathematical modelling of the present water and solute transfer pathways from the reactor zones to the surface discharge areas.

Reactor Zones 10 and 13 were characterised from the underground workings, and regional scale and local scale hydrologic models were developed for Okelobondo, just south of Oklo, and Bangombé, about 20 km to the south. The Bangombé locality has been of particular interest to SKB because it is now located near the surface, making it easy to access and sample, as well as allowing us to examine the effects of exposure of the ore to oxidising conditions in a near-surface environment. This project came to a successful conclusion in 1995 and the final reports were issued in 1996.

Oklo project, Phase 2

Near the end of the planned work period of the CEA project, Sweden joined the EU, making it possible for SKB to join CEA, ENRESA and ANDRA in development of a focused continuation of the work at Oklo. EU approved the proposal and work on a 3 year project (Phase 2) began officially in June, 1996. The mining company responsible for exploitation of the ore at Oklo has decided to mine out remaining profitable ore and close down the operation. The uncertain nature of the accessibility to the site in the future prompted an acceleration of the field fork scheduled for the Phase 2 studies. A field expedition was

Figure 15.5-2. *Drilling at the Bangombé site in Gabon.*

mounted in late 1996 to ensure that water samples and hydraulic measurements could be obtained from the Bangombé site which was, at that time, likely to be mined out, see Figure 15.5-2. Boreholes from the Phase 1 field expeditions were remeasured and new boreholes were drilled to provide a more complete picture of the site chemistry and hydrogeology. The location of the drillholes is shown in Figure 15.5-3. Samples from these drillcores and water samples collected during the expedition will be used for studies of colloid formation, uraninite alteration and near surface reaction processes.

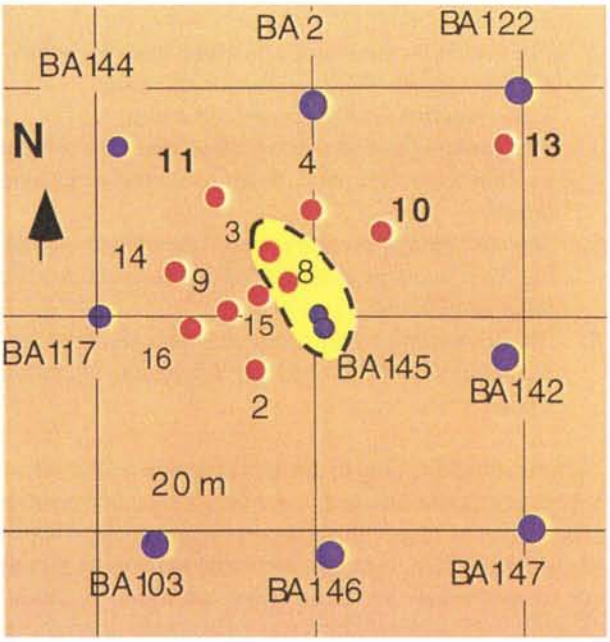


Figure 15.5-3. *Possible shape and contours of the reactor (yellow) in Bangombé. Location of prospecting holes (blue dots) and boreholes drilled for the Oklo project (red dots).*

The existence in nature of uranium ores that had $^{235}U/^{238}U$ ratios similar to those used today in enriched nuclear fuel for power production occurred because the decay of ^{235}U is much more rapid than that of ^{238}U. This means that as you go back in time $^{235}U/^{238}U$ becomes larger. Each of the uranium isotopes decay to an isotope of lead and provides a means of determining the time of formation of the uranium minerals or of some subsequent separation of the parent U from the daughter Pb. The radiogenic Pb isotopes can be distinguished from the Pb isotopes that were incorporated into the rocks when the uranium deposit and its host formations were first formed. During Phase 1 studies, a specimen from Reactor Zone 10 was found that contained uraninite crystals embedded in organic material. Detailed examination indicated that the organic matter had protected the uraninite from subsequent chemical disturbances and gave the best estimate for the age of the nuclear reactions as 1968 Ma. Samples from the nearby Reactor Zone 13 showed disturbance of their lead isotopic systems, probably due to the magma intrusion of a nearby dike /15.5-8/.

Timing of the events at Oklo, both in a relative sense and in an absolute sense, is essential. SKB has therefore established contacts with the Swedish Museum of Natural History. Their Cameca 1270 ion microscope will allow a much finer scale of resolution of mineral phases during isotope dating. Detailed studies has begun on samples from two boreholes – one from Reactor Zone 1 which is very similar to Reactor Zone 10, and one from Okelobondo, which is a region of a rather complex history at the southern end of the Oklo deposits. The isotopic studies will begin with lead isotope measurements of galena (PbS) that is associated with uraninites and noble metal fission product exsolution aggregates (separated particles). The lead now found in galenas was separated from its parent uraninite during one of the events of interest, either crystallisation of uraninite at the end of fission reactions, cooling of the uraninite below a diffusion closure temperature for lead from uraninite, or some subsequent heat event, such as the thermal period that was associated with the dike intrusion found in the vicinity of Reactor Zone 1 and 13.

Aggregates of noble metal fission products have been found in samples from Reactor Zones 10 and 13. These are of interest because technetium in spent fuel is found in a five-metal alloy phase and Tc, if it becomes oxidised, is very mobile. It is possible that the occurrence of Tc in an alloy phase will prevent oxidation and release. Laboratory studies have been undertaken as part of the Oklo Phase 2 project to synthesise alloys containing the five fission product elements found in spent fuel alloy particles and to investigate the leaching and dissolution behaviour of these synthetic particles as a function of environmental conditions and particle size. Samples of alloys containing Ru, Rh, Mo and Pd have been made using two different methods. A cold crucible method in which the alloy forms from a gas phase suspension and solidifies as a button, produced very clean material with no detected impurities. A more conventional technique of arc melting produced an alloy that contained 5-10% iron as well as four starting metals. These materials will be evaluated before selection of a synthesis route for the five component alloy. Results from the laboratory studies of both the 4 and 5 component alloys will be combined with electron microprobe and ion microscope examination of the Oklo metal aggregates to determine whether a limit to Tc oxidation can be established for the repository case.

Samples from the near surface site Bangombé are examined to evaluate the history and development of the secondary uranium minerals there. These studies, which are performed by Aarhus University, Denmark, in co-operation with University of New Mexico, USA, will help us to assess the ability of uraninite (as an analogue to spent fuel) and its alteration products to bind fission products and actinide elements under conditions of dissolution and precipitation of the uranium.

Evaluations for performance assessment

A Performance Assessment Interface Group (PAIG) has been established in the Oklo project, phase 2, to help provide the linkage between those conducting geologic studies and those with interests in using the data in site and concept performance assessments. An important task of this group is to provide ongoing advice to the Oklo project manager and those responsible for the specific field studies.

Oklo is, so far, the only place on earth with evidence of natural criticality. This is not so surprising because there are many conditions that must be fulfilled to obtain criticality in natural uranium. It's impossible today because the concentration of the isotope ^{235}U is too low but it was high enough 2000 million years ago. Long before that it was even higher but oxygen was missing in the atmosphere at that time so concentrated uranium mineralisations were not formed. Life on earth has contributed to the leaching (oxidation) and precipitation (reduction) of uranium. There were other "lucky" coincidences in Oklo such as water being present in the right proportions, that metals, which capture neutrons (e g manganese and vanadium) and follow uranium, appeared in other places in the area and that the orebody is very rich in uranium. Naudet has investigated the conditions and development of the criticality in Oklo very carefully and reported it in his book /15.5-7/. SKB has, with permission, translated this material from French to English /15.5-9/ and used the information to compare Oklo with the conditions in a deep repository for spent fuel. Reference to the Oklo phenomenon was made already in the first analysis of criticality /15.5-10/ and the conclusion this time is the same; there will not be any criticality in a repository /15.5-11/.

15.5.3 Palmottu

The new Palmottu project, supported by EU, started in November 1995. The investigated site is a uranium mineralisation at Lake Palmottu in Finland. The ore forms a 1 – 15 m thick steeply dipping zone that extends to a depth of about 300 m. It is studied as an analogue to spent fuel in a repository in granitic rock. Geology, hydrogeology, groundwater chemistry, climate etc are similar to that expected at sites in Finland and Sweden. The project is managed by GTK in Finland and SKB participates together with ENRESA and BRGM. The aims of the Palmottu project are:

- 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 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 studies to develop and refine models used for performance and safety assessments.

A new deep borehole have been drilled in Palmottu and used for sampling and investigations of the rock and groundwater /15.5-12/. The hole is inclined 60^{o} (from the horizontal plane), the first part was percussion-drilled and the rest rotary-drilled (Ø76 mm). The hole was TV-logged in the percussion-drilled part (hole length 0 – 65 m) and the rotary-drilled part (hole length 71 – 553 m). A short supplementary borehole was rotary drilled nearby, parallel to the first to get a complete core sampling of the upper part of the rock. This hole was also TV-logged covering the length interval 0 - 84.5 m. Hydraulic measurements were carried out in the two new holes and in some of the old 46 mm prospecting holes in the area. Spinner measurements, other flow-logging technique /15.5-13/ and cross hole tests were used. The "chemistry wagon", developed by SKB for site investigations, was used for sampling of groundwater chemistry. It was valuable for us to test methods and equipment for logging, hydraulic measurements and groundwater sampling /15.5-14/ and analysis that were either new or had been developed further since the investigations at the study sites in Sweden and since the preinvestigation phase at Äspö (1986 – 1990).

The data is compiled and the structural and hydraulic model of Palmottu is up-dated. The first Phase of the project will be finished and reported in spring next year (1997), and evaluated by EU before any continuation is decided.

15.5.4 Jordan

The Maqarin project started in 1990 with funding from Nagra, Nirex and Ontario Hydro. SKB has been participating since 1991. The results of Phase 1 have been published /15.5-15/. A large part of Phase 2 involved testing (validation) of chemical data and codes for calculating solubilities of radionuclides. The final report of Phase 2 is delayed for various reasons /15.5-16/. However, the project has continued with Phase 3 which is funded by EA (Environment Agency, UK), Nagra, Nirex and SKB. The third phase is co-ordinated and administered by SKB.

Cement and concrete are man-made materials and many of the mineral phases in cement paste, and the cement pore water composition, are unusual in nature. However, the examples that exist in nature are of great interest in so far as they can give us an insight into the long-term stability of concrete and its influence on the near-field environment. The last issue, concerning the high pH of pore water in Portland cement, has been the reason behind studies of hyperalkaline wells. Many of these are associated with ophiolite formations and, typically, can have a pH of up to about 11. Such wells are found, for example, in Cyprus and Oman. The water-mineral reactions which give the high pH are part of the alteration of highly reactive ultramafitic (rich in Mg and Fe) rock to serpentinite ($Mg_3Si_2O_5(OH)_4$). The hyperalkaline wells in Maqarin in Jordan has a different origin. Bituminous marl (limestone and clay minerals), with an organic content of 15 – 20%, has been burnt in situ to a cement like material after spontaneous ignition caused by pyrite oxidation. The metamorphic zone with cement like material lies within the original bituminous marl. The zone extends through the area with strongly varying thickness, of about 2 – 3 m to a maximum of 60 m. Groundwater, in contact with the metamorphic zone, develops a composition close to that of cement pore water. The pH is in the range 12 – 13 and the dominating cations are calcium, sodium and potassium. Sulphate is a common anion in addition to hydroxide. A full description of the mineralogy etc of cement-like material in Maqarin and its alteration products are given by Milodowski et al. /15.5-15, Appendix C/.

The ^{230}Th ingrowth method was used to measure a maximum age of about 0.5 – 2 Ma for the metamorphism which produced the cement zone in Maqarin /15.5-15/. Spontaneous combustion does still occur in the bituminous marl due to, for example, exposure following mechanical excavation. Temperatures of about 450^{o}C have been indicated at such events. Higher temperatures, in the range 800 – 1000^{o}C, have been indicated by mineral analysis of the old metamorphic zones. Therefore, according to Taylor (1990) we should expect belite (*C_2S* in cement nomenclature) and lime but no alite (*C_3S*). The closest we have come, so far, to belite and alite is larnite and spurrite which are similar but not identical to the

Table 15.5-1. Identified cement-like minerals in Maqarin /15.5-15/.

Mineral	Ideal formula
spurrite[a]	$Ca_5(SiO_2)_2(CO_3)$
wollastonite[b]	$CaSiO_3$
larnite[c]	Ca_2SiO_4
brownmillerite	$Ca_2(Al,Fe)_2O_5$
lime	CaO

[a] Spurrite is the mineral in Maqarin closest to alite in cement (*C_3S*).
[b] Calcium monosilicate (*CS*) is not really a component of cement which is burnt at normal high temperature (1450°C).
[c] Larnite is similar to belite in cement (*C_2S*).

Table 15.5-2. Identified cement-paste-like minerals in Maqarin /15.5-15/.

Mineral	Ideal formula
calcite[a]	$CaCO_3$
gibbsite[b]	$Al(OH)_3$
brucite[c]	$Mg(OH)_2$
portlandite	$Ca(OH)_2$
quartz[b]	SiO_2
gypsum	$CaSO_4$
ettringite	$Ca_6Al_2(SO_4)_3(OH)_{12}\ 25H_2O$
thaumasite	$Ca_6Si_2(SO_4)_2(CO_3)_2(OH)_{12}\ 24H_2O$
afwillite	$Ca_3Si_2O_4(OH)_6$
tobermorites	$Ca_5Si_6O_{16}(OH)_2\ 2\text{-}8H_2O$
jennite	$Ca_9H_2Si_6O_{18}(OH)_8\ 6H_2O$
C-S-H gel	amorphous, undefined

[a] As carbonation product in cement paste.
[b] Not a normal component of Portland cement (occurs in aluminate cement paste).
[c] Present if there is dolomite in the ballast or precipitate due to sea water intrusion.
[d] Silica present in the ballast and sometimes used as additive (e.g. silica fume).

cement minerals, see Table 15.5-1. The absence of tricalcium aluminate phases (*C_3A*) is another indication of temperatures lower than ambient in a cement kiln (1450°C).

After the combustion zone cooled down (below ca 100°C), groundwater could enter and react with the cement clinker-like phases. The low-temperature reactions with the metamorphic rock, such as hydration, carbonation and sulphatisation, created an assemblage of alteration minerals. This included calcite, apatite, ettringite, thaumasite and portlandite, see Table 15.5-2. Many of the alteration minerals in Maqarin are common to cement

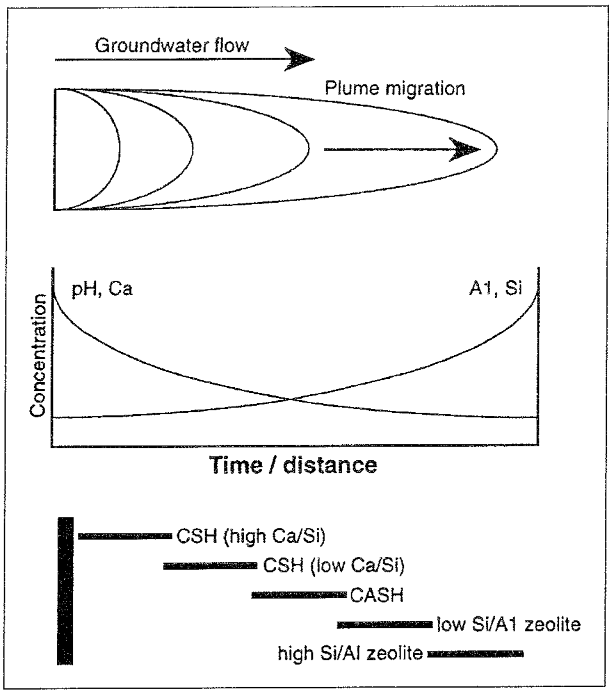


Figure 15.5-4. Schematic diagram of hyperalkaline plume migration from a cementitious source, showing hypothesised variations in fluid composition and alteration mineralogy in space and time /15.5-17/.

paste, including portlandite, ettringite and calcium silicate hydrates, see Table 15.5-2. Much of the alteration is very fine-grained and the different phases are often intimately intergrown and very hydrous. Larnite (*C_2S*) and spurrite (*C_5S_2*-carbonate) are the most readily altered, followed by brownmillerite (*C_4AF*). Alteration along fractures has caused some degree of expansion and microfracturing. Some primary calcite appears to have been removed.

The earliest hydration product appears to have been calcium silicate hydrate with a gel-like morphology which have lined or infilled fractures and other cavities. The Ca/Si-ratio is between 0.8-0.85 which is similar to tobermorite (*$C_5S_6H_9$*). Ettringite (*AFt*) is very abundant and may dominate altered lithologies. In most fractures, ettringite is accompanied by thaumasite. Afwillite (*$C_3S_2H_3$*) seems to have replaced ettringite in some veins (preceded by precipitation of thaumasite).

Progress has been made in the evaluation of the influence of the hyperalkaline groundwater on minerals in the surrounding rock /15.5-17 and 15.5-18/. It has been suggested that alkaline leachates from cement could influence the host rock in the near-field of the repository and thereby change its radionuclide retention properties. This has been evaluated from a theoretical point of view and the predicted sequence of minerals were identified in Maqarin, see Figure 15.5-4. Zeolites are formed first, when pH start to increase, followed by calcium alumina silicates and calcium silicates. Fractures will tend to seal

Figure 15.5-5. Sampling of microbes in the hyperalkaline wells in Maqarin.

Figure 15.5-6. Early coredrilling in Maqarin (1955). Some of the holes have been used by the project in addition to our own sampling holes.

as a result of the secondary minerals formed but the radionuclide retention properties are not likely to be diminished. Pore spaces in the rock behind the fractures remained open in Maqarin. Moreover, zeolites and the calcium silicates are efficient sorption substrates, at least for some of the radionuclides of interest. The general conclusion is that rock fractures will tend to seal as a result of high pH weathering which can diminish water flow in the near-field; the flow wetted surface can possibly decrease for the same reason but the bulk rock behind the fractures will still be accessible as the main sink for retention of radionuclides by matrix diffusion.

Colloids, dissolved organic material and bacteria are also being studied in Maqarin. Colloid concentration is very low in the highly mineralised groundwater which support observations of low colloid concentrations in concrete pore water. Organic matter is different from humic substances normally found in groundwater. This is being further analysed. Bacteria can survive under hyperalkaline conditions. That was demonstrated at an early stage of the investigations. Further investigations of the micro-organisms are under way, see Figure 15.5-5.

15.6 BIOSPHERE

The biosphere includes the transport of nuclides from the aquifers above the bedrock, through natural and domestic ecosystems 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 for deep groundwater,
- be transported in ecosystems, possibly causing external dose,
- accumulate in foodstuffs like plants animals and fish,
- be consumed and cause internal dose.

The modeling 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 receivers (well, river, lake) and the contaminants can be transported through different physical compartments like soils, sediments, water and air. The processes considered include chemical/physical activity (e.g. sedimentation, resuspension, evapotranspiration) and biological activity (e.g. bioturbation, bioaccumulation), and human (e.g. farming, hunting, burning).

The modeling of the biosphere is done with compartment models for which volumes and transfer factors between compartments are calculated. Solving the (linear) differential equations produces time dependent concentrations for the different compartments.

After this the accumulation in foodstuffs, intake and dose calculations are pure multiplication for each scenario. Further development of biosphere models is also discussed in the safety analysis section.

15.6.1 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 modeling /15.6-1/ and some attempts to determine a justified area of application for some models have been made in BIOMOVS /15.6-2, -3, -4, -5/ VAMP /15.6-6/.

An overview of the uncertainties in biosphere modeling /15.6-7/ demonstrated that the sources of uncertainty are generally many and dependent on the scenario. The confidence intervals normally span several orders of magnitude.

BIOMOVS

The BIOspheric MOdel Validation Study is an international co-operative 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 modeling 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 modeling /15.6-8/.

In 1991 the second phase, BIOMOVS II, was started, jointly managed by the five organizations 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 modeling and conceptual uncertainties arising with time. A general methodology and a list of FEPs (features events and processes) have been published /15.6-9/. 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 modeling /15.6-10/. In a validation scenario for 14C within the "Validation and Uncertainties" theme, we are stochastically testing the model used for the SFR safety assessment with almost all parameters as PDFs /15.6-3/ . 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 /15.6-5/ .

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 program, modeling of ^{137}Cs in lakes and uncertainty analysis has been intercompared between several working groups from several countries /15.6-11/. The implications are that simpler models with site specific parameters reflecting retention time, give the best estimates /15.6-6/ .

15.6.2 Dose factors

Land rise and gravel zones

As more interest is focused on the 10 000 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.

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 area west of Äspö was studied and six types of recipients could be identified. The long-term transfer of elements across the biosphere and geosphere was studied in a bog /15.6-12/.

The individual doses from inflow of radionuclides from a major fracture zone was also estimated /15.6-13/.

15.6.3 Radionuclide transport

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. 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 /15.6-12/.

15.6.4 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 summarized 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 /15.6-14/.

A literature survey was completed during 1993 /15.6-15/ and is now followed by an attempt to estimate the natural and seminatural levels of radionuclides in nature. Estimating the doses that some species normally get and looking for effects may add in understanding the possible effect on ecosystems.

16 OTHER LONG-LIVED WASTE THAN SPENT NUCLEAR FUEL

16.1 GENERAL

Swedish low and intermediate level waste are packaged and sent to the final repository SFR in Forsmark. Most of this waste comes from the operation of nuclear power reactors but a minor part is from research, industry and medicine which is packaged and stored in Studsvik. In the future, short-lived LLW and ILW from decommissioning of nuclear installations is also intended for disposal in an extension of SFR.

Some of the waste in Studsvik, from research mainly, is set aside when it contains too much long-lived radionuclides to be accepted for disposal in SFR. Likewise, used internal parts from the power rectors, including core components, are too rich in long-lived nuclides for SFR and are presently stored at CLAB (or at the power plants). Therefore, plans have been made to build a separate part of the deep underground repository SFL to receive long-lived LLW and ILW. This part will also be used to dispose short-lived operational waste from CLAB and the encapsulation plant, provided, of course, that SFR has been closed at that time. The operational waste and short-lived decommissioning waste will be about half of the total volume.

Design studies of repositories for long-lived LLW and ILW, and preliminary assessments of their barriers have been performed in the past /16-1 and 16-2/. Recently it has been decided to include disposal of this waste in the safety assessment report SR 97 which is intended as a part of a future MKB application. The report SR 97 is mainly directed to the issue of spent fuel disposal but it should also cover the present plans for long-lived low and intermediate level waste, and any operational waste etc that is produced after SFR has closed. This will be high-lighted in SR 97 and further information added in an Appendix to the main report. Aims and time schedule of the studies of what we refer to as "other waste" (short for other long-lived waste than spent nuclear fuel) has changed and the prestudy project has been redirected to fulfil the requirements of SR 97. The main aim for the 'Project Group Other Waste' is to have the Appendix and other materials for the main report SR 97 ready by September 97.

16.2 THE WASTE

An estimate of the amount, composition and quality of other waste was made and reported within the frame of the prestudy /16-2 and 16-3/. Included in the report was a description of the waste and waste packaging describing origin, types of materials and amounts, metal surface areas (for assessment of corrosion) and the radionuclide content /16-3/. The original aim was to make a realistic waste inventory in order to evaluate the function of the barriers in the repository concept. This "Waste Characterisation Report" is now revised with the new aim to provide an updated basis for the performance assessment of waste disposal in connection with SR 97.

16.2.1 Waste from Studsvik

LLW and ILW from Studsvik consist of waste from research, industry and medicine. A large part comes from Studsviks own research laboratories but waste from other producers, for example, institutes, universities and hospitals is also collected, treated and stored at Studsvik. The raw waste consist of activated and contaminated scrap metals, precipitation sludges, ashes, spent ion exchange resins, glove boxes, radiation sources, laboratory outfit and radiation protection equipment. Most of the waste is packaged in reinforced cubical concrete containers (side length 1.2 m) or 200 l steel drums. The Studsvik concrete container has five 105 l cylindrical holes, prepared for 85 l steel drums. The 200 l steel drums are of two types; steel drums with mixers are used for solidified sludge and double wall steel drums, with concrete in between, for ashes, refuse and scrap. Cement is used for conditioning when necessary.

Waste from Studsvik is divided up in two categories: waste suitable for disposal in SFR and waste for later disposal in a deep repository. The estimated total number of packages in the second category is presented in Table 16-1. All long-lived LLW and ILW falls in the second category.

16.2.2 Operational waste from CLAB and the encapsulation plant

CLAB and the encapsulation plant the encapsulation plant will, according to present plans, continue to operate after SFR has closed. Short-lived waste from these facilities will thereafter be sent for disposal to the deep repository SFL. The operational waste from CLAB and the encapsulation plant consists of spent ion exchange resins, filters, filter aid and solid scrap and trashes packaged in cubical concrete moulds with side length 1.2 m, i.e. the same outer dimensions as the Studsvik concrete containers. The resins and filter aid are used for the clean-up of water in the different pools and for cleaning of process water and

Table 16-1. Waste from Studsvik to be disposed of in the deep repository.

Waste type	Number of packages	Volume[a] m^3
Concrete containers		
ILW	400	691
Pu-contaminated waste	100	173
U-containing waste	1	2
Th-containing waste	3	5
Tritium containing waste	10	17
Steel drums		
Solidified sludge	1000	214
Ashes	556	120
Refuse and scrap	2390	512
Steel containers		
Decommissioning waste	12	9
Concrete boxes		
Pu-contaminated waste	29	93

[a] Total outer volume of packages.

drainage. The spent ion exchange resins are conditioned with cement in the containers (conventional method).

From the nuclear power plants CLAB receives spent fuel and core components which are repackaged and stored in water filled pools. Radioactive waste consists mainly of crud from the fuel receiving operation (cooling) and spent ion exchange resins. The amount of metal scrap and trash from maintenance and replacement of equipments is relatively small. It is expected that the encapsulation plant and, later, the treatment of core components will generate the same type and amount of waste as the receiving operation in CLAB. The calculated total amount of waste packages with short-lived operational waste from CLAB and the encapsulation plant, produced after SFR has closed, is presented in Table 16-2.

Table 16-2. Short-lived operational waste from CLAB and the encapsulation plant to be disposed of in the deep repository.

Waste type	Number of packages	Volume[a] m^3
Concrete moulds		
Ion exchange resins	1800	3110
Scrap and trash	360	622

[a] Total outer volume of packages.

16.2.3 Core components and internal parts

Used core components and internal parts from both maintenance work and decommissioning are foreseen to be disposed of in SFL 5 (see Section 16.3.3). Presently, the 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). Metal scrap cassettes, used for the interim storage of the major part of this waste, and CLAB cassettes, used for some of the metal parts, will go to final disposal together with their content of waste. Alternatively, CLAB cassettes may become separated from their waste content, decontaminated and treated as the other CLAB cassettes used for spent fuel assemblies (see Section 16.2.5) .

Used core components and internal parts have a relatively high specific activity which is induced by the neutron irradiation in the 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 from the core the surface contamination (crud) dominates. The limit, outside which waste can be accepted for SFR, lies between 0.5 – 1 m. The reactor components are in general made of stainless steel but there are also additional materials represented such as boron steel, boron carbide, hafnium, zircaloy, inconel and boron glass. The contributions to activity from ^{60}Co and ^{63}Ni are dominating in the short-time perspective but long-lived nuclides, for example ^{59}Ni, are generated by the neutron flux and the waste is characterised as long-lived waste. An inventory of the expected waste production has been made and the calculated total amounts of waste packages and their outer volumes are presented in Table 16-3.

16.2.4 Decommissioning waste from CLAB, the encapsulation plant and Studsvik

Short-lived decommissioning waste from the power reactors will be disposed of in an extension of SFR (SFR 3) but CLAB and the encapsulation plant will, presumably, be decommissioned far into the future so their radioactive parts are allocated to the deep underground repository. Decommissioning waste from CLAB and the encapsulation plant will consist of metals and concrete. The steel components are mainly parts from the cooling and cleanup system in the storage pool, i.e. tubing, pumps and tanks. The concrete waste is from buildings where it has been chipped off from potentially contaminated walls. The intended waste container is a cubic carbon steel vessel (side length 2.4 m and thickness 6 mm). Some decommissioning waste from Studsvik will also be received. Most of this is also suitable for disposal in SFR but some of the decommissioning waste from Studsvik will presumably

Table 16-3. Core components and internal parts from the power reactors to be disposed of in the deep repository.

Waste type	Number of packages	Volume[a] m^3
BWR-waste		
Core grid and spray	252	1741
Moderator tank and cover	270	1866
Control rod guide tube	45	311
Incore instrumental equipments	108	746
Boron plates	20	138
Control rods	419	2895
Compacted fuel channels	14	97
Transition pieces	1	7
Used parts	4	28
PWR-waste		
PWR Reactor tank	63	435
Burnable absorbers	51	352
PWR Internal parts	150	1037

[a] Total outer volume of packages.

Table 16-4. Decommissioning waste from CLAB, the encapsulation plant and Studsvik to be disposed of in the deep repository.

Waste type	Number of packages	Volume[a] m^3
Waste from CLAB		
Steel components	98	1352
Concrete	53	731
Waste from the encapsulation plant		
Steel components	6	83
Concrete	3	41
Waste from Studsvik		
Metallic waste	7	48

[a] Total outer volume of packages.

have to be disposed of in the deep repository, for example, some metallic parts from the research reactor R2. The estimated number of packages of decommissioning waste from CLAB, the encapsulation plant and Studsvik is presented in Table 16-4.

16.2.5 Miscellaneous waste

The storage canisters at CLAB, holding the fuel assemblies in the storage pool, will also be disposed of in the deep repository, see Table 16-5. The same cubic carbon steel vessel as for decommissioning waste is foreseen as packaging (side length 2.4 m). It has been suggested to decontaminate the storage canisters before disposal. Preliminary assessments indicate that this would simplify handling and improve the isolation of the waste nuclides which are then concentrated to less volume in a well protected part of the repository (SFL 3, see Section 16.3.1).

The transport casks and containers are used during transportation of HLW and ILW between the power plants, the interim storage and the final disposal site. The transport casks and containers will probably also be disposed of as waste, despite their very low contamination levels, see Table 16-5.

Table 16-5. Storage canisters from CLAB, transport casks and transport containers to be disposed of in the deep repository.

Waste type	Number of packages	Volume[a] m^3
Storage canisters from CLAB		
Canisters	467	6445
Transport casks and containers		
Casks for spent fuel	10	102
Casks for core components	2	20
Containers for ILW	29	784
Casks for spent fuel canisters	40	408

[a] Total outer volume of packages.

16.3 THE REPOSITORY

The intention is to dispose of this waste in an annex to the deep underground repository SFL 2 for spent fuel. Three parts are required: SFL 3, SFL 4 and SFL 5. A new design of the SFL 3-5 parts has been developed, see Section 13.2. A depth of between 300 – 400 m has been suggested for these parts of the repository and a separation from SFL 2 of about 1 km. The present design of SFL 3 and 5 is based on the experiences of SFR-BMA, see Figure 16-1. The two cavern constructions SFL 3 and 5 are identical and the tunnel system surrounding them will be used as SFL 4 (less active waste). Inner constructions of SFL 3 and 5 will be backfilled with porous concrete, a technique now used in SFR, and all other void spaces will be backfilled with crushed rock. The guiding principle is to facilitate any release of gas and to direct flowing groundwater past the inner construction which contains the waste packages. SFL 4 has no inner construction but the waste is less active and, to some extent, protected by the containers.

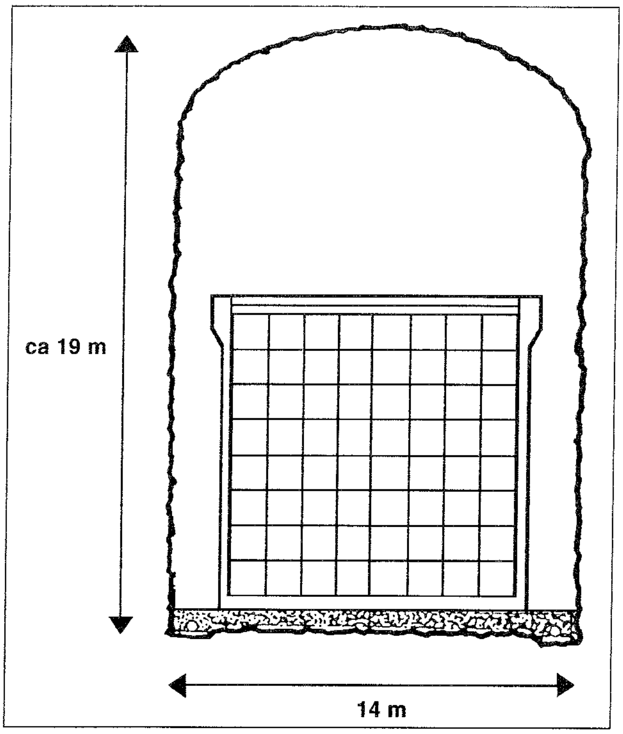


Figure 16-1. Schematic cross-section of SFL 3 (or SFL 5). The inner concrete construction, containing the waste packages, is standing on a bed of gravel. The void space between the inner construction and the rock is, finally, backfilled with crushed rock /13-4/.

16.3.1 SFL 3

SFL 3 is mainly intended for long-lived LLW and ILW from Studsvik where it has been packaged and stored (see Table 16-1). It shall also receive short-lived operational waste from CLAB and the encapsulation plant after SFR has closed (see Table 16-2). The calculated total volume of waste is 5 600 m^3 (outer volume of waste packages). The contribution from Studsvik is about 1 800 m^3 and from CLAB and the encapsulation plant together about 3 800 m^3. The calculated total activity at year 2040 is 2 10^{16} Bq. Initially the activity is dominated by ^{63}Ni in *Studsvik concrete containers with ILW* (see Table 16-1) and *CLAB/encapsulation plant ion exchange resins* (see Table 16-2). After 1000 years the activity is dominated by ^{59}Ni, see Figure 16-2. Half-life of the nickel isotopes is 93 years for ^{63}Ni and 75 000 years for ^{59}Ni.

16.3.2 SFL 4

SFL 4 is intended for low level decommissioning waste from CLAB and the encapsulation plant (see Table 16-4) as well as for transport casks, transport containers and fuel storage canisters from CLAB (see Table 16-5). The storage volume required is about 10 000 m^3 of which 2 200 m^3 is decommissioning waste, 6 600 m^3 is storage canisters from CLAB and 1 200 m^3 is transport casks and

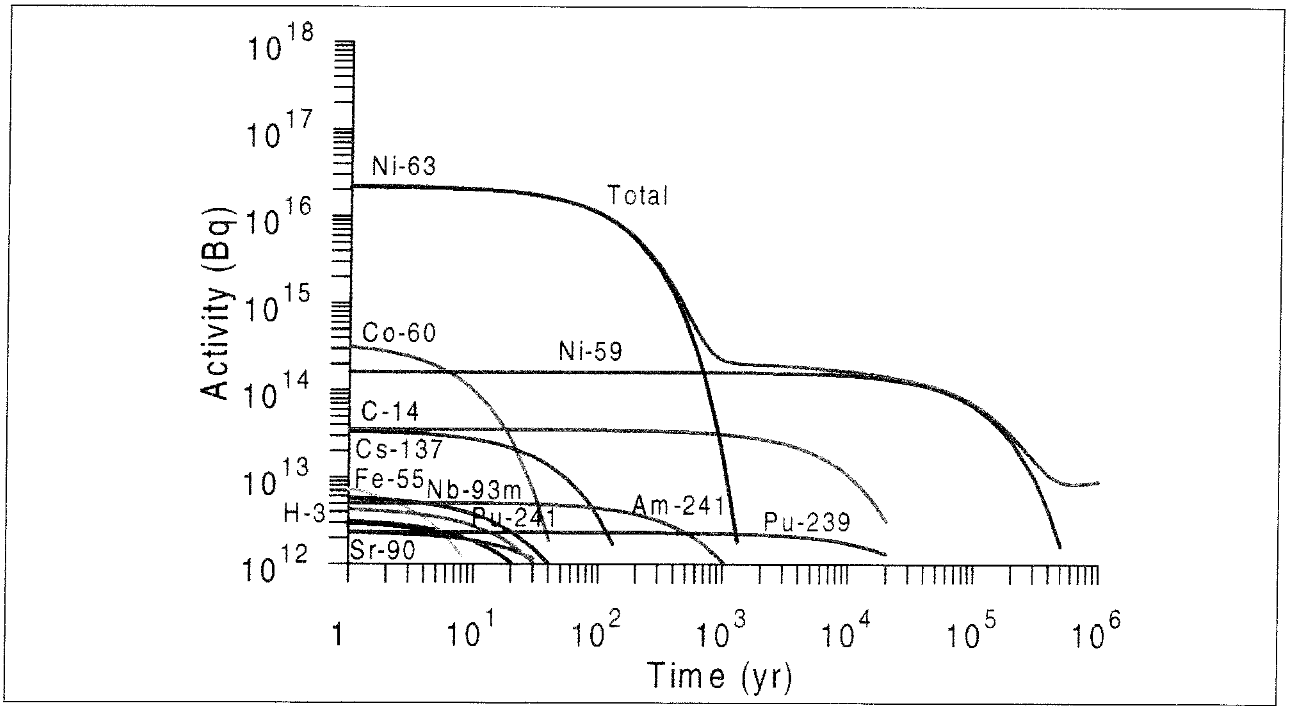


Figure 16-2. Activity content of different radionuclides in SFL 3 as a function of time (time zero is at year 2040).

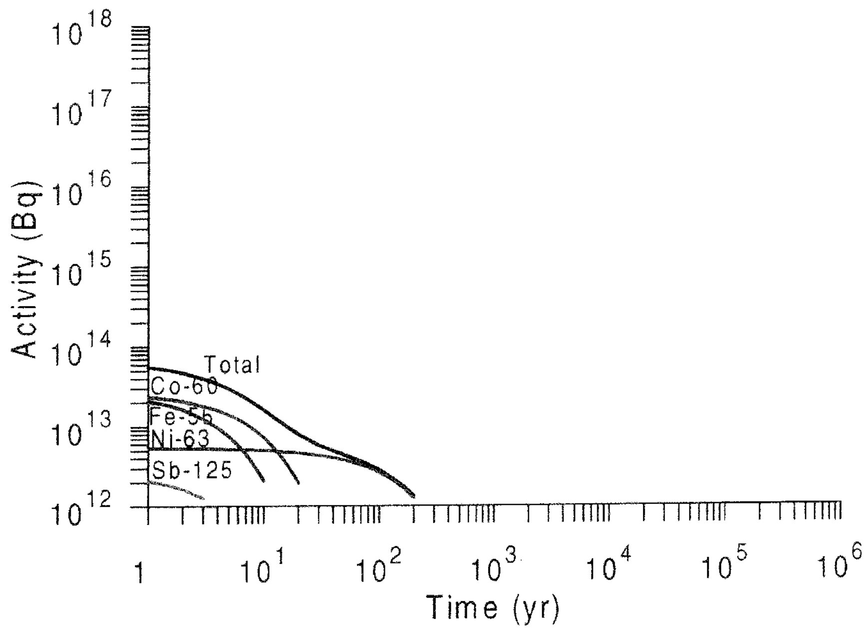


Figure 16-3. *Activity content of different radionuclides in SFL 4 as a function of time (time zero is at year 2040).*

containers. The calculated radionuclide inventory at year 2040 is 7 10^{13} Bq. The nuclides ^{55}Fe and ^{60}Co dominate during the first 20 years and thereafter ^{63}Ni, up to 1000 years, followed by ^{59}Ni, see Figure 16-3.

16.3.3 SFL 5

SFL 5 is intended for metallic core components and internal parts from the power reactors with high specific activity (see Table 16-3). This waste can, of course, arise as a result of both operation (maintenance work) and final decommissioning of a power reactor. Some decommissioning waste from Studsvik, e.g. metallic parts from the research reactor R2, will also be sent to SFL 5 (see Table 16-4). The storage volume required is about 9 700 m^3. Most of the space is needed for waste from power reactors. The contribution from Studsvik is only 48 m^3. The calculated radionuclide inventory at year 2040 is 1.4 10^{17} Bq. The nuclide ^{63}Ni dominates the first 1000 years and thereafter ^{59}Ni, see Figure 16-4.

16.4 BASIC DATA AND MODELS FOR PA

16.4.1 Concrete and hydrochemistry

Cement paste with a water to cement ratio (w/c) of about 0.38 should, in principle, be fully hydrated /16-4/. However, unhydrated cement minerals can be found even in concrete with a higher w/c /16-5/. The hydration products start to grow from the surface of the clinker grains and fill the space between the aggregate grains. The main hydration products are poorly crystalline calcium silicate hydrates and calcium hydroxide (portlandite). The calcium silicate hydrates form a gel in the cement paste and is referred to as C-S-H in concrete chemistry nomenclature (Calcium-Silicate-Hydrogel). In hardened concrete C-S-H is the principal binding phase which holds the solid grains of cement, sand and gravel together. Investigations on the molecular level of the C-S-H structure in hardened cement paste indicate a layer structure which, together with a pore solution, forms a rigid gel with pores ranging in size from macroscopic to enlarged interlayer spaces of nanometer dimensions. The C-S-H layers are expected to form subparallel groups, a few layers thick, which enclose pores of dimensions from interlayer spaces and upwards /16-4/.

Water in hardened cement paste are present in different thermodynamic states depending on if, and how, the water molecules interact with the cement phases. There will be *free water*, where the water molecules are not directly affected by the forces from the mineral surfaces, *absorbed water* which consists of about two layers of water molecules, influenced by the mineral surfaces, and, *structural water* in the minerals such as crystal water and hydroxyl groups (e.g. portlandite). The cement paste will influence the pore water. Alkali hydroxides (NaOH and KOH) in the fresh cement paste will generate a high pH of about 13 and more. If the alkali hydroxides are leached out or consumed by chemical reactions, there is enough portlandite ($Ca(OH)_2$) to keep the pH of cement pore water at

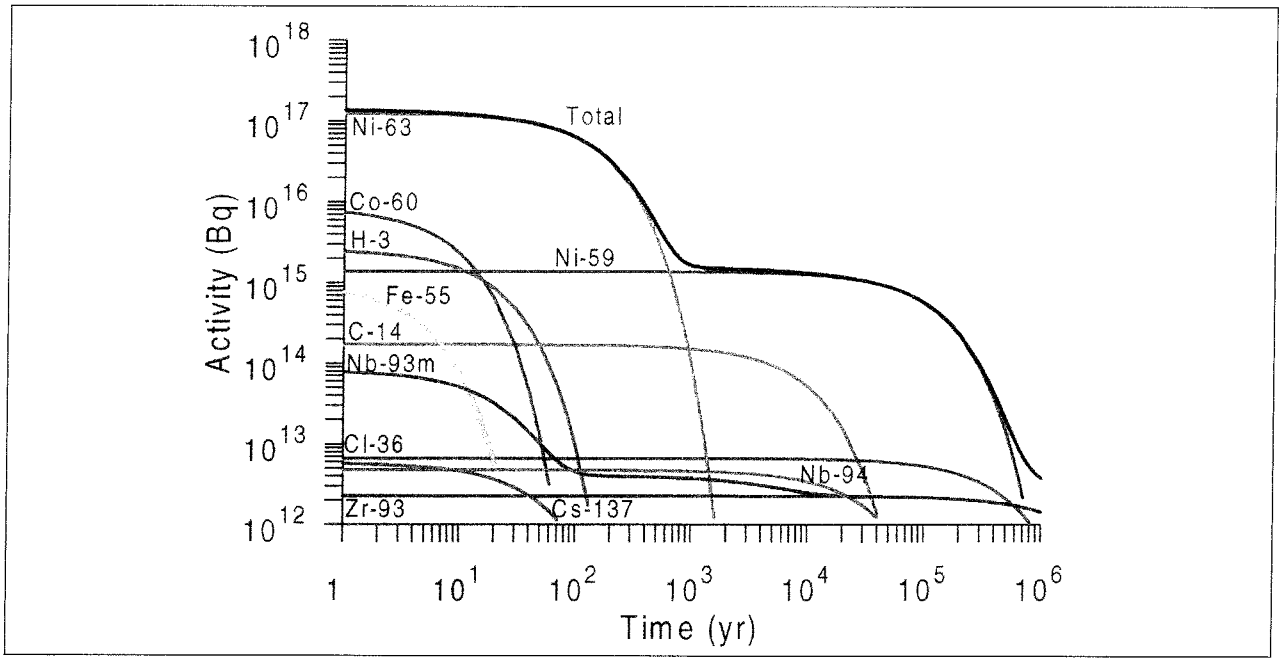


Figure 16-4. Activity content of different radionuclides in SFL 5 as a function of time (time zero is at year 2040).

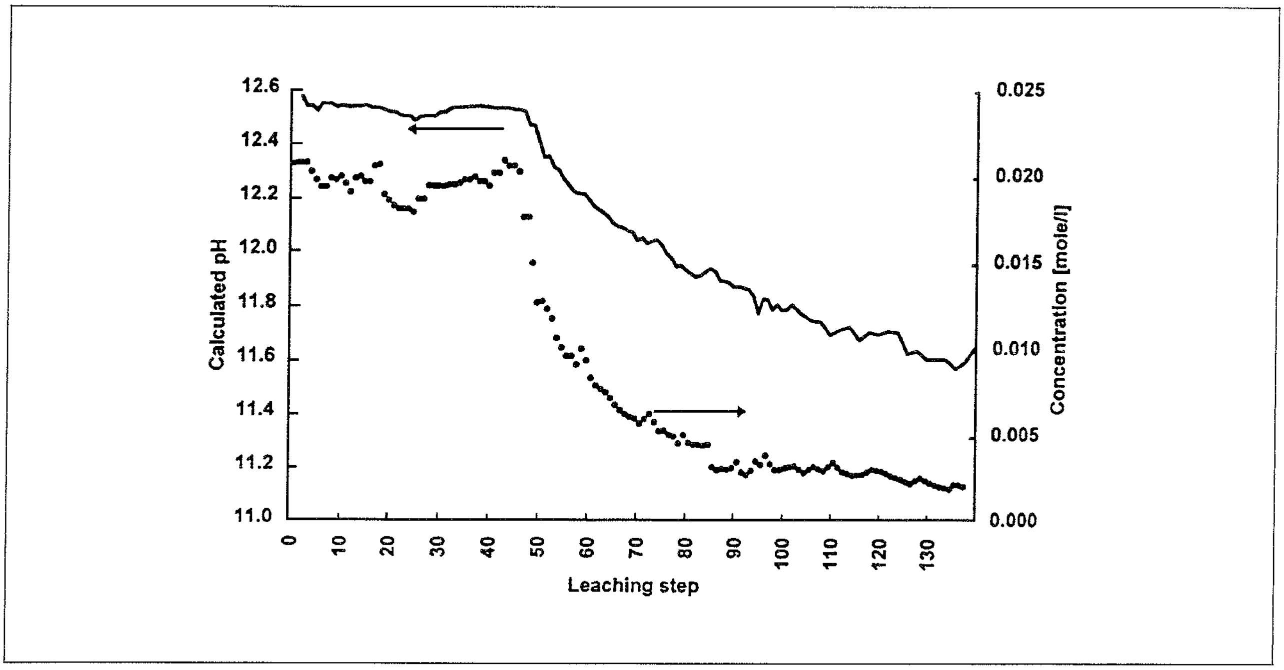


Figure 16-5. Leaching of cement paste. Concentration of calcium and pH /16-6/.

about 12.5. The capacity to control the pH at a high level can be estimated from the composition of cement, see Table 16-6, and measured in leaching experiments /16-6/.

Table 16-6. The analytical composition of Degerhamn Std Portland cement /16-5/.

Oxide	% of weight	Oxide	% of weight
CaO	64.5	K_2O	0.6
SiO_2	22.2	Na_2O	0.1
Al_2O_3	3.5	Total alkali (as % Na_2O)	0.5
Fe_2O_3	4.7	Free lime[a]	0.8

[a] Free lime is CaO from unreacted rests of burnt lime.

It is reasonable to expect that a major portion of the alkali metals Na and K can be released as NaOH and KOH by leaching. Concrete contains roughly 350 kg of cement per m^3 so the maximum release is 56 moles of alkali hydroxides per m^3 of concrete. The same way of reasoning can hardly be applied to calcium hydroxide because the calcium, from calcium silicates and free lime in the cement, will appear both in C-S-H and portlandite in the hydrated cement paste; most of it will be in C-S-H. There will also be some unhydrated clinker left with Ca bound in it. Therefore, it would be exaggerated to assume that all of the CaO in Table 16-6 is available as portlandite. Engkvist et al. /16-6/ leached samples of crushed cement paste until pH was below 12 and found a weight loss of 20 %, most of this is $Ca(OH)_2$, see Figure 16-5. Concrete contains about 350 kg of cement (ca 420 kg as hydrated cement paste) per m^3 concrete which, combined with Engkvist's results, implies that 1 m^3 of concrete can release 2300 moles of hydroxide ions from $Ca(OH)_2$.

Leaching of C-S-H will start when there is no portlandite left. The dissolution of C-S-H is incongruent with a higher Ca/Si-ratio in solution than in the solid. The Ca/Si ratio of remaining C-S-H will go down to 0.85, where dissolution becomes congruent /16-7/. The pH will successively drop to 11 and remain there during the congruent phase. A model for the dissolution of C-S-H has been developed by Berner at PSI /16-8/. For stoichiometric reasons it is not possible to release more hydroxides from C-S-H dissolution than what remains of calcium after the leaching of portlandite. The total analytical content of CaO is 65.3% (including free lime), see Table 16-6. Subtracting the portlandite part, there is less than 6000 moles of hydroxide ions which can be released from C-S-H in 1 m^3 of concrete.

16.4.2 Cement paste in old concrete

The cement paste in samples of old concrete from the wall of a water tunnel at the Porjus dam were investigated by Grudemo /16-9/ with X-ray diffraction. The tunnel was built in 1914 but, according to the analysis, no significant change in crystallinity had occurred after 65 years in a water filled inflow tunnel. It was concluded that the C-S-H phases would remain stable for a long time. Rayment /16-10/ made an electron probe analysis of C-S-H in 136 year old cement paste and found no major difference compared to modern Portland cement. The C-S-H had remained stable and still unreacted cement clinker grains were observed.

Lagerblad /16-11/ made a comprehensive study of samples from old concrete constructions throughout Sweden. The aim was to find and investigate cases relevant to the deep repository situation and preference was given to samples from water saturated environments. Thin section microscopy, SEM and X-ray diffraction were used. An overview of the study is presented in Table 16-7.

Relic grains of unhydrated cement clinker were found, up to a particle diameter of 0.3 mm, in most of the samples. Cases were also found where the grain had left a hollow shell of hydration products behind. Leaching could possibly explain that phenomenon. Calcite precipitations were found, and crystals of ettringite and portlandite, which tends to grow into voids left in the cement paste. The C-S-H in the good quality old concrete (e.g. the water tank) has a composition similar to modern cement paste and it was concluded that C-S-H is very stable if the leaching is kept down by, for example, a low hydraulic conductivity. Concrete with a relatively low w/c ratio (e.g. the inspection tunnel and the water tank) showed a positive maturity development where slowly hydrating clinker grains fill the pore spaces and the concrete gains additional strength by these reactions. The water tank in Uppsala has a quality today that would be difficult to reproduce in young concrete, at least without using superplasticisers.

The observations from the studies of old concrete can be summarised as follows:

- Rests of unhydrated clinker grains can remain for at least 100 years and possibly much longer in a dense concrete.
- The strength tends to increase with time and the hydraulic conductivity decreases in a good quality concrete stored under saturated conditions, provided there is no strong hydraulic gradient driving water through the construction.
- Concrete with a high w/c ratio leave hollow spaces and shells which can later be filled with recrystallisation and hydration products.
- Calcite is found as precipitates in the structure (due to infiltration of carbon dioxide or dissolved carbonates).

Table 16-7. Summary of observations made during the study of old concrete made by Lagerblad /16-11/.

Name	Age	Environment	Clinker	w/c[a]	New phases and crystals
Sillre/Oxsjön hydropower dam	59 y	Humid	Relic grains	>0.65	Calcite and ettringite
School building in Gävle	100 y	Dry	Relic grains	>0.7	Calcite and/or ettringite
Midskogsforsen hydro-power plant	52 y	Saturated	Hydrated	0.5	Ettringite
Rocksta mill	96 y	Wet	Hydrated	>0.7	Ettringite and portlandite
Älvkarleby hydropower plant – inspection tunnel	79 y	Dry	Relic grains	<0.3	N. D.
Älvkarleby hydropower plant – discharge chamber	79 y	Saturated	Some relic grains	0.4 – 0.6	Ettringite and portlandite
Uppsala castle water tank	90 y	Saturated	Relic grains	0.4	Calcite, ettringite and portlandit

[a] Estimate of w/c by comparison to modern concrete.

- Recrystallisation has been observed for ettringite and portlandite. The crystals are formed in void spaces in the structure.
- No recrystallisation of C-S-H is observed and this phase remain stable, at least as long as the pore water composition is controlled by portlandite (high pH and calcium concentration).
- Portlandite depletion of concrete due to leaching is less than previously assumed in our performance assessments of repository constructions.

16.4.3 Radionuclide chemistry

Solubility, sorption and diffusion of radionuclides in a concrete environment (high pH) are being investigated. The first aim is to produce a set of values to be used in transport calculations for performance assessment of repository barriers to radionuclide dispersal. Most of the information is taken from the literature but additional measurements are made in laboratories at Chalmers Technical University, Göteborg, and at University of Linköping. Experimental work has also been performed by British Geological Survey, BGS, for SKB.

The following investigations are being conducted:

- Solubility measurements of Ni, Pm, Eu and Th in cement environment.
- Measurements of radionuclide sorption on concrete (K_d-values). The element studied are: Th, Pm, Ni and Cs (IV-, III-, II- and I-valent metal ions).
- Measurements of radionuclide diffusion in cement paste and cement mortar. The elements studied are: Pm, Ni, Cs and tritium.
- 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.
- Measurements of the influence of ISA on sorption of radionuclides in cement. The elements to be studied are: Th, Pm, Ni and Cs (IV-, III-, II-, and I-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.

- Influence of high pH-weathering on rock matrix diffusion of radionuclides.

A close co-operation has been established with ANDRA, Nagra and Nirex where similar investigations are being made.

16.4.4 Hydraulic conditions

A generic modelling of the hydraulic flow in the near-field of the repository was made on the basis of the previous design of the repository. The near-field rock, tunnels, backfill and inner constructions were modelled with the continuum approach where groundwater head, flow stream lines etc were calculated by the use of a three dimensional finite difference method. This was reported at the beginning of 1996 /16-12/ but subsequently it was decided to modify the design. Therefore, additional modelling was made to describe the performance of the new design in an underground flow field. The influence of rock mass heterogeneity was studied by use of the stochastic continuum approach. The result of the calculations will be combined with the information of regional groundwater flow from the generic sites in the performance assessment study SR 97 to obtain site specific near-field flow.

Groundwater flow is of importance for the performance of the barriers to radionuclide dispersal. The strategy of the new design is to direct the near-field flow away from the waste packages by use of barriers that provide preferred flow paths around the inner construction (i.e. hydraulic cage). The hydraulic modelling has, in this case, inspired the new design and demonstrated its feasibility. The final test will be the calculation of radionuclide transport and performance assessment based on the new design and the calculated hydraulic flow-field.

16.4.5 Mass transport calculations

The computer code NUCTRAN, originally developed to calculate near-field transport from an anticipated defect canister in SFL 2, has been further developed to include the calculations of release from SFL 3-5. NUCTRAN is similar to a finite difference model for three dimensional problems but considerably less cells or compartments are necessary and computing time is saved as a result of that /16-13/. NUCTRAN for SFL 3-5 has been modified to function as a submodel to the code PROPER which is used to connect the numerical codes for radionuclide dispersal. PROPER can, for example, handle the whole chain of calculations needed to evaluate a scenario where radionuclides are leached from exposed waste, released to the near-field, transported by groundwater flow in the far-field and, finally, dispersed in the biosphere. The calculation of groundwater flow, which is needed as a basis for both near- and far-field mass-transport, is the first link in the chain of calculations.

17 ÄSPÖ HARD ROCK LABORATORY

The Äspö Annual Report /17-1/ provides a detailed description of the achievements for 1996 and the reader is referred to this publication for further information.

17.1 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 17-1. 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 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 17-2. 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 17-3.

The work with the Äspö Hard Rock Laboratory, Äspö HRL, has been divided into three phases: the pre-investigation phase, the construction phase, and the operating phase.

During the **Pre-investigation phase, 1986-1990**, 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. This phase also included planning for the construction and operating phases. These investiga-

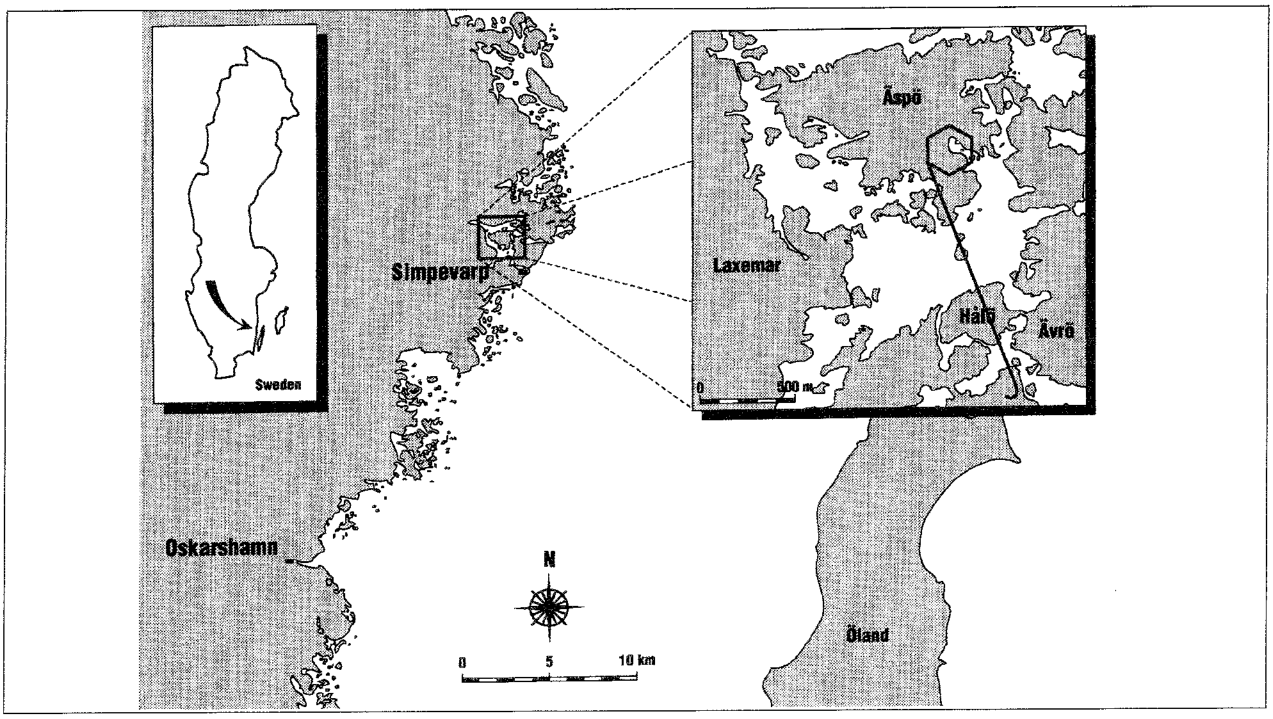


Figure 17-1. Location of the Äspö HRL.

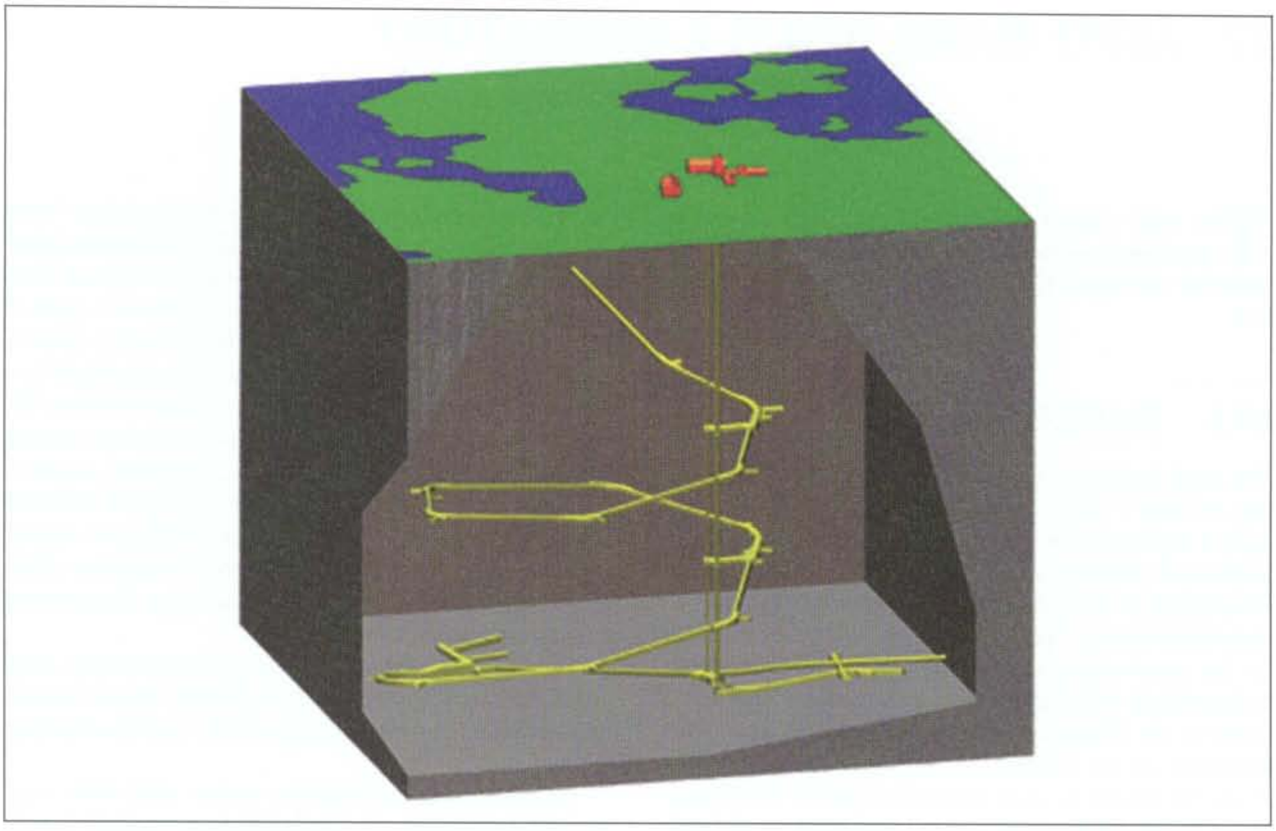


Figure 17-2. Schematic design of the Äspö HRL. The lower part of the facility has been excavated by a 5 m diameter Tunnel Boring Machine.

Figure 17-3. Aerial view of the Äspö Research Village.

tions have been summarized in Technical Reports /17-2 – 17-7/.

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) Program 1992. Since then the program has been revised and the basis for the current program is described in SKB's RD&D Program 1995.

17.2 ÄSPÖ 96

The Äspö Project began 10 years ago with geoscientific investigations on Äspö and nearby islands. Since then, bedrock conditions have been investigated by several deep boreholes, the Äspö Research Village has been built and extensive underground construction work has been undertaken in parallel with comprehensive research. This has resulted in a thorough test of methods for investigation and evaluation of bedrock conditions for construction of a deep repository. The construction of the Äspö Hard Rock Laboratory was finished in 1995. To mark this important milestone in the history of the Äspö Hard Rock Laboratory SKB organized "Äspö 96" as a combined 10-year anniversary and inauguration of the Äspö Hard Rock Laboratory facilities. Äspö 96 was attended by about 140 distinguished guests. The current status of nuclear waste management in Sweden and the role of the Äspö Hard Rock Laboratory for the Swedish and international nuclear waste programs was addressed by Carl-Erik Nyquist, President and CEO Vattenfall AB, Sten Bjurström, President of SKB, Maurice Allegre, Chairman of ANDRA, and Kjell Pettersson, Chairman of Oskarshamn Municipal Council. The Äspö Hard Rock Laboratory was inaugurated by Erik Krönmark, the Governor of Kalmar County. After the ceremony there was a guided tour of the underground facilities showing the current research projects in progress.

The Second Äspö International Seminar was held the following day in Stockholm. At the seminar, which was open to interested scientists, the results from the ten years of research at Äspö and the planned work at Äspö HRL was presented and discussed. The seminar was attended by about 120 scientists.

As part of "Äspö 96" a book, "The Äspö Hard Rock Laboratory – 10 Years of Research", was produced summarizing the results from ten years of research and plans for the future. The book was handed out to the participants of Äspö 96 and the Second Äspö International Seminar. The book has also been distributed to the individuals and organizations that normally receive SKB's Technical Reports. Additional copies can be obtained from SKB.

17.3 INVESTIGATIONS AND EXPERIMENTS – NEW RESULTS 1995

17.3.1 Allocation of experimental sites

The rock volume and the available underground excavations have to be divided between the experiments performed at the Äspö HRL. The allocation of experimental sites have been made to reduce interference between different experiments. The current allocation of experimental sites within the Äspö HRL is shown in Figure 17-4.

A permit to use short-lived radioactive nuclides for the TRUE, Radionuclide retention, and Long Term Tests of Buffer Material experiments has been obtained from the Swedish Radiation Protection Institute (SSI).

17.3.2 Verification of pre-investigation methods

The main aim of this stage goal is to demonstrate that investigations on the ground surface and in boreholes provide sufficient data on essential safety-related properties of the rock at repository level.

Predictions of a large set of rock properties were made prior to commencement of excavation of the underground facility /17-5/. These predictions were structured according to different geometrical scales and different key topics. During construction of the facility these predictions have been checked against the data collected during construction work. Comparison of model predictions and observations during construction is the basis for evaluation of characterization methods and model concepts used in the pre-investigation phase. The evaluation covers strategy for the pre-investigation, methods for data collection, analyses, predictions and evaluations.

Work on the final reports for this Stage Goal is in progress. There will be five final reports. One providing an overview of the investigations performed at Äspö and the surrounding region during the first 10 years. Three reports on the comparison of predictions based on pre investigations and outcome. Finally, there will be a report presenting the current model of Äspö based on investigations performed to date. The reports have been reviewed and are currently being updated before they will be submitted for a final review.

Regional and site scale three-dimensional groundwater flow models have been produced for Äspö. The model takes density variations due to varying salinity into account and makes use of a novel algorithm for treating the unsaturated zone. The new algorithm has provided realis-

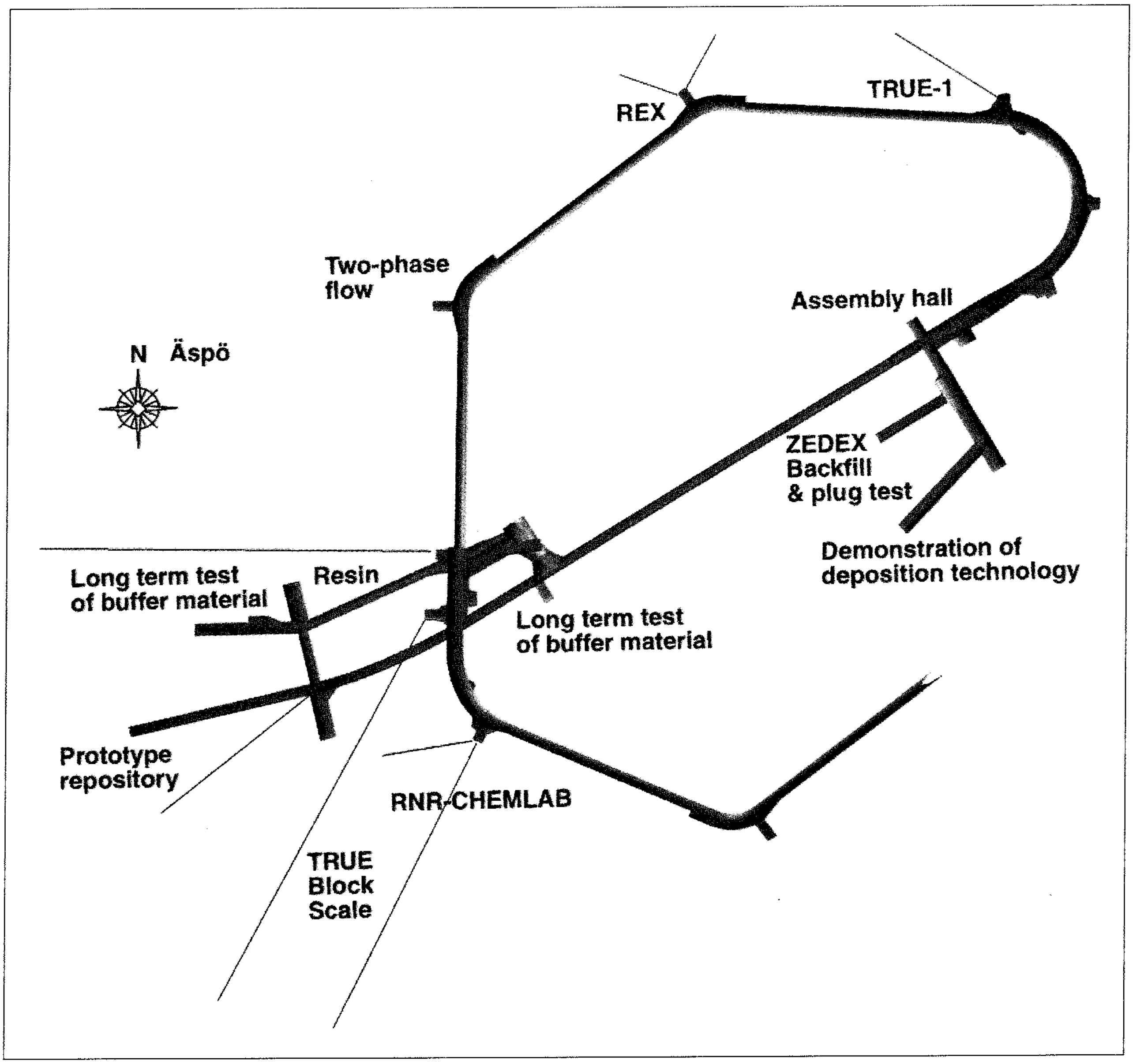


Figure 17-4. Underground excavations at the 300 – 450 m levels and current allocation of experimental sites.

tic numbers for groundwater recharge. The regional model is used to compute the boundary conditions for the site model.

17.3.3 Methodoloy for detailed characterization of rock underground

The detailed characterization of a repository will encompass investigations during construction of shafts and tunnels to repository depth. Development and testing of methodology for detailed investigations is the main aim for stage goal 2.

ZEDEX

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 decided in 1994 to perform a joint study of disturbed zone effects. The project is named ZEDEX (Zone of Excavation Disturbance Experiment). Significant in-kind contributions to the project are also provided by BMBF and NAGRA.

The ZEDEX project was started in conjunction with the change of excavation method from drill & blast to tunnel boring that took place during the summer of 1994. The originally planned experimental activities were completed and reported in 1995 /17-8/. The analysis of results obtained showed that further data collection and more thorough analysis of existing data would be beneficial for a better understanding of the extent and properties of the disturbed zone for different excavation techniques. Hence, the project parties agreed on an extension of the ZEDEX Project including additional data collection, thorough analysis of available data and predictive modeling efforts.

The experiment has been 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". A number of boreholes were drilled axially and radially relative to the test drifts to assess the properties and extent of the EDZ, see Figure 17-5.

The hypothesis set out at the start of the ZEDEX Project was that near-field disturbance (at distance of less than 2 m from the drift wall) could be reduced by the application of an appropriate excavation methods and that far-field disturbance would be independent of the excavation method. The results from the ZEDEX Project show that a division in near-field and far-field is not appropriate. Based on ZEDEX results the following division has been found more appropriate:

- there is a **damaged zone** closest to the drift wall dominated by changes in rock properties which are mainly irreversible, and
- there is **disturbed zone** outside the damaged zone dominated by changes in stress state and hydraulic head and where changes in rock properties are small and mainly reversible.

There is of course a gradational change in rock properties and rock stress with distance from the rock wall and there is hence no distinct boundary between the two zones.

The ZEDEX experiment has been performed in a rock mass with low stresses which has resulted in a mainly elastic behavior and no induced damage due to stress concentrations at the drift perimeter. The damaged zone caused by the excavation methods applied has been identified by several measurement techniques. Monitoring of AE-events is the most sensitive method which indicates minor damage due to crack opening and slip. Sparse AE-activity is not expected to correspond to measurable changes in rock properties. However, a large number of AE-events indicates intense micro-cracking and is expected to produce a macroscopically detectable increase in crack density. For the D&B drift significant AE-activity has been observed up to 1 meter from the drift wall while the corresponding extent for the TBM drift is a few tens of centimeters. Changes in seismic velocity indicate a larger increase in crack density. The dye penetration tests that have been performed in the slots sawed from the drift has shown the extent of macro fracturing, which in the floor of the D&B drift has extended to about 50 cm. The hydraulic measurements performed in the damaged zone has shown little change in permeability of the rock matrix. The larger permeabilities observed have been associated with the induced and pre-existing fractures.

The disturbed zone is characterized by elastic displacements and no induced fracturing. There are only very few AE-events observed in the disturbed zone and these have been interpreted to correspond to slip on existing fractures. The AE-event density is also similar for both the TBM and D&B drifts. The hydraulic tests performed before and after excavation have not revealed any significant changes in hydraulic properties due to excavation.

The current view of the characteristics and extent of the damaged and disturbed zones are summarized in Figure 17-6.

The results from ZEDEX indicate that the role of the EDZ as a preferential pathway to radionuclide transport is limited to the damaged zone. The extent of the damaged zone, which is the hydraulically significant part, can be limited through application of appropriate excavation methods. A limited extent of the damaged zone should also make it feasible to block pathways in the damaged zone by plugs placed at strategic locations.

Rock Visualization System

The Rock Visualization System (RVS) is developed to obtain a tool for interactive 3D interpretation of characterization data collected in boreholes, tunnels and on the ground surface. 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. The RVS system is linked to SKB's site characterization data base (SICADA) and it will hence be possible to trace all data that have been used to build a model. The system can also be used for layout of repository tunnels. The realization phase of the project started in March 96 and is still in progress. The system will be put into use during spring 97.

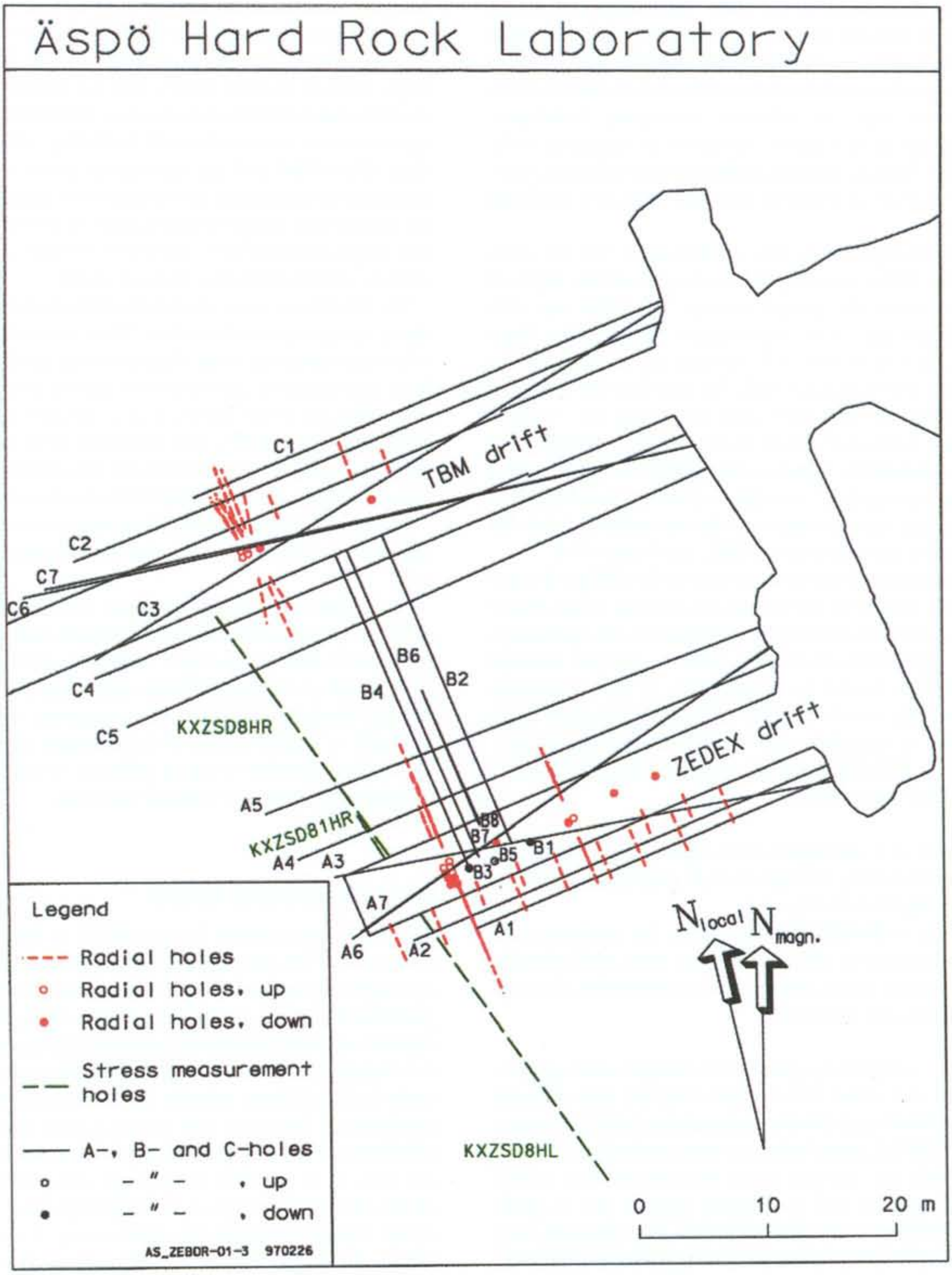


Figure 17-5. Configuration of test drifts and investigation boreholes for the ZEDEX study.

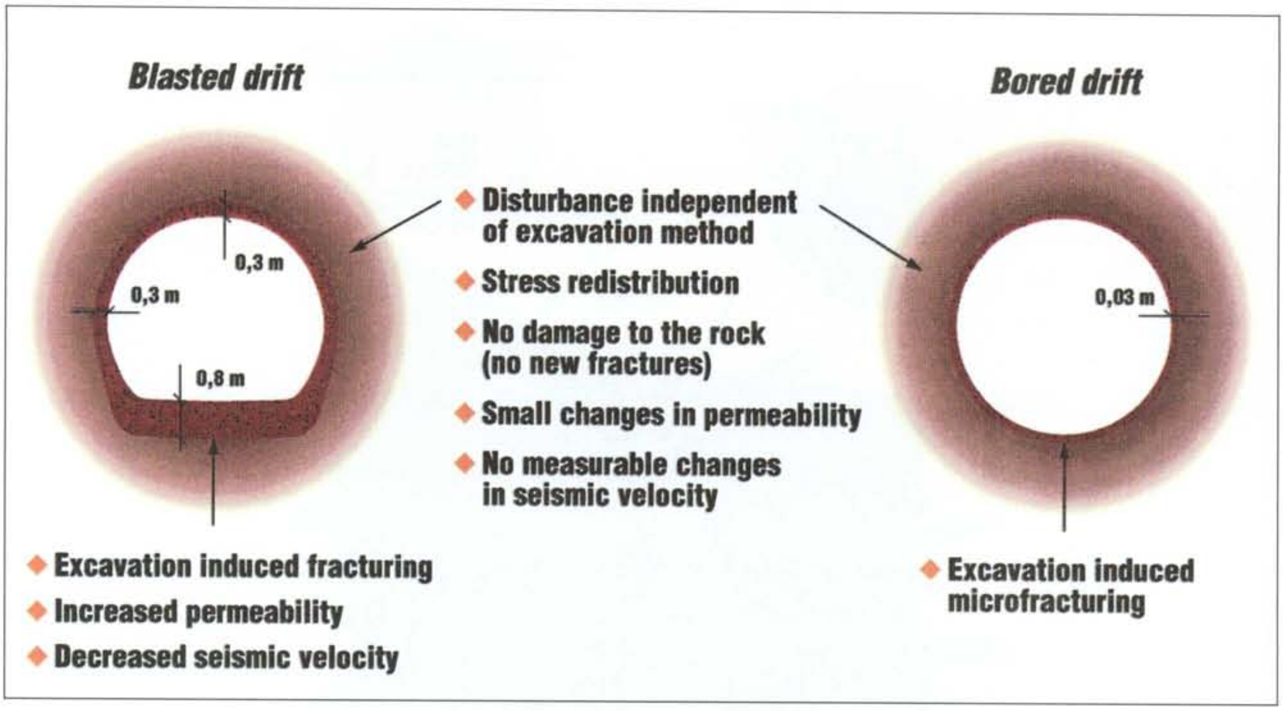


Figure 17-6. *Summary of the main findings of the ZEDEX Project. The extent of the damage zone is significantly greater in the drift excavated by blasting compared to the drift excavated by a tunnel boring machine.*

17.3.4 Test of models for groundwater flow and radionuclide migration

The rock surrounding the repository constitutes a natural barrier to release of radionuclides from a deep repository. The most important function of the natural barrier is to provide protection for the engineered barriers through stable chemical and mechanical conditions and to limit transport of corrodants and radionuclides through slow and stable groundwater flux through the repository and reactions of radionuclides with the host rock. This Stage Goal includes projects with the aim to evaluate the usefulness and reliability of different models and to develop and test methods for determination of parameters required as input to the models.

Fracture Classification and Characterization

The objectives of the Fracture Classification and Characterization project are to develop methodology for characterization of fractures with respect to tectonic evolution, infillings and wallrock alteration and to use this information for classification of fractures in terms of their importance for radionuclide transport. Detailed characterization has been made of 88 water-conducting fractures that intersect the main access tunnel. Most of the faults dip steeply and strike directions are NW-SE (dominant) and NE-SW (subordinate). Many of the faults follow pre-existing structural inhomogeneities, such as ductile shear-zones and lithified cataclastic shear-zones.

The only striking difference between individual water-conducting features is the internal fault geometry. No other distinguishing criteria (such as lithologic domains, mineralogy of fracture infills, transmissivity etc) were identified /17-9/. On the basis of the geometric arrangement of master faults and splay cracks five types of water-conducting features could be distinguished. Both observations and theoretical principles indicate that the internal geometry on which the classification is based is not a unique characteristic of a fault, i.e. the type may vary along the strike of a fault. The length of segments of the same type is in the range of meters to many decameters. The application of the classification scheme is limited to small-scale considerations. For large scale transport, the results indicate that due to a common genetic history, water flow in the underground of Äspö is dominated by one single family of water-conducting features.

TRUE – Tracer Retention Understanding Experiments

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 program has been devised for tracer tests on different scales. The program has been given the name Tracer Retention Understanding Experiments (TRUE). The experimental program is designed to generate data for conceptual and numerical modeling at regular intervals.

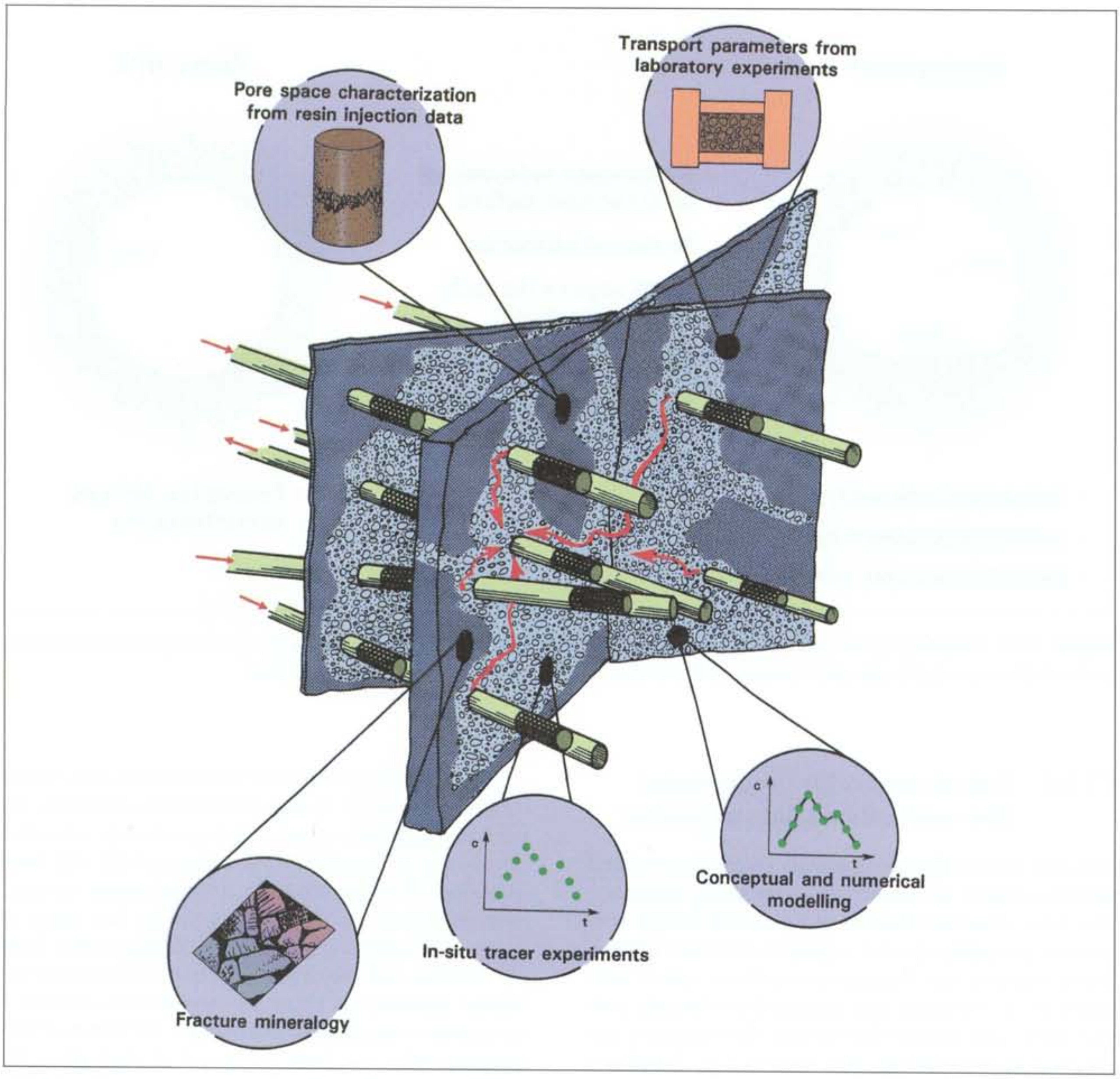


Figure 17-7. *Principal outline and components of the TRUE-1 experiment.*

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 modeling work /17-10/.

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 /17-11/, see Figure 17-7. 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 1996 a series of tracer experiments in radially converging (RC-1) and dipole flow configuration (DP1-DP4) have been performed in the feature selected for testing, Feature A, using conservative fluorescent tracers and metal complexes. These tests have also been subject to blind predictions by the Äspö Task Force. Further, a set of complementary tests have been performed with the objective of providing final support for the planned tests with sorbing tracers.

The radially converging tracer test (RC-1) was initiated in mid January 1996 using a steady flow of 0.2 l/min from a section in borehole KXTT3 where it intersects feature A. Tracer injections were performed in the four surrounding boreholes where they intersected the same feature. The experiment showed breakthrough from the two injec-

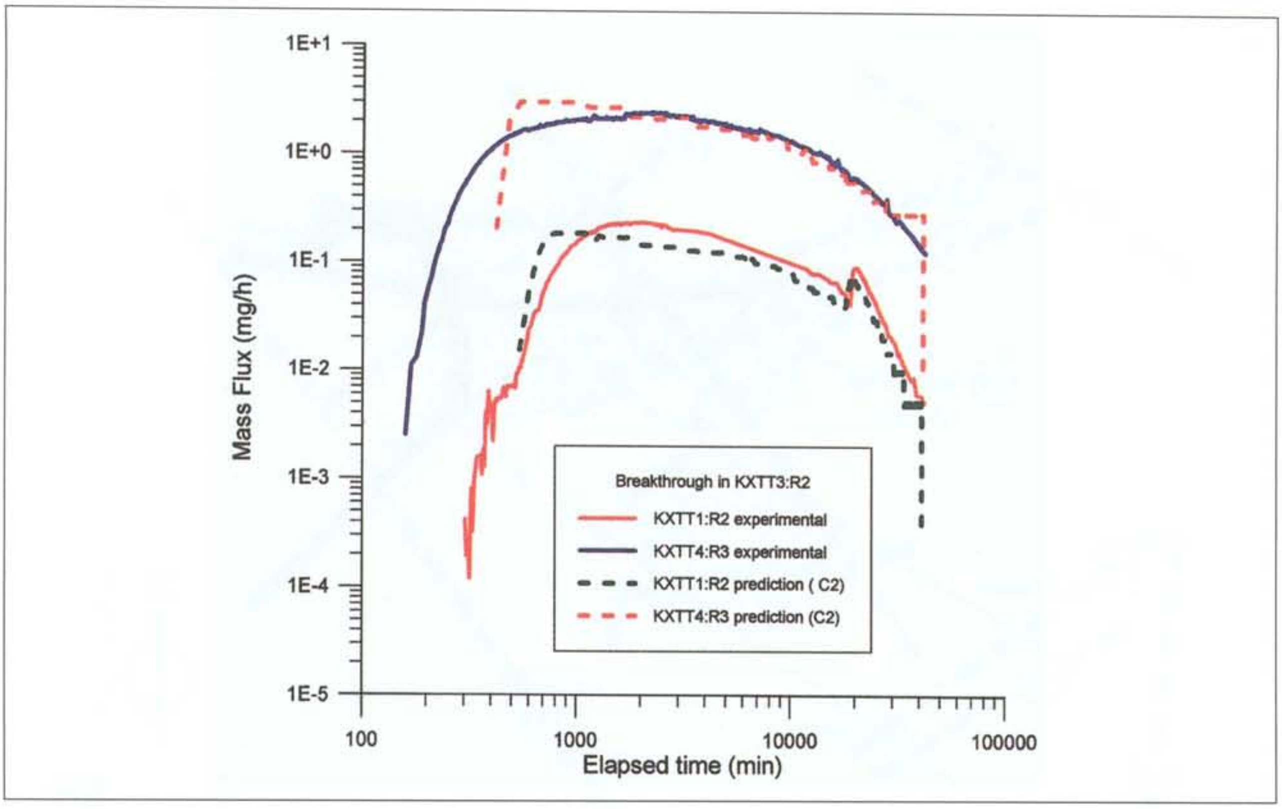


Figure 17-8. *Tracer breakthrough for radially converging tracer tests. Comparison between predictions made by SKB TRUE Project team and the corresponding experimental results.*

tion sections in KXTT1 and KXTT4 with mass recoveries of 91 and 97%, respectively, see Figure 17-8. The breakthroughs show little or no effects of processes, i.e. the breakthroughs are simply translations in time of the injection signal. No breakthrough was observed from the remaining two injection sections in KXTT2 and KA3005A, which are located farthest away from the pump hole. A subsequent stepwise increase in the pump flow to 400ml/min and finally 3.4 l/min, enabled breakthrough from the remaining two injection sections, but with very low mass recoveries. A preliminary comparison between model predictions made by the Äspö Task Force and experimental results, shows that most modeling teams predicted breakthrough from all four injections, although some teams predicted distinctly lower mass recoveries from the two injections which in-situ did not produce breakthrough. The breakthrough times predicted by the modeling teams are also in accord with those observed in the experimental results.

An experimental plan for the planned tests with sorbing tracers have been prepared. The basic idea is to perform a series of three tracer tests with one truly non-sorbing tracer (tritiated water), and a selection of weakly sorbing (Na, Ca, Sr) and moderately sorbing tracers (Rb, Ba, Cs). The tests will tentatively be performed in a radially converging flow field at Q=400, 200 and 100 ml/min. The test geometry is KXTT1->KXTT3 or KXTT4->KXTT3. During late March and April 1997, site preparations and pre-tests will be carried out. The tests with sorbing tracers are scheduled to start in May and will continue for a duration of three to four months.

The objective of the Resin technology development is to establish a technique by which a description of the pore space of a feature investigated with tracer tests can be mapped by epoxy resin injected into the feature /17-12/. The pore volume is measured in a number of sections or slices of the fracture using a combination of photographic and microscopic techniques and subsequent image processing. The obtained data is planned to be used to reduce uncertainties in the description of the heterogeneity of the studied feature.

Batch sorption and through diffusion measurements have been performed on material from the feature in which the TRUE-1 tracer experiments have been conducted. Based on the batch and diffusion experiments with Na, Ca and Sr it can be concluded that:

- Prediction of sorption capacity, K_d, for weakly ion exchangeable sorbing tracers in laboratory diffusion experiments, based on laboratory batch experiments, may overestimate the matrix diffusion related K_d if small size fractions are used. The comparisons show that large size fractions are the most representative for estimating the K_d in bulk rock samples. This is attributed to the fact that these fractions consist of primarily of polymineralic grains.

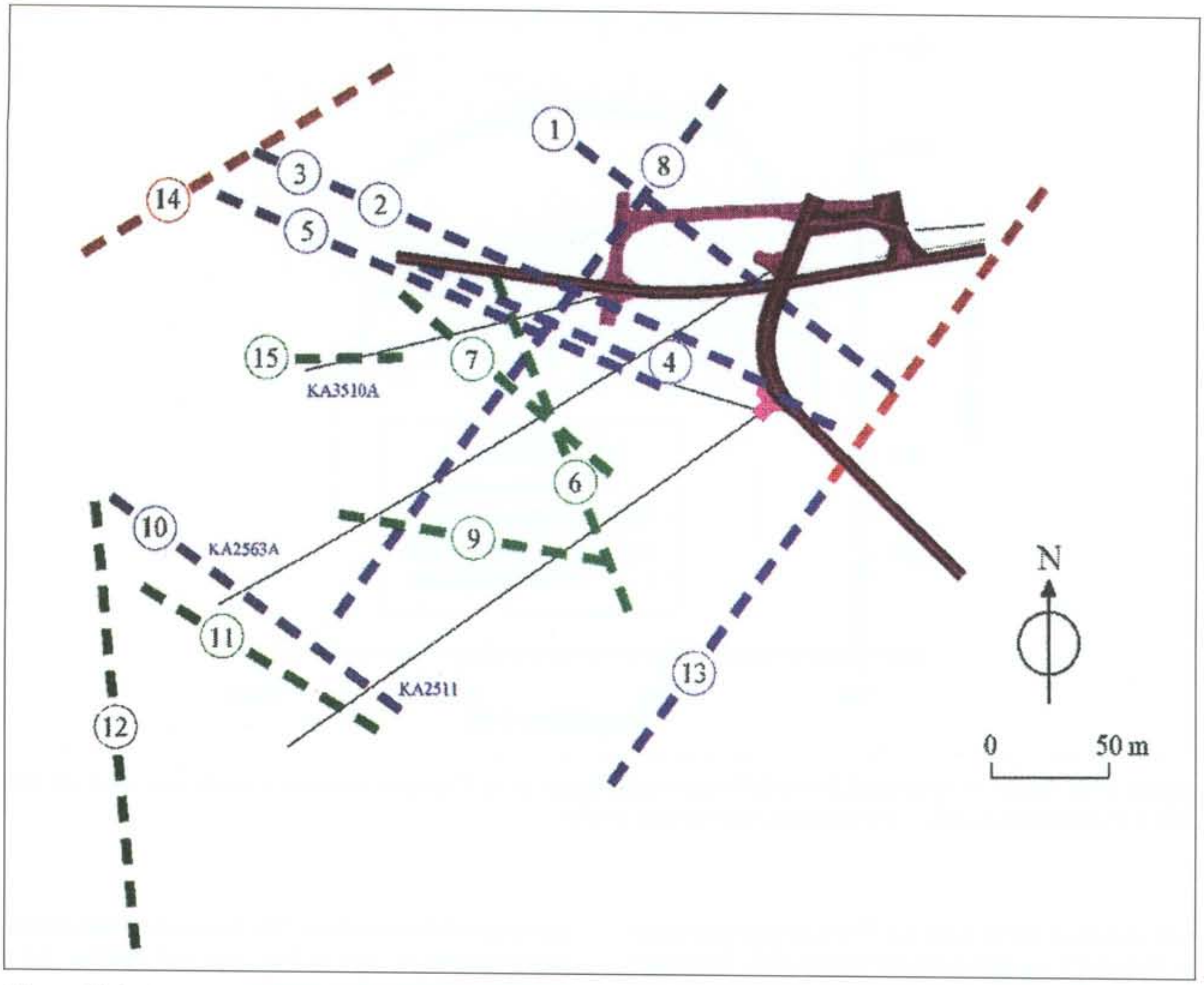


Figure 17-9. Horizontal section through structural-hydraulic model updated with characterization data collected in KA2511A, KA2563A and KA3510A. Red = certain, blue = probable, green = possible. Numbering refers to internal labelling of fracture zones, e.g. Zone #13 = NE-2 and zone #14 = EW-1.

- The observed differences in diffusivity, D_e between HTO and Na^+, Ca^{2+} or Sr^{2+} are attributed to differences in water diffusivity, D_w. No influence of the material properties on D_e have been observed.
- There is a tendency of decreasing De and α with increasing cell length, indicating that the transport porosity (and the storage porosity) decreases with increasing cell lengths. It is probable that the shorter samples overestimates the De and α since the length of the samples are of the same size as the scale of heterogeneity of the samples.

At the Pilot Resin site a test of injection techniques and equipment and resins is carried out in-situ in a fracture system close to the drift wall. Site characterization work has been carried out at the site. During August-September 1996 a series of three resin injections were performed. The injection sequence started with injection of dye-labelled water in a given section with all other holes closed. The purpose of the dye was to tag the flow paths participating in groundwater flow, this to enable comparison between the porosity impregnated by the resin. Subsequently, dye-labelled isopropyl alcohol was injected to ensure good wettability and avoidance of fingering effects. Finally injection of dye-labelled resin was performed. The injection periods lasted between 5 – 7 hours with injection pressures kept between 30 – 55 bars. Simplistic calculations predict the areal spread of the resin to be in the order of square meter(-s). Subsequent to the injections, ten short 56 mm boreholes have been drilled with the aim of obtaining a picture of the areal spread of resin. More than 50% of the drilled exploratory holes carried resin. Then three 200 mm cores were drilled parallel to three of the exploratory holes. There were problems obtaining intact core samples and a refined drilling technique using a smaller diameter pilot hole had to be developed.

The main objective of the TRUE Block Scale Experiment is to increase understanding and our ability to predict tracer transport in a fracture network over spatial scales of 10 to 100 m. The TRUE Block Scale Experiment has been initiated as a joint project between ANDRA, Nirex, Posiva, and SKB. The total duration of the project is approximately four years from the start in July 1996.

During the spring of 96 a structural-hydraulic model of the experimental level and a data set for scoping calculations were compiled. Alternative locations of the target block were considered based on the developed model. A pilot borehole, KA2563A, has been drilled to investigate the properties of the selected experimental volume in the southwestern corner of the laboratory, south of the TBM tunnel. The borehole intersected two water bearing features which produced significant water inflows to the boreholes. The rock volume beyond these features contained no major water bearing features and the rock volume was considered suitable for the planned experiment. Cross-hole seismic measurements and flow logging has been performed to further characterize the block. The updated strutural model is shown in Figure 17-9.

The REX-experiment

The REX project focuses on the reduction of oxygen in a repository after closure due to reactions with rock minerals and microbial activity. A field experiment will be performed where the consumption of oxygen in contact with a fracture surface will be studied. The field study is supported by laboratory experiments to determine oxygen reaction mechanisms and kinetics. Laboratory measurements are in progress and preparation have been made for the field experiment which will start in 1997. Preliminary measurements of dissolved methane and hydrogen in Äspö groundwaters have been performed. They have been combined with the measurements of bacteriological oxygen consumption in Äspö groundwaters. These results show that oxygen may be consumed by methanotrophic bacteria in a closed nuclear waste repository.

Radionuclide retention

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 17-10. 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

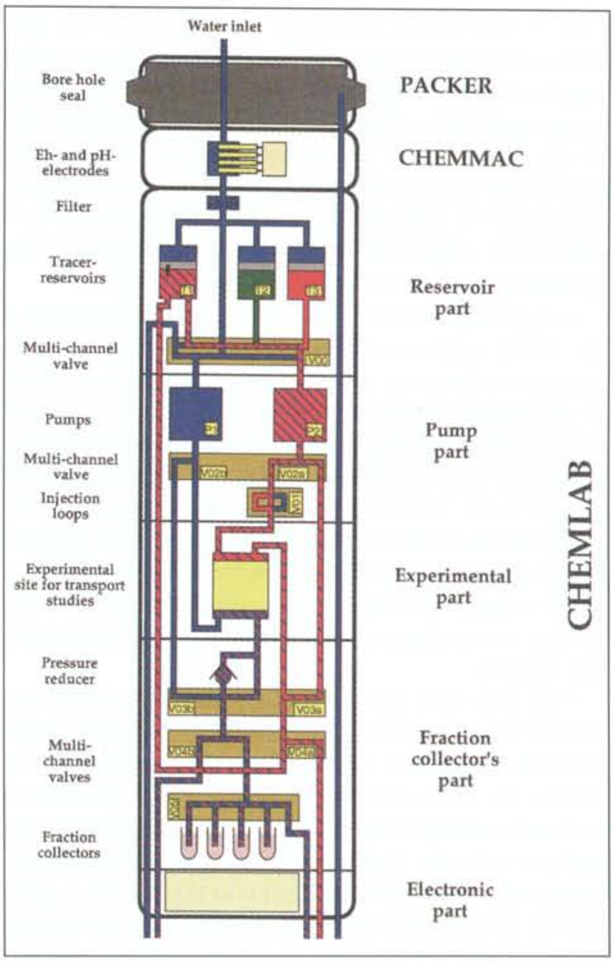


Figure 17-10. Schematic illustration of the CHEMLAB Probe.

transport in an individual fracture. In addition, the influence of naturally reducing conditions on solubility and sorption of radionuclides will be tested.

The construction of the CHEMLAB probe is completed and the probe was delivered to Äspö HRL in April 1996. An inactive test was performed to simulate a typical experiment and get experience on how to operate the probe. Some equipment problems were identified and modifications were subsequently made to the probe. The first experiment with radioactive isotopes was diffusion of ^{131}I and ^{57}Co in bentonite. The experiment started in November but had to be terminated early because a sensor indicated a too high pressure in a flow line. The pumps were automatically stopped. The CHEMLAB probe had to be taken out of the borehole and sent to the manufacturer for service.

Degassing of groundwater and two-phase flow

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 due to too low gas contents of the groundwater to cause degassing effects /17-13/.

A degassing and two-phase flow test was conducted at the TRUE resin site /17-14/. The objectives of the pilot injection-withdrawal tests with gas saturated water were to investigate whether degassing effects can be observed in borehole tests at higher gas contents and lower fracture transmissivities than the gas content and transmissivity of the previously performed pilot test. In the test water with a gas contents of about 17% was injected into the fracture. A flow reduction of 50% was observed when the pressure in the withdrawal hole was reduced to atmospheric. Degassing is considered to be the most likely explanation for this behavior.

Modeling work has shown that degassing effects are limited to low-pressure zone which has an extent on the order of centimeters for boreholes and on the order of meters for drifts /17-15/. Trapping of gas bubbles in fractures is strongly dependent on fracture roughness. The analysis indicates that bubbles of up to one centimeter length may be trapped at gradients as high as 10^4, implying that for boreholes and drifts, bubbles may get trapped throughout the low-pressure zone, provided that the extent of this zone is sufficiently large.

Laboratory tests on 200 mm diameter core samples showed significant reductions in fracture transmissivity, up to 90%, when a separate gas phase was introduced into the fracture plane. Fracture planes with relatively smooth and tabular apertures recovered their original fracture transmissivities fairly quickly, while the fracture planes that were characterized by a rough or variable aperture required several hours for the fracture transmissivities to return to their original values under single phase flow conditions. A full set of experiments have been completed using the Large Physical Model, an artificial fracture with uniform roughness with a size of about 2 by 2 m. The test results show non-linear effect due to turbulens close to the withdrawal hole and large flow reductions due to two-phase flow effects.

Task Force on Groundwater flow and transport of solutes

A "Task Force" with representatives of the project's international participants was formed in 1992. 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 modeling of groundwater flow and solute transport in fractured rock. The work in the TF is tied to the experimental work performed at the Äspö HRL and is performed within the framework of well defined and focused Modeling Tasks. The TF group should attempt to evaluate different concepts and modeling approaches. Finally, the TF should provide advice on experimental design to the Project Teams, responsible for different experiments.

The evaluation of the modeling work on Task No 3, the hydraulic impact of the tunnel excavation at Äspö, is on-going. The first part may be regarded as a direct continuation of Task No 1 and addresses how robust are site scale groundwater flow models based essentially on pumping tests results and does extrapolation from such models provide reasonable results. The second part uses the data set available from the period during excavation of the Äspö tunnel for improving the site scale groundwater flow models. The organizations taking part in Task 3 are listed in Table 17-1.

Task No 4 has been the main modeling effort within the Task Force during 1996. The scope of Task 4 has been to perform forward modeling of the radially converging tracer test (RC-1) and of the dipole tracer tests (DP1-4) carried out as part of the TRUE experiments. The model predictions have later been compared with experimental results.

Task 4 constitutes one of the few real blind predictive modeling exercises ever conducted in the field of tracer transport in fractured media. No thorough evaluation of the modeling performed has been done so far but a preliminary analysis of modeling of the RC-1 test show that /17-16/;

- Quite an impressive amount of modeling work has been performed considering the large amount of data available and the time constraints.
- Comparing the experimental result with the simulations from the eight groups, it is evident that the flow system/boundary conditions are not completely understood.
- The predicted breakthrough times are in the right order of magnitude, in some cases very good, for two out of four tracer tests.

17.3.5 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 investiga-

Table 17-1. Organisations and modelling groups of Task No 3, the Äspö tunnel experiment. SKB ICR means the Äspö International Co-operation Report Series.

ORGANI-SATION	MODELLING TEAM	REPRESEN-TATIVE	TASK 3A	TASK 3B	REFERENCE
ANDRA	Itasca Consultants	Billaux	X		DRAFT in French
CRIEPI	CRIEPI	Tanaka		X	SKB ICR 96-07
PNC	Golder Associates	Dershowitz		X	FINAL DRAFT
PNC	Hazama Corporation	Yamashita	X[1]		FIRST DRAFT
Posiva	VTT Energy	Mészáros	X	X	SKB ICR 96-06
SKB	CFE	Svensson	X[2]	X[2]	SKB HRL PR 25-91-03
UK Nirex	AEA Technology	Holton	X[1]	X	FINAL DRAFT

[1] Due to several reasons these modelling studies are not 3A as originally meant. Actually, a partly updated geological structural model has been utilised.

[2] This excercise was not performed within the framework of the Äspö Task Force. The modelling was part of the Äspö Project.

tions of the interaction between the engineered barriers and the rock will be carried out, in some cases over long periods of time.

Backfill and Plug Test

The Backfill and Plug Test includes tests of backfill materials and emplacement methods and a test of a full scale plug. It will be a test of the integrated function of the backfill material and the near field rock in a deposition tunnel excavated by blasting. The field compaction tests made in 1995 showed that a new compaction equipment was required /17-17/. During 1996 a vibrating plate has been designed and built. The vibrations are produced by the oil hydraulic of the carrier, which will be a small flexible rebuilt digging machine. The vibrating plate is equipped with a complete bottom plate, that is shaped to suite compaction close to the roof and walls with inclined compaction. The plug for the Backfill and Plug Test in the ZEDEX drift has been designed. Laboratory tests on backfill materials have been running and the development and testing of equipment for measuring THM-processes in backfill and buffer materials have continued during 1996.

Prototype Repository

The Prototype Repository Test is focused on testing and demonstrating repository system function. A full scale prototype including four deposition holes with canisters with electric heaters and highly compacted bentonite will be built and instrumented. The function of the prototype will then be monitored for several years. Certain activities aimed at contributing to development and testing of the practical, engineering measures required to rationally perform the steps of a deposition sequence are also included. Detailed planning of the Prototype Repository has been continued during 1996. The tentative layout of the Prototype repository is shown in Figure 17-11.

Demonstration of repository technology

Demonstration of deposition and retrieval of canisters will be made in the Äspö Hard Rock Laboratory. The demonstration project complements the Prototype Repository and the Backfill and Plug Test. The demonstration of deposition technology will be made in a new tunnel south of the ZEDEX drift excavated by drill and blast. Excavation of the new tunnel began in November 1996.

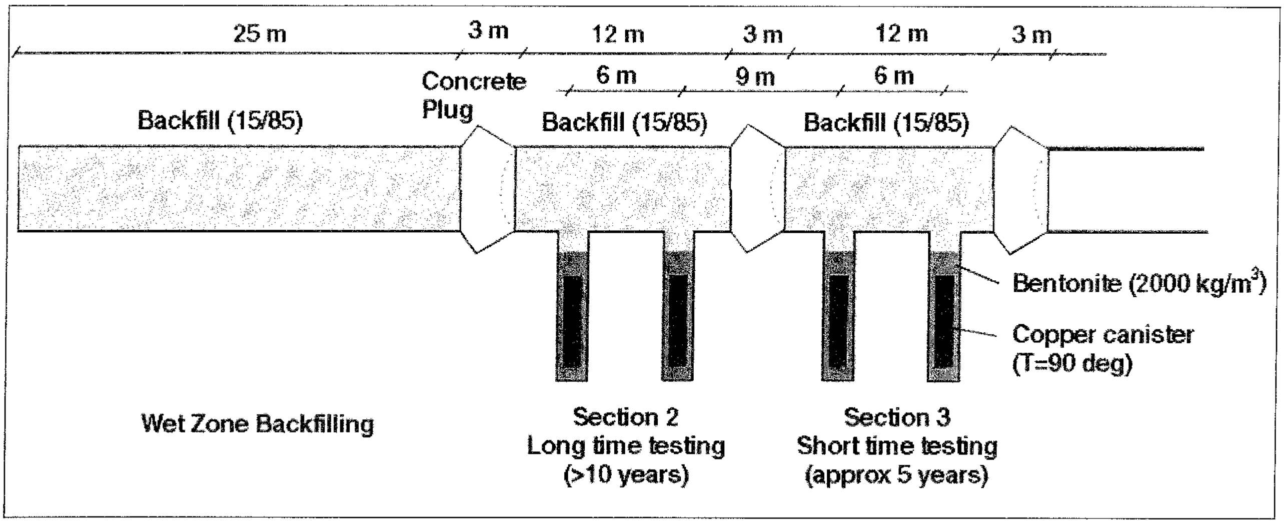


Figure 17-11. *Tentative layout of the Prototype Repository.*

Long Term Tests of Buffer Material

The Long Term Tests of Buffer Material aim to validate models of buffer performance at standard KBS-3 repository conditions, and at quantifying clay buffer alteration processes at adverse conditions. In this context adverse conditions have reference to e.g. super saline ground water, high temperatures, high temperature gradient over the buffer, high pH and high potassium concentration in clay pore water. Further, related processes regarding microbiology, radionuclide transport, copper corrosion and gas transport are also studied. Prefabricated units of bentonite blocks surrounding a copper tube with an electrical heater have been placed in vertical boreholes. The boreholes have a diameter of 30 cm and a length of about 4 m. The first test parcel, S1, designed to simulate normal repository conditions, was put in the test hole in October and its temperature has successively been increased to 90°C. The second test parcel, A1, was inserted into its borehole in mid November. The temperature in this borehole will be increased to 120°C in order to test the buffer under adverse conditions, e.g. super-saline groundwater, high temperatures, high pH, and high potassium concentration in clay pore water. Data concerning temperature, total pressure, pore water pressure and humidity in the two test parcels have been produced during test period and no major divergence from expected values has been found. Figure 17-12 shows the temperature distribution in parcel S1 at the end of December 1996. The swelling pressure of the bentonite was around 4 MPa in both tests and full water saturation has been achieved in all positions equipped with moisture gauges.

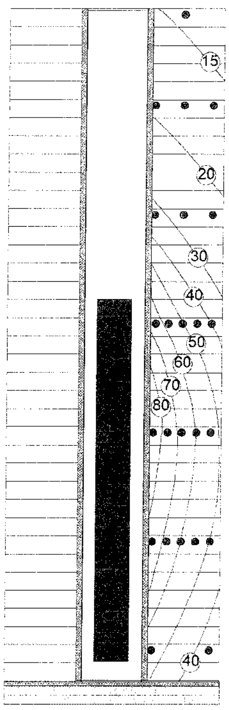


Figure 17-12. *Principle drawing of parcel S1 showing thermocouple location (dots) and isotherms for the conditions on December 31, 1996. The horizontal scale is enlarged four times compared to the vertical scale.*

Cracks in rock caused by mechanical excavation

The work with modeling the cracks caused by cutters or bits at mechanical excavation in crystalline rocks has continued with studies on the influence of mechanical properties of rock on the indentation depth and crack length. The main factors governing the indentation event have been identified and functional relationships have been established relating either the indentation depth or the length of radial/median cracks to the various quantities characterizing the physical event, namely the indentation force, the shape and the size of the indenter and the properties of the rock /17-18/.

Studies of induced cracks in the TBM tunnel wall in Äspö Hard Rock Laboratory and in borehole wall in the research tunnel at Olkiluoto, Finland has been made. Rocks at these two places are diorite and gneiss respectively. The basic crack types defined by laboratory indentation tests were found in the studied samples but with some variations. For the different excavation methods TBM caused deeper and longer cracks in the walls than was caused by the button cutters of the blind hole boring machine. It is remarkable that only few and short cracks were found in the side walls independent of method. The densely cracked layer is less than 10 mm deep and the subsurface cracks do not penetrate deeper than 10 to 20 mm into the side wall.

17.3.6 Data management

One of the main objectives with the Äspö Hard Rock Laboratory is to test and develop 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 site characterization.

The new database, SICADA, developed by SKB which was put into operation in 1995 will be one of SKB's most important database systems. The system has been further developed during 1996 and the following applications currently exist; Diary, Finder, Retriever, Project, and WWW-Retriever. The WWW-Retriever makes it possible to retrieve data from SICADA through SKB's internal home page on SKB's Intranet.

17.4 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. Eight organizations from seven countries are currently (December 96) participating in the Äspö Hard Rock Laboratory in addition to SKB. They are:

- Atomic Energy of Canada Limited, AECL, Canada.
- POSIVA OY, Finland.
- Agence Nationale 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.
- Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle, NAGRA, Switzerland.
- Bundesministerium für Bildung, Wissenschaft, Forschung und Technologie BMBF, Germany.

Multilateral projects are established on specific subjects within the Äspö HRL program. These projects are governed by specific agreements under the bilateral agreements between SKB and each participating organization. The ZEDEX project and the TRUE Block Scale Experiments are examples of such projects.

18 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.

The possibility for partitioning and transmutation is still attracting considerable interest. SKB has since the early 1990s supported work in this area at the Royal Institute of Technology (KTH) in Stockholm and at the Chalmers Institute of Technology (CTH) in Göteborg. The support from SKB is increased from 1997 and is coordinated with other work by the various groups in Sweden active in this field. The work supported by SKB at KTH is emphasized on systems and safety studies and at CTH on studies of processes for partitioning. Both groups have a broad international cooperation.

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.

18.1 KTH WORK ON ACCELERATOR-DRIVEN TRANSMUTATION

The work on Accelerator-driven Transmutation at the Royal Institute of Technology aims to assess the potential of the Accelerator-driven Systems (ADS) for the nuclear waste transmutation on the safe and economical way. ADS has a potential to overcome the drawbacks with the reactor based transmutation of nuclear waste. The Accelerator Transmutation of Wastes (ATW) concept has as its goal the realization of a high-power nuclear system, driven by a powerful neutron source, external to its subcritical core, not relying on delayed neutrons and reactivity changes for its power management.

An ATW system consists of an intense neutron source which is driven by a high power proton accelerator (1 – 2 GeV, 20–200 mA). The neutron source is surrounded by a subcritical blanket containing the nuclear waste to be transmuted. The neutron flux in the blanket (10^{15-16} $cm^{-2}s^{-1}$) is 5 – 100 times higher than in a ordinary reactor which means that the fissile inventory in the blanket can be reduced accordingly compared to the conventional reactors. The waste is dissolved in e.g. molten lead-bismuth or molten salt (mixture of Li- and Be-fluorides) in different proposed Accelerator-Driven Transmutation Technologies (ADTT) concepts with fast or thermal neutron spectrum, respectively. Special proliferation resistant processes have been proposed to prepare the fuel for the ATW system from spent nuclear fuel or weapons plutonium and to remove the fission products (volatiles, transition metals and lanthanides) from the molten salt.

There are several options for utilization of accelerator-driven transmutation technologies, however, the KTH group is concentrating on an economical way of transmuting of long-lived nuclear waste. The economical way in reality implies concurrent energy production. The ATW concept can, due to its comparatively larger neutron availability and flexibility of operations, also destroy some nuclides that cannot be incinerated in critical reactors.

18.1.1 System studies

Since 1992 different ATW systems have been studied in order to assess transmutation performance, safety requirement, and economical and technological constraints. First systems based on heavy water moderator and fuel suspensions / solutions were assessed /18-1/. Those systems have, however, problems to fulfil basic requirements concerning performance safety and economics. Then the molten-salt based thermal neutron systems have been studied showing promising results /18-1, -2/. In the mean time the Japanese program OMEGA was focused on using fast neutron spectrum in liquid fuel molten-chloride systems or fast-breeder like subcritical devices with solid fuel and sodium cooling /18-1, -3, -4/. In 1995 Carlo Rubbia and his CERN-based group proposed a system based on solid fuel (Th fuel cycle for energy production and Pu/ minor actinide fuel for transmutation) and liquid lead coolant /18-5/. CEA-Cadarche has also started a broad research project (ISAAC and GEDEON) to assess the feasibility of ADS based on a fast neutron spectrum /18-1/. Thus it became necessary to carefully investigate the transmutation performance of ADS depending on the neutron spectrum of the system. The system studies in 1996 were focused on the impact neutron spectrum on the performance of ATW. An extensive analysis of the simulation results were performed and will be published /18-6/. The results of this analysis show that liquid lead as a coolant and neutron propagator opens interesting possibilities to keep constant transmutation rates for many isotopes in a wide range of concentration in the fuel. It makes it also possible to reach the resonance cross section region for some fission products like ^{99}Tc and ^{129}I.

18.1.2 Simulations, optimization and design of spallation target

One important component of an accelerator-driven transmutation system is an efficient target for neutron production. In principle, the most efficient neutron production is obtained when the high power proton beam from the accelerator is stopped in a heavy material like lead, bismuth, tungsten or uranium. The protons split the atoms of the heavy materials by a number of complex nuclear reactions (spallation reactions). Each proton produces a large number of neutrons (about 30 neutrons/ 1 GeV proton) which number depends on the energy of the incoming proton and to a less extent the type of heavy material.

Being a part of the extensive international collaborative effort the KTH group has decided to concentrate a part of its research and management activities to the spallation target design and manufacturing. In a cooperation with several international partners (Los Alamos (USA), CEA-Cadarache (France) and PSI (Switzerland)) they participate in an ISTC (International Science and Technology Center in Moscow) project to manufacture a 1 MW liquid lead-bismuth target. The title of the project is : *Pilot flow lead-bismuth target of 1 MW for accelerator-based systems (ISTC project # 559).* The cost for this project is 1 M$ and is fully covered by ISTC.

The purpose of the project is to develop a heavy metal flow target which posses the best features for producing neutrons at a high power proton accelerator. Thus, the key technical problems of a flowing lead-bismuth 20 MW-power target should be investigated. Such a technical base will be established by the design of a pilot lead-bismuth 1 MW-power target. It is planned that the pilot target will be tested at the LANSCE accelerator at LANL (LANSCE: 800 MeV, 1.5 mA linear proton accelerator) or as an alternative at the Paul Scherrer Institute (PSI), Villigen, Switzerland.

Extensive simulations have been performed to optimize the size and neutron production for such a target using the FLUKA-96 high energy transport code /18-7/. Also the radiotoxicity of the spallation products were investigated /18-8/.

The calculations of heat generation and potential material damages around the active part of the target are being performed in collaboration with Politecnica, Torino.

18.1.3 Other projects

The IAEA CRP (coordinated research project) to benchmark the calculational tools used in the ADS simulations started in the middle of 1996 and is in its final phase. Extensive calculations comprising spallation neutron source simulations (with FLUKA-96), neutron transport (MCNP) and burnup simulations have been performed in order to compare the results of the simulations of the relatively simple ADS with the results of other groups participating in this CRP (France, Italy, CERN, Germany). The results of this CRP will be published in few months.

Work has started to develop the integral computer code system to simulate fully the accelerator driven system beginning from the ingoing proton beam and ending on the radiotoxicity of the residual waste. This project which is planned to continue for several years in collaboration with Dubna (Russia), Karlsruhe (Germany), CEA-Cadarache (France) and LANL (USA) will give a powerful tool to perform the feasibility studies of ADS and could be one prerequisite for conceptual design of a demonstration experiment.

Royal Institute of Technology was one of the main organizers of the 2nd International Conference on Accelerator Driven Transmutation Technologies and Applications in Kalmar. The conference was attended by 207 participants from 24 countries /18-9/.

18.1.4 Experimental activities

In 1996 the KTH group started the collaboration with CEA-Cadarache in order to participate in their experimental program on subcritical configurations in the MASURCA reactor. The MASURCA zero-power critical facility in Cadarache allows to set up a wide range of critical or subcritical configurations with fast neutron spectrum, using a variety of different fuels and simulated coolant materials (like Na), in different proportions. For the MUSE-1 experiment an existing loaded core was used (fuelled with UO_2 - PuO_2; ratio of Pu/(Pu+U) approximately equal to 0.25, and with simulated Na coolant). In the central channel of the core it was possible to load a ^{252}Cf neutron source, which was located successively at three axial position. Starting from a critical configuration without external source, the core was made subcritical, by unloading some peripheral fuel elements.

The following measurements have been performed:

- sub-criticality level,
- ^{235}U fission rate radial and axial distributions,
- φ^* measurement /18-1/.

The experiments on MASURCA will be continued and the results will be published in 1997. The intention is to extend this collaboration.

18.2 CTH WORK ON PARTITIONING

The current research project on Partitioning and Transmutation (P&T) at the Department of Nuclear Chemistry, CTH, has the primary objective to investigate separation processes useful in connection with transmutation of longlived radionuclides in high level nuclear waste. Partitioning is necessary in order to recover and purify the elements before and after each irradiation in a P&T treatment. In order to achieve a high transmutation efficiency the chemical separation process used must have small losses to various waste streams.

At present, only aqueous based separation processes are known to be able to achieve the high recovery and separation efficiencies necessary for a useful P&T process. The engineering and operation experience from wet separation processes by far exceeds those of alternative processes based on melt refining, molten salt electrolysis, pyroprocessing, and volatility. It is realistic to believe that aqueous separation techniques will continue to be far ahead of the other possibilities for a long time to come. This belief is shared by other European projects and is the main reason why research on separation processes at the department is concentrated on aqueous/organic liquid-liquid extraction systems.

18.2.1 SOLVENT EXTRACTION RESEARCH

New extraction reagents should be selective, have a high loading capacity and be burnable without solid residues. This limits the possible elements in the extractant molecules to hydrogen, carbon, nitrogen and oxygen, the so called CHON-principle. However, the CHONprinciple has also a wider meaning. Addition of chemicals to the process solution should be limited to those already present in large amounts in high level waste from conventional PUREX reprocessing of spent fuel, i.e. water and nitric acid.

Long chained amines and quaternary ammonium salts constitute one group of possible CHON-reagents. Other useful reagent classes are derivatives of malonamides, of tri-pyridyl-triazine, of picolinamides, and various oligopyridines, e.g. terpyridin and is derivatives. In order to comply with the requirements of high loading capacity, reasonable viscosity and high solubility in suitable organic diluents, the molecular weight of the reagent must not be too high. Furthermore, part of the reagent structure should have some similarity to the diluent molecules in order to maximize solubility at a given molecular weight. On the other hand, the molecular weight of the reagent should be high enough to make the reagent little soluble in the aqueous phase.

Due to the similarity in chemical behavior of lanthanides and the trivalent actinides (americium and curium) various routes can be envisaged, e.g.,

- (i) coextraction of both element groups followed by stripping and separation of the element groups (or single elements) by a second reagent having sufficient separation power,
- (ii) extraction of trivalent actinides only by using a very selective reagent.

The main strategy in our research follows route (i) above.

Long chain quaternary amines

Aliquat-336 (tricapryl-methylammonium nitrate) is a quarternary ammonium salt that extracts neutral or negatively charged species with an anion exchange mechanism. The diluent used is a dialkyl substituted aromatic compound, 1, 3-diisopropylbenzene. In the experiments done, the distribution of elements between the two phases is measured and expressed in terms of the distribution ratio, ***D***.

$$D_M = [M]_{org,tot}/[M]_{aq,tot}$$

The distribution ratio for M is defined as the ratio of the total concentration of M in the organic phase, $[M]_{org,tot}$ to the total concentration of M in the aqueous phase, $[M]_{aq,tot}$. A distribution ratio greater than unity means that M is enriched in the organic phase whereas a distribution ratio less than unity means that M is enriched in the aqueous phase.

The extraction of metal cations with Aliquat-336 will depend on the charge and radius of the metal ion, i.e., the ability to form complexes with the ligand, the ligand concentration and the extractant concentration. By varying the oxidation state of the metal and the ligand concentration it is possible to separate different elements from each other.

Some distribution ratios as a function of the nitric acid concentration are shown in Figure 18-1 /18-10, 11/. The results are given at 0.20 molar Aliquat-336 in the organic phase, except for plutonium which is given at 0.05 molar. The tetravalent elements, Th^{4+}, Np^{4+} and Pu^{4+}, all have distribution ratios above unity. These highly charged elements easily form complexes with nitrate and can therefore be separated from lower charged cations. The trivalent elements, Ln^{3+} (lanthanides), Am^{3+}, Cm^{3+}, In^{3+} and Fe^{3+} , all have distribution ratios below unity. Since the charge is lower for these elements they do not form nitrate complexes as easily as the tetravalent elements. Distribution ratios for the hexavalent elements, uranium and plutonium, as uranyl and plutonyl ions are between those of the tetravalent and trivalent elements. This is due to the fact that the charge of the metal atom in MO_2^{2+} is higher than the effective charge of the ion. Technetium as pertechnetate, i.e. an anion, shows very high distribution ratios at low nitric acid concentration. In this case pertechnetate ions can extract directly in to the organic phase without forming nitrate complexes. Nitrate ions will compete with pertechnetate ions for extractant molecules and therefore the distribution ratio is decreasing with increasing nitric acid concentration. Nitric acid is also extracted with Aliquat-336. Although the distribution ratio is low, the nitric acid consumes a large amount of extractant

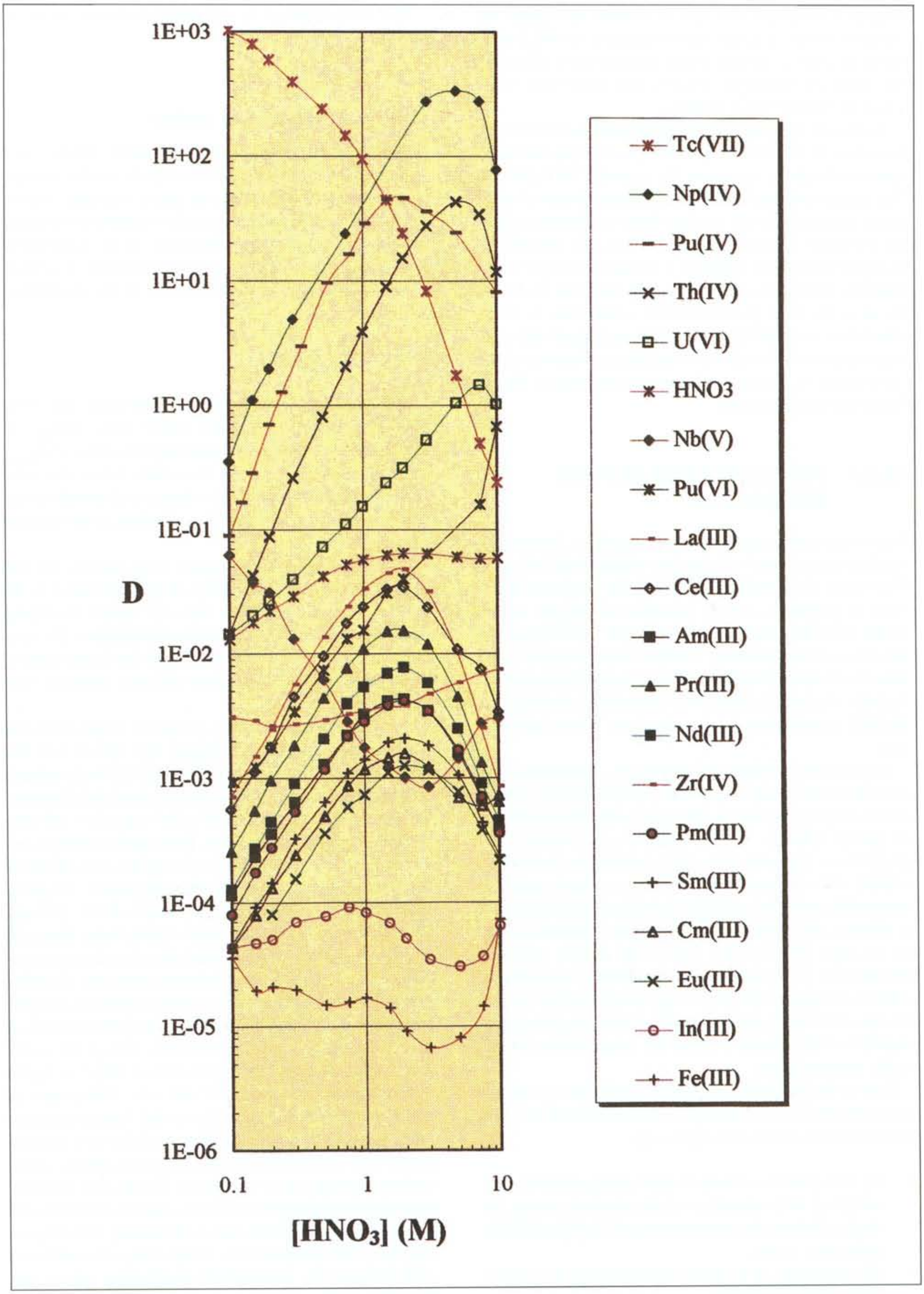


Figure 18-1. *Distribution ratios for some elements as a function of the nitric acid concentration.*

molecules since the nitric acid concentration is high. This is the reason why the distribution ratios for the tri- and tetravalent elements show a maximum when the nitric acid concentration is increased. The extraction of zirconium and niobium is difficult to explain.

Malonamides

Coextraction of trivalent actinides (Am^{3+}, Cm^{3+}) and lanthanides are considered as an interesting separation route in a partitioning process with a following step for separation of trivalent actinides and lanthanides. Malonamides have shown potential as a possible coextractant for actinides and lanthanides in such a process. In France, a separation process called the DIAMEX process, based on a malonamide is under development.

Molecular structure and electro-inductive effects are important factors when extraction by malonamides is considered. The studies at Chalmers are concentrated on structural and electronic influences on metal extraction by three different malonamides. The synthesis of these malonamides is performed in collaboration with the University of Reading. Extraction data for some actinides, lanthanides and nitric acid have been determined /18-12/, see Figure 18-2. When the nitric acid concentration is increased the phenyl substituted malonamide shows a constant increase in metal extraction. The other two malonamides do not behave in this way. Steric effects in the malonamides were evaluated by molecular modelling and it could be concluded that the phenyl groups in DMDPHTD minimize the steric hindrance in the molecule and that this might be one of the reasons for the increase in metal extraction for DMDPHTD /18-13/. The electro-inductive effects by the phenyl groups is probably also involved. By mixing the neutral malonamides with 2-bromodecanoic acid (a carboxylic acid with relatively low pK_a) a synergistic effect in the extraction is observed and a separation between Am and Eu is observed /18-14/. Because this acid violates the CHON-principle we are searching for other organic acids which could replace 2-bromodecanoic acid.

In March 1996, one graduate student finished his diploma work regarding extraction by malonamides. Two different extraction mechanisms were evaluated and a minimization program was used to fit the suggested extraction mechanism to the experimentally achieved extraction data /18-15/.

Oligopyridines

Nitrogen-donor reagents have been shown to have a potential to separate trivalent actinides from lanthanides, probably because of the more covalent character of the nitrogen-actinide bond compared to the nitrogen-lanthanide bond. TPTZ (2,4,6-tri-(2-pyridyl)-1,3,5-triazine) and oligopyridines are suggested as interesting basic structures for this purpose. Because of the hydrophilicity of the reagents, a lipophilic carboxylic acid has to be added. The carboxylic acid should have a low pK_a since it has to be dissociated to some degree in the pH range of interest (<3). 2-bromodecanoic acid has been used for this purpose. The presence of 2-bromo-decanoic acid causes a synergistic effect in the metal extraction and at the same time a separation factor between lanthanides and trivalent actinides of about 10 is achieved /18-14/.

18.3 INTERNATIONAL CO-OPERATION ON PARTITIONING AND TRANSMUTATION

IAEA

W. Gudowski of KTH has been the editor of the IAEA Status Report on Accelerator-Driven Systems /18-1/. KTH is participating in the IAEA Coordinated Research Program on the Hybrid Systems. The project is in progress and will be published in May – June 1997.

EC

A collaboration with CEA Cadarache/FR, ECN Petten/NL, KFA Jlich/DE, ENEA Casaccia/IT, FZK Karlsruhe/DE, JRC-ITU Karlsruhe/DE, AEA Techn Harwell/UK, Univ. Uppsala/SE, ENEA Bologna/IT and the groups at CTH and KTH started on 1 st of May 1996. The project is funded by the EC (IABAT project, FI4I-CT96-0012) in the IV Framework Programme. A project meeting was held on July 1 in Brussels and the first progress meeting, where all participants presented their work, was held in Marcoule, November 13-15. The next progress meeting will be held in Göteborg, Sweden, May 1997. The coordinator of this project is W Gudowski at KTH.

A collaboration between CTH and the Transuranium Institute, Karlsruhe, on calculations on Rubbia's Fast Energy Amplifier included core calculations and reprocessing losses. The work resulted in a paper to the ADTT conference in Kalmar /18-16/.

France

A collaboration has been established by the KTH group with CEA-Cadarache. The collaboration covers several subjects: in the frame of the EC-IABAT project, bilateral collaboration between RIT and CEA (e.g. participation in the subcritical experiments on Masurca experiments – see paragraph Experimental activities) and international collaboration with LANL, CERN and some smaller partners.

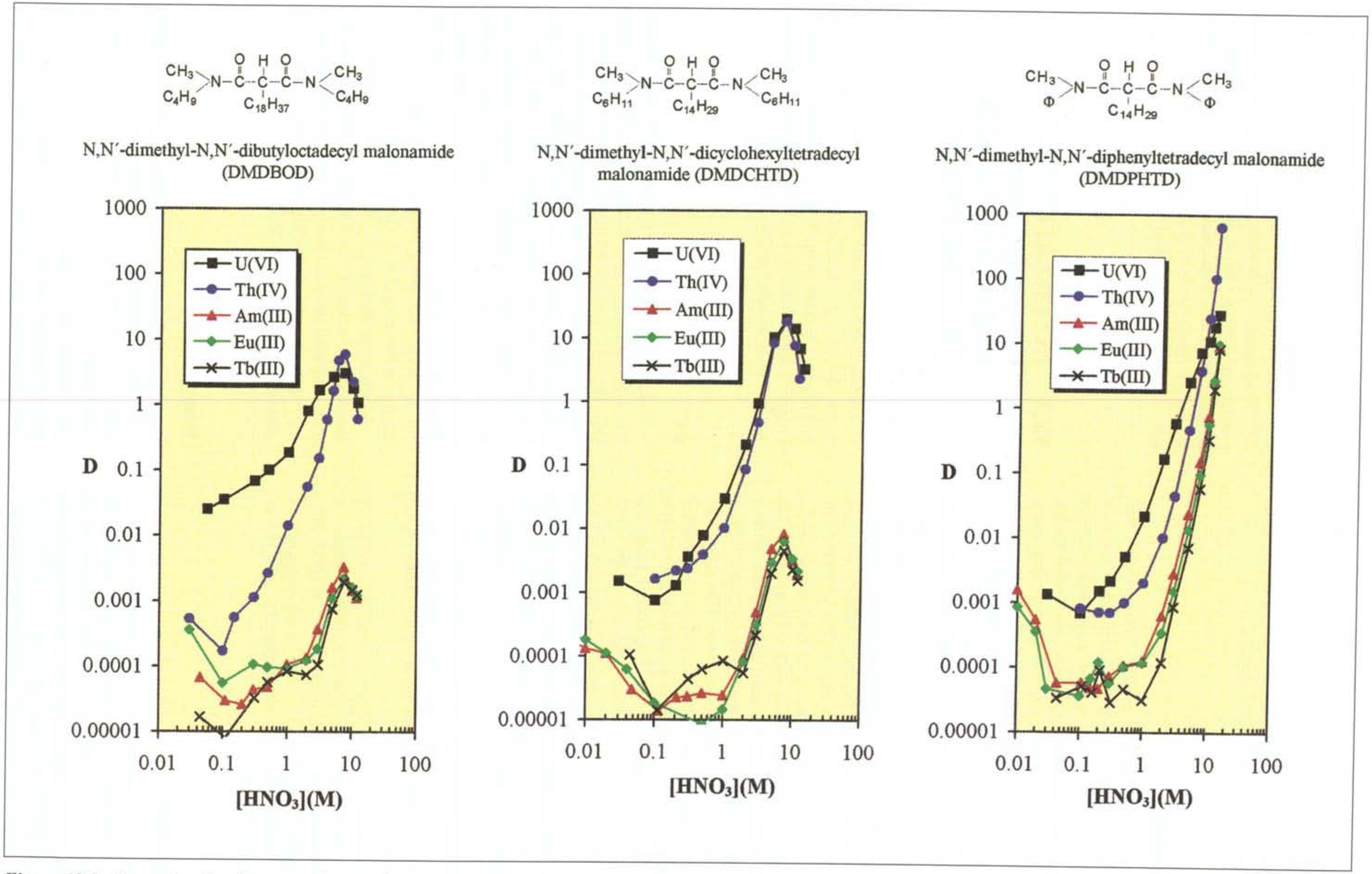


Figure 18-2. Extraction data for some elements by 0.1 M malonamide in **tert**-*butylbenzene.*

The close collaboration between the CTH-group and CEA in France was continued. Present and future work has been discussed with Charles Madic and coworkers at several occasions. Experiences with the AKUFVE system were also exchanged.

Japan

A collaboration which was initiated in 1993 between KTH as well as CTH and the JAERI-group involved in the OMEGA-project has continued under 1996. This collaboration includes information exchange and coordination of some research activities mainly in the spallation process area. CTH has also had exchange of personnel with JAERI.

Russia

A collaboration with Russia was initiated 1994 by the KTH group in the frame of the ISTC Project 17 "Feasibility Study of Principal Technologies in Accelerator Based Conversion of Military Pu and Long-Lived Radioactive Waste". This has developed into a number of projects: manufacturing of the spallation target (Obninsk – ISTC), medium energy cross-section measurements (St. Petersburg – ISTC), development of the code system for high energy charged particle transport and neutronic calculations (Dubna – bilateral collaboration).

USA

Collaboration between LANL and KTH which was initiated in 1992 has further developed under 1996. KTH has regular exchange of researchers, exchange of the research results and is coordinating some the research activities with LANL. Manufacturing of spallation target in Obninsk, Russia, is one of the results of this collaboration. The collaboration at CTH involves exchange of information and results within aqueous based partitioning processes.

18.4 GEOSCIENTIFIC APPRAISAL OF CONDITIONS IN THE DEPTH INTERVAL, 1000 – 5000 M

The Very Deep Hole (VDH) concept for storage of high level nuclear waste involves the drilling of large diameter boreholes to c 4 km depth, placing the waste in canisters, and then deploying the canisters between c 2 km to 4 km depth. For this concept, it is imperative that the geological conditions down to 5 km are fully understood. The initial step in making this appraisal was carried out in 1989 /18-17/ where a model suggesting that the upper c 1 – 1.5 km of bedrock in the Baltic Shield is considerably more fractured than that below. This suggestion was based on a review of results from the Gravberg-1 drilling project and other deep drilling projects world-wide. The concept that the upper 1 km is more fractured than the rock below was earlier put forth by Båth, /18-18/. Båth observed low velocities in the upper 1-2 km in seismic data (P-wave refraction and surface seismic wave data) and attributed these low velocities to fracturing. In the present study, the concept of increased fracturing in the upper c 1 km of bedrock is further investigated by reviewing more recent data from other deep boreholes. The earlier work /18-17/ concentrated mainly on geophysics and rock mechanics. The present study also includes these fields, but more effort has been put into understanding the geological, hydrogeological and hydro-geochemical conditions at depth /18-19/. The review work was initially divided into the five fields:

- Geology.
- Hydrogeology.
- Hydrochemistry.
- Geophysics.
- Geomechanics.

These initial reports, which include compilation, review and analyses of data relevant to each field, form the basis for an integrated study of the geological conditions down to 5 km /18-20 – -25/. In this study an attempt is made to present a consistent model which takes data from the various fields into account. The data, which are mainly extracted from deep boreholes, cover the following parameters:

- Lithology.
- Fracture mineralogy.
- Fracturing (porosity).
- Permeability.
- Pore pressure.
- Mechanical properties.
- State of stress.
- Fluid composition.
- Temperature.
- Natural seismicity.
- Bacteria.

A review of how continents evolve and the geology of Europe is included in the study since the geological setting of the boreholes studied varies greatly.

Results so far are:

– Prediction of lithology at depth based on surface geological information is difficult in volcano-sedimentary rocks due to the heterogeneous nature of these rocks. In granitic environments it is easier to predict the lithology in the upper 5 km.

- Fracture mineralogy is indicative of how an observed fracture evolved. If the minerals are in equilibrium with the surrounding rock then the fracture probably developed under ductile conditions, otherwise it developed under brittle conditions. Brittle fractures are probably the ones which are most likely to be highly permeable.

- Surface and borehole geophysical data show that the degree of open fracturing decreases significantly below c 1 km.

- Although data are sparse, permeability also appears to decrease significantly below 1 km. In the available data, the permeability is about 3 orders of magnitude lower at 5000 m than at 1000 m.

- In general, the pore pressure is close to hydrostatic in all boreholes. The one exception is the Kola borehole in the Baltic Shield of Russia where pore pressures of 40 – 50% greater than hydrostatic have been reported in the depth interval 1000 – 2800 m.

- Mechanical properties of the rock are defined in terms of strength and deformability and studies carried out near the surface are relevant for quantifying these properties at depth. As near the surface, the parameters are dependent on the volume of the rockmass under investigation.

- The stress appears to increase linearly with depth down to 5 km, although quantitative measurements below 1 km are lacking. The vertical stress is generally the intermediate stress implying the tectonic regime is strike-slip faulting.

- Pore waters are relatively fresh down to at least 500 m throughout Sweden, but become more saline below this depth. At great depth, brines are present. The depth to these brines appears to be dependent upon geographical location. High gas content may be observed in brines in Sweden and Canada. The high gas content shows that the brines have been stable for periods of millions to possibly hundreds of millions of years.

- Measurement and extrapolation of borehole data give a temperature gradient in the Baltic Shield in the range of 15 – 20°C/km. These temperature gradients are also predicted in the uppermost crust from lithospheric modeling of heat flow in the Baltic Shield.

- Sweden has only, comparatively, minor earthquake activity. However, earthquakes on the order of magnitude 5, or even 6, in the future cannot be ruled out. Some of the observed earthquakes occur at shallow levels, in the upper 5 km of bedrock. There is currently a debate on the source of earthquakes, whether they are due to stresses from post-glacial rebound or to plate tectonics.

- Investigations show that bacteria exists and flourishes at great depth, independent of the biosphere, and are able to produce large quantities of methane. This implies that previous interpretations of deep groundwater evolution may have to be somewhat modified.

- The above results are being integrated into a conceptual model for groundwater flow and evolution in the upper 5 km of the Baltic Shield.

19 INTERNATIONAL COOPERATION

An important part of SKB's 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.

19.1 SKB's BILATERAL AGREEMENTS WITH FOREIGN ORGANIZATIONS

SKB has signed formal bilateral agreements with the following organizations in other countries:

- US DOE (Department of Energy), USA,
- AECL (Atomic Energy of Canada Ltd) and ONTARIO HYDRO, Canada,
- Nagra (Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle), Switzerland,
- Posiva OY, Finland,
- JNFL (Japan Nuclear Fuel Ltd.), Japan,
- CEA (Commisariat a l'Energie Atomique), France,
- ENRESA (Empresa Nacional de Residuos Radiactivos), Spain.

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.

19.1.1 COOPERATION WITH DOE, USA

The cooperation between USDOE and SKB concerning the Äspö HRL was terminated during 1996 due to budget reductions.

19.1.2 COOPERATION WITH AECL AND ONTARIO HYDRO, CANADA

During 1996 a SKB/AECL/Ontario Hydro bilateral information exchange meeting was held in Canada.

The cooperation has mainly been concentrated to the following issues:

- Buffer and Backfill. SKB has together with ANDRA in France supported the AECL microbial investigation of the bentonite based buffer from the buffer mass heater test in URL, see section 15.2.
- Comparison of costs for the waste management system.
- VALUCLAY. An agreement was signed in 1996 between SKB, AECL and PNC on the cooperation in the VALUCLAY project (Validation of codes/models that describe the Unsaturated behaviour of engineered Clay barriers).

19.1.3 COOPERATION WITH NAGRA, SWITZERLAND

During 1996 a SKB/Nagra bilateral information exchange meeting was held in Sweden.

The cooperation has mainly been concentrated on:

- safety analysis and performance assessment,
- column experiments on high pH weathering, (see section 15.4),

- natural analogue studies, see section 15.5,
- underground construction material performance,
- other long-lived waste than spent nuclear fuel, see Chapter 16,
- fracture characterization at Äspö (see Äspö Hard Rock Laboratory Annual Report 1996).

19.1.4 COOPERATION WITH CEA, ANDRA, DCC AND IPSN, FRANCE

A bilateral information exchange agreement was signed between SKB and CEA during 1996 as well as a continuation of the Äspö HRL Project Agreement between SKB and ANDRA.

The cooperation with organizations in France has mainly concerned the following issues:

- Natural analogues. SKB is engaged in the EU sponsored natural analogue project in Oklo managed by CEA, see section 15.5.2.
- Instruments. IPSN/CEA in Cadarache, France, has been involved in the development of a borehole probe (CHEMLAB), which was delivered during 1996.
- Buffer and Backfill. As mentioned in section 15.2 above, ANDRA has participated in the work on bacteria in bentonite buffer from the Canadian URL facility, see section 15.2.
- Other long-lived waste than spent nuclear fuel. Informal exchange of experience has been established, see Chapter 16.

19.1.5 COOPERATION WITH POSIVA OY, FINLAND

SKB has a very close cooperation with POSIVA OY in many fields of the research on nuclear waste management. The following areas have during 1996 been the most active cooperation items:

- Safety analysis.
- Radionuclide retention research.
- Geochemistry.
- Exchange of experience and technology for site investigation.
- Characterization of full-scale deposition holes.
- Documentation work on relevant information on buffer and backfill materials. Standardized and recommended laboratory and field test methods etc, see section 15.2.
- Exchange of information on environmental impact assessment work in the siting processes in Sweden and Finland.
- Exchange of information on copper canister manufacturing and welding methods.
- Alternative disposal methods.
- SKB and POSIVA scientists have during 1996 had numerous meetings where information and experience have been exchanged.

19.1.6 COOPERATION WITH JNFL, JAPAN

During 1996 the cooperation has been carried out through study visits at SKB facilities and through informal information exchange meetings.

19.1.7 COOPERATION WITH NIREX, UK

Though there is no formal agreement on general information exchange with NIREX, general cooperation work has been performed outside Äspö HRL work in which NIREX is participating. The areas where this cooperation has been made are:

- Column experiments on high pH weathering, see section 15.4
- Natural analogue studies, see section 15.5.
- Other long-lived waste than spent nuclear fuel, see Chapter 16.

19.1.8 COOPERATION WITH ENRESA

SKB cooperates with ENRESA within several of the EU-projects, (see section 19.10; Oklo, Palmottu and Catsius Clay). Cooperation was initiated between SKB and ENRESA in the framework of the bilateral information exchange agreement signed during 1996 and the Äspö HRL agreement signed in the beginning of 1997.

19.2 PARTICIPATION WITHIN EU

SKB has participated in the following committees and groups during 1996 within the EU:

STC, Scientific and Technical Committee, 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.

Club of Agencies
SKB is represented by Sten Bjurström in the Club of Agencies, which is a cooperation between the implementing organizations in the member countries.

Ad Hoc group on Safety aspects of retrievability of radioactive waste and *Ad-hoc working group on the fourth report on the situation and prospects of radioactive waste management in the European Union* where SKB is represented by Tönis Papp.

Contract Implementation and Evaluation Group (CIEG) where Bo Sundman is the SKB representative.

Nuclear Safety and Safeguards which is the coordinating committee on the programme Nuclear Safety and Safeguards. Tönis Papp from SKB has participated as an expert in the group.

Natural Analogue Working Group (NAWG) is an international group working with natural analogues and their use in the safety assessment modelling. The group is organized by EU. SKB has been represented in this group since its start in 1985. One of SKB consultants, Dr John Smellie, has been the chairman of the group until 1996.

SKB participates in a number of projects within the Nuclear Safety and Safeguards programme:

The Oklo Natural Analogue Project
SKB participates in the second phase of the Oklo Natural Analogue Project, managed by CEA (see section 15.5.2).

The Palmottu Natural Analogue Project
SKB participates as a full member in the new phase of the Palmottu Natural Analogue Project, which started in 1995 (see section 15.5.3).

DECOVALEX II
SKB participates in the second phase of the project DECOVALEX (DEvelopment of COupled models and their VALidation against Experiments), (se section 15.3.5). The second phase started in November 1995.

Planning of the participation in the projects *PAGEPA* (Palaeohydrology and geoforecasting for performance assessment in geosphere repositories for radioactive waste disposal) and *EQUIP* (Evidence form Quaternary Infills for Palaeohydrogeology) was started during 1996.

Clay Microstructure. SKB is sponsoring Clay Technology in an EU project on the microstructural changes on clay, (see section 15.2.1).

Catsius Clay (Calculation and Testing of behaviour of Unsaturated Clay) is another project where SKB participates through Clay Technology, (see section 15.2.2)

Spent Fuel. Cooperation work on spent fuel was initiated during 1996 between SKB and ITU, Karlsruhe.

19.3 COOPERATION WITHIN OECD NUCLEAR ENERGY AGENCY

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 1996 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. Tönis Papp is the member from SKB.

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. Lars Olof Ericsson is the member from SKB.

IPAG (Integrated Performance Assessment of Deep Repositories) is a forum for the exchange of experience and information on recently completed performance assessment studies. Allan Hedin is the member from SKB.

The *GEOTRAP* project was launched during 1996 by the workshop "Field Tracer Transport Experiments: Design, Modelling, Interpretation and Role" in the prediction of Radionuclide Migration. The workshop was the first in a series of five and was hosted by GRS in Köln, Germany.

An implementation committee meeting was held in Paris 3 – 4 April 1996 for the forthcoming GEOTRAP project.

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. Hans Forsström is the member from SKB.

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. 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. Fred Karlsson is the member from SKB.

Liaison Committée for Co-operative Programme on Decommissioning; Technical Advisory Group.

19.4 COOPERATION WITHIN IAEA

Cooperation has during 1996 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, *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 is participating in the Standing Technical Committee for Disposal within the RADWASS programme.

Furthermore SKB participates in the *Waste Technology Advisory Committee (WATAC)* and the *Technical Committee on revision of the IAEA Regulations for the Safe Transport of Radioactive Material.*

19.4.1 VAMP

SKB has, through Studsvik EcoSafe, participated 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). The program ended during 1996.

19.4.2 BIOMASS

The Division of Radiation and Waste Safety has initiated a new research program on transport of radionuclides and radiological effects on the biosphere called *BIOMASS.* SKB will participate in the program.

19.5 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.

19.6 OTHER INTERNATIONAL COOPERATION

19.6.1 BIOMOVS

As indicated in section 15.6 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 1996 taken active part in the scenario definition work where the RES methodology work, see section 14.3.2, has been used. The BIOMOVS study was reported and completed during 1996.

19.7 INTERNATIONAL COOPERATION IN THE ÄSPÖ HARD ROCK LABORATORY

As is mentioned in Chapter 17 the Äspö HRL has gained great international interest. The following organizations have up to the end of 1996 signed agreements to cooperate in joint work at the Äspö HRL:

- AECL, Canada,
- ANDRA, France,
- BMBF, Germany.
- CRIEPI, Japan,ANDRA, France,
- Nagra, Switzerland,
- PNC, Japan,
- POSIVA OY, Finland,
- UK NIREX, UK,
- USDOE, USA.

USDOE terminated their participation during 1996.

Discussions on the participation of ENRESA, Spain, in the Äspö HRL started during 1996. An agreement was signed early 1997.

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 1996 /19-1/.

20 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 96,
- in the series of SKB Technical Reports, in contributions to scientific journals, symposia and conferences in different subject areas, see Appendix 2,
- in SKB Rapport,
- in SKB HRL International Cooperation Reports,
- in SKB HRL Progress Reports,
- in SKB Djupförvar Projektrapporter,
- in SKB Inkapsling Projektrapporter,
- in SKB CLAB Etapp 2 Projektrapporter.

Documentation for internal use (in Swedish) is prepared in internal reports (Arbetsrapporter, Progress and Project Reports, 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.

20.1 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 1996 Technical Reports are included in part IV of this Annual Report.

20.2 CONTRIBUTIONS TO PUBLICATIONS, SEMINARS ETC

The contributions to conferences, symposia and scientific journals have been extensive during 1996, see Appendix 2.

Both SKB own staff as well as the contractors of SKB have been involved in this work.

20.3 SKB BIBLIOGRAPHICAL DATABASES

20.3.1 BIBAS

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 1996 about 12 530 references. The software used to manage the database is AskSam which has a powerful free-text search capability.

20.3.2 INIS

The INIS database (1976-) is also availiable in the SKB network. All references in this database are installed on an ERL server with all data on hard disk. New references are added twice per year. Users search the database using a client program on their PCs (client-server at it's best). Performance is excellent, typical search times are less than 5 seconds, very complex ones below 10 seconds. This fast response gives the user a new interactive way to work with retrieving documents, targeting in on the desired area by many repeated scans and direct review of abstracts.

20.4 THE GEOLOGICAL DATABASE SYSTEM GEOTAB

GEOTAB is the old name of the geological database, holding all data from borehole investigations. In 1994 the data was transferred to the SICADA system (see Chapter 17 Äspö Hard Rock Laboratory, section 17.3.6 Data management). For security reasons the GEOTAB system is still running, but no new data are added. The system is planned to be shut down in 1997.

20.5 COMPUTER GROUP

The computer resources at SKB have been consolidated into a new "Computer group". This group consist of four people, of which two was transferred from the R&D department and two were recruited during 1996. The task of the group is to supply a base IT platform, spanning from almost supercomputing (PA analysis) to normal office business.

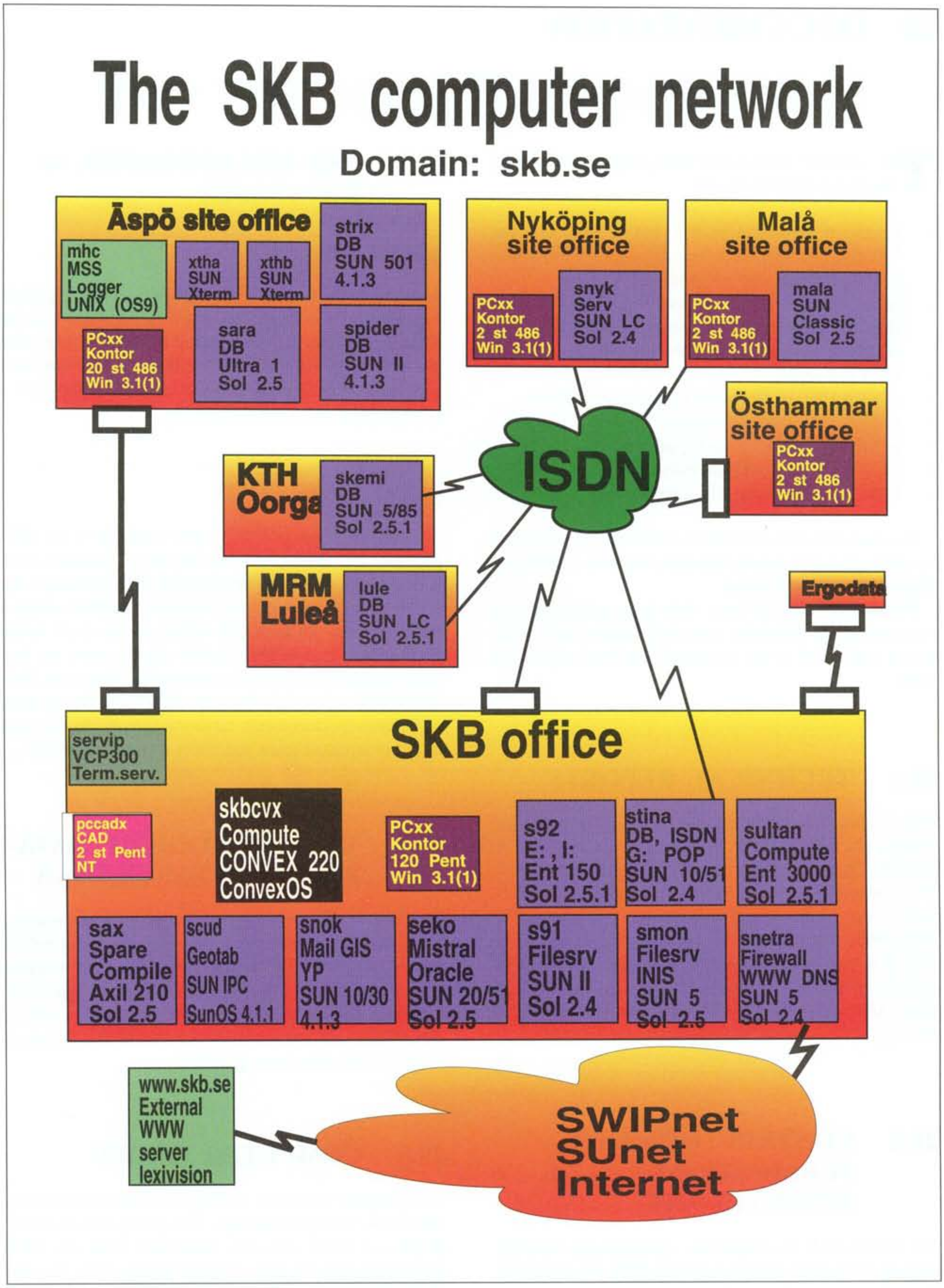


Figure 20-1. The SKB computer network.

20.5.1 Computing resources

Computing resources have previously been focused at the CONVEX C2 machine which has been doing a great job since 1989 with it's two vector processors. During 1996 there was a need for more capacity to cope with the needs for the PA SR-97. Surprisingly it was not easy to find a reasonably priced new machine that had significantly higher capacity compared to the CONVEX C2. The final solution was a combination of a reasonably fast server and a load sharing system (LSF) making it possible to use servers and workstations as one virtual machine.

The new server is a SUN Enterprise 3000 machine with four processors and 1 Gb memory. Each processor is about twice as powerful as the Convex processor.

The LSF system is currently configured with seven machines, CONVEX C2, SUN Enterprise 3000, SUN Enterprise 150, SUN SPARC 20, SUN SPARC 10 and two SUN SPARC 5. This adds up to a total capacity of eight times the CONVEX C2, which means almost 400 Mflops (millions floating point operations per second).

20.5.2 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 is running the legal accounting system.

The different data media coped with are Exabyte tapes (2, 5 and 7 Gb), QIC tapes (0.15 and 0.3 Gb), CDs, Syquest 88Mb, ZIPdrive (100Mb), 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.

20.5.3 PC network

The about 160 PCs at SKB are connected to the network using the TCP/IP stack PC-NFS from SUN Microsystems. Although this technology has put high demands on the PCs during previous years, the fruits of this strategy is now harvested as easy integration of Intranet, Mail and WWW techniques. The prevailing operating system is MSDOS with Win 3.11, but the Win NT system is under review and plans are made for an introduction in late 1997, when security problems have been solved.

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 Pentium with at least 16 Mb memory running DOS 6 Windows 3.11, MS-Office programs and others.

20.5.4 Wide area network (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,
- three offices in Nyköping, Östhammar and Malå and
- four consultants in Uppsala, Stockholm, Göteborg and Luleå.

The LANs in Stockholm and Äspö are connected via ethernet bridges, operating over a leased 64kbps line.

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 256 kbps leased line Figure 20-1 shows an overview of the WAN with nearly all connections.

20.5.5 Local area network (LAN)

Only one standard protocol is used in the network TCP/IP. TCP/IP is used by all connected computers (nodes) and also used for PC networking, terminal sessions, NIS, DNS, mail and file transfer. This facilitates implementation of new communication applications.

The cabling is a 9 year old "thin wire ethernet" operating at 10 Mbps, tied together with switches and hubs into a 100 Mbps TP backbone. In Figure 20-2 the cabling is drawed with a thickness of the connections proportional to capacity$^{(1/3)}$, thus mimicking a tube with that capacity.

20.5.6 Mail

The electronic mail system at SKB is supplies each individual with full mail-access to the whole internet world and also via a gateway, to most X-400 operators. The clients are fully MIME compliant and users are not bound to any specific PC or workstation, but can read their mailboxes at any PC. The performance of the mail client is not acceptable though, sometimes it takes several minutes to send a 2 Mb document.

All employees at SKB has the address firstname.lastname@skb.se and all users has an address on the form

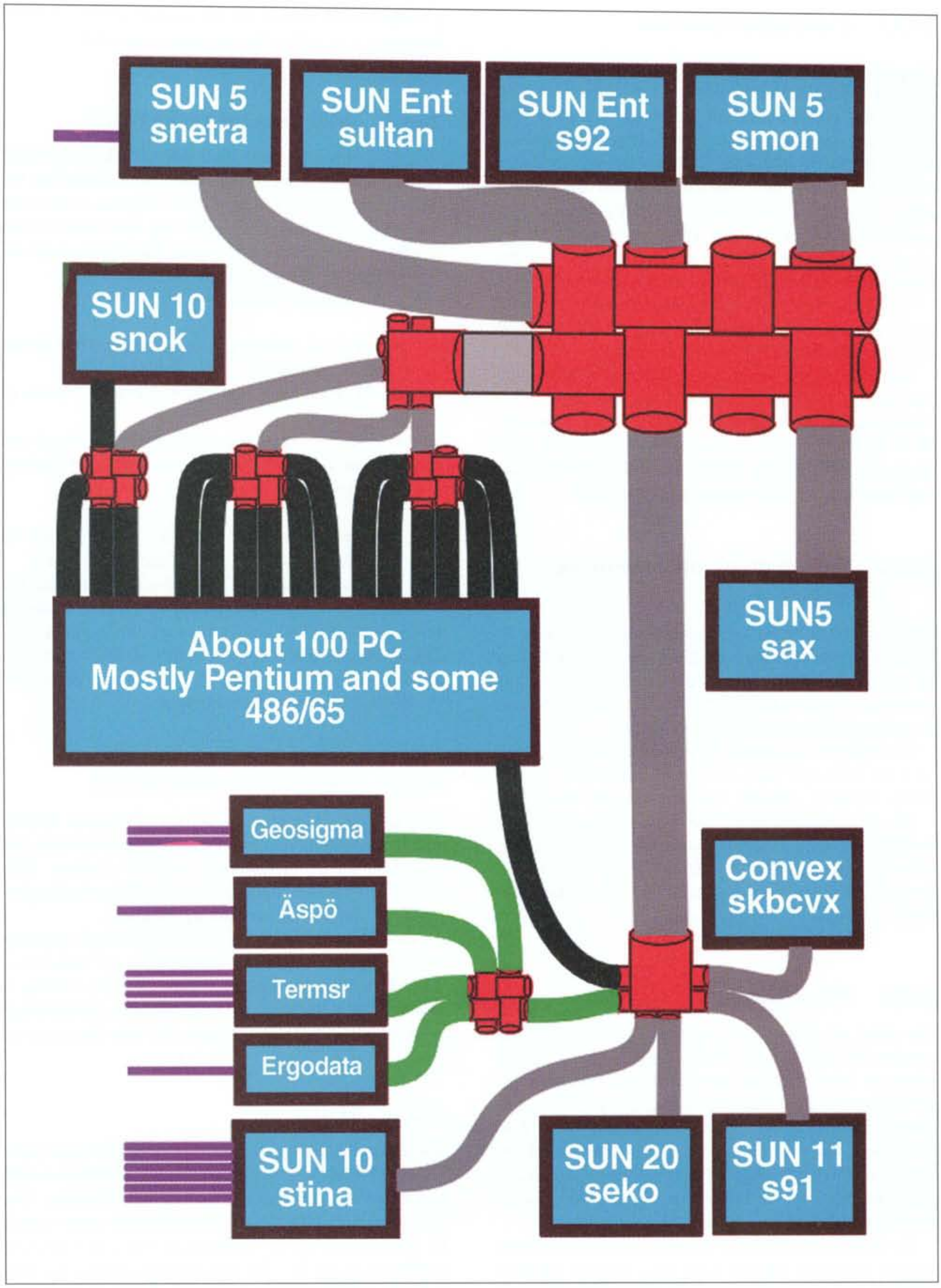


Figure 20-2. Cabling system at SKB office.

orgid@skb.se where "id" is the person's initials, the preceding "org" identifies the organization and "skb.se" identifies the SKB network domain.

20.5.7 Internet

The connection to the Internet (256 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 more unlimited, although all traffic passes a monitored firewall. All workstations and PCs have direct connection (via the firewall) to the internet and the use of common tools as WWW is increasing.

20.5.8 World Wide Web

At the website hhtp://ww.skb.se/ SKB has some 50 pages of information mimicing the printed information material. Some extra functions are implemented for taking contact with the information department and ordering material. The contents of this site is controlled by the information department and the layout and technical operation is managed by a consultant.

20.5.9 Intranet

The Intranet at SKB went into operation in may 1996 and has been a moderate success. It contains major parts of the strategical documentation, instructions, phone books, press releases, the internal news and some lighter stuff. The aims are to make everyone that is responsible for information, capable to publish it at the Intranet for immediate distribution, thus reducing the flow of sparsely read paper. (Today each employee produce about 3 000 pages per month.) Currently about ten people practice publishing at the intranet, but a rapid increase can be expected as the editing tools become more user friendly.

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