SKB

TECHNICAL REPORT

92-46

SKB ANNUAL REPORT 1992

Including Summaries of Technical Reports Issued during 1992

Sockholm, May 1993

SVENSK KÄRNBRÄNSLEHANTERING AB
SWEDISH NUCLEAR FUEL AND WASTE MANAGEMENT CO

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SKB ANNUAL REPORT 1992

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FOREWORD

The Annual Report on SKB's activities during 1992 covers planning, constructing and operational activities 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. With the central repository for final disposal of low and medium level waste, SFR, and the central interim storage facility for spent fuel, CLAB, in operation, SKB can take care of all radioactive waste produced inside Sweden for a long time ahead.

For the remaining facility – the final repository for spent nuclear fuel – comprehensive research and planning activities is well under way, aiming at a principal decision on disposal methods and site around the mid 90s.

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 hope this Annual Report will be of interest and that it will enhance the international information exchange.

Stockholm in May 1993

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 1992.

Lectures and publications during 1992 as well as reports issued in the SKB technical report series are listed in part IV.

Part V contains the summaries of all technical reports issued during 1992.

SKB is the owner of CLAB, the Central Facility for Interim Storage of Spent Nuclear Fuel, located at Oskarshamn. CLAB was taken into operation in July 1985 and to the end of 1992 in total 1 684 tonnes of spent fuel (measured as uranium) have been received. Transportation from the nuclear sites to CLAB is made by a special ship, M/S Sigyn.

At Forsmark the final repository for Radioactive Waste – SFR – was taken in operation in April 1988. The repository is situated in crystalline rock under the Baltic Sea. The first construction phase includes rock caverns for 60 000 m^3 of waste. A second phase for additional 30 000 m^3 is planned to be built and commissioned around the year 2000. At the end of 1992 a total of 11 000 m^3 of waste have been deposited in SFR.

SKB is in charge of a comprehensive research and development programme on geological disposal of nuclear waste. The total cost for R&D during 1992 was 192.3 MSEK of which 24.8 MSEK came from participants outside Sweden.

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.
- Construction 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 May 1993 there are 7 foreign organizations participating in the Äspö HRL project.

Cost calculations for the total nuclear waste management system, including decommissioning of all reactors, are updated annually. The total cost is estimated to 55 billion SEK.

SKB also handles matters pertaining to prospecting and enrichment as well as stockpiling of uranium as strategic reserves for the Swedish nuclear power industry.

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 1992 about 80 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 1992 successful public information activities have been carried out using mobile exhibitions in a tailor-made trailer and on the SKB ship M/S Sigyn.

In September 1992, SKB submitted its third RD&D-Programme to the authorities according to the Swedish Act on Nuclear Activities.

CONTENTS

SKB ANNUAL REPORT 1992

Part I

Overview of SKB Activities

CONTENTS PART I

Page

1 GENERAL BACKGROUND

1.1 THE SWEDISH NUCLEAR POWER PROGRAMME

The nuclear power programme of Sweden consists of 12 nuclear reactors located at four different sites and with a combined capacity of 10 000 MW net electric power. Main data and location of the 12 units are shown in Figure 1-1. The nuclear power plants generated about 44% of the total Swedish electric power produced in 1992.

Swedish reactors

Reactor		Power MW$_e$	Commercial operation	Energy availability in 1992 %
Oskarshamn 1	BWR	440	1972	50
Oskarshamn 2	BWR	605	1974	60
Oskarshamn 3	BWR	1160	1985	90
Barsebäck 1	BWR	600	1975	68
Barsebäck 2	BWR	600	1977	56
Ringhals 1	BWR	795	1976	55
Ringhals 2	PWR	875	1975	78
Ringhals 3	PWR	915	1981	84
Ringhals 4	PWR	915	1983	90
Forsmark 1	BWR	970	1980	88
Forsmark 2	BWR	970	1981	90
Forsmark 3	BWR	1155	1985	90

1.2 LEGAL AND ORGANIZATIONAL FRAMEWORK

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

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

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

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

The total central staff of SKB is about 65 persons. The organization is presented in Appendix 1. For the bulk of the work a large number of organizations and individuals outside SKB are contracted. As a whole about 600 persons are involved in SKB waste handling and research work.

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

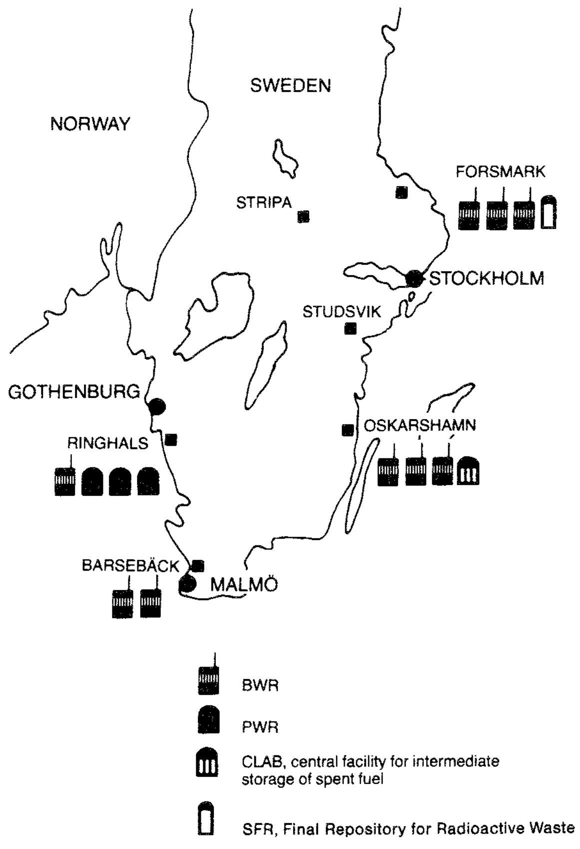


Figure 1-1. The Swedish nuclear power programme.

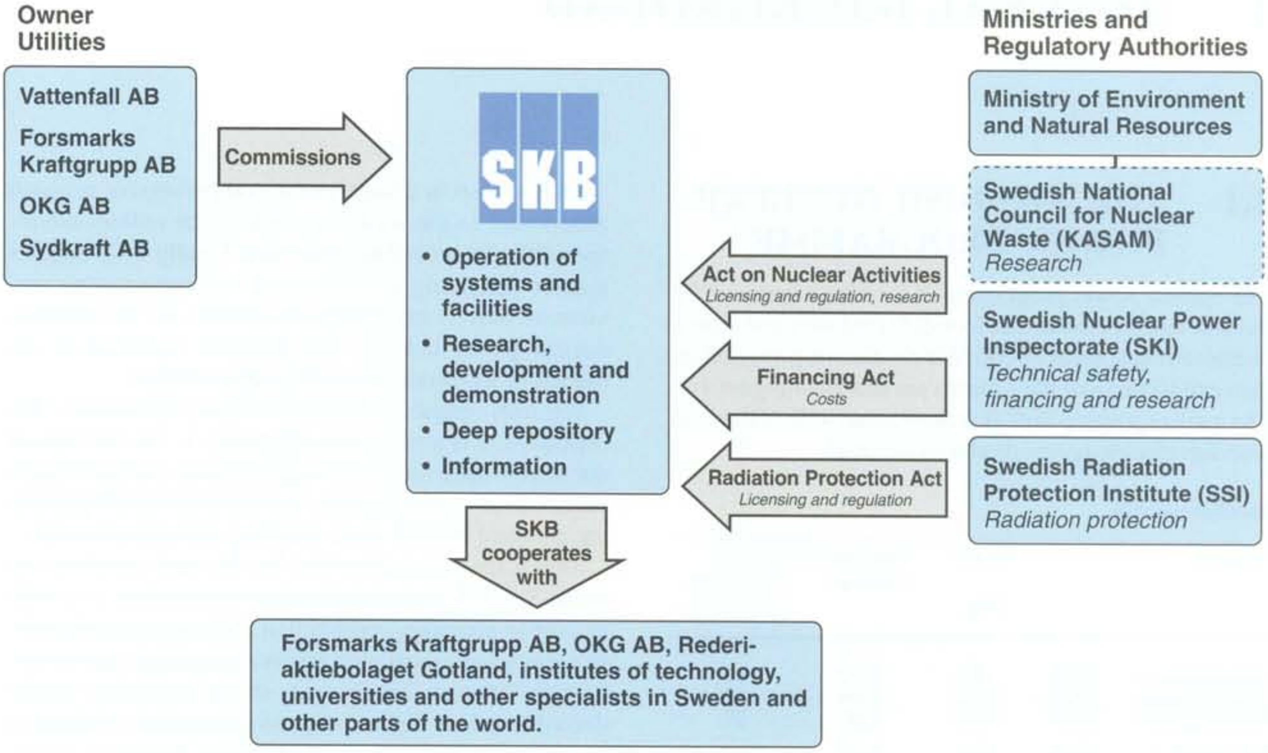


Figure 1-2. Legal framework for activities of SKB.

There are three important laws which regulate the nuclear activities.

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

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

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 National Institute for Radiation Protection (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 ever since 1984 been 0.019 SEK per kWh.

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

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

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

The four competent authorities have separate funds for the research needed to fulfil their obligations. SKI also support additional waste management research beside the SKB programme.

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 heat flux and radiation at first. Contains long-lived nuclides	4 400 canisters (7 900 tU)
2 Transuranic-bearing waste	Waste from the Studsvik research facility	Solidified in concrete	Low- to medium-level. Contains long-lived nuclides	1 500 m^3
3 Core components and internals	Scrap metal from inside reactor vessels	Untreated or cast in concrete	Low- to medium-level. Contains certain long-lived nuclides.	19 700 m^3
4 Reactor waste	Operating waste from nuclear power plants etc.	Solidified in concrete or bitumen. Compacted waste	Low- to medium-level. Shortlived	91 500 m^3
5 Decommissioning waste	From dismantling of nuclear facilities	Untreated for the most part	Low- to medium-level. Shortlived	111 500 m^3

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 be generated when the reactors and other facilities are dismantled.

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

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

The first construction stage of the Swedish Final Repository for Radioactive Waste, SFR, was taken into operation in 1988. SFR may later on be extended to accommodate waste also from the decommissioning of the nuclear reactors. For spent fuel a central interim storage facility, CLAB, was taken into operation in July 1985. This facility has with current configuration a capacity of 5 000 tonnes of spent fuel.

The spent fuel will be stored in CLAB for about 40 years. It will then be encapsulated in a corrosion-resistant canister and deposited at depth in the Swedish bedrock. According to the new time schedule presented in the RD&D-Programme 92 SKB plans to expand the CLAB facility at the end of the '90s with an encapsulation plant in order to make encapsulated fuel available for disposal around 2008.

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

More details about the planning in the RD&D-Programme 92 are given in chapter 10.

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

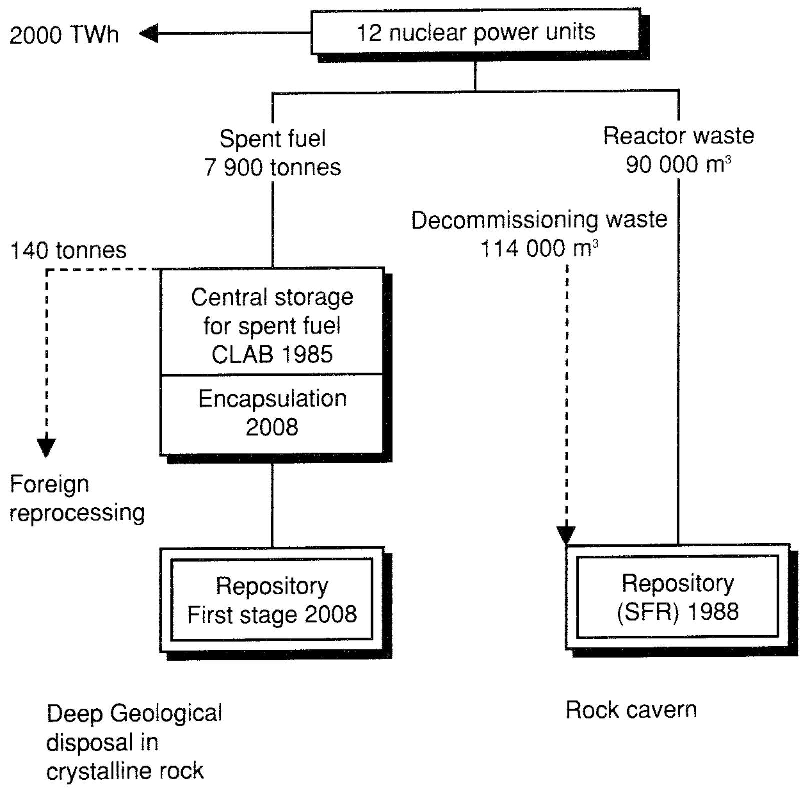


Figure 1-3. *Main system for management of radioactive waste in Sweden.*

2 NUCLEAR FUEL SUPPLY

In the front end of the nuclear cycle SKB handles matters pertaining to prospecting and enrichment as well as stockpiling of uranium as strategic reserves for the Swedish nuclear power industry. SKB also provides assistance at the request of its owner utilities in uranium procurement.

2.1 NATURAL URANIUM

The Swedish nuclear power programme has an annual demand of about 1 600 metric tonnes of natural uranium. This demand could be higher or lower depending on a number of factors, which means that the planning of supply must be flexible.

The demand for the period 1992 up to 2 001 is 16 300 tonnes. At the end of 1992, the Swedish utilities had contracts for supply of 8 300 tonnes during the same period. Most of the supply is based on long-term contracts. As the prices on the spot market were low in 1992, some spot quantities were purchased.

Natural uranium is delivered to Sweden mainly from Canada and Australia, but also from USA.

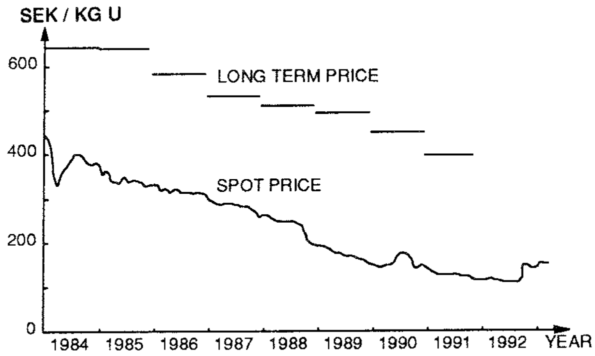


Figure 2-1. *Long term and spot prices for uranium.*
Long term price = Average price for long term deliveries to the European Community.
Spot price = Average spot price each month published by the German company NUKEM for non US origin uranium.

Exploration

Uranium occurs in relatively high concentrations in certain parts of the Swedish precambrian rock. SKB has therefore earlier been conducting exploration at a number of places in northern Sweden. Mineralizations containing at least 6 000 metric tonnes of uranium have been found with concentrations higher than 1 000 g uranium per ton ore. These ores constitute important reserves for the future.

As uranium supply is abundant and the market price is low, SKB stopped exploration at the end of 1985.

Ranstad

Sweden has considerable uranium resources. Most of the proven reserves consist of relatively low-grade shale deposits near Ranstad with about 300 g uranium per ton of shale. These deposits are not exploitable at the present low price of imported uranium.

Market-prices

Figure 2-1 shows the price situation for uranium during the last years. Spot prices were low in 1992.

The average price for long term deliveries in 1984-1991 to the European Community was considerably higher than the spot prices for the same delivery years.

2.2 CONVERSION

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

The world conversion capacity is around 60 000 tonnes of uranium per year while the demand is about 50 000 tonnes per year.

The Swedish utilities utilize conversion services from Canada, USA, United Kingdom and France.

2.3 ENRICHMENT

Up to 1983, enrichment deliveries to the Swedish utilities were dominated by DOE in the USA and Techsnabexport in the USSR.

The European enrichment industry became price competitive in the beginning of the 1980-ies. During the period 1983-1985 Swedish utilities signed contracts for deliveries from Western Europe, which started already 1984.

For the period 1992-2000, most of the deliveries to Sweden will come from EURODIF with an enrichment plant in France, from URENCO with enrichment plants in the Netherlands, the United Kingdom and in Germany and from Russia with an enrichment plant near Jekaterinburg. Deliveries from the US (DOE) will continue on a reduced scale.

This situation gives a reliable supply with deliveries from four different suppliers of enrichment.

Russia also delivers enriched uranium which means that the corresponding quantity of natural uranium and conversion services is of Russian origin.

2.4 FABRICATION OF FUEL ASSEMBLIES

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

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

Fuel fabrication at ABB Atom was around 280 tonnes of UO_2 for nuclear fuel for BWR and PWR during 1992. Of this volume about 140 tonnes were exported to Finland, Germany, Switzerland, Belgium and France.

The fuel assembly design, SVEA, where the fuel rods are devided in four minibundles with 4 x 4 or 5 x 5 rods separated by a water cross, is now the dominating BWR fuel in Sweden. All of the ABB Atom BWR deliveries in 1992 were of this design.

The SVEA fuel utilizes the energy from the inner fuel rods in a better way, which means that 8-10 % more energy can be produced from a given amount of enriched uranium compared with the earlier type of fuel.

2.5 NUCLEAR FUEL STOCKPILE

The Swedish Nuclear Fuel and Waste Management Co is on behalf of the utilities responsible for stockpiling enriched uranium and zircaloy corresponding to an electricity production of 35 TWh. This amount has been decided by the Swedish parliament.

Uranium in the above mentioned stockpile, in fuel under fabrication and at the nuclear power stations is sufficient for about two years of operation of all 12 units.

2.6 COSTS

The costs for the front end supply and services of the nuclear fuel cycle in 1992 in Sweden were as shown in Table 2-1 (the production of nuclear electricity was 60.8 TWh in 1992):

Table 2-1. Costs for the front end of the nuclear fuel cycle

	SEK/kWh	Million SEK in 1992
Natural uranium	0.008	490
Conversion	0.001	60
Isotope enrichment	0.007	430
Fuel fabrication	0.008	490
Strategic stockpile	0.001	60
Total front end	**0.025**	**1 530**

The costs for nuclear fuel have decreased during the recent years which is shown in Table 2-2.

Table 2-2. Costs for nuclear fuel 1983-1992.

Year	SEK/kWh
1983	0.038
1984	0.038
1985	0.035
1986	0.031
1987	0.028
1988	0.028
1989	0.028
1990	0.027
1991	0.025
1992	0.025

3 INTERIM STORAGE OF SPENT FUEL, CLAB

3.1 GENERAL

The Swedish interim spent fuel storage facility CLAB located on the Simpevarp peninsula adjacent to the Oskarshamn nuclear power station, was taken into active operation in July, 1985, see Figure 3-1.

The facility has five underground storage pools with a capacity of 5 000 tonnes of uranium. 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 tonnes uranium per year which corresponds to the handling of about 100 fuel transport casks and some 10-20 casks containing reactor core components. For the operation SKB has contracted OKG AB, one of the SKB's shareholders, operating three reactors at the site.

3.2 OPERATING EXPERIENCES

By the end of 1992 CLAB had been in operation for 7.5 years. The performance of the facility has been excellent. Improvements have gradually been introduced along with the experiences gained. In total 1684 tonnes of uranium from the 12 Swedish reactors have been shipped to the facility and placed in storage.

In 1992 56 casks containing spent fuel assemblies from the Swedish reactors were received together with one shipment of fuel residues from past irradiation examination at Studsvik. The total quantity shipped to CLAB during the year amounted to 170 tU. In parallel to the fuel receiving activities 32 BWR assemblies and 32 PWR assemblies have been transferred from old canisters to new compact storage canisters, see section 3.3.

The total occupational dose in 1992 was 135 mmanSv, which is still very low, but considerably more than the year before. The increase can be attributed to two main reasons: work with modification of the existing handling equipment in connection with the introduction of new storage canisters, and with inspections of transport cask internal parts.

The release of radioactivity to the environment during the 7 first years of operation has been negligible, amount-

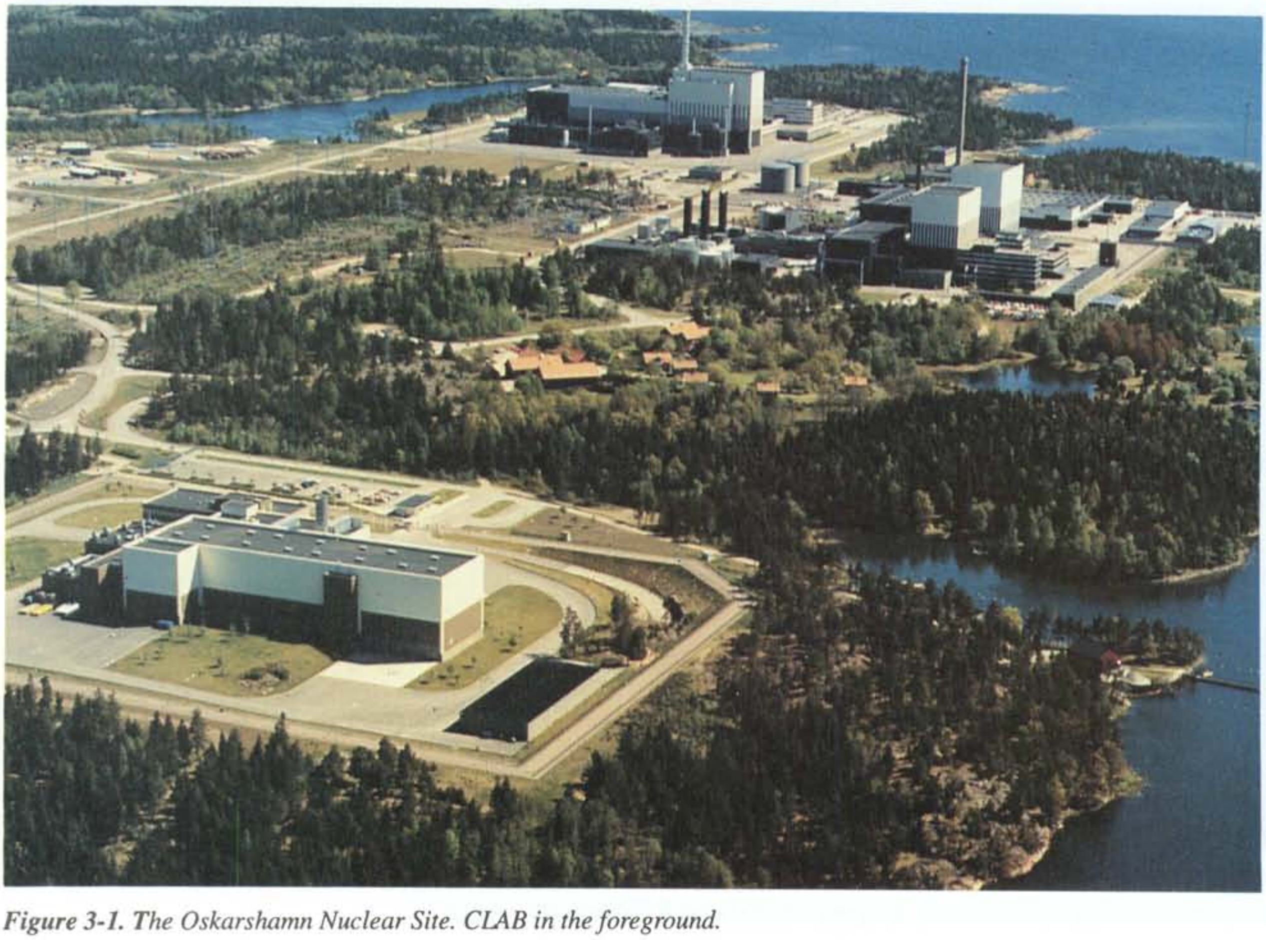


Figure 3-1. The Oskarshamn Nuclear Site. CLAB in the foreground.

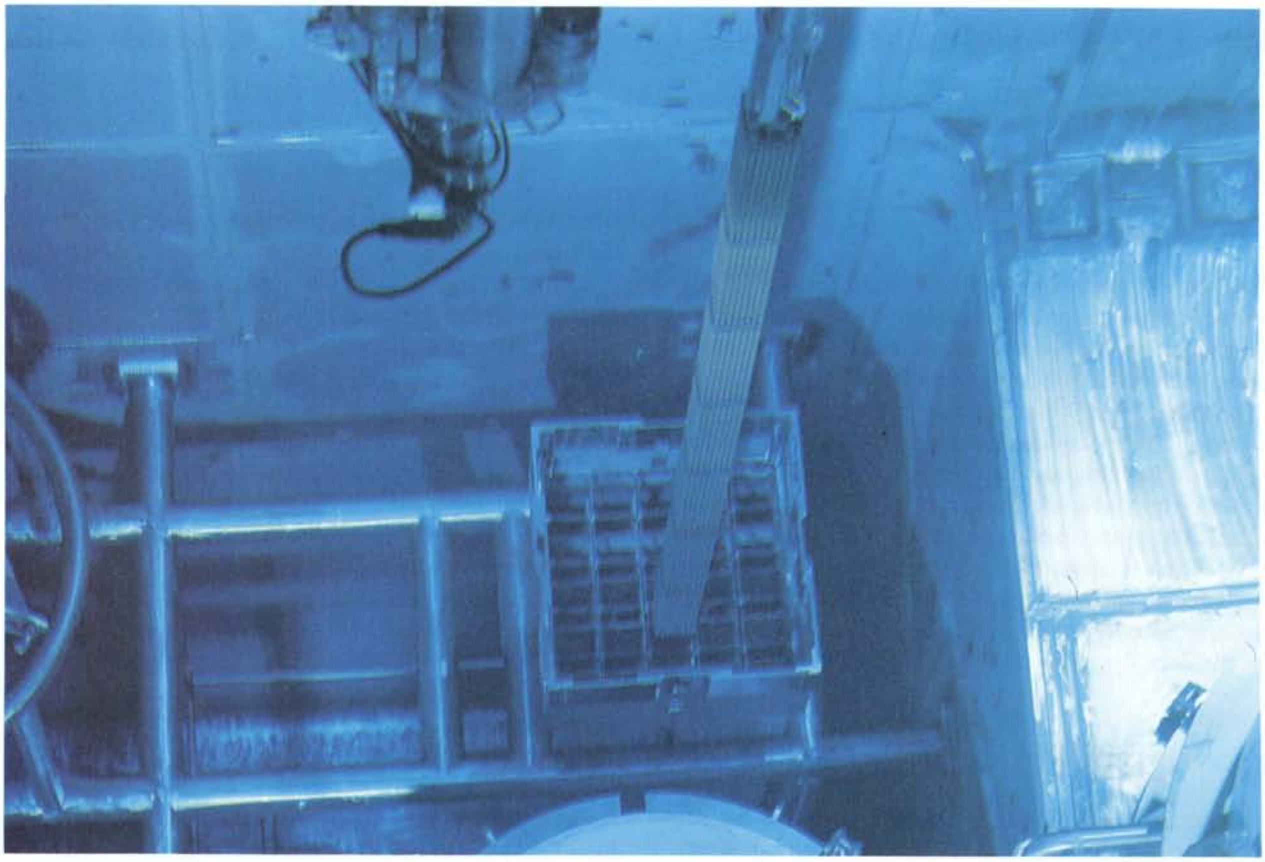


Figure 3-2. *A BWR fuel assembly being loaded into a compact storage canister.*

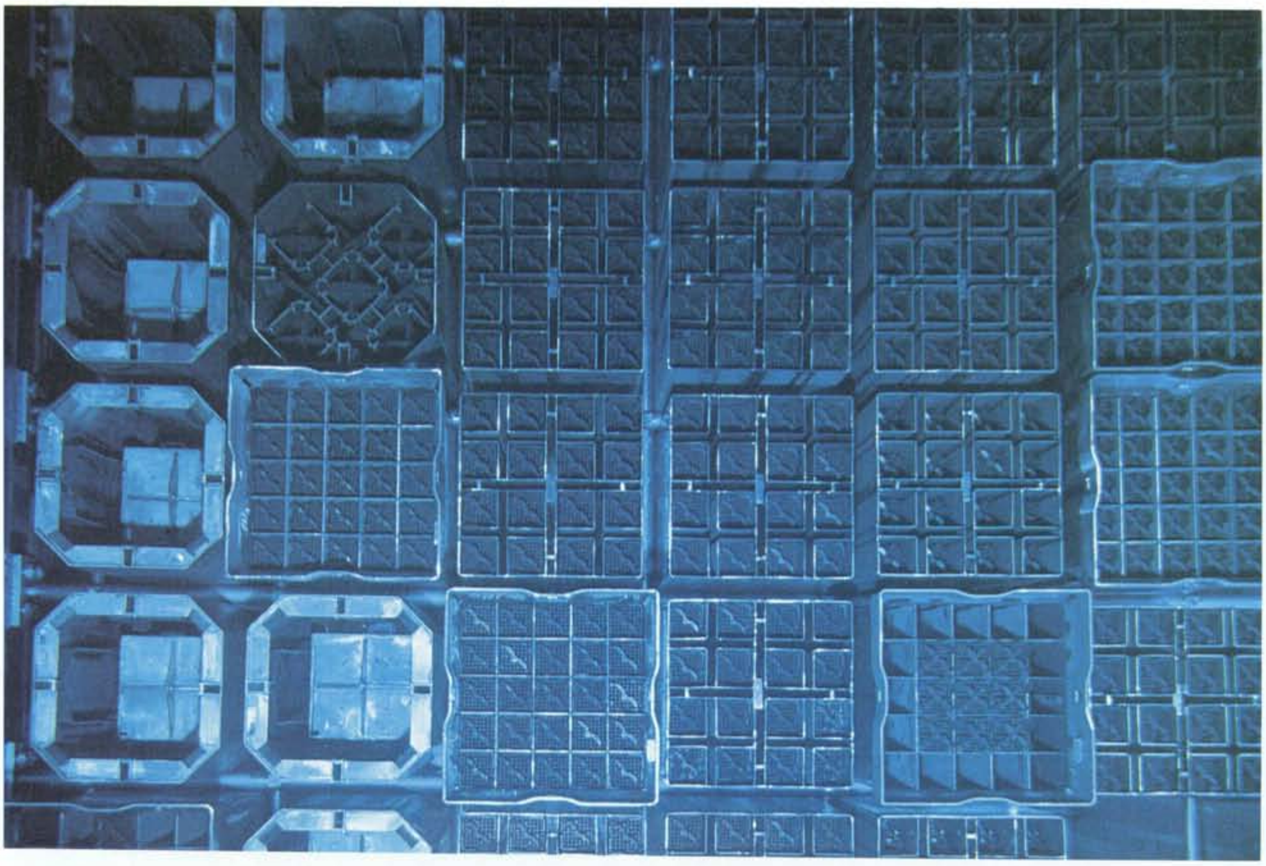


Figure 3-3. *Different canisters in the storage: Octagonal canisters to the left are used for core components. The difference between old canisters and the new compact canisters for BWR fuel can be seen.*

ing to around 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 Studsvik. The operating procedures and involved equipment have been quite easily adapted to the different casks.

3.3 INCREASED STORAGE CAPACITY

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

This has been achieved by using new compact storage canisters with borated stainless steel as neutron absorbing material, allowing the number of fuel assemblies to be increased from 16 to 25 and from 5 to 9 per canister for BWR respectively PWR fuel, see Figure 3-2 and 3-3. Due to this, a new cavern with pools will not be needed until around 2003-2004.

The new canisters came into regular operation in the autumn 1992 and are used for fuel arriving from the reactors and for fuel unloaded from old type canisters. These old canisters are decontaminated and conditioned before being shipped away from the facility.

4 TRANSPORTATION

4.1 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.

4.2 OPERATING EXPERIENCES

In 1992 the ship, M/S Sigyn, sailed around 45 000 n.m. during 165 days. The transports with spent fuel and reactor waste from the Swedish reactors to the CLAB facility and the repository SFR have been performed without disturbances and in accordance with the annual planning. In total 56 transport casks with spent fuel, and 120 IP-2 containers (ATB) with reactor waste have been transported with the transportation system during the year, see Figure 4-1. Like earlier years, no measurable dose rates have been registered to the ship's crew.

A new radar system has been installed at the ship making the operation of navigation system even more reliable.

When the ordinary transport schedule has permitted, M/S Sigyn has been used on commercial basis for transports of heavy equipment, i.e. transformers, generators etc. During 1992 three transports of UF_6-cylinders from Russia to Sweden have been performed with M/S Sigyn.

During the summer period, when normally no scheduled transports are done, M/S Sigyn was used as a floating exhibition of the Swedish nuclear waste handling system, making a voyage along the Swedish coast and visiting 18 cities including the capital, Stockholm.

Figure 4-1. Loading of ATB-container on board M/S Sigyn.

5 FINAL REPOSITORY FOR RADIOACTIVE WASTE, SFR

5.1 GENERAL

The Swedish Final repository for Radioactive Waste, SFR, was put into active operation in April, 1988. It is a repository built in the bedrock under the Baltic Sea close to Forsmark nuclear power plant. 60 metres of rock covers the repository caverns under the sea bed, see Figure 5-1. 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 waste intended for disposal in SFR originates from the operation of Sweden's 12 nuclear power reactors and CLAB. This waste contains mainly short-lived radionuclides and is classified as low- and intermediate level waste. A small amount of similar waste from research and medical activities will also be disposed of in SFR. 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 can also be treated in the same way, if required. These categories are classified as intermediate level waste (ILW) and need shielding during handling and transport. Low level waste (LLW) is treated in different ways, mainly compacted and enclosed in standard freight containers.

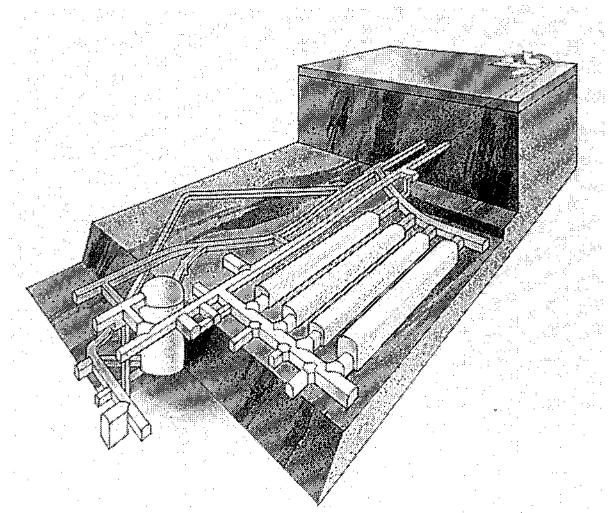


Figure 5-1. Overview of tunnels and storage chambers in the first construction stage of SFR.

At the end of 1992 a total of 11 000 m^3 of waste have been deposited in SFR. All waste producers have delivered waste. The experiences from the operation have been good and the doses to the personnel have been very low.

5.2 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 groundwater flow inside the caverns and through the waste.

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

Waste containing a minor part of the activity content are disposed of in 160 m long caverns with various cross sections. The cavern with the largest cross section, BMA, is equipped with machines for remotely controlled handling, similar to those used in the silo, see Figure 5-2.

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

5.3 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 1992 25 waste types (of a total of about 40) were accepted for disposal. In 1992 disposal has been carried out in the rock chambers and in the silo.

5.4 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.

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.

5.5 OPERATION

The operation of SFR has been subcontracted to Vattenfall AB, the operator of the nuclear reactors at Forsmark, and is closely integrated in the local organization. The staff for operation and maintenance of SFR consists of about 20 people.

In full operation the facility has an annual disposal capacity of about 6000 m^3. During the first years of operation SFR has successively been put into active operation area by area, starting with the rock chambers. Up till the end of 1992 a total of 11 000 m^3 of waste has been deposited.

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.

The operating experience is good both with regard to handling and availability. To overcome some remaining problems with high moisture content in the repository air an air drying system has been taken into operation

During 1992 the first sections in the low-level waste rock chamber (containing drums filled with ashes) was sealed.

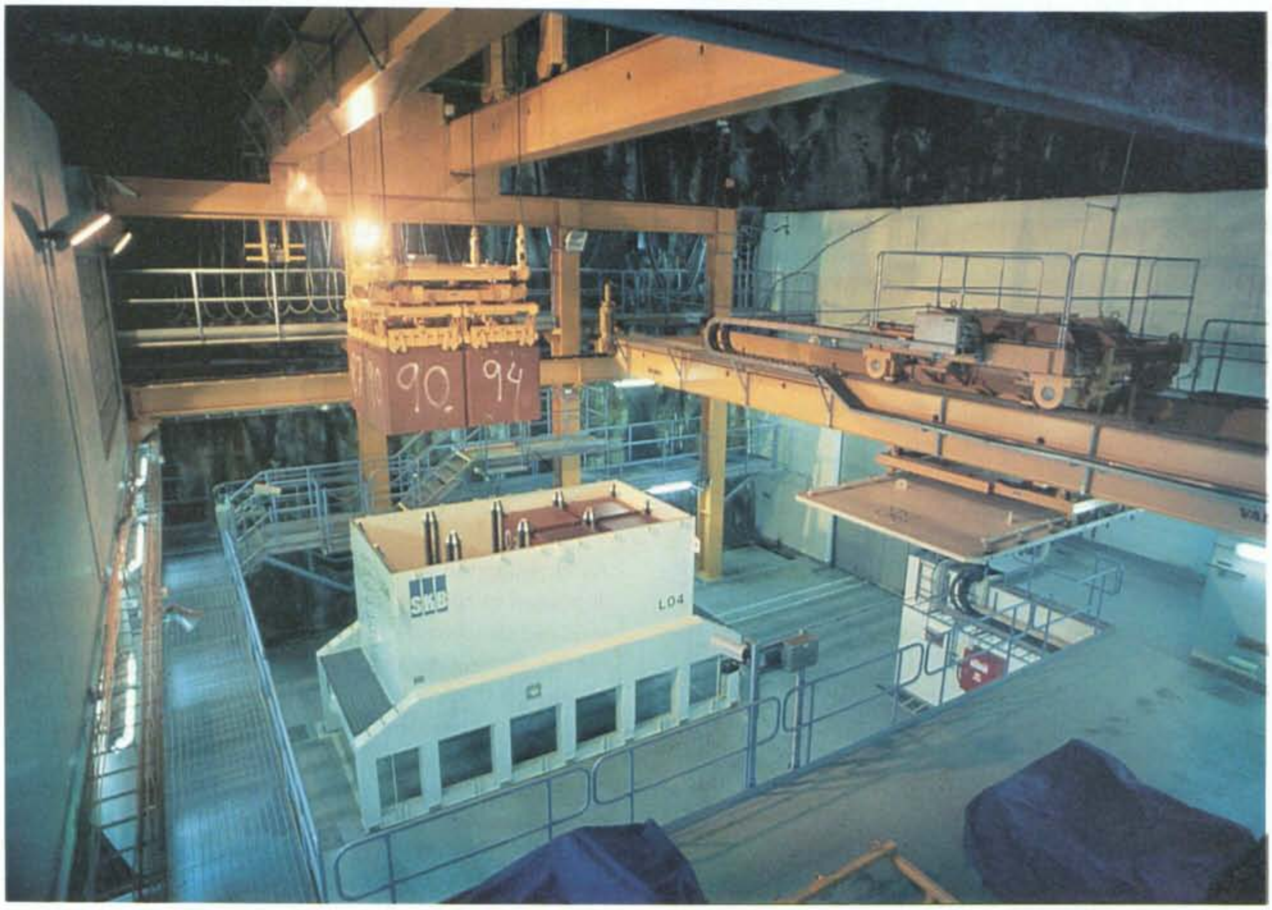


Figure 5-2. The operational waste is transported in special transport containers. In SFR the waste packages are unloaded with remote-controlled handling equipment.

6 RESEARCH AND DEVELOPMENT

6.1 GENERAL

According to the act on nuclear activities (SFS 1992:1536) the owners of Swedish nuclear power plants must together establish a comprehensive programme for the research and development and other measures that are needed i order to take care of all radioactive wastes from nuclear plants in a safe way.

The Swedish nuclear utilities have commissioned the Swedish Nuclear Fuel and Waste Management Co., SKB, to establish the programme required by the law. The programme must be submitted to the Nuclear Power Inspectorate every three years starting 1986. The first programme was submitted by SKB in September 1986 /6-1/, the second programme in September 1989 /6-2/ and the third programme in September 1992.

The third programme /6-3/ was presented in one main report and three background reports. The programme is called RD&D-Programme 92, where RD&D stands for Research, Development and Demonstration. The reason for the change of name compared to previous R&D programmes is to underscore the fact that, starting with the work at the Äspö Hard Rock Laboratory and the plans presented in the programme, the emphasis has shifted towards demonstrating different parts of the selected disposal system. The main report describes the programme in its entirety. The background reports provide more detailed accounts of the R&D work during the period 1993-1998, of the Äspö Hard Rock Laboratory programme and of Siting of a deep repository.

The programme was executed under the leadership of SKBs division for research and development. The staff of the division was 25 persons in 1988. (Note however that there was a major reorganization of SKB early 1993 as a consequence of the new programme.) Some 250 scientists, engineers, specialists and technicians were engaged under contracts with universities, technical institutes, research laboratories, engineering firms and industry. A list of contractors to the SKB R&D-programme is included as Appendix 6 in this report. The results were reported in 45 technical reports in the SKB-TR-serie, in numerous progress reports and working reports and in communications to several international meetings and to scientific magazines. A list of the more important publications is given in Appendix 2.

The work done during 1992 have in general followed the 1989 programme with some amendments during the last quarter as a consequence of the new programme presented in September.

The expenditures on research and development within the SKB budget for 1992 were 179.2 MSEK as compared to 166.8 MSEK in 1991. The increase was due to the increasing work at the Äspö Hard Rock Laboratory.

SKB is also the managing participant in the international Stripa Project which was almost completed during 1992. The expenditures for this project were 13.1 MSEK of which 4.8 MSEK were SKB contributions and 8.3 MSEK came from participants outside Sweden. The total turnover of the R&D-division was thus 192.3 MSEK.

This chapter gives a short overview of the R&D-activities during 1992. The RD&D-Programme 92 is briefly summarized in chapter 10 and a more detailed account of the progress of R&D-activities is given in chapters 11-24 and in a separate annual report for the Äspö Hard Rock Laboratory /6-4/.

6.2 ALTERNATIVE SYSTEM STUDIES

The reference design for spent fuel disposal for SKBs various R&D activities and other studies has been the KBS-3 method published in 1983. The method was extensively evaluated and found acceptable with respect to safety and radiation protection. It formed the basis for the government's approval of starting the reactors number 11 and 12 of the Swedish nuclear power programme.

Beside the KBS-3 method several different methods of disposing spent fuel deep in the bedrock have been or are being studied by SKB. The WP-Cave was evaluated between 1986 and 1988. During 1987 to 1989 a method of disposal in Very Deep Holes, at between 2 and 4 km depth, was developed. These studies have been reported in previous Annual Reports.

During 1990 a Very Long Hole concept was defined. The layout is similar to the Swiss NAGRA's "Project Gewähr" published in 1985 although that concept was developed for vitrified HLW from reprocessing. The studies and development work on alternative repository designs were since late 1990 coordinated in one project called "Project Alternative System Studies", PASS. The main goal for PASS was to evaluate and rank the various alternatives studied. The evaluation considered mainly technological feasibility, long-term and operational safety and differences in cost. The PASS-project was completed in 1992 /6-5/.

The concepts considered in PASS were KBS-3, Very Long Holes, Very Deep Holes and also a design with horizontal emplacement of KBS-3 type canisters in medium-long tunnels – called Medium Long Holes, MLH. For each concept also some alternative canister designs were evaluated. Some studies within PASS were made in cooperation with TVO.

The main conclusions from PASS were:

* A canister with a capacity of 12 BWR assemblies or equivalent thermal load (KBS-3 and MLH) is preferred before a canister with twice the load (VLH).
* A composite canister made of copper and steel is selected as the new reference design. The steel gives mechanical stability and the outer copper container gives corrosion protection. The lead-filled copper canister is studied as back-up.
* The KBS-3 design is maintained as the reference design.

6.3 SAFETY ANALYSIS

Most of the activities within the Safety Assessment Group during 1992 have been connected to completion of the SKB 91 safety assessment and follow-up work to that, and to the planning of the future activities as presented in the new RD&D programme published in september 1992 . Although SKB 91 was not reported until maj 1992, the background and goals for the assessment, the methodology used, the results obtained and the conclusion drawn have already been summarized in SKB Annual Report 1991, and in SKB RD&D-Programme 1992.

The work on improvements of the near-field and far-field modelling has continued.

6.4 SITING OF A FINAL REPOSITORY FOR SPENT FUEL

The siting of a deep geological repository for spent nuclear fuel and other long-lived waste is one of the main remaining tasks within the Swedish Nuclear Waste Management programme. During 1992, the preparation of the RD&D-Programme 92 has implied activities relating to strategy and planning for site selection and site characterization. Studies have also been made on several supporting activities, such as the status of site selection programmes in other countries or the legislation regulating the siting and construction of a deep repository. Parallel to the planning activities a comprehensive GIS (Geographical Information System) database for geoscientific, environmental and other regional data have been made operational within SKB.

It was concluded in the RD&D-Programme 92 that the scientific and technical premises necessary for implementing a safe repository for spent nuclear fuel exist in Sweden and that it is most probably possible to meet the stipulated geological requirements in many parts of the country. When selecting regions for further siting studies other important factors must also be carefully considered, such as plans for land use, transportation of spent nuclear fuel, public opinion and landownership. The programme therefore involves two new components:

- Feasibility studies of those municipalities which express an interest in having a closer examination made of their premises for a deep repository before any selection of sites for field investigations are made.
- Construction of the repository in stages. The first stage being planned for about 10% (800 tonnes) of the projected total amount of spent fuel.

To provide everybody with the same basic information a summary of the RD&D-Programme 92 was distributed to all 286 local municipalities in Sweden, together with a letter in which SKB offered to provide further information and to make a feasibility study in case there would be a mutual interest from a municipality and SKB to study the issue further. In a feasibility study, fundamental facts are gathered and evaluated on transportation-related matters, on social and geological premises and on other important factors for siting a deep repository in the municipality. With the aid of a feasibility study both SKB and the municipality can, at an early stage and without committing themselves, obtain a preliminary data and facts of the premises. Based on such a study decisions can then be taken whether the possibility of siting a deep repository in the municipality is worth examining more closely.

Site-investigations of potentially suitable areas (to be selected after several feasibility studies and general overview studies) are planned for at least two sites in different municipalities. These site investigations will constitute preliminary characterization of the site and give the bases for selecting one site for detailed characterization.

6.5 SPENT NUCLEAR FUEL

A significant part of the experimental effort during 1992 has been devoted to the detailed characterization of spent fuel before and after corrosion tests. The aim of the work has been to identify corrosion sites and to attempt to correlate corrosion processes with fuel structure and radionuclide distribution.

Particular attention is given to examination of the zone at the pellet rim. This zone exhibits significantly higher local alpha activity and fission product contents, than in the bulk of the pellet. Characterization of all reference fuels used in the programme is now completed, although the analysis and evaluation of the data are not yet fully completed.

The fuel corrosion programme was started in 1982 and specimens from the first series (series 3) are still being corroded, having now reached a cumulative contact time of over 4000 days. A second series using a PWR fuel (series 7) rod was started in 1986 and in 1990 a series of experiments on a BWR stringer rod was started (series 11). In 1991 a series (series 13) of experiments was started, using a fuel rod that had been power-bumped up to a maximum linear power of 43 kW/m.

During 1992, the methods for solution analyses were changed. An inductively coupled plasma mass spectrometer (ICP-MS)was installed in the Fuel Laboratory at Studsvik during the late autumn of 1991 and commissioned for radioactive use in June 1992. With the ICP-MS, the analysis procedures has been very much simplified, allowing the analysis of a number of elements without previous separation. For most elements, the detection limits have also been lowered considerably. In addition to implementing the new analysis procedures for experiments to be analysed during the latter part of 1992, much work has been devoted to re-analysis of archive solutions from earlier experiments.

A detailed experimental study has been performed of the oxidation products of uraninite and its fine-grained variety, pitchblende. Samples of uraninite and pitchblende annealed at 1200°C in H_2, and untreated pitchblende were sequentially oxidized in air at elevated temperatures. Untreated pitchblende during oxidation behaved similarly to irradiated UO_2 in spent nuclear fuel; whereas, reduced pitchblende more closely resembled non-irradiated UO_2.

When studying oxidative dissolution of UO_2, thermodynamic and kinetic data for relevant oxidized natural solid phases are needed. Dissolution experiments are being performed on a selected and well characterized natural samples of alteration chain of uraninite (i.e., uraninite, schoepite, uranophane). The experiments are performed using a synthetic granitic groundwater as a leachant, in contact with air at 25°C.

6.6 CANISTERS

As for the past few years, the studies during 1992 have been focused on long-lived canisters with copper as the outer corrosion barrier. The studies have included investigations of the chemical and mechanical stability of copper and investigations of the corrosion of carbon steel under aerobic and anaerobic conditions.

The reference group for mechanical integrity of canisters, which was formed in 1990 has finalized its work during 1992. One of the conclusions of the reference group was that the copper-steel canister was a favourable alternative from the mechanical point of view. The main thrust of the work during most of 1992 has therefore been concentrated on this alternative. The reference group also recommended that an oxygen-free copper (Cu-OFP) containing a small amount of phosphorus should be selected as candidate canister material for further studies. The reason for this was the very low creep ductility found for oxygen-free copper at elevated temperatures.

Stress relaxation tests have been performed on Cu-OFP to provide data for modelling the deformation behaviour of copper canisters during long-term service. The stress relaxation tests at 75 to 150°C showed that the initial stress relaxation rate is rapid; the stress falls by at least 30% within the first 200 hours.

In order to estimate the risk for creep fracture of the weld region during the design life of the canister, the temperature, strain and stress fields during welding and after cooling were calculated using the finite element code NIKE-2D. The clearing between the steel and the copper cylinder is crucial for the residual stress distribution.

The welding residual stresses are found to redistribute without lowering the maximum values during the waiting period. A very low amount of void growth is predicted for this type of copper during the deposition period. Thus, a very long time to rupture can be estimated.

A rock shear movement perpendicular to the canister's axis could threaten its integrity. In order to illucidate the consequences of such an event stresses and strains in hot isostatically pressed (HIP) canisters as well as in copper-steel canisters have been calculated for a movement of the surrounding rock of 10 cm. It was concluded from the calculations that the effect of a rock shear movement depends more strongly on the density of the bentonite than on the canister type. For the copper-steel canister the maximum strain at the cylindrical part is 2.5%. It can thus be concluded that a rock shear movement of the size considered poses no immediate threat to the integrity of the canister.

A review of the current knowledge of copper corrosion under repository relevant conditions has been made during 1992. The main conclusions are that general corrosion is very unlikely to lead to canister failure and that processes other than corrosion are more likely to endanger the integrity of the canister. Some corrosion processes, however, will require further attention. The most important ones being stress corrosion cracking and pitting corrosion under mildly oxidizing conditions.

6.7 BUFFER AND BACKFILL

One important issue which has been studied is the alteration processes of clay in the repository environment. A joint test, completed in 1992 between SKB and CEA, France, on MX-80 bentonite that had been exposed to high temperatures and gamma radiation during a year, showed no significant differences between the samples exposed to radiation and those which were not. This is in agreement with the general opinion that the direct effect of gamma radiation is small. In another work samples were taken from clay layers on southwestern Gotland which now consist of hydrous mica but are assumed to have bentonite origin. The objective was to study the conversion parameters. No bentonite was, however, identified at 75 m depth, as was the case at 500 m depth in the closely situated borehole at Hamra.

A second important issue during 1992 has been modelling of the physical properties of buffers. The material models are applied to calculation programs, although primarily the ABAQUS FEM program, for analysis of different functions and scenarios. The work is supported by laboratory tests. Triaxial tests as well as swelling/compression tests have continued with different bentonite qualities. An extensive test on bentonite swelling into slots was conducted aiming at reflecting swelling into open fractures. It was concluded that a model exists for calculating the behaviour of water saturated bentonites with different pore water at different temperatures between 5 and 90°C. The model, however, is providing linear relationships in some cases where the relationship is non-linear. The computer program also lacks data of non-saturated bentonite buffers.

6.8 GEOSCIENCE

The ancient Swedish crystalline 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.

During 1992 the geoscience programme has among others involved the following R&D activities.

Thermal, hydrological and mechanical processes might have a mutual influence on each other to a greater or lesser extent and affect the rock mass behaviour. SKB has initiated a three year programme for a better understanding of the detailed flow regimes within fractures or fracture intersections. The programme will mainly be done on a laboratory experimental scale and by means of a bi-axial cell equipment.

Besides regional groundwater modelling under today's climatic situation, it is essential to shed light on the hydraulic conditions in connection with future glaciations and deglaciations. A time dependent glaciation model of Scandinavia has been developed for the coming 120 000 years. The model also includes surface boundary conditions for loading and groundwater flow.

The swedish nuclear waste management programme has focused on granite and gneiss as the major candidate host media for a repository. A study has been elaborated which summarizes and examines existing geoscientific knowledge of relevance in assessing the potential merits of gabbro as a repository host rock.

In conclusion, there are obvious difficulties associated with siting a repository in gabbro, due to lack of sufficiently large gabbro bodies. In comparing gabbro with granitic rocks, no decisive differences can be demonstrated on the basis of the present state of knowledge, neither with respect to repository construction, nor as regards repository performance.

To obtain a better understanding of the water flux in a regional perspective, surrounding Äspö HRL, and at depths exceeding 1 000 m, a separate borehole was drilled in autumn 1992. The coredrilling was carried out in the Laxemar area near the Simpevarp peninsula in the municipality of Oskarshamn and reached a depth of 1700.5 m. The coredrilling is the deepest one in Scandinavia. After concluded drilling, geophysical, hydrochemical, geochemical and hydraulic investigations will now be performed in the hole.

6.9 CHEMISTRY

The chemistry program covers geochemistry, radionuclide chemistry, and migration. The investigations of groundwater and geochemistry have been concentrated to the Äspö Hard Rock Laboratory. Geochemical research is also being conducted within the frame of the natural analogue investigations. Important has also been the joint TVO/SKB study of groundwater sample quality. The aims of this exercise is to improve site investigations and selection of data for modelling.

The preinvestigation phase data from Äspö have been used to make an evaluation of the hydrogeochemistry. Geohydrological and geological information were used in order to support and control the results. However, using the hydrochemistry information it was also possible to develop a more detailed picture of the groundwater flow situation under Äspö.

During 1992 sampling of groundwater has been concentrated to drillholes along the Äspö entrance tunnel. The tunnel runs slightly inclined for 1.5 km from the entrance to a depth of about 200 m. Part of the tunnel is under land and part of it under sea. A distinct difference in composition has been obtained, which probably reveals some of the processes that form the groundwater composition below the sea sediments.

The redox experiment in the Äspö entrance tunnel has continued. The main aim of this project is to find out the consequences on local redox conditions when a repository is kept open during it's phases of construction and operation. So far no oxygen breakthrough has been observed and the capacity to withstand oxidation depletion is higher than expected.

The thermodynamic constants that determine the solubility and other chemical behaviour of thorium and neptunium in reducing, carbonate containing water have been measured in the laboratory. This is relevant for radionuclides in deep groundwater or bentonite pore water. Conclusions have also been drawn on plutonium(IV) by analogy. This does not cover the influence of the humic substances that exists in groundwater. However, separate experiments have been performed which elucidate this aspect. Fundamental experiments have also been performed on the importance and transport properties of colloidal particles.

Microbe analyses have been concentrated to the Äspö groundwater. Microbe strains are identified and their concentrations measured. It is important to find how they influence the geochemical processes. New technique is being tried in order to identify the bacteria.

Surface complexation and ion exchange are being tested as models to describe in a more fundamental way sorption of radionuclides on minerals and diffusion of radionuclides i bentonite backfill.

Efforts are being made to develop a probe for in situ studies of radionuclide chemistry and migration. The design of the CHEMLAB probe is being made by IPSN/CEA at Cadarache.

6.10 STRIPA PROJECT

The international Stripa project was almost completed in 1992. The experiments were finished by June 30, 1991 and all reports from the principal investigators with their evaluations have been published by the end of 1992. Two major overview reports on Natural Barriers and on Engineered Barriers respectively were still in the final preparation stage together with a general summary report. These three reports are expected to be ready for distribution in the spring of 1993.

The Stripa project was officially finished by a seminar in Stockholm organized by OECD/NEA and SKB October 14 – 16, 1992. The seminar attracted some 150 scientists from more than a dozen countries.

Chapter 20 gives a thorough summary of the results of the Stripa Project.

6.11 NATURAL ANALOGUE STUDIES

The last reports from the Poços de Caldas project have been printed. The results and the investigations have been summarized and published as a special issue of Journal of Chemical Exploration. The content of this issue is also available as a book from Elsevier Co, Amsterdam 1993.

The uranium mineralization at Cigar Lake in northern Saskatchewan, Canada, has been studied by AECL as a natural analogue to deep disposal of spent fuel since 1984. SKB joined the project in 1989 and Los Alamos National Laboratory, supported by US DOE, participates since 1991. The three year phase starting in 1989 was finished in 1992. The Cigar Lake project has been divided up into a number of discipline oriented tasks: colloids, geology, hydrogeochemistry, hydrogeology, mineralogy and lithogeochemistry, nuclear reaction products, ore mineralogy, organics and microbiology and finally radiolysis. The results of these studies are used for performance assessment related evaluations and modelling. Four performance assessment objectives were chosen: 1) The evaluation of thermodynamic equilibrium codes and their databases; 2) The role of colloids, natural organic substances and microbes for the migration of radionuclides; 3) The stability of UO_2 and the influence of radiolysis on UO_2 dissolution and radionuclide release; 4) The testing of mass-transport models for radionuclide migration through clay barriers. The evaluations and model calculations within these four tasks are being summarized.

The 2 billion years old reactor zones in Oklo, Okelobondo and Bagombe are being investigated as analogues to waste repositories. The study is directed by the French CEA and supported by CEC. Organizations from other countries including SKB are participating in the study. Our involvement has during 1992 to a large extent been devoted to the fossil reactor in Bagombe, which is situated about 20 km away from Oklo and Okelobondo. This reactor zone is close to the ground surface but has been seemingly well preserved. A number of core drilled holes have been made at Bagombe during September to December 1992. Core samples were collected, hydraulic measurements were made and groundwater samples taken. SKB also participates in the studies at Oklo and Okelobondo.

The natural hyperalcaline conditions found in Jordan have been investigated as an analogue to the concrete-rock environment anticipated for underground disposal of low- and intermediate level waste. The first phase of the project was jointly funded by NAGRA, NIREX and Ontario Hydro. Together with NAGRA and NIREX, SKB has been supporting the second phase of the study. One important objective of the study has been to make blind calculations of trace element solubilities and speciation in the alcaline waters. Different teams have been involved using different data bases. The trace elements U, Th, Ra, Se, Pb, Ni, Sr and Sn have been regarded.

6.12 BIOSPHERE

The biosphere studies treat the transport of radionuclides from the bedrock via primary receptors (e.g. sediments), redistribution in nature and finally calculate the exposure dose to man and other biota. The activities have gradually switched from investigation of general data, processes and methods, to confirmation of the models and methods used and acquisition of relevant site specific data. International cooperation is very important in the process of model confirmation. SKB follows and take active part in several international projects i.e. BIOMOVS II, VAMP, PSAC and PAAG. Site specific data are studied in the Gideå area since the Tjernobyl fallout in 1986, and in the Äspö area.

In the bedrock safety analysis SKB 91, one deterministic set of dose conversion factors was used as a measuring stick to compare dose rate from different nuclides. These factors represented a pessimistic, but not unrealistic situation with a self supplying farm situated at the release point.

6.13 INTERNATIONAL COOPERATION

SKB recognizes the need for extensive international cooperation and information exchange. Both written and informal agreements with organisations in other countries dealing with nuclear waste have been established. Exchange of up-to-date information are carried out through joint projects, meetings, joint seminars or short visits of specialists to the other signatories' facilities.

During 1992 the most active general cooperative work has been performed together with TVO in Finland, CEA in France and AECL in Canada. Also the work in different working groups and projects put together by international organisations like CEC EURATOM, OECD/NEA and IAEA has a high priority in the SKB programme.

During 1992 the number of participating organizations in the cooperative work at the Äspö Hard Rock Laboratory has grown considerably. Four organizations have signed contracts during 1992: Agence Nationale Pour la Gestion des Déchets Radioactifs (ANDRA) of France, Teollisuuden Voima Oy (TVO) of Finland, UK NIREX of the UK and the United States Department of Energy (USDOE). The organizations that earlier had signed contracts for participation were Atomic Energy of Canada Limited (AECL), the Power Reactor & Nuclear Fuel Development Corporation (PNC) of Japan and the Central Research Institute of the Electric Power Industry (CRIEPI) of Japan.

In October 1992 the Fourth International Symposium on the OECD/NEA Stripa Project was held in Stockholm. The Symposium marked the endpoint of the project and summarized the findings from the work conducted at Stripa Phase III.

Other important meetings in Sweden 1992 are among others the "Rock Mechanics meeting" in Stockholm in March, the "Seminar on the dynamics of the behaviour of radionuclides in contaminated forests" in Stockholm in May and the "Spent Nuclear Fuel Workshop" held in Visby in September.

7 COST CALCULATIONS

7.1 COST CALCULATIONS AND BACK-END FEE

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

Each year SKB calculates the future electricity production and the future costs for the back-end operations related to this electricity production. The results of the 1992 calculations were presented in PLAN 92 /7-1/. The total future electricity production (from 1992) was estimated to be about 1 280 TWh, if all twelve reactors are operated to year 2010. Up to the end of 1991 about 740 TWh have been produced making a total of about 2 000 TWh in the Swedish programme. For this production a fuel volume of about 7 900 tonnes of U is required.

The total future back-end costs were estimated to be about GSEK 46.4 (price level of January 1992). Up to and including 1992 already SEK 8.7 billion have been spent. The total cost for the back-end of the nuclear fuel cycle is thus about SEK 55 billion. The breakdown of the costs are roughly (old reprocessing costs excluded):

Transportation of waste	4%
Interim storage of spent fuel	19%
Encapsulation and final disposal of spent fuel and long-lived waste	39%
Final disposal of operational and nuclear power plant decommissioning waste	5%
Decommissioning and dismantling of nuclear power plants	22%
Miscellaneous including R&D, pilot facilities, and siting	11%

Based on SKB's cost calculations and a discussion about the time of operation of the reactors and the estimated real interest rate, the government has decided that the fee for 1993 shall be SEK 0.019 per kWh on an average. This is the same fee as for the last nine years.

The fee is periodically paid into funds at the Bank of Sweden. These funds are administrated by The Swedish Nuclear Power Inspectorate (SKI), who in 1992 took over this responsibility from the National Board for Spent Nuclear Fuel, SKN. The total sum in the four funds was at the end of 1992 about GSEK 11.5, an increase by GSEK 1.8 since 1990.

7.2 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.

7.3 DECOMMISSIONING OF NUCLEAR POWER PLANTS

During 1992 SKB's engagement continued in the international cooperate programme, which is sponsored by OECD/NEA. SKB is responsible for the programme coordinator function. This programme comprises 22 decommissioning projects in 11 countries. The majority of the projects are small first generation power demonstration reactors.

The projects include all stages of decommissioning from preparation for a long-term rest and surveillance period of the plant to a total dismantling. Examples of the latter are the Shippingport reactor, where dismantling was completed in 1988, the Japanese JPDR reactor, where dismantling is in progress and the reactor pressure vessel was removed in 1990, and the Niederaichbach reactor.

Earlier studies of the dismantling of the Swedish reactors have shown that there is no immediate need for substantial decommissioning R&D in Sweden. A study of the possibilities to remove the reactor pressure vessel in one piece, and transport it intact for disposal is, however, in progress.

The costs of decommissioning constitute a large part of the total backend costs. The costs used are based on a study performed in 1986. A new cost study has been initiated to update the costs based on the latest developments. In this a detailed study of the decommissioning of one of the newest BWRs is performed.

8 CONSULTING SERVICES

8.1 BACKGROUND

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

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

Since 1984 there is a special group, NWM (Nuclear Waste Management), within SKB for marketing and management of external services. For each assignment a tailored project team is organized with due consideration of the competence required, see Figure 8-1. It may be experts from SKB's own staff or from groups contracted for different tasks in the Swedish radioactive waste management programme.

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 some 80 assignments have been accomplished for organizations in Australia, Belgium, Canada, former Czechoslovakia, Hungary, Japan, Lithuania, 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 and facility design.

8.2 NWM WORK DURING 1992

During 1992 SKB was contracted by organizations in Belgium, United Kingdom, Finland, Japan, Canada, Lithuania, Spain, South Korea and Taiwan. Further services have been discussed with these countries and also with Australia, Romania, Slovenia and USA. Some general frame agreements have been signed. In all, twelve assignments have been concluded, distributed over ten countries.

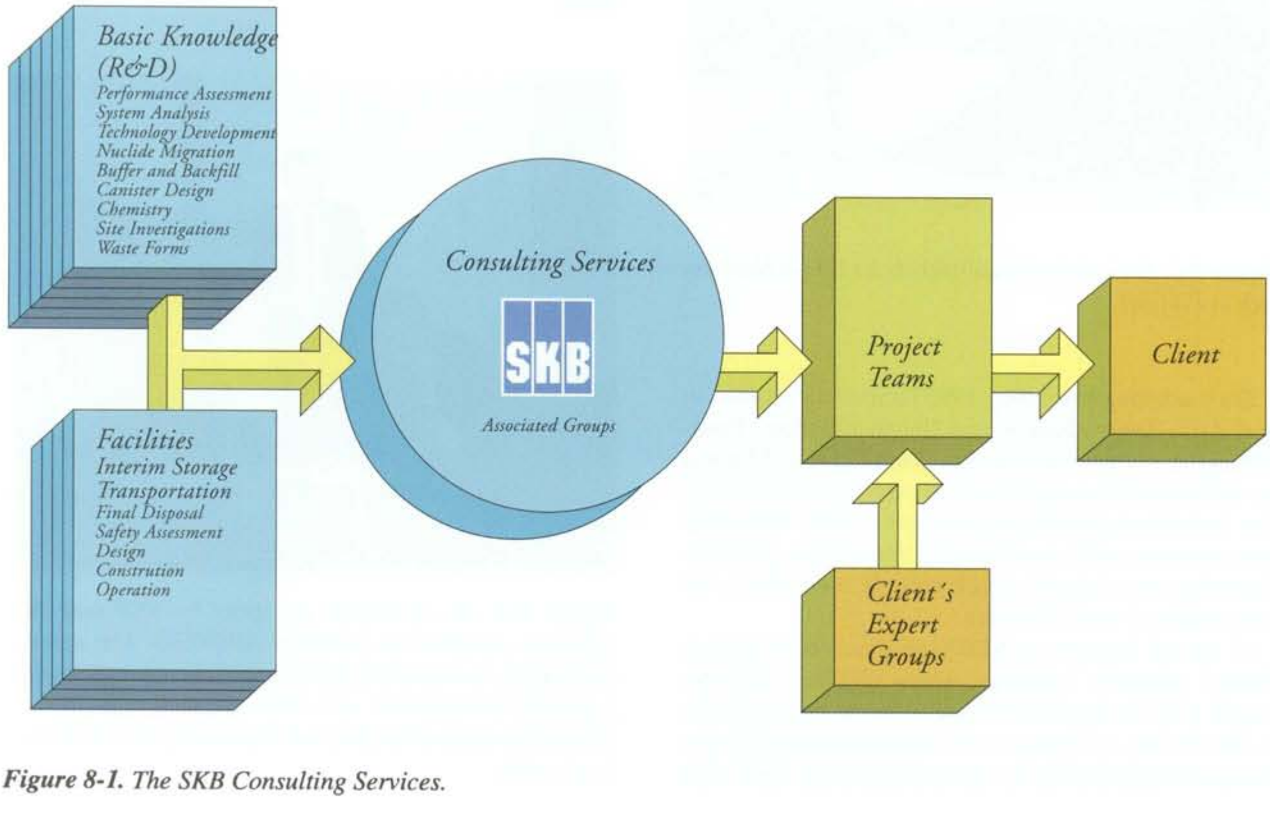


Figure 8-1. The SKB Consulting Services.

Figure 8-2. Representatives from SKB visiting Lithuania, just above the reactor at Ignalina-1 nuclear power unit.

Figure 8-3. Rock stress measurements for IVO at Hästholmen in Finland.

The marketing has during 1992 focused on contacts in East Asia, Eastern Europe and United Kingdom. Possibilities for continued activities in South Korea and Taiwan are still regarded as good and are carefully followed-up. The waste management programmes in these two countries, however, suffer continuously from delays, primarily depending on a negative public attitude connected to the localization of waste facilities.

Of special interest are SKB's assignments in Eastern Europe, primarily Lithuania, where SKB has been entrusted with the responsible task to be the prime advisor to the Ministry of Energy in the implementation of a dry interim storage facility for spent nuclear fuel. SKB shall also assist in the preparation of a national long-term radioactive waste management programme and in the solving a number of acute waste-related problems, see Figure 8-2.

Discussions with NIREX in United Kingdom has resulted in a contract on borehole radar measurements in the Sellafield area.

For ENRESA, the Spanish organization corresponding to SKB, studies regarding a suitable design of a final repository for spent nuclear fuel in granite have continued. A reference concept has been presented and further developed by INITEC, ENRESA's Spanish Consultant with SKB as advisor.

For IVO in Finland SKB has measured rock stresses at Hästholmen, the site where IVO intends to build a repository for low- and intermediate-level waste, see Figure 8-3. By using high water pressure the rock in a packed-off section of a borehole is fractured and the rock stresses can then be determined. The close cooperation with TVO in Finland includes some commissions to SKB. During 1992 a study has been made regarding the possibilities to use natural analogues for validation of safety assessments and modelling.

For ONDRAF, the organization corresponding to SKB in Belgium, experience and know-how have been transferred from the SFR facility regarding gas generation processes, volumes, requirements on the properties of backfill concrete etc.

For organizations in Japan SKB has compiled a report on the procedures for legal permits for the Äspö Hard Rock Laboratory. Know-how about bentonite clay has been transferred to the Japanese construction company SHIMIZU, see Figure 8-4. Supplementary equipment for borehole radar measurements has been delivered to South Korea.

Figure 8-4. An agreement is signed by SKB and the Japanese construction company SHIMIZU. The agreement refers to consulting services in the preparations for a possible construction of a SFR-type facility in Japan. The facility is intended for final disposal of intermediate-level waste.

AECL of Canada has entrusted SKB to review their long-term performance and safety assessment. The assessment refers to the Canadian reference concept for spent fuel disposal.

A number of assignments from i.a. Radwaste Administration in Taiwan, Hungary and former Czechoslovakia have been concluded.

9 PUBLIC AFFAIRS AND MEDIA RELATIONS

9.1 GENERAL

The Swedish nuclear power utilities are obliged by law to take all measures necessary to ensure safe handling of radioactive waste. SKB is responsible for the fulfilment of this obligation. In order to realize its radioactive waste management programme SKB has to obtain the confidence of the community. This calls for the comprehension of its present and future activities not only by scientists and technicians but also by politicians and the general public.

A well-founded and lasting confidence can only be established through demonstration of the high scientific and technical standards prevailing in the work of SKB. It is thus essential that those who have to judge the performance of SKB, or are otherwise affected by its activities, possess a wide knowledge of the aspects of radioactive waste management.

It is imperative to provide knowledge that enables the community to put the issue of radioactive waste management in its proper perspective. The issue must neither be exaggerated nor played down. Impartial information and open discussions are necessary for the community's justified demands for clear insight. Such information and discussions are also a prerequisite for the democratic decision-making process, and should enable an objective debate. In practice, the costs for the waste management is paid by today's consumers of electricity. They, in particular, are entitled to comprehensible, impartial and objective information.

The aim of the SKB information activities is to widen and deepen the community's knowledge regarding:

- Radioactive waste; its properties, its potential dangers and present and future quantities of waste.
- Basic ethical and technological principles that govern the work. The radioactive waste is and will be handled responsibly, putting high demands on safety. The planned systems have to be designed to ensure that no environmental or financial burden be put on future generations.
- The Swedish system which has been developed and built up by SKB and which is now handling all radioactive waste on a long term basis.
- The research, development and planning work carried out by SKB and others and the existing know-how concerning the options for the very long-term isolation of the waste.

9.2 SKB INFORMATION ACTIVITIES

The most efficient way of bringing forward information is through a two-way communication process between people. For such purpose SKB has established a comprehensive exhibition programme. Almost all the year round a mobile trailer visits schools, towns and rural districts as well as public exhibitions, see Figure 9-1. For four years running exhibitions have toured the country and during 1992 the SKB exhibition trailer visited 27 districts. In the autumn of 1992, a small, flexible, mobile trailer, which can easily enter schools' playgrounds and market-places of small towns, was added to the SKB information equipment.

During the course of the exhibitions SKB staff gave information about the handling and disposal of radioactive waste and the research work carried out. As previous years, the visitors were able to study models and dummies of the equipment used in the handling of waste as well as proper, life-size equipment such as e.g. casks used for the transport of waste. Also as previous years, the public showed great interest in the exhibitions. There were around 100.000 visitors representing the general public, schools at the level of upper secondary education, local political decision-makers as well as various associations.

As usual in the summer months an exhibition was arranged on board the transport vessel M/S Sigyn, see Figure 9-2. During 1992 she called at 15 ports. This again enabled SKB to meet, face to face, with members of the general public, students, political decision-makers as well as representatives of industry. 76.000 visitors were received. Apart from visiting the exhibition in the cargo hold they were allowed to enter the captain's bridge where the ship's crew demonstrated navigation and radio equipment. A new feature was a combination of walking and quiz competition for adults and children. Vital questions during the discussions concerned other countries' systems for the management of radioactive waste, and whether there exist different systems for the handling of spent fuel.

SKB also took part in public exhibitions such as the Energy Exhibition at Jönköping, the Nora Market, Books and Libraries in Gothenburg, "Women Know How", Technical Exhibition, and Education-92 in Stockholm. This SKB exhibition was also presented at specially arranged information events at Ringhals and Forsmark. About 10.000 people visited these exhibitions.

The facilities of CLAB, SFR and the Äspö Hard Rock Laboratory are open to visitors. Here permanent exhibi-

Figure 9-1. *A mobile exhibition trailer visits schools, towns and rural districts almost all year round, to inform people about radioactive waste management. In 1992, the SKB trailer mostly visited schools.*

Figure 9-2. *The SKB ship M/S Sigyn is used each summer as a floating exhibition hall, visiting harbours all around the Swedish coast. The picture shows a SKB guide, demonstrating a dummy fuel element.*

tions are found all year round. During 1992 SFR received approximately 19.000 visitors, and CLAB and Äspö around 6.500 and 2.000 respectively.

SKB information activities at schools expanded during 1992, see Figure 9-3. 250 classes were visited and altogether 5.500 students were informed about SKB. SKB also attended special educational projects treating a certain theme such as e.g. the environment, where students were given the opportunity to discuss issues and put questions. Further education for teachers was arranged.

9.3 SKB INFORMATION MATERIAL

In the course of the year a new research programme RD&D-Programme 92 was presented which involved updating of a great part of SKB information material. A number of information pamphlets were produced, which in a popular-science-manner describe the contents of some of SKB's technical reports.

"Lagerbladet", the SKB newsletter, was distributed to more than 25.000 subscribers.

During the year a new educational pack, "In Depth" was prepared for use in schools. The material is written for youths, 16-19 years of age, and contains both information, discussions and activities for the students.

"Speaking mirrors" were placed at the airports of Umeå and Kalmar in 1992. The mirrors show a multislide presentation providing information about the activities of SKB.

SKB further has a wide choice of information material such as brochures, reports, video cassettes, overhead transparencies including scripts, audio cassettes, mini-exhibitions etc. An additional channel of information is the "Pekoteket" which is an interactive computer programme where audio and visual information can be obtained by pointing at a computer screen.

In order to facilitate for the public to obtain information, SKB offers Freepost on letters of inquiry to the company as well as a nationwide 020 number where information material can be ordered around the clock at the cost of a local call. The Freepost service is available also from abroad under the address, FRISVAR, Svensk Kärnbränslehantering AB, S-110 05 Stockholm, Sweden.

***Figure 9-3.** Radioactive waste on the agenda. During 1992, SKB expanded its information activities at schools, visiting today's high school students – tomorrow's decision makers.*

REFERENCES PART I

Chapter 1

1-1 **NEW SWEDISH NUCLEAR LEGISLATION.**
Swedish ministry of industry, DSI 1984:18

Chapter 6

6-1 **R&D-Programme 86, Parts I-III. Handling and final disposal of nuclear waste. Programme for research, development and other measures.**
SKB, September 1986

6-2 **R&D-Programme 89, Parts I-II. Handling and final disposal of nuclear waste. Programme for research, development and other measures.**
SKB, September 1989

6-3 **RD&D-Programme 92**
Treatment and final disposal of nuclear waste. Programme for research, development, demonstration and other measures.
SKB, September 1992

6-4 **Äspö Hard Rock Laboratory Annual Report 1992.**
SKN Technical Report TR 93-08, Stockholm

6-5 **Project on Alternative Systems Study (PASS) Final Report.**
SKB Technical Report TR 93-04, Stockholm

Chapter 7

7-1 **PLAN 92**
Costs for management of the radioactive waste from nuclear power production.
SKB Technical Report TR 92-24, Stockholm

SKB ANNUAL REPORT 1992

Part II

Research and Development during 1992

CONTENTS PART II

Page

Page

10 RD&D-PROGRAMME 92

10.1 GENERAL

According to the act on nuclear activities (SFS 1992:1536) the owners of Swedish nuclear power plants must together establish a comprehensive programme for the research and development and other measures that are needed in order to take care of all radioactive wastes from nuclear plants in a safe way.

The Swedish nuclear utilities have commissioned the Swedish Nuclear Fuel and Waste Management Co. – SKB – to establish the programme required by the law. The programme must be submitted to the Nuclear Power Inspectorate every three years starting 1986. The first programme was submitted by SKB in September 1986 /10-1/, the second programme in September 1989 /10-2/ and the third programme in September 1992.

The third programme /10-3/ was presented in one main report and three background reports. The programme is called RD&D-Programme 92, where RD&D stands for Research, Development and Demonstration. The reason for the change of name compared to previous R&D programmes is to underscore the fact that, starting with the work at the Äspö Hard Rock Laboratory and the plans presented in the programme, the emphasis has shifted towards demonstrating different parts of the selected disposal system. The main report describes the programme in its entirety. The background reports provide more detailed accounts of the R&D work during the period 1993-1998, of the Äspö Hard Rock Laboratory programme and of Siting of a deep repository.

The programme was executed under the leadership of SKBs division for research and development. The staff of the division was 25 persons in 1992. (Note, however, that there was a major reorganization of SKB early 1993 as a consequence of the new programme.) Some 250 scientists, engineers, specialists and technicians were engaged under contracts with universities, technical institutes, research laboratories, engineering firms and industry. A list of contractors to the SKB R&D-programme is included as Appendix 6 in this report. The results were reported in 45 technical reports in the SKB-TR-serie, in numerous progress reports and working reports and in communications to several international meetings and to scientific magazines. A list of the more important publications is given in Appendix 2.

The work done during 1992 have in general followed the 1989 programme with some amendments during the last quarter as a consequence of the new programme presented in September.

The expenditures on research and development within the SKB budget for 1992 were 179.2 MSEK as compared to 166.8 MSEK in 1991. The increase was due to the increasing work at the Äspö Hard Rock Laboratory.

SKB is also the managing participant in the international Stripa Project which was almost completed during 1992. The expenditures for this project were 13.1 MSEK of which 4.8 MSEK were SKB contributions and 8.3 MSEK came from participants outside Sweden. The total turnover of the R&D-division was thus 192.3 MSEK.

The remaining part of this chapter gives a brief overview of the main content of the RD&D-programme 92.

10.2 THE SITUATION TODAY

The existing Swedish system for the management of radioactive waste has been developed and built up systematically on the basis of proposals put forth in the mid-1970s and on the research and development work initiated with the KBS Project during the latter half of the 1970s. The parts that have not yet been finalized are an encapsulation plant for spent nuclear fuel etc. and a final repository for long-lived waste, particularly spent nuclear fuel.

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

Similar studies have been and are being carried out in most countries with significant nuclear power programmes. Owing to the stringent requirements introduced in the so-called Stipulation Act in 1977, the work in Sweden got under way with great determination and ample resources. This has given the Swedish activities an internationally recognized position and led to broad international cooperation.

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

After having examined safety, technical feasibility and other aspects for a number of different alternatives, work in Sweden has now reached a point where it should be concentrated to a main line.

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

To bring about the desired long-term isolation, a final repository for spent nuclear fuel is designed according to the multi-barrier principle. The spent fuel consists primarily of uranium dioxide, a ceramic material that has low solubility in groundwater. The most important long-lived radionuclides – which are formed in conjunction with irradiation in the reactor, e.g. plutonium – are embedded in the ceramic material and are likewise low-soluble in water. The fuel is enclosed in a canister with good mechanical strength and made of a material with a long corrosion life. The canisters are placed in specially arranged chambers in the rock and surrounded with a buffer material. The materials in the engineered barriers have documented long-term stability and the repository only affects the natural conditions in the rock slightly.

The safety assessments have shown that excellent conditions exist for designing the near field in the repository so that the radioactive materials are kept isolated for more than one million years. Moreover, the rock has a great capacity to sorb the radionuclides that dominate the radiotoxicity of the fuel and thereby constitutes an additional barrier.

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

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

10.3 GENERAL PLAN FOR FURTHER WORK

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

In the planning of the 1992 RD&D-Programme, SKB considered this possibility of building and commissioning the repository in stages. The result is that SKB finds that a demonstration phase has considerable advantages. The present programme thereby calls for completion of the research, development and demonstration work by first building the final repository as a deep repository for demonstration deposition of spent nuclear fuel. When the demonstration deposition has been completed, the results will be evaluated before a decision is made whether or not to expand the facility to accommodate all the waste. This plan also makes it possible to consider whether the deposited waste should be retrieved for alternative treatment. The latter option means that it must be possible to retrieve deposited fuel during the period the facility is being operated for demonstration purposes. The siting process is only affected to a limited extent by whether the planning applies to a deep repository for demonstration deposition or to a complete deep repository. The requirements on background information from SKB in the different phases (pre-investigation, detailed characterization, construction of repository) are essentially the same.

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

- the siting process with all its technical, administrative and political decisions,
- the process and the methods for step-by-step investigation and characterization of the deep repository site,

- system design and construction,
- full-scale encapsulation of spent nuclear fuel,
- the handling chain of spent nuclear fuel from CLAB to deposition in the repository,
- the operation of a deep repository,
- the licensing of handling, encapsulation and deep disposal, including the assessment of long-term safety,
- (retrievability of the waste packages).

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

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

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

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

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

Figure 10-1 shows a timeschedule for the facilities that are needed to dispose of the long-lived radioactive waste.

The following additional units will thus be needed for a demonstration deposition of spent nuclear fuel:

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

Figure 10-2 shows a timeschedule for the encapsulation plant and Figure 10-3 for the deep repository up to the completion of demonstration deposition.

In principle, the interim storage period of 40 years can be retained for further planning even with the timeschedule for demonstration assumed here. SKB believes that the demonstration can be completed within about 20 years. Thus, as is evident from Figure 10-2, it is possible to follow this up with final disposal of the remaining fuel and waste immediately after 2020 if the decision to do so is made in about 20 years.

10.4 SITING

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

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

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

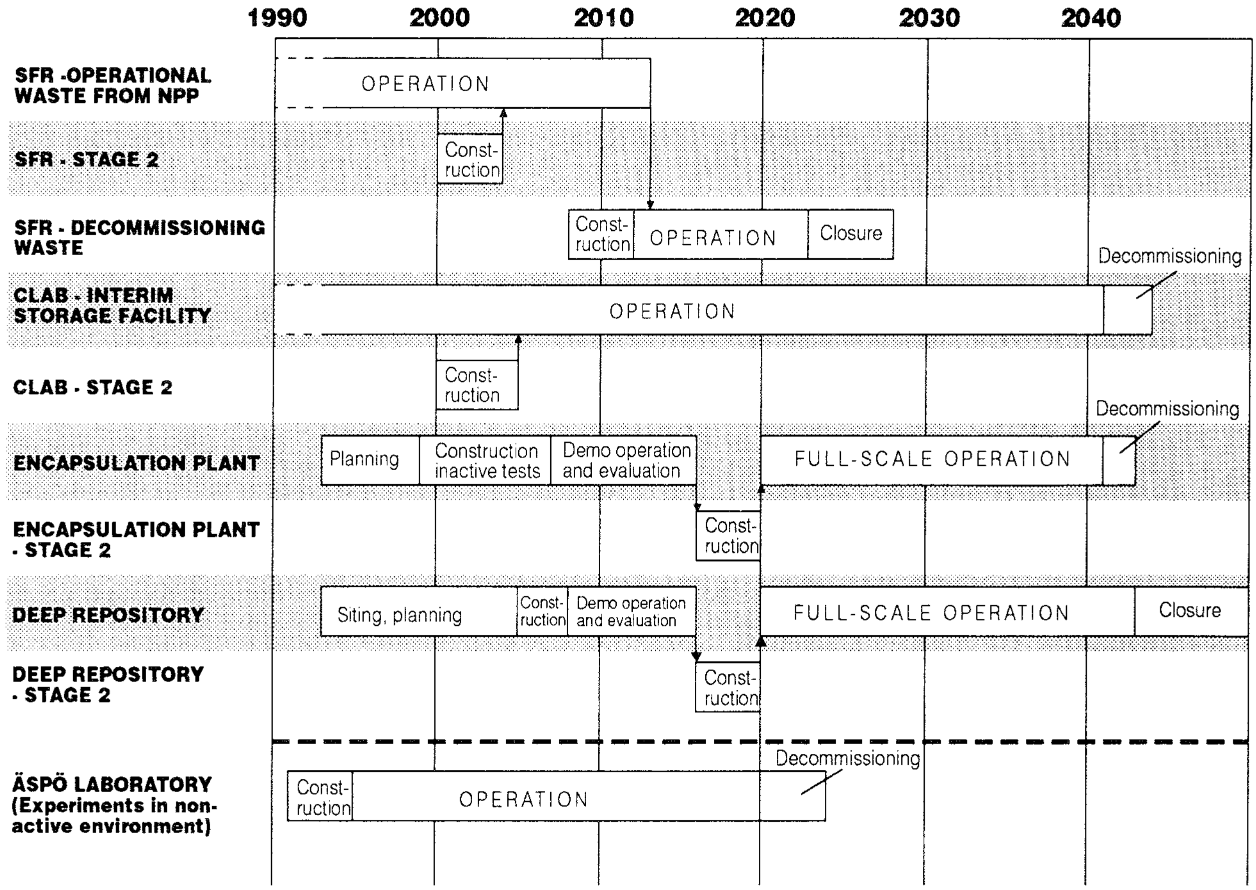


Figure 10-1. Approximate timeschedule – facilities for management of the waste products of nuclear power.

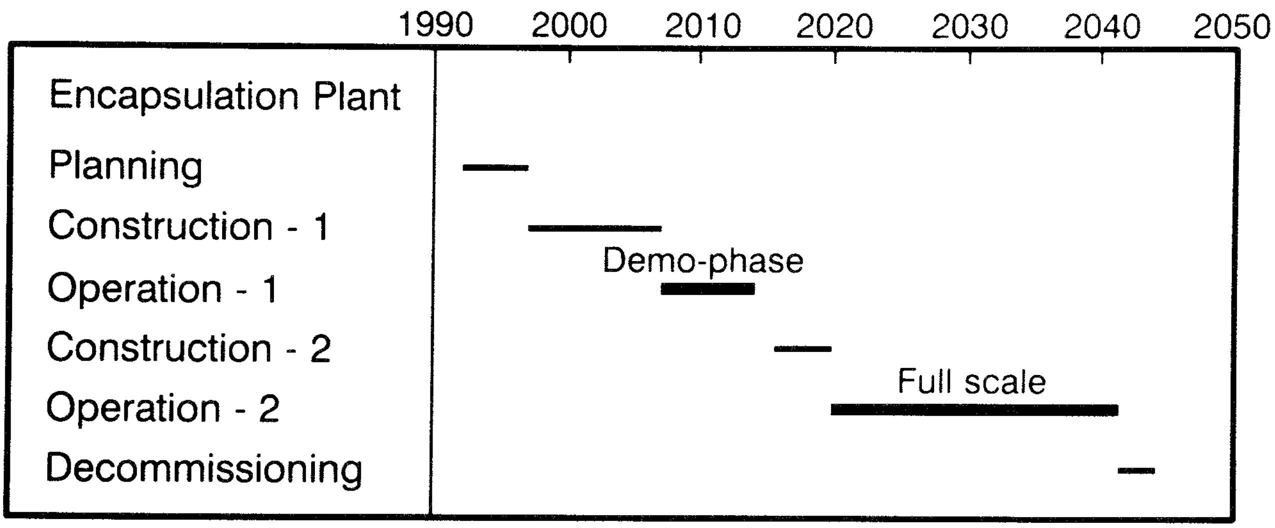


Figure 10-2. Preliminary timeschedule for the encapsulation plant.

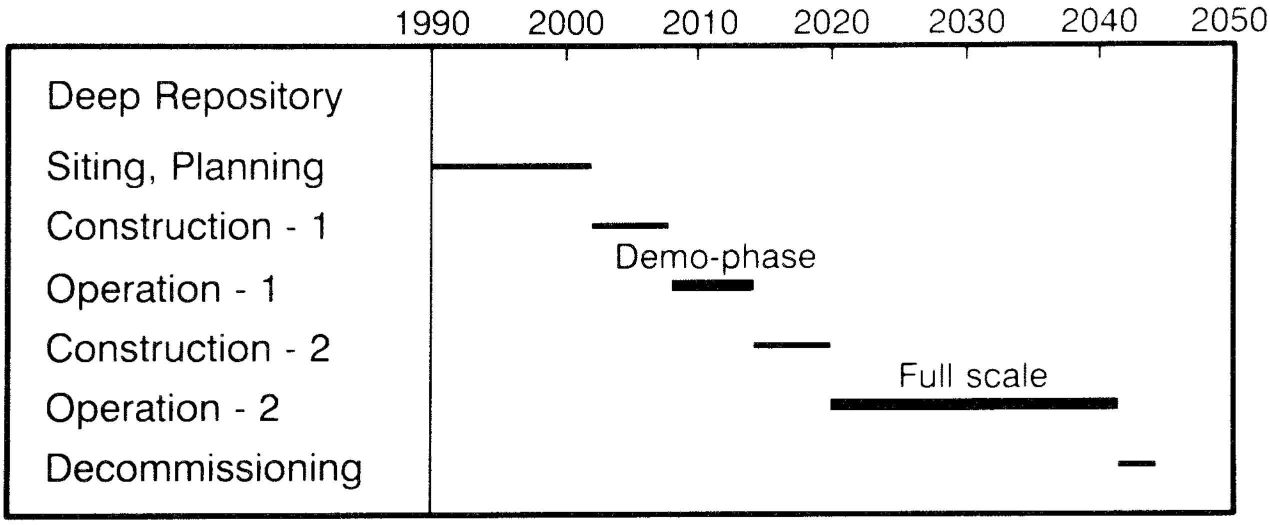


Figure 10-3. Preliminary timeschedule for the deep repository.

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

Based on these guidelines, the work of siting and construction of the deep repository is planned to proceed in the following stages, see Figure 10-4:

Stage 1:

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

Stage 2:

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

Stage 3:

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

Stage 4:

Commissioning. Demonstration deposition.

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

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

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

Of the canister alternatives studied, the composite canister holding 12 BWR assemblies is chosen as the main alternative for the continued work. This canister consists of a steel container, which provides mechanical protection, surrounded by a copper container, which provides long-lasting corrosion protection. Since the canister is a vital barrier, some additional development should be conducted for the alternative of a lead-filled copper canister as a reserve alternative to the composite canister.

Of the studied repository designs, the KBS-3 design is kept as the main alternative for further work. In connection

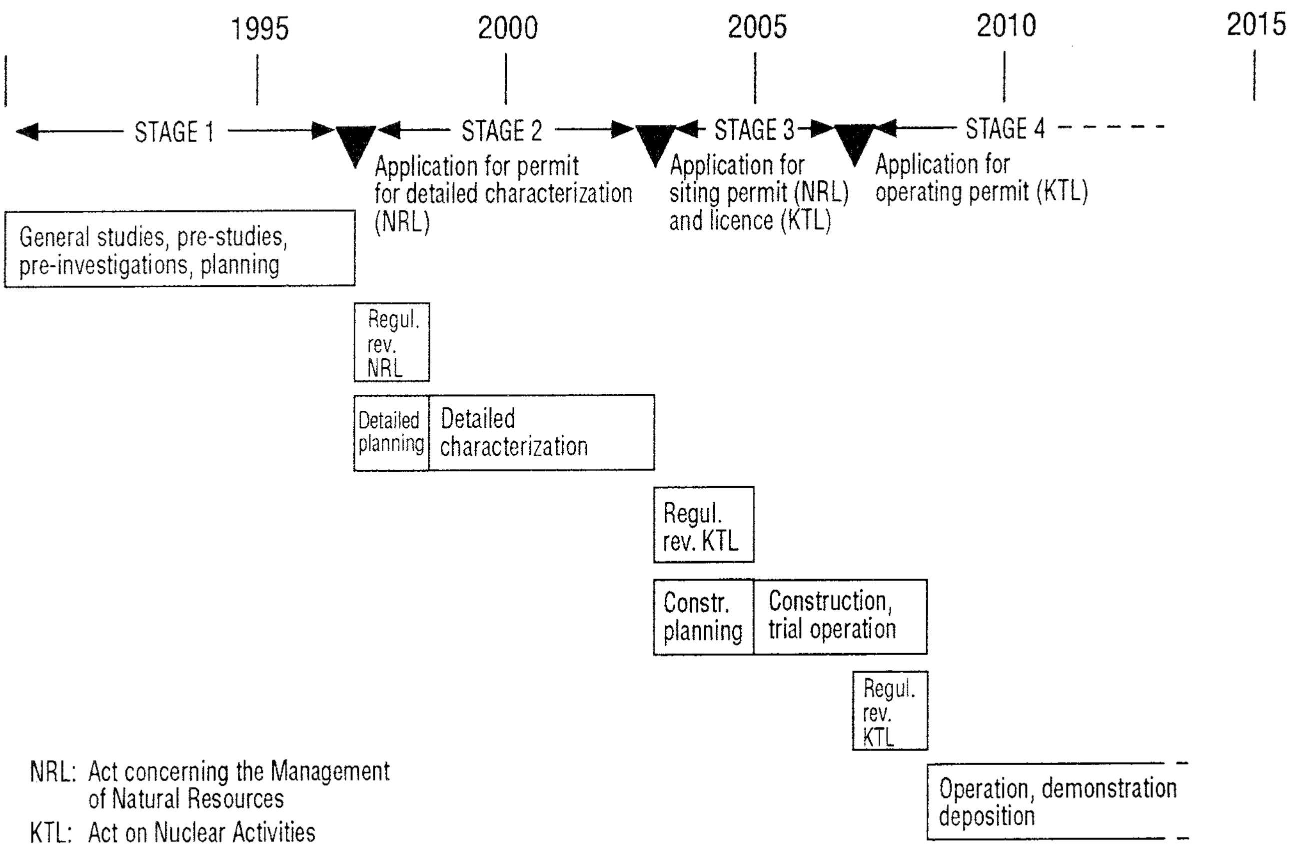


Figure 10-4. Example of timeschedule for the deep repository up to the completion of demonstration deposition. The timeschedule gives the earliest possible completion dates.

with adaptation to local conditions on the selected site, this design can be further optimized, whereby technically closely-related variants of the design can be given further consideration.

10.6 OUTLINE OF THE RD&D-PROGRAMME

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

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

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

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

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

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

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

11 ALTERNATIVE SYSTEMS STUDY (PASS)

11.1 BACKGROUND

The feasibility of direct disposal of spent nuclear fuel was demonstrated in Sweden in 1984 when the Government, after extensive domestic and international review of the KBS-3 system, declared that it "has found that the method in its entirety can be approved with regard to safety and radiation protection." This system has also been accepted by the Government as the reference system for the annual cost calculations, which aim at estimating the back-end fee on nuclear power production /11-1/. The design is today basically maintained the same as in 1984, although minor adjustments are annually made in accordance with progresses in technical developments.

In parallel with the ongoing development of the KBS-3 system also alternative system designs have been evaluated and compared with the reference system.

During 1986 to 1989 the WP-Cave system was developed and compared with the KBS-3 design /11-2/. That system was at that time the most carefully studied system besides KBS-3. The result of the comparison was that KBS-3 was considered to provide major advantages. This result was reported in the SKB Annual Report of 1988 /11-3/. The WP-Cave has not been further studied.

Three other systems have also been defined, evaluated and compared. These are called Deep Boreholes (Very Deep Holes, VDH), Long Tunnels (Very Long Holes, VLH), and Medium-Long Tunnels (Medium Long Holes, MLH). The canister diameter is different for each of these systems and for each diameter different designs have been developed.

The technical and economical evaluation and comparison of these three systems with the KBS-3 system as well as the evaluation and comparison of each system's canister alternatives have been conducted in one project named "Project on Alternative Systems Study, PASS" /11-4/. The repository systems in the study are shown in Figure 11-1.

11.2 PASS ALTERNATIVES

11.2.1 Systems

11.2.1.1 KBS-3 (abbreviation for Nuclear Fuel Safety)

The assumed layout for the KBS-3 design is illustrated in Figure 11-2, which is based on a ramp from the surface down to the repository level at about 500 m depth. The repository itself is here assumed to be divided into only two blocks.

The repository consists of a number of parallel deposition tunnels, which are interconnected by a central tunnel for transports and communication. In reality the layout of each tunnel is flexible and can be adapted to the local conditions of the rock. Also the layout of the whole repository is flexible and can be separated in several parts dependent on the existence of major discontinuities, which should be avoided, and the size of suitable blocks of host rock.

The canisters, with an outer diameter of 0.8 – 0.88 m, are deposited in a bentonite buffer in vertical holes below the deposition tunnels at a suitable distance mainly decided by the thermal load in each canister and the thermal properties of the bedrock and of the bentonite buffer. In ordinary granite the center distance is on the order of 6 m. In PASS the assumed load per canister results in a requirement of 3745 canisters for the whole Swedish program of spent nuclear fuel /11-4/.

11.2.1.2 VDH – Deep Boreholes

The principle of the Deep Borehole alternative is illustrated in Figure 11-3. The canisters are stacked on top of each other in lined boreholes between 4 000 and 2 000 m depth. The outside diameter of the canister is limited by the hole diameter that is possible to bore down to the repository depth. A casing is needed in order to prevent the rock walls from caving in. This results in a canister diameter of 0.5 m in a 0.8 m diameter borehole.

Bentonite clay is used as buffer around the canisters and to fill up caved-in volumes. The canisters are also separated by a bentonite plug. Some 11 235 canisters of this size are equivalent to the 3745 of KBS-3 type. A total of 38 deep holes are estimated to be required. In case of rod consolidation the number of canisters decreases by a factor of 2, i.e. to 5 617, and consequently the number of deep holes to 19.

11.2.1.3 VLH – Long Tunnels

The name Long Tunnels emanates from the prime objective with the layout which is to have the repository situated far away from the industrial area on the surface. The studied system has the layout illustrated in Figure 11-4.

The repository is located at about 500 m depth. The canisters are placed in the horizontal deposition tunnels in a row and surrounded by bentonite. The diameter of the canisters is chosen as large as possible with consideration to maximum fuel load per canister but also possible weight for handling during deposition. The choice has been a diameter of 1.6 m containing twice the load of the KBS-3 canister size. Thus 1 873 canisters of VLH type are required for the Swedish program with the assumptions used in PASS.

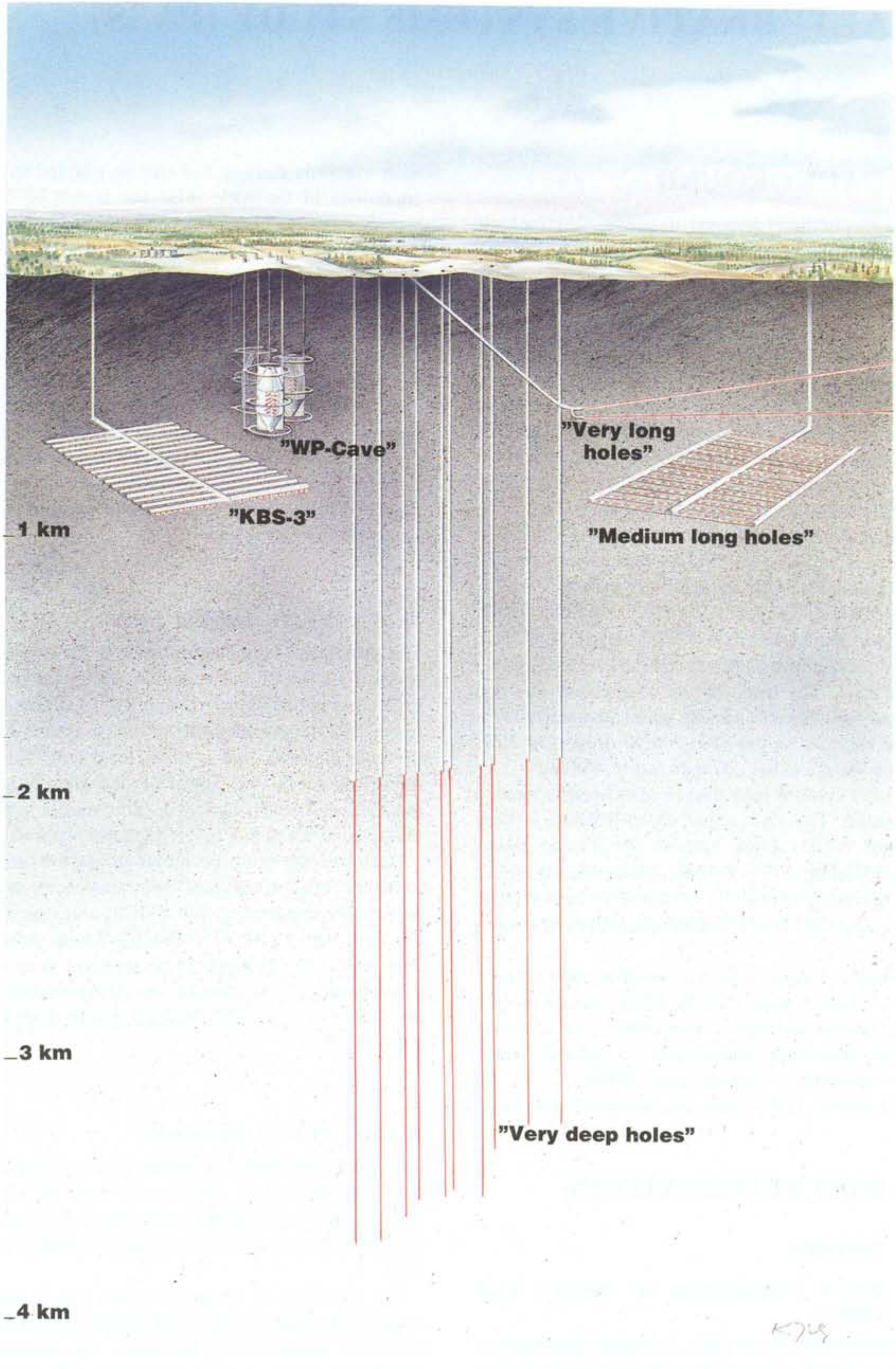


Figure 11-1. *Alternative designs of deep repository.*

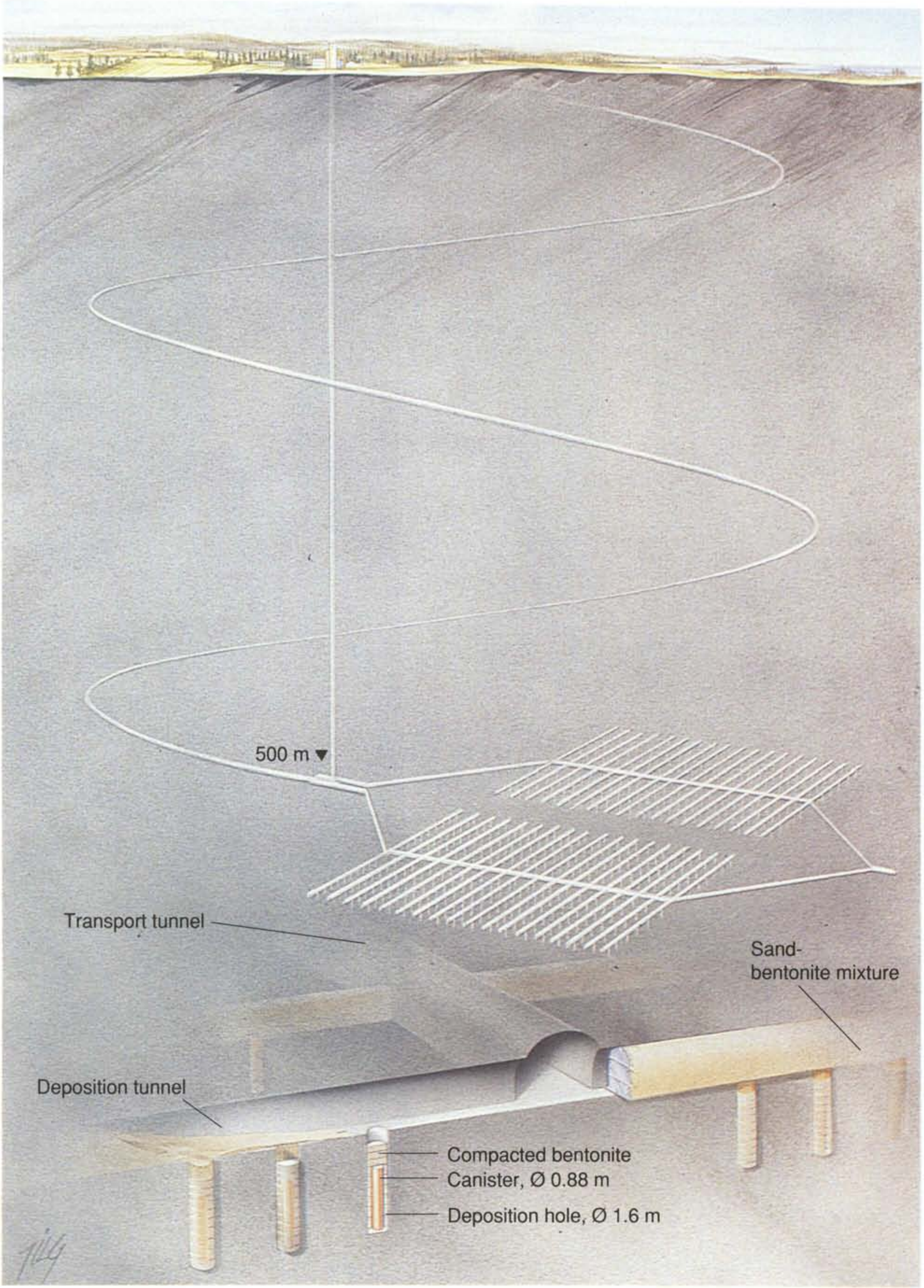


Figure 11-2. *KBS-3 design of a deep repository.*

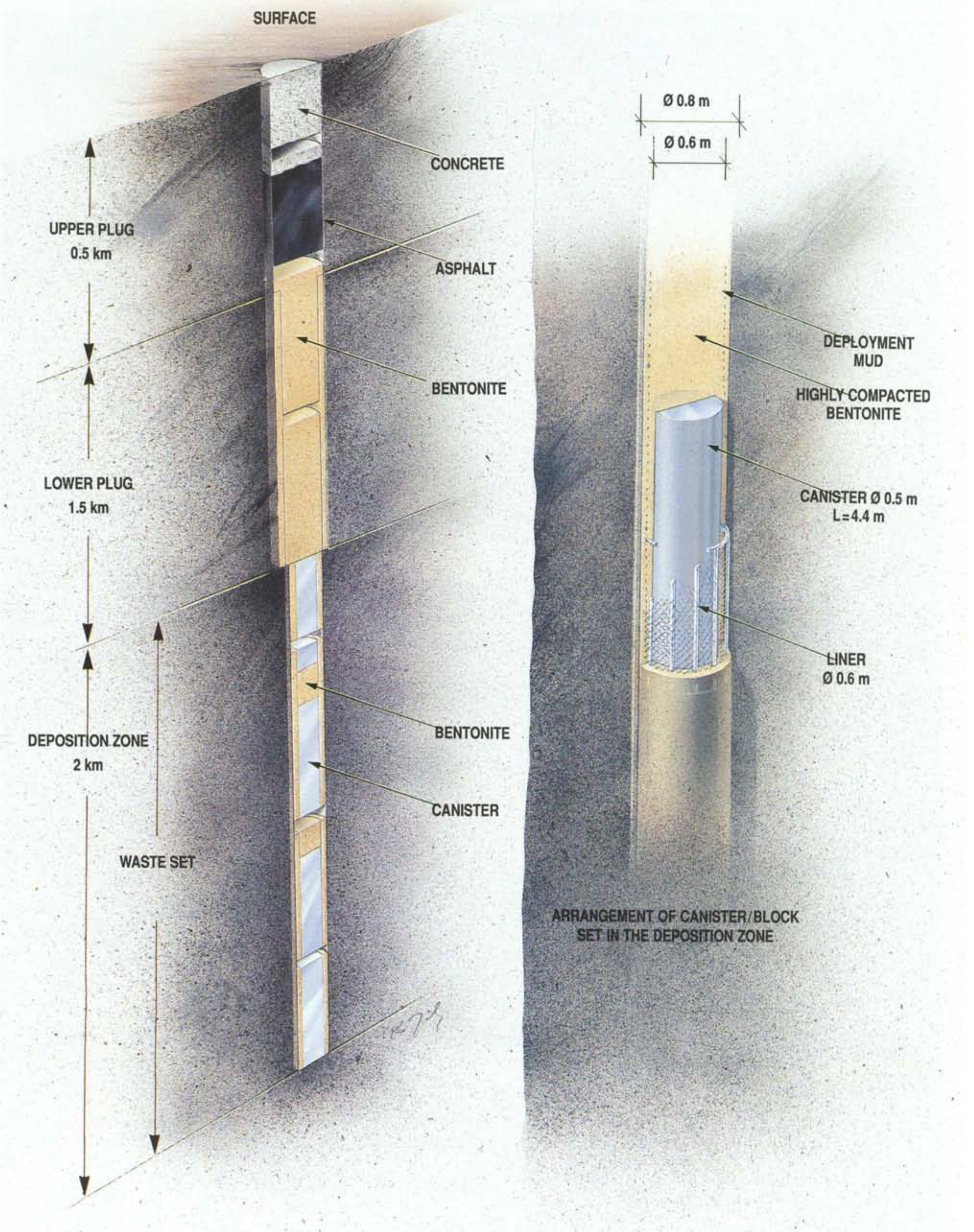


Figure 11-3. *VDH design of a deep repository.*

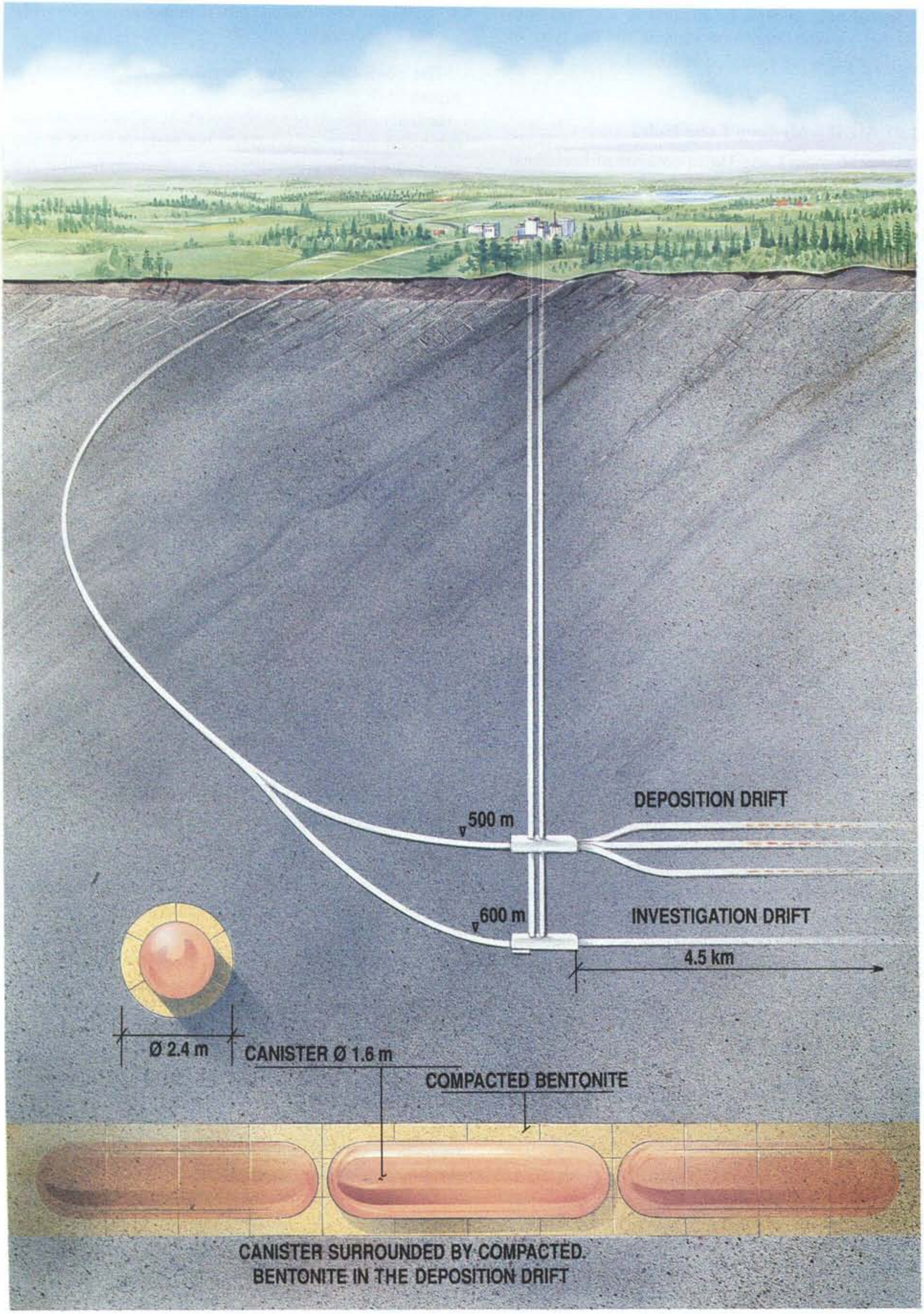


Figure 11-4. *VLH design of a deep repository.*

The layout may as well be adapted to a design with parallel but shorter deposition tunnels for the type of blocks of rock that are specified in KBS-3.

11.2.1.4 MLH – Medium Long Holes

The system Medium Long Holes consists of horizontal parallel holes with KBS-3 size canisters placed in rows and surrounded by bentonite, see the illustration in Figure 11-5. The repository layout conforms to the KBS-3 as well as major prerequisites for the flexibility in location for instance with respect to discontinuities in the bedrock, volumes of host rock blocks and repository depth. The additional side tunnels in MLH are required to provide access for mounting the reamer head on the drill string when boring the deposition tunnels. Blind boring of these tunnels is not considered as the base case in the study. The deposition process is further assumed to be operated from these side tunnels.

11.2.2 Canister alternatives

11.2.2.1 KBS-3

The considered alternatives are illustrated in Figure 11-6:

* Steel canister with copper shell (composite canister). TVO's ACP (Advanced Cold Process) canister is based on the same principle but has a somewhat smaller outer diameter in its basic version: 0.822 or 0.802 m.
  The internal free volume can be filled out with particulate material in the cold state.
* Copper canister filled with cast lead. The lead filling is made slightly over the solidification temperature of pure lead, which is 327°C, whereafter the whole canister is allowed to cool in a controlled manner. This design has been the reference design for the annual cost calculations of the back-end costs, now recently in PLAN 92 /11-1/.
* Copper canister fabricated by Hot Isostatic Pressing, HIP.
* Steel canister filled with cast lead. The lead filling process is similar to the one for the copper/lead canister.
* Steel canister. The inner free volume is filled with a particulate material in the cold state.

11.2.2.2 VLH

* Steel canister with copper shell (composite canister) with hemispherical ends. The inner free volume can be filled with a particulate material in the cold state.
* Composite canister but with flat ends.
* Steel canister with hemispherical ends. (Composite canister without copper shell).
* Steel canister with flat ends. (Composite canister without copper shell).

The considered copper alternatives are illustrated in Figure 11-7.

Initially a design of a selfsupported copper canister was discussed and proposed /11-5/ but early in PASS it was concluded that such a design is not technically feasible due to material creep.

11.2.2.3 VDH

The considered alternatives are:

* Titanium canister with concrete fill. The fuel assembly is either conditioned intact (non-consolidated) or after dismantling (rod consolidation).
* Copper canister fabricated by HIP.

The titanium/concrete canister is illustrated in Figure 11-8.

Initially ta selfsupported titanium canister was also proposed /11-6/ but based on information from AECL, Canada, /11-7/ it was found that, at the hydrostatic pressure at VDH depths, the canister wall had to be uneconomically thick in order to avoid the risk for buckling due to material creep.

11.2.2.4 MLH

The alternatives are the same as described for KBS-3.

11.3 COMPARISON METHODOLOGY

With the vast number of differences that had to be handled in the study a systematic approach was deemed necessary. The choice was made to organize the questions involved in accordance with a hierarchical structuring. The principle of this, see Figure 11-9, is that the "Goal" to achieve is at the top and below is a level of elements that each is a part of the goal, and all elements together make the goal. Each of these elements can in turn be divided into one or more lower levels. At the base of the structure are the alternatives among which a choice is made to achieve the "Goal" in the best way.

In PASS two hierarchical schemes were applied: one for ranking of canister alternatives and one for ranking of repository systems.

The main goals consequently were:

- Canister for deep disposal of the spent fuel;
- System for deep disposal of the spent nuclear fuel.

The first level below each of these two goals consists of three headings which are equal for both. The three interim goals are:

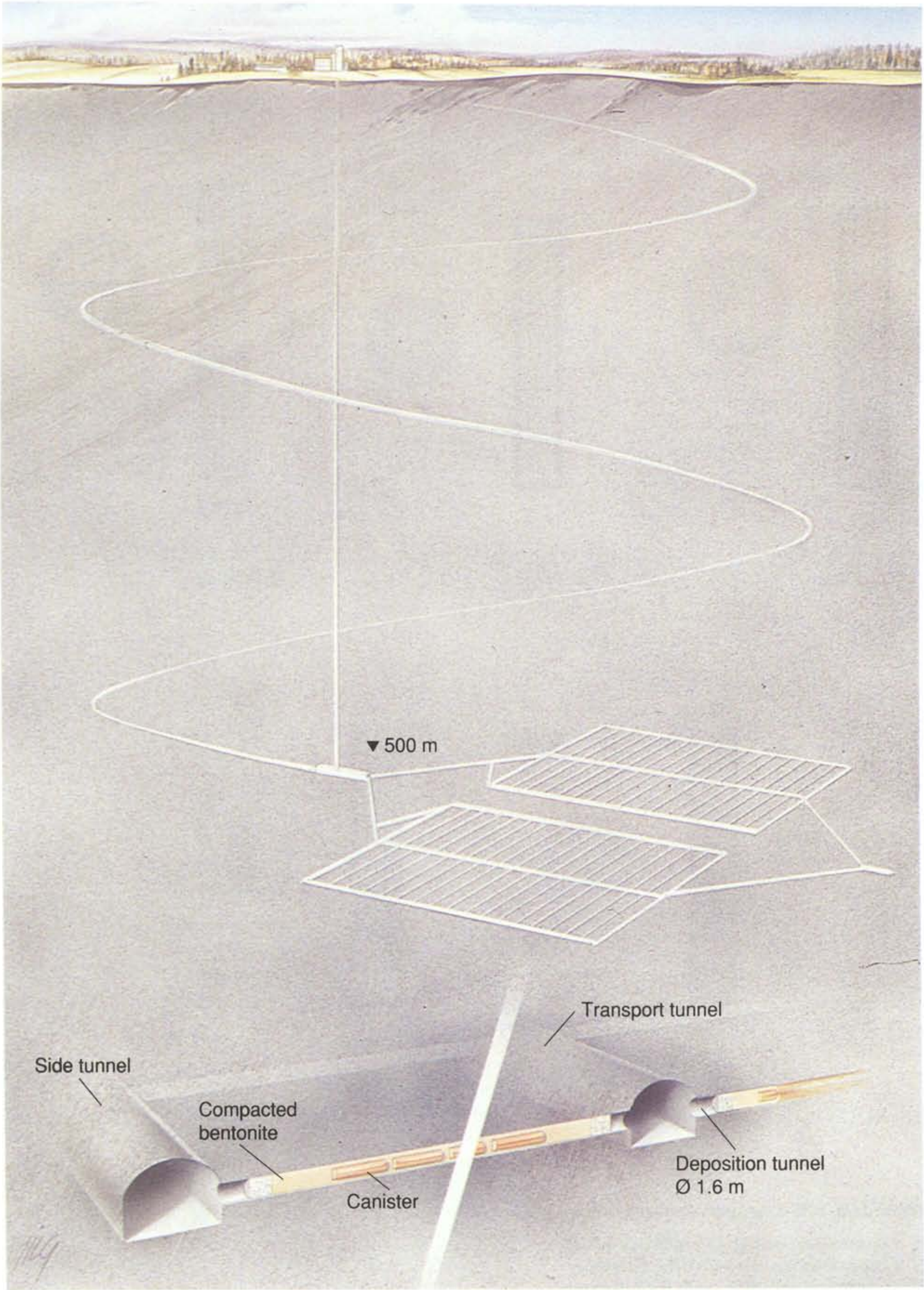
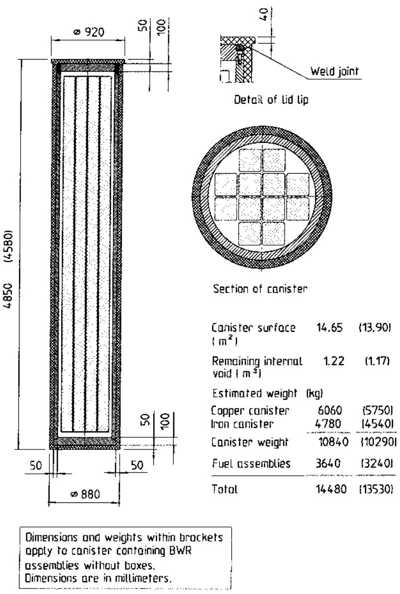


Figure 11-5. *MLH design of a deep repository.*



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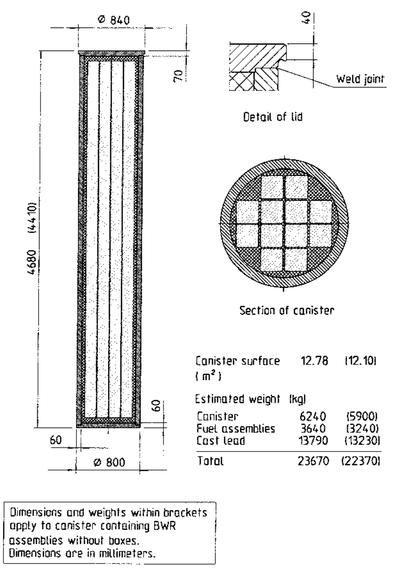

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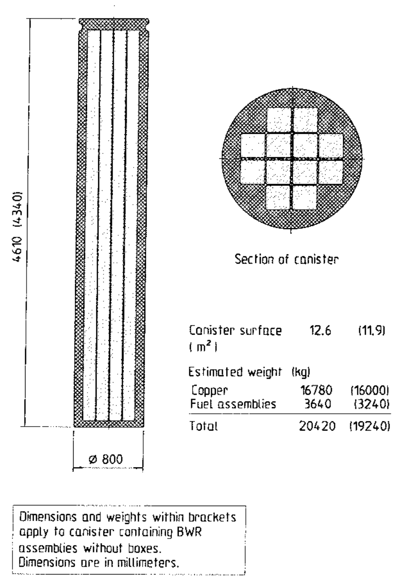

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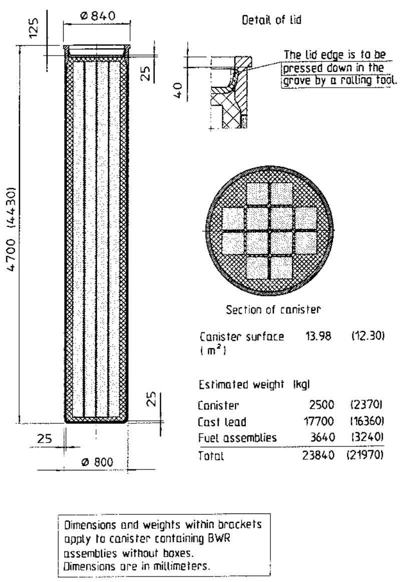

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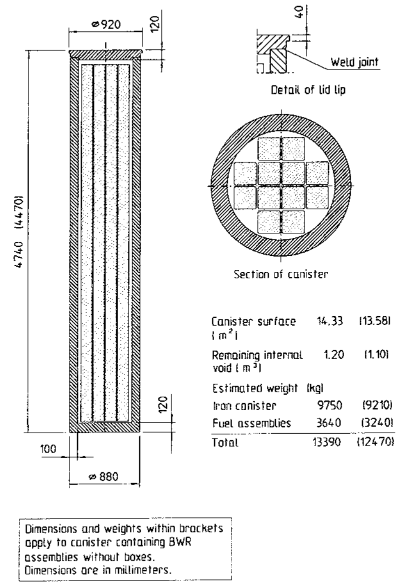


5)

Figure 11-6. *KBS-3 canister alternatives:*

1) *Copper/steel canister with BWR assemblies.*
2) *Copper/lead canister with BWR assemblies.*
3) *Copper canister - HIP - with BWR assemblies.*
4) *Steel/lead canister with BWR assemblies (Gripsholm).*
5) *Steel canister with BWR assemblies.*

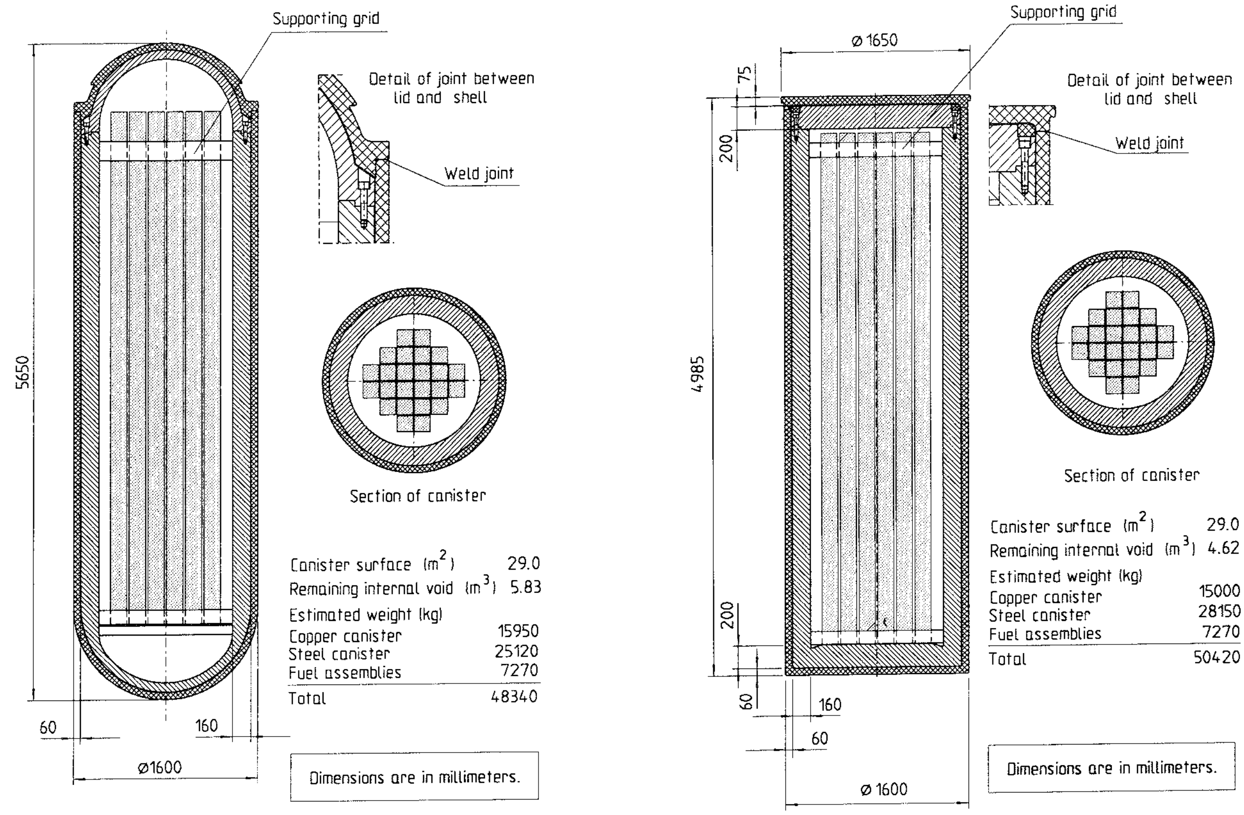


Figure 11-7. VLH canister alternatives. Copper/steel canister with hemispherical ends, BWR assemblies (to the left). Copper/steel canister with flat ends, BWR assemblies (to the right).

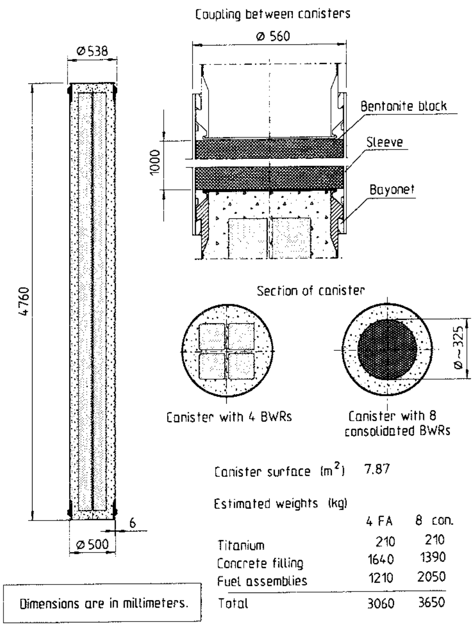


Figure 11-8. VDH canister. Titanium canister with concrete filling.

* Technology". The scope comprises methods and processes for obtaining encapsulation (canister comparison) and deep disposal (repository system comparison) respectively with the quality required for achieving the necessary long-term performance.

* "Long-term performance and safety". The scope comprises stipulated requirements and criteria as well as the sensitivity of the performance of the different barriers to existing uncertainties and to various events in the geological environment in the repository after sealing.

* "Costs". The scope comprises all aspects and factors that distinguish the canister alternatives and the repository systems respectively, so that cheaper or more expensive methods, equipment or materials can be chosen.

The levels below these are specific for the canister comparison and repository system comparison respectively.

The practical approach adopted was to start with the comparison and ranking of the different canister alternatives for each repository system. Based on the result from this analysis one prime canister alternative was chosen for each of the systems, whereafter the systems were compared. As a result of this approach the ranking of repository systems came to encompass only four alternatives.

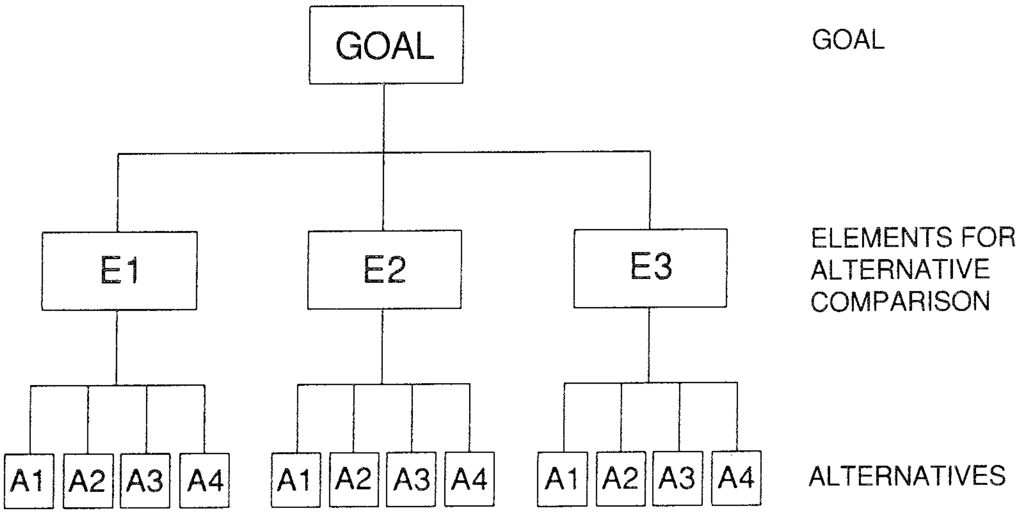


Figure 11-9. *Principle of hierarchical structuring.*

11.4 CANISTER COMPARISON

11.4.1 Comparison regarding long-term performance and safety

11.4.1.1 Corrosion

The differences due to corrosion are basically independent on the repository design. The repository depth is, however, of importance as the temperature increases with depth, and maybe also the salt content in the ground water.

The result is that copper is considered to provide the longest lifetime at repository depth of about 500 m. No difference is identified between the copper canister alternatives.

In the very deep repository, based on the limited information available about conditions at great depths, the copper alternative is as well considered to provide the longest lifetime. Titanium has been investigated by AECL of Canada, who has predicted corrosion lifetimes of 1 200 to 7 000 years /11-8/, which is the basis for estimating a corrosion lifetime on the order of 1 000 years for the VDH canister design.

11.4.1.2 Mechanical integrity

During the last two years questions related to the mechanical integrity of the canister have been analyzed in detail by a group of Swedish experts /11-9/. The focus has been on KBS-3 size canisters in the study and the major questions have been the effect on the canister due to rock displacements across the disposal hole, residual stresses in copper canisters after sealing, and creep deformations and creep relaxations of stresses in the copper shell.

The result was that the copper/steel (composite) canister was judged to be the most favourable of the copper alternatives. The reason is the good mechanical strength of the steel body.

11.4.1.3 Barrier against radionuclide migration at defect canister

For normal conditions in the repository the radionuclides will be confined in the copper and lead-filled steel canister alternatives for a very long time. The difference between these alternatives is interesting only in case of a defect canister. The plain steel canister has a shorter lifetime.

When assuming a defect in the copper the lead-filled copper alternative is considered to provide the best barrier due to the good corrosion resistance of the lead. In case the water anyway penetrates the lead only parts of the canister's inventory may be accessible for leaching at one time. The same effect may be expected from the HIP canister, although this canister only has one metal as protection against corrosion.

The composite alternatives have the same potential life time, but provide less effective barrier against migration of radionuclides, if the canister is penetrated.

The lead-filled steel alternative features the good corrosion resistance of the lead filling, which is of the same importance as for the lead-filled copper alternative in case of a defect canister. The short life of the steel is, however, a disadvantage for this alternative. The available data today on the performance of the lead-filled alternative are considered to be too limited for ranking this canister design against the others.

The plain steel canister has a shorter life which consequently means that the safety margin for this canister is less than for the other analyzed alternatives.

The conditions during disposal according to the VDH system are such that several canisters of any of the considered designs may be defect after disposal. Both concrete filled titanium and HIP designs are therefore judged to provide equal barrier potential.

11.4.1.4 Conclusion

KBS-3 and MLH

All copper alternatives as well as the lead-filled steel design provide high safety potential with respect to corrosion and barrier-performance in case of defect canister. When considering the low probability for initially defect canisters in any of the designs, the copper canisters are all ranked in top, see Table 11-2. The selfsupported steel canister is placed lower. The lead-filled design has not been ranked in the PASS context.

VLH

The copper alternatives are preferred in front of the plain steel alternatives.

VDH

The alternatives studied are considered equal.

11.4.2 Comparison regarding technology

11.4.2.1 KBS-3/MLH

Steel is the most common structure metal of the studied ones and production and sealing of the steel canister alternative is the most well known and tested operations. Thus is the steel canister the most advantageous from the technological point of view.

For the copper alternatives the production and sealing of the copper shell/canister are judged to be similar in all alternatives. The conditioning in a "cold" state is, however, considered advantageous over conditioning in a "hot" stage (cast lead filling).

The HIP alternative is special from a conditioning point of view due to the combination of a high temperature and a high pressure. Specific requirements are due for the copper powder to use. In case of failure in the process a mending operation may be difficult to accomplish.

The lead filling of the lead/steel canister is suffering from the same disadvantages as the lead filling of the copper canister.

11.4.2.2 VLH

By the same reasons as presented for the KBS-3/MLH alternatives is the plain steel canister highest ranked, in front of the copper/steel canister designs, in questions concerning technology.

11.4.2.3 VDH

The concrete-filled titanium canister with non-consolidated fuel assemblies is the most simple alternative from the technological point of view. Consolidation is more complex and requires substantial additional facilities in the encapsulation plant.

The HIP alternative has the same concerns as the somewhat larger KBS-3/MLH canister.

11.4.3 Comparison regarding costs

11.4.3.1 General

The costs for encapsulation have been calculated in the same way as in the PLAN 92 study /11-1/. For the comparison the relative difference is of importance and these values for the different canister sizes are presented in Table 11-1. The comparison factor is based on the present value, which is the discounted value as per January 1992 for a real rate of interest of 2.5%. All alternatives have the same construction and operation times.

Table 11-1. Cost comparison of studied canister alternatives.

Canister alternatives	Comparison factor
KBS-3/MLH	
Copper/steel	1.24
Copper/lead	1.27
HIP	1.14
Steel	1.02
Steel/lead	1.00
VLH	
Copper/steel	1.15
Steel	1.00
VDH	
Titanium/concrete non-cons.	1.00
Titanium/concrete cons.	1.06
HIP, non-cons.	1.05

11.4.3.2 KBS-3/MLH

Steel and steel/lead designs are according to the comparison factors in Table 11-1 ranked highest. HIP is placed third, and copper/steel and copper/lead are placed fourth.

11.4.3.3 VLH

The plain steel design is estimated to be about 15% more expensive than the copper/steel design, which is less than between the design types of KBS-3 size.

11.4.3.4 VDH

The total costs are approximately the same for all alternatives. In fact the cost for encapsulation of non-consolidated assemblies is somewhat lower than for the alternative with rod consolidation, although twice as many canisters are required.

11.4.4 Ranking of canister alternatives

11.4.4.1 KBS-3/MLH

The results of the interim comparison with respect to "Long-term performance and safety", "Technology" and "Costs" are summarized in Table 11-2. A weighing-together of the three separate comparisons has resulted in a ranking of the copper canisters ahead of the pure steel alternative. The main draw-back for the steel canister is the short lifetime.

Table 11-2. Ranking of KBS-3/MLH canister alternatives.

Canister alternatives	Long-term performance and safety	Technology	Costs
Copper/steel	1	2	4
Copper/lead	1	4	4
HIP	1	5	3
Steel	4	1	1
Steel/lead	-*	2	1

* cannot be ranked due to incomplete data

Among the copper alternatives is the copper/steel canister ranked ahead of the lead-filled canister. The major factors being the low temperature during conditioning and the higher mechanical strength of the copper/steel canister. The HIP canister is ranked after these due to fabrication-related reasons.

Thus the copper/steel canister is ranked highest followed by the copper/lead canister. All other alternatives come after these, but no effort has been made to rank them in relation to each other.

11.4.4.2 VLH

The copper/steel canister is ranked highest due to corrosion reasons.

The difference between flat and hemispherical ends has not been analysed in detail. Both designs are considered equal in PASS for canisters which besides this difference have the same design.

11.4.4.3 VDH

The concrete-filled titanium canister is ranked highest, ahead of the HIP canister. Because the question of non-consolidated or consolidated assemblies has a major impact on the size of the repository area, these alternatives are not separated in the canister ranking phase.

11.5 REPOSITORY SYSTEM COMPARISON

11.5.1 Compared systems

For each of the studied repository systems a reference design was defined. These have been adjusted in details in PASS in such a way that only "true" differences between the systems should remain for the comparison. It, however, meant that the designs were not optimized in all aspects although all important features of each system were carefully incorporated. The compared systems are:

* KBS-3 with copper/steel canister, see Figure 11-10.
* MLH with the same canister as for KBS-3, see Figure 11-11.
* VLH with copper/steel canister, see Figure 11-12.
* VDH with concrete-filled titanium canister, see Figure 11-13.

11.5.2 Comparison of ranking with regard to "Technology"

The selected way of structuring the questions resulted in a sub-level (second level) consisting of

* Technical feasibility
* Construction
* Deposition and sealing
* Human intrusion

For each of these a number of second as well as a third level elements were defined /11-10/.

The comparison and evaluation of the complex set of questions, which comprised pros as well as cons for each system, were made according to the Pairwise Comparison principle. The exercise was made individually by six technical experts with connections to the work conducted by SKB and TVO of Finland. The principle is to compare two elements at the same hierarchic level for each combination of two elements. The comparison is made from the top of the hierarchy and down to the bottom level. Each comparison results in a numerical evaluation of the importance of one of the elements over the other with respect to the next higher level. The numerical scale in use ranges

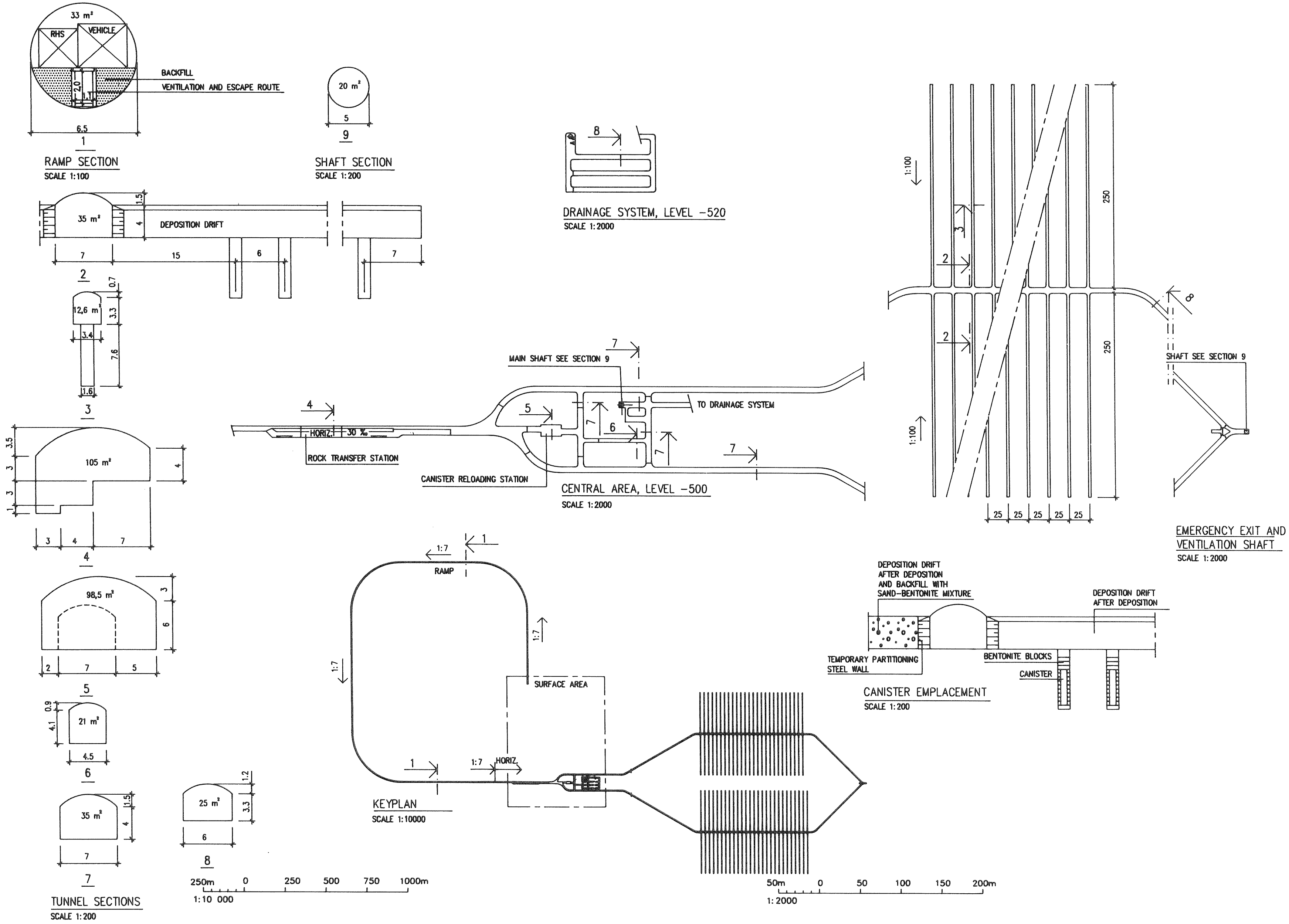


Figure 11-10. Basis for comparison – KBS-3.

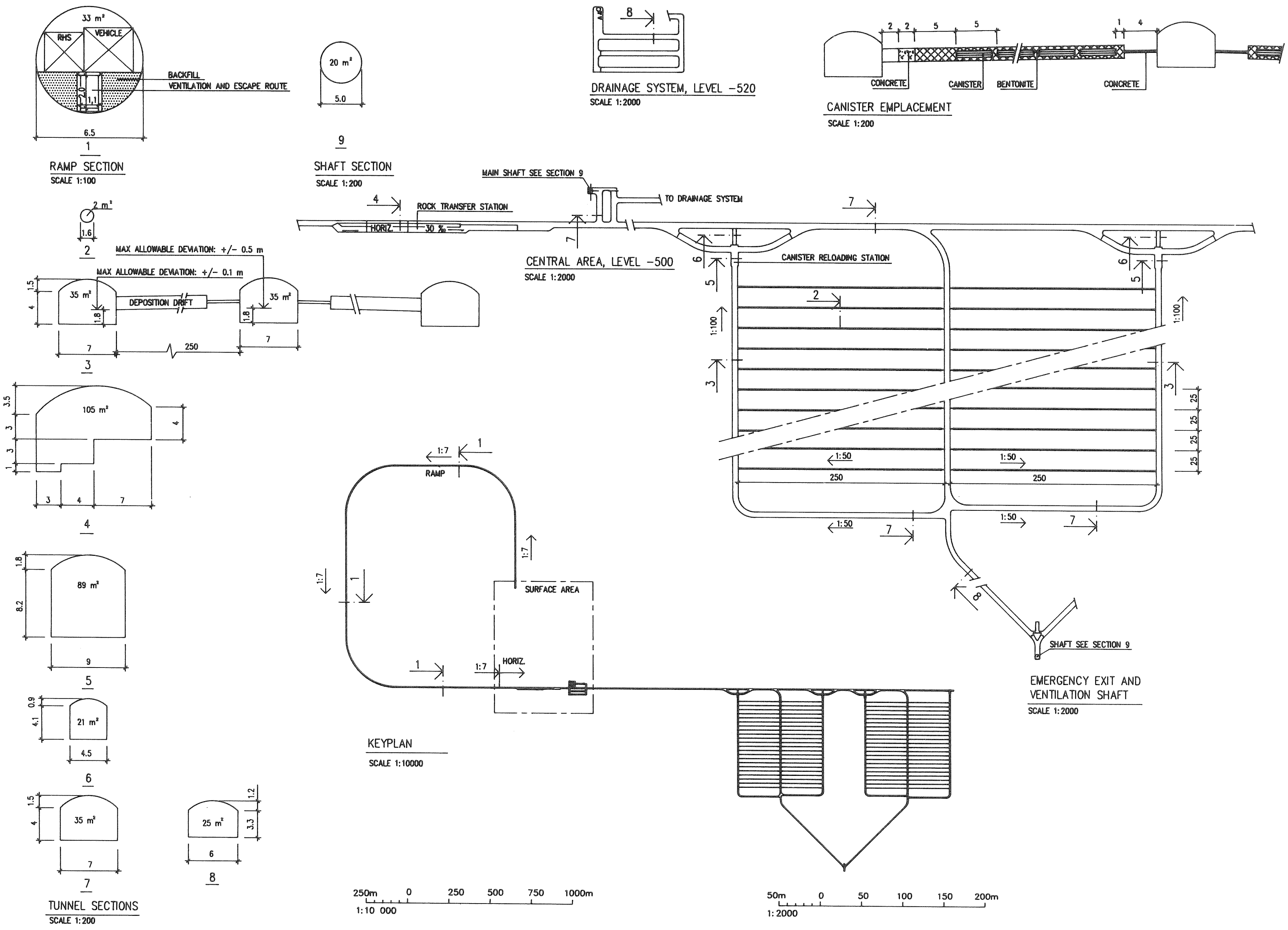


Figure 11-11. Basis for comparison – MLH.

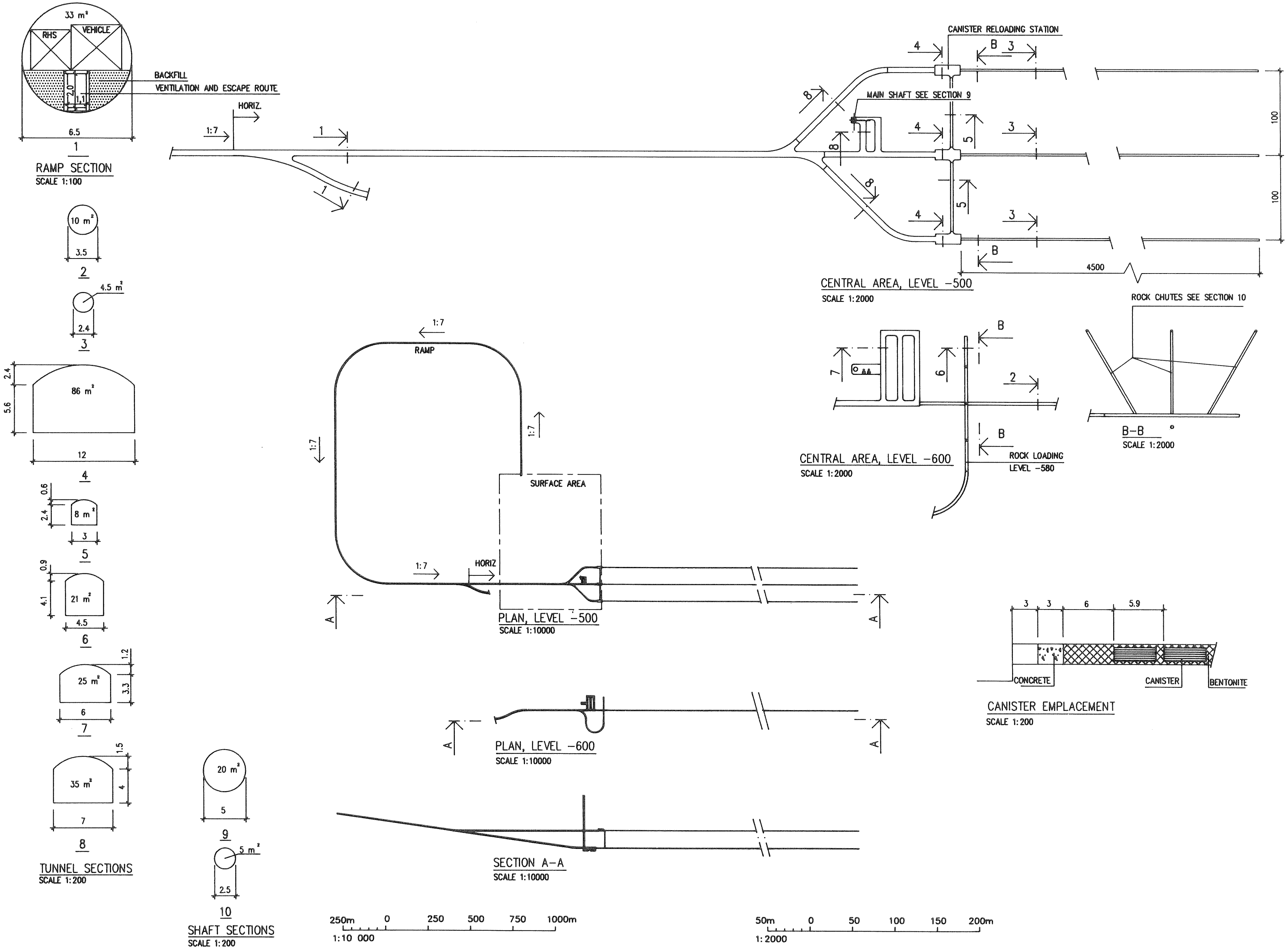


Figure 11-12. Basis for comparison – VLH.

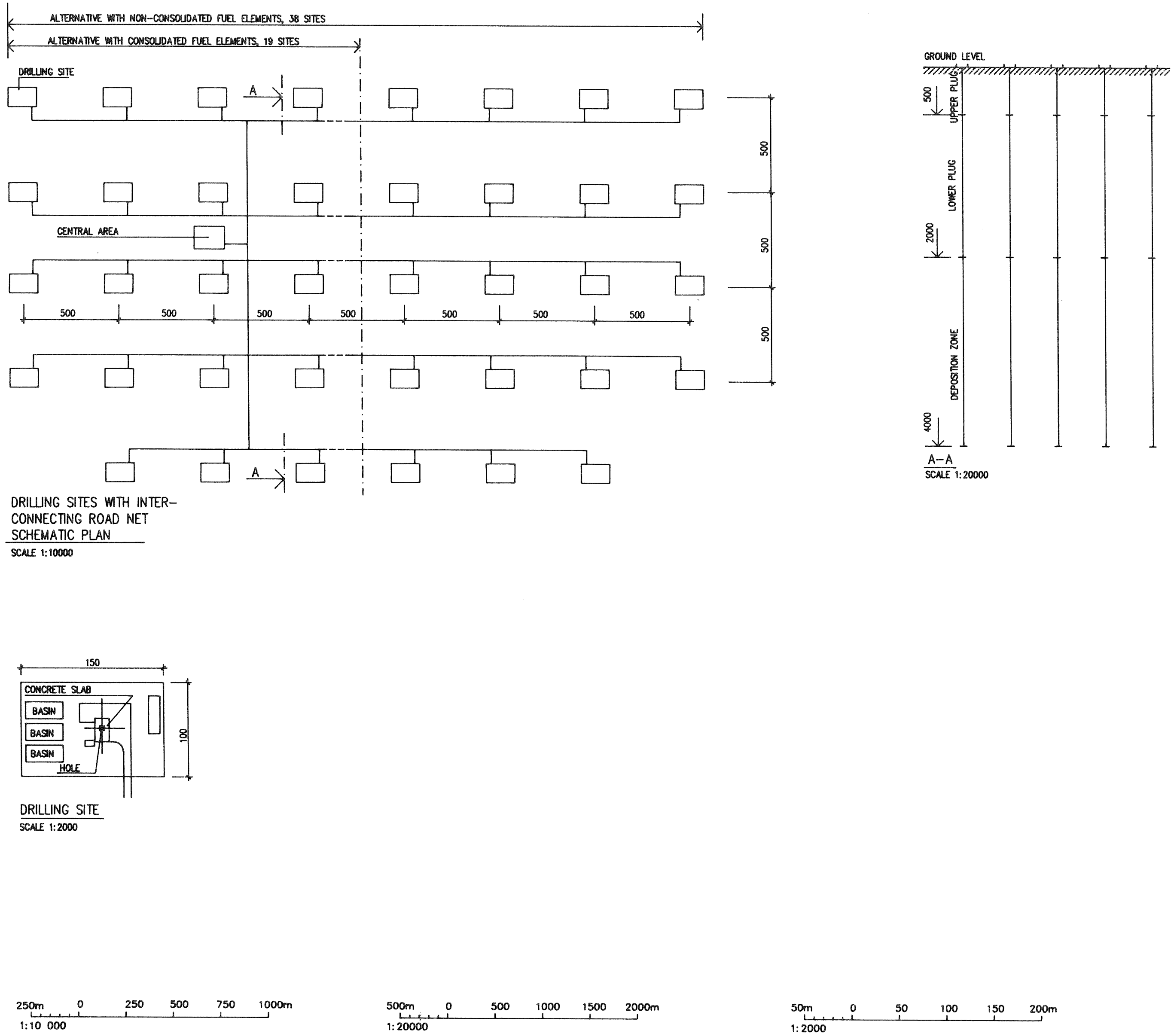


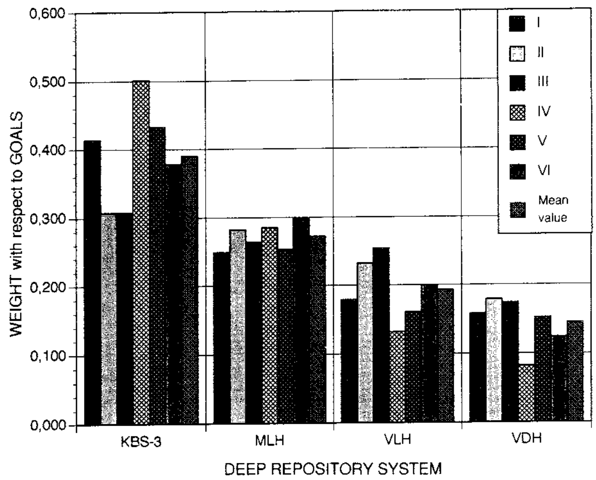
Figure 11-13. Basis for comparison – VDH.

from 1 to 10 and the evaluation is expressed in terms of so and so many times more important. Finally are the systems compared and evaluated with respect to each question on the bottom level (in accordance with Figure 11-9). The judged numbers and the reciprocals respectively go into the mathematical evaluation.

The mathematical result of the comparison/evaluation is shown in Figure 11-14. Each expert is indicated by a Roman numerical. As is shown each expert arrived at the same ranking with KBS-3 as the preferred alternative. The weight numbers given are ratio-related as a consequence of the code for the mathematical evaluation /11-10/.

The sensitivity in the result of the selected group of experts has been analysed and one result is shown in Figure 11-15. Mathematical evaluations were made for cases where Human intrusion, and Human intrusion and Technical feasibility were excluded. The ranking of the systems also in these cases turned out the same /11-10/.

Due to this the result of the exercise was judged to be clear. But could the result be due to the fact that the selected experts subconsciously favoured the most "established" system? This concern was investigated by comparing the correlation between the numbers given on each level, which showed a satisfactory low correlation, i.e. that the experts gave numbers which did not match each others. It was further noted that each system was given pros as well as cons by the experts in the comparison. With these circumstances it is obvious that the expert arrived at the same final conclusion but from different basic standpoints in details.



	I	II	III	IV	V	VI	Mean value	CoV (%)
KBS-3	0,414	0,308	0,308	0,501	0,433	0,378	0,390	19
MLH	0,249	0,282	0,264	0,285	0,253	0,298	0,272	7
VLH	0,179	0,232	0,254	0,132	0,161	0,200	0,193	23
VDH	0,158	0,179	0,174	0,082	0,152	0,124	0,145	25

Figure 11-14. Summary of the result from the six experts. Each is denoted by Roman numericals.

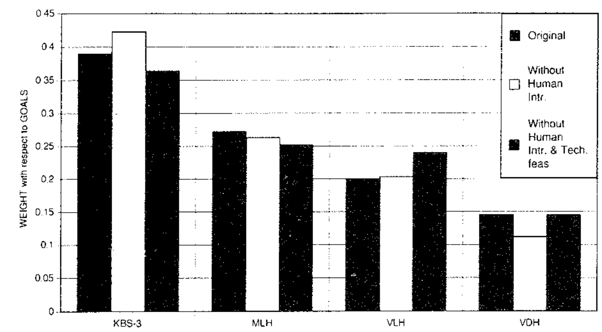


Figure 11-15. Mean value for the experts of the total result ("Technology") of evaluation with all elements (Original); with "Technical feasibility", "Construction" and "Deposition/Sealing" (without Human Intr.); and with "Construction" and "Deposition/Sealing" (without Human Intr. and Tech. feas.).

The major technical merit of the KBS-3 system was expressed as having the disposal of each canister as one closed operation, while the other systems entail a certain interdependence or serial connection. The reason for placing VDH at the bottom is the uncertainty associated with the disposal process.

The result was considered that clear, that a continued pairwise comparison with more experts was not judged necessary for ranking the systems with respect to "Technology".

11.5.3 Comparison of ranking with regard to "Long-term performance and safety"

The organization of the Long-term performance and safety related questions were made in the same hierarchical way as for technology. But the comparison turned out to be much less complicated because the differences between the four systems were fewer /11-11/. Another important factor in the analysis is that the systems have been designed in a way that earlier have been judged to provide sufficient margins for a safe isolation of the spent fuel.

The study consisted of a comparison of each individual barrier as well as of the barrier systems as a whole.

11.5.3.1 Comparison of individual barriers

Canister

In the three systems at about 500 m depth the copper protection is considered to provide the same isolation potential. In order to discuss any difference it has to be assumed that there is a defect in the canister. And in such a case the KBS-3 system is somewhat advantageous due to the upright position of the canister. In a horizontal canister the defect may be in the bottom part whereby the effect could be a pulsing pumping-out of water due to the evolution of hydrogen gas inside the canister.

The VDH canister is also given a con because of the potentially earlier release.

Bentonite buffer

The KBS-3 is considered to have the best potential due to the larger amount of buffer material when including the backfilling of the deposition tunnels.

The VDH buffer is the thinnest and also the most difficult to control after disposal.

Disturbed rock zone

This barrier is denoted with a con for the KBS-3 system due to the larger zone around the disposal drift when blasted.

The VDH is possibly advantageous compared with KBS-3 but in case of break-outs in the hole at depth the situation can be the opposite.

Far field

KBS-3/MLH are considered to have pros over VLH due to the contact with less regional fracture zones. The long extension of VLH is anticipated to cross also major fracture zones.

Compared with VDH KBS-3/MLH are denoted with cons due to the much longer transportation paths in VDH and the potentially strong barrier effect of a salt gradient.

Biosphere

Both VLH and VDH have pros compared with KBS-3/MLH because of the greater dilution potential of a wider spread repository.

11.5.3.2 Repository systems

The KBS-3 system has been the subject of detailed safety analyses. The latest, SKB 91, concluded that the major retardation for migration of dissolved radionuclides is in the near-field barriers comprising the conditioned spent fuel, the bentonite buffer and the near-field rock. Although this analysis was conducted for the system with the lead-filled copper canister the conclusion is applicable also for the system with the copper/steel canister.

An analysis has also been made for the VLH system, assuming the copper/steel canister, with the same models for the near-field performance. The conclusions were the same as in the SKB 91 study. Based on these two results the conclusion was drawn that a similar analysis of the MLH system with any of the copper canister designs would result in concurring judgements.

The VDH system was not modelled because it was early in the study clear that this system would not compete for the top ranked position. With respect to long-term safety the VDH has a large potential in the far-field rock and specifically in a high and increasing salinity of the ground water at depth as indicated in the Gravberg boring (about 6 500 m depth) and in Russia and Ukraine /11-12/. It is on the other hand not known whether the high salinity is common or not at 2 000 - 4 000 m depth in the Swedish crystalline bedrock. Only one area has been penetrated to that depth (Siljan area in central Sweden). In the course of the study it has also been concluded that the VDH system represent a less robust disposal method than the other three systems because of the higher uncertainty involved in the control of the canister and the bentonite buffer conditions during and after disposal.

11.5.3.3 Conclusion

The conclusion is that all the three systems at about 500 m depth (KBS-3, VLH and MLH) are equal from a Long-term performance and safety point of view. The are further judged to be more robust than the VDH, which mostly is relying on one barrier. The isolation potential is low in the analysed design in case this barrier fails.

11.5.4 Comparison of ranking with regard to "Costs"

All the four compared systems have been cost calculated in the manner adopted in the annual calculation of all back-end costs. The systems have been adjusted in factors like construction and disposal times, rock excavation and backfill volumes etc., so that cost differences obtained really are describing the differences between the systems. The result is summarized in Table 11-3 /11-13/.

The ranking in PASS then placed MLH in front of KBS-3 as the difference is considered to be significant although minor. It is significant because the volume of underground excavation is less for MLH. The number in MSEK, however, is minor and estimated to MSEK 400 (discounted value) out of MSEK 8 000 for Common Facilities, Encapsulation Plant and Deep Repository.

KBS-3 and VLH are placed second. The equal total costs for the systems are dependent on two major factors: 1) The costs for the canisters in VLH are higher per tonnes of spent fuel due to a higher metal content, 2) The costs for excavation and backfilling are higher for the KBS-3 system.

The VDH system is substantially more expensive than the other three systems.

The result is not changed when analysing optimization potentials underground for each system.

Table 11-3. **Summary of cost calculation for Common Facilities, Encapsulation Plant and Deep Repository. The comparison factor is based on the present value as per January 1992 for a discount rate of 2.5%.**

System	Comparison factor
KBS-3	1.05
MLH	1.00
VLH	1.06
VDH non-consolidated	2.58
VDH consolidated	1.70

11.5.5 Summary of ranking of repository systems

Table 11-4. Summary of results from the three interim comparisons of repository systems. A ranking of 1 is the best.

System	Long-term performance and safety	Technology	Costs
KBS-3	1	1	2
MLH	2	1	1
VLH	3	1	2
VDH	4	4	4

The result of the three interim comparisons are shown in Table 11-4. The merging into one verdict is arrived at by considering the major factors that differ between the systems. In the first place a grouping into three categories can be made.

A KBS-3 and MLH
B VLH
C VDH

The VDH system is placed last in all three interim comparisons. This is indisputable for Technology and Costs. For Long-term performance and safety the outcome is very much a matter of judgement. The lower ranking in this interim comparison is basically due to the fact that the long term isolation potential is mostly dependent on only one barrier, the geosphere, which today in Sweden is known to only a limited extent at greater depths. An improvement of the engineered barriers in order to enhance the multibarrier principle is judged technically feasible but with higher costs as a consequence. However, the costs are already in the analyzed design not competitive.

The major reason for placing VLH behind both KBS-3 and MLH is the large and heavy canister. This introduces higher uncertainties in the disposal process as well as higher costs for the canister. In comparison with KBS-3 the canister costs are compensated for by the lower volume of rock excavation, but the KBS-3 technology is judged supreme. In comparison with MLH the costs are slightly a disadvantage for the VLH, and this is not compensated for by Technology. Even if the repository design would be the same for the two systems (a number of parallel disposal drifts within a rectangular rock block) the MLH disposal method is judged preferable over the VLH method.

KBS-3 and MLH are both top ranked. The choice between them is based on an evaluation of Technology versus Costs. For Technology the main factor placing KBS-3 in front of MLH is the deposition procedure. The KBS-3 method is based on a rather simple technique for placing the bentonite blocks in the disposal hole, which even can be man-inspected. The canisters are lowered down into the buffer and can be positioned with small means. After filling the hole the deposition process is closed for the next to start. In MLH the emplacement of the buffer takes place close to a deposited canister and has therefore to be remotely controlled. The canister is inserted into the disposal position horizontally by pushing the canister into the hole in the bentonite buffer. The disposal process is not ended until the whole disposal drift is filled, which limits the flexibility with regard to interruptions and planned pauses.

11.6 CONCLUSION FOR RD&D PROGRAM

The conclusions from the comparison and ranking in PASS are:

* Canister with a capacity of 12 BWR assemblies or equivalent thermal load (KBS-3 and MLH) are recommended over canisters with twice the load (VLH).
* Copper/steel canister design is recommended.
* KBS-3 system is recommended.

Due to this the future RD&D programme is based on:

* KBS-3 as maintained reference system.
* Copper/steel canister (composite canister) as new reference canister design.

The lead-filled copper canister is studied as back-up for the reference canister design. No back-up is studied for the repository design.

12 SAFETY ANALYSIS

12.1 GENERAL

Most of the activities within the Safety Assessment Group during 1992 have been connected to finalization of the SKB 91 safety assessment and follow-up work to that, and to the planning of the future activities as presented in the new RD&D programme published in September 1992. Although SKB 91 was not reported until May 1992, the background and goals for the assessment, the methodology used, the results obtained and the conclusion drawn have already been summarized in SKB Annual Report 1991, and in SKB RD&D-Programme 1992.

This overview of the activities will focus on activities during 1992 not reported in SKB 91 and on the ongoing planning for future activities. Activities on the development of databases or models to describe the performance of the spent nuclear fuel or the man-made barriers are discussed in the respective sections.

12.2 SCENARIOS

The assessment of future scenarios occupies a central place in the safety analysis. Several attempts have been made in different countries to formalize this type of assessments and establish some general procedure which would limit the risk of overlooking important phenomena or environmental premises.

The models used in the safety analysis shall clarify the performance of the repository in connection with all the features, events and processes (FEPs) that can affect the repository. The methodology to build scenarios is to identify such FEPs, to sort and screen them and to combine them. The resulting number of scenarios are then sorted and screened and finally, in an expert judgement step, the most interesting combinations are chosen for calculations and evaluations.

To have a good overview, and to make the above scenario development process transparent, a good presentation methodology will be needed. Preliminary methodology work has been initiated during 1992.

12.3 RADIONUCLIDE TRANSPORT IN THE NEAR FIELD

Release calculations in a repository of the very long tunnel type:

The VLH and KBS-3 are two alternative designs for high level waste disposal. The repositories differ mainly in the layout and the canister design. A study of comparison of the nuclide release from the two designs has been made /12-1/. This study forms a part of the background materiel in PASS.

In the VLH-repository for high level wastes, the canisters are placed horizontally in long boreholes. Compacted bentonite will be used to backfill the tunnel. The repository will be located in a uniform rock mass with good properties for storage and tunnelling. In the disturbed rock zone in the vicinity of the tunnel, the water flows mainly along the tunnel. In fractures and in fracture zones the water flows around the tunnel.

The nuclide releases through different pathways were calculated for several cases. Pu-239 was chosen to study the role played by different pathways for the release. The numerical code used in the calculations is the compartment model /12-2/. The results are summarized below.

If one canister is damaged, the nuclides escape through the damage into the bentonite. Some go to the disturbed zone and some into the fractures and fracture zone. Several cases were calculated to quantify the role of the different pathways on the nuclide release into the water. The disturbed zone accounts for 70% of the release into the water. Fractures intersecting the tunnel account for the remainder. The release is quite sensitive to assumptions used to define the boundary conditions in the fracture and in the disturbed zone. The nuclide release is strongly influenced by the size of the damage in the canister wall. The dominant transport resistance is in the small hole (damage) in the canister. For hole sizes larger than 1 cm^2 area, the resistance of the pathways from the bentonite begins to play a role in the nuclide escape.

The influence of the location of a fracture zone was investigated by varying the distance between the zone and the canister nearest to the zone. For distances larger than 0.5 m, the zone has a minor importance for the release of nuclides. The disturbed zone accounts for most of the release. The influence of the location of fractures along the tunnel was also investigated. This was done for the case when all canisters leak. The location that gives the largest release is when the fracture is opposite to the damage. However, it has no large significance for the total release.

The concentration build up in the disturbed zone was investigated. For the non-sorbing nuclides that have a large decay constant (I-129), the effluent concentration downstream does not change with the travelled distance in the disturbed zone. For some sorbing nuclides (Cs-137, Ra-229, and Pu-239) the relative concentration decreases considerably with the distance downstream of a leaking canister, depending mainly of the sorption coefficient and the decay constant.

The nuclide releases from the VLH and the KBS-3 repositories were compared assuming that the damage in the canister wall is a small hole and only one canister leaks. The nuclides chosen for the comparison were I-129, Cs-137, and Pu-239. The releases are of the same order of magnitude for a given nuclide. For the sorbing nuclides the uptake in the water is very small compared with the release from the canister. The reason is that for these nuclides the sorption in the bentonite gives the nuclides time to decay. The non-sorbing nuclides with low decay constant (I-129) do not decay much.

Small differences were found for the nuclide escape from the two repositories types. The KBS-3 gives somewhat larger nuclide escape than the VLH-repository. They are mainly due to differences in the thickness of the canister wall, and the larger inventory in the VLH case. The differences are well within uncertainties in the assumptions.

Verification of the compartment model for calculating radionuclide release in the near field of a repository:

The compartment model /12-2/ mentioned earlier has been designed to calculate radionuclide release in the near field of a repository. The model is a simplification of a full, three-dimensional, integrated, finite-difference model. The simplifications are supplemented by introducing analytical solutions at sensitive zones. The model is fast, flexible and simple in concept and use. Because of the rough discretization compared with a full three-dimensional code, the results obtained with the compartment model are not exact. In this paper, an attempt is made to verify the compartment model with analytical solutions and a known numerical code, TRUCHN. From the verification analysis, it may be concluded that the accuracy of the compartment model is similar to that of models that use a very detailed discretization.

12.4 NEAR FIELD CHEMISTRY

Development of "CHEMFRONTS", a Coupled transport and geochemical program to handle reaction fronts:

Water flow through porous media is of interest in various fields. Many geochemical reactions are caused by the water flow and infiltration of reactive species such as hydrogen ions and oxygen. Prediction of geochemical reactions for geological time periods of thousands and maybe millions of years are useful for matters such as the final disposal of nuclear and other hazardous waste, and the degradation of concrete.

The objective of the project was to develop a coupled geochemical and transport computer program to predict geochemical reactions over geological time scales. The program should be able to handle sharp reaction fronts such as the redox and dissolution fronts that occur in reducing bedrock and concrete. It should be written in FORTRAN77 to be portable, and be easy to modify.

The computer program, CHEMFRONTS /12-3/, is based on the quasistationary state approximation. The species in the aqueous phase are assumed to be in equilibrium, whereas the solid phase dissolves and precipitates with a kinetic reaction rate.

CHEMFRONTS can be used for calculating complicated problems with many simultaneous moving fronts including redox fronts. The program has been verified by applying it to examples found in the literature. The results from CHEMFRONTS compare well with those from programs based on other models. Computations of problems taken from natural analogues as Poços de Caldas and Cigar Lake have shown encouraging results.

Thermodynamic modelling of bentonite-groundwater interaction and implications for near field chemistry in a repository for spent fuel:

Predictions of near field geochemistry are made using a thermodynamic model for bentonite/groundwater interaction /12-4/. This model is a refinement and extension of the model developed by the senior author. It is based on recent experiments performed at high solid/water ratio and adapted to the Swedish type of HLW repository design. Thus, from the obtained experimental results on solution composition, the model includes chemical reactions resulting from both the impurities and the main clay fraction within the bentonite. Ion exchange reactions are treated both with and without the contribution of edge sites. Due to its thermodynamic basis, the model exhibits prediction capability over a wide range of conditions in terms of solid/water ratio.

The modelling of repository conditions implies, due to the lack of experimental information, simplifications with regard to thermodynamic properties of the bentonite. This mainly involves the non-consideration of the temperature effect and of the acid/base properties of the solid. Nevertheless, our results yield insight into important processes affecting porewater chemistry. Thus, the model suggests that proton exchange reactions may exert a strong control on calcite dissolution within highly compacted bentonite. Estimations of chemical changes over time in the bentonite were done on the basis of a mixing tank model. These results indicate transformation of Na-bentonite to Ca-bentonite over time. The extent of this process, however, critically depends on the amount of carbonate present in the bentonite.

12.5 MODELLING OF TRANSPORT IN THE FAR FIELD

A detailed account of many of the calculation models used by SKB for groundwater movements and radionuclide transport within safety assessment is provided in chapter 8 of the SKB 91 report /12-5/. Different conceptual models and strategy for modelling of transport in the far field are discussed there. The following section therefore provides only a brief summary of the models used, together with some supplementary information on the models not utilized in SKB 91 and other studies within the far-field programme of safety assessments.

One of the demands on the calculation model for geohydrology to be used in SKB 91 was that it should be possible to take into account the spatial variability in the hydraulic properties of the rock. Modelling was therefore done with a stochastic continuum model, HYDRASTAR. Using the Monte Carlo technique, realizations of the hydraulic conductivity and potential fields were generated conditioned upon measured borehole data.

The stochastic approach for groundwater flow calculations has been further developed. During the year, most efforts have been concentrated on improving the documentation and the result presentation.

NAMMU, a general code for groundwater flow and radionuclide migration calculations, will be a valuable tool also for future safety assessments. Release 6.1 is available and some of the features to mention are simplified pre- and postprocessing facilities, improved numerical algorithms, options for borehole model, sensitivity analysis and improved documentation.

It was not deemed possible in SKB 91 to use a "discrete fracture" model for simulations of the groundwater flow in a block on the order of 25 km conditioned upon the local fracture patterns around each measurement section.

However, a discrete fracture approach may be utilized in order to generate basic data for conditional simulation with HYDRASTAR in terms of block conductivities derived from fracture statistics and pumping tests. A preliminary study, /12-6/, indicates that conductivities estimated from simulated packer tests, using the discrete fracture approach, and equivalent block-scale conductivities shows poor correlation. High uncertainty should therefore be taken into account in estimating the properties of a stochastic continuum from packer tests data.

Observations in rock of the channelling of the water flow and rapid transport pathways have also led to the development of special channelling models for simplified calculation of nuclide transport, e.g. in the safety assessment for SFR.

Mixing between channels can be foreseen for large-scale calculations, and for this purpose a stochastic model consisting of a network of channels has also been developed /12-7/. Thus, it is the water-conducting channels and not the fractures in the rock that constitute the basis for the model.

A comparison has been made with the traditional advection-dispersion approach and a pure channelling model. The comparisons were done using the same water residence time distribution, mean residence time and dispersion, and the same flow-wetted surface. It was found that the water-residence time distribution has a very strong impact on the solute transport of nonsorbing and sorbing species. The channel network exhibits a strong retardation if compared with the other models and dampens the channelling effects for sorbing species.

A research project, at the department of Land and Water Resources at the Royal Institute of Technology, concerning the important issue of methods for upscaling of field data into effective values for model parameters when predicting groundwater flow in fractured rock, has been finalized. The thesis has been published /12-8/.

The study deals with numerical calculations of heterogeneity of groundwater flow and solute of transport in hypothetical blocks of fractured hard rock. The major conclusions are that heterogeneity gives rise to anisotropy in the upscaling process and also that the choice of support scale is crucial for the modelling of solute transport.

This project inspired another study where the techniques for interpretation and regularization of single-hole hydraulic tests were studied. Due to lack of data, hydraulic conductivity measurements on a small scale are usually scaled-up in order to fit a comparatively coarser numerical discretization. The Monte Carlo method has been utilized in order to investigate the relationship between the conductivity value interpreted and the corresponding radius of influence in conjunction with single hole tests in heterogeneous formations. The results of the simulations indicate that the traditional methods for scaling-up hydraulic conductivity data, assuming a constant lateral scale of support and a multinormal distribution, may lead to an underestimation of the persistence and connectivity of transmissive zones in highly heterogeneous porous media /12-9/.

A realistic synthetic transmissivity field and dipole test conditions similar to the actual field tests at Finnsjön were used to demonstrate the calibration and validation procedure applied to a stochastic continuum model /12-10/. The results show that calibration on a local scale is insufficient for validating the model on another larger transport scale. More measurement data on relevant scales can reduce the uncertainty of the simulation. The study is a contribution to the INTRAVAL, phase 2.

The further refinement of models for transport in the far field will be oriented towards the goal of being able to carry out performance assessments on the candidate sites and an integrated safety assessment in 1996. Arriving at this goal will require follow-up assessments to SKB 91, further refinement of various model concepts, and verification and validation procedures of the calculation models that are intended to be used in the future. A large part of these activities will be coupled to the international Task Force for groundwater movements and nuclide transport in fractured rock within the Äspö project. INTRAVAL is also an important forum for questions surrounding validation.

The method that has been used in SKB 91 for stochastic continuum modelling of groundwater movements will be further refined. The calculation programs HYDRASTAR and INFERENS will be improved. HYDRASTAR has been specially developed for the needs of SKB 91 and needs to be generalized.

13 SITING OF A FINAL REPOSITORY FOR SPENT FUEL

13.1 GENERAL

The siting of a deep geological repository for spent nuclear fuel and other long-lived waste is one of the main remaining tasks within the Swedish Nuclear Waste Management programme. During 1992, the preparation of the RD&D-Programme 92 /13-1/ has implied activities relating to strategy and planning for site selection and site characterization. Studies have also been made on several supporting activities, such as the status of site selection programmes in other countries /13-2/ or the legislation regulating the siting and construction of a deep repository /13-3/. Parallel to the planning activities a comprehensive GIS (Geographical Information System) database for geoscientific, environmental and other regional data have been made operational within SKB.

13.2 STRATEGY AND OVERALL SCHEDULE

SKB concludes in the RD&D-Programme 92 that the scientific and technical premises necessary for implementing a safe repository for spent nuclear fuel exist in Sweden and that it is most probably possible to meet the stipulated geological requirements in many parts of the country. When selecting regions for further siting studies other important factors must also be carefully considered, such as plans for land use, transportation of spent nuclear fuel, public opinion and landownership. The RD&D-Programme 92 therefore involves two new components:

- Feasibility studies of those municipalities which display an interest in having a closer examination made of their premises for a deep repository before any selection of sites for field investigations are made.
- Construction of the repository in stages. The first stage being planned for about 10% (800 tonnes) of the projected total amount of spent fuel.

To provide everybody with the same basic information the RD&D-Programme 92 was distributed to all local municipalities in Sweden (286), together with a letter in which SKB offered to provide further information and to make a feasibility study in case there would be a mutual interest from a municipality and SKB to study the issue further. In a feasibility study, fundamental facts are gathered and evaluated on, for example, transportation-related matters and the societal and geological premises for a deep repository in the municipality. With the aid of a feasibility study, both SKB and the municipality can, at an early stage and without committing themselves, obtain a preliminary idea of the premises and decide whether the possibility of siting a deep repository in the municipality is worth examining more closely by site-investigations in some potentially suitable area (to be defined within the feasibility study) of the municipality.

13.3 PLANNING

During 1992 planning and preparation activities concerning the first stage of the siting programme have been carried out. This stage includes general overview studies of siting factors, the feasibility studies of interested municipalities, the site characterization programme on two candidate sites and supporting activities necessary for the application for the detailed investigations (which will include shaft/tunnel to repository depth). These planning activities will continue during 1993 and consist of the following:

- programme for feasibility studies,
- programme for geoscientific site investigations,
- quality assurance programme,
- programme for safety assessments,
- studies and analyses of the technical premises and of the repository system,
- programme for preliminary environmental impact statement,
- programme for local involvement, information and socio-economic studies.

A generalized schedule for the proposed siting and construction of the repository is shown in Figure 13-1.

13.4 SITE INVESTIGATION PROGRAMME

During 1992 work started to evaluate the experiences from the study site investigations, carried out at about 10 sites, and the investigations for the Stripa Project, Äspö Hard Rock Laboratory, SFR and CLAB. The evaluation will, together with experiences from corresponding site investigations in other countries, mainly Finland and Canada, form the basis for the site investigation programme of the two candidate sites. The planning for the site investigations also includes the following activities:

- instrumentation needs, site investigation methods and available resources,
- organisation and personnel resources,

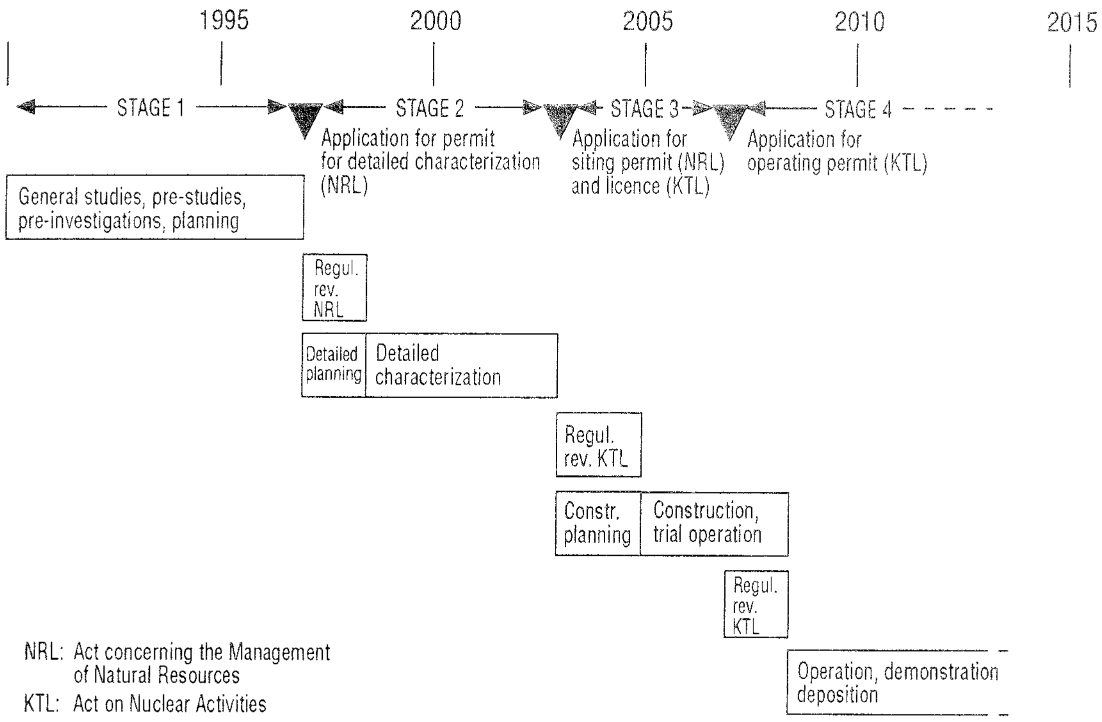


Figure 13-1. *Overall schedule for the siting and construction of the deep repository for demonstration deposition. The earliest possible times for the completion of the various stages are given.*

- data management and documentation,
- local information.

Regarding instrumentation and personnel needs, SKB have a rather complete set of instruments and have access to a well trained personnel for site investigations. However, the need to study two sites in parallel will demand developing/purchasing of new instruments and probably training of new personnel. Planning for these activities are in progress.

Most investigation methods, planned to be used at the candidate sites, have been extensively tested and no need for major developments are foreseen. However, for some methods further development and testing are justified. One example is reflection seismic. This method is probably the only one that have potential for identify subhorizontal fracture zones and subhorizontal rock boundaries in crystalline rock. Such information is highly appreciated at an early stage of a site investigation programme, preferably before any drilling has been made. Earlier test using reflection seismic at Finnsjön and Äspö have not been successful. However, there are reasons to believe that recent developments have improved the usefulness of the method. A joint study by SKB and TVO therefore started in 1992. This study includes both reprocessing earlier test surveys, optimizing the data acquisition system for typical Swedish/Finnish overburden and bedrock conditions, and tests surveys.

Data management and documentation activities are aimed to provide a rationale data flow from the field survey to the central database. This is of importance for many reasons, one is quality assurance. During 1992 work started to produce detailed instructions and technical manuals for tests and data management, along with planning for reporting and data storage.

Some site investigation activities at Äspö and Laxemar have been video recorded. The objective is to produce a video and a brochure that can be used for local information regarding instruments and methods for site investigations.

13.5 ENVIRONMENTAL IMPACT ASSESSMENT

SKB plans to prepare a preliminary Environmental Impact Assessment (MKB) at an early stage of the siting work. During 1992 work started on a first version that will be published during the end of 1993. This first version will mainly be based on generic data and will concentrate on non-radiological environmental impacts.The purpose is to use this MKB as a basis for discussions with municipalities, local populations and regulatory authorities of the facilitys environmental impact. A full MKB, including radiological safety during operation and long-term disposal, and the results of the dialogue process between SKB and the local municipalities at the candidate sites, will be presented as a part of the support needed for the application to conduct detailed characterization, see Figure 13-1.

13.6 DATABASE

During 1992 a GIS database was made operational at the SKB office. The database will be used for both geological and non-geological regional analyses. In the first phases of the siting program, i.e. overview studies and the feasibility studies, available data concerning various siting factors will be compiled and evaluated. Later in the programme, during the site investigation phase, the database will also be used to store and evaluate air-borne and surface data acquired during field surveys.

The database contains typical geographical data, such as administrative borders, municipalities, roads, railways, digital elevation data, coastlines, lakes and rivers, etc. Other non-geological data stored in the database are land use, landowners (forest companies), population density, areas of national interest for nature conservation, restricted areas, culturally protected or archaeological interesting areas etc.

Geological data stored in the database, or in the process of being stored, are distribution of rock types and Quaternary deposits, airborne geophysical data, gravimetrical data, lineaments and deformation zones, exploration permits and exploitation concessions, hydrogeological properties determined from well data, geochemical data, seismicity, post-glacial rebound etc.

An example of how the database is used is shown in Figure 13-2, in which the maximum extension of the post-glacial sea in Sweden has been delineated by combining digital elevation with data of the highest post-glacial coastline from different parts of Sweden. The map is used for several reasons, one is for interpreting groundwater characteristics and history.

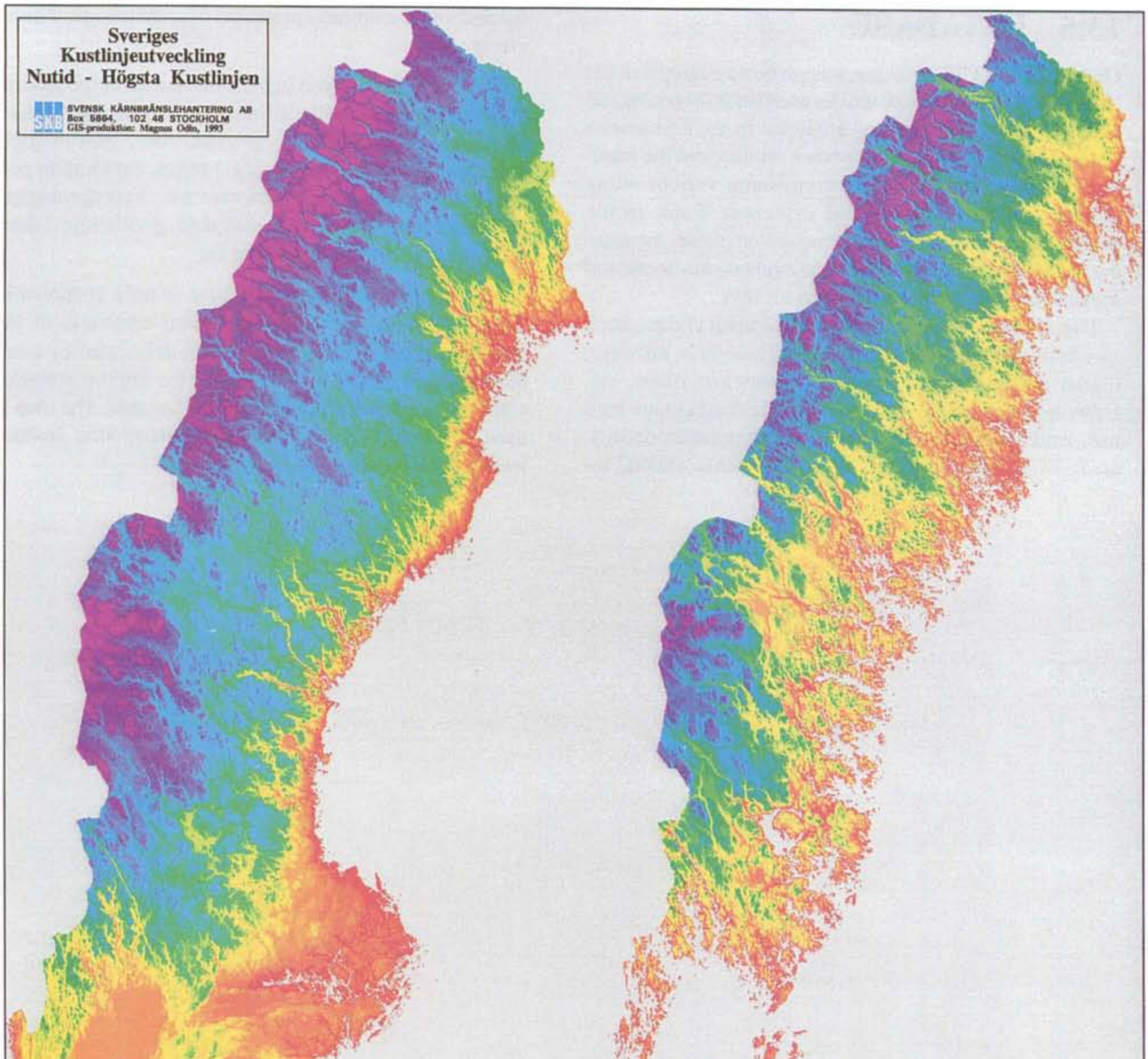


Figure 13-2. *The maximum extension of the post-glacial sea in Sweden immediately after deglaciation (right). Since the ice sheet in southern Sweden retreated about 3000 year before northernmost part of the country, the map as a whole does not reflect the extension of sea level at any specific time. The current Swedish coastline is shown to the left. (Data sources: Elevation data – National Land Survey. Postglacial sea-level elevation data – N-O Svensson, Department of Quaternary Geology, University of Lund).*

14 SPENT FUEL

The cooperation with other groups in the world performing similar studies has continued during 1992, through the spent fuel workshop that was held in Visby and arranged by SKB. More direct cooperative work has also been performed with Atomic Energy of Canada Ltd.

An account of the status of SKB's programme for spent fuel studies has recently been published /14-1/.

14.1 FUEL CHARACTERIZATION STUDIES

A significant part of the experimental effort has been devoted to the detailed characterization of spent fuel before and after corrosion tests. The aim of the work has been to:

- identify corrosion sites,
- attempt to correlate corrosion processes with fuel structure and radionuclide distribution, and thus with the fuel's irradiation history, burnup linear heat rating etc..

One correlation is of course already fairly well understood; the Instant Release of cesium and iodine nuclides.

Several methods and tools are currently used for the fuel characterization, the most important ones being:

- measurement of the radial variation of grain size and porosity using scanning electron microscopy,
- systematic examination of polished and pseudo-fracture surfaces at high magnification using scanning electron microscopy,
- measurement of the radial concentration profiles of selected fission products, Nd, Cs, Sr, Mo and Xe using wave length dispersive X-ray microprobe analysis,
- semi-quantitative determination of radial variation of alpha activity using alpha-track etching/photodensitometry of exposed plastic films.

Particular attention is given to examination of the zone at the pellet rim. This zone exhibits significantly higher local alpha activity and fission product contents, than in the bulk of the pellet. It is worth noting that a 30 µm wide zone contains about 2% of the total pellet activities of fission products and actinides.

Characterization of all reference fuels used in the programme is now completed, although the analysis and evaluation of the data are not yet fully completed. The fuel used to date are:

- Oskarshamn 1 BWR fuel, 41 MWd/kgU,
- Ringhals 2 PWR fuel, 42 MWd/kgU,
- Ringhals 1 BWR fuel, stringer rod with burnup varying along the rod from 21 MWd/kgU to 48 MWd/kgU,
- Ringhals 1 short fuel rod refabricated from a full length BWR fuel rod with a burnup of 44-48 MWd/kgU. The rod was power-bumped up to a maximum linear power of 43 kW/m.

The type of result obtained is illustrated in the diagrams in Figures 14-1 and 14-2. Figure 14-1 shows the radial variation in burnup of the Oskarshamn 1 BWR reference fuel. Although the bulk burnup is 41 MWd/kgU, the rim burnup attains a value of about 65 MWd/kgU. The width at half peak for this effect is about 130 µm. The formation and isotopic composition of the fuel's alpha-emitting inventory are dependent on not only the burnup, but also on the irradiation history and, of course, the local inventories will vary with decay time after irradiation. Figure 14-2 shows the alpha radial variation in the Oskarshamn 1 fuel at a decay time of about 13 years. The effect is much more pronounced than for burnup distribution and has a width at half peak height of about 35 m. The porosity in the fuel is also increases towards the periphery of the fuel and increases from about 1% in the bulk of the pellet to 5% in the rim region. This increased porosity at the fuel rim is also illustrated in the photo micrograph in Figure 14-3.

The structure and characteristics of the rim region develop successively with increasing burnup. This is illustrated by the scanning electron micrographs in Figure 14-4. The micrographs are from the Ringhals 1 stringer rod, which has a burnup increasing from 21 MWd/kgU to 48 MWd/kgU along the rod. The figure shows fuel fracture surfaces at the pellet periphery at three locations with bulk burnup of 21 MWd/kgU, 36 MWd/kgU and 48 MWd/kgU. The specimen with lowest burnup displayed predominantly transgranular fracture. At higher burnup (36 MWd/kgU) the fission gas bubble network was so developed that intergranular fracture was observed. Bubble sites and particles (or subgrains) of recrystallized UO_2 can be observed on grain edges and faces. In the specimen with 48 MWd/kgU burnup, the characteristics of the high burnup rim are well established, with increased porosity and loss of grain structure manifested by the formation of sub-micron particles at pore and grain surfaces and edges.

It is possible that the recrystallization process involves redistribution of certain nuclides to the particle surface, with enhanced propensity for corrosion or leaching as result. The increased porosity and alpha-dose rate may also contribute to making the rim region more prone to corrosion than the bulk of the pellet. Some evidence for this can be found in Figure 14-5. The comparison between corroded and uncorroded specimens shows that enlarge-

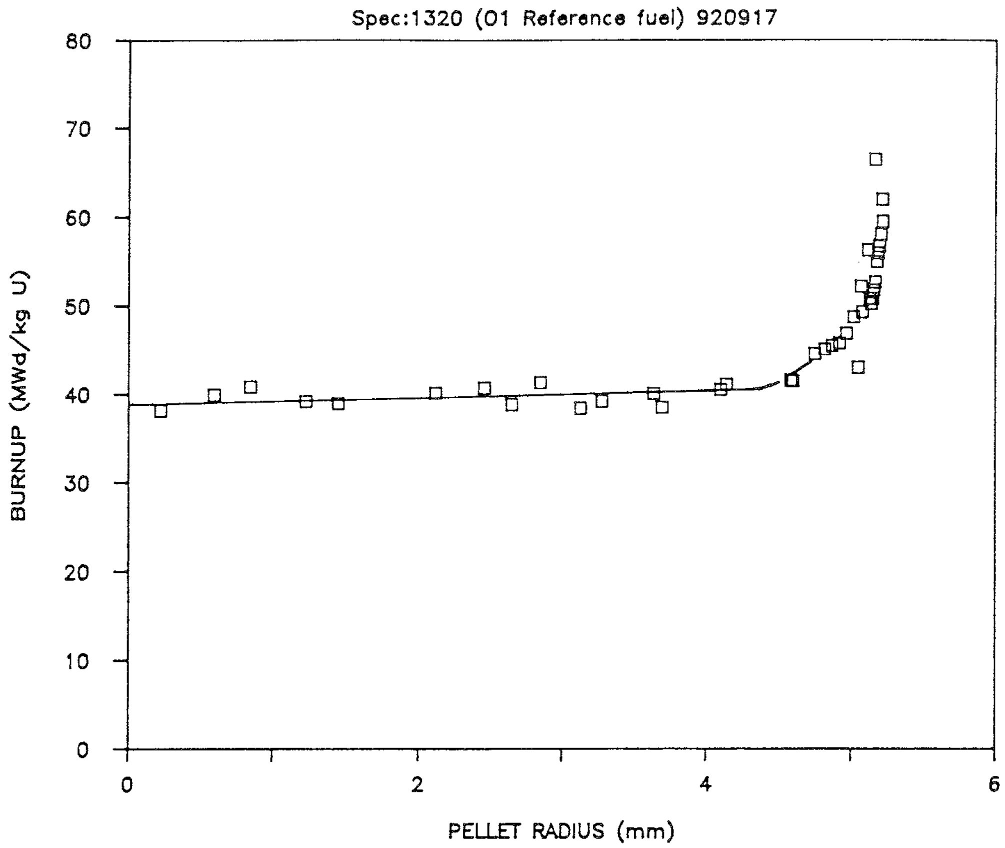


Figure 14-1. Radial variation of burnup of the Oskarshamn 1 reference fuel.

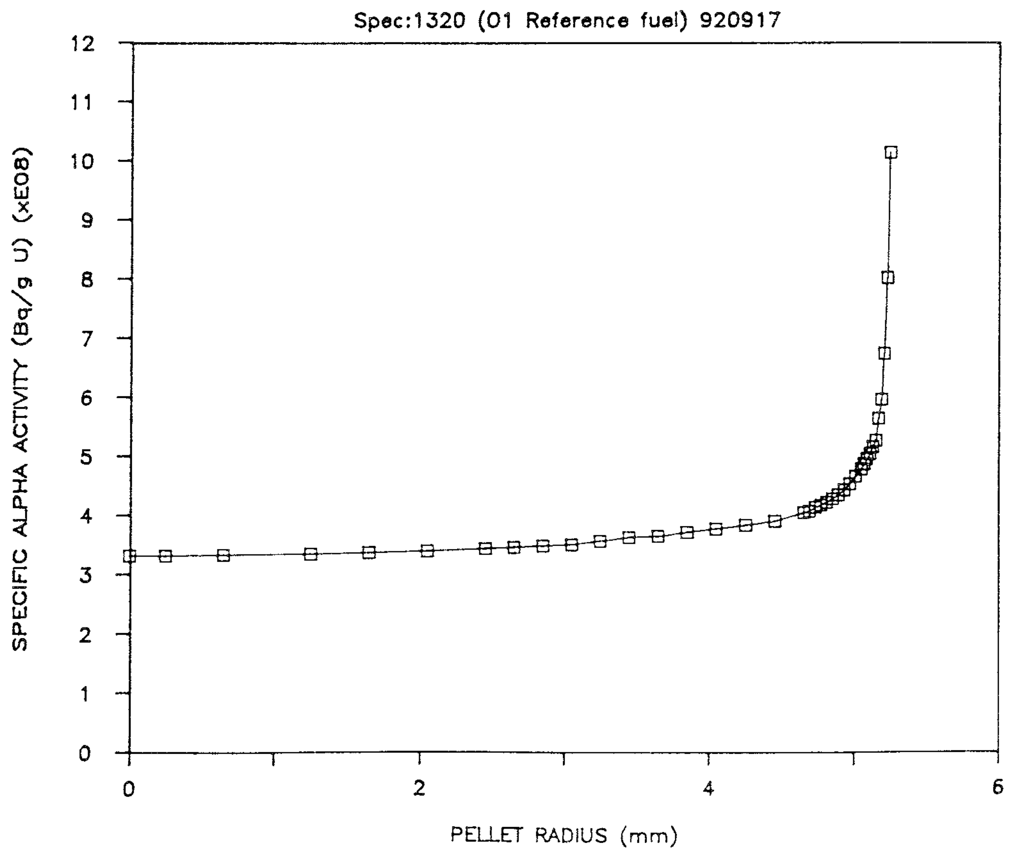


Figure 14-2. Radial variation of alpha activity of the Oskarshamn 1 reference fuel.

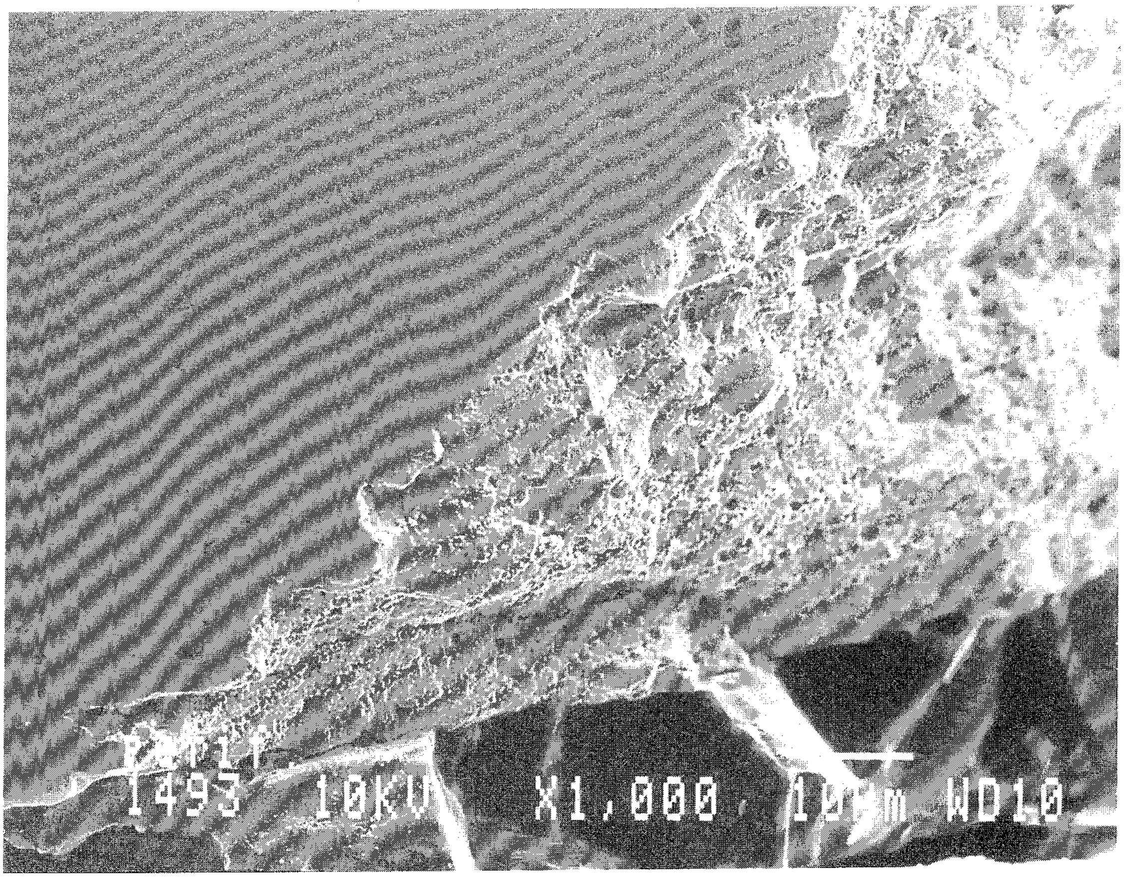


Figure 14-3. Increased porosity at the rim region on Oskarshamn 1 reference fuel.

ment and interlinkage of the peripheral porosity has occurred. This is particularly apparent in specimen 3-26, see Figure 14-5 (c), which shows that corrosion has occurred at the pellet periphery and at grain boundaries. The latter resulting in grain pull-out. Inspection of fracture surfaces of the two corroded specimens, however, showed that the small particles found in the reference material were still present. Although it is impossible to quantify their population densities, a visual assessment suggested that the corrosion process did not seem to markedly decrease their number.

The presence of crystals of UO_3 hydrates on the surface of specimen 3-26 should also be noted. After removal from the deionized water and exposed to the in-cell periscope lamp, all surfaces of the two fragments in 3-26 became covered in a few seconds with a yellow, needle-like coating, which was shown by X-ray diffraction to be dehydrated schoepite.

The increased porosity observed after corrosion could be interpreted as a preferential attack at the periphery of the fuel, but could also be a result of the grain pull-out mentioned above. Which one is a dominant process should be reflected in the solution analysis. The isotopic composition of in the first hand uranium should be observed in the water phase. The improved analytical capability when using inductively coupled mass spectrometry (ICP-MS, see below) allows analysis of individual isotopes of the predominant elements. Further studies will have to be performed before it can be established to what extent the rim region contributes to the overall corrosion of the fuel.

14.2 SPENT FUEL CORROSION STUDIES

The fuel corrosion programme was started in 1982 and specimens from the first series (series 3) are still being corroded, having now reached a cumulative contact time of over 4000 days. A second series using a PWR fuel (series 7) rod was started in 1986 and in 1990 a series of experiments on a BWR stringer rod was started (series 11). In this series the burnup varied along the fuel rod from 21 MWd/kg U to 49 MWd/kg U. For the first two series, fuels with an average burnup of about 42 MWd/kg U were used. In 1991 a series (series 13) of experiments was started, using a fuel rod that had been power-bumped up to a maximum linear power of 43 kW/m.

14.2.1 Solution analysis

During 1992, the methods for solution analyses were changed. An inductively coupled plasma mass spec-

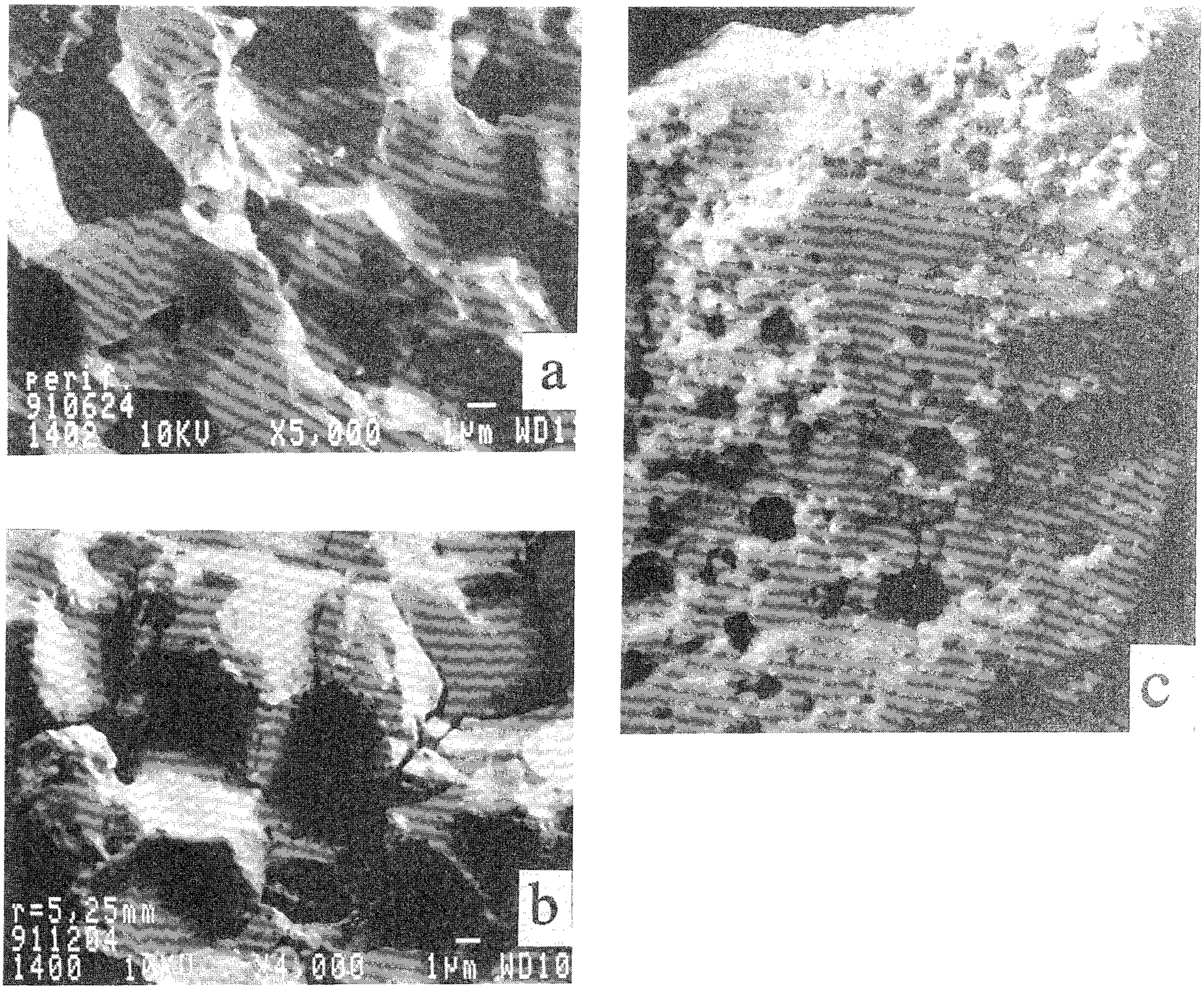


Figure 14-4. Ringhals 1 stringer rod: fuel fracture surface at pellet periphery. Bulk burnup (a) 21 MWd/kgU, (b) 36 MWd/kgU, (c) 48 MWd/kgU.

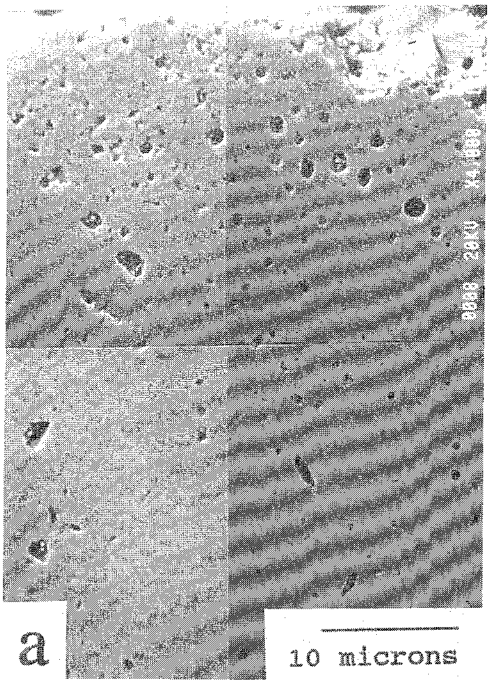
trometer (ICP-MS) was installed in the Fuel Laboratory at Studsvik during the late autumn of 1991 and commissioned for radioactive use in June 1992. Up to that date, elemental analysis was performed by laser fluorescence for uranium, alpha-spectrometry for plutonium, gamma-spectrometry for the major gamma-emitting fission products and radiochemical separation and beta-counting for ^{90}Sr and ^{99}Tc. For strontium, technetium and molybdenum optical emission inductively coupled plasma spectrometry was also used.

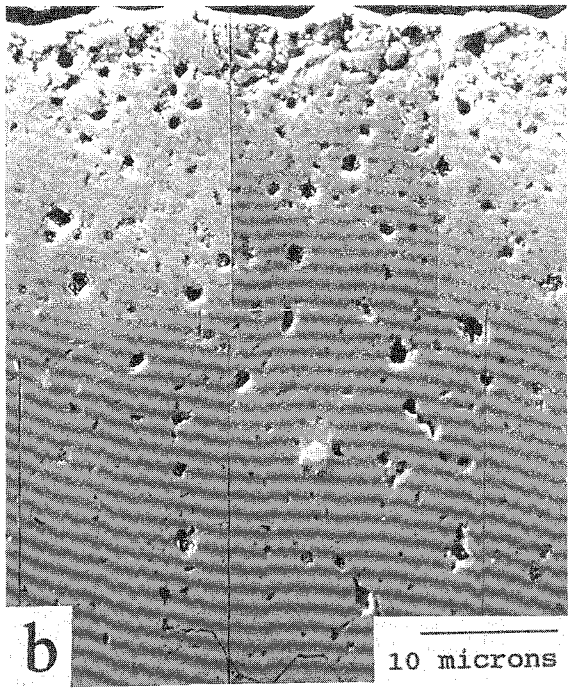
With the ICP-MS, the analysis procedures could be very much simplified, allowing the analysis of a number of elements without previous separation. The instrument has also be used for determining the inventories of nuclides in the fuel. In addition to implementing the new analysis procedures for experiments to be analysed during the latter part of 1992, much work has been devoted to re-analysis of archive solutions from earlier experiments. The results have been shown to be in good agreement with those obtained by the radiometrical methods.

In addition to more rapid analysis, the ICP-MS is capable of detecting elements that were previously not possible to analyse, either because of low concentrations in solution, such as the rare earth elements, or because of low specific activity, such as neptunium.

The isotopic composition of fuel leach solutions are very different from what occurs in natural and some development work was necessary before the instrument could be used for routine analysis, e.g. establishing methods for correcting for mass overlaps etc. In solutions with high uranium concentrations, such as oxidizing groundwater, corrections must also be made for molecular uranium species. Examples of the results obtained using ICP-MS for analysis are shown in Table 14-1 for deionized water.

Figures 14-6 and 14-7 show mass spectra of the actinide region of a deionized water after contact with BWR fuel (Figure 14-6) and for comparison the same spectral region for the same fuel after dissolution in nitric acid, i e the fuel inventory (Figure 14-7). A comparison between the two





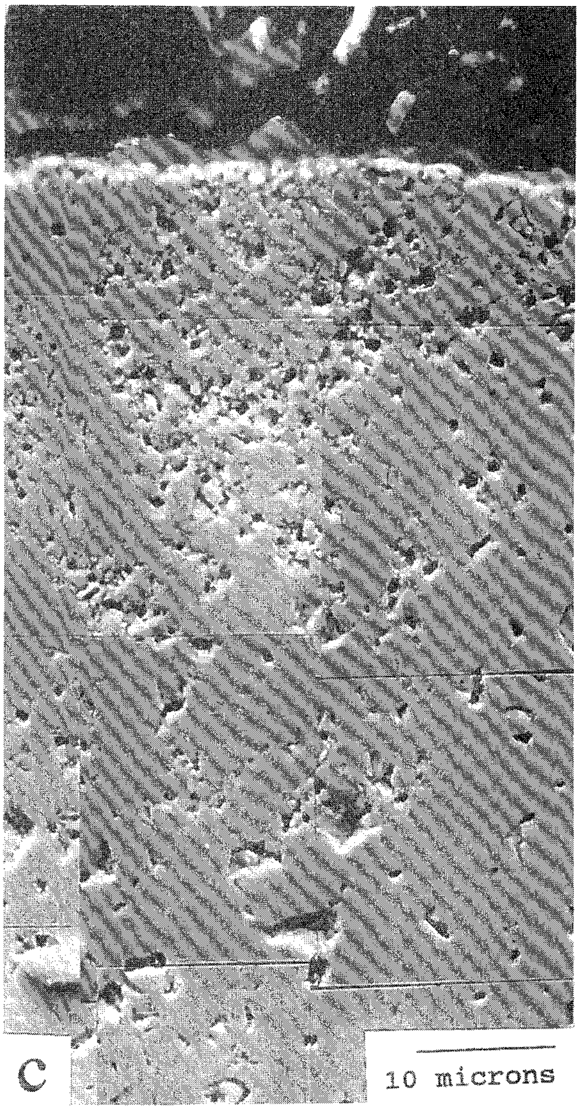


Figure 14-5. *Photo mosaics of fuel rim zone. (a) Uncorroded fuel, (b) corroded in groundwater 1427 days, (c) corroded in deionozed water 1521 days.*

spectra shows clearly the incongruent release of actinides from the fuel under oxidizing conditions.

14.3 NATURAL ANALOGUES

The use of natural uraninite (UO_{2+x}) as an analogue for the longterm corrosion products of the UO_2 in spent nuclear fuel requires the careful analysis of the alteration products of uraninite under experimental and natural conditions. A detailed experimental study has been performed of the oxidation products of uraninite and its fine-grained variety, pitchblende /14-2/. Samples of uraninite and pitchblende annealed at 1200°C in H_2, and untreated pitchblende were sequentially oxidized in air at 180-190°C, 230°C, and 300°C. All samples were analyzed by x-ray diffraction. Uraninite and untreated pitchblende oxidized to the U_4O_9-type oxide, and their x-ray symmetry remained isometric up to 300°C. Reduced pitchblende, after oxidation to UO_{2+x} and U_4O_9-type oxides, transformed into α-U_3O_8 at 300°C.

Two major mechanisms were found to control uraninite and untreated pitchblende stability during oxidation:

1) Th and/or lanthanide elements maintain charge balance and block oxygen interstitials near impurity cations;

Table 14-1. ICP-MS data correction routine illustrated with specimen 1 in series 3, 15th contact with DI-water.

XL\LDFILES\3X15

ICP-MS:DATA CORRECTION PROGRAMME. 49MWd/kgU LIBRARY — 49

SPECIMEN	DATE	RUN	BASED ON 100 ppb NATURAL In INTERNAL STANDARD !
3-1-15C	93-01-21	A259	

NUCL	MASS	COUNTS	CORRN	MAJOR	RATIO/In
Kr	82	81	2,10E+01	6,00E+01	7,80E-05
Kr	83	0	2,08E+01	-2,08E+01	-2,70E-05
Kr	84	103		1,03E+02	1,34E-04
Rb	85	12915		1,29E+04	1,68E-02
Kr	86	434	3,13E+01	4,03E+02	5,23E-04
Rb	87	27602		2,76E+04	3,59E-02
Sr	88	14515		1,45E+04	1,89E-02
Y	89	793		7,93E+02	1,03E-03
Sr Zr	90	11753	0,00E+00	1,18E+04	1,53E-02
Zr	91	16		1,60E+01	2,08E-05
Zr	92	105		1,05E+02	1,36E-04
Zr	93	0		0,00E+00	0,00E+00
Zr	94	72		7,20E+01	9,36E-05
Mo	95	15507		1,55E+04	2,02E-02
Mo Zr	96	1419	0,00E+00	1,42E+03	1,84E-03
Mo	97	18572		1,86E+04	2,41E-02
Mo	98	20005		2,00E+04	2,60E-02
Tc	99	1290058		1,29E+06	1,68E+00
Mo Ru	100	22640	6,82E+00	2,26E+04	2,94E-02
Ru	101	0		0,00E+00	0,00E+00
Ru	102	35		3,50E+01	4,55E-05
Rh	103	457		4,57E+02	5,94E-04
Pd Ru	104	27	2,54E+01	1,57E+00	2,04E-06
Pd	105	101		1,01E+02	1,31E-04
Pd	106	141		1,41E+02	1,83E-04
Pd	107	5		5,00E+00	6,50E-06
Pd	108	138		1,38E+02	1,79E-04
Ag	109	57		5,70E+01	7,41E-05
Cd Pd	110	6086	1,13E+00	6,08E+03	7,91E-03
Cd	111	3345		3,35E+03	4,35E-03
Cd	112	1905		1,91E+03	2,48E-03
Cd	114	2008		2,01E+03	2,61E-03
In	115	769562		7,70E+05	1,00E+00
U	119 2+	263		2,63E+02	3,42E-04
I	127	245		2,45E+02	3,18E-04
I Xe	129	1308	4,04E+02	9,04E+02	1,17E-03
Te Xe	130	105	6,28E+01	4,22E+01	5,48E-05
Xe	131	286	3,25E+02	-3,87E+01	-5,03E-05
Xe	132	412	4,12E+02	0,00E+00	0,00E+00
Cs	133	107213		1,07E+05	1,39E-01
Ba Xe Cs	134	1897	2,30E+03	-3,99E+02	-5,19E-04
Cs	135	42125		4,21E+04	5,47E-02
Ba Xe	136	629	1,36E+02	4,93E+02	6,40E-04
Cs Ba	137	94287	1,97E+03	9,23E+04	1,20E-01
Ba	138	10334		1,03E+04	1,34E-02
La	139	1360		1,36E+03	1,77E-03
Ce	140	86		8,60E+01	1,12E-04
Pr	141	110		1,10E+02	1,43E-04
Ce Nd	142	55	6,28E+00	4,87E+01	6,33E-05
Nd	143	101		1,01E+02	1,31E-04
Nd	144	194		1,94E+02	2,52E-04
Nd	145	131		1,31E+02	1,70E-04
Nd	146	118		1,18E+02	1,53E-04
Sm Nd	148	101	4,91E+01	5,19E+01	6,74E-05
Sm Nd	150	1	2,44E+01	-2,34E+01	-3,04E-05
Sm	152	26		2,60E+01	3,38E-05
Eu	153	8		8,00E+00	1,04E-05
Gd	156	55		5,50E+01	7,15E-05
Bi	209	635998		6,36E+05	8,26E-01
U	234	8		8,00E+00	1,04E-05
U	235	19		1,90E+01	2,47E-05
U	236	22		2,20E+01	2,86E-05
Np	237	5670		5,67E+03	7,37E-03
U	238			0,00E+00	0,00E+00
Pu	239	3318		3,32E+03	4,31E-03
Pu	240	1685		1,69E+03	2,19E-03
An	241	456		4,56E+02	5,93E-04
An	242	143		1,43E+02	1,86E-04
An	243	10		1,00E+01	1,30E-05
An	244	0		0,00E+00	0,00E+00

NUCL	InCorrF	ppb	a/o	CORRD ppb
Rb-85	0,710	**2,2621**	31,88	**2,2621**
Rb-87	0,710	**4,8345**	68,12	**4,7924**
Sr-88	0,839	**2,1514**	55,26	**1,6547**
Sr-90	0,839	**1,7420**	44,74	**1,7420**
Y-89	0,806	**0,1224**		**0,1224**
Zr-90	0,778	**0,0000**	0,00	
Zr-91	0,778	**0,0026**	8,29	
Zr-92	0,778	**0,0168**	54,40	
Zr-93	0,778	**0,0000**	0,00	**0,0000**
Zr-94	0,778	**0,0115**	37,31	
Zr-96	0,778	**0,0000**	0,00	
Mo-95	0,785	**2,4566**	19,85	**2,4566**
Mo-96	0,785	**0,2248**	1,82	**0,2248**
Mo-97	0,785	**2,9421**	23,77	**2,9421**
Mo-98	0,785	**3,1691**	25,60	**3,1691**
Mo-100	0,785	**3,5855**	28,97	**3,5855**
Tc-99	0,922	**173,9990**		**173,9990**
Ru-100	0,908	**0,0009**	10,14	**0,0009**
Ru-101	0,908	**0,0000**	0,00	**0,0000**
Ru-102	0,908	**0,0048**	52,05	**0,0048**
Ru-104	0,908	**0,0035**	37,82	**0,0035**
Rh-103	0,830	**0,0685**		**0,0685**
Pd-104	0,801	**0,0002**	0,40	**0,0002**
Pd-105	0,801	**0,0157**	26,05	**0,0157**
Pd-106	0,801	**0,0219**	36,37	**0,0219**
Pd-107	0,801	**0,0008**	1,29	**0,0008**
Pd-108	0,801	**0,0214**	35,60	**0,0214**
Pd-110	0,801	**0,0002**	0,29	**0,0002**
Ag-109	0,800	**0,0089**		**0,0089**
Cd-110	0,650	**1,1641**	45,60	**1,1641**
Cd-111	0,650	**0,6400**	25,07	**0,6400**
Cd-112	0,650	**0,3645**	14,28	**0,3645**
Cd-114	0,650	**0,3842**	15,05	**0,3842**
Te-130	0,500	**0,0105**		**0,0105**
I-127	0,200	**0,1523**	21,33	**0,1517**
I-129	0,200	**0,5619**	78,67	**0,5619**
Cs-133	0,834	**15,9864**	44,19	**15,9864**
Cs-134	0,834	**0,1477**	0,41	**0,1477**
Cs-135	0,834	**6,2812**	17,36	**6,2812**
Cs-137	0,834	**13,7651**	38,05	**13,7651**
Ba-134	0,951	**-0,0522**	-3,22	**-0,0522**
Ba-136	0,951	**0,0644**	3,97	**0,0644**
Ba-137	0,951	**0,2578**	15,90	**0,2578**
Ba-138	0,951	**1,3513**	83,35	**1,3513**
La-139	0,953	**0,1775**		**0,1775**
Ce-140	1,018	**0,0105**	63,83	**0,0105**
Ce-142	1,018	**0,0060**	36,17	**0,0060**
Pr-141	0,943	**0,0145**		**0,0145**
Nd-142	0,867	**0,0009**	1,01	**0,0009**
Nd-143	0,867	**0,0145**	16,19	**0,0145**
Nd-144	0,867	**0,0278**	31,10	**0,0278**
Nd-145	0,867	**0,0188**	21,00	**0,0188**
Nd-146	0,867	**0,0169**	18,92	**0,0169**
Nd-148	0,867	**0,0070**	7,88	**0,0070**
Nd-150	0,867	**0,0035**	3,91	**0,0035**
Sm-148	0,867	**0,0074**		**0,0074**
Sm-150	0,867	**-0,0034**		**-0,0034**
Sm-152	0,867	**0,0037**		**0,0037**
Eu-153	0,867	**0,0011**		**0,0011**
Gd-156	0,867	**0,0079**		**0,0079**
Bi-209	0,797	**99,2**		**99,2**
U-234	0,810	**0,001**	0,027	**0,001**
U-235	0,810	**0,003**	0,065	**0,003**
U-236	0,810	**0,003**	0,075	**0,003**
U-238	0,810	**0,0**		**0,0**
"U-119"	7,31E-03	**4,5**	99,832	**4,5**
Np-237	0,799	**0,8825**	Ratio to Pu-240	**0,8825**
Pu-239	0,630	**0,6549**	1,969	**0,6546**
Pu-240	0,630	**0,3326**	1,000	**0,3326**
An-241	0,630	**0,0900**	0,271	**0,0900**
An-242	0,630	**0,0282**	0,085	**0,0282**
An-243	0,630	**0,0020**	0,006	**0,0020**
An-244	0,630	**0,0000**	0,000	**0,0000**

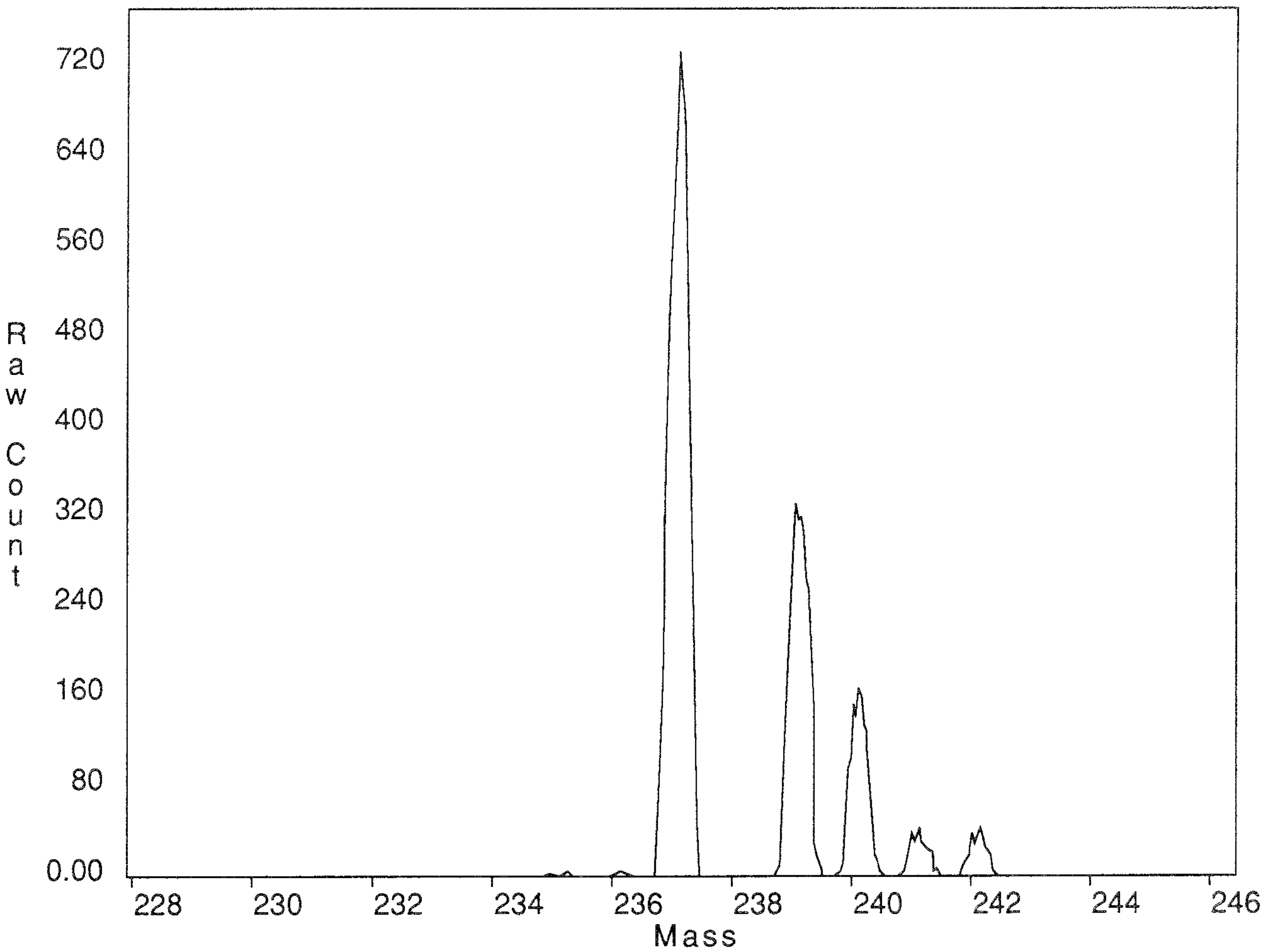


Figure 14-6. *Mass spectrum (actinide region) of Oskarshamn 1 reference fuel corroded in deionized water.*

2) the uraninite structure saturates with respect to excess oxygen and radiation-induced oxygen interstitials. Untreated pitchblende during oxidation behaved similarly to irradiated UO_2 in spent nuclear fuel; whereas, reduced pitchblende more closely resembled non-irradiated UO_2.

Based on the results of previous studies, attempts were also made to confirm the occurrence of U_3O_7 in naturally altered samples /14-2/. An analysis of the data in the literature, as well as experimental efforts (x-ray diffraction analysis, electron microprobe analysis, scanning electron microscopy and analytical electron microscopy) to identify U_3O_7 in samples from Cigar Lake, Canada, failed to provide conclusive evidence of the natural occurrence of tetragonal α-U_3O_7. Most probably, reported occurrences of U_3O_7 are mixtures of isometric uraninites of slightly different compositions.

Under oxidizing conditions, the higher oxides of UO_2 (e.g., U_3O_7 or U_3O_8) are unlikely for form in nature due to the presence of water which will cause the formation of uranium oxide hydrates.

When studying oxidative dissolution of UO_2, thermodynamic and kinetic data for relevant oxidized natural solid phases are needed. Dissolution experiments are being performed on a selected and well characterized natural samples of alteration chain of uraninite (i.e., uraninite, schoepite, uranophane). The experiments are performed using a synthetic granitic groundwater as a leachant, in contact with air at 25°C. An account of the experimental procedures together with some preliminary results are given in /14-3/.

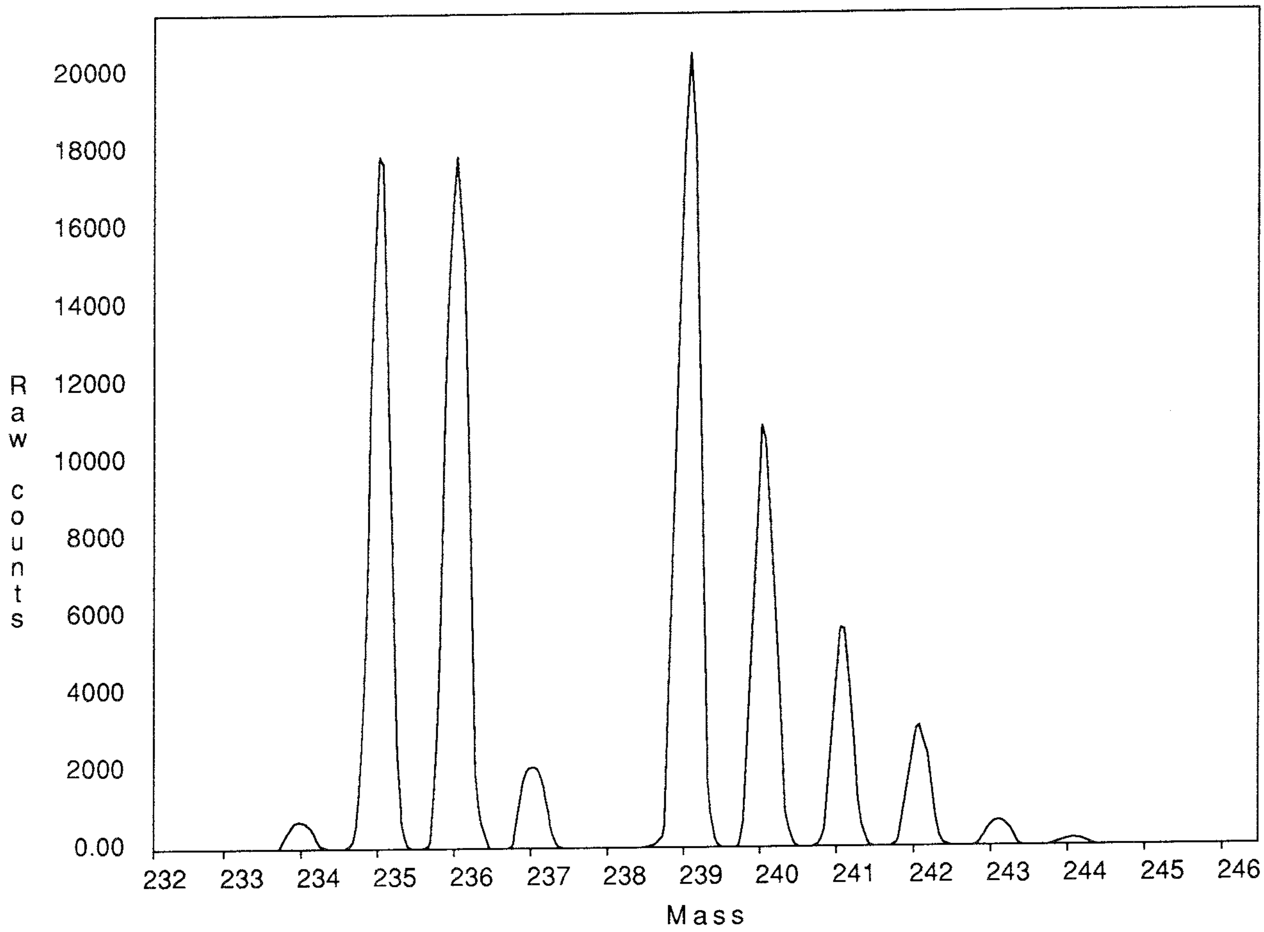


Figure 14-7. Mass spectrum of the actinide inventory of Oskarshamn 1 reference fuel.

15 CANISTERS

As for the past few years, the studies during 1992 have been focused on long-lived canisters with copper as the outer corrosion barrier. The studies have included investigations of the chemical and mechanical stability of copper and investigations of the corrosion of carbon steel under aerobic and anaerobic conditions.

The reference group for mechanical integrity of canisters, which was formed in 1990 has finalized its work during 1992 and the conclusion and recommendations of the reference group has been reported /15-1/ (see below). One of the conclusions of the reference group was that the copper-steel canister was a favourable alternative from the mechanical point of view. The main thrust of the work during most of 1992 has therefore been concentrated on this canister alternative. A copper-steel canister loaded with 12 BWR fuel bundles is shown in Figure 15-1.

15.1 MECHANICAL PROPERTIES

15.1.1 Creep

Creep tests have been carried out on oxygen-free high purity copper (Cu-OF), oxygen-free copper containing small amounts of phosphorous (50 ppm)(Cu-OFP) and oxygen-free copper containing 0.15% silver (Cu-OFS) at temperatures between 180 and 450°C /15-2/. Some Cu-OF batches exhibited poor ductility in the higher temperature range and ruptured at creep strains of less than 1% while another batch produced acceptable ductility values of about 10% elongation at fracture. The differences in ductility were attributed to a combination of sulphur content and grain size. However, the mechanisms

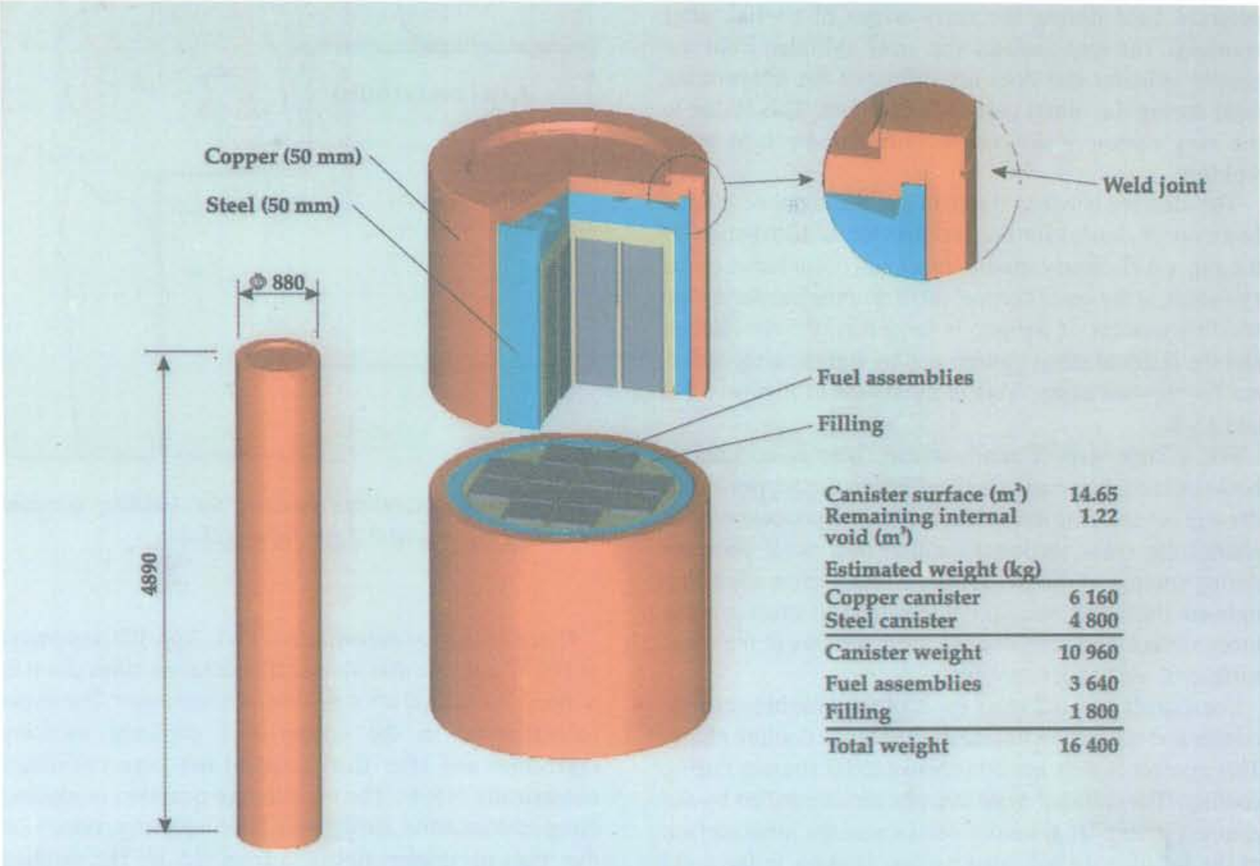


Figure 15-1. Principle design of the copper-steel canister for 12 BWR fuel elements.

behind the loss of creep ductility have not yet been fully established. Specimens of Cu-OFP and Cu-OFS ruptured at creep strains of 30% or greater. It was speculated that small additions of phosphorus or silver could increase the solid solution of sulphur in copper and therefore reduce the risk of grain boundary segregation and embrittlement or that an element like phosphorus co-segregates with sulphur and competes for grain boundary sites.

Stress relaxation tests have been performed on Cu-OFP to provide data for modelling the deformation behaviour of copper canisters during long-term service /15-3, App. III/. The stress relaxation tests at 75 to 150°C showed that the initial stress relaxation rate is rapid; the stress falls by at least 30% within the first 200 hours.

15.1.2 Thermo-mechanical analysis

In order to estimate the risk for creep fracture of the weld region during the design life of the canister, the temperature, strain and stress fields during welding and after cooling were calculated using the finite element code NIKE-2D /15-3/. The calculations were performed for two different gaps 2mm and 0.2mm, respectively, between the copper and the steel cylinders. The calculations showed that the gap between the copper and the steel components will have a small influence on the temperature field during the early stages of cooling after welding. The gap isolates the steel cylinder from the copper cylinder and does not influence the temperature field during the initial part of the cooling. This is due to the very narrow weld zone obtained in electron beam welding.

The clearing between the steel and the copper cylinder, however, is crucial for the residual stress distribution. If the gap is sufficiently small contact will occur between the cylinders, at the inner vertical surface of the lid, during the welding process. If the gap is large this will not happen and the residual stress pattern will be significantly different for the two cases. This is illustrated in Figures 15-2 and 15-3.

For a large gap (2 mm), where there is no contact between the copper and steel cylinders, the copper cylinder will act as a butt welded pipe. The stiff copper end will restrict the cross sectional rotation and radial decrease during cooling of the weld region. This creates a bending moment that is the resultant of tensile axial stresses at the inner surface and compressive axial stresses at the outer surface of the copper canister.

For a small gap (0.2 mm), the contact made between the copper and the steel will remain during the cooling phase. This contact results in compressive axial stresses during cooling. The residual axial stresses are controlled by the contacts giving large tensile values near the inner surface of the lid that are balanced by low stresses in the weld region. The accumulated effective plastic strain in the heat affected zone were found to be about 5%, which is much lower than the ductility of this copper alloy.

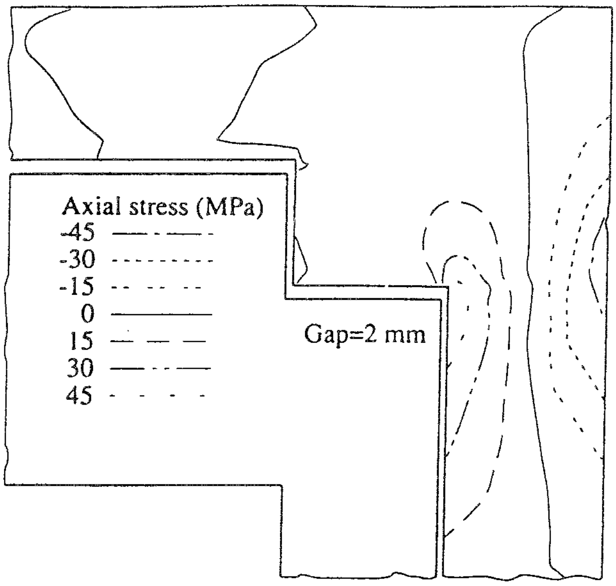


Figure 15-2. Calculated isolines for welding residual stresses for a gap of 2.0 mm (after /15-2/).

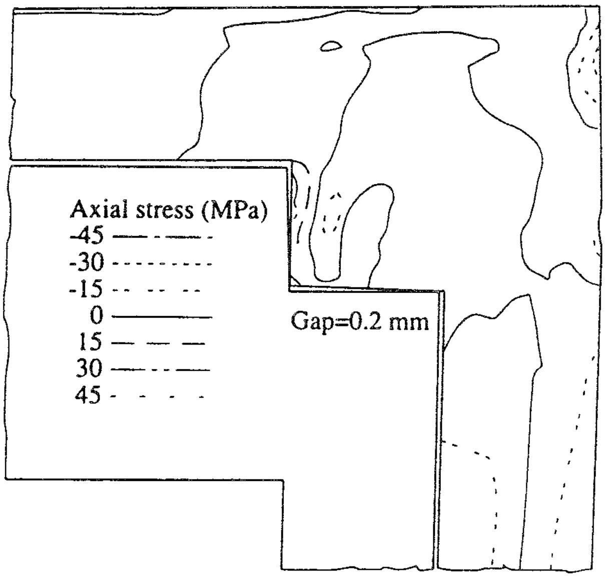


Figure 15-3. Calculated isolines for welding residual stresses for a gap of 0.2 mm (after /15-2/).

Creep relaxation experiments /15-1, App. III/ performed at 100°C indicate that almost all relaxation takes place in a short time period after closure of the canister. The stress redistribution in the copper-steel canisters awaiting deposition and after final disposal has been calculated numerically /15-4/. The constitutive equation modelling creep deformation during this time employs values on materials parameters obtained from /15-1/. The welding residual stresses are found to redistribute without lowering the maximum values during the waiting period. A very low amount of void growth is predicted for this type of the copper during the deposition period. Thus, a very long time to rupture can be estimated.

15.1.3 Tectonic movements

A rock shear movement perpendicular to the canister's axis could threaten its integrity. In order to illucidate the consequences of such an event stresses and strains in hot isostatically pressed (HIP) canisters as well as in copper-steel canisters have been calculated for a movement of the surrounding rock of 10 cm. The modelling of the bentonite in the deposition hole and of the copper is described in detail in /15-5/. In the HIP canister residual stresses in the order of 50 MPa occur along the cylindrical surface. These have been included as initial conditions for the tectonic calculations. For the copper-steel canister no residual stresses were modelled, since these are very localized to the weld region and are not expected to greatly affect the behaviour due to a rock movement.

It was concluded from the calculations that the effect of a rock shear movement depends more strongly on the density of the bentonite than on the canister type as can be seen in Table 15-1. As can be seen in the table, the maximum plastic strain in the copper after a 10 cm rock displacement at 1/4 distance from the end is 1.5% at the cylindrical surface for the HIP canister at a bentonite density $\rho_m = 2.05$ kg/dm^3. The lid is uneffected by the rock movement, but retains the 3% strain caused by the manufacturing. For the copper-steel canister the maximum strain at the cylindrical part is 2.5%. The corresponding value for a symmetrical shear movement is lower (1.2%). It can thus be concluded that a rock shear movement of the size considered poses no immediate threat to the integrity of the canister.

An attempt to include further creep deformation of the copper and bentonite after a rock shearing movement was also made. Due to very time-consuming calculations, the FEM runs could not be continued beyond 30 years after the shear event. A conservative extrapolation indicated that such deformation during a period of 10^5 years could lead to strains in the order of 6% on the surface of the cylindrical surface and 36% on the lid. It thus appears that the creep deformation under sustained rock shear can be a potential detrimental mechanism for the canister. Which importance that should be attached to to this fact is uncertain since the probability is very low that the canister has to survive 10^5 years after an such an unlikely event as a rock shear.

15.2 COPPER CORROSION

A review of the current knowledge of copper corrosion under repository relevant conditions has been presented during 1992 /15-6/. The main conclusions in this report are that general corrosion is very unlikely to lead to canister failure and that processes other than corrosion are more likely to endanger the integrity of the canister. Some corrosion processes, however, will require further attention. The most important ones being stress corrosion cracking and pitting corrosion under mildly oxidizing conditions. A preliminary estimate of length of the aerobic period indicate that mildly oxidizing conditions may prevail for some hundred years /15-6, App. 1/. During that period of time, pitting corrosion is possible /15-7/ and work aiming at modelling the pitting corrosion is in progress /15-8/.

Table 15-1. Maximum plastic strain in the copper at different processes and bentonite densities (ρ_m).

Canister type	Process	ρ_m kg/dm^3	Plast. strain % surface	Lid
HIP	HIP		0.3	3.0
	Swelling	2.05	0.0	0.0
	Shear 1/4	2.05	1.2	0.0
	Shear 1/2	1.93	0.0	0.0
	Shear 1/2	2.14	4.0	0.0
Cu/Fe	Swelling	2.05	0.2	0.4
	Shear 1/2	2.05	1.2	0.7
	Shear 1/4	2.05	2.5	1.0

16 BUFFER AND BACKFILL

16.1 CLAY ALTERATION

16.1.1 General

Two major studies on bentonite alteration were completed in 1992:

* Final analysis of MX-80 bentonite exposed to high temperatures with and without gamma radiation. This has been a joint project between SKB and CEA, France /16-1/.
* Investigation of a clay profile on southern Gotland of presumed value for documentation of smectite/illite conversion /16-2/.

16.1.2 MX-80

The purpose of the MX-80 study was to expose a 70 mm long water saturated sample of clay to a constant temperature gradient of 40°C (max. T=130°C, and min. T=90°C), with and without applying strong gamma radiation, during a period of one year under chemically open conditions (water in the sample in contact with outer water volume). The pressure of the weakly brackish Allard solution was maintained at 1.5 MPa throughout the tests, which were conducted at CEA's laboratory at Saclay in France. Detailed mineralogical and chemical analyses were then performed both in France and in Sweden. The physical properties before and after treatment were determined in Sweden.

The major mineralogical alterations were the disappearance of feldspars and formation of sulphates. The montmorillonite appeared to be largely unaffected except for slight dissolution and transformation to non-expansive minerals ("10 Å minerals") and possibly pseudo-chlorite. A small portion of the montmorillonite may have been converted to highcharge smectite like beidellite, yielding some free silica that contributed to the slight cementation that took place at the hot end. Hydrogen gas formed at the corrosion of the steel at the hot end may have displaced water and caused some drying. This may have partly isolated the steel from the clay porewater and retarded further corrosion and migration of iron in ionic form to the clay. Still, iron was found up to a distance of a centimetre from the steel plate and it is concluded to have caused strengthening and cementation effects by cation exchange and formation of amorphous oxy-hydroxide iron components and possibly also transformation of some montmorillonite to chloritetype minerals.

The physical properties were not significantly changed in either experiment. Thus, the hydraulic conductivity remained unchanged and the rheological behaviour underwent moderate or small changes. Some cementation was observed as demonstrated by creep testing and recording of the swelling ability of samples.

No significant difference was seen between samples exposed to radiation and those which were not. The direct effect of gamma radiation is known to be very small and none of the radiation-induced processes could be proven. One observation, however, was that some alterations had a tendency of taking place faster in the irradiated clay.

16.1.3 Gotland clay

Clay layers on southwestern Gotland are of presumed smectite origin (bentonite) and their geological history and location with respect to potassium-bearing water make them interesting as possible geological evidence of in-situ smectite to "illite" (hydrous mica) conversion. Core sampling to 75 m depth showed that rather thin clay layers are frequent down to about 65 m from the present sea level. A shallow part of the clay/siltstone/sand-stone sequence had been investigated at an earlier stage, indicating that the clay layers originated from ash falls, and the recent study supported the conclusion that they were initially bentonites.

All the layers were found to have almost identical mineral contents characterized by "illite", quartz and chlorite despite the smectitic microstructural morphology, see Figure 16-1. Estimating their probable initial smectite content from the present granulometry and clay content, and assuming that they were exposed to 80-100°C through their stratigraphic position in Devonian time, complete conversion to "illite" would be expected provided that potassium was supplied from the over- and underlying sediments. This is plausible since the same process of potassium migration by diffusion has been found to explain the partial conversion of the Hamra clay bed, which is located in the same area but 500 m deeper down.

16.2 MATERIAL MODELLING

16.2.1 General

The modelling of the physical properties of clay based buffer and backfill materials started in 1986 with a total stress model of the undrained properties and has since then been developed to comprise also effective stress, pore pressure and time-related strain. The material models that have been derived from laboratory tests are being applied to calculation programs like the finite element program ABAQUS for calculation of various functions and scenarios. Finally, the material models and calculation tools are being checked by verification tests

Fgure 16-1. Electron micrograph of sample from 8 m depth showing oriented stacks of flakes.

in the laboratory or in the field. The three different parts includes the following steps:

Material modelling:
- Undrained total stress.
- Drained total stress.
- Effective stress (including pore water pressure and water flow).
- Thermomechanics.
- Influence of temperature and pore water composition. *
- Influence of clay composition and additives. *
- Improved general material model. *
- Unsaturated materials. *

Verification tests:
- Rock shear tests.
- Canister displacements tests. *
- Clay swelling and compression tests. *
- Laboratory "standard" tests. *

Calculations (ABAQUS):
- Small scale laboratory tests. *
- Large scale verification tests. *
- Functions and scenarios. *
- Buffer Mass Test. *
- Alternative programs. *

* means that work is going on.

The work during 1992 and some important results will be summarized below.

16.2.2 Laboratory tests

The material modelling is primarily made from the results of tests with the following laboratory equipment:

Equipment	Test objects
Triaxial apparatus	
Compression/ swelling device	
Shear apparatus	Surface interaction
Oedometer	Swelling pressure and hydraulic conductivity
Creep apparatus	Creep properties

The tests can be made at different temperatures and drainage conditions.

16.2.3 Preliminary material model

A revised Drucker-Prager plasticity model based on the effective stress theory has been used so far. The model is illustrated in Figure 16-2. The relations between the average stress p and the Mises stress q at failure (failure surface) and at plastization (yield surface) are linear (right diagram). The stress space is divided into one elastic zone without limit at high stresses (no "cap" and no over-consolidation dependency) and one plastic zone between the yield surface and the failure surface. The elastic zone has been modelled as "porous elastic" which implies semi-logarithmic relation between the void ratio e and the average stress p and a linear relation between the Mises stress q and the shear strain ε (left diagram). The plastic zone is generally non-linear as shown in Figure 16-2. The parameters and their values are described in several reports /16-3, 16-4 and 16-5/.

16.2.4 Triaxial tests

During 1992 several triaxial tests and compression/swelling tests have been made, especially the following types:

- Drained tests on Na-bentonite MX-80 with high salt content in the pore water.
- Low pressure tests on Na-bentonite MX-80 and Ca-bentonite.
- Tests on Greek Ca-bentonite converted to Na-bentonite.

Figure 16-3 shows a compilation of the results of triaxial tests. The Mises stress q at failure is plotted as a function of the average effective stress p' for different materials at different salt content in the added pore water and at different temperatures. If otherwise not stated the tests refer to room temperature and distilled water.

The figure shows that the q-p relation for all types can be approximated by a straight line in the double logarithmic diagram yielding Eqn 1.

$$q=b\,(p/p_0)^a \tag{1}$$

where a is the inclination and b is the level of the curve

(q at $p=p_0$).

The test results also show that the q-ε relation is not linear and that the volume change during drained tests, at which the average stress is kept constant, is very small with a dilatancy (volume increase) of about 1% during the entire test.

16.2.5 Swelling/compression tests

The compression/swelling tests are also important for the material modelling. Figure 16-4 shows examples of oedometer tests with measurement of axial and radial pressure. All three samples have initially been compacted to the void ratio e=0.65. They have been water saturated at constant volume and then allowed to swell under stepwise reduced load until the axial effective stress reached about 100 kPa. The swelling was then followed by compression through corresponding stepwise reloading back to the effective stress of about 10 MPa. The actual time dependence is not shown in the figure, but the plotted values correspond to complete consolidation after a long time. The figure shows that the three different materials (Na-bentonite and Ca-bentonite with distilled water and

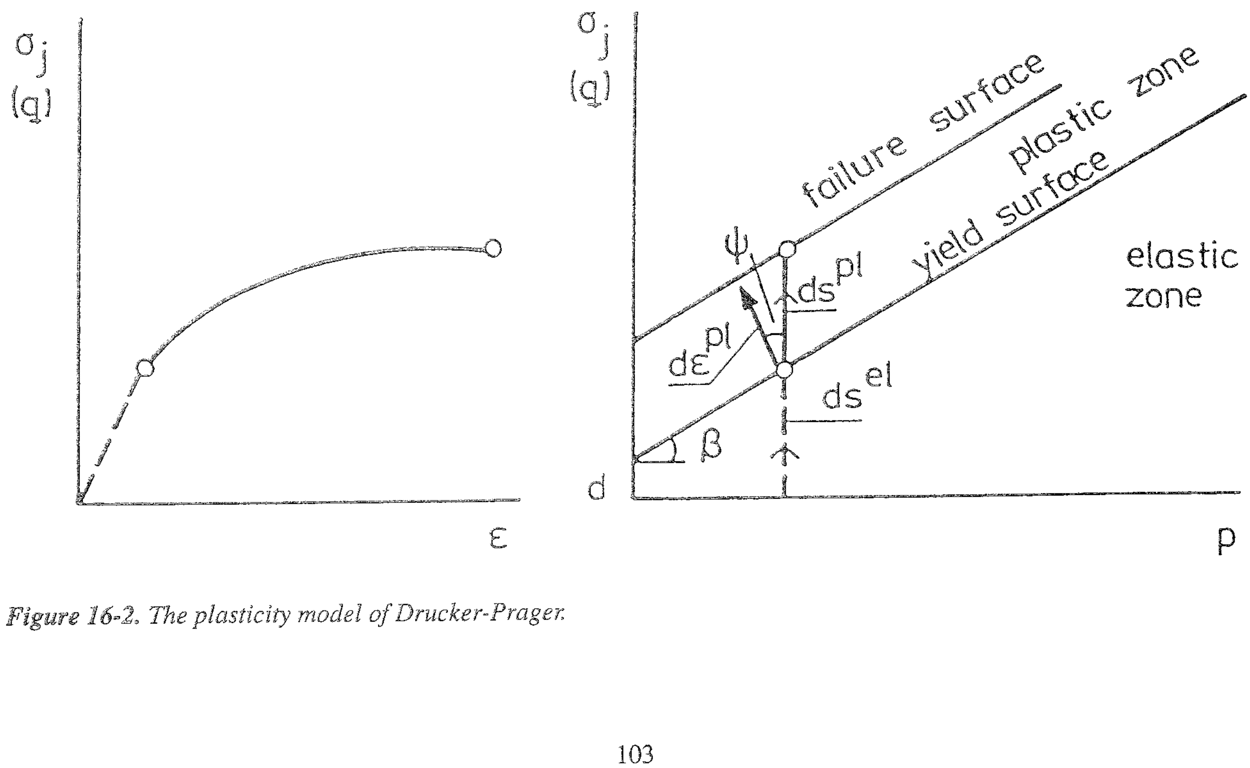


Figure 16-2. The plasticity model of Drucker-Prager.

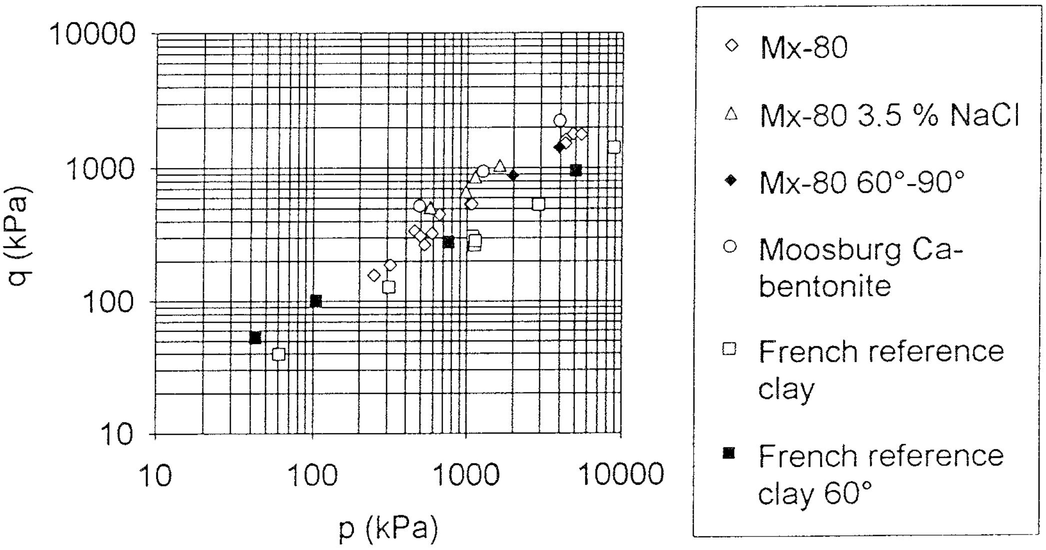


Figure 16-3. Compilation of failure stresses for different bentonites, temperatures and pore water compositions.

Na-bentonite with salt water) behave in a similar way but with the differences specified in Table 16-1.

Table 16-1. Influence of bentonite type and pore water on the measured pressure at compression and swelling. The radial pressure σ_r (similar at compression and swelling) and the axial pressure at swelling σ_{as} and at compression σ_{ac} are shown at the void ratio e=1.0.

Bentonite/water	e	σ_r kPa	σ_{as} kPa	σ_{ac} kPa
Na/dest.	1.0	2000	1050	3400
Ca/dest.	1.0	1100	320	1000
Na/3.5% salt	1.0	1500	400	1500

The tests show that

- the materials behave in a similar fashion with a hysteresis factor of about 3 for the axial pressure at swelling and compression, but without hysteresis for the radial pressure,
- there is a clear difference in pressure between the materials with 2-3 times higher pressure for the Na-bentonite with distilled water than the Na-bentonite with salt water and the Ca-bentonite.

16.2.6 Slot swelling tests

An extensive test series of bentonite swelling into slots or fractures has been performed. The tests were made by use of cylinders with 40 mm height and 50 mm diameter and with a 1-4 mm slot at mid-height of the cylinder. By letting water into the 90 mm diameter slot and into the filter stones at both ends of the sample, the bentonite has been allowed to swell radially into the slot. Figure 16-5 shows a photo of the sample and the extruded material in the 1 mm slot after removal of the upper part of the equipment.

The results show that at first a very soft gel penetrated into the slot, but with time, more and more material was extruded by which the soft material was compressed. The results are used for studying the penetration ability of different clay materials and as model verification tests.

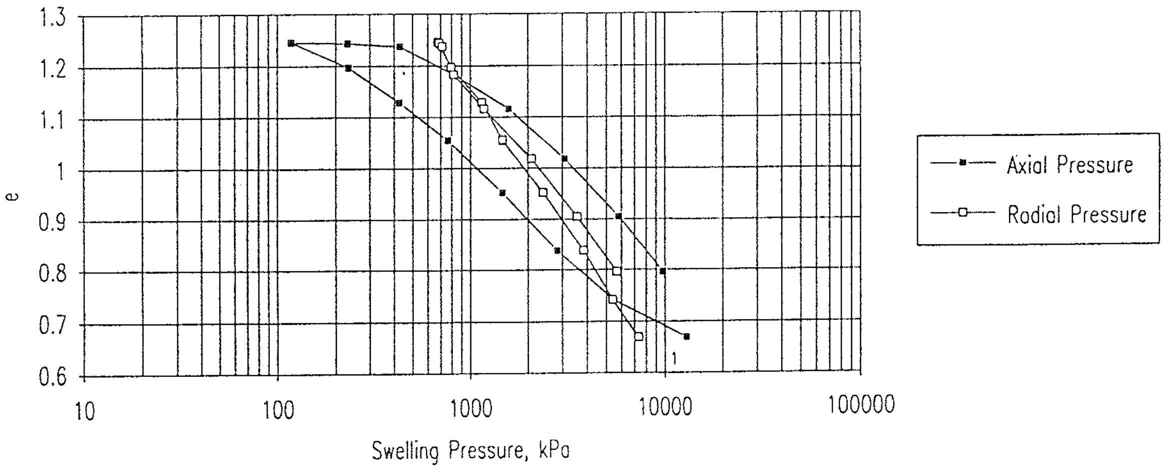
16.2.7 ABAQUS modelling

Functional material models are now at hand for the following materials:

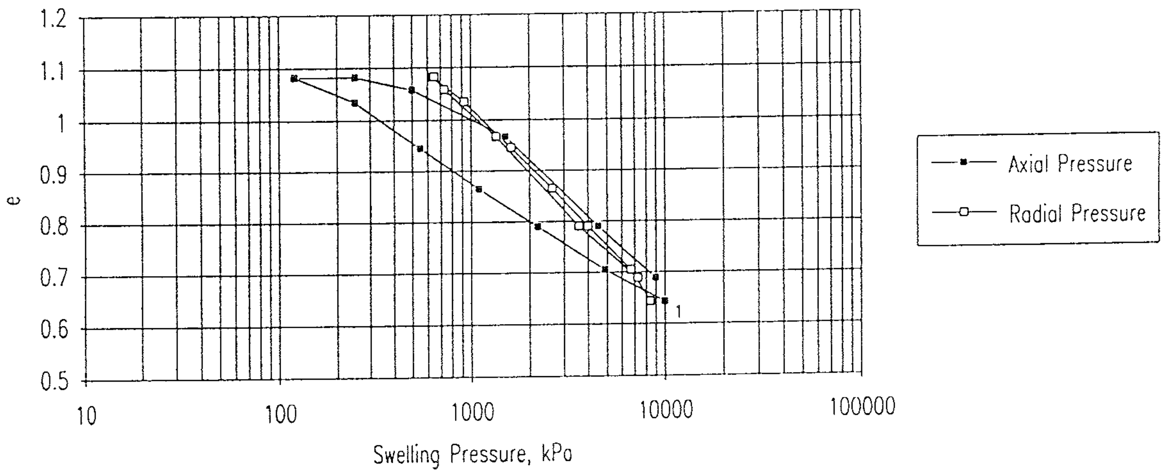
- Na-bentonite with and without salt in the pore water at temperatures between 5°C and 90°C.
- Ca-bentonite at the same conditions.
- French reference clay.

The models have been tested in several verification calculations as drained and undrained triaxial tests, swelling/compression tests and slot penetration tests as well as on a large scale in the Settlement test in Stripa. Two examples will be presented.

The swelling/compression tests have been modelled as an axisymmetric element mesh with a radial symmetry plane through the centre of the sample. The radial boundary was modelled by frictional elements that allow axial but not radial deformation. Axial drainage through the end face was also modelled.



Expansion-Compression cycle. Moosburg.



Expansion-Compression cycle.Mx-80

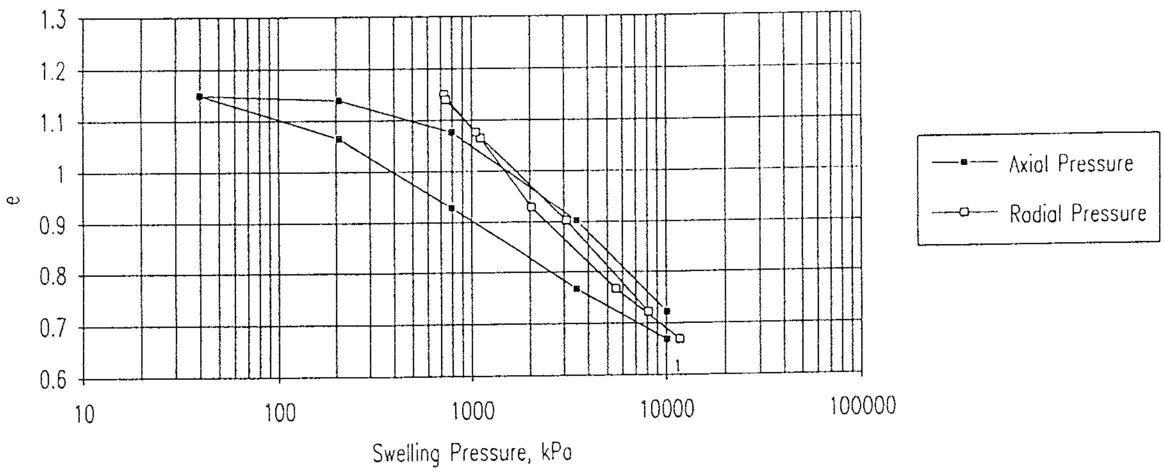


Figure 16-4. *Results of swelling/compression tests made on the following materials (from above)*

- *Na-bentonite (MX-80) saturated with distilled water.*
- *Ca-bentonite saturated with distilled water.*
- *Na-bentonite (MX-80) saturated with 3.5% salt water.*

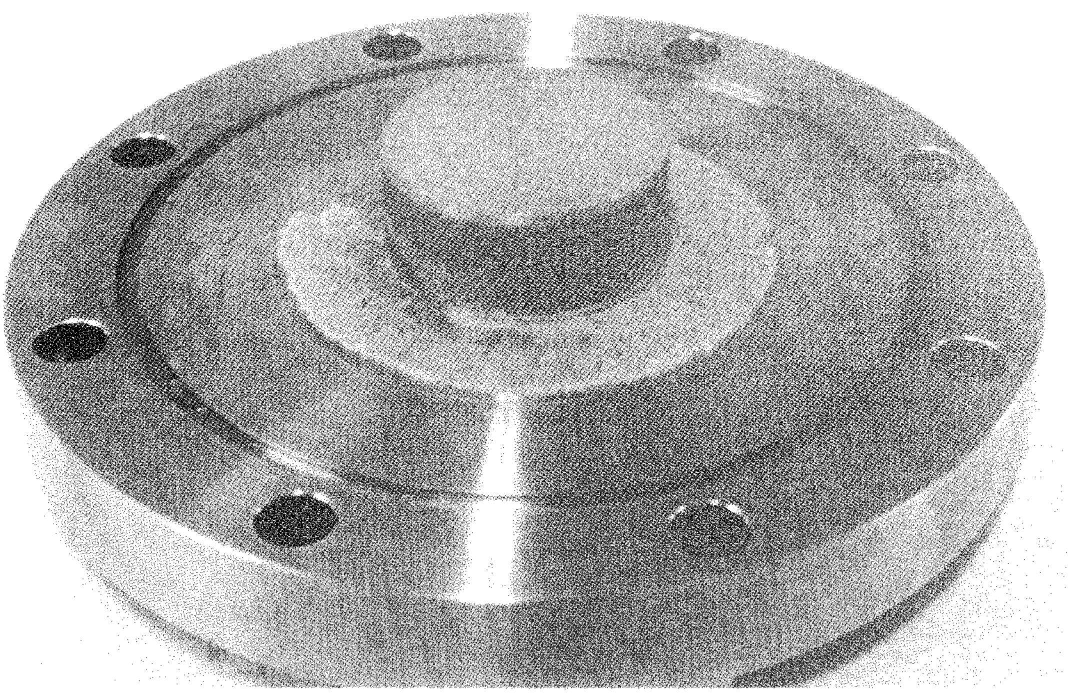


Figure 16-5. Photo of a sample with bentonite that has swelled into a slot with 1 mm aperture.

The calculation was made with the initial void ratio of the bentonite of e=0.74 and the swelling pressure 5 000 kPa, applying the derived material model with Drucker-Prager plasticity and porous elasticity. The calculation corresponds to two tests with initial unloading and swelling until e=1.25 and then reloading and compression. Figure 16-6 shows a comparison between the calculated and measured radial and axial pressure as a function of the void ratio. The figure illustrates the following:

- The measured and calculated radial pressures at unloading as well as reloading agree very well.
- The measured and calculated axial pressures at unloading agree fairly well except at very low pressure.
- The measured and calculated axial pressures at reloading do not agree very well. The calculated hysteresis effect is much smaller than the measured one.

The **slot penetration tests** have been simulated with a similar element mesh as the swelling/compression tests but with a radial slot in the middle of the sample. The horizontal symmetry plane intersects the centre of the slot and water is available at the end face and at the swelling front in the slot.

The calculation is complicated since the large swelling makes the elements deform strongly. Since the program cannot handle this, 4 new element meshes had to be introduced during the calculation (remeshing).

Some results of the calculation at different time after test start are shown in Figure 16-7 and Figure 16-8. The figures show the calculated void ratio after 3, 9, 23 and 90 days. One finds that the void ratio is very high at the front (e=2.2–2.8) until the penetration has reached the end of the slot after about 25 days and that the penetration is followed by a compression with decreasing void ratio in accordance with the measurements. However, comparison with the measurements also shows the following:

- The measured penetration is much faster than the calculated one.
- The measured void ratio at the front is much higher than the calculated one.
- After 90 days the measured void ratio gradient in the slot is higher than the calculated gradient but the measured and calculated average void ratio in the entire slot are the same.

The model can obviously not simulate the very soft gel that swells quickly in the beginning of the test while it otherwise yields good results. The main reason for this is that the model has a linear failure envelope with a cohesion that results in too high shear resistance at low stresses while the real failure envelope is curved without cohesion.

16.2.8 Concluding remarks

The tests and calculations have shown that there is a functioning material model for different water saturated bentonites with different pore water at different temperatures between 5°C and 90°C that can be used for all kinds of calculations of physical processes including water

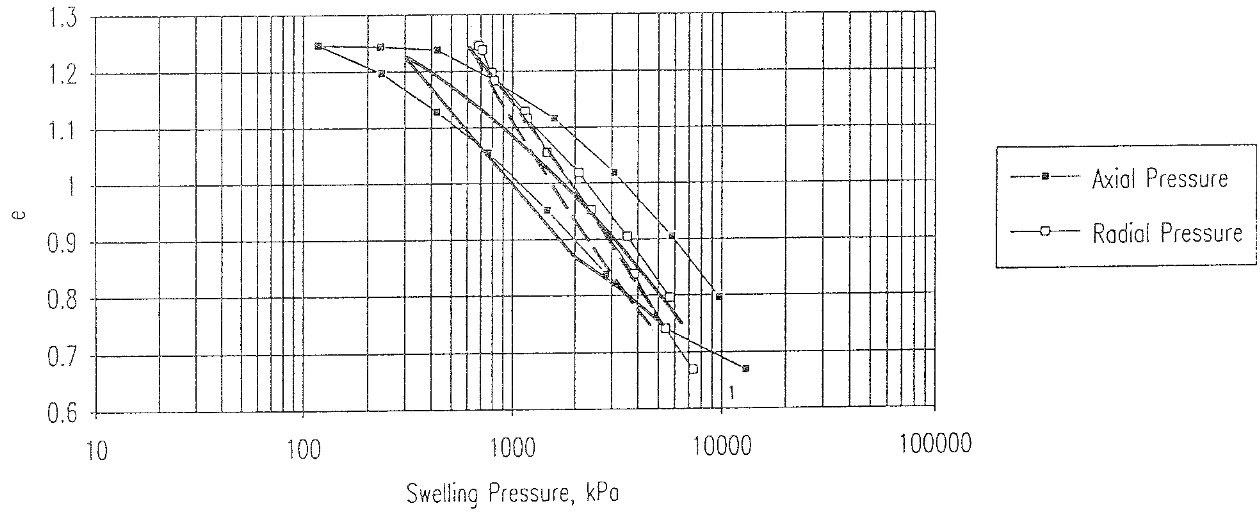


Figure 16-6. Comparison between calculated and measured axial and radial pressure at a swelling/compression test.

______ Calculated axial pressure.
\- - - - - Calculated radial pressure.

flow and thermomechanics. However, the tests also show that the material model has some flaws and an improved version is aimed at with the following main changes:

- The failure envelope is curved according to the relation in Eqn 1.
- A cap is added to the model with a yield surface at a defined average stress that is equal to the highest average stress applied to the bentonite. At lower stresses the deformation is elastic and at higher stresses the material will plasticize.
- The stress-strain behaviour on shearing is non-linear and the material may be strain softening after failure.

The other parts of the behaviour with e.g. the porous elastic *p-e* behaviour, the effective stress theory and the small dilatancy at failure seem to model the properties of the bentonite based buffer material very well.

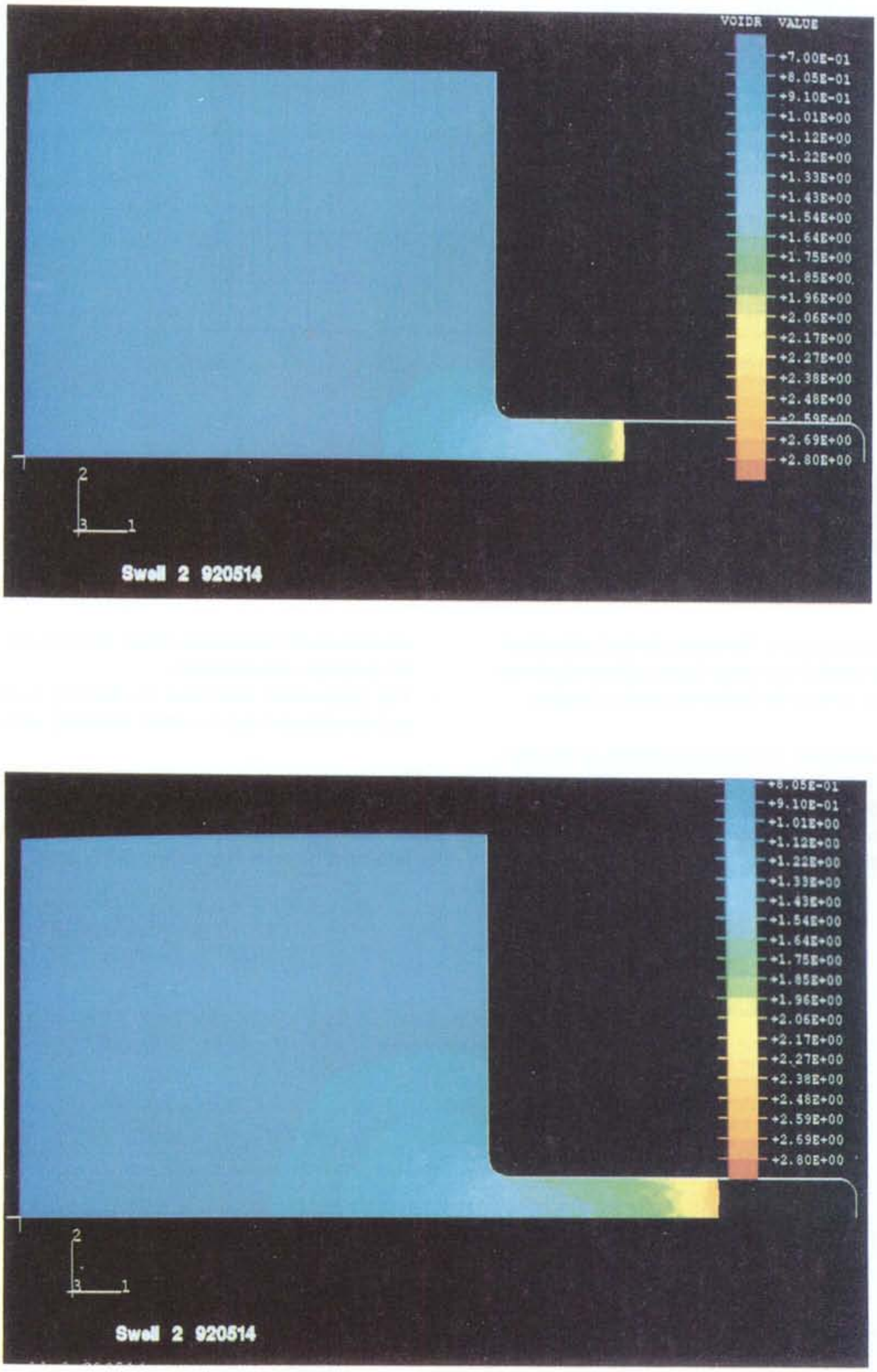


Figure 16-7. *Calculated void ratio at the slot penetration test after 3 (upper) and 9 days.*

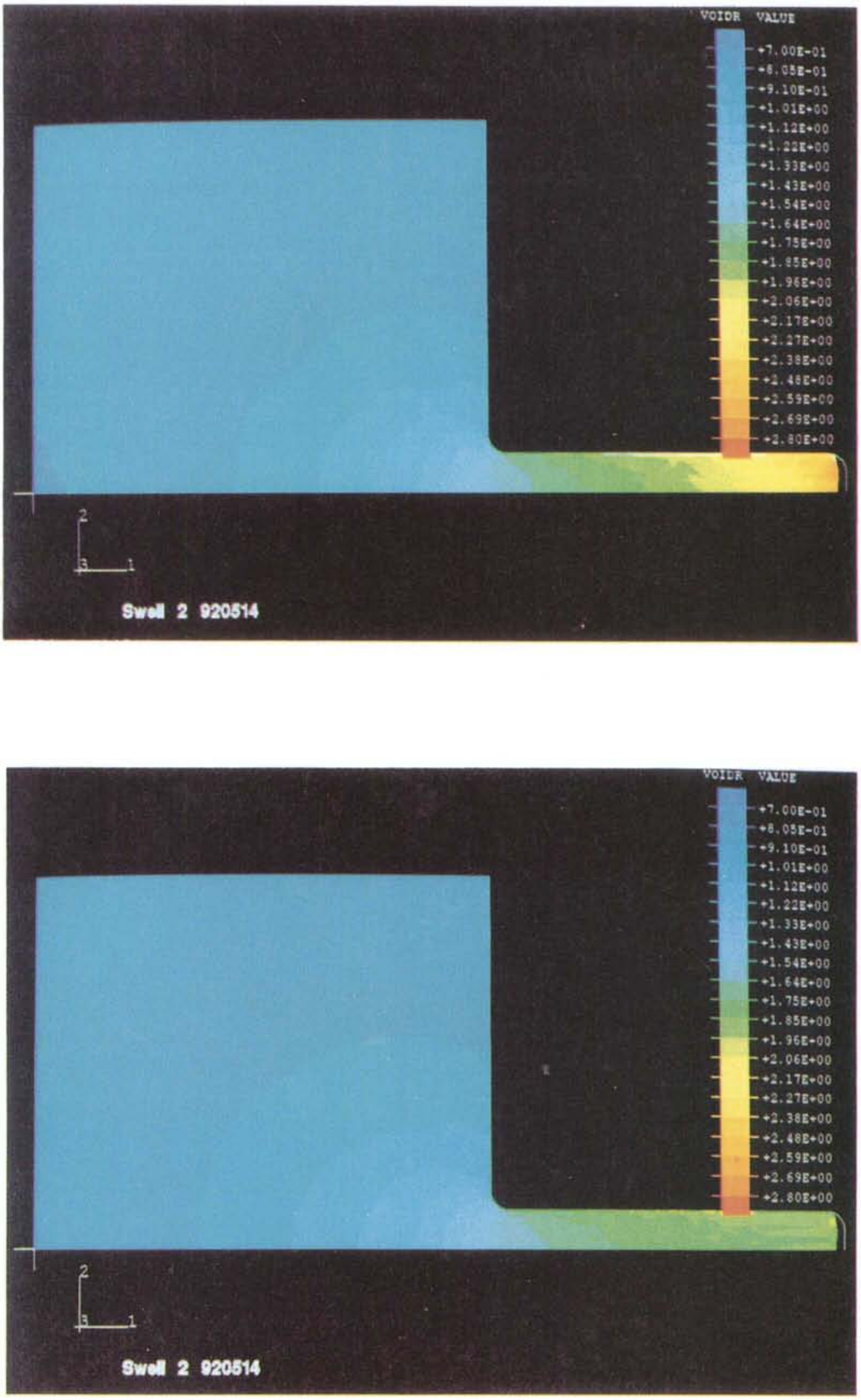


Figure 16-8. *Calculated void ratio at the slot penetration test after 23 (upper) and 90 days.*

17 GEOSCIENCE

17.1 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. The reducing chemistry of the groundwater is of great importance for the life of the canister and for the slow dissolution of the fuel matrix. Groundwater chemistry is determined for the most part by the mineral composition of the rock, which is 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 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 nuclides in the rock. The water flux is generally determined by the topography of the ground surface and by the permeability of the rock, which is in turn dependent on its fracture content.

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

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

- The Stripa Project.
- The Äspö Hard Rock Laboratory.
- Project Alternative System Studies (PASS).
- Safety assessments (e.g. SKB 91).
- Natural analogues.
- The siting programme.

The overall objectives and main activities of the geoscience programme 1987-1992 are expressed in the SKB R&D-Programme 86 and in the current programme 1990–1996 that was released in September 1989. During 1992 the geoscience programme has involved the following main tasks:

- Groundwater Movements in Rock.
- Bedrock Stability.
- Geological Overviews in Repository Siting.
- The Laxemar Deep Drilling Project.
- Development of Instruments and Methods.

17.2 GROUNDWATER MOVEMENTS IN ROCK

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 and the driving forces. The importance of small density contrast for the overall groundwater flow distribution has been recognized.

The conceptualization of the groundwater flow distribution is important for the overall assessment of radionuclide transport, both non-sorbing and sorbing. The hypothesis that groundwater flow occurs in more or less dependent channels or pathways needs thorough studies.

17.2.1 Analysis of well test data

The in-situ determination methodology of permeability and storage properties in fractured rock is an essential topic. The analysis techniques have been improved through the Stripa and Äspö projects, particularly when it comes to interference tests. The application of probabilistic models to infer the hydraulic properties of fractures is described in /17-1/. The development has continued and new projects have commenced on the subject.

17.2.2 Coupled processes

Thermal, hydrological and mechanical processes might have a mutual influence on each other to a greater or lesser extent and affect the rock mass behaviour, especially the nearfield surrounding the canisters.

In this context SKB has initiated a three year programme for a better understanding of the detailed flow regimes within fractures or fracture intersections. The programme will mainly be done by means of experiments on a laboratory scale. The research tasks include collection of input data for calculation models and development of measurement methods for fracture characterization and porosity.

In the DECOVALEX project initiated by SKI (international cooperative project for the DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation), development and verification of coupled thermo-hydro-mechanical models is taking place. SKB is a member of the Steering Committee of the DECOVALEX. On behalf of SKB a single fracture test case on a coupled stress-flow problem has been modelled using the UDEC-code /17-2/. Within the DECOVALEX project SKB emphasizes the analytical approaches for a better understanding of the calculation results and their dependence on boundary conditions and dimensionality /17-3/.

17.2.3 Numerical modelling efforts

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

It would be an advantage and hopefully a greater degree of realism would be achieved if hydraulic (hard) input data are complemented with geoscientific (soft) information. An attempt on a flow field problem showing this methodology was published during 1992. A stochastic approach when using a synthetic data set compared the effect of hard and soft conditioning. The specific conclusions of the study are that conditioning on soft data reduces the uncertainty of solute arrival time and there is also an indication that the soft data gives an improvement in characterizing channelling effects, although the later statement requires further studies /17-4/.

Besides regional groundwater modelling under today's climatic situation, it is essential to shed light on the hydraulic conditions in connection with future glaciations and deglaciations. During the ongoing three year programme a time dependent glaciation model of Scandinavia has been developed /17-5/ for the coming 120 000 years. A surface boundary condition for loading and groundwater flow, including the distribution in time and space of glacial, permafrost and temperate/boreal conditions, glacier loading, glacially imposed shear stresses, glacier- and permafrost derived meltwater fluxes is presented.

A continuation is now aimed at predicting the hydraulic regime in a linear swathe of terrain, approximately parallel to an average ice sheet flow line, over the next 120 ka. Furthermore the work will include testing the validity of the climatic and geo-hydrographical model. The swathe of terrain is represented as a 2-dimensional broad-path transect (about 50 by 600 km) along the line Stripa-Äspö. A 2D groundwater flow model, developed at the University of Edinburgh, is applied to the selected transect, in order to evaluate the effects of variations in a number of properties and boundary conditions. The project comprises scoping calculations to identify reasonable ranges of boundary conditions for the geohydraulic model and to determine sensitivities.

A comprehensive literature survey of relevant data for the selected transect has been carried out. The data collected comprises topography, lithology and drift, groundwater chemistry, geothermal properties, rock mechanical data and lake statistics. A general conceptual model for the distribution of hydraulic conductivity is proposed and quantitative models for different lithological units along the transect are developed on basis of available hydraulic test data. A model for the dependence of hydraulic conductivity on effective stress is chosen and the effect of an overlying ice sheet is briefly investigated /17-6/.

SKB supports modelling efforts at different scales. During the last decade much of scientific and civil engineering interest have been devoted to the behaviour of the disturbed zone surrounding a drift or tunnel in crystalline rock. From hydraulic point of view it is now a well-known phenomenon that the average hydraulic conductivity decreases nearby the tunnel walls. Several explanations are discussed in literature and one of these considers that two-phase flow conditions occur. SKB has supported a modelling exercise which studies the two-phase flow regimes in the disturbed zone.

17.2.4 The implication of fractal dimension

A fractal, or a fractal set, is "a shape made of parts similar to the whole in some way". A fractal set may be described by its "dimension", which is an indication of the degree of heterogeneity or roughness of the shape. Fractal systems have similar geometries at different scales, such that measurements made at one scale can be used to predict geometries at other scales.

Since much of geology and hydrogeology is controlled by the geometry of geologic features such as faults, fractures, and stratigraphy, many researchers have proposed the use of fractal dimension as an index for replicating geologically realistic fracture patterns.

An extensive literature survey /17-9/ was carried out on the application of fractals for comparison of hydrogeologic environments. Although no applications were identified that directly applied fractals as a hydrogeologic index, personal contacts in the oil industry indicated that fractal dimension is used to select locations for production and exploration wells. Extensive references were found to empirical studies of the fractal nature of fracture geometry and fractured rock hydrogeology. These papers demonstrated that consistent and meaningful fractal dimensions can be derived from lineaments, fracture maps, and fracture surfaces, and that the fractal dimension can be used to compare different geologic geometries. A further significant indication of the potential usefulness of fractal indices is the relationship found between rock block size distributions and the fractal dimension for a wide variety of fracture patterns cited in the literature.

Numerical simulations carried out with the FracMan model /17-10/ indicate that fractal dimension appears to be a useful index for fracture connectivity and block formation. Since fracture connectivity is strongly related to both large scale and small scale radionuclide transport, this indicates potential usefulness of the fractal dimension as an index for site comparison. Surprisingly, fractal dimension seems to have as strong an effect on connectivity and block formation as fracture intensity, at least within the range simulated.

Based upon both the literature survey, and numerical simulations, it appears that fractal dimension can be used to distinguish geologic environments. However, further study will be required to determine which values of fractal dimension are preferable for repository location in particular geological environments.

17.3 BEDROCK STABILITY

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

The objectives are to:

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

17.3.1 Tectonics and seismic activity

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

A brief resumé of the studies conducted at SKB during the period 1992 is provided here. See also SKB's Annual Reports /17-11,17-12/.

It is essential to have an understanding of the brittle tectonic evolution of the Baltic Shield in order to be able to describe the fracture systems and previous movements in the Swedish bedrock. The stress situations that have arisen during continental drift, "the paleostress field", can be roughly analyzed based on the presence of different types of hypoabyssal rocks, fracture sealings, fracture minerals etc. and related to different time periods. It is further possible based on isotope data to indicate former vertical loads from sedimentary strata that have eroded away. Successive glaciations have also affected the load situation in the Swedish crystalline basement. The Swedish bedrock has thus in all likelihood been subjected to forces in all directions. During the current tectonic regime, spreading from the Mid-Atlantic Ridge, there are therefore no signs that new fractures or zones will appear in the upper part of the crust in Sweden. If movements take place in the earth's crust in Sweden, they will accordingly take place as reactivations in existing fracture zones or faults. A synopsis of the brittle tectonics of southern Sweden is presented in /17-13/. Some other regional studies funded by SKB exhibit similar results with movement activation /17-14/, /17-15/, /17-16/, see Figure 17-1.

The seismic activity in the Baltic Shield is mainly controlled by the plate-tectonic processes and ongoing land uplift. The results of seismic measurements /17-17, 17-18, 17-19/ show that most of the stress that causes earthquakes has a compression direction of N60W, which is approximately perpendicular to the continental movement from the Mid-Atlantic Ridge. The globe-spanning database "World Stress Map Project" also shows relatively good agreement between the plate tectonics of the earth's crust and the greatest horizontal principal stresses, i.e. compressions. Certain deviations occur in the stress field in the Baltic Shield, which can be explained as effects of the glaciation-uplift processes /17-20/.

If the deglaciation is relatively rapid, the earth's crust is subjected to regional stress differences which can trigger movements in already existing weakness structures. The deglaciation phase following the most recent glaciation was significantly faster in northern Sweden than in the southern parts of the country. This is considered to be an important cause of the neotectonic and postglacial movements that have been interpreted in, for example, the Lansjärv area /17-20/.

17.3.2 Glaciation scenario

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 purpose was to describe when future ice ages can be expected and what changes in the geosphere occur in connection with them /17-21, /17-22/, /17-23/.

As mentioned above further refinement of the dynamics in a future glaciation scenario has taken place with the development of a time-dependent model of the glaciation in Scandinavia. The model has been calibrated with our knowledge of the Weichsel glaciation and then with re-

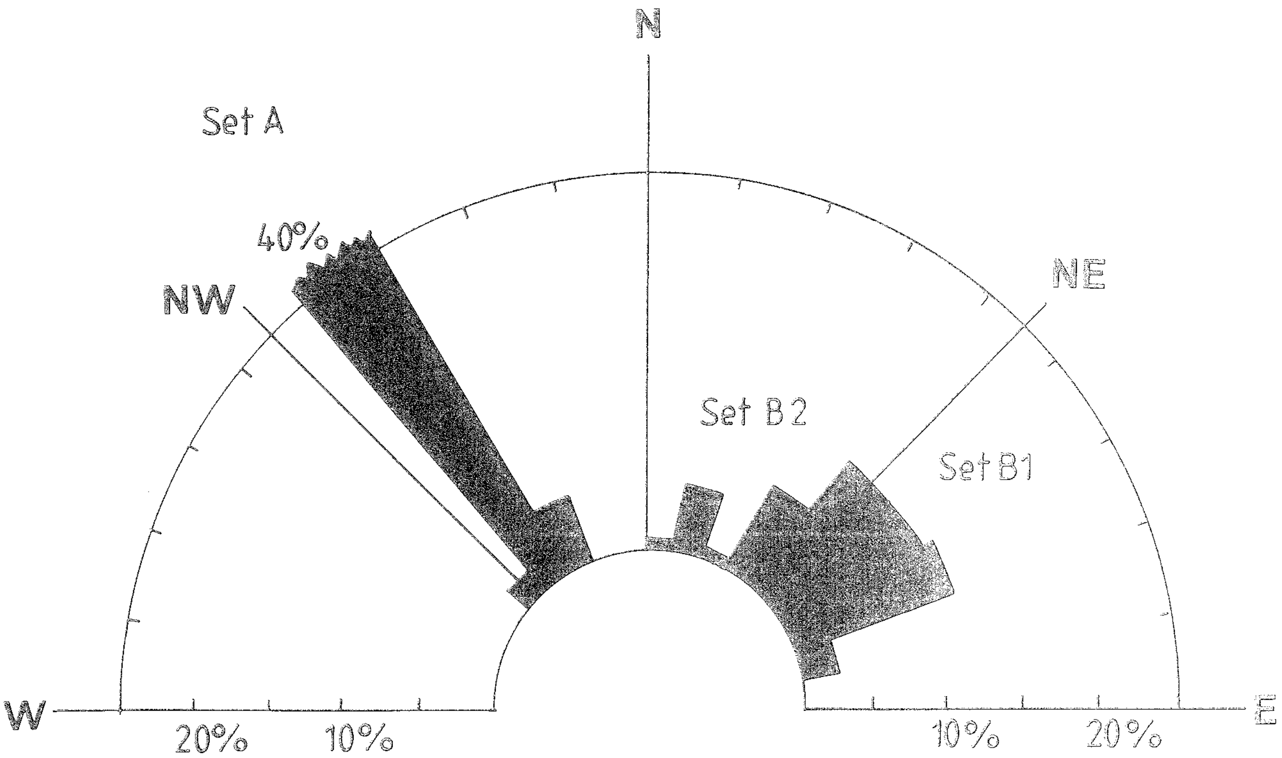


Figure. 17-1. Example of fracture mapping results from Hornsudden, Öland. Air photo of ground control area and summary of fracture orientation statistics from scanline and ground control data. (From ref /17-16/)

spect to moraine structures, soil thicknesses, erosion traces etc. /17-5/.

Based on future glaciation scenarios, the impact on a repository can be judged with different types of calculation models. To begin with, the stability within a repository area has been analyzed with respect to an ice load and simultaneous water pressures. A sensitivity study has also been carried out. The work has been exemplified with data from the Finnsjön study site. In summary, the results say that the movements will be taken up in already existing zones. The total peak relative movement in a zone within the area amounts in the normal case to about 0.05 m. If an extreme situation prevails with low in-situ stresses in the rock, the importance of the pore pressure would increase, resulting in a total movement of about 0.5 m. Current repository concepts within SKB assume that canisters will not be deposited in such zones /17-24, 17-25/.

17.3.3 Deglaciation rebound

The land uplift after the most recent glaciation has normally been studied in relation to sea level. It is, however, possible to study the total movement and distribution of the land uplift via the tilting and "rebound" of oblong lakes. By dating sedimentation levels at different depths, it is possible to analyze the general land uplift within a region. SKB has initiated studies on several sites in the country in order to obtain better knowledge of the distribution of the land uplift /17-26/. Based on Quaternary geological studies it is also possible to analyze any shoreline displacements, which could be a sign of postglacial movements. Such studies are being conducted in Värmland, the results of which will be published in 1993.

17.3.4 Fault dating methods

During the past few years new dating techniques and new methods of investigating geological structures have implied research into the direct dating of faults. SKB has initialized a project on this subject. During 1991 and 1992 the sampling activities were undertaken within a comparative study on different dating techniques. Specimens from a fault in the access ramp to the Äspö Hard Rock Laboratory are to be analyzed. Petrographic, Palaeomagnetic, Electron Spin Resonance (ESR) and Isotope techniques will be employed in attempts to assess the age of the most recent movements on the fault. The study will be reported during 1993 /17-27/.

17.4 GEOLOGICAL OVERVIEWS IN REPOSITORY SITING

During 1992 SKB organized the framework for the siting process with all its technical, administrative and political decisions. To some extent geological activities were performed within the Siting project (see Chapter 13). However, a few reports closely related to the siting tasks were elaborated within the general SKB Geoscience programme.

17.4.1 Compilation of study sites

During the period from 1977-1986 SKB performed surface and borehole investigations of 14 study sites for the purpose of assessing their suitability for a repository of spent nuclear fuel. The next phase in the SKB site selection programme will be to perform detailed characterization, including characterization from shafts and/or tunnels, of one or two sites. The detailed investigations will continue over several years to provide all the data needed for a licensing application to build a repository.

It is presently not clear if anyone of the Study sites will be selected as a site for detailed characterization. Other sites with geological an/or socioeconomical characteristics judged more favourable may very well be selected. However, as a part of the background documentation needed for the site selection studies to come, summary reports have been prepared for eight Study Sites. These reports include scope of activities, main results, uncertainties and proposed need of complementary investigations. The six compiled study sites are: Gideå, Fjällveden, Sternö, Kamlunge, Klipperås and Finnsjön. During 1992 Technical Reports on Sternö, Kamlunge, Klipperås and Finnsjön were published /17-28/, /17-29/, /17-30/, /17-31/, see Figure 17-2.

17.4.2 Gabbro as a host rock for a repository

The Swedish nuclear waste management programme has focused on granite and gneiss as the major candidate host media for a repository for spent nuclear fuel. Mafic rock types, in particular gabbro, has been suggested as alternatives, and the potential suitability of gabbro with respect to nuclear waste disposal has been studied and discussed at several occations over the past twelve years.

A study has been elaborated which summarizes and examines existing geoscientific knowledge of relevance in assessing the potential merits of gabbro as a repository host rock. Implications in terms of site selection, repository construction and post-closure repository performance are also discussed /17-32/.

In the first part of the report, an overview is given of the sources of data available for the present study. A description of the scope of research and main results of previous investigations relating to gabbro, and conducted within the Swedish radwaste programme, is also included.

The main part of the report compiles existing knowledge on the geological, geohydrological, geochemical and geomechanical properties of gabbro. This part is based mainly on three recently completed supporting studies, more narrow in scope than the present work, by penetrating deeper into the data available and matters of

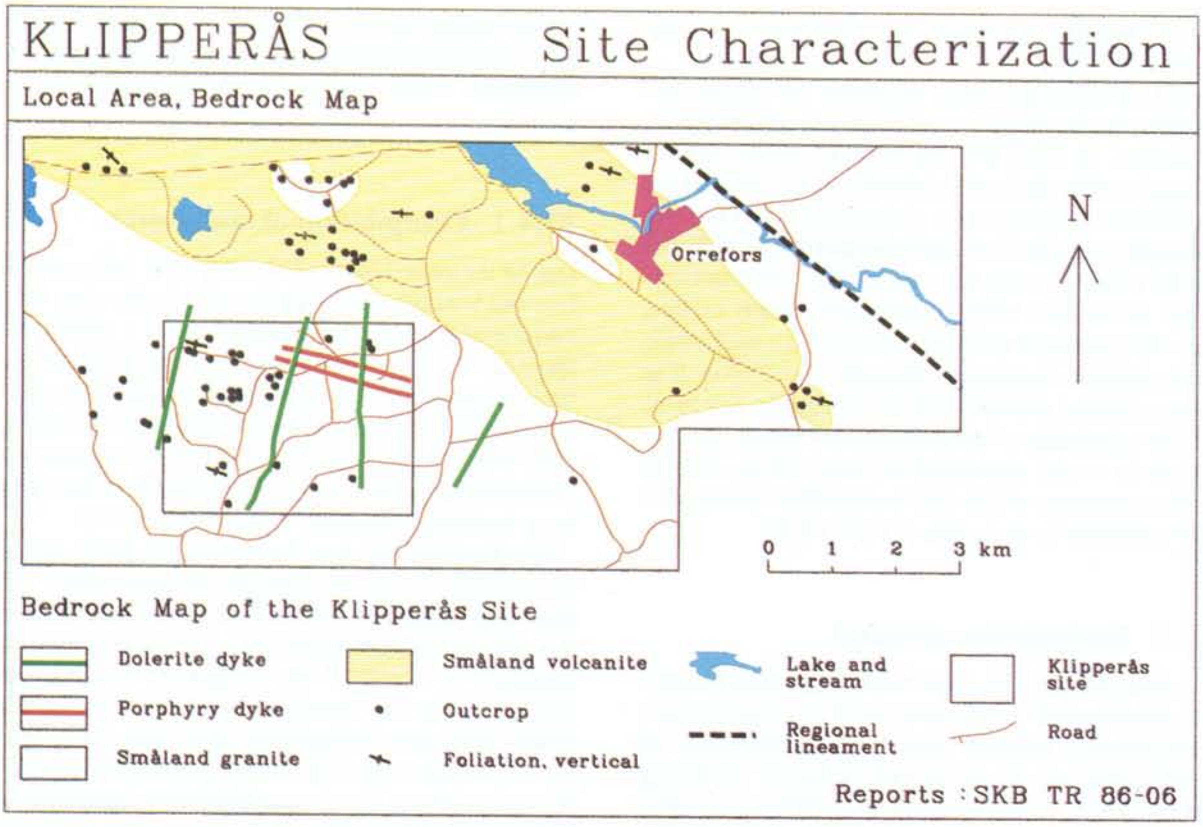


Figure 17-2. Generalized map of the bedrock at the Klipperås Study Site and its surroundings. (From ref. /17-30/)

concern within the different geoscientific disciplines. The presentation also draws upon a study conducted by SKB in cooperation with TVO – the organization managing the Finnish radwaste programme /17-33/, /17-34/, /17-35/, /17-36/.

With respect to the feasibility of selecting a site for a repository, there are obvious disadvantages associated with the gabbro-alternative. This is because the number of gabbro bodies available in Sweden, that have sufficient size to appropriately host a repository is limited, probably to about some ten. Out of these majority are found in the northernmost part of the country. An important factor in this context is the fact that gabbro intrusions typically have much larger surface areas than depth extensions. Furthermore, the required flexibility to avoid major structures within the selected rock body imposes additional volume requirements, see Figure 17-3.

Thus, in a site selection situation presuming gabbro, the number of potentially suitable locations would be very limited from start. Adding further discriminating factors of non-geological character, such as accessibility, alternative use of land, landownership and population density, it is believed that few gabbros, if any, would in reality be acceptable for siting a repository.

Considering then the construction of a repository, no decisive difference in conditions offered can be identified, when comparing gabbro to granitic rocks. Generally speaking, both rock types provide good construction feasibility and, in fact, poses similar problems related to large-scale structural features.

With respect to repository performance, it is clear that the mineral chemistry of basic rock offers potential advantages as compared to granite/gneiss. This includes high sorption capacity and potential for self-healing of fractures. The significance of these processes is, however, not well verified experimentally.

Available data on the hydraulic conductivity of gabbro show, on average, somewhat lower values than for granite/gneiss. Results are, however, not consistent. Furthermore, the differences indicated are typically a factor of five or less, which is not decisive with respect to repository performance. An important characteristic displayed by all gabbro bodies investigated, is that they are rather frequently interested by granitic dykes. These are often highly fractured and exhibit higher hydraulic conductivity than the gabbro host rock.

The relatively poor thermal energy transport capacity of gabbro implies some disadvantages, in terms of higher thermomechanical loads on the rock in the post-closure phase. Differences with respect to granitic rocks are, however, moderate. It is envisaged that they can, if desired, be balanced by design measures. This will, however, negatively affect repository volume requirements and also costs.

In conclusion, there are obvious difficulties associated with siting a repository in gabbro, due to lack of suffi-

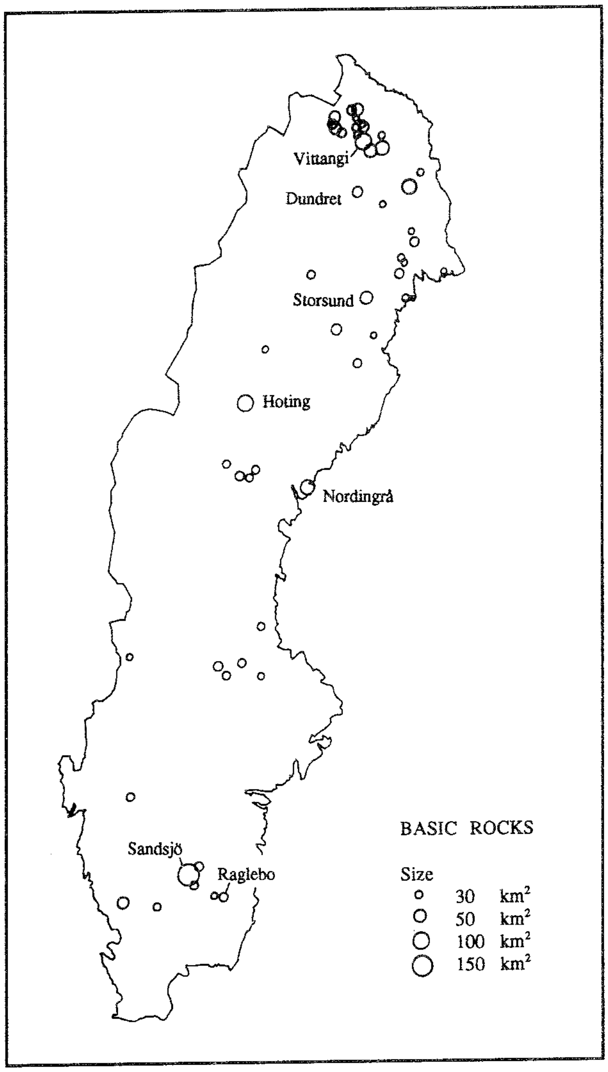


Figure 17-3. Location of basic rock bodies larger than 20 km^2. (From ref. /17-32/)

ciently large gabbro bodies. In comparing gabbro with granitic rocks, no decisive differences can be demonstrated on the basis of the present state of knowledge, neither with respect to repository construction, nor as regards repository performance.

17.4.3 Possible strategies for geoscientific classification

Certain standardized evaluation systems have been developed during the past decade for classification of the vulnerability of the groundwater reservoirs and groundwater resources. This is primarily intended to reduce the risk of contamination of superficial groundwater accumulations. The classification system has now been further refined to permit a rough assessment of disposal sites for radioactive waste from the hydrogeological point of view. Since knowledge of different sites in an evaluation phase can vary with respect to the extent of completed mapping surveys and pre-investigations, it is essential that the uncertainty level for the hydrogeological parameters be weighed into the assessments /17-37/. This possible strategy for geoscientific classification will be further studied and evaluated during 1993.

17.5 THE LAXEMAR DEEP DRILLING PROJECT

The natural groundwater flux at repository level is not necessarily controlled by the local flow gradients, but is more likely controlled 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 1 000 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 and reached a depth of 1700.5 m. The coredrilling is the deepest one in Scandinavia. After concluded drilling, geophysical, hydrochemical, geochemical and hydraulic investigations will be performed in the hole.

17.6 DEVELOPMENT OF INSTRUMENTS AND METHODS

17.6.1 Reflection seismics

The study of the reflection seismic method with a small vibroseismic source for detecting horizontal or sub-horizontal reflectors at moderate depths in crystalline rock was not successful. The data processing showed no sub-horizontal fracture zone in the investigated rock volume.

However, in view of the potential importance of the method at early stages of site investigations SKB decided to continue the development effort of the seismic method. The strategy is to use the well-characterized sub-horizontal major fracture zone at the Finnsjön site as a reference for features which has to be detected. The first step, to re-process an old seismic survey at this site, seems to be promising. Later on, the development work will concentrate on the performance of the seismic field work.

17.6.2 Point dilution probe

The development of the point dilution probe for in-situ measurements of groundwater flow is almost completed. Two different flow cells can be used in the probe, one conductivity cell with which the dilution of saline tracer pulses can be measured and one light transmission cell with which the dilution of colour tracer pulses can be measured. From the dilution curve the groundwater flow across a packed-off borehole section can be determined.

The point dilution probe is designed for 56 mm boreholes or wider and flow measurements can be made at depths down to 1500 m.

17.6.3 Core drilling

The 1700 m deep borehole at Laxemar, see section 16.5, was drilled according to the telescope-type drilling method, i e uppermost part drilled with larger diameter (for optimizing post-completion testing) and air-lift pumping maintained during drilling (for minimizing contamination). The core drilling was this time carried out by the wire-line technique with a larger drill rig than normally used for SKB drilling. Among drilling parameters recorded, the water into and out from the borehole, giving the water balance during drilling, is of interest for evaluating the pumping effect and water conducting sections of the borehole. The drilling diameter was 76 mm, except for the uppermost 200 m where 8" was used, and the drilling time was less than three months.

17.6.4 Measurement techniques at the Äspö HRL

Developments or improvements of methods and instruments related to activities at the Äspö HRL are more or less continuously ongoing. The Hydro Monitoring System has been increased and testing tools have been adapted to the conditions in the tunnel environment. Details are described in Äspö reports.

17.6.5 Evaluation of pre-investigation methods

The evaluation of pre-investigation methods are an important issue prior to the site investigations at the candidate sites for the deep repository. Based on experiences from the investigations at Äspö, with conceptual modelling of the site and prediction work, a first evaluation report will be written. This work has been initiated and will later on be followed by a second evaluation report when the outcome from the Äspö validation work is finished.

17.7 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 point of views could be ventilated. The following seminars have been arranged with participation of the authorities and different experts in the broad field of geoscience:

- Offshore high resolution geophysical surveys.
- Stochastic continuum simulation of mass arrival using a synthetic data sheet.
- Direct fault dating trials at the Äspö Hard Rock Laboratory.
- Sensitivity study of rock mass responds to glaciation at Finnsjön.
- Gabbro as a host rock for a repository.
- Tectonics and Paleostress regimes in the southern part of the Baltic Shield during the last 1 200 Ma.
- Far field modelling (in co-operation with SKI).

Besides these open discussions it is of great importance to present and assess the ongoing R&D work within the international scientific society. The SKB Geoscience programme encourages the involved consultants and researchers to attend international meetings as well as to publish papers in scientific journals /17-38/, /17-39/, /17-40/, /17-41/, /17-42/.

18 CHEMISTRY

18.1 GEOCHEMISTRY

18.1.1 General

The investigations of groundwater and geochemistry, including evaluation of data were concentrated to the Äspö HRL site during 1992, as they were under 1991. The groundwater sampling has been made in the entrance tunnel and mainly from probing holes in the tunnel. Additional sampling was made from permanently packed off sections in the deep boreholes on Äspö. The results of the analyses have been used in the evaluation of the tunnel section 700 – 1475 m and compared to the predictions set up before the construction work started.

Since much of the work is carried out within the Äspö HRL project the geochemical periodic meetings have been connected to the periodic meetings of the redox project, see below.

Since late 1991 a groundwater quality classification exercise has been carried out together with SKBs Finnish sister organization TVO. One of the aims is to give a basis for optimizing the groundwater sampling activities in the repository site investigations, another aim is to identify the high quality data which is necessary for quantitative modelling of groundwater rock interaction.

18.1.2 Final evaluation of geohydrochemical pre-investigations

The pre-investigation phase at Äspö was finished in 1991 with a united conceptual modelling exercise using data from geological, geohydrological and groundwater chemical measurements. However, a renewed evaluation of the geohydrochemical data has been made in relation to the reported geohydrological and geological conditions /18-1/. This critical evaluation is much more complete than the common modelling. However, the results do not contradict each other. The main difference is that the latter (final) evaluation of the data give a more refined and detailed picture of the groundwater flow situation at Äspö. The groundwater flow model based on these data is presented in Figure 18-1.

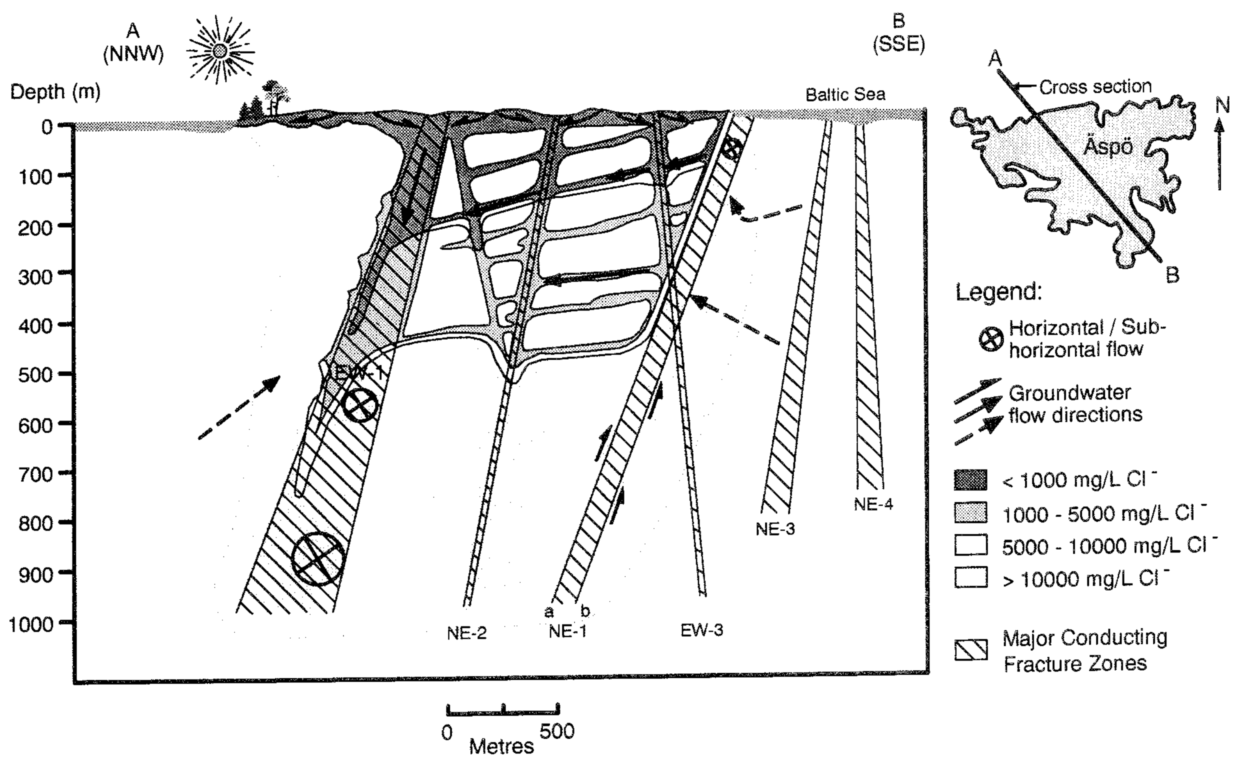


Figure 18-1. Conceptual groundwater flow model /18-1/.

The main conclusions/experiences from this evaluation are:

- Two main meteoric water bodies exist superimposed on deep regional groundwaters. These consist of a marine-derived water and a floating lens of fresh (to brackish) water, with a narrow dispersion zone at the interface located at around 50 m depth.
- At greater depths the highly saline regional groundwaters, which are welling up very slowly from depth, also mix to limited degrees with the marine-derived waters along an interface which extends approximately from 400 – 500 m depth.
- There are at least three sources of meteoric waters which are entering and mixing within the upper 500 m: a) recent fresh to brackish near-surface waters, b) modern Baltic Sea, and c) deep saline waters. Subordinate amounts of ancient seawater and glacial melt water also contribute.
- The major hydrogeochemical character of the groundwaters can be explained by the mixing of waters from two major sources; shallow fresh/brackish vs. deep saline.
- The three-dimensional fracture grid system which characterizes the island of Äspö results in the active hydraulic mixing, circulation and redistribution of the groundwaters, particularly within the upper 100 – 200 m, but also down to 400 m.
- The lack of any systematic groundwater and fracture filling isotopic trends reflects the complexity of the mixing processes.

18.1.3 Comparison between collected tunnel data and predictions

The sampling of groundwater in the Äspö tunnel and on site analyses are conducted by the local staff.

The results of the analyses serve two main purposes:

- to evaluate the groundwater chemical situation and compare the results with the predictions,
- to evaluate the groundwater flow situation and compare the results with the predictions.

In both cases, but especially for the groundwater flow situation the evaluation is much broader than the previous comparison with the predictions.

During 1991 the results of the investigations made in the tunnel section 0 – 700 m were reported. During 1992 the first predicted tunnel section was investigated. The comparison between the outcome and the prediction for the tunnel section 700 – 1475 m is presently under way.

Groundwater samples have been collected from 81 different locations in the tunnel between 700 and 1475 m. Most samples are from probing holes 20 m deep. During grouting of the fracture zone NE-1 seven adjacent boreholes were carefully monitored for changes in groundwater composition. The samples, collected at the drilling of the probing holes have been analyzed for bicarbonate, chloride and pH.

On the basis of the results from the first sampling campaign a few boreholes were selected for renewed sampling. This second sampling campaign was more carefully done than the first one. Also many more parameters were analyzed in these samples. The water was filtered on-line through a 0.45 micron filter and preserved for specific analyses. This arrangement is necessary to prevent influences from air.

In the section below the sea between Hålö and Äspö very high bicarbonate concentrations have been found. Compared to the sea water, the sulphate concentrations are surprisingly low. These observations indicate that a microbial sulphate reduction takes place. The existence of the sulphate reducing bacteria Desulfovibrio and sulphur isotope data further strengthen this conclusion. No evaluation has been made so far.

The early analyses of the water sampled in the probing holes are important, because they give the initial groundwater composition before the drainage to the tunnel has caused a major disturbance of the natural conditions. If the tunnel creates a major change in the groundwater flow, this can be followed in the changes of composition in subsequent groundwater samples.

Changes in the groundwater chemistry and especially the chloride concentration are used as indicators of solute transport, together with the natural tracers tritium and ^{18}O. In addition to this, the groundwater flow and the measured electrical conductivity of selected borehole sections at Äspö have been used to evaluate the changes in the water salinity, caused by the drainage into the tunnel.

In the tunnel section 700 – 1475 m, groundwaters have been collected for analyses of deuterium, tritium and ^{18}O. In Table 18-1 some results are presented.

The tritium data in Table 18-1 indicate that there is a measurable proportion of modern water in the analyzed samples. The deuterium data are also close to the value of the Baltic Sea. It is therefore likely that the drainage into the tunnel has caused water from the overlaying Baltic sea to enter the rock mass.

Table 18-1. Deuterium, tritium and ^{18}O data of groundwater samples from the tunnel section 700 – 1475 m.

Borehole	^{18}O ‰ SMOW	Deuterium ‰ SMOW	Tritium Bq/l
SA1062B	-7.7	-58.0	1
SA1077A	-7.5	-58.7	2
SA1094	-7.3	-60.3	2
SA1111B	-7.7	-60.3	3
SA1163B	-7.9	-61.9	3
SA1210A	-7.4	-61.5	2
SA1229A	-8.1	-63.6	2

SMOW = Standard Mean Ocean Water

18.1.4 Redox experiment

The objectives of the experiment are to evaluate the oxygen reduction rate in a water conducting fracture zone and to evaluate the effects of having an oxidizing front penetrating a previously reducing fracture system. Both the effects on the groundwater chemistry and on the fracture minerals will be investigated.

Background and purpose

During the operating phase, when a repository is kept open for the inplacement of the spent fuel canisters, the drainage of groundwater into the tunnels will cause an enhanced water circulation in the surrounding rock mass. This water circulation causes oxygenated surface water to be transported to great depth. An increase in the amount of the infiltrating surface water by one order of magnitude might cause oxygenated water, which is normally reduced at a few tens of metres, to be drawn to several hundred metres depth. Such a situation might cause an oxidation of the fracture minerals in the water conducting fractures all the way from the surface down to the repository. The consequences of this scenario would be that, in the post closure phase, radionuclides oxidized by the radiolysis might be transported in an oxidized form through the geosphere from a leaking canister up to the surface.

The geochemical data obtained from the site investigations made during the past ten year all over Sweden, clearly indicate that the oxygen in the infiltrating water is reduced in the soil and in the uppermost part of the bedrock. At a depth of 100 metres the water is reducing with a typical iron concentration of 1 – 10 mg/l. Only in one borehole out of 30 – 40 has oxygen been measured in samples from more than 100 m depth. The prevailing reducing conditions are also seen in mines and tunnels in the rock where iron precipitates on the walls is due to the inflow of reducing iron rich water which has been oxidized by the oxygen in the air.

The effect of the drainage into the tunnel system has not been specifically investigated in previous work. The experiment is located to a fracture zone at 70 metres depth below the surface. Breakthrough of oxidizing surface water was expected to occur at this location.

Operation

Before the tunnel reached the fracture zone, one bore hole was drilled through it. This was sampled immediately. The result of the sample analysis indicated that the water had been stagnant with a salinity of approximately twice that of the surrounding Baltic Sea. Two weeks after the tunnel had reached the fracture zone, the salinity and the iron content of the inflowing water had decreased to zero. Afterwards the salinity and the iron content increased again.

The breakthrough of the surface water and the oxygen was calculated based on the estimated water inflow to the tunnel but before the freshwater reached the tunnel /18-2/.

The drillcores have been carefully mapped with respect to fracture and bulk rock minerals. The same boreholes will be overcored after oxygen break through and the mapping repeated. The most suitable analyses will be made in order to find out how much of the minerals have reacted.

Water samples are regularly collected and analyzed for main constituents and redox sensitive elements. Isotope analyses are also made in order to facilitate the interpretation of the complex flow situation. During 1992 the chemistry in the fracture zone has been monitored. During this time the data has been evaluated successively.

TVO joined the Äspö project with a specific wish to participate in the redox experiment.

Reporting

The results of the initial investigations were published as a progress report /18-3/ and the results from 1992 in another one /18-4/. The evaluation of the observed groundwater chemical processes taking place in the fracture zone was done in 1992.

18.1.5 Groundwater sampler for "stagnant water"

Sampling of groundwater from low-conductivity rock is a very important issue as these samples contribute to the understanding of the interaction between rock and water. However, this sampling is very difficult to perform, due to the small amount of water available. For the construction of a sampling system the first criteria is to minimize the "dead" water volume in the sampling equipment. Therefore very slim boreholes, 36 mm or 46 mm, will be used. Critical for the development work is the choice of materials for all components being in contact with the water, in order to avoid trace metal contamination.

The prototype of a packer system has been constructed and a functional test of this prototype in a borehole is going on. The next step in the development of the sampling system will be the construction of a miniature flow cell for measurements of Eh and pH at these very low water flows. The planning for this work has just been initiated.

18.1.6 The CHEMLAB probe

The migration of radionuclides is a very slow process, due to the sorption of the nuclides to the rock minerals. It is also necessary to maintain a full control of the parameters involved to be able to fully interpret the experiments. These kind of experiments are therefore usually performed on rock samples in laboratories. However, it is very difficult to artificially create the right in-situ conditions in a laboratory. Therefore, within the Äspö HRL programme, radionuclide migration experiments are planned to be carried out in a "borehole laboratory", the CHEMLAB probe. Migration experiments will be made on rock samples inside the probe. The CHEMLAB will be placed in a borehole, and the water for the experiments will be taken from the borehole. The experiment and all the radionuclides will be confined to the inside of the probe.

IPSN/CEA in Cadarache, France has been contracted for the development work. The design work of the CHEMLAB has been almost finished. Details of the system, such as chromatography pumps, valves, couplings etc have been constructed and tested. The planning of supporting laboratories, test site laboratory in the tunnel and test preparation laboratory located at the SKB CLAB facility, has been initiated.

18.2 RADIONUCLIDE CHEMISTRY

18.2.1 Solubility and speciation

The solubility and formation of complexes in carbonate containing solutions of thorium /18-5/ and neptunium(IV) /18-6/ have been investigated. The aim is to calculate the solubility and speciation of tetravalent actinides in contact with pure groundwater or bentonite pore water. Under the normal reducing conditions in deep groundwater thorium will always appear in the tetravalent state and neptunium will occur as neptunium(IV). The solubility is low and the dominating complex around neutral pH is anionic with three hydroxy groups and one carbonate group per central actinide(IV) atom, see Table 18-2. A similar behaviour is expected for U(IV) and Pu(IV), because they are also actinides. Preliminary calculations show that hydroxo-carbonate complexes are also formed for uranium and plutonium and will influence their chemical properties under repository conditions. For example the solubility of $Pu(OH)_4$ is increased at high pH and high carbonate concentrations typical for bentonite pore water due to the formation of plutonium-hydroxo-carbonate complexes /18-6/.

The experiments on the kinetics of thorium dioxide dissolution, using a plug flow reactor, have continued. The dissolution is related to the formation of a thorium carbonate complex. A summary of results is being produced.

SKB has supported participation of Swedish experts in the international OECD/NEA project TDB in order to compile and evaluate thermodynamic databases for important radionuclides. SKB is also participating in the CHEMVAL project which is organized by CEC and engaged in the validation of geochemical codes.

18.2.2 Organic complexes, colloids and microbes

Results from investigation of natural analogues to radioactive waste disposal, for example the Poços de Caldas analogue, have indicated a binding between Th^{4+} and aquatic humic substances /18-7/. Laboratory experiments have now confirmed this interaction /18-8/. Ultrafiltration was used to study complex formation between fulvic acids and the ions Eu^{3+}, Th^{4+} and UO_2^{2+}. A stronger binding was noted for Eu^{3+} and Th^{4+} as compared to UO_2^{2+}, which can be explained by the higher charge density of the two former ions. The results underline the necessity to regard radionuclide interaction with aquatic humic substances in performance assessment of deep disposal.

Table 18-2. Equilibrium constants for thorium and neptunium(IV) in carbonate containing solution.

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

a Reference /18-5/
b Reference /18-6/

The investigation of the influence of pH, ionic strength and fulvic acids on the size distribution and surface charge of colloidal particles of quartz and ferric hydroxide has been published /18-9/. This will also be part of a doctoral thesis presented at the Dept. of Water and Environmental Studies at University of Linköping presented by A. Ledin. It was found that fulvic acids can stabilize ferric oxid colloids att high pH.

The laboratory experiments with goetite colloids have continued. Parameters such as pH and water flow are important for the transport of goetite colloids in a quartz filled column.

Sampling of colloids in natural deep groundwater is difficult. This has been clearly demonstrated by the use of a photo correlation spectrometer down in the Hard Rock Laboratory at Äspö. Colloids were directly measured in groundwater from packed off sections in a borehole. The generation of new colloids even at seemingly secure sampling procedures could be followed by the instrument /18-10/.

Sampling and analysis of microbes in groundwater has continued. Our efforts in this field are concentrated to the

Äspö Hard Rock Laboratory, see Table 18-3. Earlier investigations of microbes from deep groundwater in Stripa, mid-Sweden, have been published. It was found that deep groundwater bacteria are capable of assimilating CO_2 /18-11 and 18-12/. Further experiments of metal (radionuclide) uptake on bacteria have been reported /18-13/.

Identification of a selected DNA-sequence is being tried as a way to identify deep groundwater bacteria. This so called 16S-rRNA method has indicated the existence of the sulphate reducing bacteria *Desulphovibrio baculatus* in groundwater samples from Äspö. Previous investigations have shown the existence of iron(III)-reducing bacteria. The strain *Shewanella putrefaciens*, which is capable of iron(III) reduction, was isolated and reported /18-14/. The competition between sulphate-reducing and iron(III)-reducing bacteria has recently been pointed out as an important process for groundwater chemistry /18-15/.

Table 18-3. Sampling of bacteria in groundwater at Äspö Hard Rock Laboratory. The sampling points are ranging from surface water down to 600 m in drillholes.

Sampling location	Depth m	Date[b]	Total number of bacteria ml^{-1}	Sulphate reducing bacteria
Surface water	0	92	1,400,000	n.a.
HBH 02	10	91	1,140,000	n.a.
		92	890,000	neg.
HBH 01	40	91	–	n.a.
		92	3,800	neg.
KR 0012 B	70	91	–	n.a.
		92	38,000	n.a
KR 0013 B	70	91	16,000	neg.
		92	41,000	n.a.
KR 0015 B	70	91	39,000	n.a.
		92	17,000	n.a.
SA 0644 B		91	14,000	n.a.
SA 0813 B		92	5,900	pos.
SA 0923 A		92	12,000	n.a.
SA 0958 A		91	11,000	n.a.
SA 0992 A	50-200[a]	91	14,000	n.a.
SA 1009 A		91	11,000	n.a.
SA 1062 B		92	6,900	pos.
SA 1327 A HB		92	16,000	pos.
SA 1420 A		92	12,000	neg.
KBH 02		91	28,000	n.a.
KAS03	544-626	92	63,000	pos.

a Sampling in probing holes along the entrance tunnel to the Äspö Hard Rock Laboratory.
b Sampling made 1991-12-02 (91) and 1992-12-02 (92).
n.a Not analyzed.
neg. Negative.
pos. Positive.

18.2.3 Sorption and diffusion

The evaluation of the surface complexation method as a way to describe radionuclide sorbtion on mineral phases continues.

Previously unpublished results from experiments with sorption of Sr, Ba and Ra on rock minerals and soil samples have been summarized /18-16/. The influence of parameters such as pH and ionic strength in the simulated groundwater was tested. The cation exchange capacity of the substrate was also tested. The analogous behaviour of these alkaline earth metal ions and the increased sorption in the order $Sr^{2+}<Ba^{2+}<Ra^{2+}$ were demonstrated, see Figure 18-2. In Figure 18-2 the effect of ionic strength can be observed. The unfilled symbols are measurements in a simulated groundwater twice as concentrated (TDS) as for those marked with filled symbols. The dependence on ionic strength was only tested for Sr and Ba. The measurements for Ra were too scattered (alpha spectrometry). The radionuclides used were ^{85}Sr, ^{133}Ba and ^{226}Ra. It was concluded that the sorption process is dominated by ion exchange.

A lack of understanding of the process of radionuclide diffusion through bentonite has been identified when summarizing data for the SKB 91 performance assessment. This may have implications for the description of the stationary diffusion release scenario. Therefore further

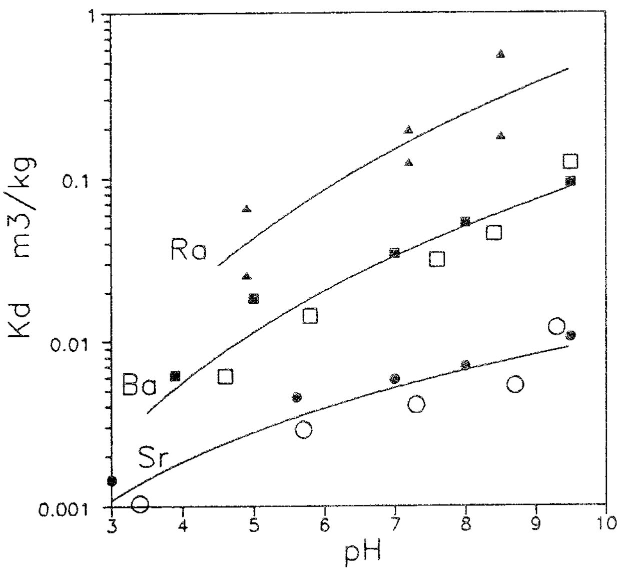


Figure 18-2. Sorption (K_d) of Sr, Ba and Ra on crushed granite as a function of pH /17-3/. The unfilled symbols refer to a different water composition with half the content of dissolved solids (TDS) as compared to the reference water for filled symbols.

diffusion experiments and modelling evaluations have been initiated in order to clarify this issue.

18.3 VALIDATION OF TRANSPORT MODELS

18.3.1 Laboratory experiments

Equipment is being built to study water flow through a natural open rock fracture. The fracture has been overcored by diamond drilling and the core brought into the laboratory for the experiment. A detector system is being constructed in order to follow the break through of a tracer substance.

Previous experiments with migration of redox sensitive radionuclides through open rock fractures have pointed to the necessity of making separate experiments in order to elucidate the kinetics of reduction and sorption /18-17/. Such experiments have been performed with technetium and neptunium. Strictly reducing conditions, simulating deep groundwater, are kept throughout the experiments. Reduction of the radionuclides occurs both in the water phase and on the surface of iron(II) containing minerals. The reduction of pertechnetate ions to technetium(IV) in solution is expected to be slower than the reduction of neptunyl ions to neptunium(IV), due to the fact that the former reaction is a three electron process and the second is a one electron process.

An cell for the measurement of radionuclide diffusivity in bentonite clay has been constructed. The intention is to install the cell in the CHEMLAB probe and use it for measurements at in situ conditions.

18.3.2 Tracer test at Äspö

Dilution measurements

The deep boreholes on Äspö are equipped with a permanent packer arrangement in which two of the packed off intervals can be used for circulation of water to the surface. In the circulating volume a colour tracer is added which can be easily analyzed. Within this loop a dilution of the colour is due to groundwater flow through the packed off section. Thus, by analysing the tracer concentration at regular time intervals, the groundwater flow can be calculated down to the limit of molecular diffusion, which is less than 0.1 ml/min.

The dilution measurements have been made in the core drilled boreholes KAS02, KAS04 and KAS06 during the tunnel construction /18-18/. The results are listed in Table 18-4.

Table 18-4. Groundwater flow in packed off borehole sections during the construction of the tunnel.

Borehole	Code	Section	Flow (ml/min)					
		(m)	LPT-1[A]	NGI[B]	NG2[C]	LPT-2[D]	TP1[E]	TP2
KAS02-4	B4	309-345	1.1	–	fm	2	1.0	0.5
KAS02-2	B2	800-854	fm	–	fm	4	3.0	2.5
KAS04-2	D2	332-392	28	–	12	–	4.7	4.3
KAS06-5	F5	191-249	197	25	27	ph	3.0	2.5
KAS06-1	F1	431-500	79	52	25	ph	96	119

– = No measurement, ph = Pump hole, fm = Failed measurement.
A = (LPT-1) August 1989, B = (NG1) September 1989, C = (NG2) June-August 1990,
D = (LPT-2) October 1990, E = (TP1) February 1992.

19 THE ÄSPÖ HARD ROCK LABORATORY

The scientific investigations within SKB's research programme are part of the work of designing a final repository and identifying and investigating a suitable site. This requires extensive field studies regarding the interaction between different engineered barriers and host rock.

A balanced appraisal of the facts, requirements and evaluations presented in connection with the preparation of R&D-Programme 86 led to the proposal to construct an underground research laboratory. This proposal was presented in the aforementioned research programme and was very positively received by the reviewing bodies.

In the autumn of 1986, SKB initiated field work for the siting of an underground laboratory in the Simpevarp area in the municipality of Oskarshamn. At the end of 1988, SKB made a decision in principle to site the facility on southern Äspö about 2 km north of the Oskarshamn nuclear power plant, see Figure 19-1. Construction for the Äspö Hard Rock Laboratory started on October 1, 1990 after approval was obtained from the concerned authorities.

The work with the Äspö Hard Rock Laboratory, HRL, has been divided into three phases: the pre-investigation, the construction and the operating phase.

The pre-investigation phase aimed at site selection for the laboratory, description of the natural conditions in the bedrock and predictions of changes that will occur during construction of the laboratory. The investigations have been summarized in six Technical Reports /19-1 – 19-6/. The construction of the access ramp to a depth of 460 m will be used to check the predictive models set up from the pre-investigation phase, to develop methodology for construction/testing integration and to gather more data on the bedrock as a basis for improving models on groundwater flow and radionuclide migration. A programme for the operating phase has been set up, /19-7/. The operating phase is very much focused on research and development of models for transport of groundwater flow and radionuclide migration, tests of methods for construction and handling and pilot tests of important parts of a repository system.

Some highlights for 1992 are as follows:

The excavation work for the Äspö Hard Rock Laboratory commenced in October 1990. By the end of 1992, 1,924 m of the tunnel had been excavated, corresponding to a depth of 255 m below the surface. Construction and investigations of the bedrock are being carried out in parallel as planned. Documentation of the bedrock is being done in the tunnel as well as by means of measurements in more than 140 measuring points in surrounding boreholes. The measurement system is now automated and delivers data on-line to the site office. Collected data are compared with the models of the bedrock set up prior to the start of construction. The Scientific Advisory Committee follows these evaluations.

A methodology for tunnelling and data collection in conjunction with passage of major zones of weakness was tested during the year. A highly conductive zone of weakness (NE-1) was passed at a depth of 180 m below the surface. Extensive investigations of the zone were conducted on passage. Special studies were performed to refine the theory and practice of grouting the rock.

SKB's RD&D-Programme 92 gives an account of the results obtained in the project to date. It also presents a plan for the experiments to be conducted after the conclusion of the construction phase.

The Äspö HRL has aroused great international interest. Participation agreements have previously been signed with Atomic Energy of Canada Limited (AECL), with the Power Reactor & Nuclear Fuel Development Corporation (PNC) of Japan, and with the Central Research Institute of the Electric Power Industry (CRIEPI) of Japan. Four more countries signed agreements on active participation in the project during 1992: Agence Nationale Pour la Gestion des Déchets Radioactifs (ANDRA) of France, Teollisuuden Voima Oy (TVO) of Finland, UK NIREX of the UK and the United States Department of Energy (USDOE). An important part of the cooperation concerns models for groundwater flow and radionuclide migration. A Task Force has been formed, and the first meeting was held during the year. Some ten model groups are currently working on evaluation of the combined pumping test and radioactive tracer test (LPT2) which was carried out in conjunction with the pre-investigations for the Äspö HRL.

The planning and design of Äspö village is completed. Construction work will begin in the spring of 1993.

A separate Annual Report 1992 has been prepared for the Äspö Hard Rock Laboratory /19-8/ and the reader is referred to this publication for a more detailed account of the achievements for 1992.

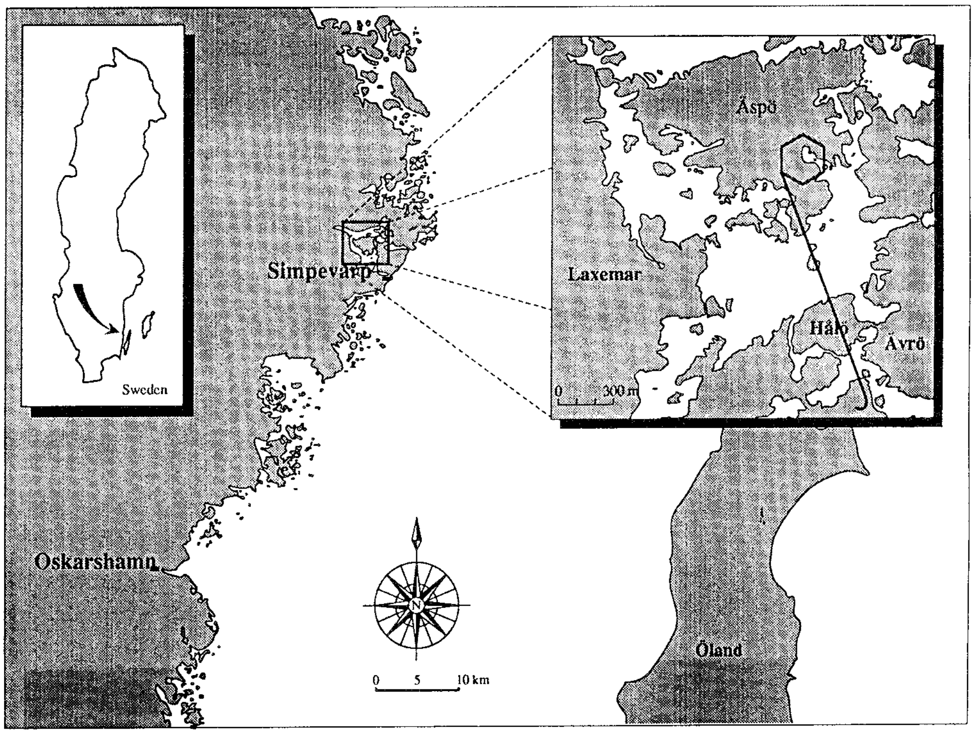


Figure 19-1. *Location of the Äspö Hard Rock Laboratory .*

20 NATURAL ANALOGUE STUDIES

20.1 THE POÇOS DE CALDAS PROJECT

The last reports from the project were printed in 1992. A summary of the results and their implications together with articles on the underlying investigations has been published as a special issue in Journal of Geochemical Exploration /20-1/. A reprint from this issue has also been bound into the form of a book by Elsevier Co, Amsterdam 1993. A summary of the results and their implications can also be found in the SKB series of reports /20-2/ (also available as reports from NAGRA, NTB 90-33, and UK DOE, WR 90-055). Short reviews of the results and implications have been made /20-3 and 20-4/. Applications to performance assessment of spent fuel disposal have been made and examples of such exercises have been presented /20-5/.

20.2 THE CIGAR LAKE PROJECT

The Cigar Lake uranium mineralisation is situated in northern Saskatchewan, Canada. It has been studied by AECL as a natural analogue to deep disposal of spent fuel since 1984. SKB joined the project 1989. The Los Alamos National Laboratory participates in the study since 1991, supported by US DOE. The three year project phase from 1989 to 1992 is being summarized and evaluated. Two annual reports from this period have been produced /20-6 and 20-7/. A summary of the results with emphasis on the analogue aspects and the implications of modelling activities related to performance assessment has been presented /20-8/.

The Cigar Lake uranium ore deposit is situated at a depth of 430 m in sandstone, see Figure 20-1. The deposit

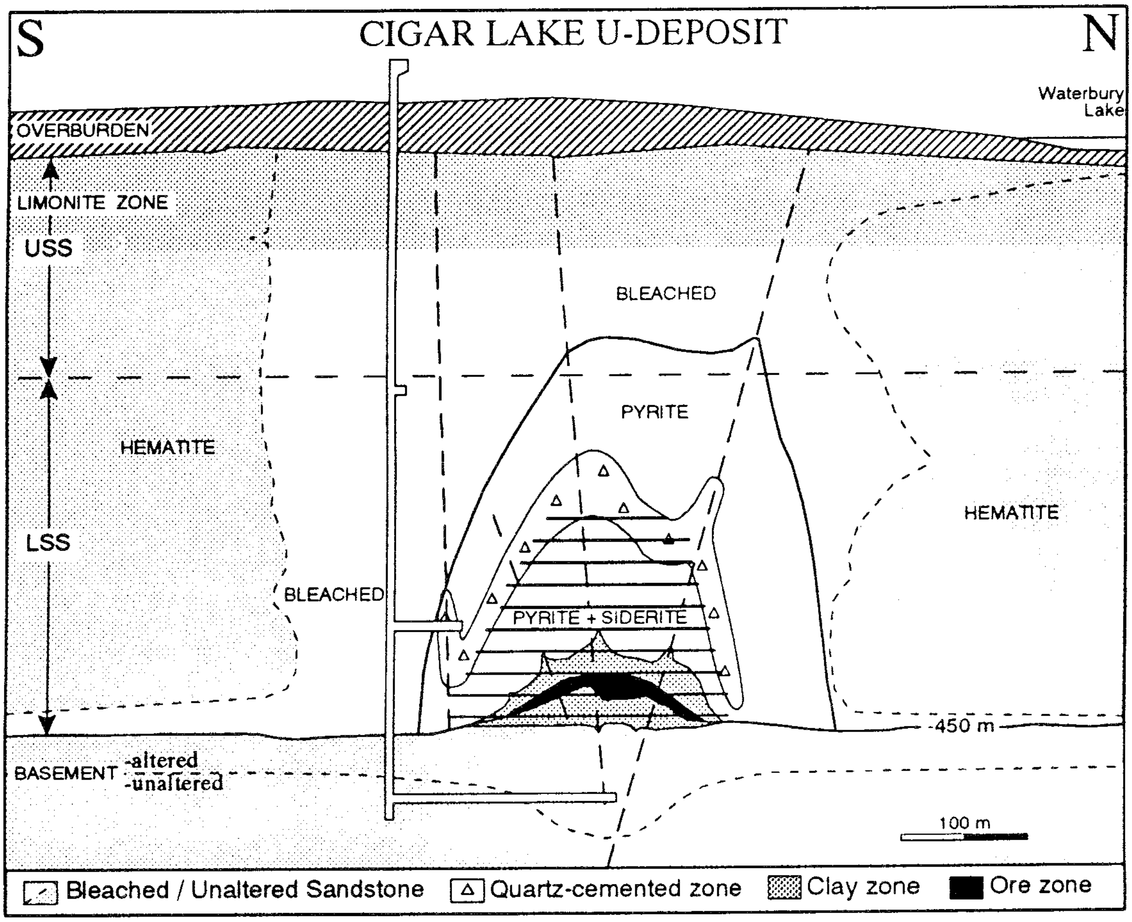


Figure 20-1. Schematic cross section through the Cigar Lake deposit showing the uranium mineralization and its host rocks, including the lithologic characteristics related to hydrothermal alteration and weathering. (USS = upper sandstone; LSS = lower sandstone).

is very concentrated with an average grade of 12% and with local concentrations of 55%. The ore body is about 2 km long, 25-100 m wide and 1-20 m deep. It is situated in a clay matrix and surrounded by a 5-30 m thick clay halo. There is no direct indication of its existence on the ground surface.

The uranium ore and the clay were formed by hydrothermal activity at a depth of more than 3 km and about 1.3 billion years ago. The overlying sandstone is hydrothermally altered. Underlying the sandstone is the Precambrian basement consisting of metapelites and gneisses. The ore is located at the contact between the sandstone and the basement rock.

The uranium minerals are primarily uraninite and pitchblende with minor amounts of coffinite. The clay in and around the ore zone consists mainly of illite and chlorite with small amounts of siderite (iron(II) carbonate) and calcite. The ore also contains sulphides and arsenides of nickel, cobalt, molybdenum, lead, zinc, manganese, copper and iron. The lead is formed by radioactive decay of the uranium.

The ore was formed when reducing hydrothermal fluids at temperatures 150 to 100°C were discharged along a vertical graphite containing fracture zone in the basement into the oxidizing uranium containing solutions in the sandstone. Uranium precipitated and pyrite was formed. The hydrothermal reactions in the sandstone dissolved quartz and transformed feldspars in sandstone and underlying basement rock into clay minerals. The clay mineral content in the clay rich halo is up to 60%. The clay at the ore contact has a red-brown colour from iron(III) minerals such as hematite and ferrihydrites.

20.2.1 Databases and interpretive evaluation and modelling

The analogue project has been divided up into a number of discipline oriented tasks: colloids, geology, hydrogeochemistry, hydrogeology, mineralogy and lithogeochemistry, nuclear reaction product geochemistry, ore mineralogy, organics and microbiology and finally radiolysis. The aim for each task is to provide databases and to carry out interpretive modelling in each discipline. The results of these exercises in the form of accepted conceptual models and reference databases are used in the final step, which is the performance assessment related evaluation and modelling. Four performance assessment objectives were selected:

- The evaluation of thermodynamic equilibrium codes and their databases.
- The role of colloids, natural organic substances and microbes for the migration of radionuclides.
- The stability of UO_2 and the influence of radiolysis on UO_2 dissolution and radionuclide release.
- The testing of mass-transport models for radionuclide migration through clay barriers.

The objectives are in fact to the largest part devoted to near-field issues of performance assessment for spent fuel repositories.

The geological and hydrogeological database for Cigar Lake contains information produced by the exploration and mining companies. Over 200 boreholes were drilled, cores collected, geophysical logging carried out, pumping tests made, piezometers installed etc. A shaft down to the ore and horizontal drifts has been excavated. Additional hydrogeological measurements were also made specifically for the analogue study.

A mineralogical and lithogeochemical database has been compiled for the analogue project. Likewise, a set of reference groundwater compositions has been selected from analysis results of water samples collected over a period of seven years. The reference set was carefully reviewed by geochemical experts before release to the performance assessment related modelling.

Numerical hydrogeological flow models have been applied for the dual purpose of explaining the existing hydraulic conditions and to test codes used in performance assessments such as the 3D hydraulic flow model NAMMU. The conceptual hydraulic model of Cigar Lake is illustrated in Figure 20-2.

The local flow conditions and the hydrochemical conditions have been summarized and explained by interpretive modelling as a base for the performance assessment related evaluation.

The clay and ore zones have very low conductivities as compared to the host sandstone. Therefore residence times calculated for imagined non interacting particles released within the ore are 18,000 to 85,000 years.

There is relatively little carbonates in the sandstone. Infiltrating meteoric water reacts with clay (illite and kaolinite) and iron minerals. Thus the pH and redox conditions of the recharge water are defined. The oxidizing reactions are leaving a zone of precipitated ferrihydrites down to about 100 m. The groundwater in the ore zone is reducing. Iron sulphides provide a considerable redox capacity and UO_2 is stable.

20.2.2 Modelling and evaluations related to performance assessment

Thermodynamic equilibrium codes

Solubility and speciation of radionuclides and other safety relevant elements are obtained by equilibrium calculations using numerical codes and their databases. The code PHREEQE has been used and the database ZZHATCHES. For plutonium the database SKBPU was used. Good agreement was obtained for U, Th, Ba and Cu. Less agreement was found for Sr, Mo, As and Cr due to poor data on solubility limiting phases (Sr, Mo and As) or uncertainty in redox potential for the element (Cr^{6+}/Cr^{3+}). The elements Ni and Pb were below detection limit. In no case were the solubilities predicted too low.

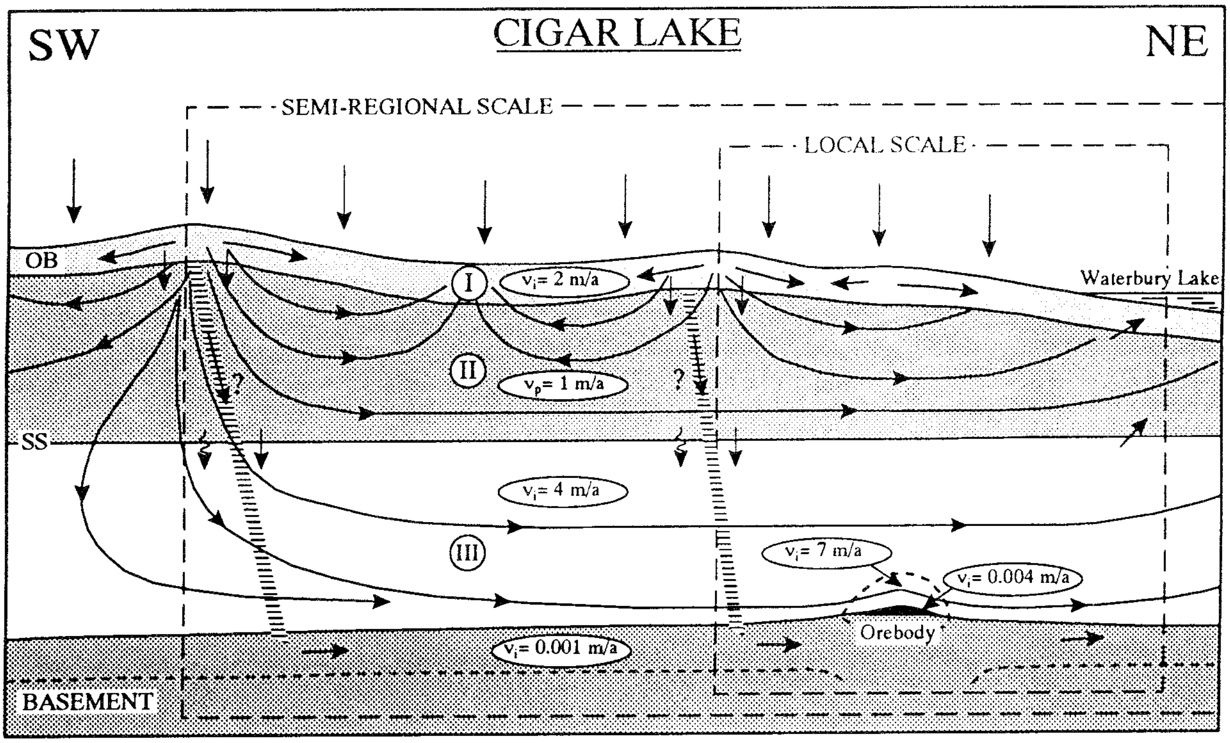


Figure 20-2. Schematic section through the Cigar Lake deposit showing the conceptual hydrogeologic model with three distinct flow regimes (I-III) and calculated particle velocities (v_i = integrated velocities, v_p = point velocities). (OB = overburden, SS = sandstone).

Colloids, organic material and microbes

Particles in the deposit have been analyzed and their capacity to carry radionuclides measured. It was indicated that some radionuclides like U, Th and Ra can remain fixed on particles for thousands of years and thereby become a case of irreversible-sorption-transport if the particles are mobile.

It was also found that particles in the ore zone are distinct from that in the host sandstone. This indicates that the transport of particles from the ore through the clay has been negligible.

The TOC (Total Organic Carbon) content in all surface and groundwater is less than 2 mg/l. Higher contents up to 11 mg/l have been measured in the ore zone. Only 15-25% are humic substances except in the ore zone where the humic content is less than 2%. The humic fraction is largely low molecular weight fulvic acids with ^{14}C ages in the ore zone of up to 15,000 a. No significant influence of complexation was found for either U^{4+} or U^{6+}.

Microorganisms are present in all groundwaters and seem capable of surviving the radiation fields. Anaerobic bacteria are about 10 times more abundant than aerobic bacteria. Sulphate reducers, iron related and denitrifying bacteria are common in the deposit.

Radiolysis and the stability of UO_2

The concentrated high-grade uranium ore at Cigar Lake within the low conductive clay offered the possibility to study the potential influence of radiolysis on uranium ore and near field chemical conditions as an analogue to spent fuel being exposed to groundwater inside a bentonite clay buffer.

The radiation from the ore will cause radiolysis of water close to the ore mineral surfaces. Oxidants will be generated such as OH-radicals H_2O_2 and O_2 and reducing species such as hydrogen. Hydrogen is not very reactive under the prevailing conditions and is therefore expected to diffuse out of the system through water filled connected pores and fractures. A net oxidation effect is consequently expected. The oxidants can in principle react with dissolved components in the groundwater: Fe^{2+}, HS^- and DOC (Dissolved Organic Content), or with solid components such as minerals with Fe(II) and sulphide or solid organic material or U(IV) in uranite, see Figure 20-3. The last reaction would be analogous to spent fuel oxidation by radiolysis which have been treated in performance assessment by both SKB and AECL. In the SKB approach it was conservatively assumed (SKB 91) that oxidation of UO_2 proceeds beyond the U_3O_7 stage causing damage and release to the original crystal lattice structure of UO_2. It is also assumed in the performance assessment exercise, that oxidants generated will create an oxidized near-field around the waste. Observations of the hematised zone at Cigar Lake can in a qualitative way be interpreted as caused by radiolytic oxidation. However, there is no sign of oxidation of the uranite beyond U_3O_7 or any extensive loss of uranium by dissolution for at least the last million years. Hydrogen has been found in the water of the ore zone which would support the assumption of radiolysis.

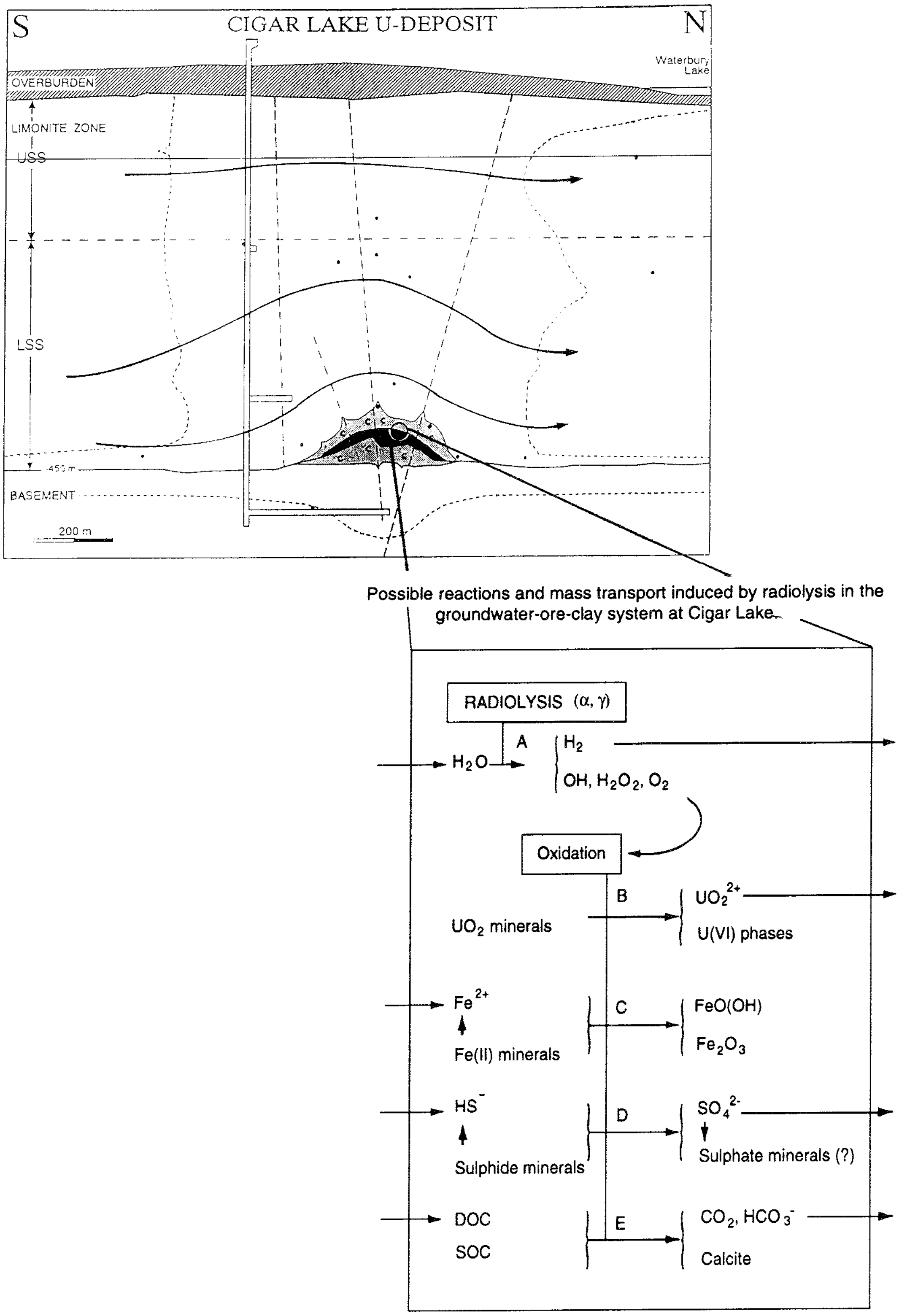


Figure 20-3. *Diagramatic vertical profile through the Cigar Lake uranium deposit showing the major rock units, the clay (c) and ore (black) zones, the vertical shaft and the main groundwater flow directions. The clay/ore contact, where radiolytic reactions might be expected, is ringed. Schematically illustrated is an analysis of possible oxidation reactions caused by radiolysis; the production of hydrogen and oxidizing species (reaction A), and some possible oxidation reactions accounted by these species (reactions B-E).*

Radiolysis kinetic models have been used to calculate the yield of hydrogen and oxidized species in the ore zone. The calculations can account for hydrogen and the oxidized zone with Fe(III) but the assumption of extensive UO_2 oxidation is not valid. The currently used model also overestimated the extent of net oxidation in the ore zone. So far the present results from Cigar Lake support the conclusion that oxidation in the near-field of exposed spent fuel cannot be ruled out. However, the net changes are overestimated by currently used models. It is also concluded that UO_2 must not be the component that consumes the oxidants generated but other constituents in the repository may react instead.

Near-field mass transport

Mass transport models are being applied to the observed conditions in the ore zone. In principle these are the same kind of models that are used to assess the mass transport in the near-field of a radionuclide waste repository and the situation is very similar. In both cases transport in the clay containing barrier is restricted to diffusion or flow at very low conductivity. Outside the dense clay zone the flow conductivity is higher and advective transport dominates. For the Cigar Lake near-field the modelled transport situation is schematically illustrated in Figure 20-4.

Constituents that can in principle be affected by mass transports have been analyzed in the ore zone and outside in the sandstone. Uranium has been studied but there is negative evidence of current uranium release, and according to isotope analysis there has been no uranium migration for at least one million years back.

In the ore, nuclide reactions generate elements such as helium and also radionuclides such as tritium, ^{14}C and ^{36}Cl. The rate of production can be calculated. These constituents are very mobile and it should be expected that a steady state builds up where production rate is balanced by near-field release and, for the radioactive constituents, also radioactive decay. Analysis of concentrations inside the ore zone and outside combined with information of groundwater flow in the sandstone around the ore can be used to calculate the release by a steady state near-field release model. Surprisingly good agreement between generation rate and release rate has thus been obtained for helium. For a decaying constituent such as tritium it was found that most of the generated ^3H decays within the ore body and clay.

Coupled mass transport with geochemical reactions in the clay zone has also been tested in order to simulate the development of the oxidized hematised zone in the clay. Reasonable agreement was obtained.

20.3 OKLO

The study of the fossil reactor zones at Oklo, Okelobondo and Bagombe as natural analogues to waste repositories is being directed by the French CEA and supported by CEC /20-9/. SKB and organizations from other countries such as Japan, USA etc are participating in the study. SKB has during 1992 mainly been participating in the investigation of the fossil reactor in Bagombe which was found about 20 km away from the zones in Oklo and Okelobondo, but into the same geological formation /20-10/.

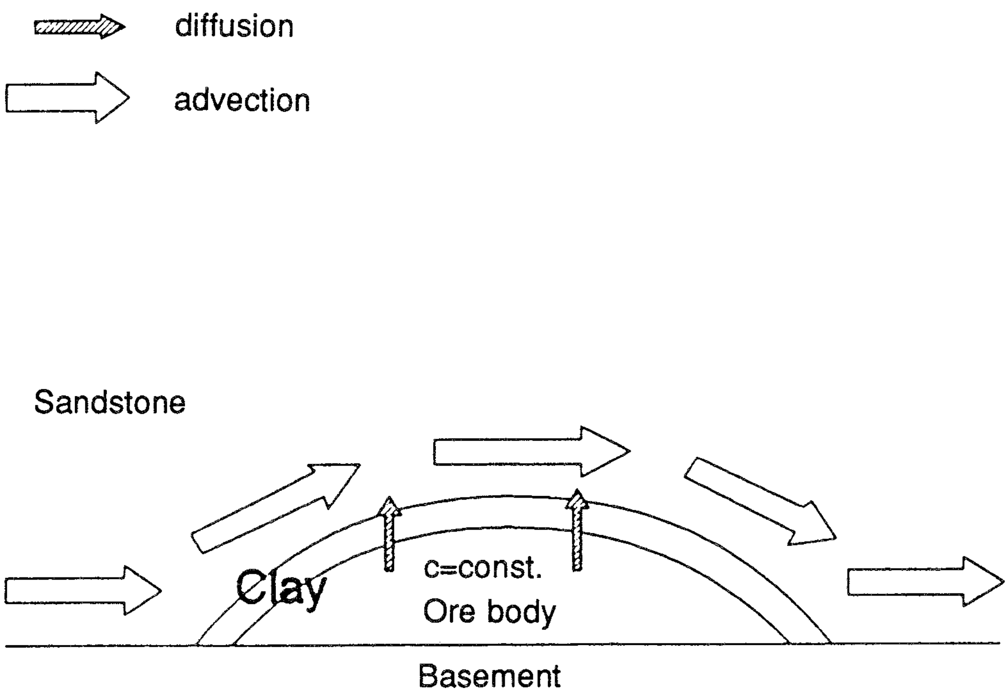


Figure 20-4. Schematic view of transport modelling at Cigar Lake.

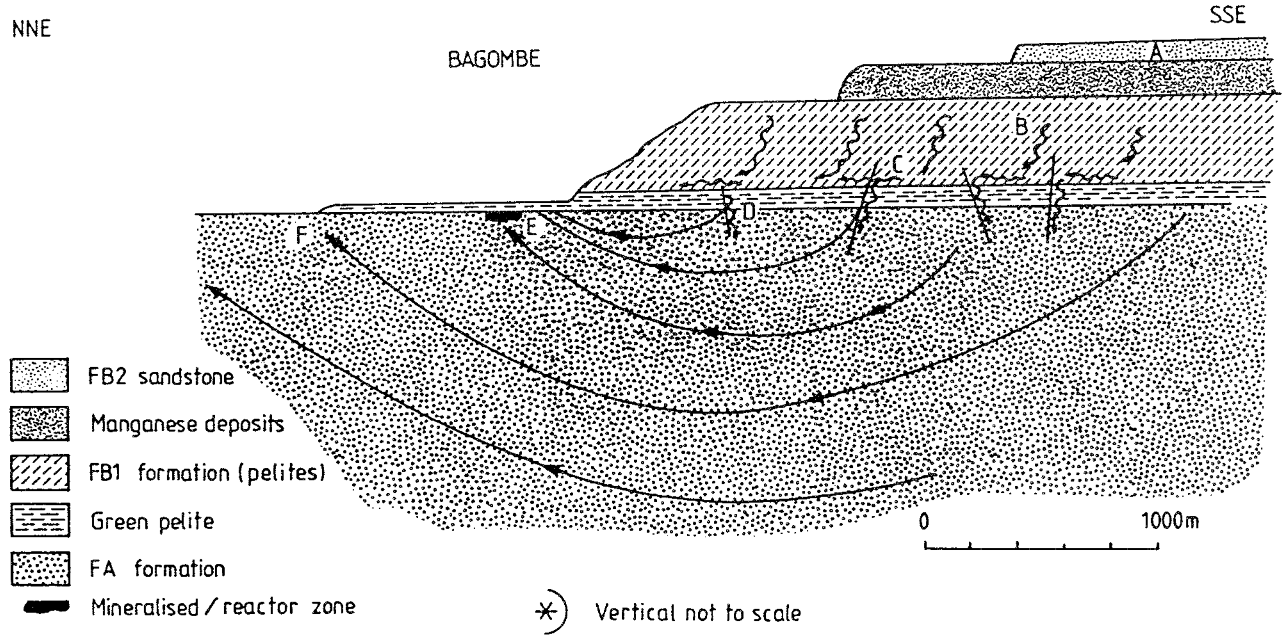


Figure 20-5. Bagombe: simplified conceptual groundwater flow model.

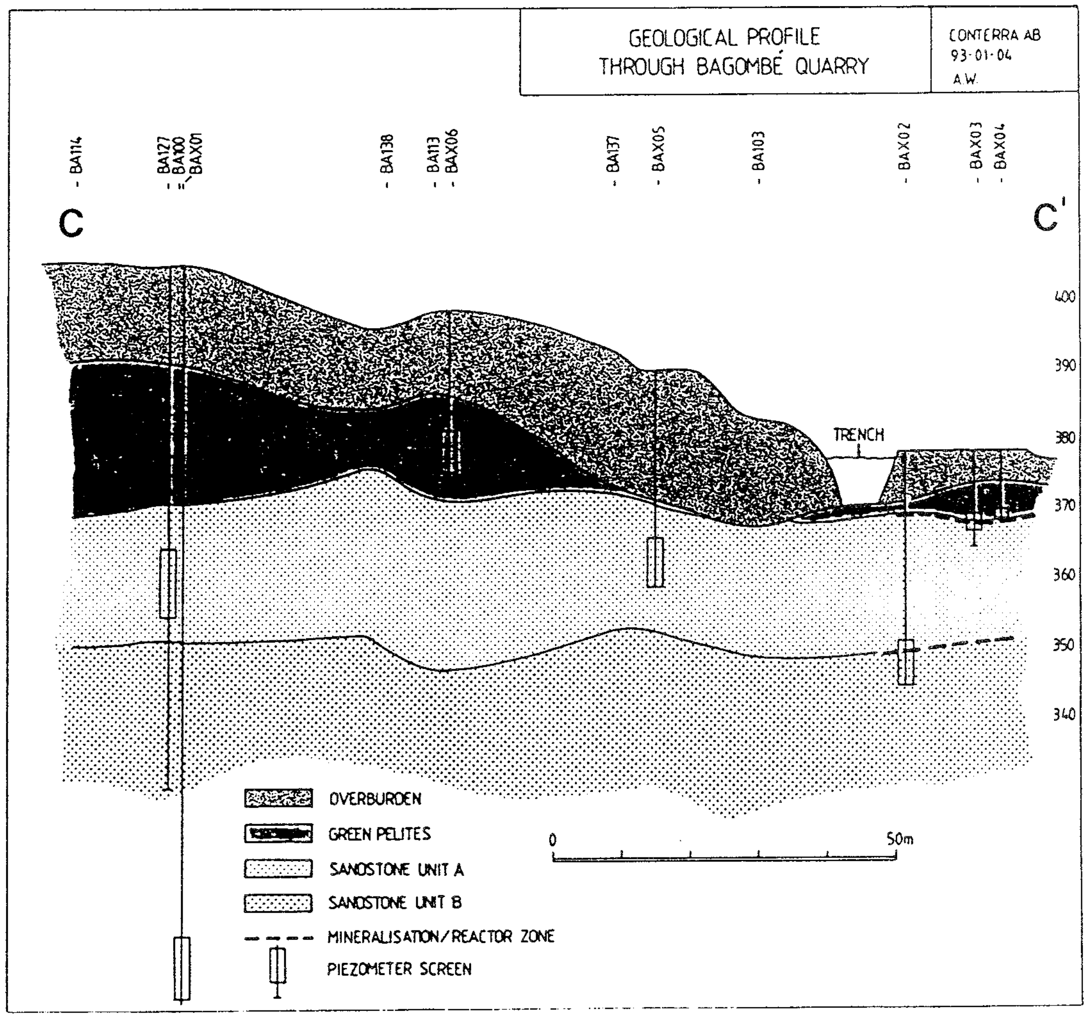


Figure 20-6. Bagombe conceptual geological section showing the location of the boreholes and piezometer positions.

The Bagombe reactor is close to the ground surface (about 30 m). The reactor zone has been well preserved due to its position in a recharge zone which prevents extensive oxidation to reach down to the uranium ore, see Figure 20-5.

From September to December 1992, a number of boreholes were drilled at Bagombe, see Figure 20-6, core samples were collected, hydraulic measurements and groundwater sampling performed.

To avoid contamination and hydraulic short-circuiting in order to obtain representative groundwater samples, bentonite was used to seal the borehole sections in addition to the packer system being used. Sampling and analysis will continue.

SKB also participates in the part of the analogue project which is centered around Oklo and Okelobondo. Mineral and groundwater samples are being analyzed and interpreted and the hydrogeology of the ore is being characterized for evaluation of the analogue properties.

20.4 OTHER ANALOGUE STUDIES

In addition to the previously described projects SKB is also participating in other studies. At Palmottu in Finland a uranium mineralization in granite is being investigated /20-11/. SKB is following the project as an observer. SKB has also been following the work within the Alligator Rivers Analogue Project, ARAP, by participation in the INTRAVAL project. The INTRAVAL project is managed by the Swedish Nuclear Power Inspectorate, SKI. The ARAP project is evaluating a uranium mineralization in Australia as an analogue to radionuclide dispersal and the project has been reported within INTRAVAL /20-12/.

In Jordan hyperalkaline groundwater has been found that was generated by water flowing through natural occurrences of burnt limestone /20-13/. The heat that burnt the lime was generated by spontaneous ignition of pyrite containing bituminous marl and subsequent combustion of the bitumen. Water reacting with burnt lime, CaO, produced the mineral portlandite, $Ca(OH)_2$ and the groundwater obtained a pH of around 12.5. Other typical cement phases were also produced and the reactions of these phases and the reactions between the rock minerals and the hyperalkaline water can be traced in the rock samples and groundwater samples. Young such occurrences from present day to thousands of years of age can be studied at Maqarin near the Yarmok River, see Figure 20-7. Older examples are found in Central Jordan, see Figure 20-8. Together with NAGRA and NIREX, SKB has been supporting the investigations as a natural analogue to the use of concrete in a waste repository. The area of Maqarin is rich in minerals and one objective of the investigation is the validation of geochemical codes for calculating solubility and speciation of radionuclides at high pH /20-14/.

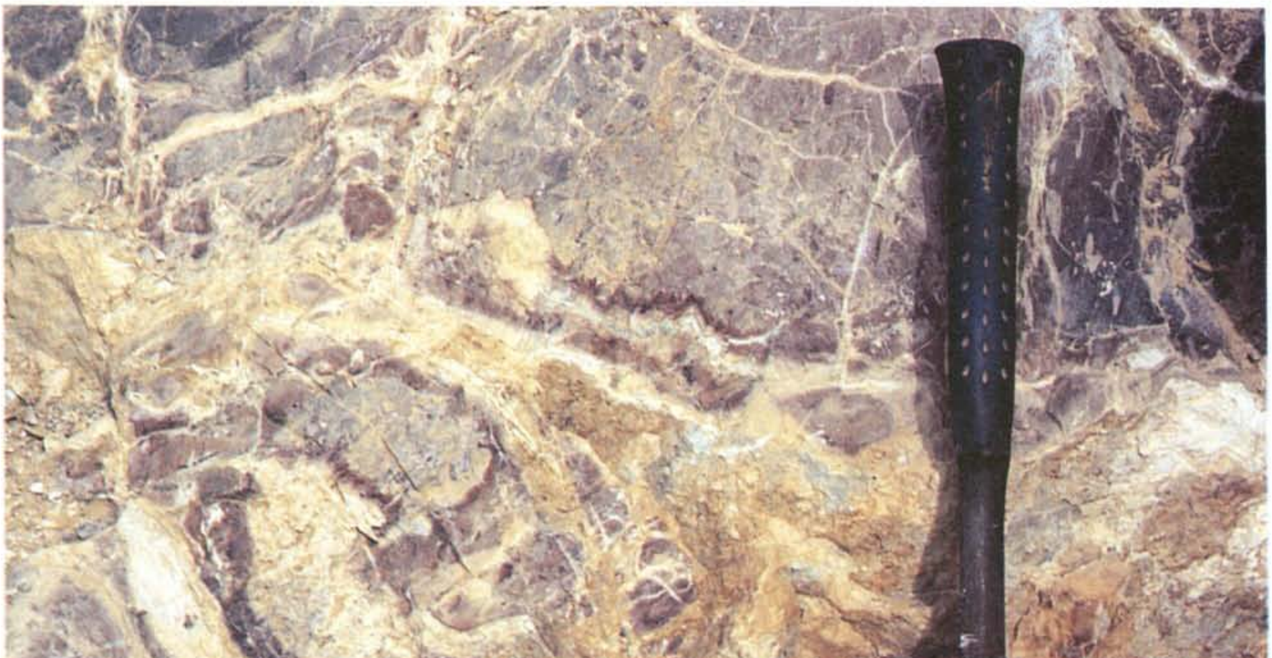


Figure 20-7. *Fresh seepages of hyperalcaline groundwater in Maqarin, Jordan.*

Figure 20-8. Fossil hyperalkaline reactions mineralogically preserved in Central Jordan. Age well above 10 000 years.

21 THE BIOSPHERE

The biosphere studies shall describe the transfer of radionuclides from the deep groundwater to biota and the subsequent dose to man from these nuclides. As the exact knowledge of systems and processes is not at hand, coarse compartment models have to be used to describe this nuclide transport. As direct validation of the models is impossible, the confidence in these models can only be increased by applying them to other short term problems like atmospheric fallout and by internationally comparing the results from different groups. This has been the basis for the Chernobyl studies and the participation in BIOMOVS. To increase the knowledge of the processes and parameters of systems, as they appear today, is the basis for the studies at the Äspö site.

The biosphere studies have to bridge not only the spatial distance between bedrock and man, but also the timescales involved.

The end point of any assessment study is yearly dose to man. A human being can be expected to exist for no more than 100 years and his living habits probably change each 10 years. Changes in land exploitation can occur from 10 to 1000 years. More dramatical changes in nature, as lakes drying up, eventuate in 1000 to 10000 years, iceages and changes in sea level in 10000 to 100000 years, while geological processes operate in timescales of several million years. This timescale thus covers six factors of ten.

The assumption that all processes are in a semi steady state, would simplify the treatment of biosphere. It is possible though, that some reservoir could accumulate radionuclides (not necessary with higher concentration) during a long time period. Due to some change in, for instance, land exploitation these nuclides can be released during a relatively shorter time period, giving a higher dose during that shorter time. One example of this is the ageing lake, where the sediment of a lake evolves into farmland. Other examples like accumulation in peat /21-1/ exist but no systematic study has been accomplished. The ageing lake example /21-2,3/ shows that these processes can occur but the effect in the assessment is limited.

21.1 VALIDATION OF MODELS: BIOMOVS

BIOMOVS (BIOspheric MOdel Validation Study) is an international cooperative study initiated in 1985 to test models designed to calculate the environmental transfer and bioaccumulation of radionuclides and other trace substances. To SKB this has been an opportunity to test the widely utilized modelling tool BIOPATH and the uncertainty tool PRISM in several applications. The first study was run for five years and ended in 1990. BIOMOVS I forcefully demonstrated the short-comings of our present capabilities for biosphere modelling /21-4/. Older models involving well studied pathways and relatively shortlived radionuclides (e.g. ^{137}Cs and ^{131}I) need improving, but the newer models for the longer lived radionuclides and less well studied pathways inspire little confidence. With values assigned to basic parameters differing between modellers by 3, 4 and even 5 factors of ten, with estimates of uncertainties about these parameters by the individual modellers covering a similar range, and with, in the worst cases, little or no overlapping of the uncertainty ranges, the situation clearly demands a remedy. Ideally, any modelling group given a specific scenario should calculate levels of activity in any commodity that agree to well within a factor of 2. BIOMOVS has shown that such a target is a long way off even for the older well studied pathways.

In 1991 the second phase BIOMOVS II was started at a workshop in Vienna in October. It is jointly managed by five organizations:

- The Atomic Energy Control Board of Canada (AECB),
- The Atomic Energy of Canada Limited (AECL) Research,
- Centro de Investigaciones Energeticas Medioambientales y Technologicas, Spain (CIEMAT),
- Empresa Nacional de Residuos Radioactivos SA, Spain (ENRESA),
- Swedish Radiation Protection Institute (SSI).

As all important exposure pathways, scenarios and reasons for differences in predictions, could not be addressed in BIOMOVS I, some of the goals and justifications remain valid for BIOMOVS II. The primary objectives of BIOMOVS II are threefold, namely:

- to test the accuracy of the predictions of environmental assessment models for selected contaminants and exposure scenarios,
- to explain differences in model predictions due to structural deficiencies, invalid assumptions and/or differences in selected input data, and
- to recommend priorities for future research to improve the accuracy of model predictions.

SKB has during 1992 actively taken part in the work with emphasis on the theme "Reference Biosphere Scenario for Long Time Assessment" as we believe it is of great value to get an international consensus how to deal with the modelling and conceptual uncertainties arising with time. Within this working group a list of FEPs (features events and processes) have been compiled during 1992. The proposed methodology and FEP list will be used and tested in a related theme "Complementary studies." In this theme a specific site is modelled by 10 different groups and the results and approach compared. The site specific data has been collected from a valley in Wellingen in Switzerland. The results are discussed with emphasis on the processes involved. SKB is also taking part in a validation study concerning ^{14}C in lakes. During 1992 plans were also made for a "model complexity" group trying to evaluate the effect of more complex model approach, and a "natural analogues" group where models can be tested against the nuclide content in different parts of the biosphere as a result of naturally radionuclide sources.

21.2 VALIDATION OF MODELS: VAMP

SKB is 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) /21-6/. In this programme modelling of ^{137}Cs in lakes and uncertainty analysis is intercompared between several working groups from several countries. The lake model and the codes BIOPATH and PRISM are the tools SKB tries to validate in this study. Some preliminary results have been reported in /21-9/.

For most of the lakes the predicted to observed ratio (P/O) was within a factor of three for water and fish. Overestimation was most pronounced in a Norwegian lake, probably due to the snow cover during fallout. The main difficulty seams to be the estimation of the water concentrations. If these estimations are close to the measured real concentrations, then the calculated concentration in fish will also be realistic. The uncertainty analysis showed that the source of uncertainty was different between the lakes.

21.3 VALIDATION OF MODELS: PSAC 1b

This international OECD/NEA exercise "Probabilistic Systems Assessment Code (PSAC) User Group of the OECD Nuclear Energy Agency" PSACOIN level 1b deals with the verification of codes used in biospheric modelling and uncertainty analyses. Most of the work was done during 1991 and a report was prepared during 1992.

The results showed good agreement between the 7 participating codes. The major uncertainty in the results was due to the transport processes and not to the exposure pathways. The parameters, that were most critical for total uncertainty, were not the same ones in the beginning as in the end of the time period studied.

21.4 SITE SPECIFIC STUDIES OF RECIPIENTS AT ÄSPÖ

The long term transport from the geosphere to the biosphere has been addressed for a specific site in this project.

Postglacial and glacial sediments and soils have been studied in the archipelago around Äspö, with special interest in the influence of discharging groundwater.

This project is divided into the following three phases:

Phase I	Prestudy to find out what data is available, a preliminary fieldstudy and planning,
Phase II	Recipient studies regarding surface and ground water and water flows through the sounds, estuaries and coastal area,
Phase III	Recipient evolution – modelling of the likely evolution of the coastal region in the time perspective of 1000 to 10000 years.

In analyzing the mineral composition and natural radiation in sediments and soil samples at the site, it will hopefully be possibly to draw some conclusions about the history of long term radionuclide transport. Comparisons with future situations when flow patterns may have changed, may also be valuable.

Sediment samples (0.5-6 m sediment depth) from the waters around Äspö have been taken at 22 sites. 15 of these are concentrated to the area SE Äspö. Soil samples (0.4-2 m depth) from 4 sites on Äspö have been characterized and in some cases neutron activation analysis was used. Measurements with gammameter have been made along two well determined profiles, complementing previous measurements /21-5/. The sediment cores need long time for evaluation but they can be stored for future more extensive analysis.

About 30 elements have been studied in the solid sediment phase (Na, Cs, Br, Fe, Th, U and REE). In the pore water Na, K, Ca, Br, Cl, SO_4, NO_3 and others were analysed.

Some anomalities concerning Cs, Rb, Cl and Br, indicate that the rising groundwater influences the sediments. The composition of the groundwater (types of ground- and marine waters) was addressed by factorial analyses. Samples from above the fracture zone NE1 show a pore water with some baltic water, mixed either with surface groundwater or with older more saline marine waters /21-8/. This is consistent with the very variable and complicated flow pattern which is assumed in the fracture zone /21-9,10,11/.

A primary model approach has been made. This model is aimed for estimating the radionuclide transport from a costal repository at a site like Äspö. Modelling this area will, via uncertainty analysis, indicate where our knowledge may be weak and more data collection can be fruitful.

The flow patterns in surrounding bogs have been recorded for use in this coastal zone model.

A second model tries to describe the dynamics in groundwater flow for the shallow parts of Äspö and the coastal areas in the time perspective of the last ice age up till now. In this model the evaluation of the sediment cores is an important input.

21.5 SKB 91 SAFETY ASSESSMENT

The special safety study SKB 91 was finalized during 1992. It did in general not deal with the uncertainties in the biosphere but used a set of constant dose conversion factors. These factors related release rate from the far field directly to dose to individuals in a critical group. They apply to a relatively conservative situation that gives high doses but still has a high probability of occurring sometime within the studied timescale. Three cases were studied, one central case with radionuclides reaching biosphere via a lake and partly via a well, one with all radionuclides reaching the well and one case where the sea is the recipient.

This standard biosphere constituted of a well and lake with adjacent farming-land /21-12,13,14,15/. It was assumed that a fraction of 1% of the activity reached a well directly while the remaining part was directly diluted into lake water. Ten exposure pathways originating from activity in well and lake water were considered. The ecosystem was assumed to be similar to present conditions in Sweden. So was also the diet and living habits. No delay or reduction of activity by accumulation in the interphase geosphere and biosphere was considered. A seven compartment model of the studied biosphere was designed. The BIOPATH code was used for solving the differential equations and calculating the doses. Adults and five year old children were considered.

The uncertainty in the results due to the uncertainty in input parameter values were examined with the PRISM-system. The major contribution to the uncertainty in the results was the dilution volume for the nuclides in the groundwater. This dilution was studied by varying reservoir volumes as well as varying the fraction of activity reaching the water in the well.

Earlier calculations of the doses from these long-lived nuclides showed that the internal exposure dominates the exposure for the nuclides considered. The only external exposure considered is from ground. This represents staying on the fields and the garden plots.

The drinking water from the well is the dominant pathway for most nuclides.

21.6 THE DISTRIBUTION OF RADIONUCLIDES IN SOILS AND SEDIMENTS

The modelling of transport in soils and sediments have been heavily relying on the sorption assumption expressed as a single K_d-value. There is a strong need for techniques to model this important part of the biosphere models in a better way /21-16,17,18/. Thus, a project was initiated in 1989 to better understand the long-term modelling of the accumulation of nuclides in sediments and soils. This can be achieved by:

- extending the understanding of sorption phenomena relevant to both the biosphere and the geosphere,
- using thermodynamic data and sorption data to explain and hopefully reduce uncertainty within the biosphere modelling (as the big intervals of uncertainty today mostly are the sequel of the highly variable K_d-values found in the literature).

The study has been delayed but reported in 1992. Some main difficulties were:

- Organic substances wont fit into thermodynamics.
- Redox fronts in organic matter.
- Biological processes as bioturbation.

For I these difficulties have had great impact and this nuclide had to be dropped from the study.

The results show that the work with surface complexion model for actinides increase the understanding of both laboratory measurements as well as studies of natural systems. The surface complexation model could estimate the dependence of K_d as a function of important chemical parameters.

The power of the surface complexation model is that equilibrium constants, obtained under well laboratory conditions on well determined minerals, easily can be used to estimate sorbtion under a much wider variety of conditions. K_d-value for Ra could be more precisely determined if the Ca concentration in the environment was known. It was not possible to estimate the feasibility of the method for Cs.

21.7 THE CHERNOBYL FALLOUT

In order to utilize the Chernobyl fallout for validation of nuclide migration models in the shallow groundwaters and the upper soil layer, samples have been collected and measurements have been made in two Swedish areas since 1986 /21-19,20,21,22/. Model evaluations are currently performed using both compartment models /21-20/ and continuous flow models /21-23/. Another main issue is the chemical properties of the observed radionuclides, currently studied by migration in soil /21-22/.

Measurements of radionuclides originating from the Chernobyl accident in samples of deep and superficial ground waters, soil profiles and dwell sediment from the Gideå and Finnsjön areas have been performed. The studied radionuclides are: ^{54}Mn, ^{60}Co, ^{110m}Ag, ^{106}Ru, ^{125}Sb, ^{134}Cs and ^{137}Cs. As expected there is a strong correlation between groundwater table fluctuations and precipitation and temperature fluctuations at different periods during the year.

The measurements of water from the deep core drillhole KGI02 over 3 years indicate an activity pulse of long-lived radionuclides, present in the Chernobyl fallout, at all sections (28-96 m, 97-106 m and 107 m-). This is surprising since the water flow at these depths is very low (approx. 0.05 l/min). The flow is not affected by the fluctuations in precipitation indicating that there is no short circuit and that the radionuclides are transported quite fast through the bedrock. The ^{106}Ru peak arrives 263 days after the fallout to the 96-107 m level, while the peak of ^{60}Co and ^{137}Cs arrives at 599 respectively 516 days. This transport speed, discussed in /21-23,24,25/, is surprising as some previous experiments /21-26/ have shown that i.e. Cs is strongly sorbed. Other experiments have shown a minor amount migrating almost without any retention /21-27/. A possible explanation can be the speciation or the existence of organic complexes, colloids or particles. The speciation analysis have shown that Cs is not transported in the cationic Cs^+ form, as it is found in the anion exchanger or charcoal in the deep groundwater. In the surface waters, on the other hand, it is transported as a cation.

The migration of radionuclides in the soil profiles shows that the transport in till is relatively slow compared to sand and peat (the profiles were sampled each year since 1986). Other conclusions from this study are:

- ^{60}Co moves relatively fast with 50% of the activity found in the upper 5 cm of sand and till.
- ^{106}Ru seems to move very fast and 50% of the activity is found in the upper 7 cm in sand.
- ^{110m}Ag has moved very moderately but it should be observed that this nuclide is difficult to measure because of the low activity.
- ^{125}Sb seems to move very fast with 50% of the activity found in the upper 7 cm in till.
- ^{134}Cs and ^{137}Cs can be found with 50% of the activity in the upper 3 cm in sand and till.

Measurements of radionuclide content in sediment profile samples taken in a dwell indicate a very fast migration through the sediment. This is shown by an almost straight radionuclide concentration profile versus depth.

The data from the area has been compiled into models using both compartment and finite element methods. To get a better understanding of the total flows a soil map has been prepared for the area /21-28/. The modelling results serve as a quality control on the measurements and indicate among other things that only about 5% of the initially deposited ^{137}Cs has left the area after 5 years /21-29/.

22 INTERNATIONAL COOPERATION

An important part of SKBs programme is to follow the corresponding research and development work conducted in other countries and to participate in international projects within the field of nuclear waste management.

These efforts give positive results in many ways e.g.:

- contributions to method- and model development,
- broadened and strengthened databases,
- exploration of alternatives for repository and barrier design, material selection etc.,
- insights in programmes to broaden the public confidence in waste management systems.

The international work gives a perspective to the domestic programme and is an aid to the SKB strive for maintaining state-of-the art in relevant scientific areas of nuclear waste management.

22.1 SKBs BILATERAL AGREEMENTS WITH FOREIGN ORGANIZATIONS

SKB has signed formal bilateral agreements with the following organizations in other countries:

- USA – US DOE (Department of Energy),
- Canada – AECL (Atomic Energy of Canada Ltd) and ONTARIO HYDRO,
- Switzerland – NAGRA (Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle),
- France – CEA (Commissariat à l'Energie Atomique), ANDRA, DCC and IPSN,
- EC – EUROATOM,
- Finland – TVO and IVO,
- Russia – former SCUAE, current name of organisation unclear,
- Japan – JNFL (Japan Nuclear Fuel Ltd.).

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-year 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.

22.2 COOPERATION WITH TVO, FINLAND

A regular exchange of experience and technology for site investigation is taking place. Furthermore, Finnish representatives are included in the reference group for the Hard Rock Laboratory.

Regarding waste canisters SKB and TVO have continued the evaluations of production methods and costs of an Advanced Cold Process Canister with an inner steel canister and an outer corrosion shield of copper. Models of canisters on a scale of 1:4 have been jointly developed for the purpose of demonstration.

SKB is also following the investigations at a uranium mineralization in Palmottu as an observer.

The joint work and information exchange on alternative designs for waste repositories (the PASS-project) was completed in 1992.

22.3 COOPERATION WITH CEA, FRANCE

22.3.1 Clay

SKB is currently cooperating with CEA in clay studies. The cooperation has included coordination of research projects and information exchange regarding relationships between the microstructure, mineralogy etc of smectite clays and the influence of temperature and irradiation. Hydrothermal tests and irradiation have been carried out during year-long experiments in the laboratory. The 4-year test conducted at Stripa with highly compacted French smectite clay in a simulated deposition environment at approx. 170°C has been completed, see section 16.1. The results were evaluated together with laboratory tested radiated samples of the french clay and SKB reference clay MX-80. The cooperation has provided good opportunities for comparisons between the two countries' reference clays for buffer materials, methods for measurement of properties, swelling pressure, hydraulic conductivity, thermal conductivity etc, and technical methods for deposition.

22.3.2 Natural analogues

SKB is engaged in the CEC sponsored natural analogue project in Oklo which CEA is managing, see section 20.3.

22.3.3 Instruments

IPSN/CEA in Cadarache, France, has been contracted for the development work of a borehole probe (CHEMLAB), see section 18.1.6.

22.4 COOPERATION WITH AECL, CANADA

22.4.1 Characterization of the 240 level of URL

AECL and SKB signed in April 1987 an agreement on cooperation for characterization of the 245 Level in the Underground Research Laboratory situated in a granitic batholite in Manitoba, Canada.

During 1992 the URL Characterization Program has continued. The program comprises a broad spectrum of activities such as: geological mapping, testing of rock properties, geomechanical and geophysical measurements, acoustic emission monitoring, microseismic monitoring, hydrogeological monitoring etc. The operating phase experiments include studies of solute transport in highly as well as moderately fractured rock. Furthermore rock strength, rock stress and rock yield are major engineering issues relevant to excavation stability and sealing system integrity. A mine-by experiment has commenced in order to improve underground characterization methods including monitoring instrumentation and data management systems. A buffer/container experiment is in progress to collect a database for improvement and validation of models of coupled processes that will occur in highly compacted clay buffer and surrounding rock mass.

22.4.2 Natural analogues

Concerning the joint AECL/SKB work at Cigar Lake see section 20.2.

22.5 COOPERATION WITH EURATOM, CEC

22.5.1 COCO

The working group COCO (Colloids and Complexes) was formed by CEC to explore the importance of colloids and organic complexes for the migration of radionuclides. An important part of the cooperation is comparative experiments with different methods used at different laboratories. SKB is supporting the participation of a Swedish specialist active within the field.

22.5.2 CHEMVAL

The first phase of the CEC project CHEMVAL for verification and validation of chemical equilibrium calculation programs and coupled models for geochemistry transport was finalized and reported during 1990. A new phase of the CHEMVAL project called CHEMVAL2 started up during 1991 with participants from the EC countries, Sweden, Finland and Switzerland. The project will run from 1991-1994 and will comprise temperature effects, ion strength effects, organic complexes, sorption, coprecipitation and coupled geochemical transport, see section 18.2.1.

22.5.3 Natural Analogue Working Group

Natural Analogue Working Group (NAWG) is an international group working with natural analogues and their use in the safety assessment modelling. It's organized by CEC.

SKB has been represented in this group since its start in 1985. Presently one of SKB consultants, Dr John Smellie, is the chairman of the group.

22.6 COOPERATION WITHIN OECD NUCLEAR ENERGY AGENCY

22.6.1 RWMC

One of OECD/NEA's principal areas of cooperation is radioactive waste management in the member countries. These questions are dealt with by the **Radioactive Waste Management Committee (RWMC)**, where SKB is represented through Per-Eric Ahlström. Some work is carried out in joint international projects, and working groups are formed to facilitate information exchange or prepare material as a basis for joint opinions or coordination.

Seminars and workshops are arranged within important areas to document and discuss the state of development and the direction of future work.

The groups and projects within the area of radioactive waste management where SKB during 1992 was providing personnel or funding are listed below.

PAAG (Performance Assessment Advisory Group) functions in an advisory capacity to RWMC in matters pertaining to cooperation on means and methods for performance and safety analyses of final disposal systems.
Member from SKB: Tönis Papp

SEDE (Site evaluation and design of Experiments for Radioactive Waste Disposal) functions in an advisory capacity to RWMC in matters pertaining to the activities of experimental work in the member countries. The group will organize a workshop in 1992 on the use of palaeohydrogeological evidence in site characterization.
Member from SKB: Lars-Olof Ericsson

PSAG (Probabilistic Safety Assessment Group) is a cooperation group between those who develop and those

who use mathematical models for probabilistic analyses of repository systems. The emphasis lies on coordinating the development and comparing the quality of the models.
Member from SKB: Nils Kjellbert

Cooperative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects is a forum for information exchange and cooperation on various decommissioning projects all over the world.
Member from SKB: Hans Forsström. SKB is also sponsoring a programme coordinator, Shankar Menon, Studsvik Energiteknik AB.

Expert Group on Geochemical Modelling and Data deals with matters of common interest within geochemistry, including the buildup of a common thermodynamic database TDB and augmentation of the database for sorption data, SDB.
Member from SKB: Fred Karlsson

The Stripa Project. (See Part V)
Members from SKB: P-E Ahlström (chairman of Joint Technical Committee), Hans Carlsson, SGAB (member of Joint Technical Committee), Bengt Stillborg (Project Manager) and Karl-Erik Almén (assistant Project Manager)

Working Group on the Assessment of Future Human Actions at Radioactive Waste Disposal Sites deals with different aspects on human intrusion into waste repositories. The group was initiated in 1990.
Member from SKB: Torsten Eng.

22.6.2 TDB

The TDB Project (Thermochemical Data Base) is under the direction of OECD/NEA. The goal is to develop a chemical thermodynamic database for a number of elements that are of importance for the safety assessment of the final disposal of radioactive waste. The development of the database entails not only collecting and storing published data, but also critical review. Review is carried out by a group of international experts selected for each element. At present the work is concentrated on neptunium, plutonium, americium and technetium. The uranium database was the first to be completed.

The TDB Project is a very important effort to develop a well documented, reviewed and internationally accepted database. SKB is supporting the activity and Swedish experts are participating in the review work. For SKB, as well as for other participants, it will naturally be necessary to have an operational database available before TDB for different calculation purposes. However, the results from TDB will be incorporated as they become available. A good example of this is the Uranium Database at SKB.

22.6.3 INTRAVAL

INTRAVAL is an international project whose purpose is to validate calculation models for radionuclide transport in the geosphere. The project is a follow-up of the previous projects HYDROCOIN and INTRACOIN. All of these projects were initiated by SKI, which also appointed the secretariat that coordinates the work within INTRAVAL.

A total of 14 test cases were included in the project phase I, which involved evaluation of results of selected laboratory tests, field tests and studies of natural analogues. In many of the cases, it was possible for different model groups to perform predictive modelling before the measurement results had become available.

Five of the fourteen test cases were SKB-linked:

- laboratory tests of migration in overcored fractures/KTH,
- tracer tests at Finnsjön within the fracture zone project/SGAB,
- Stripa 3D migration/KTH,
- Poços de Caldas Project,
- colloid transport/BGS,
- redox front/KTH.

The detailed results of INTRAVAL phase I were published during 1991.

Phase II of INTRAVAL started in 1990. This phase emphasizes on validation efforts based on field studies and natural analogues. The number of test cases are less than in phase I and cover validation issues like scale dependency, heterogeneity and coupled processes. SKB is supplying data for this study as indicated in section 12.5.

22.7 COOPERATION WITHIN IAEA

Cooperation has during 1992 also been conducted within the International Atomic Energy Agency, IAEA, concerning the management of radioactive waste.

The cooperation is conducted in different ways, including the publication of reports consisting of:

- proceedings from international symposia,
- guidelines and standards within established areas of activity,
- status reports and methodology descriptions within important areas undergoing rapid development.

IAEA has an expert advisory group for its waste management programme (the International Waste Management Advisory Committee, INWAG) and arranges for information exchanges within different special areas through Joint Research Programmes. IAEA publishes an annual catalogue on current research projects within the waste management field in the member countries.

An important new IAEA initiative is the RADWASS programme to work out international safety standards and guidelines. SKB will participate in the Standing Technical Committee for Disposal within the RADWASS programme.

22.7.1 VAMP

SKB is participating in an IAEA/CEC program on "Validation of Models on the Transfer of Radionuclides in Terrestial, Urban and Aquatic Environment and Acquisition of Data for that Purpose" (VAMP), see section 21.2.

22.8 OTHER INTERNATIONAL COOPERATION

22.8.1 BIOMOVS

As indicated in section 21.1 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 1992 taken active part in the reference scenario definition work as well as a validation study concerning ^{14}C in lakes.

22.8.2 DECOVALEX

Interest in developing coupled models has increased in recent years. The purpose is to be able to describe conditions in the near field of a repository in particular with greater realism. Within the framework of the DECOVALEX project (international cooperative project for the DEvelopment of COupled models and their VALidation against EXperiments in nuclear waste isolation), development and verification of coupled thermo-hydro-mechanical models is being conducted. SKI initiated the project during 1992 and is also the organization in charge of its execution. Nine countries are participating in the project which will run for three years.
Member from SKB: Lars-Olof Ericsson

22.9 INTERNATIONAL COOPERATION IN THE ÄSPÖ HARD ROCK LABORATORY

As is mentioned in Chapter 19 the Äspö HRL has gained great international interest. The following organizations have up to the end of 1992 signed agreements to cooperate in joint work at the Äspö HRL:

- AECL, Canada,
- PNC, Japan,
- CRIEPI, Japan,
- ANDRA, France
- TVO, Finland,
- UK NIREX, UK,
- USDOE, USA.

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 1992 /22-1/.

23 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 92 and Plan 92,
- in the series of SKB Technical Reports, in contributions to scientific journals, symposia and conferences in different subject areas, see Appendix 2,
- in SKB Arbetsrapporter,
- in internal SKB memos,
- in technical memos and notes.

Further, the bulk of basic data from geological site characterization activities, spent fuel studies etc. are collected and stored in the electronic data base systems at SKB.

23.1 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 1992 Technical Reports are included in part V of this Annual Report.

23.2 CONTRIBUTIONS TO PUBLICATIONS, SEMINARS ETC

The contributions to conferences, symposia and scientific journals have been extensive during 1992, see Appendix 2.

Both SKB own staff as well as the contractors of SKB have been involved in this work.

23.3 SKB GEOLOGICAL DATA BASE SYSTEM

The data from the geological site investigations, including the Äspö Hard Rock Laboratory, is managed by and brought together in GEOTAB, a common database system. The aim of this database system is threefold, namely to

- facilitate retrieval and combination of data from different disciplines,
- provide an archive, independent of the different data collecting contractors,
- assure the quality of measurements and calculations performed.

23.3.1 Technical

This database is a so called relational database, giving the investigator the possibility to freely select and combine information. The stored data can be kept at the high initial quality due to the implied data structure. In 1992 the old database was prepared to be ported from a VAX system running the database manager MIMER to a SUN (UNIX) workstation running INGRES. This port lead to better response times, better integration with other programs (PC and UNIX), better consistency control and better auditing and logging. The codes are generally written in either the language C, using 3-G calls to the database manager, or in the 4-GL languages VISION ocr WINDOWS-4GL (INGRES). Typical response times are 1 second to 2 minutes for a selected retrieval from two combined tables with 1.000 records in each.

23.3.2 Structure

The data tables are free tables, but to facilitate retrieval they are also hierarchically structured in "sciences", "subjects" and "methods". The set of tables making up a "method" are normally one or two "flyleaf" tables, a comment table, some data tables and possibly some calculated data tables. The "flyleaf" tables contain information about who, when, equipment and other features of the measurement. A set of "methods" makes a "subject" and a set of "subjects" and "methods" makes a "science".

The data acquisition techniques are documented in technical reports /23-1, 2, 3, 4, 5/. As new measuring methods and data acquisition techniques are applied, new methods and tables are created and the documentation is completed with working reports. Overview /23-6/ and Users Guide /23-7/ are of course important documents for the occasional user. All documentation is in English.

23.3.3 Content

The data stored in the database is of course limited to what can be captured in letters and digits. The open concept, however, allows other programs to directly interact with the database thus extending the use of the database to geometrical or graphical information.

The database now contains surface data from 43 sites and data from 489 boreholes in many of these. Total borehole length that is investigated is about 150 km.

23.3.4 Statistics

Data are as mentioned above structured in "sciences", "subjects" and "methods" and tables. Currently there are about 500 tables containing 4932 columns and about 4.1 millions of tupels (lines). In addition there is one log table corresponding to each data table and some 50 tables containing system and meta data (information about the hierarchical structure), totalling to more than 1000 tables. The integrity of the data is maintained with a system of insert and update rules, currently about 1700 rules and 1250 procedures.

Total data volume is about 1 Gbyte.

23.3.5 QA routines

New data is continuously fed into the system with a time lag varying between one day and some weeks, depending on which quality-assurance routines that must be applied. In some cases the primary data is collected in dBase format, checked and directly transferred into GEOTAB. After entry in GEOTAB the stored data is checked again by the investigator and signed off.

23.4 SKB BIBLIOGRAPHICAL DATABASE

SKB has built up a database containing bibliographical data and abstracts on all reports currently available in the SKB library. The database, called BIBAS, contained by the end of 1992 about 8 000 references. The software used to manage the database is AskSam which has a powerful free-text search capability.

23.5 COMPUTER SYSTEM AT SKB

23.5.1 Computer network – LAN and WAN

The computers managed by SKB are placed in three locations; the office at Brahegatan, the computer room at Birger Jarlsgatan (both in Stockholm) and at the office of Äspö Hard Rock Laboratory, north of Oskarshamn. The computers at all three sites are connected to local area network (LAN) of the physical type "ethernet". The three LANs are connected via two pairs of ethernet bridges, operating over 64kbps lines, making the three segments appear as one.

Two standard protocols are used in the network – TCP/IP and DECNET. TCP/IP is used by all connected computers (nodes) and used for PCnetworking, terminal sessions, mail and file transfer. The mail systems in all multiuser machines (including the VMS/VAX) are integrated and externally connected to the E-mail international mailing system, covering 90% of all UNIX machines worldwide. The more proprietary but well known DECNET is also used for terminal sessions and file transfer.

The networking software used for PC networking is PCNFS from SUN Microsystems. The main use is to keep a common file system, making document transfer very easy and the common software and standards consistent throughout the company. The servers can be one or several UNIX computer with NFS and currently 2 SUN386i and 3 SUN Sparc workstations with 10 Gbyte on 10 disks, 3 2.3 Gb Exabyte and 4 QIC tape stations are used. 2 CD-Rom stations are also available. A PC in this LAN is served by several file servers simultaneously and, to improve performance, one server has been sited at Äspö.

As SKB is contracting several companies for different work in the computer system a wide area network (WAN) for terminal lines has emerged during the years. Currently 58 lines are connected to the computers in the computer room. Of these 2 are local, 9 are used as dialup lines (2 in Gothenburg) and the rest connected via multiplexors and leased lines to 9 different sites in Stockholm and to Luleå and Gothenburg. The system is very open in the sense that an user at any node can log into any other node (except PCs), depending on his rights.

23.5.2 Minisupercomputer

The CONVEX C210 was during 1991 upgraded to a 2-processor vector computer. It has been very easy to operate, running 24 hours a day with no major problems and with the expected vector capacity of about 24 Mflops (floating point operations per second). The operating system is a BSD UNIX 4.3 system with system V extensions. To facilitate communication and migration into this system the CONVEX also provides VAX commands, DECNET, the VAX editor EDT and direct batch queues operated from the VAX 11/750. This software enables all consultants with access to the VAX system to directly access the CONVEX as well. The current hardware configuration is 128 Mbyte main memory, 6 Gbyte on 6 disks, a 6250 bpi tape drive, 2 ethernet transceivers and 16 asyncronous ports.

23.5.3 Minicomputer

The VAX 11/750 is now conceptually more than 13 years old and does not cope very well with the computing demands of today. However, it is reasonably good in reading and writing to disks and is currently intensively used for storing data and archiving backups from the other machines. The machine configuration now includes 12 Mbytes main memory, 2.3 Gbytes on 5 disks, a 2.3 Gbyte Exabyte and a 1600 bpi tape station, ethernet transceiver and 40 asyncronous ports. The software is rather conventional but includes a TCP/IP suite from Carnegie Melon to make the VAX communicate with the UNIX world.

23.5.4 Workstations and measuring system

Currently the 5 SUN workstations are mainly used as PC network servers but they are of course also used as personal workstations.

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 the SUN workstation and accessible from the all the terminals in the WAN.

REFERENCES

Chapter 11

11-1 SKB
1992
PLAN 92. Cost for management of the radioactive waste from nuclear power production.
SKB Technical Report TR 92-24, Stockholm

11-2 SKB
1989
WP-Cave – assessment of feasibility, safety and development potential.
SKB Technical Report TR 89-20, Stockholm

11-3 SKB
1988
SKB Annual Report 1988.
SKB Technical Report TR 88-32, Stockholm

11-4 SKB
1993
Project Alternative Systems Study (PASS). Final report.
SKB Technical Report TR 93-04, Stockholm

11-5 Sandstedt H, Wichmann C, Pusch R, Börgesson L, Lönnerberg B
1991
Storage of nuclear waste in long boreholes.
SKB Technical Report TR 91-35, Stockholm

11-6 Juhlin C, Sandstedt H
1989
Storage of nuclear waste in very deep boreholes: feasibility study and assessment of economic potential.
SKB Technical Report TR 89-39, Stockholm

11-7 Hosaluk L J, Teper B
1985
Status of alternative container design for fuel waste disposal.
Proceedings of the nineteenth information meeting of the nuclear fuel waste management program, volume I, Atomic Energy of Canada Ltd., Pinawa

11-8 Lawrence L H, Shoesmith D W, Ikeda B M, King F
1991
Lifetime of titanium and copper containers for the disposal of used nuclear fuel.
Scientific Basis for Nuclear Waste Management XV. Symposium held November 4-7. 1991, Strasbourg, Franc. Materials Research Society Symposium Proceedings, volume 257, pp 439-446. MRS, Pittsburgh, Pennsylvania

11-9 Nilsson F
1992
Mechanical integrity of canisters.
SKB Technical Report TR 92-45, Stockholm

11-10 Olsson L, Sandstedt H
1992
Project on Alternative Systems Study – PASS. Comparison of technology of KBS-3, MLH, VLH and VDH concepts by using an expert group.
SKB Technical Report TR 92-42, Stockholm

11-11 Birgersson L, Skagius K, Wiborgh M, Widén H
1992
Project on Alternative Systems Study – PASS. Analysis of performance and long-term safety of repository concepts.
SKB Technical Report TR 92-43, Stockholm

11-12 Scientific Industrial Company on superdeep drilling and comprehensive Investigation of the Earth's Interior, NEDRA
1992
Characterization of crystalline rocks in deep boreholes. the Kola, Krivoy Rog and Tyrnauz boreholes.
SKB Technical Report TR 92-39, Stockholm

11-13 Ageskog L, Högbom T
1992
Project on Alternative Systems Study – PASS. Cost comparison of repository systems.
SKB Technical Report TR 92-44, Stockholm

Chapter 12

12-1 Romero L, Moreno L, Neretnieks I
November 1992
Release calculations in a repository of the very long tunnel type.
SKB Technical Report TR 92-29, Stockholm

12-2 Romero L, Moreno L, Neretnieks I
October 1991
A compartment model for solute transport in the near field of a repository for high level waste.
SKB Technical Report TR 91-48, Stockholm

12-3 Nyman C
February 1993
Development of "CHEMFRONTS", a coupled transport and geochemical program to handle sharp reaction fronts.
Licenciate thesis, Royal Institute of Technology KAT 93/02

12-4 Wanner H, Wersin P, Sierro N
November 1992
Thermodynamic modelling of bentonite-groundwater interaction and implications for near-field chemistry in a repository for spent fuel.
SKB Technical Report TR 92-37, Stockholm

12-5 SKB 91
Final disposal of spent nuclear fuel. Importance of the bedrock for safety.
SKB Technical Report TR 92-20, Stockholm

12-6 Geier J, Doe T
1992
Discrete fracture modelling of the Finnsjön rock mass – Supplementary interpretations of actual and simulated well test data.
SKB Arbetsrapport AR 92-45, Uppsala

12-7 Moreno L, Neretnieks I
1991
Fluid and solute transport in a network of channels.
SKB Technical Report TR 91-44, Stockholm

12-8 Follin S
1992
Numerical calculations on heterogeneity of groundwater flow.
SKB Technical Report TR 92-14, Stockholm

12-9 Follin S
1992
On the interpretation of double-packer tests in heterogeneous porous media:
Numerical simulations using the stochastic continuum analogue.
SKB Technical Report TR 92-36, Lund

12-10 Kung C S, Cvetkovic V, Winberg A
1992
Calibration and validation of a stochastic continuum model using the Finnsjön dipole tracer test. A contribution to INTRAVAL phase 2.
SKB Technical Report TR 92-35, Stockholm

Chapter 13

13-1 R&D-Programme 89, Parts I-II. Handling and final disposal of nuclear waste. Programme for research, development and other measures.
September 1989
SKB, Stockholm

13-2 McEwen T, Balch C
1992
Review of the status of siting radioactive waste repositories.
SKB Djupförvar PR 44-93-001 (in print)

13-3 Tunbrant S
1992
Sammanställning och analys av tillämpliga lagar vid lokalisering av ett djupförvar för använt kärnbränsle.
("Summary and analysis of legislation applicable to the siting of a deep repository for spent nuclear fuel".)
SKB Progress Report LOK 92-08, Stockholm

Chapter 14

14-1 Forsyth R S and Werme L O
1991
Spent fuel corrosion and dissolution.
J. Nucl. Mater. 190 pp. 3-19

14-2 Janeczek J and Ewing R C
1993
Oxidation of Uraninite.
SKB Technical Report TR 93-17, Stockholm

14-3 Casas I, Cera E and Bruno J
1993
Kinetic studies of natural uranium minerals for the long-term evolution of spent nuclear fuel under oxidizing conditions.
Mat. Res. Soc. Symp. Proc. Vol 294, pp. 521-526

Chapter 15

15-1 Nilsson F (Compiled by)
1992
Mechanical integrity of canisters.
SKB Technical Report TR 92-45, Stockholm

15-2 Henderson P J, Österberg J-O and Ivarsson B G
1992
Low temperature creep of copper intended for nuclear waste containers.
SKB Technical Report TR 92-04, Stockholm

15-3 Josefson B L, Karlsson L, Lindgren L-E and Jonsson M
1992
Thermo-mechanical FE-analysis of butt-welding of a Cu-Fe canister for spent nuclear fuel.
SKB Technical Report TR 92-27, Stockholm
Also presented at the 12th International Conference on Structural Mechanics in Reactor Technology, in Stuttgart, Germany, Aug 15-20, 1993

15-4 Josefson B L, Karlsson L and Häggblad H-Å
1993
Stress redistribution and void growth in butt-welded canisters for spent nuclear fuel.
SKB Technical Report TR 93-01, Stockholm

15-5 Börgesson L
1992
Interaction between rock, bentonite buffer and canister. FEM calculations of some mechanical effects on the canister in different disposal concepts.
SKB Technical Report TR 92-30, Stockholm

15-6 Werme L, Sellin P and Kjellbert N
1992
Copper canisters for nuclear high level waste disposal. Corrosion aspects.
SKB Technical Report TR 92-26, Stockholm

15-7 Taxén C
1991
Pitting corrosion of copper with sulphide as the reaction determining reactant.
SKB Arbetsrapport AR 91-40, Stockholm

15-8 Taxén C
1992
Pitting corrosion under moderately oxidizing conditions.
SKB Arbetsrapport AR 92-43, Stockholm

Chapter 16

16-1 Pusch R, Karnland O, Lajudie A, Decarreau A
1992
MX 80 clay exposed to high temperatures and gamma radiation.
SKB Technical Report TR 93-03, Stockholm

16-2 Pusch R
1992
Investigation of a clay profile on southern Gotland of presumed value for documentation of smectite/illite conversion.
SKB Arbetsrapport AR 92-74, Stockholm

16-3 Börgesson L
1990
Interim report on the laboratory and theoretical work in modelling the drained and undrained behaviour of buffer materials.
SKB Technical Report TR 90-45, Stockholm

16-4 Börgesson L
1992
Interaction between rock, bentonite buffer and canister. FEM calculation of some mechanical effects on the canister in different disposal concepts.
SKB Technical Report TR 92-30, Stockholm

16-5 Börgesson L
1992
Interim report II on the pilot settlement test in Stripa.
SKB Arbetsrapport AR 93-09, Stockholm

Chapter 17

17-1 Osnes J D et al.
1992
Analysis of Well Test Data – Application of probabilistic models to infer hydraulic properties of fractures.
SKB Arbetsrapport AR 92-54, Stockholm

17-2 Rosengren L, Christianson M
1992
DECOVALEX. Test Case 1, Coupled Stress Flow Model.
SKB Arbetsrapport AR 92-49, Stockholm

17-4 Kung C S, Huan W X, Cvetkovic V, Winberg A
1992
Stochastic continuum simulation of mass arrival using a synthetic data set. The effect of hard and soft conditioning.
SKB Technical Report TR 92-18, Stockholm

17-5 Boulton G S, Payne A
1992
Simulation of the European ice sheet through the last glacial cycle and prediction of future glaciation.
SKB Technical Report TR 93-14, Stockholm

17-6 Wallroth T, Gustafson G
1992
Sub-surface conditions produced by future climate changes including glaciation. Data support for modelling.
SKB Arbetsrapport AR 92-77. Stockholm

17-7 Larsson E
1992
Two phase flow in the disturbed zone around a drift in rock.
SKB Arbetsrapport AR 92-76, Stockholm

17-8 Mandelbrot B B
1983
The Fractal Geometry of Nature – Updated and Augmented. W.M. Freeman. New York

17-9 Dershowitz W et al.
1992
The implication of fractal dimension in hydrogeology and rock mechanics.
SKB Technical Report TR 92-17, Stockholm

17-10 Dershowitz W et al.
1991
FracMan version ß2.3. Interactive Discrete Feature Data Analysis, Geometric Modelling and Exploration Simulation.
Project Report 913-1058. Golder Associates Inc., Redmond

17-11 SKB Annual Report 1990
SKB Technical Report TR 90-26, Stockholm

17-12 SKB Annual Report 1991
SKB Technical Report TR 91-64, Stockholm

17-13 Larsson S-Å, Tullborg E-L
1993
Tectonics regimes in the southern part of the Baltic Shield during the last 1 200 Ma – A review.
SKB Arbetsrapport AR 93-12 , Stockholm

17-14 Andreasson P-G, Rodh A
1992
The Protogine Zone Geology and Mobility during the last 1.5 Ga.
SKB Technical Report TR 92-21, Stockholm

17-15 Wannäs K-O, Flodén T
1993
Tectonic framework of the Hanö Bay area, Southern Baltic Area.
SKB Technical Report (in print), Stockholm

17-16 Milnes A G, Gec D G
1992
Bedrock Stability in the southeastern Sweden – evidence from fracturing in the Ordovician limestones of northern Öland.
SKB Technical Report TR 92-23, Stockholm

17-17 Slunga R
1985
The seismicity of southern Sweden, 1979-1984, final report.
FOA report C20572-T1, ISSN 0347-3694, Stockholm

17-18 Slunga R, Nordgren L
1987
Earthquake measurements in southern Sweden, October 1 – March 31, 1987.
SKB Technical Report TR 87-27, Stockholm

17-19 Slunga R
1989
Earthquake mechanisms in northern Sweden, October 1987 – April 1988.
SKB Technical Report TR 89-28, Stockholm

17-20 Muir-Wood R
1993
A review of the seismotectonics of Sweden.
SKB Technical Report TR 93-13, Stockholm

17-21 Ahlbom K, Äikäs T, Ericsson LO
1991
SKB/TVO Ice Age Scenario.
SKB Technical Report TR 91-32, Stockholm

17-22 Eronen M, Olander H
1990
On the worlds ice ages and changing environments.
TVO Report YJT-90-13, Helsinki

17-23 Björk S, Svensson N-O
1992
Climatic changes and uplift patterns – past, present and future.
SKB Technical Report TR 92-38, Stockholm

17-24 Rosengren L, Stephansson O
1990
Distinct element modelling of the rock mass response to glaciation at Finnsjön, central Sweden.
SKB Technical Report TR 90-40, Stockholm

17-25 Israelsson J, Rosengren L, Stephansson O
1992
Sensitivity study and rock mass response to glaciation at Finnsjön, central Sweden.
SKB Technical Report TR 92-34, Stockholm

17-26 Påsse T
1993
The glacio-isostatic uplift and the enstatic rise expresses in mathematical terms.
SKB Technical Report (in preparation), Stockholm

17-27 Maddock R
1993
Direct fault dating trials at the Äspö Hard Rock Laboratory.
SKB Technical Report (in preparation), Stockholm

17-28 Ahlbom K et al.
1992
Sternö Study Site. Scope of activities and main results.
SKB Technical Report TR 92-02, Stockholm

17-29 Ahlbom K et al.
1992
Kamlunge Study Site. Scope of activities and main results.
SKB Technical Report TR 92-15, Stockholm

17-30 Ahlbom K et al.
1992
Klipperås Study Site. Scope of activities and main results.
SKB Technical Report TR 92-22, Stockholm

17-31 Ahlbom K et al.
1992
Finnsjön Study Site. Scope of activities and main results.
SKB Technical Report TR 92-33, Stockholm

17-32 Ahlbom K, Leijon B, Liedholm M, Smellie J
1992
Gabbro as a host rock for a nuclear waste repository.
SKB Technical Report TR 92-25, Stockholm

17-33 Leijon B
1992
Geomechanical and rock engineering characteristics of gabbro.
SKB Progress Report LOK 92-01, Stockholm

17-34 Smellie J
1992
Gabbro: Geological and hydrogeochemical features.
SKB Progress Report LOK 92-03, Stockholm

17-35 Liedholm M
1992
The hydraulic properties of different greenstone areas.
SKB Progress Report LOK 92-07, Stockholm

17-36 Ahlbom K
1992
Gabbro as a host rock for a repository. Swedish experiences.
SKB Progress Report LOK 92-02, Stockholm

17-37 Rosén L, Gustafson G
1992
Possible strategies for geoscientific classification for high-level waste repository site selection.
SKB Arbetsrapport AR 92-57, Stockholm

17-38 Påsse T
1992
Mathematical formulas for the glacio-isostratic uplift in Fennoscandi and the global enstatic rise.
International Geological Correlation Programme 274 Annual Meeting, Wellington, New Zeeland.
Geol. Soc. of New Zeeland publ. 65A

17-39 Israelsson J, Stephansson O
1992
Sensitivity analysis of rock mass response to glaciation of the Finnsjön Site, Sweden.
In: Aggregate 1992 and The Finnish Symposium on Rock Mechanics 1992. pp. IV/9-12

17-40 Rosengren L, Stephansson O
1993
Modelling of Rock Mass Response to Glaciation at Finnsjön, Central Sweden.
Tunnelling and Underground Space Technology, Vol. 8, No. 1, pp 75-82. Pergamon Press Ltd., Great Britain

17-41 Rehbinder G
1992
Thermally induced flow of two immiscible stratified liquids in a porous body at a low Reyleigh number.
Int. J. Heat and Fluid Flow, Vol. 13, No. 2, June, 1992. Butterworth-Heinemann

17-42 Pusch R, Stanfors R
1992
The zone of disturbance around blasted tunnels at depth.
Int. J. of Rock Mechanics and Mining Sciences & Geomech. Abstr. Vol 29, No 5 pp. 447-456, Pergamon Press

Chapter 18

18-1 Smellie J, Laaksoharju M
1992
The Äspö Hard Rock Laboratory: Final evaluation of the hydrogeochemical pre-investigations in relation to existing geological and hydraulic conditions.
SKB Technical Report TR 92-31, Stockholm

18-2 Banwart S, Gustafsson E
1991
Scoping calculations of surface water and redox front breakthrough.
SKB Progress Report PR 25-91-06, Stockholm

18-3 Banwart S, Laaksoharju M, Nilsson A-C, Tullborg E-L, Wallin B,
1992
The large scale redox experiment. Initial characterization of the fracture zone.
SKB Progress Report PR 25-92-04, Stockholm

18-4 Banwart S, Gustafsson E, Laaksoharju M, Nilsson A-C, Tullborg E-L, Wallin B
1993
Redox processes in granitic coastal aquifer: Characterization of the large scale experimental site and some initial results.
SKB Progress Report PR 25-93-03, Stockholm

18-5 Östhols E, Bruno J, Grenthe I
1993
On the influence of carbonate on mineral dissolution III: The solubility of microcrystalline ThO_2 in CO_2-H_2O media.
Geochemica et Cosmochimica Acta (submitted)

18-6 Eriksen T E, Ndalamba P, Spahiu K Bruno J
1993
Solubility of neptunium(IV) hydrous oxide in neutral to alkaline solutions. Effect of carbonate complexation.
Journal of Inorganic Chemistry (submitted)

18-7 Miekeley N, Coutinho de Jesus H, Porto da Silveira C L, Linsalata P, Morse R, Osmond J K
1991
Natural series radionuclide and rare earth element geochemistry of waters from the Osamu Utsumi mine and Morro do Ferro analogue study sites, Poços de Caldas, Brazil.
SKB Technical Report TR 90-17, Stockholm

18-8 Nordén M, Albinsson Y, Ephraim J H, Allard B
1992
A comparative study of europium, thorium and uranium binding to an aquatic fulvic acid.
In Scientific Basis for Nuclear Waste Management Materials Research Society

18-9 Ledin A, Karlsson S, Allard B
1993
Effects of pH, ionic strength and a fulvic acid on size distribution and surface charge of colloidal quartz and hematite.
Journal of Applied Geochemistry, Vol 8, No 4 (in press)

18-10 Ledin A, Karlsson S, Düker A, Allard B
1993
On the applicability of photon correlation spectroscopy for in situ measurement of concentration and size distribution of colloids in deep groundwaters.
Anal. Chim. Acta (submitted)

18-11 Pedersen K, Ekendahl S
1992
Incorporation of CO_2 and introduced organic compounds by bacterial populations in groundwater from the deep crystalline bedrock of the Stripa mine.
Journal of General Microbiology, Vol 138, p. 369-376

18-12 Petersen K, Ekendahl S
1992
Assimilation of CO_2 and introduced organic compounds by bacterial communities in groundwater from Southeastern Sweden deep crystalline bedrock.
Microbial Ecology, Vol 23, p. 1-14

18-13 Wallberg M, Brynhildsen L, Allard B
1991
Metal binding properties of Klebsiella oxytoca.
Water Air Soil Poll. Vol 57/58, p. 579-587

18-14 Pedersen K, Ekendahl S
1990
Distribution and activity of bacteria in deep granitic groundwaters of Southeastern Sweden.
Microbial Ecology, Vol 20, p. 37-52

18-15 Chapelle F H, Lovley D R
1992
Competitive exclusion of sulfate reduction by Fe(III)-reducing bacteria. A mechanism for producing discrete zones of high-iron groundwater.
Ground Water, Vol 30, p. 29-36

18-16 Ittner T, Allard B, Thorén A
1993
Radium, barium and strontium sorption on soil and rock samples. A laboratory study of the analogous behaviour of the alkaline earth metals.
(Manuscript in preparation)

18-17 Eriksen T E
1988
Radionuclide transport in a single fissure. A laboratory flow system for transport under reducing conditions.
SKB Technical Report TR 88-28, Stockholm

18-18 Ittner T
1992
Groundwater flow measurements (TP2) in selected sections at Äspö after tunnel passage of fracture zone NE-1.
SKB Progress Report PR 25-92-17, Stockholm

Chapter 19

19-1 Stanfors R, Erlström M, Markström I
1991
Äspö Hard Rock Laboratory. Overview of the investigations 1986 – 1990.
SKB Technical Report TR 91-20, Stockholm

19-2 Almén K- E, Zellman O
1991
Äspö Hard Rock Laboratory. Field investigation methodology and instruments used in the pre-investigation phase, 1986 – 1990.
SKB Technical Report TR 91-21, Stockholm

19-3 Wikberg P, Gustafson G, Rhén I, Stanfors R
1991
Äspö Hard Rock Laboratory. Evaluation and conceptual modelling based on the pre-investigations 1986–1990.
SKB Technical Report TR 91-22, Stockholm

19-4 Gustafson G, Liedholm M, Rhén I, Stanfors R, Wikberg P
1991
Äspö Hard Rock Laboratory. Predictions prior to excavation and the process of their validation.
SKB Technical Report TR 91-23, Stockholm

19-5 Smellie J, Laaksoharju M
1992
The Äspö Hard Rock Laboratory: Final evaluation of the hydrogeochemical pre-investigations in relation to existing geologic and hydraulic conditions.
SKB Technical Report TR 92-31, Stockholm

19-6 Rhén I, Svensson U (eds), Andersson J-E, Andersson P, Eriksson C-O, Gustafsson E, Ittner, Nordqvist R
1992
Äspö Hard Rock Laboratory. Evaluation of the combined longterm pumping and tracer test (LPT2) in borehole KAS06, SKB.
SKB Technical Report TR 92-32, Stockholm

19-7 RD&D-Programme 92
Treatment and final disposal of nuclear waste. Programme for research, development, demonstration and other measures.
Background report, Äspö Hard Rock Laboratory.
SKB September 1992, Stockholm

19-8 SKB
1992
Äspö Hard Rock Laboratory. Annual Report 1992.
SKB Technical Report TR 93-08, Stockholm

Chapter 20

20-1 Chapman N A, McKinley I G, Shea M E, Smellie J A T (editors)
1993
The Poços de Caldas project. Natural analogues of processes in a radioactive waste repository.
Journal of Geochemical Exploration Vol 45, No 1-3, 1993

20-2 Chapman N A, McKinley I G, Shea M E, Smellie J A T
1991
The Poços de Caldas project. Summary and implications for radioactive waste management.
SKB Technical Report TR 90-24, Stockholm

20-3 SKB Annual Report 1990
SKB Technical Report TR 90-46, p 109-117, Stockholm

20-4 RD&D-Programme 92
Treatment and final disposal of nuclear waste. Programme for research, development, demonstration and other measures.
Background Report.
SKB, September 1992, Stockholm

20-5 Karlsson F, Smellie J A T, Höglund L-O
1992
The application of natural analogues to the Swedish SKB-91 safety performance assessment.
In the proceedings of the 5th CEC Natural Analogue Working Group, Toledo, Spain, 6-9 October 1992.
Com. Europ. Commun. Publ. (in press)

20-6 Cigar Lake project, Annual Report of the AEC/SKB Cigar Lake Project
1990
Phase 1: 1989-1990, AECL, Pinawa

20-7 Cigar Lake Project, Second Annual Report of the AECL/SKB Cigar Lake Project.
1991
Year 3, 1990-1991, AECL, Pinawa

20-8 Cramer J J, Smellie J A T
1992
The AECL/SKB Cigar Lake analogue study. Some implications for performance assessment.
In proceedings of the 5th CEC Natural Analog Working Group. Toledo, Spain Oct 6-9, 1992.
Com. Europ. Commun. Publ. (in press)

20-9 Blanc P L
1992
Oklo as a natural analogue for radionuclide transfer processes in a waste geological repository: Present status of the program.
In proceedings of the 5th CEC Natural Analogue Working Group. Toledo, Spain Oct 6-9, 1992.
Com. Europ. Commun. Publ.(in press)

20-10 Winberg A, Smellie J
1992
Oklo natural reactor study.
Preliminary reporting of CEA/EdM/SKB activities at Bagombe performed during September-December 1992

20-11 Suksi J, Ahonen L, Niini H
1992
The Palmottu analogue project.
Progress Report 1991.
Geological Survey of Finland Report YST-78, Espoo

20-12 Duerden P ed
1992
The international INTRAVAL project. Phase 1. Case 8. The Alligator Rivers natural analogue.
OECD/NEA publication, Paris, 1992

20-13 Tweed C J, Milodowski A E
1992
An overview of the Maqarin National analogue project. A natural analogue study of a hyperalcaline cement groundwater system.
In proceedings of the 5th CEC Natural Analogue Working Group. Toledo, Spain Oct 6-9, 1992.
Com. Europ. Commun. Publ. (in press)

20-14 Alexander W R ed.
1992
A natural analogue study of cement-buffered hyperalkaline groundwaters and their interaction with a sedimentory host rock.
NAGRA Technische Bericht NTB 91-10, Wettingen
(in print)

Chapter 21

21-1 Landström O, Sundblad B
October 1986
Migration of Thorium, Uranium, Radium and ^{137}Cs in till soils and their uptake in organic matter and peat.
SKB Technical Report TR 86-24, Stockholm

21-2 Sundblad B, Bergström U, Evans S, Puigdomenech I
September 1988
Long-term dynamics of a lake ecosystem and the implications for radiation exposure.
SKB Technical Report TR 88-31, Stockholm

21-3 Smith G
May 1989
Scenario B5; Ageing of a Lake.
BIOMOVS TR 5 National Institute of Radiation Protection, Stockholm, Sweden

21-4 Jones C H
September 1990
Scenario B6; Transport of radionuclides to root-zone soil from contaminated groundwater. BIOMOVS TR 9 National Institute of Radiation Protection, Stockholm, Sweden

21-5 Zeevert Th, Jones C
1992
Modelling the geosphere-biosphere interface. Experiences gained from two test scenarios in the BIOMOVS study; Modelling the geo-biosphere interface Vol 1 No 3/4 Sept/Dec 1992

21-6 IAEA
1992
VAMP Progress Report No 4, June

21-7 Nordlinder S, Bergström U
1992
A dynamic model of the Cs-137 concentration in fish applied on seven different lake eco systems – A VAMP scenario: Presented at the 6th Nordic Radioecological Seminar, Torshavn, Faroe Islands, 14-18 June

21-8 Sundblad B
Augusti 1992
Långsiktig överföring mellan geosfär och biosfär beskriven genom sedimentstudier vid Äspö HRL. Presented at the Nordic Hydrological Conferense in Alta (Norway)

21-9 Smellie J, Laaksoharju M
1992
Hydrochemical investigations in relation to existing geological and hydraulic conditions.
SKB Progress Report PR 25-91-05 1992, Stockholm

21-10 Smellie J, Laaksoharju M
1992
Final evaluation of the hydrochemical pre-investigations in relation to geological and hydraulic conditions.
SKB Technical Report TR 92-31 1992, Stockholm

21-11 Wikberg P
1991
Äspö Hard Rock Laboratory. Evaluation and conceptual modelling based on the pre-investigation 1986-1990.
SKB Technical Report TR 91-22, Stockholm

21-12 Bergström U, Nordlinder S
April 1990
Individual radiation doses from unit releases of long lived radionuclides.
SKB Technical Report TR 90-09, Stockholm

21-13 Bergström U, Nordlinder S
July-September 1991
Uncertainties Related to Dose Assessment for High Level Waste Disposal. Nuclear Safety Vol 32 No 3

21-14 Bergström U, Nordlinder S
November 1990
Dose conversion factors for major nuclides within high level waste.
SKB Technical Report TR 90-35, Stockholm

21-15 Bergström U, Nordlinder B
May 1991
"Individual doses from radionuclides released to the Baltic coast".
SKB Technical Report TR 91-41, Stockholm

21-16 Kelmers A D, Meyer R E, Blencoe J G, Jacobs G K
1987
Radionuclide Sorbtion Methodologies for Performance Assessments of High-Level Nuclear Waste Repositories: A Perspective Gained from an NRC Workshop. Nuclear Safety vol.28 No.4 (1987) 515

21-17 Elert M, Argärde A-C
1985
Modelling of the interface between the geosphere and the biosphere – Discharge through a sediment layer. Project SSI P295-84. Swedish National Institute of Radiation Protection, Stockholm, Sweden (1985)

21-18 Elert M, Argärde A-C, Ericsson A-M
1988
Modelling of the interface between the geosphere and the biosphere – Discharge through a soil layer. Kemakta AR 88-23. Swedish National Institute of Radiation Protection, Stockholm, Sweden (1988)

21-19 Gustavsson E, Skålberg M, Sundblad B, Karlberg O, Tullborg E-L, Ittner T, Carbol P, Eriksson N, Lampe S
December 1987
Radionuclide deposition and migration within the Gideå and Finnsjön study sites, Sweden: A study of the fallout after the Chernobyl accident. Phase I, initial survey.
SKB Technical Report TR 87-28, Stockholm

21-20 Carbol P, Ittner T, Skålberg M
1988
Radionuclide Deposition and Migration of the Chernobyl fallout in Sweden.
Radiochimica Acta 44/45, 207-212 (1988)

21-21 Liljenzin J O, Skålberg M, Persson G, Ingemansson T, Aronsson P O
1988
Analysis of the Fallout in Sweden from Chernobyl. Radiochimica Acta 43,1-25 (1988)

21-22 Ittner T
December 1990
Long term sampling and measuring program. Joint report for 1987, 1988 and 1989. Within the project: Fallout studies in the Gideå and Finnsjö areas after the Chernobyl accident in 1986. SKB Technical Report TR 91-09, Stockholm

21-23 Ittner T, Gustavsson E, Nordqvist R
June 1991
Radionuclide content in surface and groundwater transformed into breakthrough curves. A Chernobyl fallout study in a forested area of northern Sweden.
Swedish Nuclear Fuel and Waste Management Co. 1991.
Presented at XXVII-CSI Pre-Symposium: MEASUREMENTS OF RADIONUCLIDES AFTER THE CHERNOBYL ACCIDENT June 6-8, Bergen, Norway, (1991).
SKB Technical Report TR 91-28, Stockholm

21-24 Nordqvist R, Gustavsson E, Ittner T
1991
Transport of the Chernobyl fallout in soil and bedrock – A field study.
Applied Geochemistry, 1991

21-25 Gustavsson E, Carbol P, Nordqvist R, Skålberg M
1992
Transport of radionuclides from the Chernobyl fallout in surface water and groundwater – Gideå study site – IV.
Submitted to The Science of the Total Environment

21-26 Abelin H, Neretnieks I, Tunbrant S, Moreno L
May 1985
Final report of the Migration in a single fracture – experimental results and evaluation.
Stripa Project Technical Report TR 85-03, Stockholm

21-27 Landström et al.
January 1981
Migration experiments in Studsvik.
SKBF/KBS Technical Report TR 83-18, Stockholm

21-28 Ittner T, Tammela P-T, Gustavsson E
June 1991
Soil map, area and volume calculations in Orrmyrberget catchment basin at Gideå, Northern Sweden.
SKB Technical Report TR 91-29, Stockholm

21-29 Sundblad B, Nordqvist R, Jönsson S
1992
The turnover of ^{137}Cs within a forest ecosystem described by two modelling approaches – Gideå study site – VI.
Submitted to The Science of the Total Environment

Chapter 22

22-1 See 19-8

Chapter 23

23-1 Sehlstedt S, Stark T
1991
Description of geological data in SKBs database GEOTAB Version 2.
SKB Technical Report TR 91-01, Stockholm

23-2 Sehlstedt S
1991
Description of geophysical data in SKBs database GEOTAB Version 2.
SKB Technical Report TR 91-02, Stockholm

23-3 Andersson P, Gerlach M
1991
Description of tracer data in SKBs database GEOTAB Version 1.
SKB Technical Report TR 91-05, Stockholm

23-4 Eriksson E, Sehlstedt S
1991
Description of background data in SKBs database GEOTAB Version 2.
SKB Technical Report TR 91-06, Stockholm

23-5 Gerlach M
1991
Description of hydrogeological data in SKB's database GEOTAB. Version 2.
SKB Technical Report TR 91-07, Stockholm

23-6 Eriksson E, Johansson B, Gerlach M, Magnusson S, Nilsson A-C, Sehlstedt S, Stark T
1992
GEOTAB: Overview.
SKB Technical Report TR 92-01, Stockholm

23-7 Ergodata
1990
GEOTAB: Users Guide Version 1.8.2.
SKB Technical Report TR 90-34, Stockholm

SKB ANNUAL REPORT 1992

Part III

SUMMARY OF THE INTERNATIONAL STRIPA PROJECT 1980 – 1992

CONTENTS PART III

24 SUMMARY OF THE INTERNATIONAL STRIPA PROJECT 1980–1992

24.1 INTRODUCTION

The Stripa Project was an international cooperative research effort organized to 1) develop techniques to characterize geologic sites in granite that are potentially suitable for the disposal of heat-generating radioactive wastes, and 2) examine engineered barrier materials and designs that could enhance the long-term safety of a repository system at such sites. The project was jointly undertaken by as many as nine member countries of the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). The project consisted of three phases, beginning in 1980 and ending in 1992. The Swedish Nuclear Fuel and Waste Management Company (SKB) managed the investigations of the project, carried out principally at the Stripa mine in Sweden, under the direction of a Joint Technical Committee composed of representatives from participating countries.

The scientific programme completed by the Stripa Project is the result of an evolutionary process covering some thirteen years of work, with shifts in emphasis arising in response to technical findings, administrative needs and changing priorities in the national programmes of participating countries.

The Stripa Project has been successful in demonstrating the use of the observational method for the evolutionary characterization of a volume of crystalline rock and for providing information relevant to the engineering of a repository. Important technical contributions have been made in two main areas:

1. development and demonstrated application of new equipment and methodologies for site characterization, and
2. development and in situ evaluation of materials and construction methods for engineered barriers.

In particular, the Stripa Project has achieved important results in the development of:

- methodologies, including techniques and tools, for the geophysical, geochemical and hydraulic characterization of fractured rock,
- procedures for comparing numerical predictions with measurements of groundwater flow and solute transport in fractured rock,
- methodologies to obtain data to assess radionuclide migration in saturated crystalline rock,
- methodologies for characterization of rock masses from the regional to the local scale and for development of conceptual site models, and
- performance evaluation for selected engineered barriers (buffer, backfill, seals), including an appreciation of their longevity.

The Stripa Mine is located in an old mining district in Sweden, approximately 250 km west of Stockholm. Figure 24-1 shows exactly the location of the Stripa mine.

24.2 PROJECT STRUCTURE

24.2.1 Management

The International Stripa Project, under the aegis of the OECD Nuclear Energy Agency, was based on agreements signed by the participating countries; a separate agreement was signed for each of the three phases of the project. The agreements stated detailed terms and conditions for management and finance and outlined the general scope of the scientific programme.

Responsibility for supervision and financing of the research programme resided with the JTC, which was composed of managers representing each of the national organizations. The JTC also provided information on the general progress of work to the Steering Committee of the OECD Nuclear Energy Agency, through the NEA Committee on Radioactive Waste Management.

A recognized expert (principal investigator) was selected to lead each research activity. In Phases 1 and 2, the design and conduct of the experiments were periodically reviewed by two TSGs. In Phase 3, the two TSGs were combined into one. Also in Phase 3, a Task Force on Sealing Materials and Techniques and a Task Force on Fracture Flow Modelling were established to guide the efforts to seal fractured rock and to evaluate the validity of groundwater flow and solute transport models, respectively. The subgroups and task forces were composed of experts from the participating countries. The organization of the project is illustrated schematically for Phases 1 and 2 in Figure 24-2 and for Phase 3 in Figure 24-3.

The Research and Development Division of the Swedish Nuclear Fuel and Waste Management Company (SKB) acted as the host organization and provided the project management. SKB was responsible for the mine operations and for the procurement of equipment and material for experimental work. Meetings of the TSGs, the JTC, the principal investigators and the project management were held regularly to review the progress of the project.

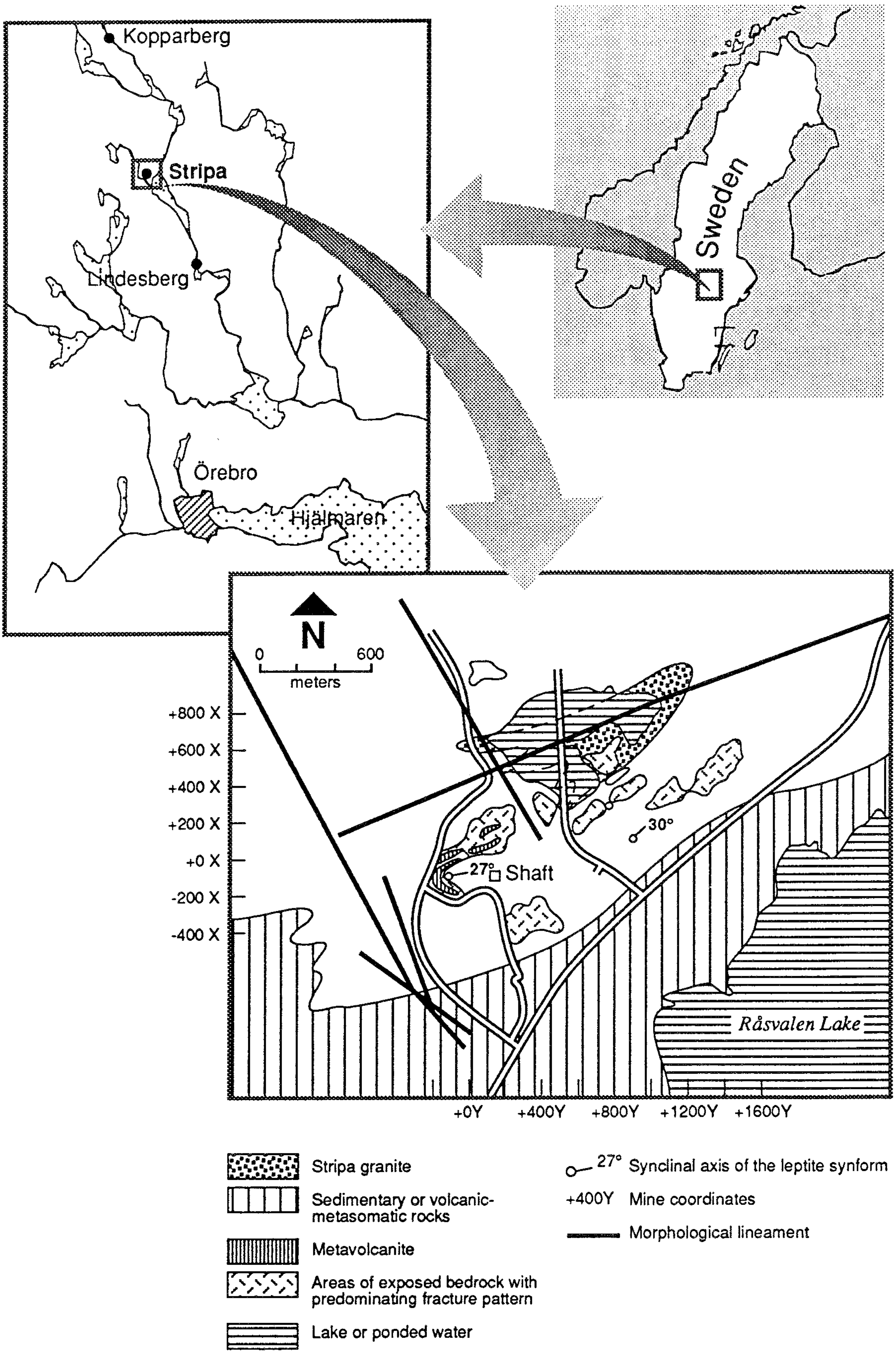


Figure 24-1. *Location of the Stripa mine (from /24-14/).*

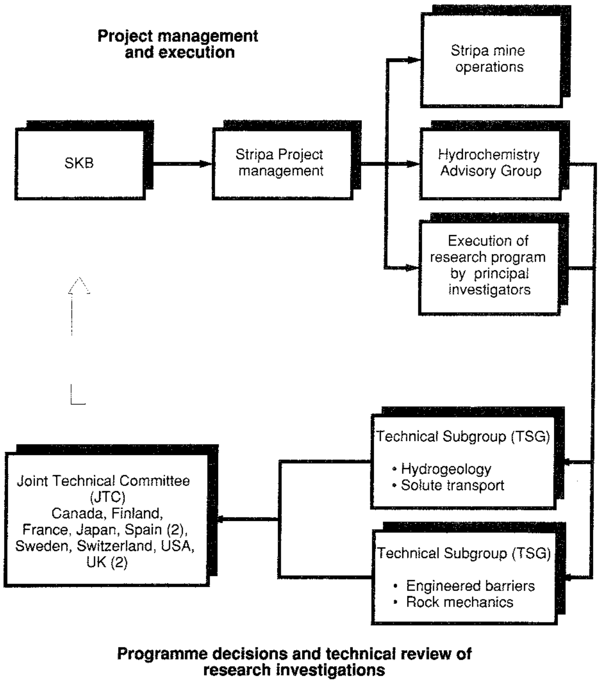


Figure 24-2. *Organization of Phases 1 and 2 of the Stripa Project.*

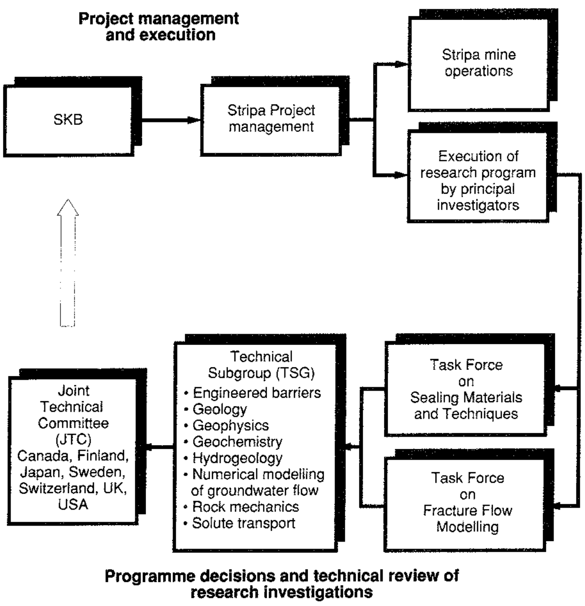


Figure 24-3. *Organization of Phase 3 of the Stripa Project.*

High-quality scientific work was assured through the recruitment of principal investigators of high standing in the international scientific community. These investigators were assembled into research groups, which conducted all research activities. Continuity of principal investigators and key operational staff minimized delays and disruptions in the scientific programme and led to a high level of motivation of the technical personnel throughout the project.

Owing to the international nature of the project, specific formal quality assurance and control procedures that would be consistent with those of each of the individual national programmes could not be adopted. The quality of the research was assured by reliance on the knowledge and expertise of the individual researchers, the implementation of accepted good scientific practice and thorough documentation of the work. During the course of the project, quarterly progress reports were produced by the various research teams. Over 170 technical reports were issued and many articles were published in the refereed scientific literature.

At the outset in 1980, four technical subgroups were established to guide investigations in the separate areas of hydrogeology, solute transport, engineered barriers, and rock mechanics. Later that year, the subgroups were reorganized into two subgroups, as shown above, and a separate advisory group for hydrochemistry was established.

At the outset in 1986, the JTC combined the two technical subgroups into one technical subgroup, and established a task force to guide the investigations dealing with sealing materials and techniques. In 1987, the JTC established a second task force to guide the activities dealing with groundwater flow modelling. The activities of the Hydrochemistry Advisory Group were then discontinued.

The participants in the Stripa Project and variations in participation in the three phases are shown in Table 24-1.

Table 24-1. Participants in the Stripa Project.

Canada	Atomic Energy of Canada Ltd (AECL Research)
Finland	Teollisuuden Voima Oy (TVO), Imatran Voima Oy (IVO), Ministry of Trade and Industry
France (Phases 1 and 2)	Commissariat ' l'Energie Atomique (CEA), Agence Nationale pour la Gestion des Dichets Radioactifs (ANDRA)
Japan	Power Reactor and Nuclear Fuel Development Corporation (PNC)
Spain (Phase 2)	Junta de Energia Nuclear (JEN)
Sweden	Swedish Nuclear Fuel and Waste Management Company (SKB)
Switzerland	National Cooperative for the Disposal of Radioactive Waste (NAGRA)
United Kingdom (Phases 2 and 3)	Department of the Environment (UK DoE)
United States	Department of Energy (US DOE)

24.2.2 Financial

The participants agreed to assign funds for each of the three phases of the project. In accordance with the signed agreement, the total cost of Phase 3 was adjusted for inflation on the basis of the Swedish Index for Consultants. The remaining budget was adjusted in January of each year. Also, the initial cost estimates of Phases 1 and 2 were revised because of changes in the programme of work, as agreed by the JTC, and inflation.

The total cost of the International Stripa Project, Phases 1 to 3, as borne by the participating countries in accordance with the agreements, was as follows:

Phase 1 (1980–1985)	47 MSEK
Phase 2 (1983–1988)	66 MSEK
Phase 3 (1986–1992)	144 MSEK
TOTAL COST	257 MSEK

This total project cost does not account for costs borne by the individual member countries as part of their participation in the JTC, technical subgroups, and task force activities. Also, the total expenditure does not include the cost of a number of technical contributions borne by member countries and provided at no expense to the Stripa Project.

24.3 NATURAL BARRIERS

24.3.1 Introduction

As the search for geological sites for nuclear waste repositories progressed in various countries, several classes of rock formations emerged as preferred candidates. Crystalline rocks, such as granites and diorites, are abundant in many parts of the world and are attractive on at least two counts. The rock strength is high, so that it is easy to construct stable excavations, and the intrinsic rock permeability is low (i.e. the resistance to groundwater flow is high). In many cases, however, the rock mass is traversed by systems of discrete planar fractures that are the consequence of various phases of tectonic activity through geological history. The fractures may range from closely-spaced microfractures that are almost invisible to the naked eye, to extensive fractures and fault systems in which spacing varies from metres to hundreds of metres and more. When these fractures are interconnected, their resistance to groundwater flow may be much lower than that of the intact rock.

Radionuclides carried by groundwater usually move through a rock mass at a different rate than water. The nuclides may be sorbed onto rock fracture surfaces and/or into the rock matrix, depending on the chemistry of the radionuclides, groundwater and rock surfaces; and so may travel much more slowly than groundwater. This might not be the case for radionuclides attached to colloidal suspensions in groundwater. Thus, characterization of groundwater flow and radionuclide transport in fissured rock is clearly a topic of primary importance in assessing the suitability of a particular rock mass for hosting a waste repository.

24.3.2 Groundwater flow and solute transport

The rate of groundwater flow through a rock mass depends on two main factors: 1) the resistance to flow, determined by the characteristics of the constricted pathways through which water flows; and 2) the driving force, usually the "hydraulic gradient".

Most groundwater theory and experience concerns flow in porous, granular materials, where the material structure provides many interconnected pathways and the overall resistance to water flow is essentially uniform, even over a very small volume. This volume, above which there is no significant change in the specific flow resistance (known as the "representative elementary volume"), is a measure of the homogeneity of the material. If the resistance is the same in all directions through the volume, the material is said to be isotropically permeable; if not, it is anisotropic.

In porous rocks under moderate hydraulic gradient, the flow can be described by a simple and well-known linear relationship introduced by Darcy in 1856. In this relationship, the flux is equal to the product of the flow resistance, or hydraulic conductivity, and the hydraulic gradient.

From the viewpoint of hydraulic properties, there are important differences between porous materials and crystalline rocks. The latter are formed from a molten rock mass; after solidification, during cooling, they develop thin, flat cracks or fractures of various sizes. Exposure to tectonic forces over geological time can extend these fractures and/or introduce new sets of fractures. Each set is usually planar, with discrete spacing between each fracture, and has a dominant orientation. Some sets may be relatively open and highly conductive, whereas others may have been sheared. The sheared fractures may become filled with fault gouge and may act as barriers to water flow. The hydraulic conductivity of relatively thin planar fractures is much more sensitive to changes in pressure, acting within (fluid pressure) or outside (rock stresses) the fractures, than is the hydraulic conductivity of porous granular media. Such media are formed of aggregated particles and show a more uniformly distributed connected porosity through which water can flow. Moreover, field observations suggest that fractures in crystalline rocks tend to be distributed "log-normally" – that is, with many short, closely-spaced fractures or cracks and progressively fewer, longer and more widely spaced fractures.

How, then, can flow in a fractured rock be characterized? Can a representative elementary volume be assumed, at least for the larger scale regional groundwater flow studies, so that an "equivalent porous medium" may be defined that allows Darcy's law to be used? Is there a scale (e.g., around individual excavations) below which such an approximation is invalid? How do stress changes induced by excavations or by temperature changes in the rock mass affect groundwater flow and transport, and/or the interpretation of *in situ* experimental observations? Can flow in fractured rock be predicted with reasonable accuracy? Can any adverse features of flow in fractured rock masses be improved by appropriate engineering design of the repository? These questions concerning groundwater flow and transport need to be addressed in order to establish the suitability of fractured crystalline rocks as sites for high-level radioactive waste repositories.

To answer such questions, it is necessary to study the behavior of the fractures directly in the rock mass. The investigations in the Stripa Project were designed, in part, with this purpose in mind.

24.3.3 Site characterization

Characterization, in a broad sense, means quantification of the geologic, hydrologic, geochemical and geomechanical characteristics of a rock mass. These characteristics represent a wide range of physical parameters and attributes of the rock, ranging from intrinsic properties such as permeability and electrical conductivity to larger scale conditions and processes such as the regional hydraulic gradient and tectonic activity. The end product of these characterization activities is an understanding of the properties, conditions and ongoing processes within the rock mass as they exist today, as well as an understanding of how such characteristics evolved to their current states.

The preliminary selection of a potential site for a repository for heat-generating radioactive waste may be based only on limited data from regional tectonics and structural geology, as inferred from surface observations supplemented by geophysical investigations and a small number of vertical drill holes. Underground access to a potential repository site adds important and more detailed insight to the general understanding obtained from the initial regional characterization studies. When a large volume of rock is exposed at depth in the horizontal or subhorizontal direction, features that may have major significance, but which the regional study may have missed (e.g., steeply inclined, water-conducting fractures), can be revealed. Underground access also provides an opportunity:

1. to confirm, by direct measurement, the values previously assumed for the properties of the rock mass (e.g., homogeneity, permeability, *in situ* stress state), as well as the characteristics of specific structural features and hydraulic anomalies, and

2. to determine whether the introduction of the repository excavations would lead to local, or more extensive, irreversible effects (e.g., creation of an excavation disturbed zone with enhanced permeability around the periphery of the openings); whether these effects would be exacerbated by the introduction of heat-generating waste; and whether these effects can be addressed sufficiently by engineering measures to ensure that the overall isolation capability of the proposed repository site is not compromised.

24.3.4 Elements of the natural barriers investigations

The Stripa Project was established in recognition of the need for a more comprehensive *in situ* research programme to address critically important basic questions concerning groundwater flow and radionuclide transport related to isolation of heat-generating radioactive waste in fractured crystalline rock. Because several countries were considering the possibility of crystalline rock for waste repositories, an international effort seemed appropriate.

As described earlier, the Stripa Project developed in three overlapping phases during the period 1980-92. For the Natural Barriers investigations, these phases were, see Figure 24-4:

Phase 1 (1980-85). A learning exercise to evaluate the possibilities and limitations of existing characterization methods and experimental techniques.

Phase 2 (1983-88). Development of improved characterization procedures and techniques.

Phase 3 (1986-92). Application of characterization procedures and techniques involving: (i) hydrogeologic characterization of the SCV (Site Characterization and Validation) site in the Stripa mine; and (ii) predictions of groundwater flow and transport in the SCV site, and comparison of predictions with observations.

Within the various phases, three parallel component research activities can be identified:

1. *Characterization of the groundwater flow system*, on both the **regional scale,** within which the mine was situated, and a **local scale**, within the mine itself. The regional-scale system was simulated by means of a hydrogeologic equivalent porous media (EPM) model. The calculated flow system was confirmed by comparison with mine pumping rates and by "isotopic" dating of groundwater. The local-scale models considered flow through discrete fracture systems, with predictions of flow into the boreholes in the SCV block and, later, into the SCV drift.

2. *Development of non-destructive in situ tests* for locating and characterizing discrete fracture zones within a crystalline rock mass. The tests were based on radar, seismic and hydraulic techniques and also involved both regional- and local-scale studies.

3. *Investigation of the solute transport properties of rock fractures in granite.* This activity involved tracer migration experiments in single fractures and fracture systems, and studies of channelling of flow within fractures.

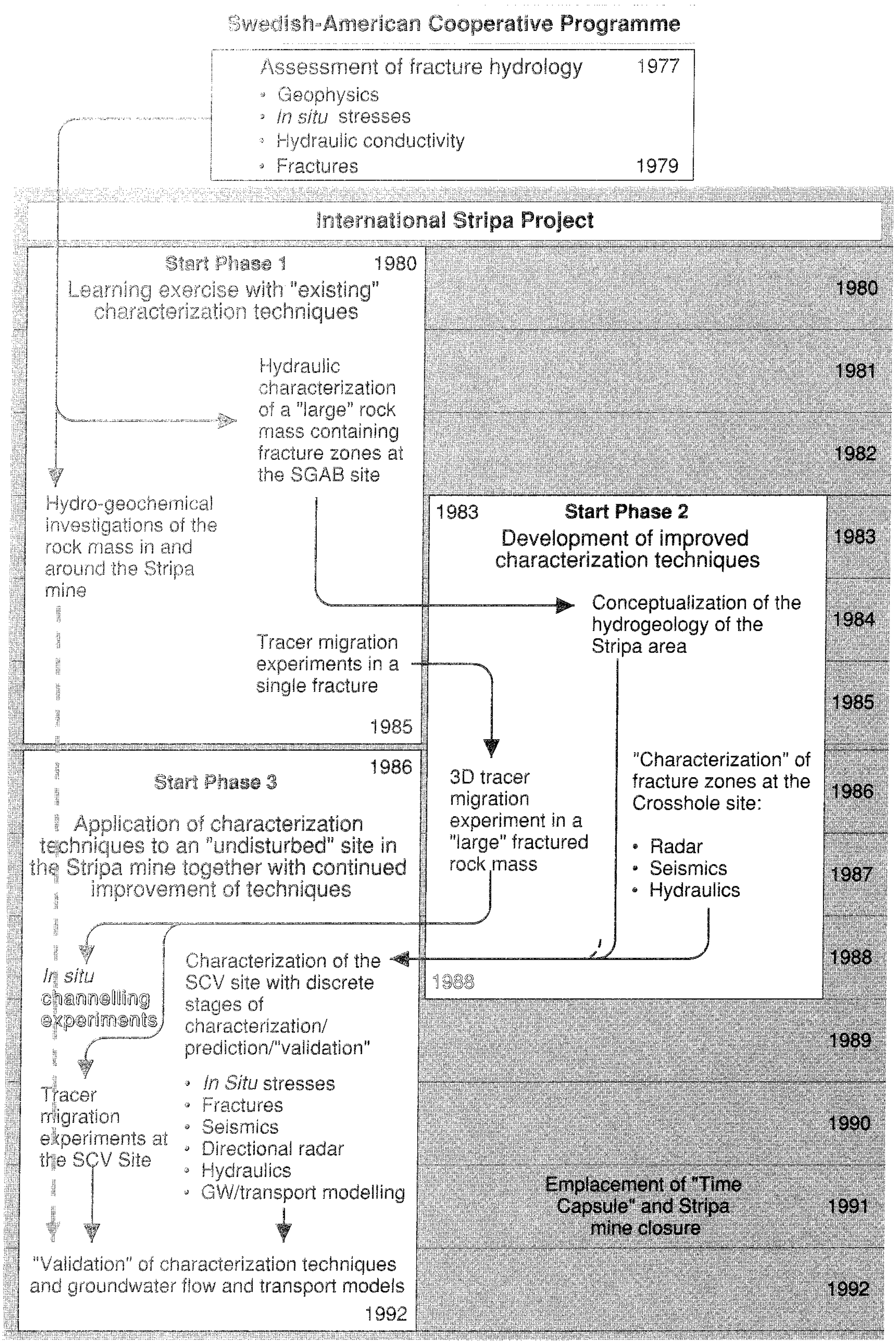


Figure 24-4. Evolution of the characterization, geochemistry, tracer migration, and modelling activities during the Stripa Project.

The development of conceptual hydrogeologic models at Stripa began during Phase 2 and was continued as a significant component of Phase 3.

All three components were brought together in the SCV programme, which concluded the Stripa Project. The progression of the various components is described below, following the outline shown in Figure 24-5.

24.4 REGIONAL-SCALE SITE CHARACTERIZATION

The Stripa mine had operated since the middle of the fifteenth century, and the general geology of the rock mass in and around the mine was well documented. Taking advantage of the existing background information, the project investigations were focussed on evaluating the main structural features of the geological setting. These features were the discrete zones of intensive fracturing in the granite, and systems of less intensive fracturing contained within the rock between the discrete fracture zones. The current groundwater hydrology and its associated geochemistry are products of the disturbance induced by the mining activities over many years, together with the groundwater conditions at the surface and in the surrounding rock masses. Similarly, the state of stress in the rock mass had been influenced by the mined excavations. Thus, when the Stripa Project began in 1980, the natural state of the rock mass had accommodated the disturbances induced by underground mining over several centuries.

24.4.1 Conceptual model of the Stripa area

The conceptual model of the hydrogeology of the Stripa area was based on information obtained from pre-Stripa-Project studies and from data collected during Phases 1 and 2 /24-1/. These geological, geochemical and hydrologic data were gathered by means of 1) observation of drill cores and exposed rock on the surface and in the Stripa mine; 2) laboratory tests of the groundwater geochemistry and rock fracture characteristics; and 3) hydraulic tests in boreholes drilled from the surface and at various locations within the Stripa mine. On the basis of the geologic and hydrologic data, a 3-D finite element model of the Stripa area, including the Stripa mine, was developed and calibrated against the rate at which groundwater was pumped out of the mine and against groundwater transit times inferred from the isotope studies. This finite element model, which represented the average rock and the fracture zones as equivalent porous media (EPM) with different hydraulic properties, covered a volume of rock with a plan area of about 100 km^2 and a depth of 3 km. In addition, several successively smaller sub-models were developed to represent in greater detail the rock mass that contained the rock excavations within the Stripa mine. These models were used in Phase 3 to obtain the hydrologic boundary conditions around the SCV site in order to support modelling of the groundwater flow and transport within the site.

The calculated inflow to the Stripa mine was within 10% or less of the measured pumping rate, which was good agreement, considering the large dimensions of the modelled region and the relative sparseness of subsurface geologic information and hydraulic data. The quantification of the hydraulic conductivity of the rock mass as a non-linear function of depth, together with the inclusion of the significant fracture zones, were important features of the model. Assumptions of a depth-independent hydraulic conductivity and different bedrock units produced inflow rates to the mine that were considerably less than the pumping rate. The computed hydraulic gradients within the vicinity of the mine were consistent with measurements made in surface boreholes, and the discharge points of groundwater were consistent with the locations of lakes in the area. The general pattern of head isopotentials indicated that groundwater recharge and shallow groundwater flow within 3 km of the mine influenced the groundwater discharge into the deepest levels of the mine. This result supports the hypothesis, based on geochemical data, that mixing of shallow and deep waters occurs in some regions of the mine.

The computed travel times of the groundwaters through the rock mass at Stripa were much shorter than those based on interpretations of geochemical and isotopic data. However, groundwater flow models based on equivalent porous media models are known to predict fluxes more accurately than travel times.

24.4.2 Hydrochemical characterization of the Stripa area

In 1980, a Hydrochemistry Advisory Group (HAG) was established to determine the origin and evolution of deep groundwaters within the Stripa granite through the study of their geochemistry /24-2/. In addition, the HAG was asked to identify processes and mechanisms of water-rock interactions that might occur at the depths being considered for the disposal of heat-generating waste in crystalline rock. Between 1985 and 1988, water samples were collected from packed-off water-bearing zones in boreholes within the Stripa mine. The water samples were subjected to routine and specialized analyses by the HAG members at various laboratories in Europe and North America. Results indicated that the salinity in the groundwater at Stripa was not caused by the presence of ancient sea water. The composition of the groundwater could not be matched with that of sea water, even after reasonable allowances were made for dilution and ion exchange. Studies of the rubidium-strontium chronology and the ^{87}Sr to ^{86}Sr ratios of rocks, minerals and fluids suggest that the Stripa granite was intruded 1.71 Ga (1 Ga = 10^9 years) ago and that later hydrothermal activity, perhaps 1.63 Ga ago, formed moderately high-temperature minerals along fractures. Isotopic ratios indicate that sea water diluted by groundwater, in a mixture very depleted in rubidium and strontium, invaded the granite along fractures, probably less than one million years ago.

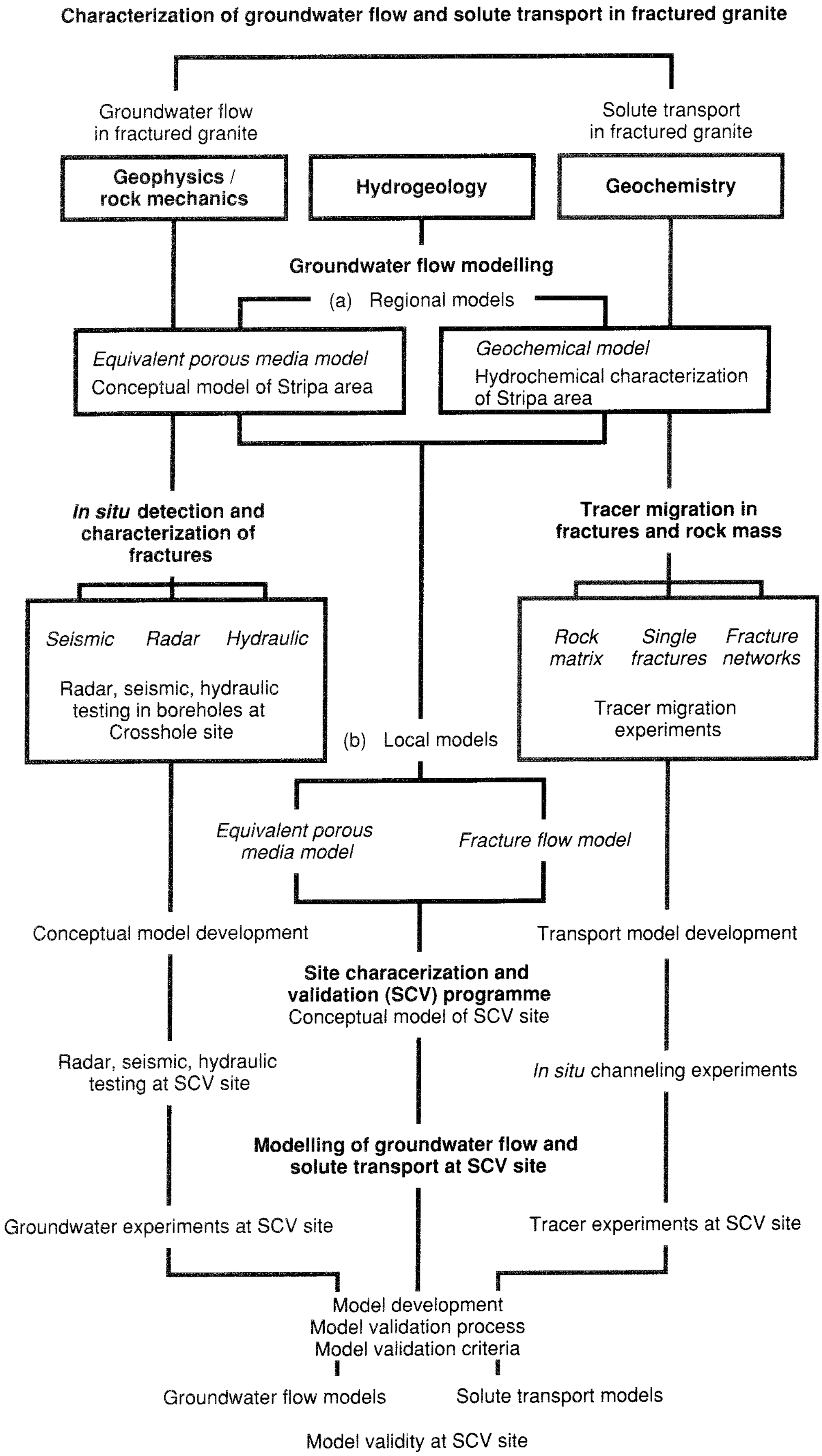


Figure 24-5. Relationships among the natural barriers investigations.

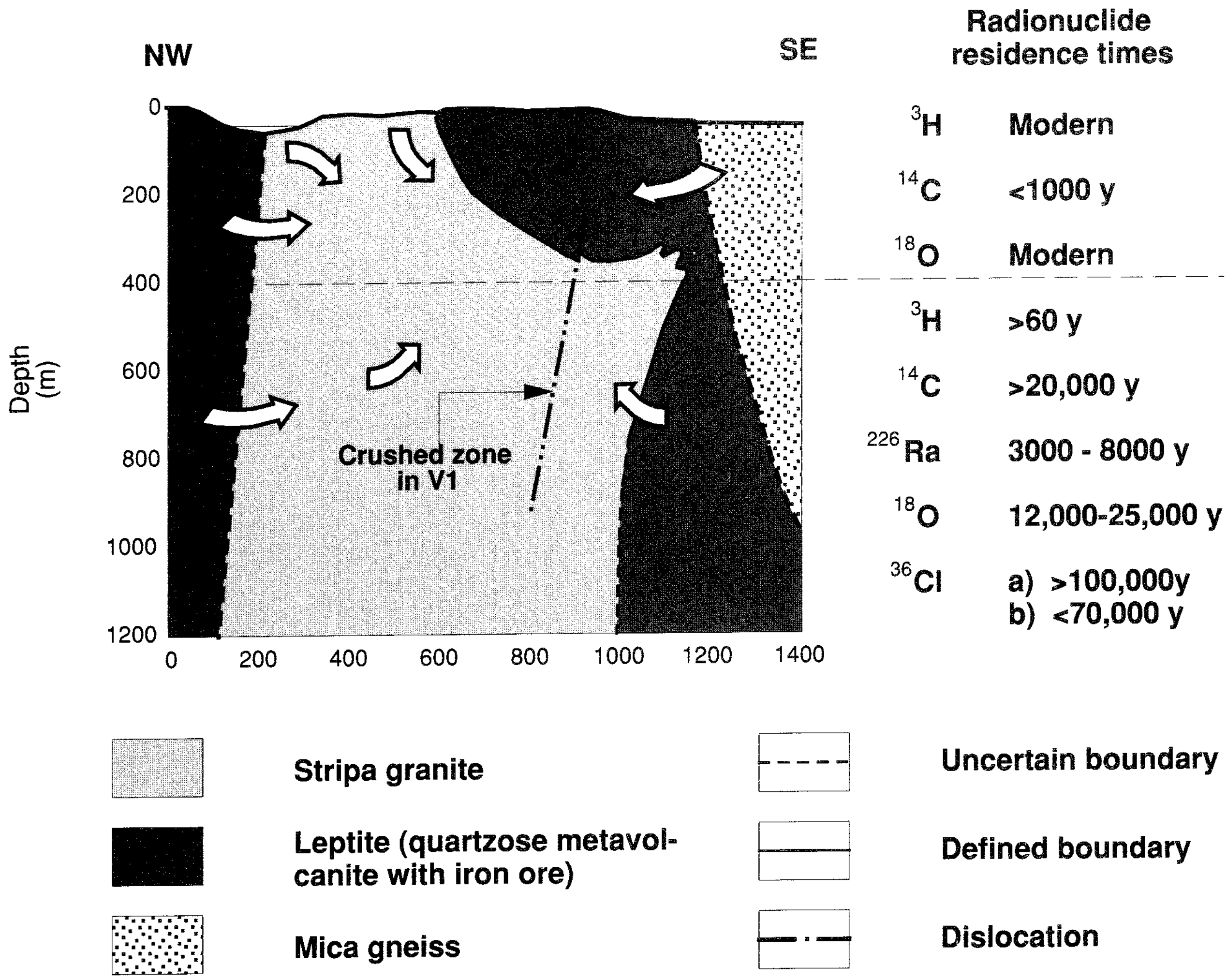
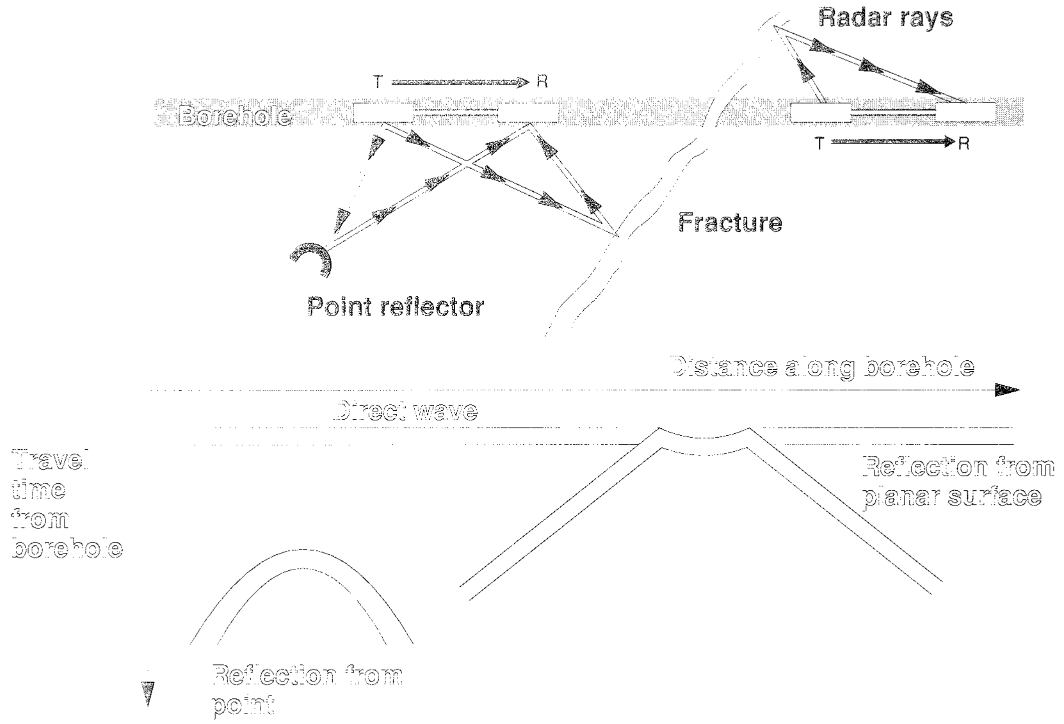


Figure 24-6. *Regional model of hydrochemical conditions at Stripa /24-1/.*

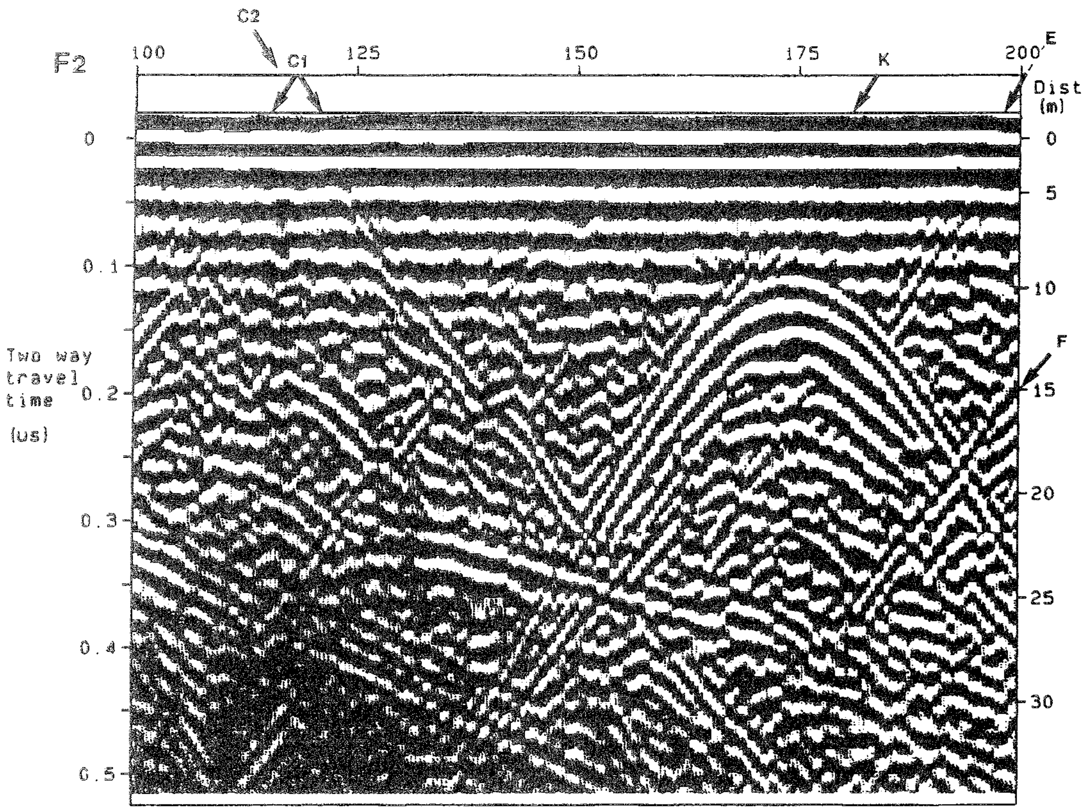
Shallow water and deep water are drawn into the area of the mined openings, forming a mixed water.

This was the first time that many techniques had been used in a single study to determine the origins and ages of groundwaters. These techniques included isotopic disequilibrium, buildup of radiogenic gases, decay of cosmogenic radionuclides, buildup of *in situ* generated radionuclides, and matching changes in concentrations of stable nuclides with known climatic fluctuations. A noteworthy achievement of the research work at Stripa was the clear demonstration, for the first time, that a number of so-called "cosmogenic radionuclides" are generated in the subsurface in quantities that greatly exceed the cosmogenic components. *In situ* production of the "key radionuclides" ^{36}Cl, ^{37}Ar, ^{39}Ar, ^{85}Kr, and ^{129}I takes place in the granite at Stripa at rates that exceed the atmospheric production of these radionuclides. The concentration differences between the *in situ* and atmospheric production are indicative of groundwater "residence time". Additionally, measurements were made of the stable nuclides ^{2}H, ^{18}O, ^{34}S, ^{37}Cl, ^{40}Ar, and ^{87}Sr, as well as the unstable nuclides identified above, together with ^{3}H, ^{14}C, ^{226}Ra, ^{234}U. A large number of these measurements were difficult to make with existing analytical techniques owing to the low concentrations and slight differences among samples from different locations and because the constituent had never, or only rarely, been measured before. The HAG finding that ^{36}Cl can be produced *in situ*, while previously it was thought to be generated only by the action of cosmic radiation in the atmosphere and by nuclear explosions, was particularly significant. This radioisotope can be used to measure the natural neutron flux in the subsurface.

The cross-section of the Stripa area, see Figure 24-6, illustrates the pattern of groundwater circulation and the residence times, interpreted from the presence of stable and unstable nuclides. The upper 300 to 400 m of the Stripa granite have been invaded in various locations by modern surface waters, small amounts of which have penetrated deeper than 800 m. Concentrations of ^{14}C in dissolved organic and inorganic carbon indicate an isola-



a) *Scheme of the borehole-radar testing configuration with characteristic patterns generated by reflections of the electromagnetic waves by a planar surface and a point.*



b) *Radar reflection map from measurements in borehole F2 at the crosshole site. C1, C2, K, E and F indicate fracture zones in the granitic rock mass.*

Figure 24-7. *Borehole radar testing method /24-6/.*

tion time of several thousands of years, particularly for the relatively deep groundwater. Dissolved chloride at depths of 300 to 600 m, and possibly deeper, has migrated into the granite from the surrounding metamorphic rocks. Below 600 m, the residual concentrations of ^{226}Ra and ^{36}Cl suggest that most of the water has been isolated from the atmosphere for at least several thousands of years and possibly as long as a few hundred thousand years, even though lateral migration of the water has taken place.

In the view of the HAG, the use of geochemistry to predict the long-term movement of radionuclides over distances of several kilometres is not a straightforward characterization method, especially for fractured rocks of plutonic origin. The presence of brackish groundwater, or water with a high pH or high helium content, indicates a rather "static" groundwater system. These kinds of geochemical conditions do not develop rapidly and suggest, qualitatively, that the ages of such waters would be of the order of thousands of years. The most reliable quantitative indicators of water age and, correspondingly, rates and directions of regional groundwater flow are ^{3}H, ^{14}C, ^{39}Ar, and ^{36}Cl, provided that the chemistry of the stable elements is understood and the *in situ* production of the radionuclides is minimal or quantifiable.

24.5 LOCAL-SCALE SITE CHARACTERIZATION

The need to develop tools and methods to evaluate the hydrogeologic characteristics of fractured rock masses was recognized from the start of the Stripa Project. The emphasis was placed on development of remote sensing techniques in boreholes; so that the minimal disturbance of the natural state of the rock during testing would permit to obtain data reflecting *in situ* conditions. This approach offers great promise for the cost-effective exploration of the large volumes of rock required for geologic disposal sites. The data collected by remote sensing from a few strategically placed boreholes could be used to make a preliminary assessment of the suitability of the site for long-term containment of radioactive waste, and to guide additional studies in evaluating specific features and anomalies.

24.5.1 Conventional characterization techniques

In Phase 1, the characteristics of fractures from drill core and from observations in boreholes and drifts were determined using conventional techniques /24-3/. A televiewer was used to scan the walls of boreholes to determine fracture locations and orientations, complementing the core-logging observations. In Phase 2, a more comprehensive description of the fracture systems in the Stripa granite was developed /24-4/. This description was based on information obtained previously from core logging, borehole televiewer scans and fracture mapping in the various drifts; it was enhanced during the remainder of the project with additional borehole and core data and drift mapping. The compressive strength and elastic deformation properties of the granite were obtained by laboratory tests on drill core and by an *in situ* block test in the mine. The influence of normal and shear loadings on those fracture characteristics relevant to water flow through the fractures, as well as aperture variations within fracture planes, were investigated in the laboratory, using large-diameter drill cores obtained from the mine. *In situ* stresses were measured in boreholes at various locations in the mine, using overcoring and hydraulic fracturing techniques.

24.5.2 Remote sensing techniques

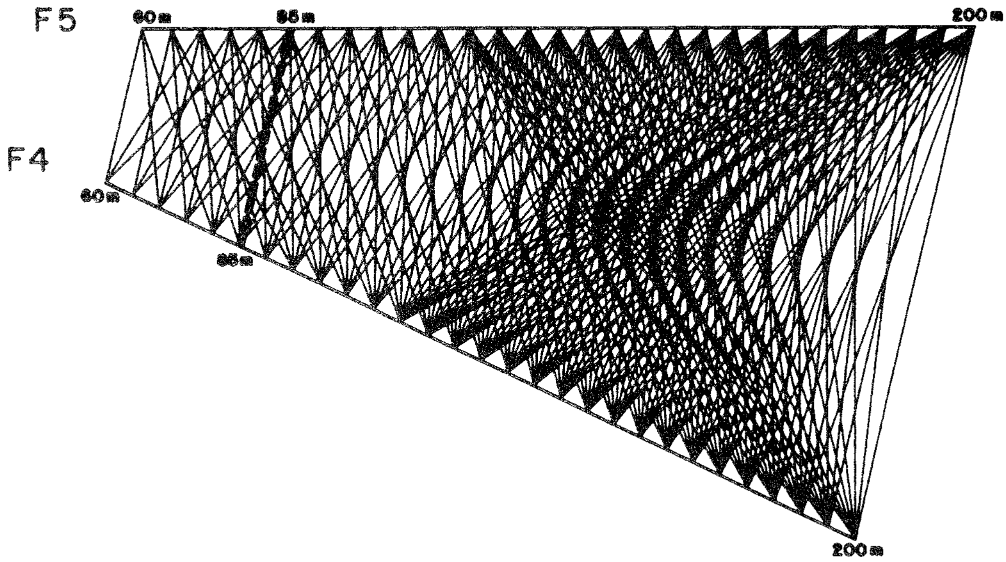
In Phase 2, a programme was undertaken to develop remote sensing techniques based on radar, seismic and hydraulic testing, and requiring only a few boreholes, for the detection and characterization of fracture zones /24-5/. These methods can provide data on the electric, elastic and hydraulic properties of rock, including, specifically, the geometric characteristics and the water flow properties of the fracture zones. The results of hydraulic testing initiated in Phase 1 provided a basis for design of the Phase 2 activities, including the selection of the borehole array and testing objectives. Much of the development and testing of these methods was carried out at the SGAB (Swedish Geological Company) site and, later, at the Crosshole site, as described below (Figure 24-9).

Radar testing

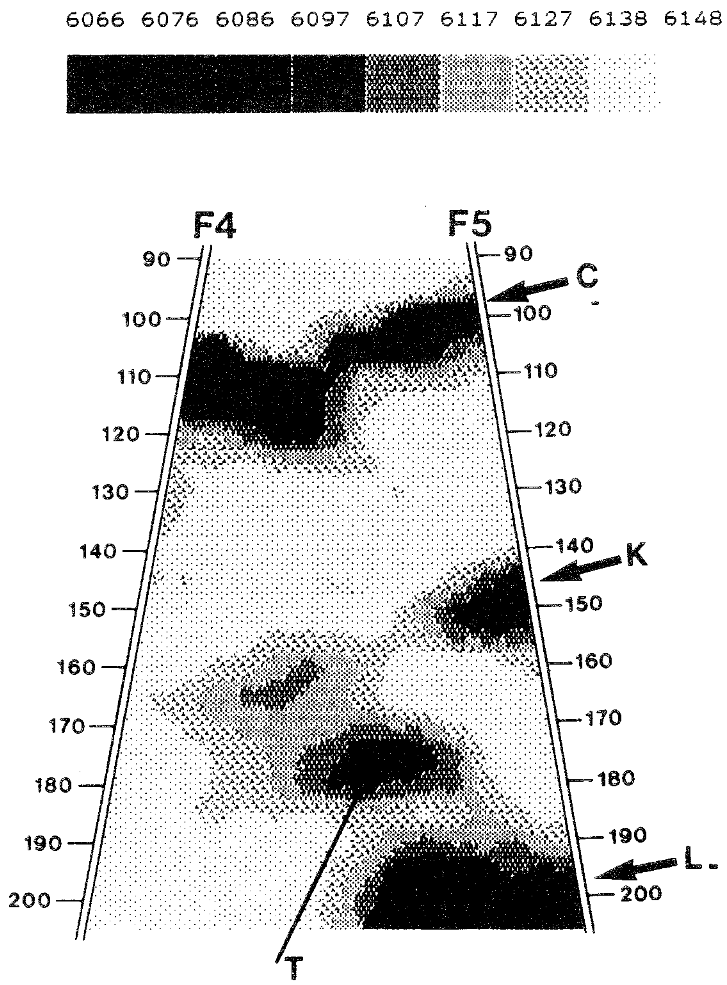
Between 1983 and 1985, the radar system was developed and designed for field work on a production basis. Since then, the system has been used to conduct borehole surveys at more than 25 sites throughout the world in a variety of rock types /24-6/. This short-pulse radar system, which is capable of both single borehole and crosshole measurements, displays and processes the data directly in the field. The radar senses localised changes in the dielectric constant and electrical conductivity of the rock mass usually associated with the increased water content of fracture zones. A schematic of the operation of the radar system and an example of a radar scan are shown in Figure 24-7. The location and geometric characteristics of the fracture zones at the Crosshole site, as determined by the radar technique, were consistent with interpretations based on crosshole seismic data. A directional antenna developed for the radar system in Phase 3 made it possible to determine the orientation of fracture zones from measurements in a single borehole.

Seismic testing

The crosshole programme included the development of large-scale and small-scale crosshole seismic techniques



a) Source-to-receiver ray paths between boreholes.



b) P-wave tomographic reconstruction of plane section between boreholes at the Crosshole site. C, K, L and T indicate fracture zones in the granitic rock mass.

Figure 24-8. *Borehole seismic testing methods /24-7/.*

/24-7, 24-8/. The crosshole seismic technique, as illustrated in Figure 24-8, involves the placement of a chain of receiving units in one borehole, and the detonation of an explosive source, or impact by a mechanical device, in a second borehole. The location and geometric characteristics of fracture zones existing between the boreholes can be derived from the results of tomographic inversions of the travel-time data. The small-scale crosshole seismics tests at the Crosshole site produced a detailed image of the structure between boreholes. In contrast, the large-scale tests performed at the Gideå site, located in northern Sweden and consisting of a migmatized gneiss with distinct fracture zones and dolerite dikes, revealed only the major structural features. In Phase 3, the seismic techniques were improved through the development of a reflection technique to determine the location and orientation of structural features, using a high-frequency seismic source located in a borehole. "Image source" theory and travel-time / distance profiles obtained with different seismic source/receiver locations were used to determine the feature orientation. The geometric characteristics of fracture zones within the plane can be identified by tomographic analysis from the amplitude and velocity profiles obtained from a number of different detector and source locations.

Hydraulic testing

Throughout the crosshole programme in Phase 2 and the SCV programme in Phase 3, determined efforts were made to integrate the hydrologic investigations with those involving remote-sensing geophysical techniques. The aims of the hydraulic testing were to measure the distribution of hydraulic properties within extensive fracture zones and to determine the hydraulic connections between fractures zones. The computer-controlled hydraulic equipment for single-borehole testing and crosshole sinusoidal hydraulic testing provided, on site, real-time interpretation of the data /24-9/. In addition, methods of analysis were developed to obtain information on the geometries of flow channels within fracture systems from single-borehole packer-test data.

24.5.3 Crosshole site programme

The Crosshole site, shown in Figure 24-9, was located on the 360 m level of the Stripa mine and consisted of a rock mass defined by six boreholes, drilled from the end of the drift in a fanlike array, to outline a tilted pyramid with a height and base of about 200 m. The site encompassed 3 million m^3 of fractured, granitic rock and was used for testing a variety of characterization methods /24-5/. Radar, seismic and hydraulic tests were conducted in the six boreholes in both the single-borehole and crosshole modes. A combination of single-borehole radar reflection measurements and tomographic inversion of crosshole radar and seismic data were used to define the geometrical characteristics of four major fracture zones, as well as several other less prominent zones, within the site.

The head distribution and hydraulic conductivities of the rock mass were determined by, first, evaluating regions of major groundwater flow with single-borehole hydraulic tests; and then comparing the results with those obtained from the single-borehole radar reflection tests in order to identify regions where anomalous features could be investigated by crosshole hydraulic testing.

The collected results from the fracture analysis of the core logs and the remote sensing tests in the boreholes provided the basis for developing a conceptual model of the hydrogeology of the Crosshole site. Furthermore, the applicability and effective "dimension" of the characterization methods were evaluated, as given in Table 24-2. The ordering in the table indicates the hierarchy of test methods that might be considered for sequential determination of the hydraulic characteristics of a saturated granitic rock mass. This hierarchy is representative of the increasing dimension of measurement, as well as the increase in time required for testing and data interpretation.

The research at the Crosshole site demonstrated that it was possible to characterize the geologic and hydrologic features of a saturated, fractured granitic rock mass of some 3 million m^3 and to define its hydrogeology more reliably and realistically than had been thought possible previously.

The programme illustrated the value of a well-coordinated characterisation effort, in which geophysical test results provided guidance for subsequent hydrologic testing of those regions of the rock mass containing hydraulic conduits. The investigations resulted in the development of a hierachy of test methods that could be applied sequentially to characterize a site containing water-bearing fracture zones.

24.5.4 Tracer migration experiments

In Phase 1, a groundwater tracer test was conducted in a single fracture in granitic rock that intersected a drift on the 360 m level of the Stripa mine /24-10/. The purpose of the test was to investigate the sorption and retardation of radionuclides during transport by natural water flow through the fracture over distances of 5 to 10 m; and, concurrently, to evaluate the extent and influence of channelling within the fracture plane. Non-sorbing tracers, used to characterize the water flow within the fracture, were injected from boreholes drilled into the fracture and collected in short sampling holes drilled into the exposed fracture in the drift. The shapes of the breakthrough curves could be interpreted in terms of the presence of channels of different transport lengths. Because the experimental data fit equally well various mathematical models of transport that incorporate matrix diffusion and channelling separately, the transport mechanism, or mechanisms, could not be uniquely determined. Subsequently, six sorbing tracers were injected into the fracture plane. These tracers did not reach the sampling holes during the test period, so portions of the fracture were excavated and tracer concentrations in the rock

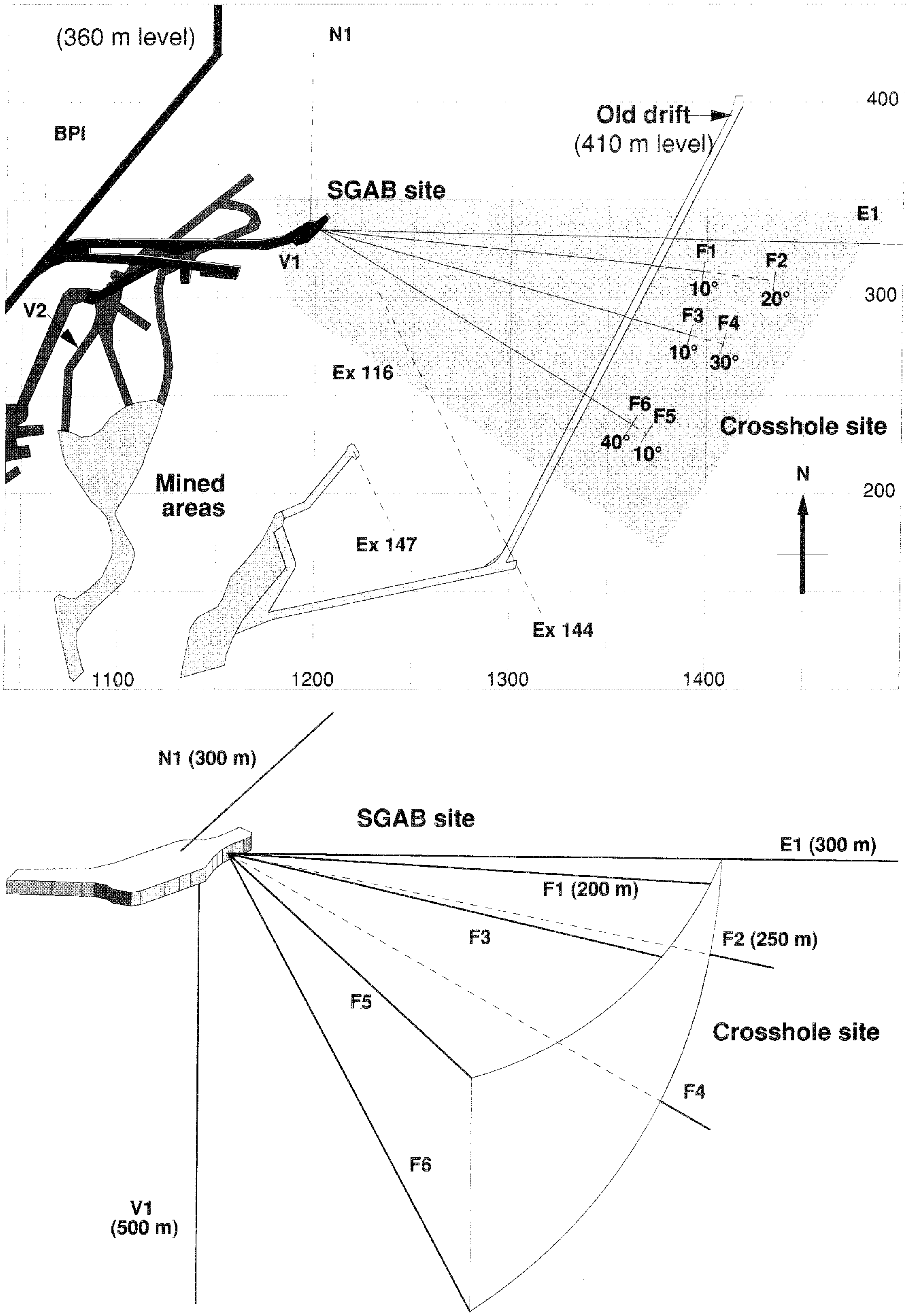


Figure 24-9. *Plan and perspective views of the borehole arrays at the SGAB and Crosshole sites /24-9/.*

The SGAB site was used for the hydraulic investigations in Phase 1 and the Crosshole site for the radar, seismic, and hydraulic investigations in Phase 2. All of the boreholes were drilled from the end of a drift located on the 360 m level.

Table 24-2. Applicability and effective "dimension" of characterization methods used at Crosshole site (modified after /24-5/.

Characterization technique	Primary information	Secondary information	Dimension of measurement
Logging of rock core	Geometrical features of rock mass	Fracture characteristics	1D
Laboratory tests on rock core	Physical properties of rock matrix	Fracture properties	1D
Single-borehole geophysical logging	Electrical properties of rock matrix/mass	Borehole deviation	1D
Single-borehole hydraulic test	Hydraulic properties of rock mass/fracture zones	Geometrical features of fracture zones	1D (minor 3D)
Crosshole radar and seismic tests	Geometrical features of rock structure	Transport properties	2D – 3D
Crosshole hydraulic tests	Hydraulic properties	Geometrical features of fracture zones	Pseudo 3D

were measured. It was found that the tracers had diffused into the rock matrix to depths of several mm or more near the injection point.

In Phase 2, a large-scale "three-dimensional" tracer experiment was designed and conducted /24-10/. The purpose of this experiment was to determine the flow porosity and the dispersion characteristics of a large fractured mass of granite and to investigate further the features of channelling. A 100-m-long drift with small side drifts forming a cross, as shown in Figure 24-10, was excavated at the 360 m level in the mine. Tracers were injected continuously for more than 1.5 years at nine points in three vertical boreholes at distances up to 55 m in the roof, and flow into the drift was collected in plastic sheets glued onto the roof and the upper part of the walls of the drift. The average travel times from the injection points to the plastic sheets varied between 85 and 290 days. The flow porosity of the granite within 10 m of the drift was about double the flow porosity of the relatively undisturbed rock. The apparent combination of independent pathways in the rock and fractures interconnected by channels resulted in quite variable flow rates, making it difficult to determine the dispersivity characteristics of the flow system.

The results of this large-scale tracer experiment supported the view that some fraction of the groundwater flow took place in discrete channels outside the fracture zones that had been previously identified as the main conduits. *In situ* tests were therefore conducted in the Stripa mine in Phase 3 specifically to address the geometric and hydraulic aspects of channelling in single natural fractures. The data from the single-borehole tests, over fracture lengths of about 2 m, provided information on the transmissivity and correlation lengths of conductive fractures and their possible relation to local fracture aperture values, as well as the distance between channels and the widths of channels within fractures. Pressure pulse tests and tracer tests with the "multipede" borehole-injection system provided information on the interconnection and mixing between channels, longitudinal and transverse dispersion, and residence volumes. The test data indicated that, on average, 25% or less of the natural fracture plane was open to water flow during inflow to the open excavation. The channel widths ranged from a few millimetres to a decimeter, and occurred in clusters that were decimeters wide and separated by distances of 0.5 m to 1 m.

From the results of the three series of tests described above, two very important conclusions can be derived.

1. The existence of channels in natural fractures seriously complicates the analysis of tracer transport data. It is difficult to identify the mechanisms of nuclide retardation and to quantify the pathway lengths and the dispersivity characteristics.

2. The characteristics of the groundwater flow field and the migration pathways in a volume of rock must be carefully quantified before conducting an *in situ* tracer test. This level of understanding is necessary in order to be able to properly analyze the tracer test data.

In Phase 3, considerable effort was expended to define the geometric and flow characteristics of the H fracture zone at the SCV site by radar and hydraulic testing before initiating the tracer experiment in that fracture zone.

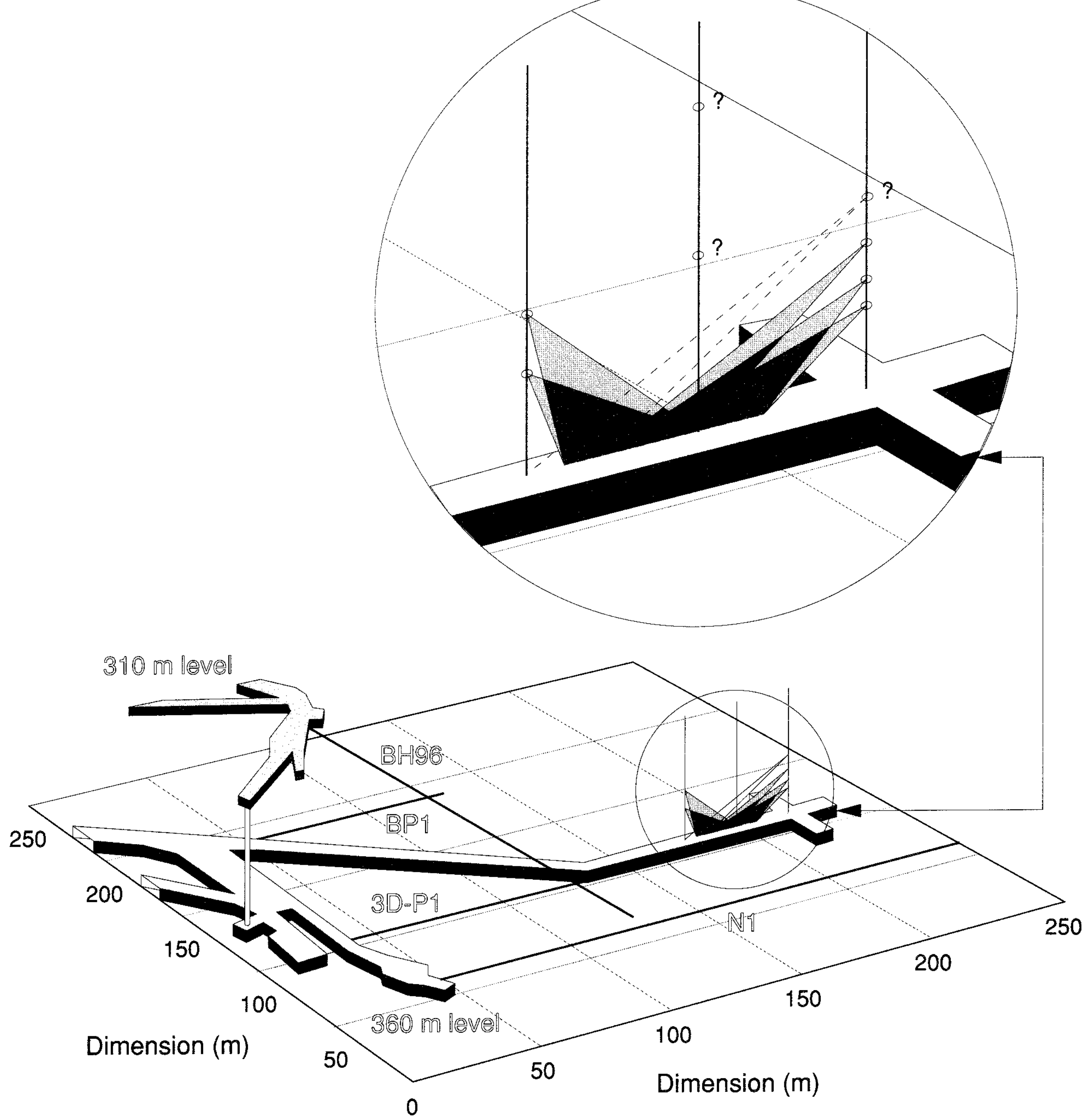


__Figure 24-10.__ Schematic diagram of the configuration of the 3D migration experiment /24-10/. The shaded areas indicate the locations at which tracers were observed in the drift.

24.6 SITE CHARACTERIZATION AND VALIDATION PROGRAMME

In Phase 3, the Site Characterization and Validation (SCV) programme was undertaken to develop and apply an advanced site characterization methodology that integrated different tools and methods in order to /24-11/:

- predict the distribution of groundwater flow and transport pathways in a specific volume of fractured Stripa granite,
- support the efforts to develop flow and transport models for fractured rock; and
- evaluate the validity of such models at the SCV site in the Stripa mine.

The SCV site encompassed a previously undisturbed block of granite located about 100 m north of the old mine workings and between the 360 m and 410 m levels. The dimensions of the block were approximately 125 to 150 m on a side and 50 m high, representing a volume of about 1,000,000 m^3.

The SCV programme was subdivided into five stages so that field data could be progressively compared with numerical predictions. The programme included several experiments, as summarized in Table 24-3.

In Stage I ("preliminary site characterisation"), three 200-m-long (S to N) and two 150-m-long (E to W) near-horizontal boreholes were drilled into the SCV block from the 345 m and 357 m levels. A preliminary database on existing fractures was established from core logging, borehole imaging, crosshole radar and seismic testing, and single-borehole hydraulic testing.

In Stage II ("preliminary prediction"), a hydrogeologic model of the SCV site was developed on the basis of the preliminary database. Assessments were made of the ge-

Table 24-3. Experiments conducted at the SCV site in support of the model validation activities in Phase 3.

Experiment	Measurements	Purpose
"First" simulated drift experiment	Rate and distribution of groundwater inflow to the array of six 100 m long boreholes	Comparison with predictions by equivalent porous media and fracture flow models
"Second" simulated drift experiment	Rate and distribution of groundwater inflow to the remaining 50 m long D boreholes after construction of the 50 m long Validation Drift	Comparison with predictions by equivalent porous media and fracture flow models, including effects of drift excavation
Fracture distribution in the Validation Drift	Mapping of the fractures in the roof, floor, and walls of the Validation Drift	Comparison with stochastic predictions of fracture patterns by fracture network models
Validation Drift experiment	Rate and distribution of groundwater inflow	Comparison with predictions by equivalent porous media and fracture flow models
"First" radar/saline tracer experiment	Collection of saline tracer in the D boreholes from injections in the H zone, before construction of the Validation Drift	Design of the tracer migration test; calibration of the equivalent porous media and fracture flow transport models
"Second" radar/saline tracer experiment	Collection of saline tracer in the Validation Drift from injections in the H zone	Design of the tracer migration test; comparison with predictions by equivalent porous media and fracture flow transport models; evaluation of effects of drift excavation
Tracer migration experiment	Collection of dye and metal-complex tracers in the Validation Drift and in a borehole from injections in the H zone and the "good" rock	Comparison with predictions by equivalent porous media and fracture flow transport models
Monitoring of groundwater head	Distribution of groundwater heads within and around the SCV site during (i) construction of the Validation Drift, (ii) implementation of the validation experiments, and (iii) draining of the T1 borehole	Comparison with predictions by equivalent porous media and fracture flow models

ometries and physical properties of the major fracture sets, along with numerical predictions of groundwater flow into six 100-m-long parallel boreholes known as D boreholes. Five of these boreholes were drilled at approximately equal spacing around the circumference of the planned circular cross-section of the Validation Drift. The sixth hole was drilled along the axis of the future drift. It was during this stage that the major fracture zones A,B, D, H, I, J, K, M were identified definitely at the SCV site.

In Stage III ("detailed characterization and preliminary validation"), the groundwater inflow to the D boreholes was measured and compared to the numerical predictions made in Stage II. In addition, three boreholes, referred to as C boreholes, were drilled into the central portion of the SCV site to check the accuracy of the hydrogeologic model developed in Stage II, and to obtain additional radar and seismic data. Finally, crosshole hydraulic tests were performed in the D boreholes and in a combination of the C boreholes and the boreholes drilled in Stage I.

In Stage IV ("detailed prediction"), the conceptual hydrogeologic model was refined on the basis of data obtained in Stage III. The first radar/saline tracer test was also performed. In this test, a saline tracer was injected into the H zone at a point of intersection with one of the C boreholes, and collected in the D boreholes. Borehole radar tests were conducted during the injection-collection process.

In Stage V ("detailed evaluation"), the Validation Drift was constructed in the region outlined by the first 50 m of the D boreholes and the groundwater inflow rate and distribution were measured. These measurements were compared with modelling predictions made in Stage IV. In addition, a second radar/saline tracer test was performed by injecting a saline tracer into the H zone and collecting the tracer in the Validation Drift. Finally, dye and metal complex tracers were injected into the rock mass at various distances from the Validation Drift, principally in the H zone, and collected within the drift. Predictions of the transport of the saline, dye, and metal complex tracers into the Validation Drift were made and compared with the measurements.

24.6.1 Radar, seismic and hydraulic testing

Based on the existing knowledge of the local geology of the Stripa mine and the core logs from the N and W boreholes, there was reason to believe that significant fracture zones existed within the SCV site. To determine the geometric characteristics of these zones, and to locate any other zones that may have been missed by the boreholes, crosshole radar and seismic tests and single-borehole geophysical tests were conducted in the boreholes in Stage I. The testing was conducted in semi-horizontal planes defined by the three N boreholes and the two W boreholes, and confirmed the existence of a number of major and minor fracture zones. Although these zones dipped steeply, their orientations and thicknesses could not be determined with a high degree of resolution because the testing configuration was restricted to a single horizontal plane. In Stage III, further radar and seismic testing, including single-borehole directional radar testing, expanded the database to the point where the geometric characteristics of the fracture zones could be defined with a relatively high degree of accuracy.

The hydrologic characteristics of the SCV site were evaluated by means of hydraulic packer tests in single boreholes and in the crosshole mode, and by hydraulic head monitoring with the Piezomac system that was installed in existing boreholes in the Stripa mine. The hydraulic tests focussed on the most permeable sections of each borehole, and straddle packers were used to isolate intervals of variable length containing single fractures, when possible. The testing indicated that the bulk of the groundwater flow occurred in a few highly transmissive fractures that represented about 1% of the total number of fractures. Measurements of hydraulic head indicated a pattern of high heads to the northeast of the site, with a decrease in heads to the south and west. In effect, the pattern reflected the likely movement of groundwater draining into the mine excavations. Based on the transient data from both the single-borehole and crosshole hydraulic tests, the fracture system within the SCV site was found to behave as a well-connected network. The hydraulic flow properties of the B and H zones, located predominantly within the centre of the site, were consistent with spherical to cylindrical flow geometries. The geochemical studies indicated that mixed water was formed within the site by the drawdown of shallow water and the upwelling of deep water as a result of groundwater draining from the D boreholes over relatively long periods of time.

24.6.2 Conceptual model of the SCV site

The conceptual model of the SCV site was based on a binary representation of the rock mass, distinguishing between distinct "fracture zones" and "averagely fractured rock". A "fracture zone index" was developed from a detailed analysis of data from single borehole measurements, including normal resistivity, sonic velocity, hydraulic conductivity, fracture frequency, and the occurrence of single-borehole radar reflectors. The analysis involved the calculation of a matrix of correlation coefficients for the data set, and from this, the eigenvectors for the matrix. Each eigenvector represented a particular weighing of the data so that the eigenvector with the largest eigenvalue was taken to indicate the most intensely fractured part of the rock. The frequency distribution of the fracture zone index exhibited a skewed distribution consisting of two parts, as shown in Figure 24-11. One part is a basically normal distribution centered around a mean value very close to zero, representing the "averagely fractured rock"; the other part is

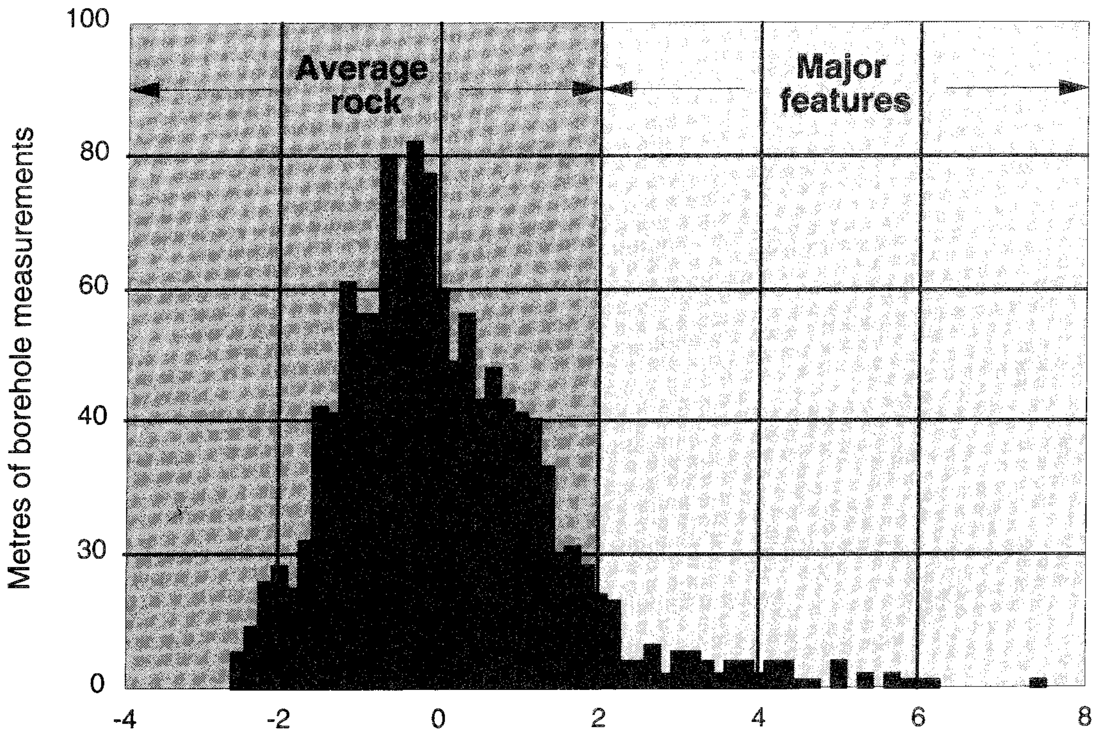


Figure 24-11. *Frequency distribution of "fracture zone index" /24-11/.*

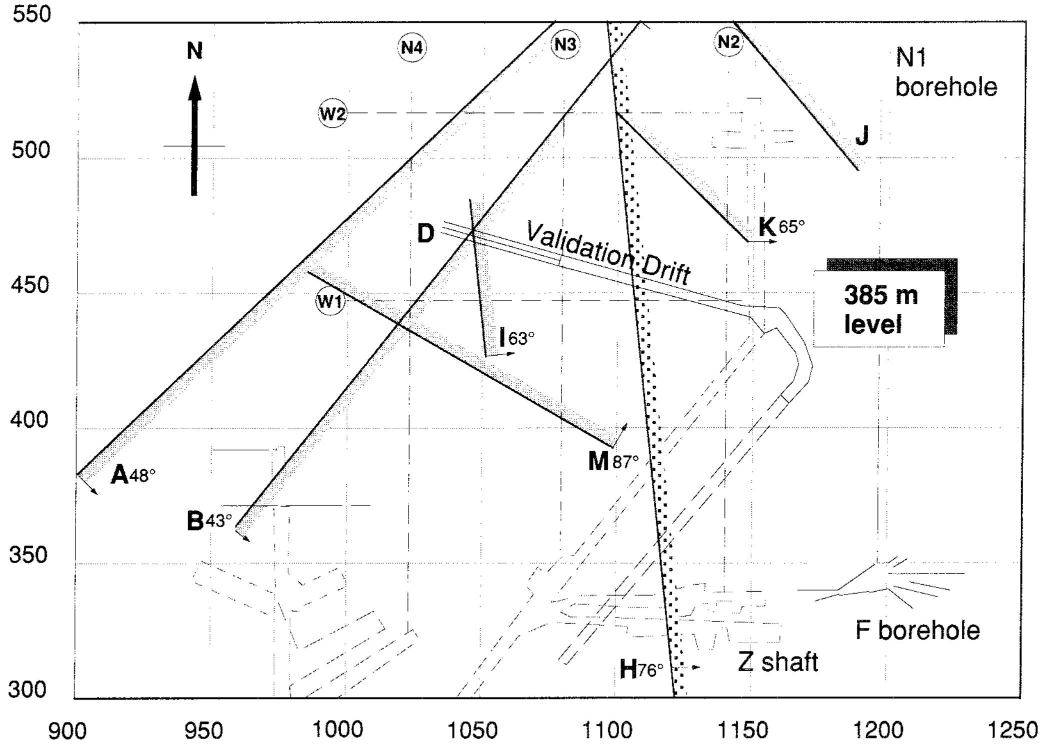
Values from the tail of the distribution (FZI>2) are designated as "fracture zones" while values less than 2 are designated as "average rock".

the tail of values greater than two, representing "fracture zones". These fracture zones occupied about 7% of the total length of boreholes at the SCV site. The geometric and hydraulic details of the fracture zones were determined subsequently by crosshole testing with radar, seismic and hydraulic methods. The geometric model, developed on the basis of the fracture zone index and the remote sensing data, was checked iteratively for consistency with the crosshole hydraulic responses, head monitoring data, groundwater geochemistry and geologic information from the corelogs and drift wall mapping.

As shown in Figure 24-12, the conceptual model of the SCV site contains three major fracture zones (A, B, and H), ranging from 2 to 12 m thick. These features are believed to extend beyond the limits of the SCV site and to connect to the ground surface. This connection between the SCV site and the surface was thought to cause the high groundwater heads observed at the site. These zones accounted for about 75% of the hydraulic transmissivity, as measured by hydraulic tests in single boreholes. In addition, three minor fracture zones (I, K, and M), were identified. Fractures D and V were inferred on the basis of radar and seismic testing, but were not confirmed by subsequent tests and observations. These minor zones, which had extensions of 50 to 100 m and provided hydraulic connections between zones A, B, and H, accounted for approximately 4% of the hydraulic transmissivity measured in the boreholes. Crosshole hydraulic testing confirmed that the conceptual model was consistent with the groundwater flow through the site.

24.6.3 Validation experiments

Prior to the outset of the SCV programme, a decision was made to perform at least two experiments to evaluate the validity of modelling approaches for simulating groundwater flow and solute transport. The first experiment (later named the Validation Drift Experiment) involved measuring groundwater flow into a drift constructed within the SCV site. This experiment was followed by a second experiment (later named the Tracer Migration Experiment), in which tracers were injected into the rock mass surrounding the drift and collected in the drift. During the development of the investigation plans for the SCV programme, it was decided that the first experiment should be preceded by an experiment (known as the Simulated Drift Experiment) that involved measuring the inflow of groundwater to an array of boreholes which outlined the periphery of the Validation Drift in the SCV site. At the time these experiments were planned, the location and size of the SCV site had been identified in a



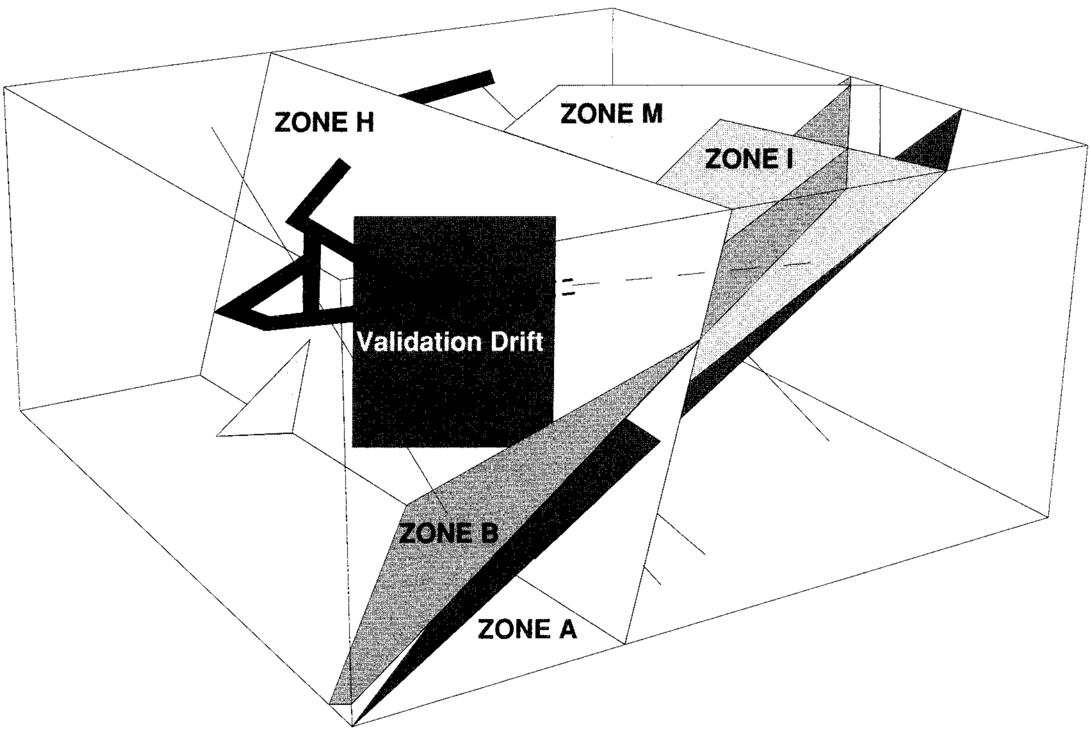


Figure 24-12. *Plan and perspective views of the conceptual model of the geologic structure at the SCV site /24-11/.*

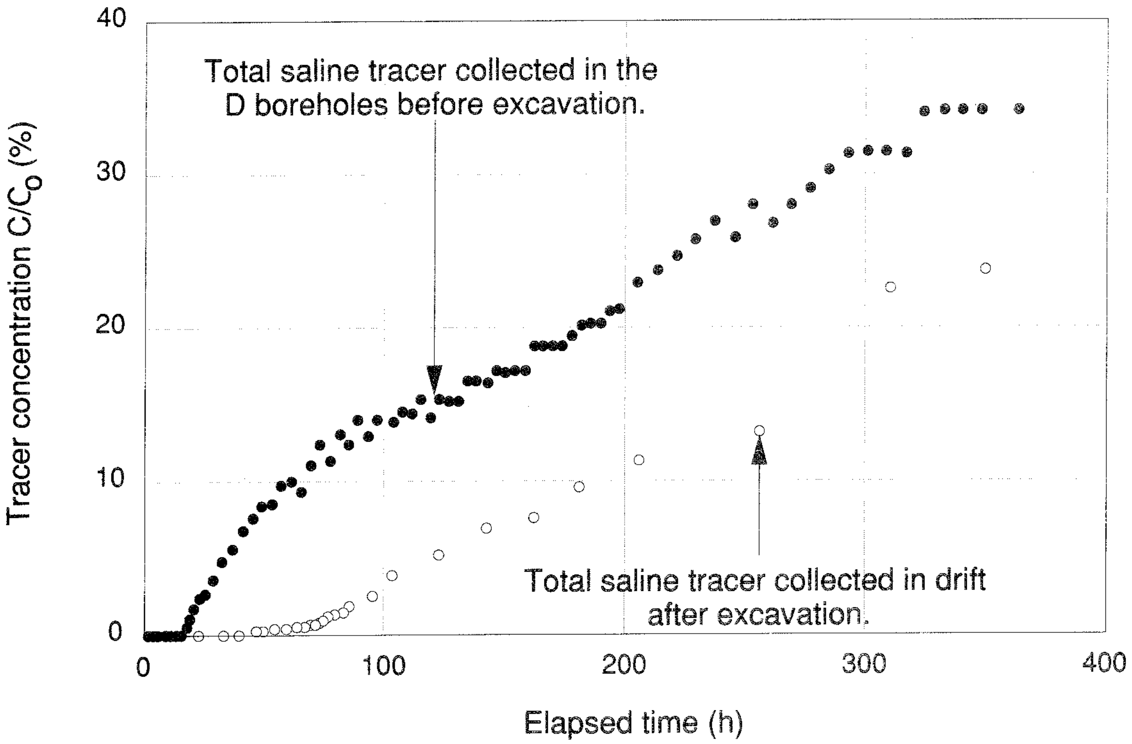
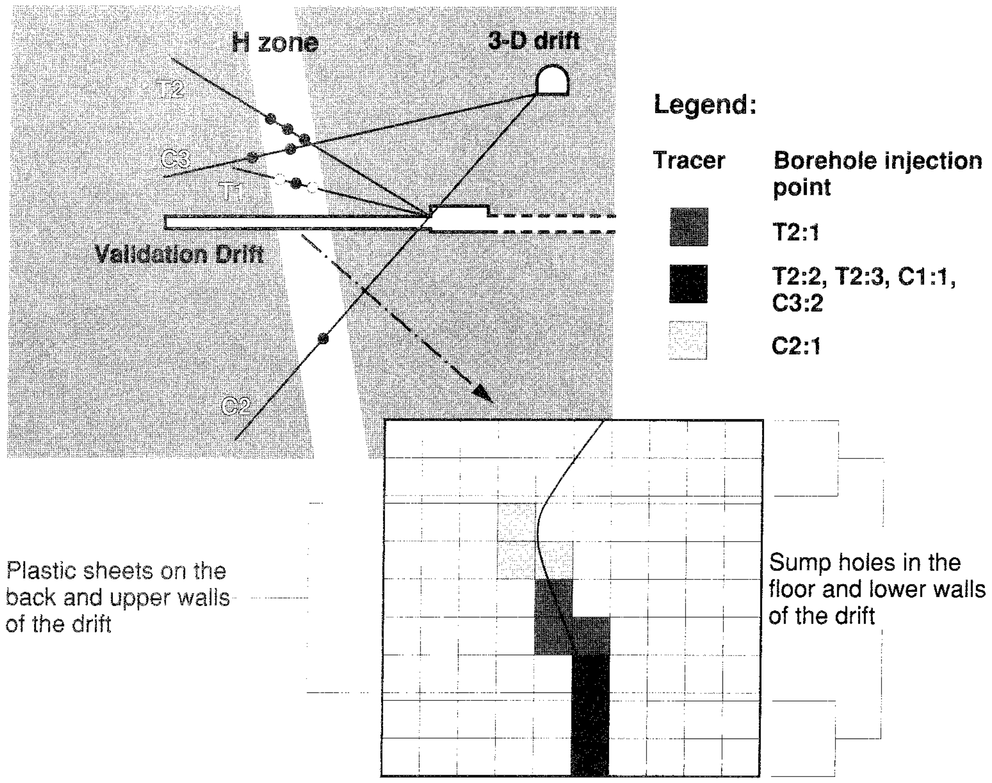


Figure 24-13. *Comparison of the total saline-tracer collection before and after excavation of the Validation Drift at the SCV site /24-11/.*

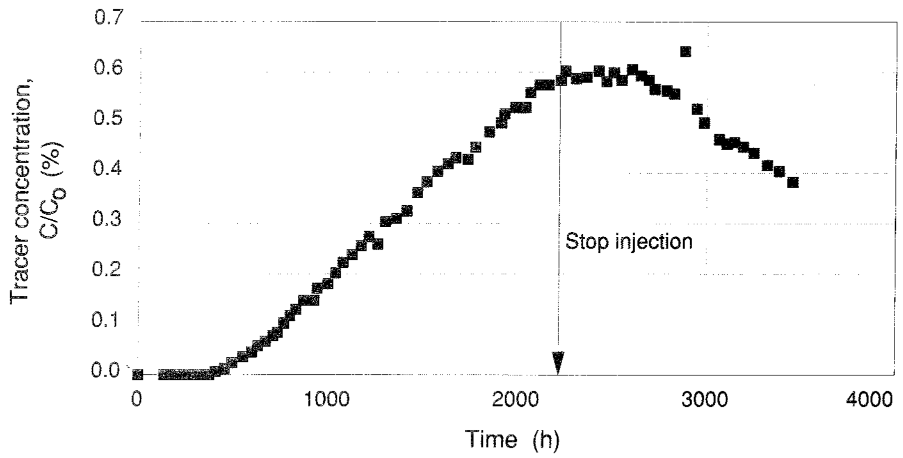
preliminary fashion, but the structural and hydraulic characteristics of the rock mass were, for all practical purposes, unknown. The orientation and dimensions of the Validation Drift would be selected after the preliminary characterization of the SCV site had been completed in Stage I of the programme. In addition, the details of the experiments could be developed only after this characterization information became available.

The list of experiments performed at the SCV site is shown in Table 24-3. In addition to the three experiments described above, five experiments were conducted during the various stages of the SCV programme to evaluate the validity of certain aspects of the modelling approaches, design the tracer migration experiment, and assess the influence of the disturbed zone around the Validation Drift on groundwater flow and solute transport. Groundwater flow to the remaining sections of the D boreholes was measured in the second simulated drift experiment, after construction of the Validation Drift. The measurement was intended to quantify precisely the inflow, mainly from the "averagely fractured rock", together with any disturbance induced by the Validation Drift. The fractures in the roof, floor and walls of the Validation Drift were carefully mapped for comparison with stochastic predictions of the patterns of such fractures by the fracture-network components of the groundwater flow and transport models. The groundwater heads within and around the SCV site were monitored continuously during the characterization, construction and experimental activities. These data were used for developing the numerical models of the hydrogeology of the site, and for comparison with the hydraulic heads predicted by the models.

Two radar/saline tracer experiments were conducted to evaluate the validity of solute transport models, provide input to the design of the subsequent tracer migration experiment, and evaluate the influence of the disturbed zone around the Validation Drift. The radar tomograms depicted the migration of tracers within the H zone before and after construction of the Validation Drift. The comparison of the saline tracer collected in the D boreholes before excavation with the saline tracer collected in the Validation Drift after excavation is shown in Figure 24-13. The disturbance created by the drift reduced the recovery of tracer to less than half of that recovered in the D boreholes, and increased the breakthrough time by many hours. Additionally, the total rate of groundwater flow into the drift was only about 10-15% of that collected in the first 50 m of the D boreholes before excavation. The radar tomograms indicated a significant redirection of tracer migration from the H zone in the vicinity of the Validation Drift to adjacent minor fracture zones. In effect, the "sink" effect was stronger in the first experiment than in the second one, even though the source strength was the same in both cases. In contrast, the measured head difference in the second experiment was approximately 340 m, compared to 65 m in the first experiment. Clearly, the disturbed zone significantly reduced the transmissivity of the rock immediately around the drift, thereby altering the magni-



Location of major tracer recovery in the Validation Drift.



Breakthrough curve for tracer injected in borehole T2.

Figure 24-14. *Schematic layout and typical results from the tracer test in the Validation Drift /24-10/.*

tudes and directions of the components of the hydraulic conductivity tensor.

As shown in Figure 24-14, the subsequent tracer migration experiment focussed principally on the migration of tracers in the H zone because most of the groundwater inflow into the Validation Drift originated from that fracture zone. The experiment used dyes and metal complexes as tracers. All but one of the injection points were located primarily in the zone above and below the drift. Significant quantities of tracers arrived at the measuring points in the drift within a few hundred hours. Most of the mean residence times for individual tracers, based on fitting the breakthrough curves with an advection-dispersion model, were calculated to range between 1,200 and 5,000 hours (50 to 208 days).

24.7 GROUNDWATER FLOW AND SOLUTE TRANSPORT MODELS

During the last stages of Phase 2 and throughout Phase 3, the characterization tools and methods were applied to develop and refine models of the hydrogeology of the Stripa mine and its surroundings. In particular, hydrogeologic models were developed for the Crosshole site in the vicinity of the 360 m level of the Stripa mine, the general area of the Stripa mine, and the SCV area in the vicinity of the 385 m level of the mine. In addition to providing a basis for the development and application of groundwater flow and transport models, these models were useful in demonstrating the applicability and credibility of the characterization tools and methods.

The principal objective of the SCV programme in Phase 3 was to integrate different tools and methods, in order to 1) predict the groundwater flow and transport in a specific volume of the Stripa granite and 2) compare the predictions against the experimental measurements made at the SCV site. In 1987, the JTC established a Task Force on Fracture Flow Modelling to guide the development of numerical models for groundwater flow and solute transport and to develop criteria to be used for their verification and validation.

The fracture flow modelling studies were carried out by three independent groups: the Atomic Energy Authority (AEA) Harwell in the United Kingdom, sponsored by the Stripa Project; and the Lawrence Berkeley Laboratory (LBL) and Golder Associates, both in the USA and sponsored by the U.S. Department of Energy (USDOE). In addition, Fracflow Consultants in Canada, sponsored by the Stripa Project, were responsible for development of the equivalent porous media (EPM) models of the Stripa area, the Stripa mine, and the portion of the mine within which the SCV site was located.

24.7.1 Model development

Development of the models began with procedures for the stochastic generation of fracture networks, was followed by modelling of groundwater flow in the SCV site, and concluded by modelling tracer transport by groundwater flow in the SCV site.

The modelling methodology developed by the group at AEA Harwell was able to account explicitly for the fractures identified by the remote sensing tests at the SCV site, as well as to generate stochastically a fracture network for parts of the rock mass at the SCV site where statistical properties of the fracture system were known /24-12, 24-13/. The AEA Harwell approach is capable of including variability of transmissivity within the fracture planes or, alternatively, of deriving a permeability tensor for the intact rock to be used in a stochastic continuum model.

The methodology developed by Golder Associates involves creation of a semi-stochastic discrete fracture system that combines deterministic information on portions of fracture zones with statistical information on the properties of hydrologically conductive fractures /24-14/; /24-15/ and /24-16/. The fractures in the intact rock are assumed to be distributed randomly, and the fracture area per unit volume of rock is increased for fracture zones identified by geophysical methods. The discrete fracture network model is shown in Figure 24-15.

The approach adopted by LBL focussed on the fracture zones that were identified by geophysical methods, and considered the intact rock to be impermeable /24-17/; /24-18/. The fracture zones were discretized into a regular grid. Each grid consisted of a combination of equally conductive channels (i.e. grid spaces) and "blocked" or non-conductive channels, in order to provide an equivalent representation of heterogeneity. The fractures within the zones were represented as a partially filled lattice of one-dimensional conductors, where the degree of filling in a fracture was selected by comparing the predicted flow in the fracture against the actual hydrological data.

The simulation of groundwater flow and transport at Stripa by the EPM modelling approach, adopted by Fracflow Consultants, used the established CFEST (Coupled Flow, Energy and Solute Transport) code /24-12/; /24-19/.

24.7.2 Model validation process

The Task Force on Fracture Flow Modelling established, first, an exercise for cross- verification of the various modelling approaches for groundwater flow and transport; and, second, a process for evaluating the validity of the models /24-20/. This process, see Figure 24-16, involved the task force, the principal investigators, the TSG and the JTC.

24.7.3 Model validation criteria

Recognizing that both definitions and requirements for model validation varied from country to country and that the issue was still being discussed at the international level, the task force decided to adopt an operational definition of validation for the SCV programme. Thus, a model would be considered to be validated for use in a given application when the model had been determined, by appropriate measures, to provide a representation of the process or system that was acceptable to an assembled group of knowledgeable experts.

In terms of the given application in the Stripa Project, the task force decided that judgments of model validity would be made only in the restricted sense of simulation of groundwater flow and solute transport in the saturated granitic rock mass in the SCV site. The "appropriate measures" were defined to mean a comparison of model calculations against field observations and/or data from

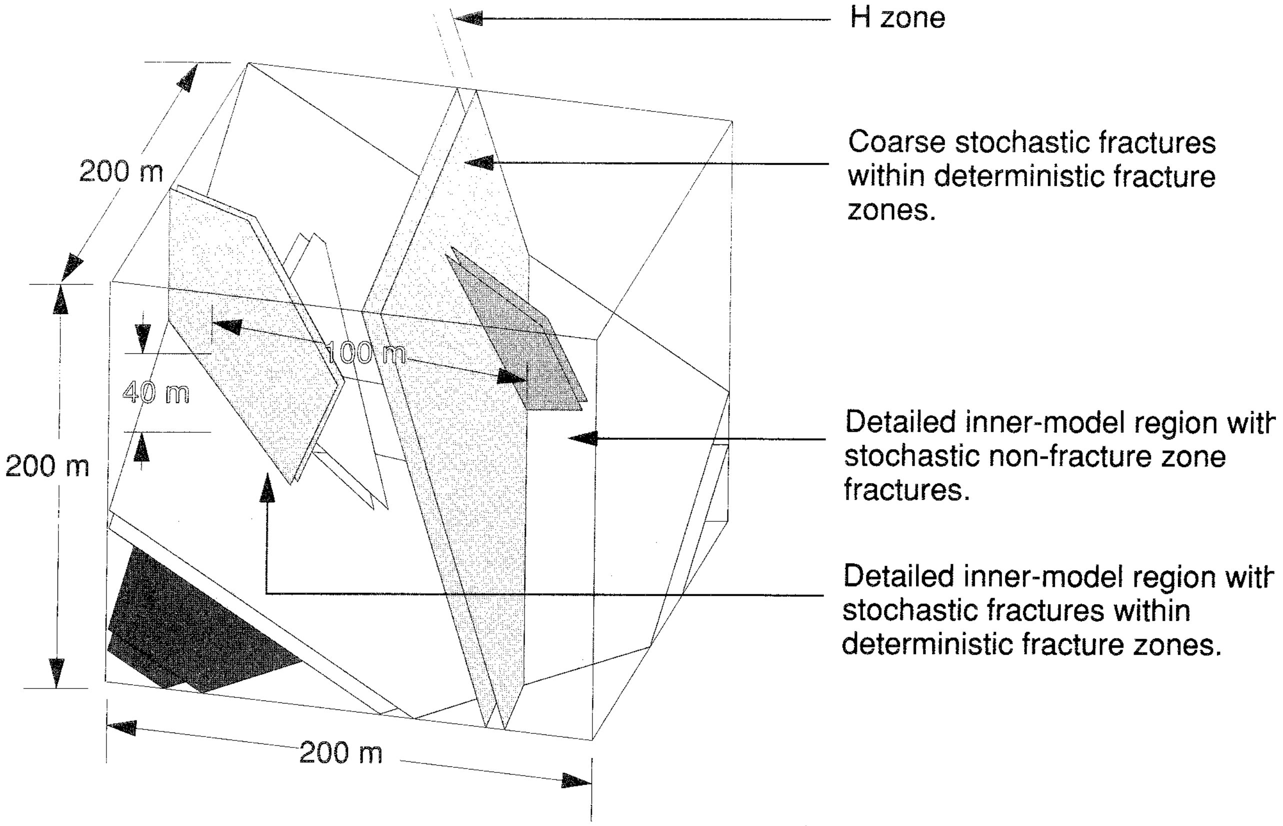


Figure 24-15. *Discrete fracture network model of the SCV site /24-14/.*

This discrete fracture network model, developed by the Golder Associates modelling group, encompasses the entire SCV site.

experimental measurements, considering the extent and nature of uncertainties in calculations, observations and measurements in relation to those allowable for purposes of the application. By decision of the JTC the members of the Task Force on Fracture Flow Modelling comprised the "group of knowledgeable experts".

With respect to the performance measures for the various validation exercises, the task force established two sets of criteria for evaluating the validity of a modelling approach and the validity of the components of the modelling approach. These criteria were stated as two sets of questions.

The first set addressed both quantitative and qualitative features.

> *Quantitative*: do the predictive calculations adequately reflect the measured values? That is, are the predictions of the correct order of magnitude as compared to the measurements?
>
> *Qualitative*: for the purposes of this application, are the predicted distribution patterns sufficiently accurate as compared to the observations? That is, are the predictions of the patterns reasonable when compared to the observations?

The second set addressed the usefulness and feasibility of a modelling approach from the larger viewpoint of general applicability.

> *Usefulness*: is the modelling approach useful for representing groundwater flow and transport in a hydrogeologic environment similar to that of the SCV site?
>
> *Feasibility*: can the data required to support fully the modelling approach be collected in a feasible and timely manner?

Evaluation of the validity of the groundwater flow and transport models involved two steps: 1) a "training exercise" and 2) a "validation exercise". In each exercise, the modellers were asked to predict the results of the validation experiments conducted at the SCV site without prior knowledge of the outcomes.

24.7.4 Groundwater flow models

The training exercise for the groundwater flow models involved predicting the total rate of groundwater flow

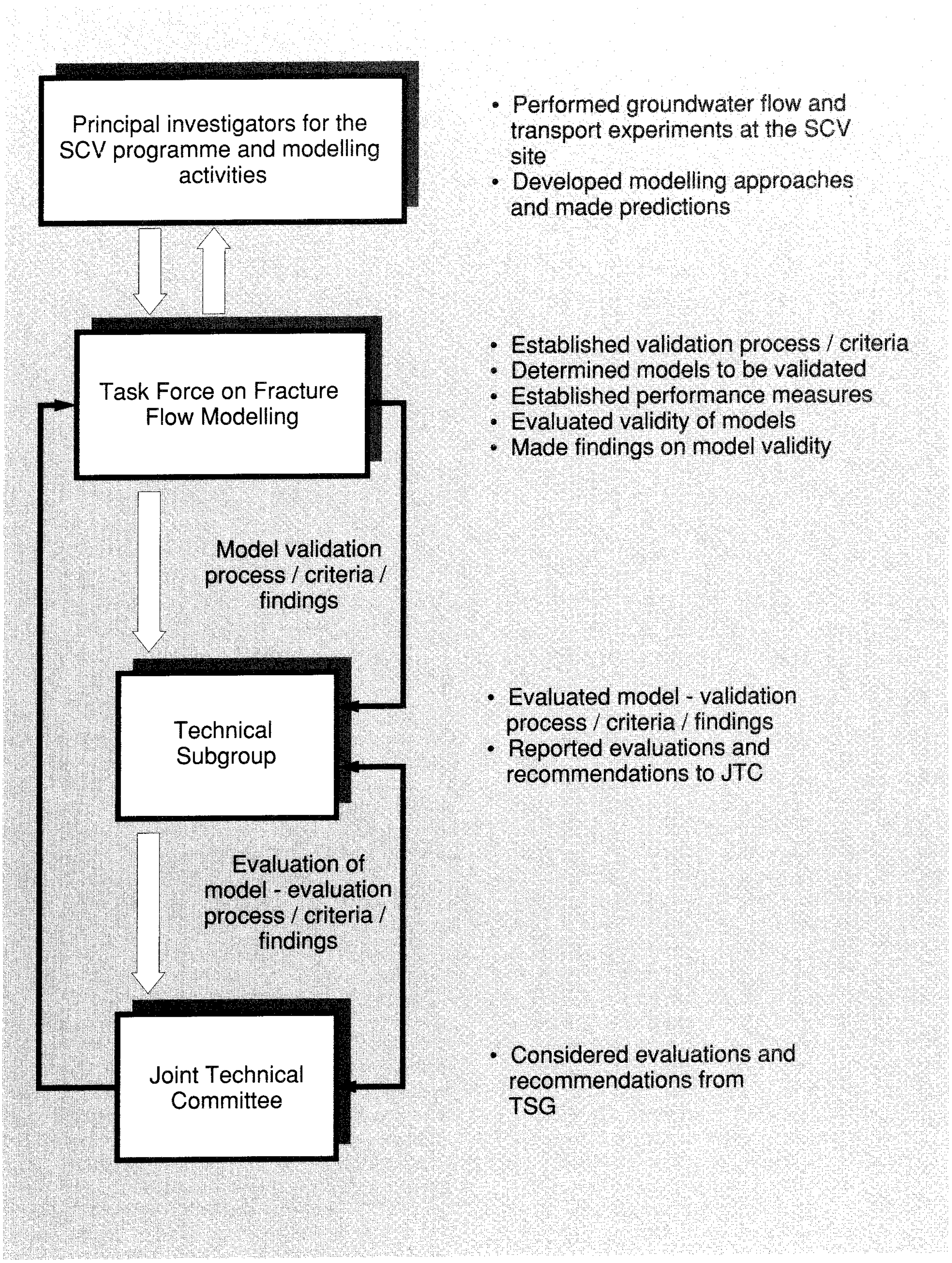


Figure 24-16. *Process for evaluating the validity of the groundwater flow and transport models at the SCV site /24-20/.*

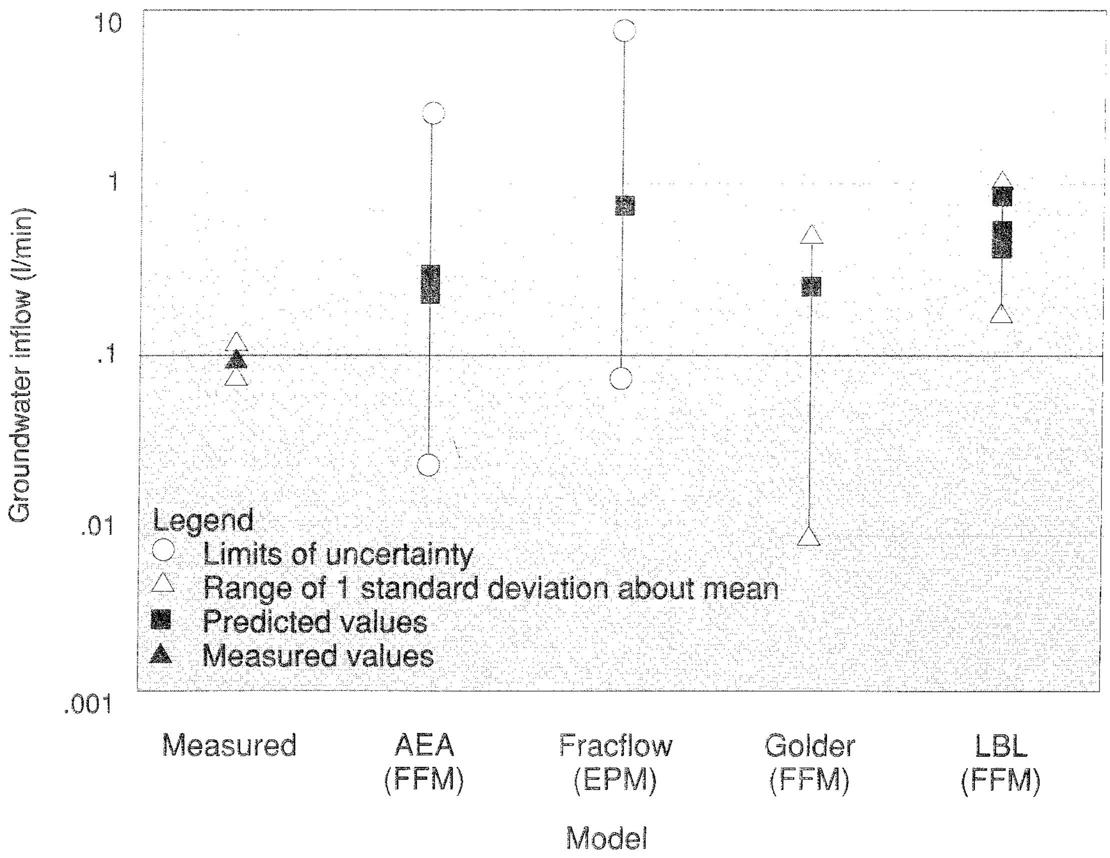


Figure 24-17. Comparison of the measurements with the model predictions for groundwater inflow into the Validation Drift /24-22/.

FFM designates fracture-flow model and EPM designates equivalent porous media model.

into the D boreholes at the SCV site, and predicting the distribution of groundwater inflow along the D boreholes /24-21/. The modellers were allowed to use the preliminary hydrogeologic model that had been developed on the basis of data gathered in Stage I of the SCV programme. The measurements at the SCV site yielded a water inflow to the boreholes in the range of 1.67 to 1.75 l/min, with the bulk of the inflow originating from the H zone and the three other major fracture zones, see Figure 24-12. The three fracture-flow modelling approaches predicted mean inflows of 1.45 l/min (Harwell/Fracflow); 1.5 l/min (Golder); and 3.1 l/min (LBL), all of which are close to the measured inflow. In addition, the models predicted that the bulk of the inflow would originate from the major fracture zones. The agreement between predictions and measurements, from both the quantitative and qualitative viewpoints, was remarkably good, and provided confidence that some acceptable level of success could be achieved in the validation of the models.

In step 2, the validation exercise involved predicting: (i) the total rate and distribution of groundwater inflow to the Validation Drift; (ii) the rates of groundwater flow into the drift from the significant fracture zones; and (iii) the response of the groundwater head distribution in the SCV site to construction of the drift /24-22/. In addition, the modellers were asked to predict (iv) the fracture patterns that would be observed in the walls of the drift, and (v) the distribution of groundwater flow into the remaining sections of the D boreholes that were located beyond the end of the drift. The modellers were allowed to use the conceptual hydrogeologic model of the SCV site that included the refinements and additional detail provided by the test data collected during Stage III of the SCV programme. Finally, the modellers had to consider the effect of stress redistribution, caused by drift construction, on the hydrologic properties of the rock mass.

Based on measurements in the Validation Drift, the groundwater inflow was considerably less, by a factor of about 8, than was expected on the basis of the inflow into the D boreholes /24-11/. Approximately 97% of the inflow originated from the H zone. Although the values of predicted groundwater inflows to the Validation Drift varied considerably from those measured, the mean values were all within a factor of 2 to 7 of the measured inflow, see Figure 24-17. The influence of stress redistribution in the rock mass because of drift construction was treated differently by the various modellers. In the EPM model used by Fracflow Consultants and in the fracture flow model used by AEA Harwell, the fracture transmissivities were modified according to the calculated change in the state of stress and a laboratory-derived relationship between transmissivity and normal stress. The fracture flow models used by Golder Associates and LBL incorporated a skin factor, based on empirical evidence, wherein the transmissivity of the rock mass within 3 to 5 m of the drift wall was reduced by about an order of magnitude.

The predictions of both the fracture patterns in the Validation Drift and the magnitude and spatial distribution of head changes in the SCV site due to excavation of the drift were reasonably good, considering the stochastic nature of the fracture networks. The principal difference between the three fracture flow modelling approaches consisted of the assumed density of fractures in the rock mass between the discrete fracture zones. AEA Harwell and Fracflow Consultants employed fracture statistics to define a network model that was used to determine the permeability of an equivalent homoge neous medium, which in turn provided the boundary conditions for more detailed models of inflow. The fracture flow model of Golder Associates included the fractures explicitly, focussing on the most transmissive fractures, with inferred fracture frequencies in the H zone and in the intact rock near the drift, and no fractures in the intact rock situated away from the drift. The LBL group modelled flow in the fracture zones only, and assumed the intact rock to be impermeable.

Considering only the effect of stress redistribution on the transmissivity of the rock mass in and around the Validation Drift, none of the modelling groups was able to explain the substantial decrease in groundwater flow into the drift. However, as the radar tomograms from the saline tracer experiments demonstrated, the disturbed zone caused a significant redirection of groundwater flow away from the drift. This suggested that the hydraulic conductivity of the rock in the immediate vicinity of the drift had been reduced, perhaps substantially.

24.7.5 Solute transport models

In order to evaluate the validity of transport models, the training exercise in step 1 involved predicting the results of the second radar/saline tracer test /24-23/. In this test, the saline tracer was collected in both the SCV drift and boreholes in the H zone. The modellers were asked to predict the saline breakthrough curves and to develop histograms of the breakthrough concentrations in the D boreholes. They were allowed to use the refined hydrogeologic model of the SCV site, the groundwater flow measurements into the Validation Drift and the results of the first radar/saline experiment. Table 24-4 compares the measurements with the predictions produced by the Golder fracture flow model. The saline breakthrough times and concentrations were well within an order of magnitude of those measured, and the predicted locations of measurable breakthrough concentrations were reasonably accurate.

Step 2 of the effort to evaluate the validity of the transport models involved predicting the outcome of the tracer test in the Validation Drift /24-23/. As discussed earlier, this test involved injecting a variety of dyes and metal complexes, principally in boreholes that intersected the H zone, and collecting the tracers in the Validation Drift. The modellers were asked to predict the tracer concentrations and arrival times in the Validation Drift and the boreholes, and the pattern of distribution of tracer arrivals in the drift. As shown in Figure 24-18, the predictions of tracer concentration agreed reasonably well with the measurements. However, the models tended to underestimate the time required for the tracers to travel from the injection points in the H zone to the collection points in the Validation Drift.

24.7.6 Assessment of model validity at the SCV site

At the close of the SCV programme, the task force made consensus judgments on the validity of the various modelling approaches in simulating groundwater flow and transport at the SCV site /24-22, 24-23/ and /24-24/. All of the modelling approaches were judged to be capable of making predictions that were the correct order of magnitude as compared to the experimental measurements. It must be noted, however, that the influence of the disturbed zone on groundwater inflow to the Validation Drift could not be totally explained by the manner in which the effects of stress redistribution were treated. Furthermore, the transport models tended to underestimate the time required for a tracer to migrate from the point of injection to the point of recovery. This discrepancy may have been due to the choice of the value for dispersivity.

The task force framed its judgments of model validity against the qualitative criterion, see section 24.7.3, in

Table 24-4. Comparison of predictions with measurements and observations for the second radar/saline tracer experiment /24-23/.

Prediction	Parameter	Prediction	Measurement
Tracer breakthrough to the	t_5	30 to 150 h	60 to 70 h
Validation Drift	t_{50}	100 to 150 h	125 to 150 h

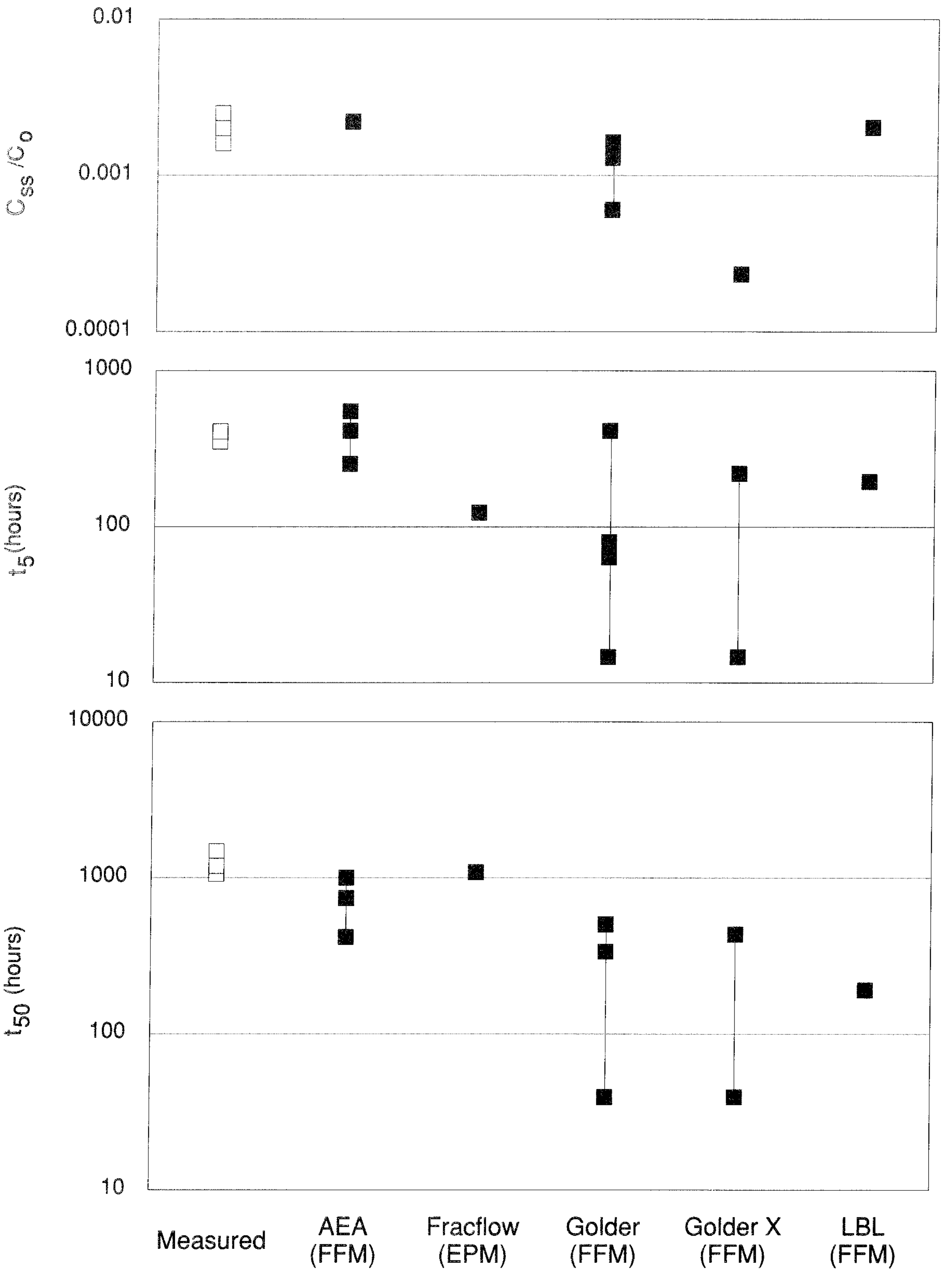


Figure 24-18. *A comparison of observed and predicted breakthrough curves for tracer injection in the first interval of borehole C2 and recovery in the Validation Drift /24-23/.*

C_{ss}/C_0 is the ratio of the "steady-state" tracer concentration measured in the drift to that injected into the borehole interval. t_5 and t_{50} are the times at which 5% and 50%, respectively, of steady-state breakthrough were observed. FFM signifies fracture flow model and EPM signifies equivalent porous media model. The vertical lines between the solid squares represent the ranges of predictions.

reference to the near field (within several excavation diameters of the drift wall) and the far field (beyond several excavation diameters). The fracture-flow modelling approach used by Golder Associates was judged to yield reasonable patterns of phenomena in both the near field and far field. A similar judgment was made for the combined application of the fracture-flow modelling approach by AEA Harwell and the EPM modelling approach by Fracflow Consultants. The very detailed stochastic modelling of the fracture network in the AEA Harwell approach yields good results in the near field, but is somewhat impractical for a large rock mass because it requires a computer with a very large storage capacity. On the other hand, the EPM model gives good results in the far field, but requires a high degree of discretization of the finite element mesh in order to achieve good results in the near field. The modelling approach chosen by LBL considered groundwater flow and solute transport through deterministic fracture zones, and assumed that the rock mass between fracture zones was impermeable. As a consequence, there was no basis on which to judge whether the model could produce reasonable patterns of phenomena in the near field. Furthermore, the number of realizations for the experiments was insufficient to demonstrate that the approach was capable of producing consistent patterns. However, the task force felt that the modelling approach had potential, perhaps more as a research tool than as an applied technique.

The task force concluded that the fracture-flow modelling approach by Golder Associates, and the combination of approaches used by AEA Harwell and Fracflow Consultants for fracture-flow and EPM modelling, respectively, were useful for simulating groundwater flow and solute transport in a hydrogeologic environment similar to the environment at the SCV site. Because the number of realizations was not sufficient to fully demonstrate the usefulness of the LBL modelling approach at the SCV site, the task force could only judge the approach to be potentially useful. The consensus judgment of the task force was that the necessary data could be collected in a feasible and timely manner to support each of the four modelling approaches. This fact was clearly demonstrated over the five years of the SCV programme. It must be kept in mind, however, that the data required to support the modelling approach by LBL are principally the product of hydraulic head testing in a groundwater system, whereas the data required to support the other three modelling approaches are derived from fracture mapping and hydraulic packer tests.

24.8 SIGNIFICANT ACHIEVEMENTS

Over a period of some thirteen years, the investigations of methods and techniques to characterize the natural barriers evolved through successive stages of learning, development and application. This evolution of activities involved the collective thinking and efforts of the group of scientists and engineers from seven to nine countries who were responsible for the planning, review, investigation and management of the project.

The significant achievements of the project, in terms of their contribution to site characterization in general, were:

- *Demonstration of a programme of progressive site characterization.* The SCV site, which encompassed about 1 million m^3 of rock, was characterized in progressive stages over a period of about five years. Based on remote-sensing data from only five boreholes in the site during the first stage of characterization, it was possible to identify the locations of the principal structural features and hydraulic anomalies. This information provided a basis for deciding upon the orientation and length of boreholes to be drilled to obtain a more detailed characterization of these features and anomalies. The test data from these additional boreholes subsequently provided the basis for determining the dimensions of the Validation Drift that was constructed in the SCV site and for designing the groundwater inflow and tracer tests in the rock mass surrounding the drift. At the outset of the SCV programme, it had already been decided that these tests would be conducted in the last stage of the programme. However, the specific details of the tests could not be determined until the principal structural features and hydraulic anomalies within the site area had been identified and initially characterized.

- *Procedure for developing a conceptual hydrogeologic model.* During the mid-years of the Stripa Project, the principal investigators developed conceptualizations of the hydrogeology of the Stripa area and the Crosshole site in the Stripa mine. The conceptual model of the Crosshole site was developed principally to illustrate the integrated use of core logs, borehole photography, and single-hole and crosshole testing with radar, seismic and hydraulic techniques. The model of the Stripa area was developed principally to complement the characterization work that was beginning at the SCV site, as well as to illustrate how the large collection of geologic, hydrologic, and geochemical data for the general area and the mine could be assembled in a consistent, realistic fashion. During the early stages of the SCV programme, a conceptual model of the hydrogeology of the SCV site was developed on the basis of core logs, drift mappings, borehole test data, and the information provided by the conceptual model of the Stripa area. In the later stages of the programme, as more data was obtained from additional boreholes, the conceptualization of the hydrogeologic characteristics of the site was refined. Through these efforts, a procedure for the rational and consistent development of a conceptual hydrogeologic model of the Stripa rock mass evolved. The procedure incorporated testing techniques and methods, consistency checks and decision points concerning the need for more data. Although the intended use of a conceptual model will differ from site to site, as will the amount of data required for details and consistency checks, this procedure serves as an example of a rational and systematic development process.

- *Demonstration of a process for evaluating the validity of numerical models.* An evaluation of the validity of models of groundwater flow and solute transport was an integral part of the SCV programme. To this end, the JTC established the Task Force on Fracture Flow Modelling, consisting of recognized experts from the member countries. The task force, in concert with the principal investigators, developed a formal process for evaluating the validity of the models that were applied at the SCV site. The process involved defining the appropriate measures for the evaluation, including the scope of the evaluation, and the criteria against which the validity of the various models and modelling approaches could be judged by a group of knowledgable experts. The process required close cooperation and coordination among the experimenters, the modelers, and the experts. At the close of the SCV programme, the experts made documented judgments of the validity of various approaches for modelling groundwater flow and transport in the SCV site. This effort constitutes a case-history example of a formalized and deliberate approach to evaluating the validity of numerical models for a very specific application.

From the particular viewpoint of a saturated, fractured granitic rock mass, the significant achievements of the project were:

- *Development of a suite of tools for hydrogeologic characterization.* These tools included the radar, seismic and hydraulic methods for both single-borehole and crosshole testing, as well as borehole photography and drill-core observations. For site conditions similar to those encountered at Stripa, including groundwater with a low salinity, tests in boreholes spaced as much as 200 m apart can be expected to produce data to identify the principal structural features and hydraulic anomalies. The hydraulic conductivities of the rock mass in the SCV site ranged from 10^{-7} m/s for the fracture zones to as low as 10^{-11} m/s for the "competent" rock.

- *Development of methods and techniques for geochemical characterization.* This work involved the development and implementation of methods and techniques to determine the history, or residence time, and origin of groundwater in the granitic rock mass at Stripa, including water sampling techniques and identification of concentrations of key radionuclides.

- Demonstration of the applicability of the equivalent porous media (EPM) approach for simulating groundwater flow and transport. The EPM modelling approach was used to simulate groundwater flow and transport in hydrogeologic models of the Stripa area, the Stripa mine and a portion of the mine containing the SCV site. These models represented volumes of rock ranging from about $300 \cdot 10^9$ m^3 to some $5 \cdot 10^6$ m^3. The calculations of groundwater inflow to the mine agreed quite well with the measured pumping rate when the models for the Stripa area and the Stripa mine were used. This agreement is a significant achievement, considering the sizes of the areas modelled and the relative sparseness of the geologic information and hydraulic data. The predictions of groundwater flow and transport within the SCV site for the validation experiments also were reasonably good, considering the relatively coarse discretization of the rock mass as compared to the dimensions of the Validation Drift.

- Demonstration of the applicability of the fracture flow modelling for simulating groundwater flow and solute transport. Three fracture flow modelling approaches were used with reasonable success to simulate groundwater flow and transport within the SCV site. The approaches differed in the manner in which the fracture networks were constructed, in the exent of the regions modelled within the site, and in the types of input data required from the characterization activities. The combination of approaches and the differences among them provided a rather robust demonstration of modelling capability and usefulness.

- Specification of requirements for design of a tracer test to identify groundwater flow paths. The tracer tests at Stripa demonstrated the importance of understanding the groundwater flow field and potential pathways in a rock mass prior to conducting a tracer test. There is little hope of quantifying the migration characteristics of a pathway unless the geometries of the pathway and the generalized groundwater flow field within the pathway are defined before tracers are injected into the pathway. This information can be obtained through the integrated use of radar, seismic and hydraulic borehole testing methods. In addition, the sorbing characteristics of the tracer substance must be determined *a priori* in the laboratory.

- Demonstration of the influence of the "disturbed" zone around an underground drift on groundwater inflow. The groundwater inflow to the Validation Drift at the SCV site was almost an order of magnitude smaller than that measured previously in the D boreholes, which outlined the drift periphery before excavation. The radar tomograms from the saline tracer tests indicated qualitatively a substantial redirection of groundwater flow in the H zone around the drift after it was constructed. However, the circumstances that caused the reduced inflow could not be explained quantitatively, even when the redistribution of the stress field, gas bubbles in the water, desaturation of the wall rock, etc., were considered. This unexplained effect was also observed at the site of the 3-D migration drift in the Phase 2 investigations.

24.9 ENGINEERED BARRIERS

24.9.1 Introduction

The construction and operation activities undertaken for a repository for heat-generating radioactive waste will provide, with time, increasingly detailed information on the performance characteristics of the engineered host rock. Features and phenomena that may be important to the performance of the repository or the safety of its operation will be uncovered. It is likely that the repository design will need to be modified in response to these findings. Moreover, engineering measures are generally planned to ensure repository performance. Although this iterative and flexible approach is commonly required in many of the engineering disciplines, it is so important to geotechnical works that it has been formalized by the term "the observational method" /24-25/. This method not only recognizes the need to observe important aspects of the rock mass and its response during construction, but also demands that engineering measures be available prior to construction to counter any reasonably foreseeable performance or safety concern. The engineered barrier studies for the Stripa Project were undertaken to provide a series of technologies that could be applied to these construction and development activities. As may be expected from an appreciation of the "observational method" the engineered barrier studies evolved throughout the programme to reflect the increasing understanding of the Stripa granite. Changing interests of the member countries as national programmes matured also influenced the Stripa work.

A wide range of materials is available for consideration for use in the construction of engineered barriers. Screening studies indicated that clay and cement-based materials were appropriate for study in the Stripa Project. Phases 1 and 2 (1981-1988) of the project focussed on the *in situ* application of swelling, bentonite-clay materials. Bentonite and cement-based grouts were studied in Phase 3 (1991-1992). For both clays and cements, the following issues were considered important to the isolation of heat-generating radioactive waste and addressed throughout the Stripa investigations: measurement of performance; application methods, including aspects of quality control; and material longevity. Specifically, *in situ* experiments yielded information related to 1) the application of laboratory-derived performance data to full-scale repository system performance and 2) the limits to which the engineering processes were effective in the Stripa granite. Laboratory, desk and modelling studies were the primary tools used to evaluate longevity.

To reflect the international interests of the programme, the studies were structured to give, as far as possible, generally applicable results.

During the evolution of the scientific investigation programme at Stripa, and concurrent with other international developments, a better understanding has developed of the natural barriers and of the interactions and necessary integration between natural and engineered barriers. The assessment of integration aspects has progressed from the near field to the far field. Thus, in Phase 1, the emphasis was on engineered barriers in the disposal holes and in the thermally affected zone of the repository; in Phase 2, investigations were aimed at plugging of repository openings such as boreholes, shafts and tunnels; and in Phase 3, considering the progress made in understanding the hydraulic features of the surrounding rock mass, as affected by the engineering processes, the emphasis shifted to sealing of natural and induced fractures in the rock.

The following chapters describe and analyse the important aspects of the engineered barriers studies, and summarises many positive results of the programme. These achievements were obtained by adjusting the programme in response to both expected and unexpected results. The adequacy with which issues could be addressed was constrained by costs and other administrative concerns. Balance was sought between the interests of the member countries and the principal investigators.

24.9.2 PHASE 1 – The Buffer Mass Test

In Phase 1, the Buffer Mass Test (BMT) was carried out to examine phenomena and processes related to plugging with clays the excavations near the heat-generating waste containers. The following aspects of repository design, construction and performance were examined:

- engineering feasibility of a specific concept for waste emplacement and plugging of disposal holes,
- materials behaviour, and
- qualification of performance models for hydro-thermo-mechanical interactions between HLW, plugging materials and surrounding rock.

The observations performed in the BMT were expected to assist in improving designs, confirming feasibility and improving understanding of ongoing processes.

Figure 24-19 presents the general layout of the BMT, which was carried out in the Swedish-American Cooperative programme (SAC) macropermeability room. The important information that the SAC investigations provided on the hydraulic properties of the rock mass close to the room could be used both in the design of the experiment and in the interpretation of the test data.

An important objective of the BMT was to investigate the transient processes of heat and water transfer in a highly compacted bentonite-based buffer and the effects of these processes on hydro-mechanical interactions between the rock and the buffer, immediately following waste emplacement in a repository. Given the short duration of the test, it was not possible to examine the radionuclide transport properties of the buffer, other than by implication.

The highly compacted bentonite (HCB) used as buffer in the BMT was prepared by statically compacting MX-80

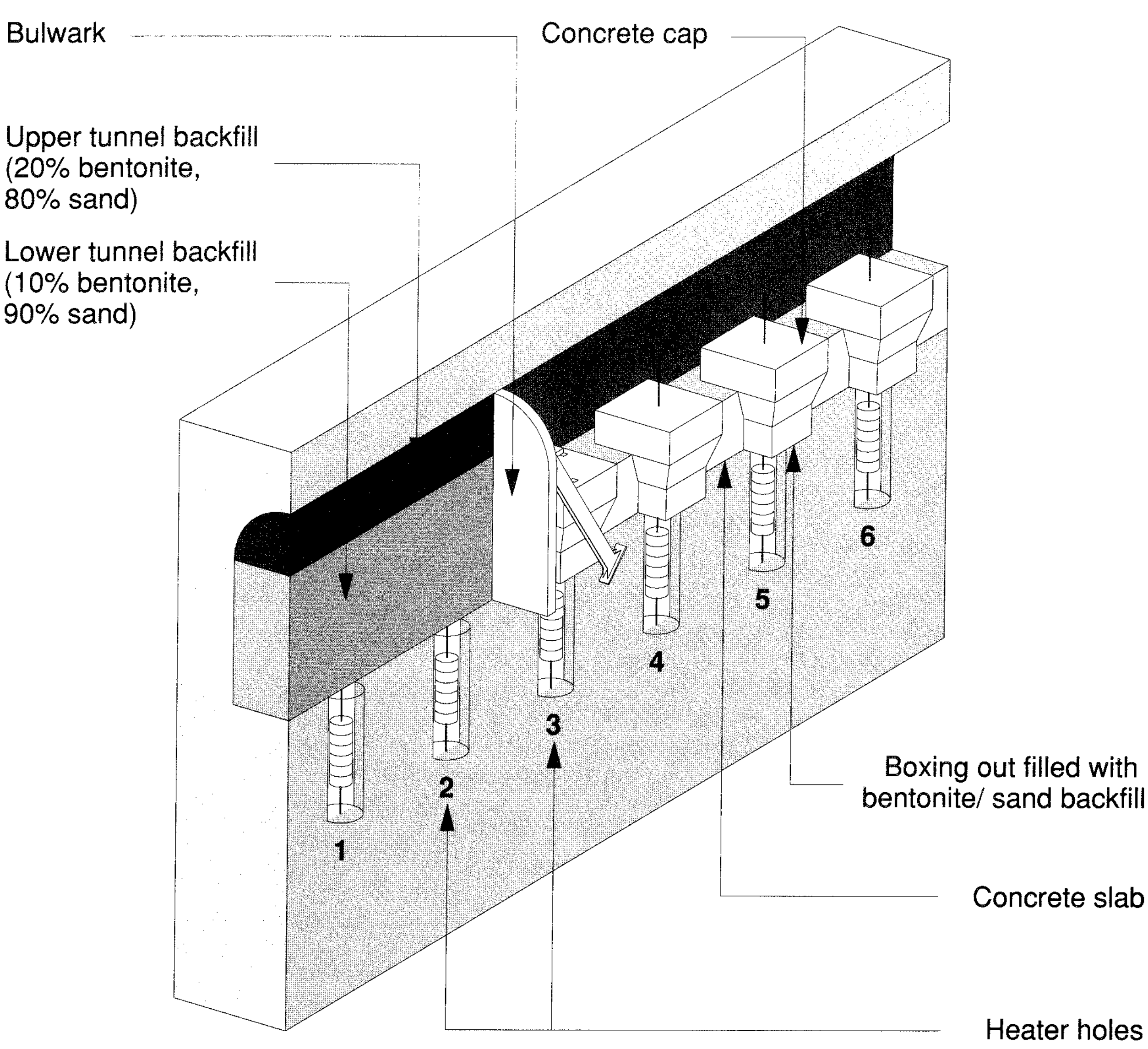


Figure 24-19. *The layout of the Buffer Mass Test.*

Six 160 mm diameter holes were drilled in line in the floor of the room. The holes were equipped with electrical heaters surrounded by highly compacted bentonite and overpacked with sand-bentonite mixtures. Temperatures, pressures and displacements were monitored over 4 years /24-29/ and /24-30/.

bentonite to a dry density of at least 1.88 Mg/m^3. The sand-bentonite tunnel backfill materials were compacted *in situ* to minimum dry clay densities of approximately 0.43 (lower backfill) and 0.37 Mg/m^3 (upper backfill). At these densities the buffer probably behaves, when intact, as an almost perfect semipermeable membrane; the backfill materials have lower densities and show lower osmotic potential.

Calculations using numerical models based on Fourier's law for heat transfer in the backfill/buffer/rock system indicated that the test needed to be run for approximately three months to allow for observations on the validity of the model. Calculations for moisture transfer in the system, using numerical models based on isothermal moisture diffusion equations, indicated that the experiment needed to be operated for one or more years.

Uncertainties were evident in the modelling exercises. These related principally to a lack of available data for the transfer parameters employed in the field equations describing heat and moisture transfer. Experience indicated that it was likely that the uncertainties were more significant in the predictions of water transfer than in predictions of heat transfer.

During water uptake, the buffer and backfill materials were expected to swell and develop swelling pressures acting against the other components of the test system. The buffer was expected to exert high ultimate pressures of 10 MPa or more against the rock and the backfills. Pressures of hundreds of kPa were expected from the backfill materials. Interactions would result in deformations of decimetres at the buffer/backfill boundary. Moreover, the swelling pressures could result in clay being extruded into open rock fractures that intersected the excavations in which the experiment was to be carried out.

These hydro-mechanical interactions, because of their largely unknown effects on the hydraulic boundary conditions acting at the buffer/backfill/rock interfaces, further decreased confidence in an ability to predict numerically heater/buffer/backfill/rock performance and interactions with precision. The observations to be made in the BMT were needed to qualify and refine the conceptual models for performance and to provide an indication of the extent to which enhanced numerical prediction capabilities were required or, indeed, possible.

The work for the macropermeability experiment carried out under the SAC agreement, along with observations made in the large-diameter boreholes (Ø = 760 mm, 3 m deep) that were diamond-drilled for the BMT, provided information used to bound the initial hydraulic conditions around the BMT room. Excavating the room at right angles to the horizontal major principal stress resulted in an excavation disturbed zone with a radial hydraulic conductivity that was less than the value of 10^{-10} to 10^{-11} m/s estimated for the surrounding rock mass. Measurable water flows occurred principally in the fractures in the rock mass, although there was evidence of flow in permeable "fracture-free" rock. Only a small fraction of the observed fractures visibly carried water. A large number of the natural fractures were infilled with minerals or otherwise blocked to the transmission of water. The eastern wall and the ceiling of the room appeared likely to provide greater access of water to the backfills than the floor, the western wall and the end of the room. Natural water inflow into the emplacement boreholes varied between holes and was higher in holes 1, 2 and 5 than in holes 3, 4 and 6. Most of the water entered the open large-diameter drillholes through discrete fractures which, reflecting the excavation disturbance, were concentrated in the upper third of the hole and sub-parallel to the floor of the room. The water inflow conditions in the wetter holes made it impossible to infill the construction gap between the HCB buffer and the rock wall with powdered bentonite and, thereby, through practicality, conditioned the layout of the experiment.

A large number of instruments was installed in the buffer and backfills to monitor changes in temperature, total pressure, pore water pressure, moisture content and displacement during the progress of the tests. Water pressure and temperature were measured in the near-field rock mass. Hole 5 was selected for extra instrumentation: the swelling forces from buffer and backfills acting on the cap of the hole and the rock displacements arising from the combined effects of temperature changes and the swelling forces were measured.

Commercially available instrumentation was used wherever possible; much of this instrumentation required modification and calibration for the harsh environmental conditions of temperature, pressure and water salinity expected in the test. Special moisture sensors were developed to monitor transients in the clay masses.

The heaters used in the six emplacement holes were specifically designed to facilitate measurements of moisture content in the HCB at a single point in time at the end of the test. During this activity, precise surveys were used to determine the deformation of the buffer/backfill interface.

It is high testimony to the care taken in the design of the heaters and the special measures taken to protect the other instrumentation that none of the heaters failed during the four years over which the BMT was carried out, and that other instrumentation suffered little from malfunction.

The response of the buffer and backfills to heating and to water supplied through the bounding rock mass depended on the original hydraulic boundary conditions, the test configuration and the interactions between the clays and the rock mass. The temperature distribution, final moisture content distributions and swelling pressures developed by the HCB buffer material were largely controlled by the rate at which water was supplied at the rock/buffer interface. The buffer in wet holes 1, 2 and 5 became saturated within the test period; the buffer in dry holes 3, 4 and 6 showed increasing water content from the heater to the buffer/rock interface, with drying having occurred near the heater. Correspondingly, swelling pressures were higher and temperatures were generally lower in the wet holes than in the dry ones.

Under the force of the swelling pressure, HCB was extruded into fractures in the rock, thereby preventing these fractures from acting as local water sources. The buffer, which originally contained construction joints, took up water through a thin layer of sealed rock and sealed itself. In accordance with expectations, this self-sealing was more pronounced in wet holes than in dry ones. Eventually, when the buffer in dry holes becomes saturated, it is expected to self-seal just as effectively.

The results tended to confirm that an isothermal moisture transfer model could be applied to moisture transfer in the backfill materials in which imposed temperature gradients are small. The isothermal model did not account for moisture transfer that occurred in the HCB buffer. The temperature data, swelling pressure results, moisture redistribution data, results from tracer tests, and retrospective analyses all support the hypothesis that an evaporation/condensation cycle was established down the temperature gradient developed in the unsaturated buffer. The conduct of the BMT showed that moisture transfer under temperature gradients can be significant in dense bentonite materials under repository conditions and, depending on repository design, may need to be accommodated in models of the performance of the near field.

The heat conduction model used tended to overestimate the temperatures to be expected in saturated systems – the ultimate condition expected in a repository.

The mechanical performance of the backfill met all expectations by exhibiting more than adequate resistance to the uplift forces from the buffer. The magnitude of heave at the buffer/backfill interface was well predicted by a simple mathematical model. The effects of swelling pressures from the buffer on the rock mass appeared to be within the accuracy of the instruments and could not be measured. However, the floor of the emplacement room exhibited heave, which was principally attributable to increases in the temperature of the rock and in accordance with understanding. The effects of this movement on water flow in the near-field rock mass could not be established.

Pore water pressures in the rock mass within 1 m of the tunnel faces appear to have been controlled by an excavation disturbed zone (EDZ). Adding to the results from the SAC macropermeability experiment, the BMT results indicated that the hydraulic properties of the EDZ were anisotropic: hydraulic conductivity parallel with the tunnel axis appeared higher than radial hydraulic conductivity. The high axial conductivity fed groundwater to the top of the emplacement boreholes and to the bottom of the tunnel backfills. These data provided significant understanding of the near-field rock mass for the design of grouting experiments carried out during Phase 3 of the Stripa Project.

24.9.3 PHASE 2 – Borehole, Shaft and Tunnel Plugging

Siting a repository for heat-generating radioactive waste will require thorough investigation of the rock mass. Despite the significant advances made in geophysical investigation methods through the Stripa Project and other programmes, it remains clear that the site investigation will include penetration of the rock by investigation boreholes. Thus, most preliminary designs for repositories are based on the assumption that investigation boreholes, if left unfilled during repository closure, may act as preferential pathways for radionuclide migration and release. The same concern exists for shafts and tunnels used to develop and access the disposal levels of a repository. Phase 2 of the Stripa Project focussed on methodologies for sealing these possible pathways using highly compacted bentonite (HCB).

In the tunnel and shaft plugging experiments, highly compacted bentonite blocks, with densities and thus with properties similar to the buffer material tested in the BMT (Phase 1), were used. The tests were configured to determine the efficiency of swelling clay in limiting flow at the interfaces between bulkheads, backfills and the excavated rock surfaces. The layouts of the tunnel and shaft plugging tests are shown in Figures 24-20 and 24-21.

In both cases, two bulkheads were constructed within the excavations to form a test cell. The inner surfaces of the bulkhead were lined with HCB and the enclosed volume was filled with sand. The inner, sand-filled part of the test cell acted as a constraint to resist bentonite swelling and could be filled with water and pressurized. Only one tunnel plug test was carried out in which the hydraulic competence of the complete concrete bulkhead and HCB gasket was tested. Two shaft plug tests were conducted: the first tested the hydraulic properties of a concrete bulkhead alone; the second measured the hydraulic performance of HCB gaskets, confined within tied steel plates.

The hydraulic tests on the constructed cells lasted approximately 12 to 24 months. The tests involved increasing or decreasing the water pressure in the sand by steps, and measuring water flows into and out of the systems. The swelling properties and water uptake of the HCB were measured. The measured responses of the system and, specifically, the HCB were compared with predictions of performance derived using laboratory data, theoretical considerations and the experience obtained from the BMT (Phase 1).

The ease with which the HCB gaskets were placed showed that the concept of incorporating these materials into designs for plugs for large excavated openings was practical.

The hydraulic testing of the tunnel plug showed that after water uptake and swelling, the HCB effectively sealed interfaces between the bulkhead and the rock and minimised water flows at the HCB/concrete interfaces. The effective hydraulic conductivity of the combined concrete tunnel plug and HCB gasket can be estimated to be between 10^{-12} m/s and 10^{-11} m/s. This is of the same order of magnitude as the measured hydraulic conductivity of the undisturbed granite. Similar results were obtained from the shaft plugging tests, from which the effective hydraulic conductivity of the concrete plug and rock/concrete interface could be estimated to have been reduced significantly by the addition of the HCB gasket.

The measurements of water uptake by the HCB generally confirmed the applicability of isothermal diffusion models derived from laboratory tests and applied in Phase 1 for the BMT. An isothermal water diffusivity parameter value of $4 \cdot 10^{-10}$ m^2/s could be used reasonably to predict moisture profiles in the maturing (wetting) HCB. The water uptake measurements also showed that presumably, at an early stage of the tests, before swelling of the bentonite had occurred, water had flowed around the HCB gaskets. This pathway had subsequently been sealed.

Three borehole plugging/sealing tests were carried out. Each test was configured to allow for investigation of different aspects of borehole plugging with HCB.

In all of the tests HCB (MX-80) was introduced into smooth-walled, diamond-drilled boreholes and observations were made on the rate of maturation (water uptake and swelling) of the HCB, the resistance of the maturing bentonite to piping under hydraulic gradients, and the shear resistance between matured bentonite plugs and the borehole wall. The differences among the three tests lay in the orientation of the boreholes (one horizontal borehole and two vertical boreholes were sealed) and in the type of plug used (one vertical borehole was plugged using techniques which were virtually identical to those used in the horizontal borehole; a different sealing system was used for the second vertical borehole).

In all cases, the sealing system consisted of hollow cylinders of HCB encased in a copper exoskeleton, consisting of either perforated tubing or mesh. The exoskele-

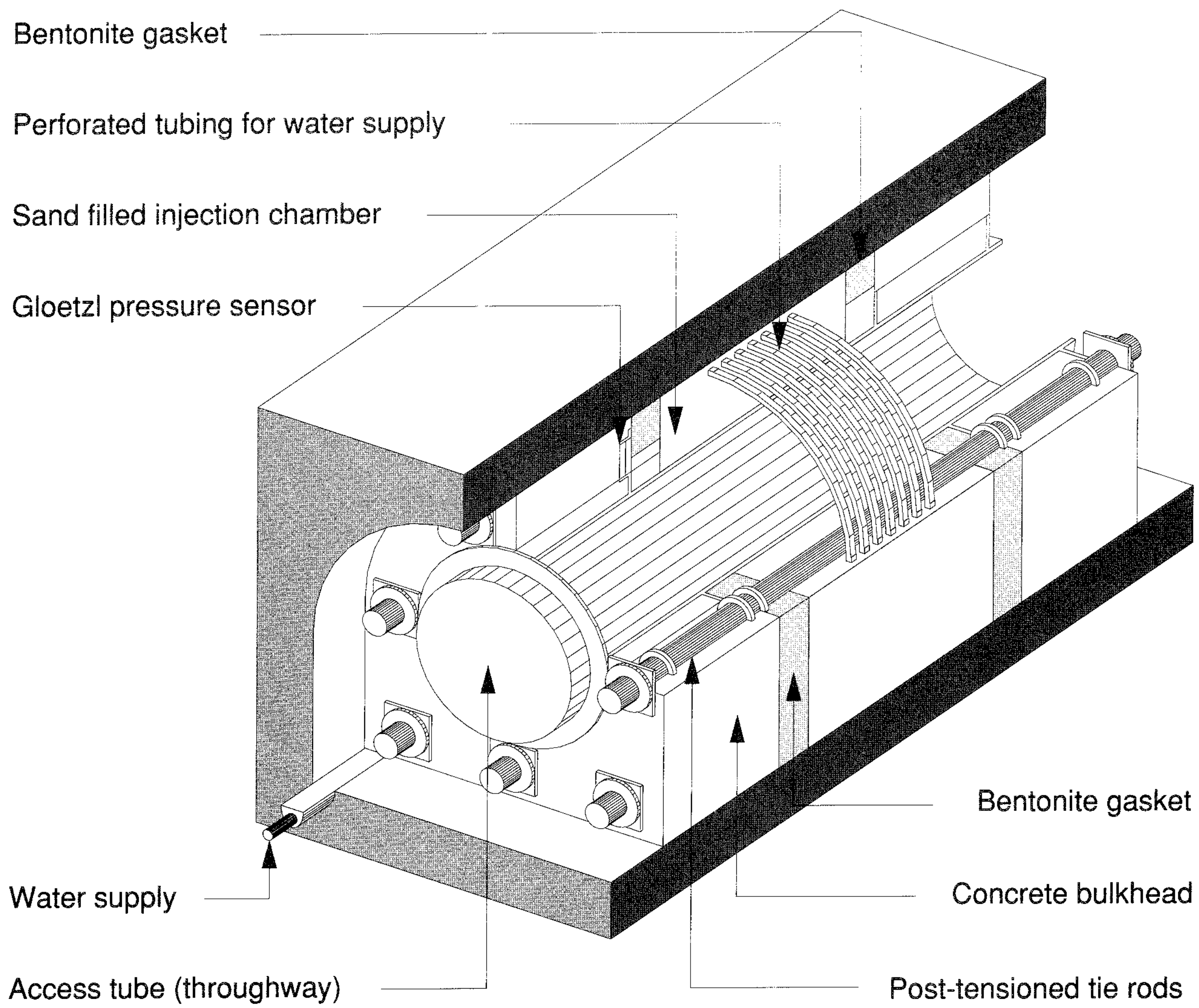


Figure 24-20. *The layout of the tunnel plug test (after /24-32/).*

HCB was placed as blocks on the inner faces of two tied concrete bulkheads. The inner steel tube (= 1.5 m) provided access to the inner bulkhead. Such a structure may be used to seal off fracture zones during repository operation and provide access for equipment and manpower.

ton provided needed rigidity to the system as it was inserted into the borehole. The perforations allowed access of water to the HCB, causing the material to swell and seal unfilled sections of the boreholes. This process is shown in Figure 24-22. The axial central hole in the HCB allowed access for instrumentation leads and hydraulic tubing.

The horizontal borehole plugging test was carried out in a 96.6-m-long, 56-mm-diameter borehole that was drilled as part of the SAC macropermeability experiment. The hole ran approximately parallel to and then continued 50 m beyond the end of the drift used for the BMT.

The vertical borehole plugging tests were carried out in two 14-m-long, 76-mm-diameter boreholes that were specially drilled between two vertically separated, parallel tunnels near the BMT area. The lower tunnel provided access to the 3-D migration experiment area.

The layout of the vertical borehole plugging tests is shown in Figure 24-23. The test arrangement included mechanisms that allowed the plugs to be subjected to high water pressure gradients so that the hydraulic properties of the sealed borehole could be examined. The resistance of the maturing bentonite to piping was specifically investigated by applying high hydraulic gradients along the length of the plug and measuring the resulting flows. Moreover, total and pore water pressure sensors were included at strategic locations along the length of the plugs to assist with the interpretation of the flow measurements and to confirm aspects of bentonite behaviour and properties. The horizontal plugging test included filters at selected locations to allow for the hydraulic tests to be carried out. No pressure sensors were used in the horizontal plug test. After the *in situ* hydraulic testing of the borehole plugs had been completed, the vertical borehole

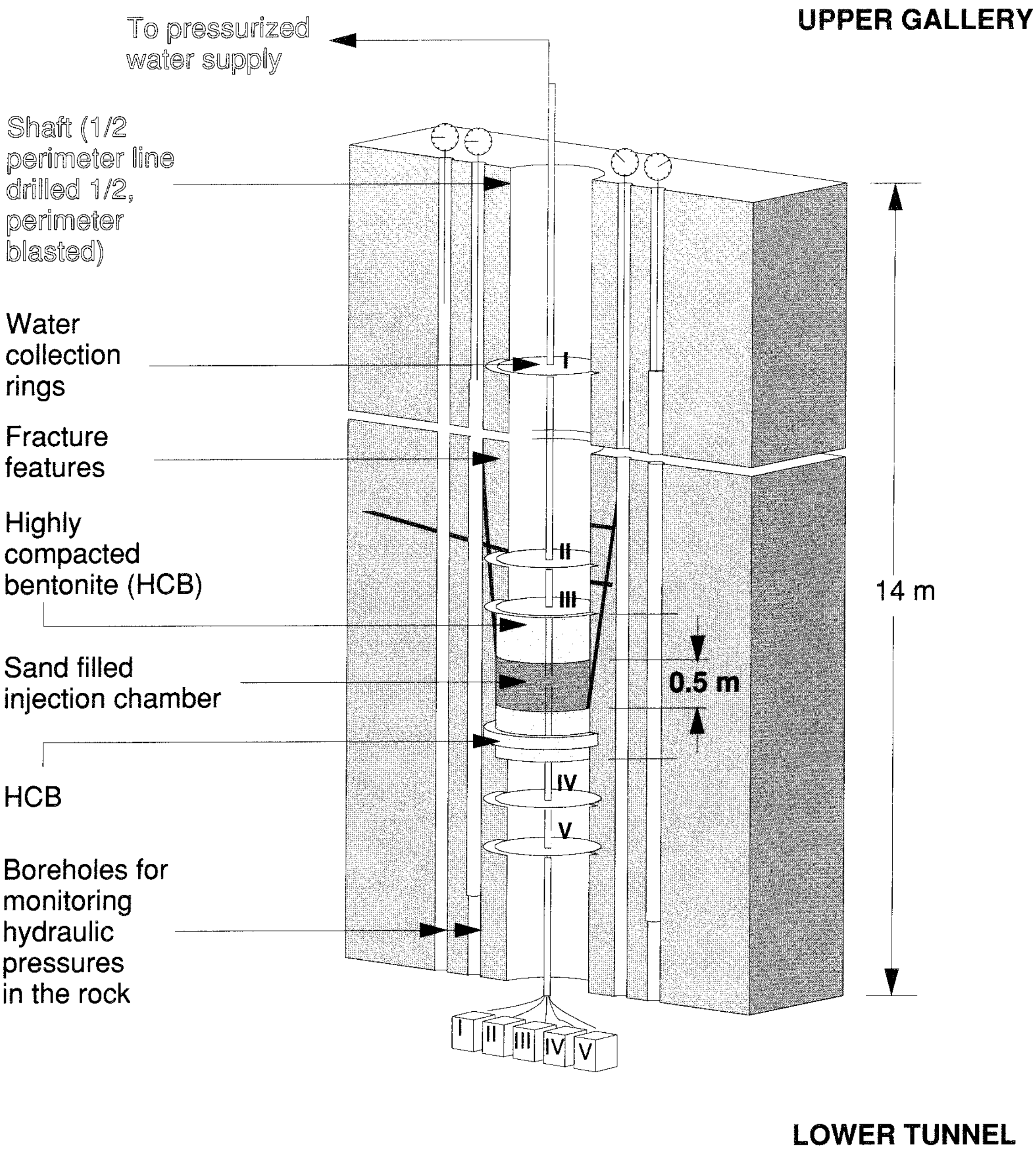


***Figure 24-21.** The layout of the shaft plug test (after /24-32/).*

Concrete bulkhead performance was first tested. A plug consisting of HCB confined by tied steel plates was then tested. The second test showed that the HCB plug had a lower hydraulic conductivity than the concrete alone. Water was lost from the test cell through the neighbouring rock.

plugs were extruded and measurements were made to evaluate the water uptake and swelling behaviour of the HCB. Owing to the configuration of the test, the horizontal borehole plug could not be extruded. At the end of the test a small volume of rock, through which the sealed borehole passed, was carefully excavated and the bentonite contained in it was examined.

The ease with which the borehole plugs were emplaced (it took only two hours to place the horizontal borehole plug) demonstrated the practicality of the design of the HCB sealing system for both horizontal and vertical borehole plugging operations.

Hydraulic testing of the horizontal plug as little as 14 days after plug installation proved that the partially matured bentonite plugs could sustain hydraulic gradients as high as approximately 450 (4.5 MPa along 1 m of plugged borehole) without piping. The effectiveness of the seal at the HCB/rock interface was well demonstrated by the instrumentation used in the vertical borehole plugging tests. The centrally located instruments registered vir-

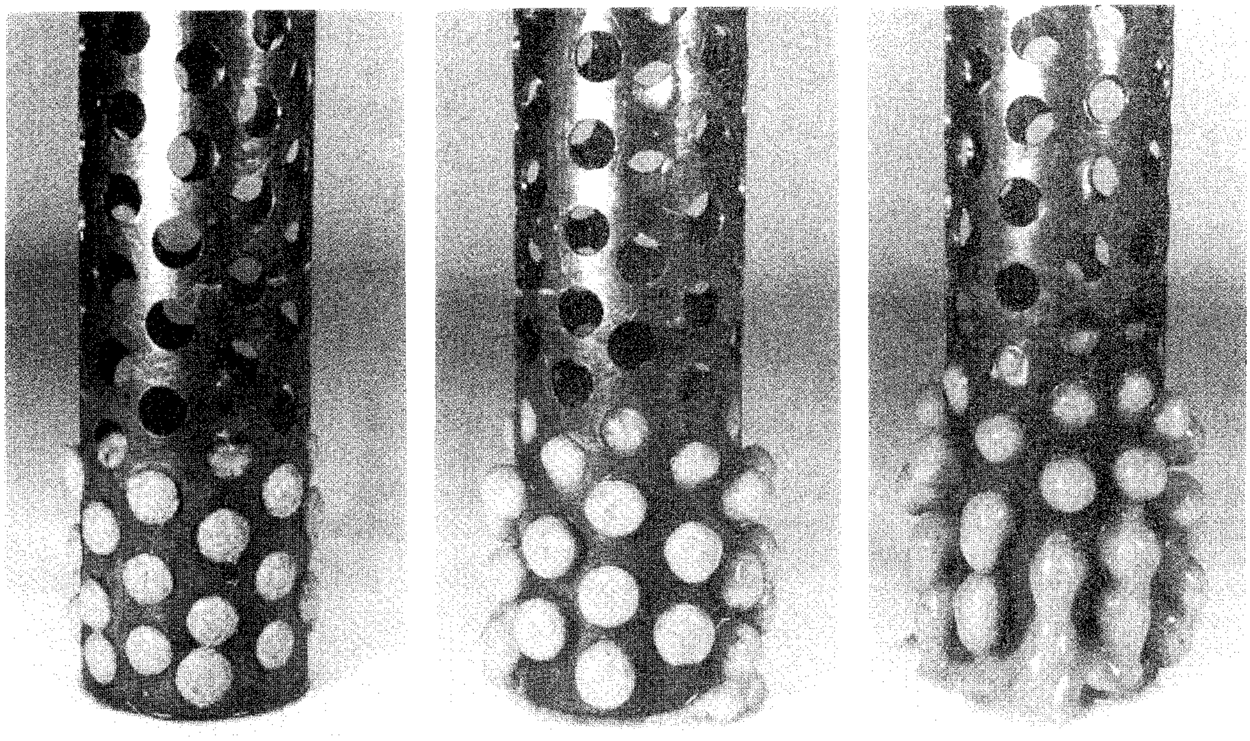


Figure 24-22. Evolution of an HCB plug.

(a) HCB contained within a perforated copper sleeve, and the effects of immersing the sealing system in water for (b) 8 hours and (c) 24 hours showing the extrusion of the bentonite through the perforations.

tually no transfer of pressure (total or pore water) from the upper water filled and pressurized chamber.

Examination of the recovered HCB plugs further indicated that the clay was virtually fully water-saturated and had expanded into the annular space between the plug and the borehole wall that was needed as a working clearance (1 to 4 mm) during plug emplacement operations. This outer film of bentonite was less dense than the inner core, indicating either incomplete maturation and consolidation of the bentonite or an ability of the bentonite to sustain significant stress gradients. The hydraulic conductivity of a borehole sealed with HCB was concluded to be between 10^{-12} and 10^{-13} m/s, which is as low as or lower than that of the intact granite at the experimental levels in the Stripa mine.

In the analyses of the BMT carried out in Phase 1, it was clear that the interactions between the rock and the bentonite strongly influenced the behaviour and function of the engineered barrier materials. Self-injection of the bentonite into fractures under the forces generated during maturation and swelling, although limited in extent and not detrimental to long-term system performance, changed the hydraulic boundary conditions acting on the clay. Similar phenomena were observed in the borehole, shaft and tunnel sealing experiments. Despite the presence of discrete fractures in the rock, the bentonite appeared to take up water uniformly and was observed to have self-injected into the wider rock fractures. Moreover, the swelling of the clay sealed construction joints in both the gaskets used in the tunnel and shaft plug tests and the borehole plugging systems. It is particularly significant that in the tunnel plugging experiment, after maturation of the bentonite and during the hydraulic conductivity testing, most of the leakage from the test cell occurred through a moderately fractured, hydraulically conductive planar pegmatitic feature, about 1 m thick, that had been intersected by the excavations. Responding to this observation, and recognizing that such variability can be expected in rock masses hosting real repositories, initial attempts were made to seal the transmissive rock by grouting with bentonite clay. This measure successfully reduced outflow from the test cell by more than 50%. This first stage of the development of grouting materials and methodologies was to become the main focus of the Phase 3 investigations into engineered barriers.

24.9.4 PHASE 3 – Grouts and Grouting

Phases 1 and 2 of the project demonstrated that excavations could be filled with materials capable of reducing the hydraulic conductivity of the excavated zones to values typical of undisturbed Stripa granite. Seals at the

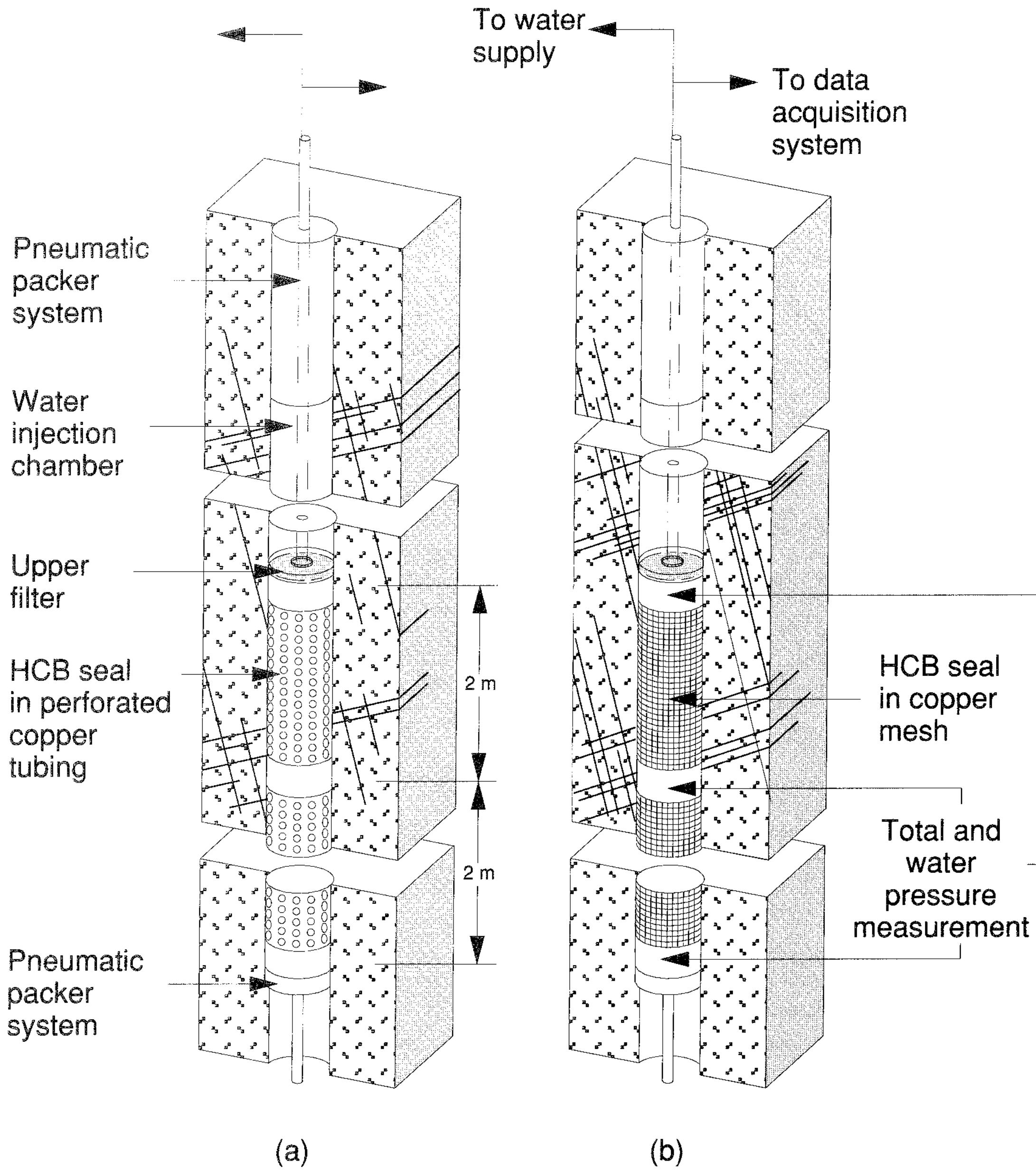


Figure 24-23. *The layout of the vertical borehole plugging tests.*

Two types of plug were tested /24-31/. One [(a) above], similar to the system used in the horizontal borehole plugging test, was encased in a perforated copper tube. The other [(b) above] was encased in a copper wire mesh. Both systems effectively sealed the boreholes. System (a) was more rigid and the investigators considered it easier to emplace than system (b).

interfaces between the excavated rock surfaces and the engineered barrier materials could be secured by means of swelling clays. Evidence from work carried out at Stripa through the SAC programme and Phases 1 and 2 of the international project, combined with observations being made elsewhere, led to a conceptual model, see Figure 24-24, that envisaged the possibility of the natural barrier to radionuclide release being bypassed by short circuits generated by the excavation disturbed zones surrounding the excavations. In this context, the SAC investigations and observations made during Phase 1 had shown that, relative to the natural properties of the host granite, the excavation disturbed zone (EDZ) may possess higher hydraulic conductivity along the axis of a tunnel and lower hydraulic conductivity radially away from the excavations. These phenomena were explained as related to the effects of rock-stress reorientation and concentration caused by the creation of the opening.

Similar phenomena observed during the SCV exercise have been accounted for, qualitatively, by similar reason-

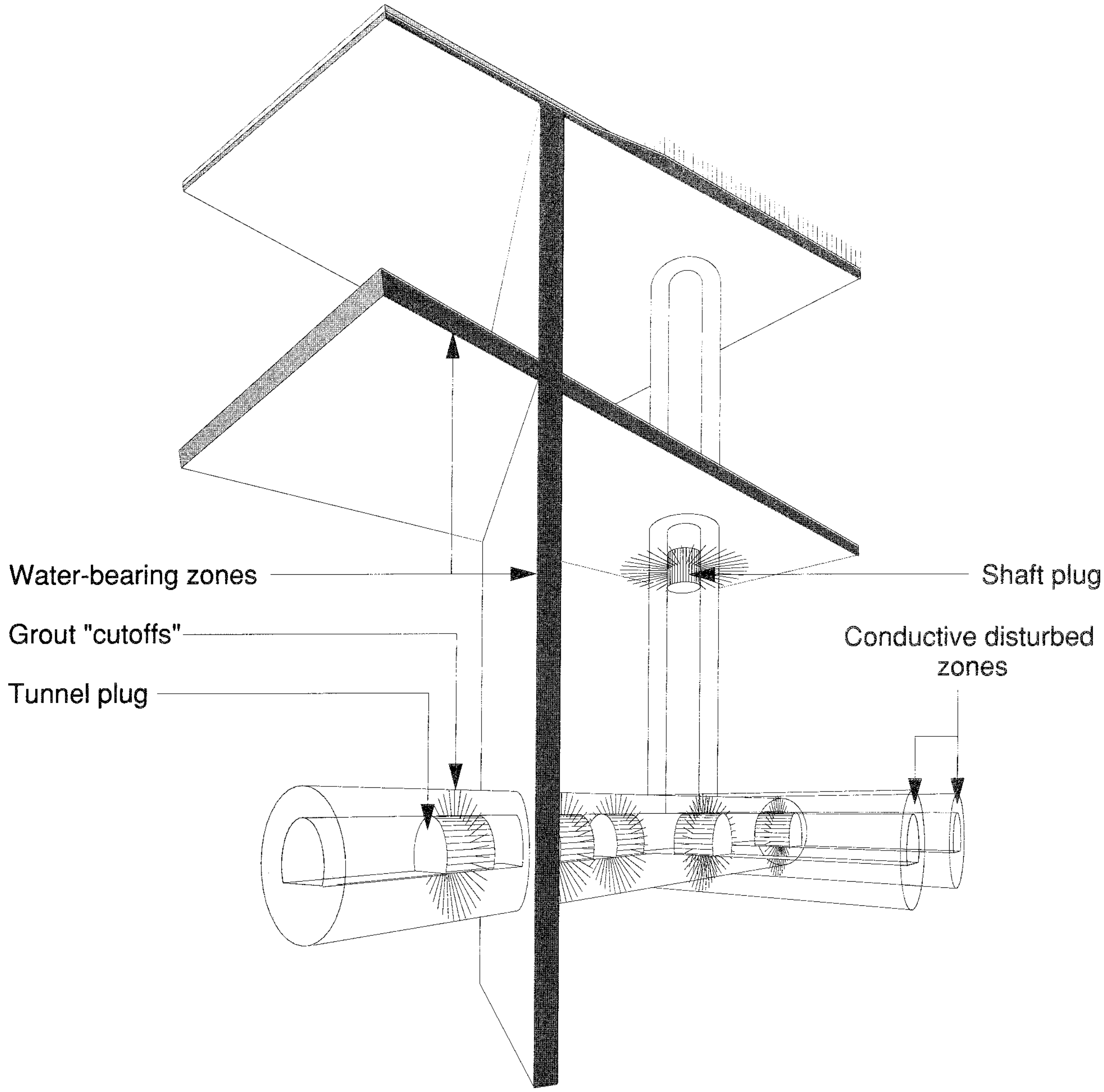


Figure 24-24. *A conceptual model of major pathways for water flow in a repository.*

This model was developed and discussed at two OECD/NEA Workshops /24-36, 24-37/. The EDZ can be seen to connect either directly to the surface or to the hydraulically active fracture zones that through Phases 1 and 2 had been identified as significant by investigations of the natural barrier.

ing. The following alternative mechanisms for the phenomena have also been suggested /24-24/:

- desaturation of the rock near the excavation surface, which was associated both with degassing of the groundwater as pressures decreased near the excavations and with the generation of gases by the combustion of explosives,
- high pressures produced during blasting, which drive water away from the excavations, and
- mineral precipitation and formation in fractures near the surface of the excavation phenomena associated with changes in hydrogeochemical conditions arising from the excavation.

Because the information available from Phases 1 and 2 on the properties of the EDZ was limited, further measurements and observations on its characteristics were needed. Moreover, with the still unproven, but conservative, assumption that the EDZ is detrimental to the isolating capacity of a repository for heat-generating radioactive waste, materials and methods to decrease the hydraulic conductivity of the zone(s) were sought. An extensive review of materials being investigated in the national programmes was undertaken /24-26/. This identified grouts based on clays, portland cements and bitumens as potentially suitable for sealing the fine fractures anticipated in the EDZ, and therefore appropriate for study in the Stripa programme. Reflecting national interests and the requirement to limit costs, the Phase 3 investigations were limited to grouting materials based on clay and portland cement.

The following activities were undertaken to address major areas of concern identified by the review:

- *Laboratory studies* of material properties, to permit the selection of grouting materials and the design of appropriate grouting equipment and procedures.
- *In situ testing* of selected grouting materials and methodologies in both the EDZ and zones of moderately fractured and hydraulically conductive rock.
- *Evaluation* of the longevity of both cement- and clay-based grouts.

With regard to the latter, reasonable arguments may be needed to demonstrate that the performance of engineered materials used in repository construction is assured for many millennia. The review highlighted a number of issues for both clay- and cement-based sealants that could be resolved through exposure in the Stripa programme.

24.9.4.1 *In situ* investigations

Within the context of the observational method as applied to geotechnical engineering, the *in situ* sealing investigations and experiments carried out in Phase 3 had the following primary objectives:

- to define the hydraulically significant characteristics of the EDZ and zones of moderately fractured, water-bearing rock, and
- to develop, apply and define the limitations of selected grouting materials and methodologies for application in repository design and construction.

Prior to conducting the *in situ* work, extensive laboratory testing, concentrating on the rheological properties of both cement- and clay-based grouting materials, was effected. The clay grouts were developed from the bentonite materials examined in Phases 1 and 2 of the project. The cement grouts were developed from materials that had been investigated through a joint USDOE/AECL project for the Underground Research Laboratory in Canada /24-27/.

At the outset of the investigations it was expected that both the clay- and what later became known as "high-performance" cement-based grouts would be thixotropic. This specific aspect of grout rheology was investigated and, assisted by theoretical and numerical modelling of the processes of grout injection, devices were designed and built to take advantage of this material property during injection. These tools allowed for vibration of the grouts during injection, causing the materials to liquefy and thereby permitted injection at minimum water contents. Both cement- and clay-based grouts are made fluid by suspending the colloidal solids in water. Decreasing the quantity of water required for liquidity increases the density of the injected grout and enhances performance. An extensive database on material properties was established that allowed for the appropriate selection of the materials to be used in the *in situ* studies.

The *in situ* experiments were carried out in the BMT room, where the EDZ was studied, and in the 3-D migration drift, where moderately fractured rock was investigated. The choice of these areas of the mine was favoured because both had been investigated previously in the project and thus their characteristic hydraulic features had been identified, with various levels of uncertainty. Studies in the BMT room focussed on the end of the room in and around heater holes 1 and 2, see Figure 24-19; the wet, eastern arm was studied in the 3-D migration drift. Developing studies of the natural barriers during the progress of the sealing experiments provided additional information on the major hydraulic features of the rock in the vicinity of the 3-D migration drift.

The principles of the tests planned for the BMT room are shown in Figures 24-25 and 24-26. Those for the tests in the 3-D migration drift are shown in Figure 24-27.

Using the specially developed injection devices, the test shown in Figure 24-25 was completed. Hydraulic testing prior to grout injection showed a clear increase in the hydraulic conductivity of the rock with increasing proximity to the surface of the excavation. Ungrouted, the rock had hydraulic conductivities in the range from $5 \cdot 10^{-10}$ to $5 \cdot 10^{-7}$ m/s. The higher values were reduced by grouting. Before heating, the hydraulic conductivity was reduced to less than 10^{-9} m/s. Heating caused the hydraulic conductivity of the grouted rock to increase to values that were intermediate between those of the untreated and the treated, but not heated, rock. Subsequent careful excavation of the grouted rock confirmed that the clay had penetrated the rock fractures, particularly those closer to the tunnel floor. This finding confirmed the hydraulic conductivity test results, which indicated that the grouting was more effective near the rock surface than distant from it. The results indicate that it is possible to grout discretely fractured rock with an initial equivalent hydraulic conductivity as low as $5 \cdot 10^{-8}$ m/s.

Employing an equivalent porous medium, finite element model, inflows into different parts of the test system shown in Figure 24-26 were calculated. The results of this flow model, which was symmetrical about the longitudinal axis of the tunnel, were checked against measured inflows at different stages of the test. This process showed that inflows into the test tunnel could be predicted with an accuracy of about 30%. The most consistent and accurate results were obtained using a model for the ungrouted rock that included a 0.5-m-thick blast-damaged zone with an

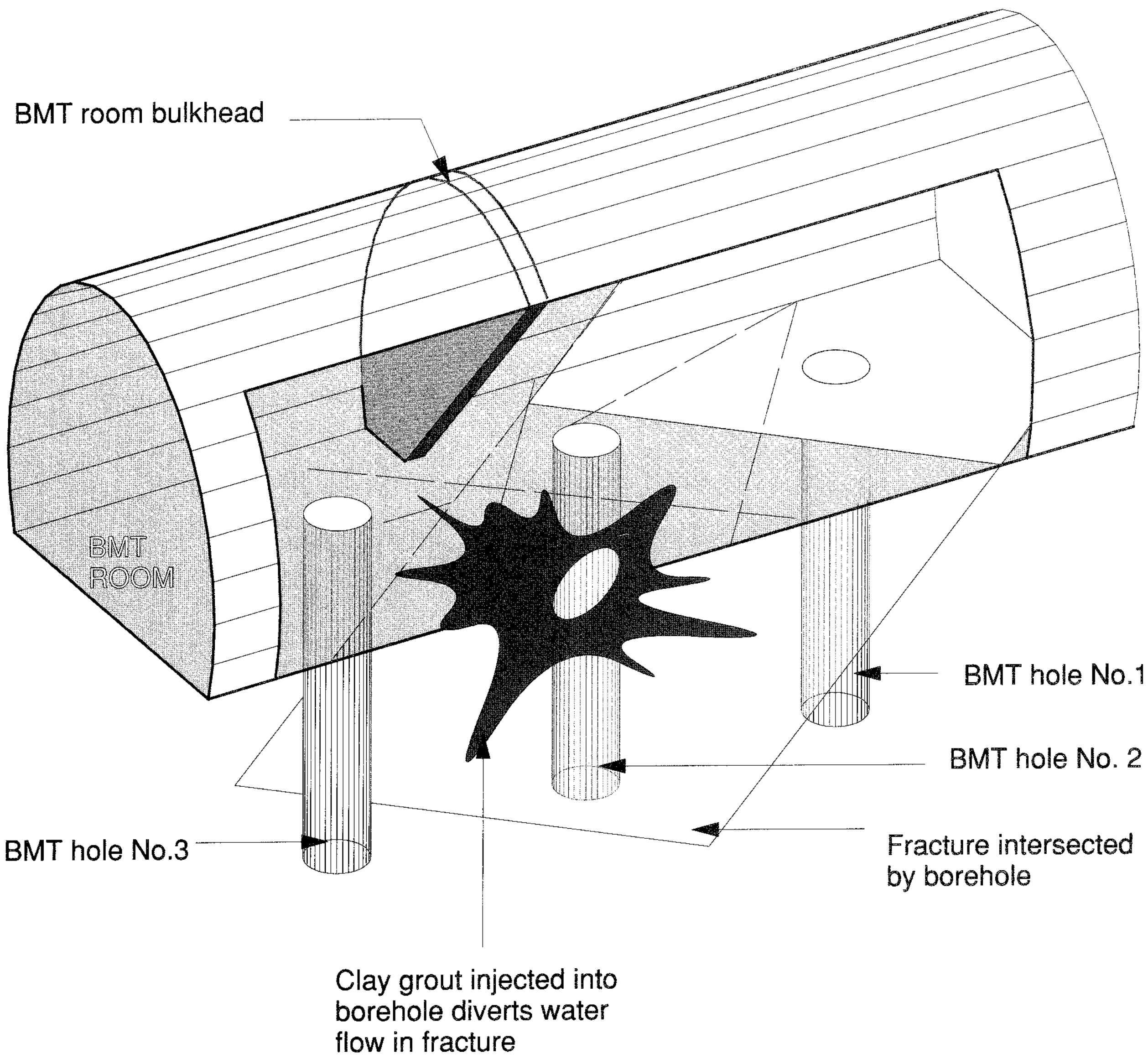


Figure 24-25. *Sealing around emplacement boreholes (after /24-38/).*

During the BMT (Phase 1) discrete water bearing fractures intersecting the emplacement boreholes had been identified. These were both natural and caused by excavation disturbance. A test was carried out to determine the extent to which these fractures could be sealed by bentonite grout. Possibly, such grouting procedures may be applied during repository construction to enhance the near-field seals.

isotropic hydraulic conductivity of $1.2 \cdot 10^{-8}$ m/s. The blast-damaged zone was surrounded by a 2-m-thick, anisotropic stress-disturbed zone with an axial hydraulic conductivity of $9 \cdot 10^{-10}$ m/s and a radial hydraulic conductivity of $2.3 \cdot 10^{-11}$ m/s. These latter figures were slightly modified locally to improve accuracy. The undisturbed rock was assigned isotropic hydraulic conductivities in the range from $3 \cdot 10^{-11}$ to $9 \cdot 10^{-11}$ m/s, depending on position and according to field measurements. The ability of the model to predict water flows and pressures at different stages of the test provided significant support to the empirical observations on the nature of the EDZ, and it was concluded that these observations were probably correct (i.e., the stress-disturbed zone in the BMT area of the Stripa granite is hydraulically anisotropic). Measurements indicated that the hydraulic properties of the stress disturbed zone were not symmetrical about the tunnel axis. This factor may account for the inaccuracy of some modelling results. Moreover it clearly demonstrates the presence of complicating natural factors that cannot be expected to be accounted for in performance models for a repository. It is not reasonable to expect that these local uncertainties can be evaluated during repository construction in the level of detail

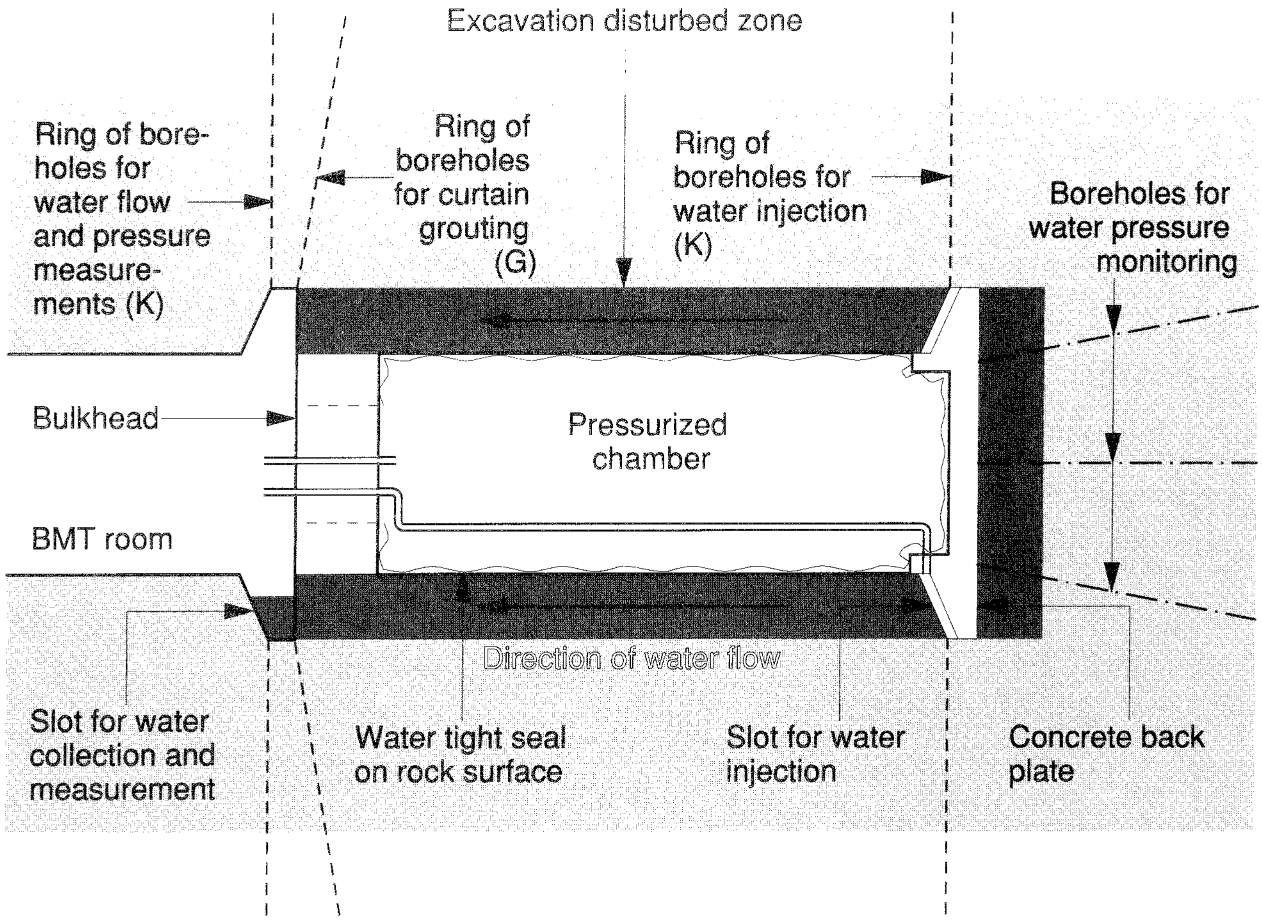


Figure 24-26. *Investigating the excavation disturbed zone (EDZ).*

The inner section of the BMT tunnel was to be sealed with a water tight lining and pressurized. Two rings of monitoring boreholes (K) were packed off in sections. This allowed for cross-hole hydraulic testing and, in combination with the end slots, for evaluation of the axial hydraulic conductivity of the EDZ, which was inferred from numerical simulations and testing to consist of a blast damaged zone and a zone influenced by stress relief. The effectiveness of cement grouts in sealing these two zones was examined. Testing of the grouting of the stress influenced zone (through the G holes) was not completed /24-39, 24-40/.

achieved in the Stripa studies. The possible existence of such local variations can be accommodated through uncertainty analyses and the use of appropriate safety factors in repository design.

Attempts were made to decrease the hydraulic conductivity of the shallow blast-damaged zone by injecting high-performance cement-based grout through shallow injection holes drilled at very close spacing into the walls of the excavation. Hydraulic testing indicated that this grouting procedure did not significantly change the hydraulic properties of the EDZ. Examination of samples of the grouted rock recovered by subsequent excavation revealed that although the grout had penetrated fissures as narrow as 40 %m, a continuous grout seal had not been generated. The grout readily penetrated new fractures generated by the blasting, but did not penetrate very well the natural fractures that contained infilling materials which obstructed grout flow and penetration. It is not certain that this result would have been obtained had it been possible to use high grout pressures. It was not possible to use high pressures when grouting the blast-damaged zone because of its close proximity to the free face of the excavation. Higher pressures than those used (800 kPa) could have significantly disturbed the rock. It was generally concluded that it was not possible to decrease the hydraulic conductivity of the blast damaged zone with the grouting procedures used. Alternative configurations for grouting or otherwise sealing off the EDZ remain to be examined.

Although simple in concept, as shown in Figure 24-27, the test to determine the ability of cement-based grouts to decrease hydraulic conductivity of moderately fractured, hydraulically conductive rock increased in complexity as more information became available on the rock properties through the progress of the investigations. The test proceeded on the initial assumption that the major conduit for

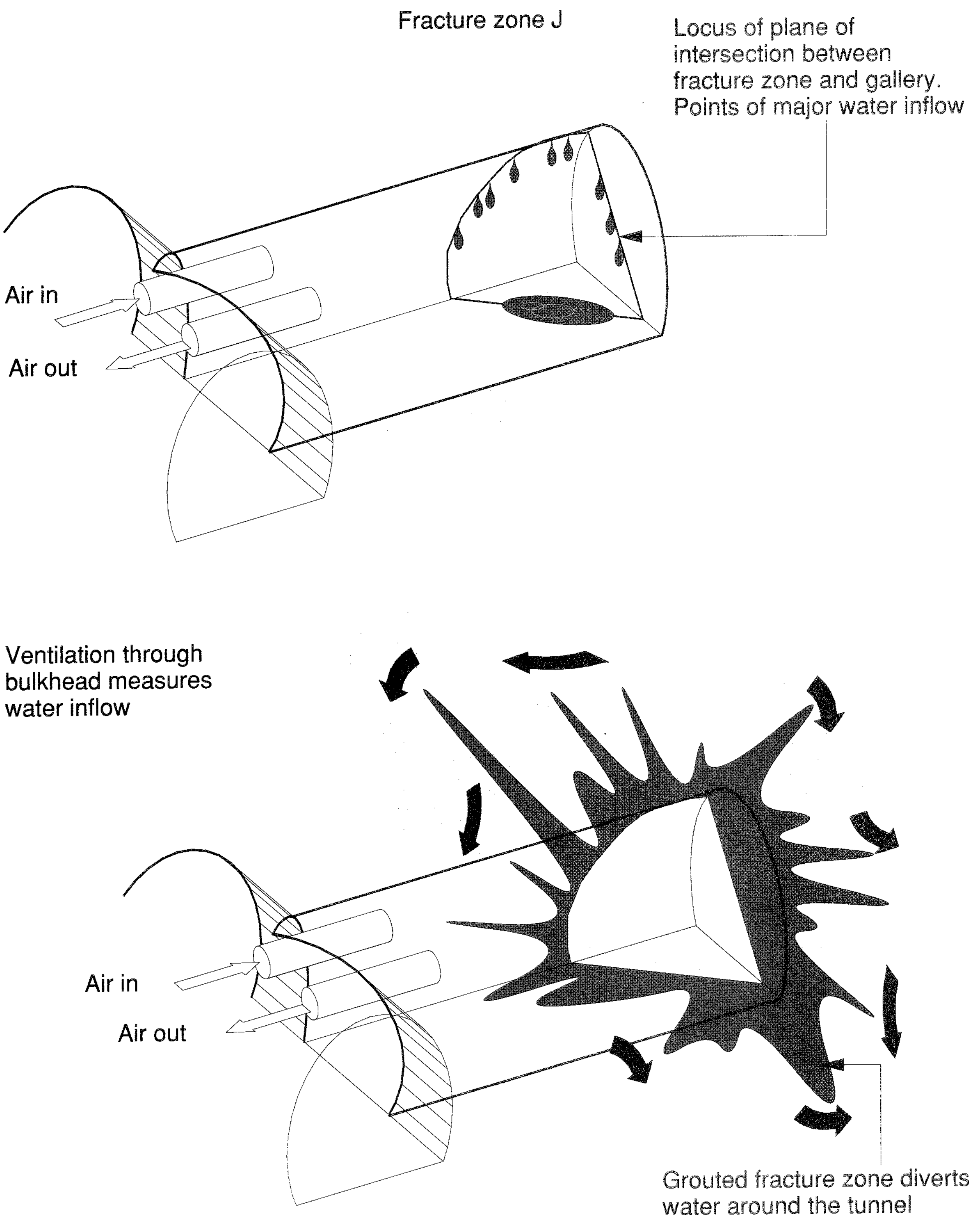


Figure 24-27. *Concept for testing the efficacy of grouting a zone of moderately fractured, water-bearing rock.*

A ventilation test was to be carried out to determine the water inflow into a rock chamber that was isolated from the rest of the mine. The major water bearing features were to be grouted (with cement grout) and the consequent changes in water inflow were to be determined by further ventilation testing /24-34/.

water inflow into the excavation was fracture zone J, which was identified by the geophysical and hydrogeological investigation methods used in the natural barrier studies for the SCV exercise. Along with extensive geological characterisation of the rock in the test area, measurements of water pressure and flows before and after an initial attempt to grout the wet rock intersected by the excavations revealed that while the J zone provided water regionally to the area of the test, the local hydrogeology was influenced by a subordinate fracture system. This subordinate system was only identified by the engineering activities associated with the grouting programme, which involved extensive drilling into the rock and probing of the hydraulic features. This experience defines the limitations of the large-scale hydrogeological characterisation methodologies developed through the Stripa Project, and clearly indicates the need for engineering designs for repository systems to be allowed to evolve with the increasing knowledge of the rock mass that will be gained through the engineering activities associated with repository construction.

The hydraulic structures were grouted with high-performance cement-based grout in two stages. As noted above, the first stage provided results that led to an increased understanding of the important hydraulic features in the rock mass; the second stage was effected in an attempt to alter the water flow paths. This was successfully accomplished. Although the total inflow into the test area was not significantly decreased, water inflow in the rock was diverted around the grouted rock volume. It is considered that, as shown by other experiments /24-27/, continued grouting could have successfully decreased the hydraulic conductivity of the accessible volume of the hydraulic structure from approximately 10^{-7} m/s to $5 \cdot 10^{-10}$ m/s or less. On the other hand, the practicality of grouting fracture zones in actual repository conditions remains to be established. Cores taken from the grouted rock contained cement-filled fractures. The morphology of the hardened cement was examined in detail and the knowledge gained of the material structure was used in the appraisals of cement grout longevity discussed in the following section.

24.9.4.2 Longevity of sealing materials

Longevity has been defined as the ability of a material to maintain its design performance through time, under the range of temperatures, pressures and geochemical conditions in the host environment. Implicitly, this definition accommodates the inevitable changes expected in sealing materials through time, with the requirement to understand the nature of possible changes and the consequences of these changes on sealing and, thus, total repository system performance.

Although some minor studies were undertaken as part of Phases 1 and 2 of the Stripa Project on aspects of the longevity of HCB seals, much of the available information on the important properties of both cements and clays rested within the national programmes for repository development. The review of sealing materials /24-26/ revealed a general consensus that HCB would behave adequately for millennia at temperatures below about 120°C. No such consensus existed for either the less dense clays to be used as grouting materials or for the portland cement-based materials. Studies were undertaken to reduce uncertainties about these materials.

For bentonite-based materials, attention focussed on providing detailed understanding of hydrothermal alteration of the mineral; particularly, reactions causing transformation of smectitic clays to hydrous clay-micas or causing cementation of the clay mass were studied /24-28/. Both of these processes could decrease the swelling capability of the bentonite and lead to a loss in long-term function. For the cement-based sealants, mechanisms causing dissolution of cement in groundwater and, thereby, an increase in the hydraulic conductivity of the grouted rock were investigated: specifically, the leaching and hydraulic conductivity properties of the materials were studied. To allow for the development and application of theoretical and numerical models of cement-grouts longevity, a database on the fundamental thermodynamic properties of cement grout phases was established and expanded. In addition to these basic studies, the mechanical stability of clay gels and unset cement pastes were investigated to determine their ability to resist erosion under the action of flowing groundwater. This information was needed to define the limiting groundwater flow conditions under which each of the materials could be applied.

To a greater or lesser extent, depending on applicability and the issues being investigated, the following three basic methodologies were applied to investigations of the longevity of both the clay- and cement-based materials:

1. investigation of natural analogues and archaeological evidence,
2. laboratory studies of basic material properties, and
3. application of the principles of thermodynamics.

Studies of the morphology of grouts recovered from the injections in the Stripa mine allowed the theoretical models to be adjusted to reflect actual, rather than supposed, material fabric.

The crystal structures of smectite minerals and hydrous clay-micas possess similar features. With particle sizes less than 2 %m, both mineral types consist of negatively charged lamellae of phyllosilicates comprised of covalently bonded silica (2 outer) and alumina (1 inner) layers. The lamellae in clay-mica are ionically bonded by K^+. In smectites the lamellae are separate and discrete. Studies of the products of reaction between bentonite, finely ground silica, groundwater and rock over a wide range of temperatures and pressures showed that when K^+ is present in the groundwater, the smectite in bentonite clay grouts will convert to hydrous mica. In contrast with HCB, for which the conversion reaction will take many tens of thousands of years, conversion in clay grouts will take a few thousands of years or less. At temperatures above about 130°C, the conversion reaction will be accompanied by the release of silica from the clay crystals and the precipitation of the silica within the clay fabric. This latter process results in cementation and embrittlement of the grout. The benefits gained by the use of flexible, swelling clays will be lost. The cementation reaction is enhanced by the presence of silica powder. Thus, while the studies showed that admixing siliceous rock flour with the clay improves the mechanical performance (piping resistance) of fresh clay-based grouts, siliceous additives to clay grouts may not be appropriate for applications aimed at sealing repositories for heat-generating radioactive waste, where temperature may exceed 75°C. The effects of the conversion of smectite to hydrous clay-mica on the performance of a grouted rock mass were assessed by reviewing the structure of the grouts uncovered during examination of the samples recovered from the *in situ* tests. It was estimated that after the grouted rock mass has undergone complete conversion from smectite to hydrous claymica, it will still possess an hydraulic conductivity that is significantly less than that of the ungrouted rock. The Stripa studies suggest that this process is controlled by the coefficient of Fickian diffusion of K^+ in the clay and the concentration of K^+ in the groundwater.

High-performance cement grouts differ from normal cements and concretes. First, high-performance cement is ground to an average particle size smaller than that of ordinary cements and allows for penetration of fine fissures. Second, it contains two performance-enhancing additives: superplasticizer (which increases the fluidity of the grout at low water/cement ratios) and pozzolanic material (finely divided particulate and amorphous silica, which, by reaction, decreases the amount of free $Ca(OH)_2$ in the hardened grout). Theoretically, both additives enhance the durability of the materials. A lower water/cement ratio results in denser, stronger, less permeable material; reduced $Ca(OH)_2$ decreases the solubility of the hydrated cement solids in groundwater. In contrast with bentonite and normal portland cements and concretes, high-performance grouts are relatively new (development began only in the 1980's) and, at the outset of Phase 3, information on basic properties such as hydraulic conductivity, microstructure and resistance to environmental forces of degradation was scant. Laboratory studies were undertaken to begin to provide these data. These studies were coupled with geochemical modelling of the changes that may occur within the fabric of the material and a numerical assessment of the effects of these changes on material performance. In the absence of other information, the modelling studies were based on contemporary understanding of normal cements and concretes, with adjustments for expectations of laboratory findings on the high-performance materials. Figure 24-28 shows the links between modelling and laboratory studies on the longevity of cement grouts.

Using the database available at the early stages of the investigations, the longevity model predicted that the grouts would endure between 100,000 and 1,000,000 years in a repository environment. The model was simplistic in that it did not account for the complex amorphous nature of some of the siliceous and aluminous phases found in cement, nor for the existence of unhydrated material and hydrated lime that was found to exist in high-performance grouts in the form of portlandite. Unhydrated materials were also found to exist in ancient cements. Later developments of the model were able to incorporate portlandite in the normative composition of the model for the hydrated cement phases.

The porosity-hydraulic conductivity relationship for conventional portland cements was found not to apply to high-performance materials, laboratory specimens of which were shown to be virtually impermeable at hydraulic gradients less than approximately 15,000. It is noted that with the mine working open, the maximum hydraulic gradient measured in the Stripa facility was approximately 2000; much lower gradients are anticipated in a sealed repository. This has significant bearing on the findings of the studies. Based on conventional wisdom, it was assumed for the longevity models that cements would degrade by water percolating through the grouts; as the water passed through, the cement solids would dissolve and the consequent porosity increase would enhance hydraulic conductivity. This dissolution model now appears to be overly simplistic /24-41/. Given the low conductivities achieved by high-performance cement grouts, substantial flow through the body of the cement does not appear likely. Rather, flow will probably be diverted to grout-rock interfacial gaps or into the surrounding rock. While flow will occur around the grout, diffusional processes will operate within the grout to alter the "mineralogy" and chemistry of the cement. This slow diffusional process virtually assures an approach to chemical equilibrium and means that void spaces represented by micro-cracks or other microporosity will be filled by the precipitation of secondary products (the chemical reaction of groundwater and cement yields secondary products that occupy more space than was occupied by the original solids).

The consequence of this new understanding is that during early repository history, cement grout performance will be dominated by surface-controlled mechanisms. Because these mechanisms are less efficient at mass removal than the processes assumed by the percolation/dissolution model, the calculated persistence times reported above are considered to be a lower bound on the longevity of intact cement grout. Several uncertainties rest with this judgment. Particularly, examination of the microstructure of the grouts injected in the Stripa mine revealed an inhomogeneous structure that was not present in laboratory-prepared and -tested materials. These inhomogeneities have not been found in in situ investigations carried out elsewhere, and the full implications of the Stripa finding need to be appraised after further *in situ* tests have established wether the result is specific to the Stripa site and to the injection technique applied in the Stripa field tests. In common with other areas of investigation, the Stripa studies into cement grout longevity have highlighted matters of practical concern that may need to be investigated

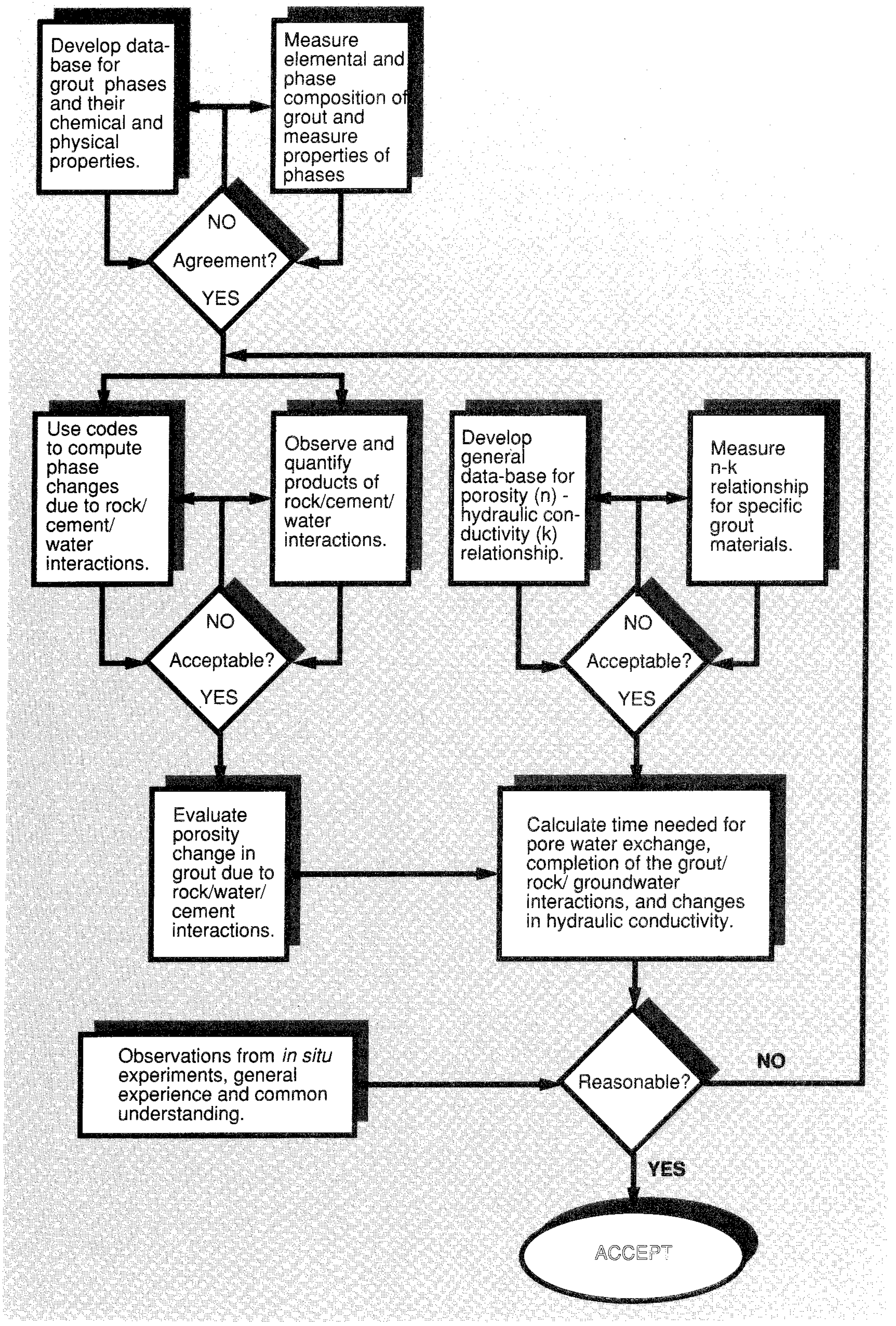


Figure 24-28. *Flow diagram for the studies on longevity of cement-based grouts.*

The laboratory studies showed that laboratory prepared grout specimens were virtually impermeable as a result of very small pore volumes and fine pore size distributions. Extremely low permeabilities are expected to enhance longevity by controlling surface processes such as leaching. It was shown that the naphthalene based superplasticizers are incorporated within the hydrated cement phases and are immobilized within the grout. The superplasticizers will not significantly contribute to the load of free organic matter present in a repository /24-35/.

in situ by national programmes, possibly during site specific repository construction and development.

24.10 SIGNIFICANT ACHIEVEMENTS

The "observational method" /24-25/ provides a basis for the investigation of the geotechnical engineering aspects of a repository for heat-generating radioactive waste. The method not only recognizes the need to observe important aspects of the rock mass and its response during repository construction, but also demands that engineering measures be available prior to construction to counter any reasonably foreseeable performance or safety concern. Some of these measures have been examined in the engineered barrier studies for the Stripa Project. The focus has been on the development of materials and methods for reducing the hydraulic conductivity of the engineered repository structure, including both the engineered openings and the accessible rock mass.

The application of both clay- and cement-based materials for sealing repositories has been considered, and limits have been defined for the longevity of these materials. It has been shown that, provided maximum temperatures do not exceed about 130°C, HCB and compacted bentonite-sand mixtures can be used to fill rooms and to seal boreholes, shafts and tunnels. These clay materials can be expected to perform effectively for tens, if not hundreds, of millennia. Less dense bentonite clay gels used for grouting rock will change to non-swelling clays within hundreds to several thousands of years. These non-swelling clays retain an ability to reduce the hydraulic conductivity of those elements of the repository system that have been treated. The longevity of cement-based materials cannot yet be as precisely prescribed as that of clay materials. However, it is reasonably certain that, applied as grouts in regions of a repository that are at temperatures below about 75°C, cement-based materials will persist in a repository environment, such as that exemplified by the Stripa granite, for very long periods of time, perhaps offering adequate performance for millions of years.

Practical methodologies for using highly compacted bentonite and compacted bentonite-aggregate mixtures to backfill engineered openings in the rock mass have been demonstrated. These methodologies result in sealed openings with hydraulic conductivities similar to those of the host Stripa granite (10^{-11} m/s), ensuring that Fickian diffusion would be the dominant mechanism of radionuclide flux within the engineered components of a repository in which these backfill materials were employed.

Some of the complex phenomena involved in the processes of heat and water transfer that will occur in bentonite clay barriers when these materials are used close to heat-generating waste containers have been observed and elucidated. It is clear that water transfer under temperature gradients can significantly influence the processes of resaturation of repositories built beneath the natural water table in the host rock. Phenomena similar to those observed during saturation of buffer and backfill materials and driven by temperature gradients may be relevant in unsaturated host media. The significance of these processes to the assessment of the performance of a repository and any need further to clarify the transfer processes must be judged on the basis of site-specific conditions and national regulatory requirements.

The water uptake and swelling properties of HCB can be effectively utilized to seal interfaces between rigid bulkheads, such as those made of concrete, and the engineered rock surfaces. Practical methods for placing such seals have been demonstrated, along with the ability of the clay systems to effectively limit water flow. Moreover, simple, practical designs for HCB systems for sealing investigation boreholes have been applied and demonstrated to be effective.

The extent to which the excavation geometry and processes influence the hydraulic characteristics of the Stripa granite have been examined. The study identified an excavation disturbed zone, composed of a blast-damaged zone immediately adjacent to the face of the excavations and a zone influenced by stress realignment and concentration surrounding the blast damaged zone. The disturbed zone did not possess the same hydraulic characteristics as the neighbouring host rock. Moreover, like the properties of the host rock, the properties of the disturbed zone were anisotropic and variable in the scale of tens of metres. Such variability can be expected in the host rocks for repository structures. It is not considered reasonable to expect that these local uncertainties can be evaluated during repository construction in the level of detail achieved in the Stripa studies. The uncertainties will have to be accommodated in any scheme for the approval of a repository design, which may have to be an iterative process to accommodate the observations made at the repository site.

Grouting techniques that were tried in the Stripa mine for treating the excavation damaged zone were not successful at decreasing its hydraulic conductivity. This experience may reflect the practical limits of grouting technology and, if such zones exist at a disposal site, their significance to repository performance will need to be assessed. It remains unclear whether or not decreasing the hydraulic conductivity of the blast-damaged zone improves repository system performance. Should decreased hydraulic conductivity of the excavation disturbed zone be needed, alternative methodologies to those examined through the Stripa investigations will have to be developed.

Major water flow paths in the accessible rock mass near the engineered openings were successfully grouted; groundwater was diverted around the grouted zones of moderately fractured rock. Thus, it may be possible locally to modify the groundwater flow paths to the benefit of total repository system performance or for construction expediency, if such action is deemed necessary. The grouting activities, like the studies of the interactions between clay backfills and the host rock, revealed significant details of the hydraulic features of the accessible rock mass. It became evident that the engineering activities associated with repository construction, operation and closure will provide information that can be used to understand and enhance the isolation capacity of a repository for heat-generating radioactive waste. This further reinforces the suggestion that detailed repository design should be an iterative process that accepts and responds to the increased knowledge of the natural barriers gained though the repository engineering activities.

24.11 CONCLUDING REMARKS

The principal objectives of the Stripa Project, when it began in 1980, focussed on the development of techniques to characterize crystalline rock masses that are potentially suitable for the construction of a geologic repository for radioactive waste, and the examination of engineering design considerations to enhance the long-term safety of the repository. Over the course of some thirteen years, research activities to satisfy these objectives were planned and carried out, principally in the granitic rock mass that contained the Stripa mine. Tools and techniques to quantify the hydrogeologic characteristics of a saturated, fractured granite were developed and demonstrated to be effective. Conceptual hydrogeologic models of the Stripa mine and its surroundings were developed and used with reasonable success for simulations of groundwater flow and transport. The characteristics of clays and cements that were considered to be important for the effective long-term sealing of fractured rock and man-made excavations were studied through numerical simulations, laboratory studies and experiments in the Stripa mine. The knowledge that emerged from these studies is impressive in extent and detail, and is destined to have application in the design of seals for geologic repositories and other underground containment structures and systems in many rock types. The field-scale demonstrations of shaft and tunnel sealing, borehole plugging, and the use of clay buffer materials in a saturated environment at elevated temperatures were both comprehensive and innovative.

These are the tangible accomplishments of the Stripa Project. The transfer of technology from the project to the member countries, and among the member countries, is in itself a significant accomplishment and must not be overlooked. Certain aspects of research programmes in the member countries are reflections of the research activities, discussions and debates that took place. In addition to some conclusions from the work of the Stripa Project, this chapter looks forward to the possible focus of future research to support the development of geologic repositories.

24.11.1 Conclusions

The main conclusions of the Stripa Project are listed below.

1. Underground access to representative depth may be necessary for assessing the safety of a repository in crystalline rock and is required for the detailed characterization work required to design a repository.

2. Due to variability and uncertainty in the geological environment, the site characterization process should be iterative and adaptive.

3. Significant advances have been made in geophysical and related techniques for identifying water-bearing fracture zones in granite. Such zones were identified up to 100 m away from boreholes.

4. Flow of water in undisturbed, fractured rock occurred in only a fraction of the discontinuities identified. Fracture flow is not uniform; zones of practically stagnant water may exist between the complicated flow channels. The influence of these flow features on radionuclide migration may be significant, but difficult to quantify.

5. Both discrete fracture flow and porous media models were able to describe groundwater flows into the excavations in the Stripa granite and changes in hydraulic conditions caused by the creation of the excavations.

6. Excavations at Stripa were surrounded by a disturbed zone with a radial permeability lower than that of the undisturbed rock. Observations to establish whether or not permeabilities parallel to the excavation were increased gave contradictory results. The significance of these permeability changes to repository design and performance assessment will depend on site specific and design specific features. If necessary, engineered control of the excavation affected zone seems possible.

7. Measurements of several naturally occurring radionuclides have established that groundwater deeper than 600 m in the Stripa area has been isolated from the atmosphere for at least several thousand years and possibly more than a hundred thousand years.

8. The geochemical studies, among other results, have demonstrated that a number of radionuclides are generated underground, in uranium-rich rocks, in quantities that exceed the cosmogenic contributions.

9. Engineered barriers can be integrated in repository designs with the functions of: a) improving waste isolation, b) backfilling excavations and c) restricting access of groundwater to the disposal zone. Aspects of barrier construction and performance of sealing systems have been successfully demonstrated. The need to apply engineered barriers to the excavation affected zone should be assessed on a case by case basis, depending on specific features of both candidate site and repository design.

10. The confidence in the longevity of certain clay- and cement-based engineered barrier materials has been increased greatly by the Stripa Project.

11. International cooperation, in relation to radioactive waste disposal, has been effective and rewarding. The recommendation that interested countries and international organizations give serious consideration to other cooperative activities similar to the Stripa Project appears to be a logical consequence and a worthwhile objective.

24.11.2 Final remarks

As noted in the Introduction, demonstration that radionuclide transport from a repository to the biosphere will occur very slowly, if at all, is a central issue in ensuring the safety of geological isolation of radioactive waste. For repositories in fractured rock masses, as are contemplated in many countries, such demonstration requires a good understanding of the scientific principles governing radionuclide transport in fractures and the availability of techniques to characterize the fracture networks in specific rock masses.

The Stripa Project was an imaginative and pioneering international venture that has made major contributions both to scientific understanding of groundwater flow and solute transport in fractured rock and the development of practical tools and procedures for characterizing and controlling groundwater flow. Scientists in all of the nine participating countries have gained much experience that can now be used to good effect in developing national programmes. Developments and progress at all stages of the thirteen years of investigations have been thoroughly documented in the comprehensive set of technical reports. These provide an invaluable archive for researchers worldwide who will be stimulated to investigate the many issues raised by the Stripa studies. Overall, the Stripa Project is a unique example of a well managed, cost-effective international collaborative project; as such it can serve as a basis upon which to build future projects, both national and international, in the pursuit of safe disposal of radioactive waste.

REFERENCES

Chapter 24

24-1 Gale J, MacLeod R, Welhan J, Cole C, Vail L
1987
Hydrogeological characterization of the Stripa site.
Stripa Project Technical Report TR 87-15, SKB, Stockholm

24-2 Davis S N, Nordstrom K (Eds.)
1992
Geochemical studies at Stripa – Final report.
Stripa Project Technical Report TR 92-19, SKB, Stockholm

24-3 Stripa
1986
Executive Summary of Phase 1.
Stripa Project Technical Report TR 86-04, SKB, Stockholm

24-4 Stripa
1989
Executive Summary of Phase 2.
Stripa Project Technical Report TR 89-01, SKB, Stockholm

24-5 Olsson O J, Black J, Cosma C, Pihl J
1987a
Crosshole investigations – Final report.
Stripa Project Technical Report TR 87-16, SKB, Stockholm

24-6 Olsson O J, Falk L, Forslund O, Lundmark L, Sandberg E
1987b
Crosshole investigations – Results from borehole radar investigations.
Stripa Project Technical Report TR 87-11, SKB, Stockholm

24-7 Cosma C
1987
Crosshole investigations – Short and medium range seismic tomography.
Stripa Project Technical Report TR 87-08, SKB, Stockholm

24-8 Pihl J, Hammarström M, Ivansson S, Morén P
1986
Crosshole investigations – Results from seismic borehole tomography.
Stripa Project Technical Report TR 87-06, SKB, Stockholm

24-9 Black J, Holmes D, Brightman M
1987
Crosshole investigations – Hydrogeological results and interpretations.
Stripa Project Technical Report TR 87-18, SKB, Stockholm

24-10 Birgersson B, Widen H, Ågren T, Neretnieks I
1992
Tracer migration experiments in the Stripa mine.
Stripa Project Technical Report TR 92-25, SKB, Stockholm

24-11 Olsson O J (Ed.)
1992
Site characterization and validation project – Final report.
Stripa Project Technical Report TR 92-22, SKB, Stockholm

24-12 Herbert A, Gale J, Lanyon G, MacLeod R
1991
Modelling for the Stripa site characterization and validation drift inflow: prediction of flow through fractured rock.
Stripa Project Technical Report TR 91-35, SKB, Stockholm

24-13 Herbert A, Lanyon G
1992
Modelling tracer transport in fractured rock at Stripa.
Stripa Project Technical Report TR 92-01, SKB, Stockholm

24-14 Dershowitz W S, Wallmann P, Geier J E, Lee G
1991a
Preliminary – discrete fracture network modelling of tracer migration experiment at the SCV site.
Stripa Project Technical Report TR 91-23, SKB, Stockholm

24-15 Dershowitz W S, Wallmann P, Kindred S
1991b
Discrete fracture modelling for the Stripa site characterization and Validation Drift inflow predictions.
Stripa Project Technical Report TR 91-16, SKB, Stockholm

24-16 Dershowitz W, Wallmann P
1992
Discrete fracture modelling for the Stripa tracer validation experiment predictions.
Stripa Project Technical Report TR 92-15, SKB, Stockholm

24-17 Long J, Karasaki K
1992
Simulation of tracer transport for the site characterization and validation site in the Stripa mine.
Stripa Project Technical Report TR 92-06, SKB, Stockholm

24-18 Long J, Mauldon A, Nelson K, Martel S, Fuller P, Karasaki K
1992
Prediction of flow and drawdown for the site characterization and validation site in the Stripa mine.
Stripa Project Technical Report TR 92-05, SKB, Stockholm

24-19 MacLeod R, Gale J, Bursey G
1992
Site characterization and validation – Porous media modelling of validation tracer experiments.
Stripa Project Technical Report TR 92-10, SKB, Stockholm

24-20 Gnirk P
1992
The process for validation of the groundwater flow and transport models in the SCV program.
Stripa Project Technical Report TR 92-28, SKB, Stockholm

24-21 Hodgkinson D
1991
A comparison of predictions and measurements for the Stripa simulated drift experiment.
Stripa Project Technical Report TR 91-10, SKB, Stockholm

24-22 Hodgkinson D, Cooper N
1992a
A comparison of measurements and calculations for the Stripa Validation Drift inflow experiment.
Stripa Project Technical Report TR 92-07, SKB, Stockholm

24-23 Hodgkinson D, Cooper N
1992b
A comparison of measurements and calculations for the Stripa tracer experiments.
Stripa Project Technical Report TR 92-20, SKB, Stockholm

24-24 Hodgkinson D
1992
A compilation of minutes for the Stripa Task Force on Fracture Flow Modelling.
Stripa Project Technical Report TR 92-09, SKB, Stockholm

24-25 Peck R B
1969
Advantages and limitations of the observational method in applied soil mechanics. Ninth Rankine lecture.
Géotechnique 19:171-187

24-26 Coons W E (Ed.)
1987
State-of-the-art report on potentially useful materials for sealing nuclear waste repositories.
Stripa Project Technical Report TR 87-12, SKB, Stockholm

24-27 Gray M N, Keil L D
1989
Field trials of superplasticized grout at AECL's underground research laboratory.
3rd International Conference on Superplasticizers and Other Additives in Concrete.
CANMET/ACI, Ottawa, Ontario.

24-28 Pusch R, Karnland O, Hökmark H, Sandén T, Börgesson L
1991b
Final report of the rock sealing project – Sealing properties and longevity of smectitic clay grouts.
Stripa Project Technical Report TR 91-30, SKB, Stockholm

24-29 Pusch R, Börgesson L, Ramqvist G
1985a
Final report on the Buffer Mass Test – Volume II: test results.
Stripa Project Technical Report TR 85-12, SKB, Stockholm

24-30 Pusch R, Nilsson J, Ramqvist G
1985b
Final report on the Buffer Mass Test – Volume I: scope, preparative field work and test arrangement.
Stripa Project Technical Report TR 85-11, SKB, Stockholm

24-31 Pusch R, Börgesson L, Ramqvist G
1987a
Final report of the borehole, shaft and tunnel sealing test – Volume I: borehole plugging.
Stripa Project Technical Report TR 87-01, SKB, Stockholm

24-32 Pusch R, Börgesson L, Ramqvist G
1987b
Final report of the borehole, shaft and tunnel sealing test – Volume II: shaft plugging.
Stripa Project Technical Report TR 87-02, SKB, Stockholm

24-33 Pusch R, Börgesson L, Ramqvist G
1987c
Final report of the borehole, shaft and tunnel sealing test – Volume III: tunnel plugging.
Stripa Project Technical Report TR 87-03, SKB, Stockholm

24-34 Pusch R, Börgesson L, Karnland O, Hökmark H
1991a
Final report on test 4 – Sealing of natural fine-fracture zone.
Stripa Project Technical Report TR 91-26, SKB, Stockholm

24-35 Onofrei M, Gray M N, Roe L
1991
Superplasticizer function and sorption in high-performance cement based grouts.
Stripa Project Technical Report TR 91-21, SKB, Stockholm

24-36 OECD/NEA
1989a
Proceedings of a workshop on "excavation response in geological repositories for radioactive waste". Winnipeg, 26-28 April, 1988.
Nuclear Energy Agency of OECD, Paris

24-37 OECD/NEA
1989b
Proceedings of a workshop on "sealing of radioactive waste repositories". Braunschweig, 22-25 May, 1989.
Nuclear Energy Agency of OECD, Paris

24-38 Börgesson L, Pusch R, Fredrikson A, Hökmark H, Karnland O, Sandén T
1991
Final report of the rock sealing project – Sealing of the near field rock around deposition holes by use of bentonite grouts.
Stripa Project Technical Report TR 91-43, SKB, Stockholm

24-39 Börgesson L, Pusch R, Fredrikson A, Hökmark H, Karnland O, Sandén T
1992a
Final report of the rock sealing project – Identification of zones disturbed by blasting and stress release.
Stripa Project Technical Report TR 92-08, SKB, Stockholm

24-40 Börgesson L, Pusch R, Fredrikson A, Hökmark H, Karnland O, Sandén T
1992b
Final report of the rock sealing project – Sealing of zones disturbed by blasting and stress release.
Stripa Project Technical Report TR 92-21, SKB, Stockholm

24-41 Onofrei M, Pusch R, Gray M N, Börgesson L, Karnland O, Shenton B, Walker B
1992
Final report of the rock sealing project – Sealing properties of cement based grout materials.
Stripa Project Technical Report TR 92-28, SKB, Stockholm

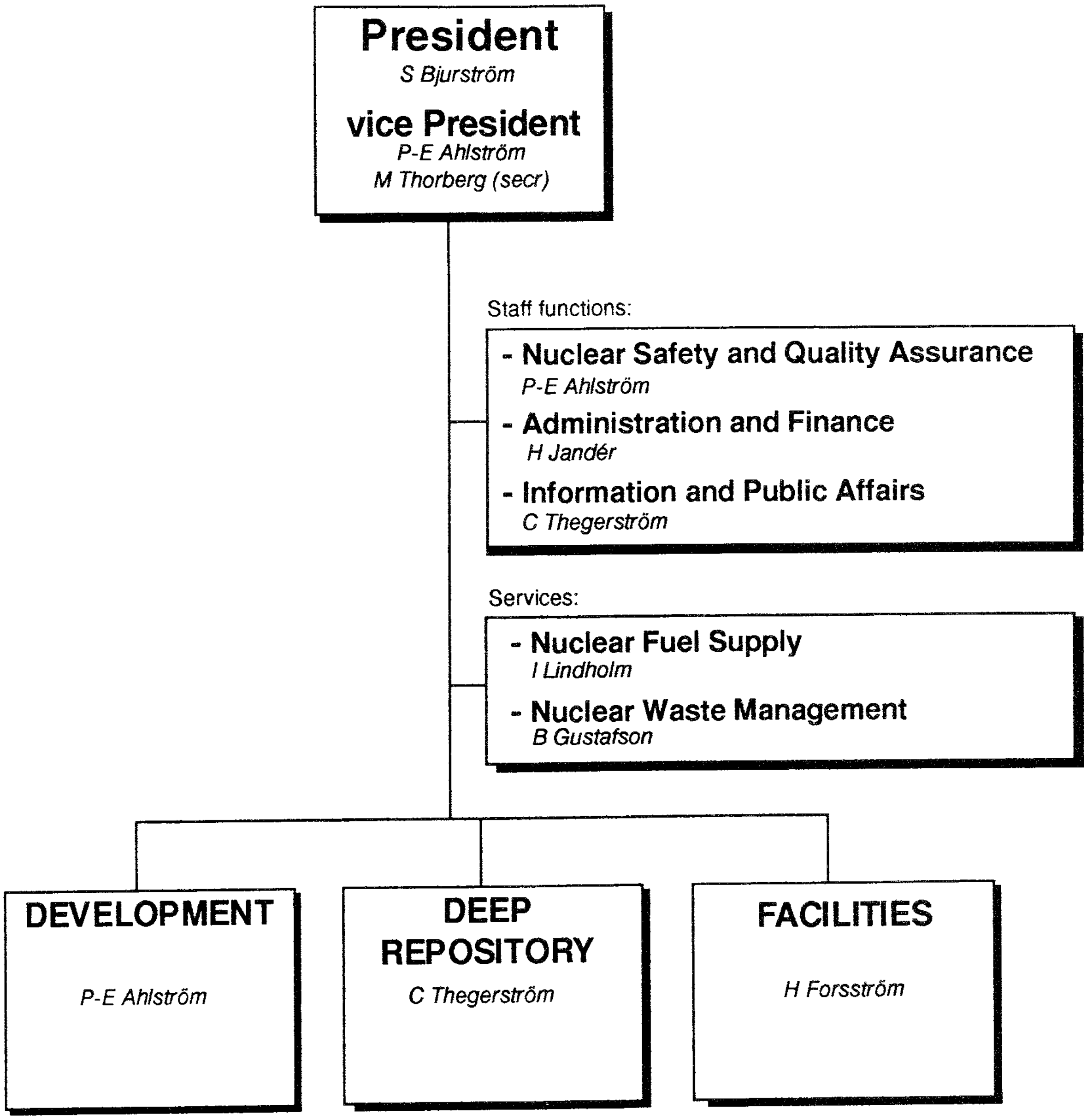
SKB ANNUAL REPORT 1992

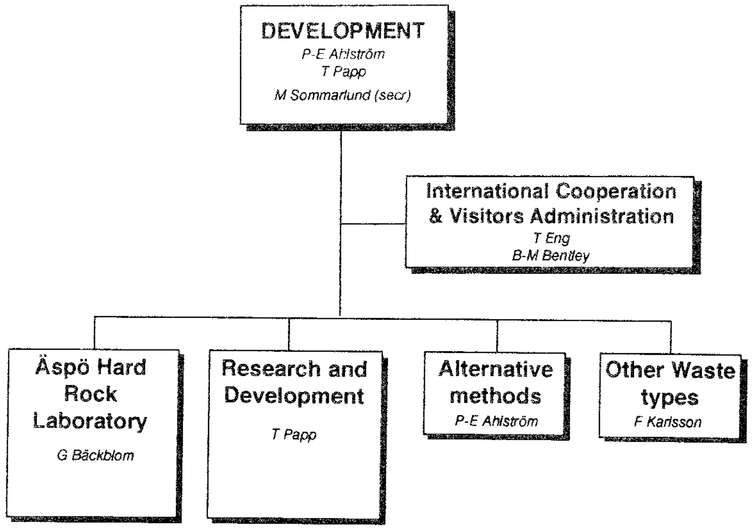
Part IV

Appendices

CONTENTS OF PART IV

Organization charts for SKB and its divisions May 1993

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International Cooperation & Visitors Administration
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Äspö Hard Rock Laboratory
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Research and Development
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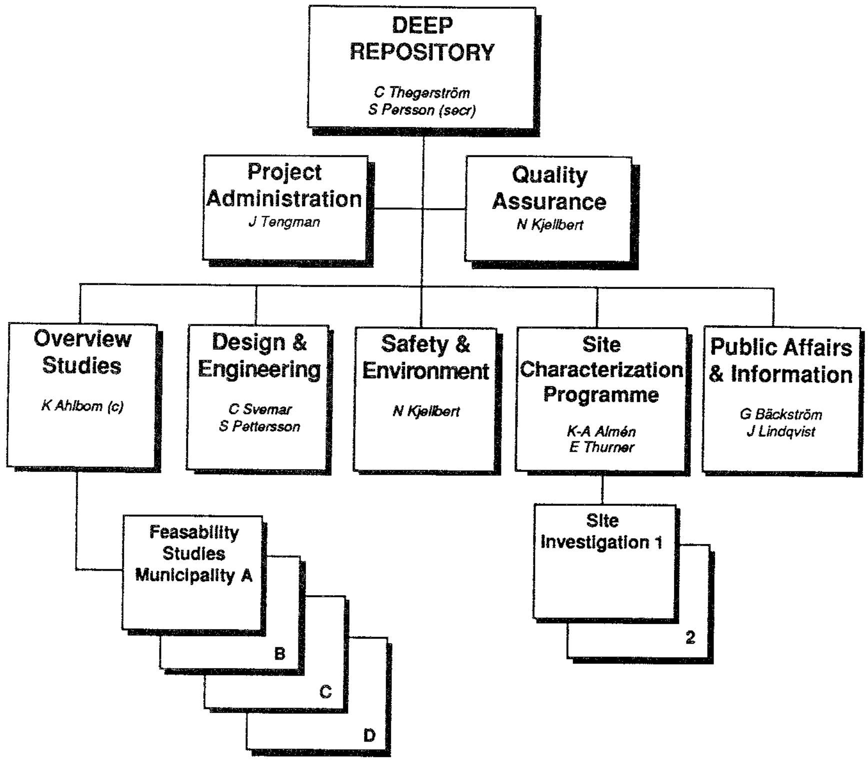
Alternative methods
P-E Ahlström
Other Waste types
F Karlsson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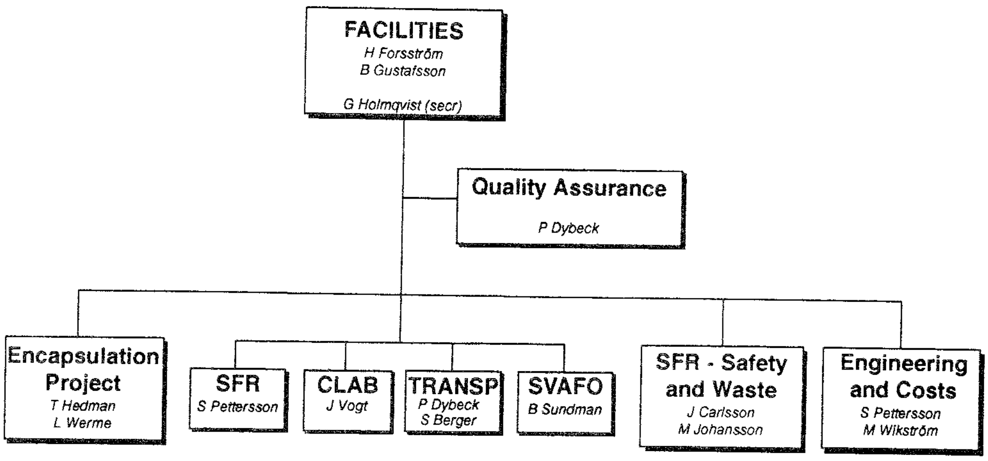
C Thegerström
S Persson (secr)
Project Administration
J Tengman
Quality Assurance
N Kjellbert
Overview Studies
K Ahlbom (c)
Design & Engineering
C Svemar
S Pettersson
Safety & Environment
N Kjellbert
Site Characterization Programme
K-A Almén
E Thurner
Public Affairs & Information
G Bäckström
J Lindqvist
Feasability Studies Municipality A
B
C
D
Site Investigation 1
2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H Forsström
B Gustafsson
G Holmqvist (secr)
Quality Assurance
P Dybeck
Encapsulation Project
T Hedman
L Werme
SFR
S Pettersson
CLAB
J Vogt
TRANSP
P Dybeck
S Berger
SVAFO
B Sundman
SFR - Safety and Waste
J Carlsson
M Johansson
Engineering and Costs
S Pettersson
M Wikström

LECTURES AND PUBLICATIONS 1992

Geological research for selecting and evaluating high-level waste repository sites in Sweden
Ahlbom, K; Ericsson, L O
In: Papers presented at the Rock Mechanics Meeting in Stockholm, March 19, 1992, pp. 93-108

Swedish high-level radioactive waste management issues
Ahlström, P-E
In: High-level Radioactive Waste Management, Proc of the Third International Conference, Las Vegas, Nevada, April 12-16, 1992, Vol. 1, pp. 33-40

Waste Management Research – The past, the present and the future
Ahlström, P-E
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Stripaprojektet – Bakgrund, organisation (The Stripa Project – Background, organization)
Ahlström, P-E
In: Papers presented at the Rock Mechanics Meeting in Stockholm, March 19, 1992, pp. 85-91

Geoscientific borehole investigations for nuclear waste disposal in Sweden
Almén, K-E
Poster at the 6th International Symposium on the Observation of the Continental Crust through Drilling, Paris, 7-10 April, 1992

Diffusion of radionuclides in sediments – in situ studies
Andersson, K; Evans, S; Albinsson, Y
Radiochimica Acta 58/59(1992), 321

Electron cycling in a granitic groundwater: Characterization of field site
Banwart, S
Invited lecture for the Workshop "Chemistry on the surfaces of particles", L. Lövgren organizer, University of Umeå, 1992

Geotechnical predictions of the excavation disturbed zone at Stripa
Barton, N; Makurat, A; Monsen, K; Vik, G; Tunbridge, L
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

In-situ tracer experiments in single fractures, fracture zones and averagely fractured rock
Birgersson, L; Neretnieks, I; Widén, H; Ågren, T
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

The Stripa Project in a Swedish waste management perspective
Bjurström, S
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Innovations in the characterization of fractured rocks developed within the Stripa Project
Black, J; Holms, D; Brightman, M
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Reconstructing past and predicting future regional components of global change: The case of glaciation in Europe
Boulton, G S; Payne, A
In: Waste Disposal and Geology, Scientific Perspectives. Proc. of the Workshop WC-1 of the 29th International Geological Congress, Tokyo 1992, ed. H. Koide, pp. 51-134

Spent fuel dissolution modelling
Bruno, J; Casas, I
Third Workshop of the Cigar Lake Analog Study, Winnipeg, Canada, June 1992

Kinetic and thermodynamic studies of natural analogues for the long term evolution of spent fuel
Bruno, J; Casas, I; Cera, E
Workshop of Spent Nuclear Fuel, Visby, Sweden, September 1992

SIMFUEL dissolution studies
Bruno, J; Casas, I; Cera, E; Duro, L
Workshop of Spent Nuclear Fuel, Visby, Sweden, September 1992

Static and dynamic SIMFUEL dissolution studies under oxic conditions
Bruno, J; Casas, I; Sandino, A
Journal of Nuclear Materials 190(1992), pp. 61-69

Testing models of trace element geochemistry at Poços de Caldas
Bruno, J; Cross, J E; Eikenberg, J; McKinley, I G; Read, D; Sandino,
Journal of Geochemical Exploration 45(1992), pp. 451-470

On the influence of carbonate in mineral dissolution: I. The thermodynamics and kinetics of hematite dissolution in bicarbonate solutions at T=25°C
Bruno, J; Stumm, W; Wersin, P; Brandberg, F
Geochimicia et Cosmochimica Acta 56(1992), pp. 1139-1147

Field and laboratory studies of the reduction and sorption of technetium(VII)
Byegård, J; Albinsson, Y; Skarnemark, G; Skålberg, M
Radiochimica Acta 58/59(1992), 239

The Äspö Hard Rock Laboratory – some notes on the current status
Bäckblom, G; Ericsson, L O
In: Papers presented at the Rock Mechanics Meeting in Stockholm, March 19, 1992, pp. 143-155

Äspö Hard Rock Laboratory – high-lights
Bäckblom, G; Stanfors, R
Paper presented at the 29th Geological Conference, Kyoto, Japan, September 2, 1992

Properties and functions of smectic clays in a repository
Börgesson, L; Pusch, R
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Sealing of fractured rocks
Börgesson, L; Pusch, R
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Trace element speciation
Casas, I; Bruno, J
Third Workshop of the Cigar Lake Analog Study, Winnipeg, Canada, June 1992

Kinetic studies of natural analogues for the long term evolution of spent nuclear fuel under oxidizing conditions
Casas, I; Cera, E; Bruno, J
Materials Research Society Symposium, Boston, USA, December 1992

The Poços de Caldas Project: An introduction and summary of its implications for radioactive waste disposal
Chapman, N A; McKinley, I G; Penna Franca, E; Shea, M E; Smellie, J A T
In: The Poços de Caldas Project: Natural analogues of processes in a radioactive waste repository, Part I. J. Geochem. Explor. Spec. Issue 45(1992), pp. 1-24

Borehole seismics
Cosma, C
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

THE AECL/SKB Cigar Lake Analogy Study: Some implications for performance assessment
Cramer, J J; Smellie, J A T
Presented by Smellie at the 5th CEC/NAWG meeting at Toledo, Spain, Oct. 5-9, 1992 (to be published in the Workshop Proc. (CEC EUR Series Report))

Studies of groundwater chemistry at Stripa: An evaluation
Davis, S
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

The role of the Stripa Phase 3 Project in the development of practical discrete fracture modelling thechnology
Dershowitz, W
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Relevance of the International Stripa Project to a national nuclear waste management program
Dormuth, K W; Whitaker, S H
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

The Swedish sea transportation system – for safety reasons
Dybeck, P
PATRAM -92, Yokohama, Japan, September 1992

Transport of irradiated nuclear fuel by sea in Sweden
Dybeck, P
First Technical Committee Meeting of the joint IAEA/IMO Working Group on the safe carriage of (TNF) by sea, London, England, December 1992

A decade of successful domestic sea transport of radioactive waste in Sweden, 1982-1992
Dybeck, P; Gustafsson, B
PATRAM -92, Yokohama, Japan, September 1992

Bacteria in groundwater from deep Swedish granitic bedrock.
Ekendahl, S; Pedersen, K; Ståhl, F
Poster presentation at the Sixth International Symposium on Microbial Ecology (ISME-6), Barcelona, Spain, 6-11 September 1992

Structure and activity of bacterial communities in deep granitic groundwater
Ekendahl, S; Pedersen, K; Ståhl, F
Poster presentation at the Annual meeting of the Swedish Society of Microbiology, Göteborg, 9-10 April 1992

Hydrolysis studies of Th using solvent extraction technique
Engkvist, I; Albinsson, Y
Radiochimica Acta 58/59(1992), 109

Hydrolysis studies of Tm using solvent extraction techniques
Engkvist, I; Albinsson, Y; Yi Min
Solv. Extr. 1990, T. Sekine (ed.), Elsevier Science Publ., 473, 1992

The solubility of $TcO_2 \cdot nH_2O$ in neutral to alkaline solutions under constant pCO_2
Eriksen, T E; Ndalamba, P; Bruno, J; Caceci, M
Radiochimica Acta 58/59(1992), pp. 67-70

The role of natural analogues in performance assessment: Applications and limitations
Ewing, R C
In: Proc of the Third International Conference on High Level Radioactive Waste Management, ed. James S. Tulenko (American Nuclear Soc.), 1992, pp. 1429-1436

Material properties: The use of natural materials to predict the long-term behaviour of nuclear waste forms
Ewing, R C; Murakarni, T
In: Waste Disposal and Geology Scientific Perspectives. Proc of the 29th International Geological Congress Workshop WC-1, Tokyo, Japan, September 4-5, 1992, pp. 285-298

Alteration of natural uranyl oxide hydrates in Si-rich groundwaters: Implications for uranium solubility
Finch, R J; Ewing, R C
Scientific Basis for Nuclear Waste Management XV, C. Sombret (ed.), Mat. Research Soc., Pittsburgh, Pennsylvania, USA, 257(1992), 465-472

Corrosion of uraninite under oxidizing conditions
Finch, R J; Ewing, R C
Journal of Nuclear Materials 190(1992), pp. 133-156

Uranyl oxide hydrates and uraninite corrosion: Relevance to "Natural Analogue" studies of spent fuel corrosion
Finch, R J; Ewing, R C
In: Proc of the Third International Conference on High Level Radioactive Waste Management, ed. James S. Tulenko (American Nuclear Soc.), 1992, pp. 332-337

Weathering of natural uranyl oxide hydrates: Schoepite polytypes and dehydration effects
Finch, R J; Miller, M L; Ewing, R C
Radiochimica Acta 58/59(1992), pp. 433-443

Kärnavfall – inte bara en teknisk fråga
Forsström, H
Seminarium "Kärnavfall i ett humanekologiskt perspektiv", Umeå, Maj 1992

Spent fuel corrosion and dissolution
Forsyth, R S; Werme, L O
Journal of Nuclear Materials 190(1992), pp. 3-19

A comparison of calcite fracture mineralogy and geochemistry for the Canadian and Fennoscandian shields
Frape, S K; Blyth, A R; Jones, M G, Blomqvist, R; Tullborg, E-L;
In: Water-Rock Interaction, Khanaka & Maest (eds.), 1992, Balkema, Rotterdam, Proc of the 7th Int. Symp. on Water-Rock Interaction-WRI-7, Park City, Utah, USA, 13-18 July, 1992

Characterizing the geometry of the Stripa fracture system for flow and transport studies
Gale, J; MacLeod, R; Bursey, G; Stråhle, A; Carlsten, S; Tirén, S
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Equivalent porus media modeling of flow and transport in the granitic rocks at Stripa
Gale, J; MacLeod, R; Bursey, G; Herbert, A
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Discrete fracture modelling of in-situ hydrologic and tracer experiments
Geier, Joel; Dershowitz, William S; Wallmann, Peter C; Doe, Thomas W
Paper presented at the ISRM Fractured and Jointed Rock Masses
Conference, Tahoe City, California, USA, June 3-5, 1992

Engineered barriers investigations within the Stripa Project
Grey, M; Gera, F
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Stripa Project modelling in the site characterization and validation program
Gustafson, G
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

High-FeO olivine rock, a potential redox-active backfill working in a nature-analogue way
Hellmuth, K-H; Lindberg, A; Tullborg, E-L
Poster presented at the 5th CEC Natural Analogue Working Group Meeting and Alligator River Analogue Project (ARAP) Final Workshop, held in Toledo, Spain, on October 5-9, 1992

Discrete fracture network modelling for Phase 3 of the Stripa Project using NAPSAC
Herbert, A; Lanyon, G; Gale, J; MacLeod, R
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

An overview of fracture flow modelling at Stripa
Hodgkinson, David P
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Alteration of uranium minerals in the Koongarra deposit, Australia – Unweathered zone
Isobe, H; Ewing, R C
Journal of Nuclear Materials 190(1992), pp. 174-187

Sensitivity analysis of rock mass response to glaciation of the Finnsjön site, Sweden
Israelsson, J; Stephansson, O
In: Proc of the Finnish Symposium on Rock Mechanics 1992, eds. Jokinen, J et al., 1992

Migration of the Chernobyl fallout in different types of forest soils – Gideå Study Site – II
Ittner, T; Carbol, P; Skålberg, M
Seminar on the dynamics of the behaviour of radionuclides in contaminated forests, Stockholm, May 18-22, 1992

Coffinitization – A possible mechanism for the alteration of spent fuel under reducing conditions
Janeczek, J; Ewing, R C
Scientific Basis for Nuclear Waste Management XV, C. Sombret (ed.), Mat. Research Soc., Pittsburgh, Pennsylvania, USA, 257(1992), 497-504

Corrosion and alteration of uraninite under reducing conditions
Janeczek, J; Ewing, R C
Journal of Nuclear Materials 192(1992), pp. 157-173

Heterogeneity and alteration of uraninite from the natural fission reactor 10 at Oklo, Gabon
Janeczek, J; Ewing, R C
In: EUR Proc of the 2nd Joint CEC-CEA OKLO Working Group Meeting, Brussels, Belgium, 1992, pp. 187-201

The structural formula of uraninite
Janeczek, J; Ewing, R C
Journal of Nuclear Materials 190(1992), pp. 128-132

Validation of models for safety assessment of high-level waste disposal in Sweden
Karlsson, F
In: "Sicherheitstechnische Aspekte bei der Endlagerung radioaktiver Abfälle", GSF-Bericht 6/92(1992), pp. 41-46

Slutförvar för radioaktivt avfall – Kemins betydelse
Karlsson, F; Neretnieks, I
Kemisk Tidskrift 9/92, pp. 24-27, Del 1; 10/92, pp. 21-24, Del 2

The application of natural analogues to the Swedish SKB-91 safety performance assessment
Karlsson, F; Smellie, J A T; Höglund, L-O
In: Proc of the 5th CEC Natural Analogue working Group Meeting held in Toledo, Spain on 5-9 October 1992. To be published in the EUR series

Dating of late quaternary faulting in northern Sweden
Lagerbäck, R
Journal of the Geological Society, London, 149(1992), pp. 285-291

Overview of separation and transmutation activities in Sweden and some observations
Liljenzin, J O; Skålberg, M
Presented at the Symposium on Separation Technology and Transmutation Systems (STATS), Washington, DC, USA, 13-14 January, 1992

Modelling flow and transport for Stripa Phase III: What did we learn?
Long, J
Presented at the Fourth International Conference on Ground Penetrating Radar, June 8-13, 1992, in Rovaniemi, Finland

Comparison of in-situ and laboratory corrosion experiments with borosilicate nuclear waste glass
Lutze, W; Ewing, R C
Scientific Basis for Nuclear Waste Management XV, C. Sombret (ed.), Mat. Research Soc., Pittsburgh, Pennsylvania, 257(1992), pp. 465-472

Long term effects of acid deposition: Implications on the performance of high level nuclear waste repositories
Nebot, J; Bruno, J
Environmental Contamination, 5th International Conference, Morges, Switzerland, September 1992

Some important mechanisms and processes in the near-field of the Swedish repository for spent nuclear fuel
Neretnieks, I
Paper presented at MRS conference in Boston, Nov/Dec 1992

A dynamic model of the Cs-137 concentration in fish applied on seven different lake ecosystems – A vamp scenario
Nordlinder, S; Bergström, U
6:e Nordiska Radioekologiska Seminariet, Torshavn, Färöarna

Groundwater chemistry and geochemical modelling of water-rock interactions at the Osamu Utsumi mine and Morro do Ferro analogue study sites, Poços de Caldas, Minas Gerais, Brazil
Nordstrom, D K; McNutt, R H; Puigdomènech, I; Smellie, J A T; Wolf, M
In: The Poços de Caldas Project: Natural analogues of processes in a radioactive waste repository, Part I. J. Geochem. Explor. Special Issue 45(1992), pp. 249-287

Simfuel dissolution studies in granitic groundwater
Ollila, K
Spent Nuclear Fuel Workshop 92 organized by SKB, Visby, September 1992

Simfuel dissolution studies in granitic groundwater
Ollila, K
Journal of Nuclear Materials 190(1992), 70-77

Simfuel dissolution studies in granitic groundwater, leaching experiments at VTT
Ollila, K
YJT-92-24, Helsinki, 1992

An integrated approach to site characterization – the Stripa Project experience
Olsson, O
Keynote address to the International Geological Congress in Kyoto, Japan, August 1992

Borehole radar difference tomography applied to mapping of groundwater flow through crystalline rock
Olsson, O
Presented at the Fourth International Conference on Ground Penetrating Radar, June 8-13, 1992, in Rovaniemi, Finland

Characterization of fractures and groundwater flow in the Stripa mine
Olsson, O
In: Papers presented at the Rock Mechanics Meeting in Stockholm, March 19, 1992, pp. 109-128

Fracture characterization in crystalline rock by borehole radar
Olsson, O; Andersson, P; Carlsten, S; Falk, L; Niva, B; Sandberg, E
Geological Survey of Canada Paper 90-4. Ground Penetrating Radar ed. by J.A. Pilon, Canada, 1992

Borehole radar applied to characterization of hydraulically conductive fracture zones in crystalline rock
Olsson, O; Falk, L; Forslund, O; Lundmark, L; Sandberg, E
Geophysical Prospecting 40(1992), 109-142

Borehole radar and its application to characterization of fracture zones
Olsson, O
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Conceptual model development within the SCV Project
Olsson, O; Black, J; Gale, J
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

The site characterization and validation project
Olsson, O
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Research on cement based grouts for the OECD/NEA International Stripa Project
Onofrei, M
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Some recent research and developments in Swedish tunnel blasting
Ouchterlony, F
SveDeFo Report DS 1992:1, chapter 4, pp. 60-68

Blasting damage in the ramp of the SKB Underground Laboratories at Äspö in Sweden – Verification of predictions based on vibration measurements
Ouchterlony, F; Nakagawa, K
In: Proc of the 24th Japan Symposium on Rock Mechanics, Japan Soc. of Civil Engineers, Tokyo, Japan, 6-7 February, 1992, pp. 395-399

Final disposal of spent nuclear fuel in Sweden. The importance of the site for long-term safety
Papp, T; Kjellbert, N
Paper presented at the International Symposium on Geologic Disposal of Spent Fuel, High-level and Alpha-bearing Wastes, Antwerp, Belgium, 19-23 October 1992

Possible effects of bacteria on trace element migration in crystalline bed-rock
Pedersen, K; Albinsson, Y
Radiochimica Acta 58/59(1992), 365/369

Assimilation of CO_2 and introduced organic compounds by bacterial communities in ground water from Southeastern Sweden deep crystalline bed-rock
Pedersen, K; Ekendahl, S
Microbial Ecology 23(1992), 1-14

Incorporation of CO_2 and introduced organic compounds by bacterial populations in ground water from the deep crystalline bed-rock of the Stripa mine
Pedersen, K; Ekendahl, S
Journal of General Microbiology 138(1992), 369-376

Costs for the Swedish radioactive waste management
Pettersson, S; Forsström, H
In: Waste Management '92 "Working towards a cleaner environment". Proc of the Symposium on Waste Management at Tucson, Arizona, March 1-5, 1992, Vol. 1, pp. 765-769

Sealing of granite rock as evaluated from Stripa experiments
Pusch, R; Börgesson, L
In: Papers presented at the Rock Mechanics Meeting in Stockholm, March 19, 1992, pp. 129-142

Sealing of boreholes, shafts and tunnels
Pusch, R; Börgesson, L
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

The buffer mass test (BMT)
Pusch, R; Börgesson, L
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

The Stripa clay sealing program – Intentions, results and afterthoughts
Pusch, R
Paper presented at the Fourth International Symposium on the OECD/NEA Stripa Project, Stockholm, October 14-16, 1992

Mathematical formulas for the glacio-isostatic uplift in Fennoscandia and the global enstatic rise
Påsse, T
International Geological Correlation Programm 274. Int. Symposium on Diversity in Coastal Evolutions in the Quaternay. Annual Meeting in Wellington, New Zealand, 1992

Movement of the redox front at a uranium mine at Poços de Caldas
Romero, L; Neretnieks, I; Moreno, L
Journal of Geochemical Exploration 45(1992), pp. 471-502

Models for calculating radionuclide release from the near field
Romero, L; Nilson, L; Moreno, L; Neretnieks, I
Presented at IHLRW Conference, Las Vegas, USA, April 1992

Modelling of rock mass response to glaciation in Sweden
Rosengren, L; Stephansson, O
In: Papers presented at the Rock Mechanics Meeting in Stockholm, March 19, 1992, pp. 21-36

Sorption of fission products on colloids made of naturally occurring minerals and the stability of these colloids
Sätmark, B; Albinsson, Y
Radiochimica Acta 58/59(1992), 155

The solubility of $(UO_2)_3(PO_4)_2 \cdot 4H_2O(s)$ and the formation of U(VI) phosphate complexes: Their influence in uranium speciation in natural waters
Sandino, A; Bruno, J
Geochimica et Cosmochimica Acta 56(1992), 4135-45

Äspö Hard Rock Laboratory. Plans for radioactive tracer experiments and facilities needed
Skålberg, M
Dept. of Nuclear Chemistry, Chalmers Univ. of Techn., Internal report KKR 921014, Göteborg, 1992

Äspölaboratoriet. Radiokemiskt baslaboratorium vid CLAB
Skålberg, M
Inst. för Kärnkemi, Chalmers tekn. högskola, Internrapport KKR 921204, Göteborg, 1992

The Gideå Study Site – II – Area description and radionuclide deposition pattern
Skålberg, M; Carbol, P; Gustafsson, E; Ittner, T; Mattiasson, L
Seminar on the dynamics of the behaviour of radionuclides in contaminated forests, Stockholm, May 18-22, 1992

Swedish activities in the Oklo Natural Analogue Project
Smellie, J A T; Winberg, A; Karlsson, F
In: EUR Proc of the 2nd joint CEC-CEA OKLO Working Group Meeting held in Brussels on 6-7 April, 1992, pp. 137-148

Undersökning av sprängskador i Äspölaboratoriet (Blasting damage investigation in the Äspö Hard Rock Laboratory)
Stanfors, R
In: Papers presented at the Rock Mechanics Meeting in Stockholm, March 19, 1992, pp. 191-203

Stochastic modelling of groundwater flow in fractured rock
Ström, A
Paper presented at the Convex User Conference 1992

Långsiktig överföring mellan geosfär och biosfär beskriven genom sedimentstudier vid Äspö HRL
Sundblad, B
Nordisk Hydrologisk Konferens i Alta, Norge, augusti 1992

The turnover of Cs-137 within a forest ecosystem described by two modelling approaches – Gideå Study Site - VI
Sundblad, B; Nordqvist, R; Jönsson, S
Seminar on the dynamics of the behaviour of radionuclides in contaminated forests, Stockholm, May 18-22, 1992

Swedish developments of spent fuel direct disposal concepts
Svemar, C
Paper presented at the International Symposium on Geologic Disposal of Spent Fuel, High-level and Alpha-bearing Wastes, Antwerp, Belgium, 19-23 October 1992

The Swedish programme for siting of a deep geological repository for spent nuclear fuel
Thegerström, C
Paper presented at the International Symposium on Geologic Disposal of Spent Fuel, High-level and Alpha-bearing Wastes, Antwerp, Belgium, 19-23 October 1992

Sammanställning och analys av tillämpliga lagar vid lokalisering av ett djupförvar för använt kärnbränsle
Tunbrant, S
Vattenfall Energisystem AB, 1992

Stabila isotoper som indikatorer på ursprung och perkolationsmönster för salta grundvatten (Stable isotopes as indicators for the origin and percolation pattern for salt groundwater)
Wallin, B
Nordiskt seminarium om salt grundvatten, Saltsjöbaden, Oktober 1992

Experiences from geohydrological investigations at the Äspö HRL site in Sweden
Wikberg, P; Smellie, J A T; Wallin, B; Tullborg, E-L; Laaksoharju, M
Presented by Wikberg at the OECD/NEA Workshop on Paleohydrogeological methods and their application for radioactive waste disposal, 9-10 November 1992 (to be published in the proceedings)

LIST OF SKB ANNUAL REPORTS 1977 — 1991

1977-78
TR 121
KBS Technical Reports 1–120
Summaries. Stockholm, May 1979.

1979
TR 79-28
The KBS Annual Report 1979
KBS Technical Reports 79-01–79-27.
Summaries. Stockholm, March 1980.

1980
TR 80-26
The KBS Annual Report 1980
KBS Technical Reports 80-01–80-25.
Summaries. Stockholm, March 1981.

1981
TR 81-17
The KBS Annual Report 1981
KBS Technical Reports 81-01–81-16.
Summaries. Stockholm, April 1982.

1982
TR 82-28
The KBS Annual Report 1982
KBS Technical Reports 82-01–82-27.
Summaries. Stockholm, July 1983

1983
TR 83-77
The KBS Annual Report 1983
KBS Technical Reports 83-01–83-76.
Summaries. Stockholm, June 1984.

1984
TR 85-01
Annual Research and Development Report 1984
Including Summaries of Technical Reports Issued during 1984. (Technical Reports 84-01–84-19)
Stockholm June 1985.

1985
TR 85-20
Annual Research and Development Report 1985
Including Summaries of Technical Reports Issued during 1985. (Technical Reports 85-01–85-19)
Stockholm, May 1986.

1986
TR 86-31
SKB Annual Report 1986
Including Summaries of Technical Reports Issued during 1986
Stockholm, May 1987

1987
TR 87-33
SKB Annual Report 1987
Including Summaries of Technical Reports Issued during 1987
Stockholm, May 1988

1988
TR 88-32
SKB Annual Report 1988
Including Summaries of Technical Reports Issued during 1988
Stockholm, December 1989

1989
TR 89-40
SKB Annual Report 1989
Including Summaries of Technical Reports Issued during 1989
Stockholm, May 1990

1990
TR 90-46
SKB Annual Report 1990
Including Summaries of Technical Reports Issued during 1990
Stockholm, May 1991

1991
TR 91-64
SKB Annual Report 1991
Including Summaries of Technical Reports Issued during 1991
Stockholm, May 1992

LIST OF SKB TECHNICAL REPORTS 1992

TR 92-01
GEOTAB. Overview
Eriksson, Ebbe 1); Johansson, Bertil 2); Gerlach, Margareta 3); Magnusson, Stefan 2); Nilsson, Ann-Chatrin 4); Sehlstedt, Stefan 3); Stark, Tomas 1)
SGAB 1); ERGODATA AB 2); MRM Konsult AB 3); KTH 4)
January 1992

TR 92-02
Sternö study site. Scope of activities and main results
Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Nordqvist, Rune 2); Ljunggren, Christer 3); Tirén, Sven 2); Voss, Clifford 4)
Conterra AB 1); Geosigma AB 2); Renco AB 3); U.S. Geological Survey 4)
January 1992

TR 92-03
Numerical groundwater flow calculations at the Finnsjön study site – extended regional area
Lindbom, Björn; Boghammar, Anders
Kemakta Consultants Co, Stockholm
March 1992

TR 92-04
Low temperature creep of copper intended for nuclear waste containers
Henderson, P J; Österberg, J-O; Ivarsson, B
Swedish Institute for Metals Research, Stockholm
March 1992

TR 92-05
Buoyancy flow in fractured rock with a salt gradient in the groundwater – An initial study
Claesson, Johan
Department of Building Physics, Lund University, Sweden
February 1992

TR 92-06
Characterization of nearfield rock – A basis for comparison of repository concepts
Pusch, Roland; Hökmark, Harald
Clay Technology AB and Lund University of Technology, Lund, Sweden
December 1991

TR 92-07
Discrete fracture modelling of the Finnsjön rock mass: Phase 2
Geier, J E; Axelsson, C-L; Hässler, L; Benabderrahmane, A
Golder Geosystem AB, Uppsala, Sweden
April 1992

TR 92-08
Statistical inference and comparison of stochastic models for the hydraulic conductivity at the Finnsjön-site
Norman, Sven
Starprog AB
April 1992

TR 92-09
Description of the transport mechanisms and pathways in the far field of a KBS-3 type repository
Elert, Mark 1); Neretnieks, Ivars 2); Kjellbert, Nils 3); Ström, Anders 3)
KEMAKTA Konsult AB 1); Royal Institute of Technology 2); Swedish Nuclear Fuel and Waste Management Co 3)
April 1992

TR 92-10
Description of groundwater chemical data in the SKB database GEOTAB prior to 1990
Laurent, Sif 1); Magnusson, Stefan 2); Nilsson, Ann-Chatrin 3)
IVL, Stockholm 1); Ergodata AB, Göteborg 2); Dept. of Inorg. Chemistry, KTH, Stockholm 3)
April 1992

TR 92-11
Numerical groundwater flow calculations at the Finnsjön study site – the influence of the regional gradient
Lindbom, Björn; Boghammar, Anders
Kemakta Consultants Co., Stockholm, Sweden
April 1992-04

TR 92-12
HYDRASTAR – a code for stochastic simulation of groundwater flow
Norman, Sven
Starprog AB
May 1992

TR 92-13
Radionuclide solubilities to be used in SKB 91
Bruno, Jordi 1); Sellin, Patrik 2)
MBT Tecnologia Ambiental, Cerdanyola, Spain 1); Swedish Nuclear Fuel and Waste Management Co (SKB), Stockholm, Sweden 2)
June 1992

TR 92-14
Numerical calculations on heterogeneity of groundwater flow
Follin, Sven
Royal Institute of Technology, Department of Land and Water Resources, Stockholm
June 1992

TR 92-15
Kamlunge study site. Scope of activities and main results
Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Andersson, Peter 2); Ittner, Thomas 2); Ljunggren, Christer 3); Tirén, Sven 2)
Conterra AB 1); Geosigma AB 2); Renco AB 3)
May 1992

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Equipment for deployment of canisters with spent nuclear fuel and bentonite buffer in horisontal holes
Henttonen, Vesa; Suikki, Mikko
JP-Engineering Oy, Raisio, Finland
June 1992

TR 92-17
The implication of fractal dimension in hydrogeology and rock mechanics. Version 1.1
Dershowitz, W 1); Redus, K 1); Wallmann, P 1); LaPointe, P 1); Axelsson, C-L 2)
Golder Associates Inc., Seattle, Washington, USA 1); Golder Associates Geosystem AB, Uppsala, Sweden 2)
February 1992

TR 92-18
Stochastic continuum simulation of mass arrival using a synthetic data set. The effect of hard and soft conditioning
Kung Chen Shan 1); Wen Xian Huan 1); Cvetkovic, Vladimir 1); Winberg, Anders 2)
Royal Institute of Technology, Stockholm 1); Conterra AB, Gothenburg 2)
June 1992

TR 92-19
Partitioning and transmutation. A review of the current state of the art
Skålberg, Mats; Liljenzin, Jan-Olov
Department of Nuclear Chemistry, Chalmers University of Technology, Göteborg, Sweden
October 1992

TR 92-20
SKB 91 – Final disposal of spent nuclear fuel. Importance of the bedrock for safety
SKB
May 1992

TR 92-21
The Protogine Zone. Geology and mobility during the last 1.5 Ga
Andréasson, Per-Gunnar; Rodhe, Agnes
September 1992

TR 92-22
Klipperås study site. Scope of activities and main results
Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Andersson, Peter 2); Ittner, Thomas 2); Ljunggren, Christer 3); Tirén, Sven 2)
Conterra AB 1); Geosigma AB 2); Renco AB 3)
September 1992

TR 92-23
Bedrock stability in Southeastern Sweden. Evidence from fracturing in the ordovician limestones of Northern Öland
Milnes, Alan Geoffrey 1); Gee, David G 2)
Geological and Environmental Assessments (GEA), Zürich, Switzerland 1); Geologiska Institutionen, Lund, Sweden 2)
September 1992

TR 92-24
Plan 92. Costs for management of the radioactive waste from nuclear power production
Swedish Nuclear Fuel and Waste Management Co (SKB)
June 1992

TR 92-25
Gabbro as a host rock for a nuclear waste repository
Ahlbom, Kaj 1); Leijon, Bengt 1); Liedholm, Magnus 2); Smellie, John 1)
Conterra AB 1); VBB VIAK 2)
September 1992

TR 92-26
Copper canisters for nuclear high level waste disposal. Corrosion aspects
Werme, Lars; Sellin, Patrik; Kjellbert, Nils
Swedish Nuclear Fuel and Waste Management Co, Stockholm, Sweden
October 1992

TR 92-27
Thermo-mechanical FE-analysis of butt-welding of Cu-Fe canister for spent nuclear fuel
Josefson, B L 1); Karlsson, L 2); Lindgren, L-E 2); Jonsson, M 2)
Division of Solid Mechanics, Chalmers University of Technology, Göteborg, Sweden 1); Division of Computer Aided Design, Luleå University of Technology, Luleå, Sweden 2)
October 1992

TR 92-28
A rock mechanics study of Fracture Zone 2 at the Finnsjön site
Leijon, Bengt 1); Ljunggren, Christer 2)
Conterra AB 1); Renco AB 2)
January 1992

TR 92-29
Release calculations in a repository of the very long tunnel type
Romero, L ; Moreno, L; Neretnieks, I
Department of Chemical Engineering, Royal Institute of Technology, Stockholm, Sweden
November 1992

TR 92-30
Interaction between rock, bentonite buffer and canister. FEM calculations of some mechanical effects on the canister in different disposal concepts
Börgesson, Lennart
Clay Technology AB, Lund, Sweden
July 1992

TR 92-31
The Äspö Hard Rock Laboratory: Final evaluation of the hydrogeochemical pre-investigations in relation to existing geologic and hydraulic conditions
Smellie, Johan 1); Laaksoharju, Marcus 2)
Conterra AB, Uppsala, Sweden 1); GeoPoint AB, Stockholm, Sweden 2)
November 1992

TR 92-32
Äspö Hard Rock Laboratory. Evaluation of the combined longterm pumping and tracer test (LPT2) in borehole KAS 06
Rhén, Ingvar (ed.) 1); Svensson, Urban (ed.) 2); Andersson, Jan-Erik 3); Andersson, Peter 3); Eriksson, Carl-Olof 3); Gustafsson, Erik 3); Ittner, Thomas 3); Nordqvist, Rune 3)
VBB VIAK AB 1); Computer-aided Fluid Engineering 2); Geosigma AB 3)
November 1992

TR 92-33
Finnsjön study site. Scope of activities and main results
Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Andersson, Peter 2); Ittner, Thomas 2); Ljunggren, Christer 3); Tirén, Sven 2)
Conterra AB 1); Geosigma AB 2); Renco AB 3)
December 1992

TR 92-34
Sensitivity study of rock mass response to glaciation at Finnsjön, central Sweden
Israelsson, Jan 1); Rosengren, Lars 1); Stephansson, Ove 2)
Itasca Geomekanik AB, Falun, Sweden 1); Royal Institute of Technology, Dept. of Engineering Geology, Stockholm, Sweden 2)
November 1992

TR 92-35
Calibration and validation of a stochastic continuum model using the Finnsjön dipole tracer test. A contribution to INTRAVAL phase 2
Kung Chen Shan 1); Cvetkovic, Vladimir 1); Winberg, Anders 2)
Royal Institute of Technology, Stockholm 1); Conterra AB, Göteborg 2)
December 1992

TR 92-36
On the interpretation of double-packer tests in heterogeneous porous media: Numerical simulations using the stochastic continuum analogue
Follin, Sven
Department of Engineering Geology, Lund University, Lund, Sweden
December 1992

TR 92-37
Thermodynamic modelling of bentonite-groundwater interaction and implications for near field chemistry in a repository for spent fuel
Wanner, Hans; Wersin, Paul; Sierro, Nicholas
MBT Umwelttechnik AG, Zürich, Switzerland
November 1992

TR 92-38
Climatic changes and uplift patterns – past, present and future
Björck, S; Svensson, N-O
Department of Quaternary Geology, University of Lund, Lund, Sweden
November 1992

TR 92-39
Characterization of crystalline rocks in deep boreholes. The Kola, Krivoy Rog and Tyrnauz boreholes
NEDRA
December 1992

TR 92-40
PASS-Project on Alternative Systems Study. Performance assessment of bentonite clay barrier in three repository concepts: VDH, KBS-3 and VLH
Pusch, Roland; Börgesson, Lennart
Clay Technology AB, Lund, Sweden
December 1992

TR 92-41
Buoyancy flow in fractured rock with a salt gradient in the groundwater. A second study of coupled salt and thermal buoyancy
Claesson, Johan; Hellström, Göran; Probert, Thomas
Depts. of Building Physics and Mathematical Physics, Lund University, Sweden
November 1992

TR 92-42
Project on Alternative Systems Study – PASS. Comparison of technology of KBS-3, MLH VLH and VDH concepts by using an expert group
Olsson, Lars 1); Sandstedt, Håkan 2)
Geostatistik Lars Olsson AB 1); Bergsäker Öst AB 2)
September 1992

TR 92-43
Project on Alternative Systems Study – PASS. Analysis of performance and long-term safety of repository concepts
Birgersson, Lars; Skagius, Kristina; Wiborgh, Marie; Widén, Hans
Kemakta Konsult AB
September 1992

TR 92-44
Project on Alternative Systems Study – PASS. Cost comparison of repository systems
Ageskog, Lars; Högbom, Thomas
VBB VIAK AB
September 1992

TR 92-45
Mechanical integrity of canisters
Compiled by Nilsson, Fred
Royal Institute of Technology, Stockholm, Sweden
December 1992

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SKB ANNUAL REPORT 1992

Part V

Summaries of Technical Reports Issued during 1992

SKB Technical Report 92-01

GEOTAB. Overview

Eriksson, Ebbe 1); Johansson, Bertil 2); Gerlach, Margareta 3); Magnusson, Stefan 2); Nilsson, Ann-Chatrin 4); Sehlstedt, Stefan 3); Stark, Tomas 1)

SGAB 1); ERGODATA AB 2); MRM Konsult AB 3); KTH 4)

January 1992

SKB Technical Report 92-02

Sternö study site. Scope of activities and main results

Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Nordqvist, Rune 2); Ljunggren, Christer 3); Tirén, Sven 2); Voss, Clifford 4)

Conterra AB 1); Geosigma AB 2); Renco AB 3); U.S. Geological Survey 4)

January 1992

ABSTRACT

Preface: During the period from 1977-1986 SKB (Swedish Nuclear Fuel and Waste Management Co) performed surface and borehole investigations of 14 study sites for the purpose of assessing their suitability for a repository of spent nuclear fuel. The next phase in the SKB site selection programme will be to perform detailed characterization, including characterization from shafts and/or tunnels, of two or three sites. The detailed investigations will continue over several years to provide all the data needed for a licensing application to build a repository. Such an application is foreseen to be given to the authorities around the year 2003.

It is presently not clear if anyone of the previously investigated study sites will be selected as a site for detailed characterization. Other sites with geological and/or socio-economical characteristics judged more favourable may very well be the ones selected. However, as a part of the background documentation needed for the site selection studies to come, summary reports will be prepared for most study sites. These reports will include scope of activities, main results, uncertainties and need of complementary investigations.

This report concerns the Sternö study site. This site was one of the first sites to be investigated by SKB. The studies at Sternö were made under severe time-constraints and with prototype borehole instrumentations. These limitations should be kept in mind when reading the report.

The report has been written by the following authors; Kaj Ahlbom and Sven Tirén (scope of activities and geologic model), Jan-Erik Andersson (geohydrological model), Rune Nordqvist (groundwater chemistry), Clifford Voss (assessment of solute transport) and Christer Ljunggren (rock mechanics).

SKB Technical Report 92-03

Numerical groundwater flow calculations at the Finnsjön study site – extended regional area

Lindbom, Björn; Boghammar, Anders

Kemakta Consultants Co, Stockholm

March 1992

ABSTRACT

The present report describes modelling efforts of the groundwater flow situation at the Finnsjön site in northern Uppland, approximately 140 km north of Stockholm. The study forms part of the SKB 91 performance assessment project, and aims at describing the travel times and travel paths from a potential repository for spent nuclear fuel located in crystalline rock, and also to calculate the flux values at repository level. The groundwater flow equations were solved with the finite element technique and made use of the NAMMU-code for stationary calculations in three dimensions.

The calculations aimed at identifying a reference case from which the boundary pressures were to be extracted and used as input for future calculations with the HYDRASTAR-code, which is based on the stochastic continuum approach. The study also comprises an analysis to investigate the model sensitivity to the conductivity contrast between rock mass and fracture zones, the sensitivity to the presence of sub-horizontal fracture zones, and the degree of establishment of the discharge area.

The fracture zones were modelled implicitly with an averaging technique.

SKB Technical Report 92-04

Low temperature creep of copper intended for nuclear waste containers

Henderson, P J; Österberg, J-O; Ivarsson, B

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March 1992

ABSTRACT

Creep tests have been carried out on oxygen-free high purity copper (Cu-OF) oxygen-free phosphorus copper (Cu-OFP) and oxygen-free copper containing 0.15 wt% Ag (Cu-OFS) at temperatures between 180 and 450°C. Some Cu-OF batches exhibited poor ductility and ruptured at creep strains of less than 1% while another batch produced acceptable ductility values of about 10% elongation at fracture. These differences in ductility were attributed to a combination of sulphur content and grain size. Specimens of Cu-OFP and Cu-OFS ruptured at creep strains of 30% or greater. It was speculated that small additions of P or Ag could increase the solid solution of S in copper and therefore reduce the risk of grain boundary segregation and embrittlement or that an element like P co-segregates with S and competes for grain boundary sites.

SKB Technical Report 92-05

Buoyancy flow in fractured rock with a salt gradient in the groundwater - An initial study

Claesson, Johan

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February 1992

ABSTRACT

Nuclear waste, deposited in canisters in rock, produces heat that will induce a buoyancy flow of groundwater in fractures. The radioactive material may then, in case of leakage, possibly reach the biosphere. The groundwater density will increase downwards with an increasing salt content. This density increase counteracts the thermal buoyancy, and it may create a natural barrier for the repository.

The aim of the study is to analyse this barrier effect and to assess the extent of upward flow. The coupled flow process for groundwater, salt and heat is studied. The equations have been analysed in great detail, and a numerical model has been developed for the case of groundwater flow in a fracture or crack plane.

The largest upward motion from the repository has been determined with the model for a wide range of heat release. Approximate formulas, which are shown to be sufficiently accurate for assessments, have been derived.

There is a very clear barrier effect. In a reference case with a salt concentration increase of 2% per km downwards, and with as much as 300 canisters (releasing all in all 0.32 TWh) placed in a limited region, the largest upward movement of groundwater from the repository region became 150 m. The result is remarkably insensitive to variations of the involved parameters (heat release, distance from canisters to fracture plane, considered time, salt concentration gradient, thermal expansivity, hydraulic conductivity of flow plane and so on).

SKB Technical Report 92-06

Characterization of nearfield rock – A basis for comparison of repository concepts

Pusch, Roland; Hökmark, Harald

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December 1991

ABSTRACT

The hydraulic conductivity of the nearfield rock controls the rate of wetting of adjacent buffer material, as well as the rate of degradation of its smectite content and of the transport of radionuclides from the buffer/rock interface.

Comparison of different repository concepts with respect to the function of the nearfield rock requires a common rock structure model, which is suggested in the report. Applying this model and 2D and 3D numerical calculations for evaluation of stress-induced structural changes, major differences between the three concepts VDH, KBS3 and VLH concerning the hydraulic conductivity of the nearfield have been identified. The importance of the orientation of the excavations turns out to be particularly obvious.

Further development of the rock structure model is concluded to offer ways of quantifying more accurately the damaging effects of blasting and TBM-drilling.

SKB Technical Report 92-07

Discrete fracture modelling of the Finnsjön rock mass: Phase 2

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April 1992

ABSTRACT

A discrete fracture network (DFN) model of the Finnsjön site was derived from field data, and used to predict block-scale flow and transport properties.

The DFN model was based on a compound Poisson process, with stochastic fracture zones, and individual fractures concentrated around the fracture zones. This formulation was used to represent the multitude of fracture zones at the site which could be observed on lineament maps and in boreholes, but were not the focus of detailed characterization efforts. Due to a shortage of data for fracture geometry at depth, distributions of fracture orientation and size were assumed to be uniform throughout the site. Transmissivity within individual fracture planes was assumed to vary according to a fractal model.

Constant-head packer tests were simulated with the model, and the observed transient responses were compared with actual tests in terms of distributions of interpreted transmissivity and flow dimension, to partially validate the model. Both simulated and actual tests showed a range of flow dimension from sublinear to spherical, indicating local variations in the connectivity of the fracture population.

A methodology was developed for estimation of an effective stochastic continuum from the DFN model, but this was only partly demonstrated. Directional conductivities for 40 m blocks were estimated using the DFN model. These show extremely poor correlation with results of multiple packer tests in the same blocks, indicating possible limitations of small-scale packer tests for predicting block-scale properties.

Estimates are given of effective flow porosity and flow wetted surface, based on the block-scale flow fields calculated by the DFN model, and probabilistic models for the relationships among local fracture transmissivity, void space, and specific surface. The database for constructing these models is extremely limited. A review is given of the existing database for single fracture hydrologic properties.

SKB Technical Report 92-08

Statistical inference and comparison of stochastic models for the hydraulic conductivity at the Finnsjön-site

Norman, Sven

Starprog AB

April 1992

ABSTRACT

The origin of this study was to find a good, or even the best, stochastic model for the hydraulic conductivity field at the Finnsjö site. The conductivity fields in question are regularized, that is upscaled. The reason for performing regularization of measurement data is primarily the need for long correlation scales. This is needed in order to model reasonably large domains that can be used when describing regional groundwater flow accurately. A theory of regularization is discussed in this report.

In order to find the best model, jacknifing is employed to compare different stochastic models. The theory for this method is described. In the act of doing so we also take a look at linear predictor theory, so called kriging, and include a general discussion of stochastic functions and intrinsic random functions. The statistical inference methods for finding the models are also described, in particular regression, iterative generalized regression (IGLSE) and non-parametric variogram estimators. A large amount of results is presented for a regularization scale of 36 metre.

SKB Technical Report 92-09

Description of the transport mechanisms and pathways in the far field of a KBS-3 type repository

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April 1992

ABSTRACT
The main purpose of this document is to serve as a reference document for the far field radionuclide transport description within SKB 91. A conceptual description of far field transport in crystalline rock is given together with a discussion of the application of the stream tube concept. In this concept the transport in a complex three-dimensional flow field is divided into a number of imaginary tubes which are modelled independently. The stream tube concept is used as the basis for the radionuclide calculations in SKB 91. Different mathematical models for calculating the transport of radionuclides in fractured rock are compared: advection dispersion models, channeling models and network models. In the SKB 91 project a dual-porosity continuum model based on the one dimensional advection-dispersion equation taking into account matrix diffusion, sorption in the rock matrix and radioactive chain decay.

Furthermore, the data needed for the transport models is discussed and recommended ranges and central values are given.

SKB Technical Report 92-10

Description of groundwater chemical data in the SKB database GEOTAB prior to 1990

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April 1992

SKB Technical Report 92-11

Numerical groundwater flow calculations at the Finnsjön study site – the influence of the regional gradient

Lindbom, Björn; Boghammar, Anders

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April 1992

ABSTRACT
The present report describes the modelling efforts of the groundwater flow situation at the Finnsjön site in northern Uppland, approximately 140 km north of Stockholm. The study forms part of the SKB 91 performance assessment project, and aims at describing the model sensitivity to changes in the prevailing regional gradient, as well as the local, with regard to both direction and magnitude. Particular emphasis has been put into the evaluation of travel times and travel paths from a potential repository, and also on flux values at repository level. The analyses were based on the finite element technique and made use of the NAMMU-code for stationary calculations in three dimensions.

The fracture zones within the modelled area were modelled implicitly with an averaging technique.

SKB Technical Report 92-12

HYDRASTAR - a code for stochastic simulation of groundwater flow

Norman, Sven

Starprog AB

May 1992

ABSTRACT
The computer code HYDRASTAR was developed as a tool for groundwater flow and transport simulations in the SKB 91 safety analysis project and its conceptual ideas can be traced back to a report by Shlomo Neuman in 1988, see the reference section. The main idea of the code is the treatment of the rock as a stochastic continuum which separates it from the deterministic methods previously employed by SKB and also from the discrete fracture models. The current report is a comprehensive description of HYDRASTAR including such topics as regularization or upscaling of the hydraulic conductivity field, unconditional and conditional simulation of stochastic processes, numerical solvers for the hydrology and streamline equations and finally some proposals for future developments.

SKB Technical Report 92-13

Radionuclide solubilities to be used in SKB 91

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June 1992

ABSTRACT

We have performed thermodynamic calculations in order to assess the solubility limits (source term) for selected radionuclides. Equilibrium solubilities for U, Pu, Np, Am, Th, Ra, Sn, Tc, Zr, Sn, Ni, Sm, Pa, Nb and Pd have been calculated in four waters, representing average fresh and saline granitic groundwaters under oxidizing and reducing conditions, respectively. The results from the calculations have been compared with the measured radionuclide concentrations in natural waters as well as in spent fuel leaching tests.

SKB Technical Report 92-14

Numerical calculations on heterogeneity of groundwater flow

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June 1992

ABSTRACT

The upscaling of model parameters, i.e. scale-dependent parameters, is a key issue in many research fields concerned with parameter heterogeneity. The upscaling process allows for fewer model blocks and relaxes the numerical problems caused by high contrasts in the hydraulic conductivity. The trade-offs are dependent on the object but the general drawback is an increasing uncertainty about the representativeness, i.e. the relation to the real world problem. The present study deals with numerical calculations of heterogeneity of groundwater flow and solute transport in hypothetical blocks of fractured hard rock in a "3 m scale" and addresses both conceptual and practical problems in numerical simulation. Evidence that the hydraulic conductivity (K) of the rock mass between major fracture zones is highly heterogeneous in a 3 m scale is provided by a large number of field investigations. The present study uses the documented heterogeneity and investigates flow and transport in a twodimensional stochastic continuum characterized by a variance in $Y = \ln(K)$ of $\sigma_Y 2= 16$, corresponding to about 12 $\log_{10}$ cycles in K. The study considers anisotropy, channelling, non-Fickian and Fickian transport, and conditional simulation. The major conclusions are: (i) heterogeneity gives rise to anisotropy in the upscaling process, (ii) the choice of support scale is crucial for the modelling of solute transport. As a consequence of the obtained results, a twodimensional stochastic discontinuum model is presented, which provides a tool for linking stochastic continuum models to discrete fracture network models.

SKB Technical Report 92-15

Kamlunge study site. Scope of activities and main results

Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Andersson, Peter 2); Ittner, Thomas 2); Ljunggren, Christer 3); Tirén, Sven 2)

Conterra AB 1); Geosigma AB 2); Renco AB 3)

May 1992

ABSTRACT

PREFACE: During the period from 1977-1986 SKB (Swedish Nuclear Fuel and Waste Management Co) performed surface and borehole investigations of 14 study sites for the purpose of assessing their suitability for a repository of spent nuclear fuel. The next phase in the SKB site selection programme will be to perform detailed characterization, including characterization from shafts and/or tunnels, of two or three sites. The detailed investigations will continue over several years to provide all the data needed for a licensing application to build a repository. Such an application is foreseen to be given to the authorities around the year 2003.

It is presently not clear if anyone of the study sites will be selected as a site for detailed characterization. Other sites with geological and/or socioeconomical characteristics judged more favourable may very well be selected. However, as a part of the background documentation needed for the site selection studies to come, summary reports will be prepared for most study sites. These reports will include scope of activities, main results, uncertainties and need of complementary investigations.

This report concerns the Kamlunge study site. The report has been written by the following authors; Kaj Ahlbom and Sven Tirén (scope of activities and geologic model), Jan-Erik Andersson (geohydrological model), Thomas Ittner (groundwater chemistry), Peter Andersson (solute transport) and Christer Ljunggren (rock mechanics).

SKB Technical Report 92-16

Equipment for deployment of canisters with spent nuclear fuel and bentonite buffer in horisontal holes

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June 1992

ABSTRACT

This study presents the predesign of equipment for the deployment of canisters in long horizontal holes. The canisters are placed in the centre of the hole and are surrounded by a bentonite buffer. In this study the canisters are assumed to have a diameter of 1.6 m and a length of 5.9 m, including the hemispherical ends. Their total weight is 60 tonnes. The bentonite buffer after homogenization is 400 mm thick, making a total package diameter of 2.4 m. The deployment system consists of four wagons for handling the canisters and the bentonite blocks. To ensure safe emplacement, every part is installed separately in its final position. This also makes it possible to use small clearances between the canisters and the bentonite blocks and between the blocks and the rock wall. With small clearances, backfilling can be avoided. Another basic design idea is that the wagons are equipped with wheels, which are in direct contact with the rock walls. Thus, rails, which have to be removed as the deployment progresses, are unnecessary. To minimize the time taken for deploying one canister, the wagons are designed so that only three trips from the service area to the deposit area are needed. Due to the radiation in the vicinity of the canisters, the wagons have to be teleoperated.

SKB Technical Report 92-17

The implication of fractal dimension in hydrogeology and rock mechanics. Version 1.1

Dershowitz, W 1); Redus, K 1); Wallmann, P 1); LaPointe, P 1); Axelsson, C-L 2)

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February 1992

ABSTRACT

Since much of geology and hydrogeology is controlled by the geometry of geologic features such as faults, fractures, and stratigraphy, many researchers have proposed the use of fractal dimension as an index for comparing hydrogeologic environments. This report describes an investigation carried out by Golder Associates Geosystem AB to evaluate the use of fractal measures within the SKB site selection, evaluation, and characterization process.

This report defines fractal dimension and the methods available for calculating fractal dimension. The report then summarizes a literature survey carried out to identify and evaluate applications of fractal methods in hydrogeology. Preliminary hydrogeological fractal numerical simulations carried out with the FracMan package (Dershowitz et al., 1991) are then presented and discussed. These numerical simulations evaluate the application of fractal methods within the context of other geometric measures such as connectivity measures, percolation probability, and block size measures.

Based upon the literature survey and numerical simulations, recommendations are presented regarding the potential usefulness of fractal approaches. Fractal dimension can be used to distinguish hydrogeologic environments, provided the limitations of the approach are explicitly recognized. Recommendations are made for fractal dimension calculation procedures, specification of fractal dimension, and the use of fractal dimension in conjunction with other measures of hydrogeologic structure and heterogeneity.

SKB Technical Report 92-18

Stochastic continuum simulation of mass arrival using a synthetic data set. The effect of hard and soft conditioning

Kung, Chen Shan 1); Wen Xian Huan 1); Cvetkovic, Vladimir 1); Winberg, Anders 2)

Royal Institute of Technology, Stockholm 1); Conterra AB, Gothenburg 2)

June 1992

ABSTRACT

The non-parametric and parametric stochastic continuum approaches were applied to a realistic synthetic exhaustive hydraulic conductivity field to study the effects of hard and soft conditioning. From the reference domain, a number of data points were selected, either in a random or designed fashion, to form sample data sets. Based on established experimental variograms and the conditioning data, 100 realizations each of the studied domain were generated. The flow field was calculated for each realization, and particle arrival time and arrival position along the discharge boundary were evaluated. It was shown that conditioning on soft data reduces the uncertainty of solute arrival time, and that conditioning on soft data suggests an improvement in characterizing channelling effects. It was found that the improvement in the prediction of the breakthrough was moderate when conditioning on 25 hard and 100 soft data compared to 25 hard data only.

SKB Technical Report 92-19

Partitioning and transmutation. A review of the current state of the art

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October 1992

ABSTRACT

The recent development in the field of partitioning and transmutation (P-T) of long-lived radioactive waste nuclides from nuclear power production is reviewed and evaluated. Current national and international R&D plans are summarized. It is concluded that P-T is technically feasible but much R&D remains to be done before it is technically mature. At present there seems to be no economic gain from P-T as compared to direct disposal of spent nuclear fuel. There seems only to be an insignificant reduction in future radiation doses by P-T when compared to current disposal plans. However, future long term research may perhaps change these conclusions. Therefore the further development in this area should be followed. Some areas where a limited research by swedish scientists could be worth while are indicated.

SKB Technical Report 92-20

SKB 91 – Final disposal of spent nuclear fuel. Importance of the bedrock for safety

SKB

May 1992

ABSTRACT

The safety of a deep repository for spent nuclear fuel has been assessed in this report. The spent fuel is assumed to be encapsulated in a copper canister and deposited at a depth of 600 m in the bedrock. The primary purpose has been to shed light on the importance of the geological features of the site for the safety of a final repository.

The assessment shows that the encapsulated fuel will, in all likelihood, be kept isolated from the groundwater for millions of years. This is considerably longer than the more than 100,000 years that are required in order for the toxicity of the waste to have declined to a level equivalent to that of rich uranium ores.

However, in order to be able to study the role of the rock as a barrier to the dispersal of radioactive materials, calculations have been carried out under the assumption that waste canisters leak. The results show that the safety of a carefully designed repository is only affected to a small extent by the ability of the rock to retain the escaping radionuclides. The primary role of the rock is to provide stable mechanical and chemical conditions in the repository over a long period of time so that the function of the engineered barriers is not jeopardized.

SKB Technical Report 92-21

The Protogine Zone. Geology and mobility during the last 1.5 Ga

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September 1992

ABSTRACT

This report treats the Protogine Zone (PZ) as the western boundary of the Southeastern Megablock (SEM), and summarizes scientific aspects of different geological and geophysical functions of the zone. A systematic inventory and a technical description of shear zones and faults in the type area of the "Schistosity Zone" ("Förskiffringszonen") are presented. The report then reviews observed and inferred activity of the zone during the last 1500 million years. This calendar includes at least eight different periods of compression or extension, tilting, uplift, magmatism etc. along the zone, in harmony with the common experience that old zones of weakness in the crust seldom heal. The network of major structures of southern Sweden is described, and the function of the PZ within this network is discussed with particular attention to east-west running lineaments within the SEM, like the Nömmen-Oskarshamn and Hörnebo-Högsby fault and shear zones. Future work should inter alia investigate if these two zones are connected with the PZ, and if movements along the PZ can reactivate the zones. A bibliography comprising c. 100 titles is included as an appendix.

SKB Technical Report 92-22

Klipperås study site. Scope of activities and main results

Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Andersson, Peter 2); Ittner, Thomas 2); Ljunggren, Christer 3); Tirén, Sven 2)

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September 1992

ABSTRACT

PREFACE: During the period from 1977 – 1986 SKB (Swedish Nuclear Fuel and Waste Management Co) performed surface and borehole investigations of 14 study sites for the purpose of assessing their suitability for a repository of spent nuclear fuel. The next phase in the SKB site selection programme will be to perform detailed characterization, including characterization from shafts and/or tunnels, of two or three sites. The detailed investigations will continue over several years to provide all the data needed for a licensing application to build a repository. Such an application is foreseen to be given to the authorities around the year 2003.

It is presently not clear if any of the study sites will be selected as a site for detailed characterization. Other sites with geological and/or socio-economical characteristics judged more favorable may very well be the ones selected. However, as a part of the background documentation needed for the site selection studies to come, summary reports will be prepared for most study sites. These reports will include scope of activities, main results, uncertainties and need of complementary investigations.

This report concerns the Klipperås study site. The report has been written by the following authors; Kaj Ahlbom and Sven Tirén (scope of activities and geologic model), Jan-Erik Andersson (geohydrological model), Tomas Ittner (groundwater chemistry), Peter Andersson (assessment of solute transport), and Christer Ljunggren (rock mechanics).

SKB Technical Report 92-23

Bedrock stability in Southeastern Sweden. Evidence from fracturing in the ordovician limestones of Northern Öland

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September 1992

ABSTRACT

The stability of the bedrock in SE Sweden with regard to radioactive waste disposal has recently been the subject of some controversy. In order to better assess the age and significance of fracturing in the Precambrian basement at the site of the Äspö Hard Rock Laboratory (HRL), near Oskarshamn, a detailed analysis of fracturing in the lower Ordovician limestones exposed along the west coast of the neighbouring island of Öland has been carried out. The limestones form continuously exposed shore platforms, in segments up to 30 m broad and several kilometres long. These, and the numerous quarries, provide ideal objects for quantitative analysis (ground and air photo mapping, scanline logging), and unique opportunities for investigating the amount of movement on the fractures, because of well-developed bedding and abundant rod-shaped fossils on the bedding surfaces.

The fracture patterns are dominated by two sets of subvertical fractures, a NW trending closely spaced and strongly orientated set (set A) and a NNE-ENE trending widely spaced and variably orientated set (set B). Only about 10% of the fractures in both sets show lateral fossil displacement, with maximum movement of 5 cm, and only 3% of the fractures show vertical displacement of bedding (maximum 8 cm).

All in all, the lower Ordovician limestones along the exposed shoreline have suffered remarkably little deformation since deposition, i.e. over the last 500 million years. Appreciable bedrock instability, if it occurred, must have been concentrated offshore, or in the unexposed segments of the coastline, where some weak indications of slight movement (changes of a few metres in stratigraphic level) have been observed. Among other recommendations for further work, geophysical investigations to test these indications are suggested.

SKB Technical Report 92-24

Plan 92. Costs for management of the radioactive waste from nuclear power production

Swedish Nuclear Fuel and Waste Management Co (SKB)

June 1992

ABSTRACT

The Swedish nuclear power utilities are responsible for adopting such measures as are necessary in order to ensure the safe management and disposal of spent nuclear fuel and radioactive waste from the Swedish nuclear power reactors. In order to fulfil this responsibility, the nuclear power utilities have commissioned SKB, the Swedish Nuclear Fuel and Waste Management Co, to plan, build, and operate the necessary facilities and systems.

This report presents a calculation of the costs for implementing all of these measures. The cost calculations are based on a scenario for management and disposal of the radioactive waste products, which has been prepared by SKB and is described in this report.

Since disposal of the high-level (long-lived) waste will not commence until some time into the 21st century, continued RD&D activities (Research, Development and Demonstration) may reveal new methods, that can affect both system design and costs. This is expected to lead to overall simplifications in the design.

The facilities and systems that exist are:

- Transportation system for radioactive waste products.
- Central interim storage facility for spent nuclear fuel, CLAB.
- Final repository for radioactive waste from reactor operation, SFR 1.

Future facilities under planning are:

- Encapsulation station for spent nuclear fuel.
- Final repository for long-lived waste.
- Final repository for decommissioning waste.

The cost calculations also include costs for research and development and for decommissioning and dismantling of the reactor plants etc.

The total future costs of the Swedish waste management system, starting in 1993, have been calculated to be SEK 46.4 billion in January 1992 prices. These costs will be incurred over a period of about 60 years. SEK 8.7 billion has been spent up to the end of 1992.

This cost calculation is presented annually to SKI, the Swedish Nuclear Power Inspectorate, which uses it as a basis to propose a fee on the nuclear electricity production in order to cover all future expenses. The fee for 1992 is on average 1.9 öre/kWh (0.019 SEK/kWh).

Before 1992-07-01 this function of SKI was the responsibility of the National Board for Spent Nuclear Fuel (SKN), which has now been incorporated in SKI.

SKB Technical Report 92-25

Gabbro as a host rock for a nuclear waste repository

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September 1992

ABSTRACT

As an alternative to granitic rocks, gabbro and other basic rock types have been investigated with respect to their suitability to host a nuclear waste repository. The present report summarizes and examines existing geoscientific knowledge of relevance in assessing the potential merits of gabbro as a repository host rock. Implications in terms of site selection, repository construction and post-closure repository performance are also discussed. The objective of the study is to provide a basis for decisions as regards future consideration of the gabbro alternative. It is found that there are rather few gabbro bodies in Sweden, that are potentially of sufficient size to host a repository. Thus, gabbro offers little latitude as regards site selection. In comparison to siting a repository in granitic rocks, this is a major disadvantage, and it may in fact remove gabbro from further consideration. The potential advantages of gabbro refer to repository performance, and include low hydraulic conductivity and a chemical environment promoting efficient radionuclide retardation. However, results from field investigations show that groundwater flow in gabbro bodies is largely controlled by intersecting heterogeneities, in particular granitic dykes, that are significantly more conductive to water than the gabbro. In the far-field scale significant to repository performance, this may reduce or eliminate the potential effects of favorable hydraulic and chemical characteristics of the gabbro itself. In conclusion, there are apparent difficulties associated with siting a repository in gabbro, due to lack of sufficiently large gabbro bodies. On the basis of the present state of knowledge, no decisive differences can be demonstrated when comparing gabbro with granitic rocks, neither with respect to repository construction, nor as regards repository performance.

SKB Technical Report 92-26

Copper canisters for nuclear high level waste disposal. Corrosion aspects

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October 1992

ABSTRACT

A corrosion analysis of a thick-walled copper canister for spent fuel disposal is discussed. The analysis has shown that there are no rapid mechanisms that may lead to canister failure, indicating an anticipated corrosion service life of several million years. If further analysis of the copper canister is considered, it should be concentrated on identifying and evaluating processes other than corrosion, which may have a potential for leading to canister failure.

SKB Technical Report 92-27

Thermo-mechanical FE-analysis of butt-welding of Cu-Fe canister for spent nuclear fuel

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October 1992

ABSTRACT
In the Swedish nuclear waste program it has been proposed that spent nuclear fuel shall be placed in composite copper – steel canisters. These canisters will be placed in holes in tunnels located some 500 m underground in a rock storage. The canister consists of two cylinders of 4850 mm length, one inner cylinder made of steel and one outer cylinder made of copper. The outer diameter of the canister is 880 mm and the wall thickness for each cylinder is 50 mm. At the storage, the steel cylinder, which contains the spent nuclear fuel, is placed inside the copper cylinder. Thereafter, a copper end is butt welded to the copper cylinder using electron beam welding. To obtain penetration through the thickness with this weld method a backing ring is placed at the inside of the copper cylinder.

In the present paper, the temperature, strain and stress fields present during welding and after cooling after welding are calculated numerically using the FE-code NIKE-2D. The aim is to use the results of the present calculations to estimate the risk for creep fracture during the subsequent design life. A large strain formulation is employed for the calculation of transient and residual stresses in the canister based on the calculated history of the temperature field present in the canister during the welding process. The contact algorithm available in NIKE-2D is used to detect possible contact between the steel and copper cylinders during the welding. To simplify the numerical calculations and reduce the computational time, rotational symmetry is assumed.

For large gap distances between the steel and copper cylinders the residual stress field is calculated to have a shape similar to that observed in butt welded pipes with maximum axial stress values at the yield stress level. For small gap distances the backing ring will come in contact with the steel cylinder which leads to large residual stresses in the backing ring. The maximum accumulated plastic strain in the weld metal and HAZ was calculated to about 5% for both gap distances.

SKB Technical Report 92-28

A rock mechanics study of Fracture Zone 2 at the Finnsjön site

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January 1992

ABSTRACT
Comprehensive field investigations at the Finnsjön study site have revealed a subhorizontal fracture zone, termed Zone 2, that exhibits anomalous characteristics in terms of high hydraulic conductivity, governing the groundwater transport pattern on a regional scale. The present study provides an assessment of the mechanical characteristics of Zone 2. Thus, estimates of the deformational characteristics of the zone, based on available borehole information, show that the zone forms a diffuse and rather moderate mechanical contrast to the surrounding bedrock. As also verified by stress measurement results, major stress anomalies attributable to the zone are therefore not to be expected. Bound estimates of stress conditions during periods of glaciation and deglaciation are also derived, and possible impacts of these loadings on the fracture zone are discussed. It is concluded that glaciation represents stable conditions, whilst the complex loading mechanisms encountered during deglaciation may trigger reactivation of structures at shallow depth. Taking the above results as an example, implications of a feature like Zone 2 on the integrity of a hypothetical repository are discussed in more general terms. Considering the likely spatial extension of the mechanical disturbances related to the repository excavations and the fracture zone respectively, it is suggested that a mutual distance of the order of one hundred metres is sufficient to avoid mechanical interaction.

SKB Technical Report 92-29

Release calculations in a repository of the very long tunnel type

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November 1992

ABSTRACT
The VLH and KBS-3 are two alternative repository designs for high level waste. The repositories differ mainly in the layout and the canister design. In the VLH repository the canisters are placed horizontally in long boreholes. In the KBS-3 repository every canister is placed vertically in repository holes in the tunnel floor. If a small hole forms in the canister wall the nuclides diffuse through this hole into the backfill surrounding the canister. From the backfill the nuclides migrate by different pathways into the mobile water in the fractures in the rock.

A study of comparison of the nuclide release in the two alternative repositories shows releases in the same order in magnitude for a given nuclide. A quantitative assessment of the importance of the different pathways, the influence of the location of the fracture and fracture zone in relation to the damage in the canister wall and the influence of the size of the small hole (damage) is made. The assessment for the VLH-repository shows that the main pathway is that to the disturbed zone and the fracture location in relation to the damage has no large influence in the release.

SKB Technical Report 92-30

Interaction between rock, bentonite buffer and canister. FEM calculations of some mechanical effects on the canister in different disposal concepts

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July 1992

ABSTRACT

An important task of the buffer of highly compacted bentonite is to offer a mechanical protection to the canister. This role has been investigated by a number of finite element calculations using the complex elasto plastic material models for the bentonite that have been developed on the basis of laboratory tests and adapted to the code ABAQUS.

The following main functions and scenarios have been investigated for some different canister types and repository concepts:

- The effect of the water and swelling pressure
- The effect of a rock shear perpendicular to the canister axis
- The effect of creep in the copper after a rock shear displacement
- The thermomechanical effects when an initially saturated buffer is used.

SKB Technical Report 92-31

The Äspö Hard Rock Laboratory: Final evaluation of the hydrogeochemical pre-investigations in relation to existing geologic and hydraulic conditions

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November 1992

ABSTRACT

Executive summary: Introduction: The Swedish Nuclear Fuel and Management Company (SKB) is currently excavating the access tunnel to an underground experimental laboratory, the Äspö Hard Rock Laboratory, planned to be located some 500 m below the island of Äspö which is located in the Simpevarp area, southeast Sweden. The construction of an underground laboratory forms part of the overall SKB strategy to test, not only the construction techniques for deep excavation, but also the various methods and protocols required to obtain a three-dimensional model of the geology and groundwater flow and chemistry, within a fractured crystalline bedrock similar to that envisaged for the final disposal of spent fuel. Äspö was chosen because it geologically represents a variety of typical crystalline bedrock environments.

The hydrogeochemical activities described and interpreted in this report form part of the initial Pre-investigation Phase (from the surface to around 1000 metres depth) aimed at siting the laboratory, describing the natural hydrogeological and hydrogeochemical conditions in the bedrock and predicting the changes that will occur during excavation and construction of the laboratory. Hydrogeochemical interpretation has therefore been closely integrated with the hydrogeological investigations and other disciplines of major influence, in particular, bedrock geology and geochemistry and fracture mineralogy and chemistry.

A large section of this report has been devoted to the detailed investigation of each individual zone hydraulically selected, tested and sampled for hydrogeochemical characterisation. The main objective was to establish the reliability or representativeness of each groundwater collected, in relation to the bedrock level sampled. Only by achieving a set of representative groundwater samples and hence a reliable set of chemical analyses, can some of the detailed hydrogeochemical studies be carried out.

The data have been used to describe the chemistry and origin of the Äspö groundwaters, models have been developed to illustrate groundwater mixing and standard geochemical modelling approaches have been employed to understand rock/water interaction processes. An attempt has been made to integrate the hydrogeochemical information with known geological and hydrogeological parameters to construct a conceptual groundwater flow model for the island.

SKB Technical Report 92-32

Äspö Hard Rock Laboratory. Evaluation of the combined longterm pumping and tracer test (LPT2) in borehole KAS 06

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November 1992

ABSTRACT

This report summarizes the results from the LPT2 experiment. The field experiment had three major parts: a pump-

ing test, a tracer experiment and a tracer dilution experiment. These are described in detail in the appendices of the report. Numerical simulations have been carried out both prior to and after the experiments. Results from these are also reported.

The longterm pumping test and tracer test performed in KAS06, called LPT2, was the first attempt to clarify the transport of solutes in the site scale of Äspö. The test was not intended to be complete regarding the transport parameters needed for nuclide transport modelling. In the operating phase of the Äspö Hard Rock Laboratory more detailed tracer tests and numerical modelling will be conducted.

The main conclusion from the field experiment is that the present conceptual model of Äspö is sound, but some modifications may be required. These include both the extension and transmissivities of fracture zones. The field experiment has also produced additional information on the properties of the fracture zones like porosity and dispersivity. The cumulative aperture of all hydraulic fractures was estimated to $10 \cdot 10^{-3} - 30 \cdot 10^{-3}$ m for two different sets of zones. Considering the estimated width of the zones the flow porosities were estimated to 0.02 – 0.1%. The dispersivities were estimated to 0.1 – 0.2 of the flow path distance and the Peclet number to 4-11.

The numerical simulations made prior to the experiment, dealing with the travel time of tracers, were found to be in reasonable agreement with the measurements. The data gathered in the field experiment will however make it possible to pursue the modelling efforts further.

SKB Technical Report 92-33

Finnsjön study site. Scope of activities and main results

Ahlbom, Kaj 1); Andersson, Jan-Erik 2); Andersson, Peter 2); Ittner, Thomas 2); Ljunggren, Christer 3); Tirén, Sven 2)

Conterra AB 1); Geosigma AB 2); Renco AB 3)

December 1992

BSTRACT

The Finnsjön study site was selected in 1977 to provide input to the KBS-1 and KBS-2 performance assessments. The site was later used as a test site for testing new instruments and new site characterization methods, as well as a research site for studying mainly groundwater flow and groundwater transport. All together, the Finnsjön studies have involved 11 cored boreholes, down to max. 700 m depth, and extensive borehole geophysical, geochemical and geohydraulical measurements, as well as rock stress measurements and tracer tests.

This report presents the scope of the Finnsjön studies together with main results. Conceptual uncertainties in assumptions and models are discussed with emphasis on the models used for the performance assessment SKB 91. Of special interest for the Finnsjön study site is the strong influence caused by a subhorizontal fracture zone on groundwater flow, transport and chemistry.

SKB Technical Report 92-34

Sensitivity study of rock mass response to glaciation at Finnsjön, central Sweden

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November 1992

ABSTRACT

The safety analysis SKB-91 of the Swedish Nuclear Fuel and Waste Management Company (SKB) paid specific attention to the glaciation scenario and related phenomena. In the first phase, Rosengren and Stephansson (1990), used the distinct element computer code UDEC to examine the response of the rock mass in the Finnsjön area to the processes of glaciation and deglaciation.

This report describes the second phase, in which the sensitivity of the results to different in-situ stresses and fault zone strength properties have been analyzed. A statistical approach was used to extrapolate the range of in-situ stresses at depth from measured in-situ stresses at shallower depths. Three different linear in-situ stress variations with depth were defined using a 99% confidence interval. For each in-situ stress case, three fault zone strength assumptions were analyzed for an ice loading sequence, involving 3 km, 1 km, 0-1 km (ice wedge) and 0 km of ice thickness. Each combination of in-situ stress and fault zone strength was analyzed with and without an ice lake, situated on top of the ice sheet. Consequently, a total of 18 models were studied.

The results indicated significant differences in stress distribution, failure (reactivation) of fault zones, and shear displacement on fault zones for some combinations of in-situ stress, fault zone strength, and ice lake pressure. Based on the results, several preliminary recommendations for repository siting are made, as well as recommendations for further study.

SKB Technical Report 92-35

Calibration and validation of a stochastic continuum model using the Finnsjön dipole tracer test. A contribution to INTRAVAL phase 2

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December 1992

ABSTRACT

A realistic semi-synthetic transmissivity field and dipole tracer test conditions similar the actual field test conditions at Finnsjön were used to demonstrate a calibration and validation procedure applied to a stochastic continuum model. A validation strategy was set up to address whether a model calibrated and possibly validated on a local model scale, also is validated when extrapolated to a far-field scale. A generated 2D realization of the material property distribution on a far-field scale, honouring the measured single hole data and also recapturing the characteristics of the field dipole tracer test, was selected as an exhaustive reference field. The results of the simulated dipole tracer test on a local scale and of a far-field natural gradient tracer test were considered as reference field results. A large number (N=100) of conditioned parametric realizations of the studied domain were generated and the dipole tracer experiment was simulated for each realization. The porosity of the 2D aquifer model was used to calibrate the stochastic simulations on a local scale. An alternative calibration using a defined index of deviation was also applied. Subsequently, also the natural gradient tracer test was simulated on a far-field scale for the ensemble of realizations. It was found that the realizations which based on the defined index of deviation compared best with the local scale reference results did not recapture the characteristics of the reference far-field results. It was thus concluded that calibration/validation of a model on a local scale is insufficient for validating the model also on another larger transport scale. In order to reduce the uncertainty in the far-field simulations more data on the corresponding transport scale are required.

SKB Technical Report 92-36

On the interpretation of double-packer tests in heterogeneous porous media: Numerical simulations using the stochastic continuum analogue

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December 1992

ABSTRACT

Flow in fractured crystalline (hard) rocks is of interest in Sweden for assessing the post-closure radiological safety of a deep repository for high-level nuclear waste. Depending on the scale of the problem under study different modelling concepts are used. For simulation of flow and mass transport in the far field different porous media concepts are often used, whereas discrete fracture/channel network concepts are often used for nearfield simulations. Due to lack of data, it is generally necessary to have resort to singlehole double-packer test data for the far-field simulations, i.e., test data on a small scale are regularised in order to fit a comparatively coarser numerical discretisation, which is governed by various computational constraints. It is interesting to note that single-hole fixed-interval packer test data are also used as the basis for derivation of the hydro-geologic properties of discrete fracture models, despite the different assumptions regarding the geometry of flow. Obviously, techniques that improve the interpretation and the regularisation of single-hole double-packer tests are of paramount interest. In the present study the Monte Carlo method is used to investigate the relationship between the transmissivity value interpreted and the corresponding radius of influence in conjunction with single-hole double-packer tests in heterogeneous formations. The numerical flow domain is treated as a two-dimensional heterogeneous porous medium with a spatially varying diffusivity on a 3 m scale. Two methods which have been traditionally used for interpreting constant-head injection tests within the Swedish nuclear waste repository programme, namely Moye's and Jacob-Lohman's formulae, are compared and ambiguities observed in relation to real tests are discussed. The Monte Carlo simulations demonstrate the sensitivity to the correlation range of a spatially varying diffusivity field. In contradiction to what is tacitly assumed in stochastic subsurface hydrology, the results show that the lateral support scale (e.g., the radius of influence) of transmissivity measurements in heterogeneous porous media is a random variable, which is affected by both the hydraulic and statistical characteristics. If these results are general, the traditional methods for scaling-up, assuming a constant lateral scale of support and a multinormal distribution, may lead to an underestimation of the persistence and connectivity of transmissive zones, particularly in highly heterogeneous porous media.

SKB Technical Report 92-37

Thermodynamic modelling of bentonite-groundwater interaction and implications for near field chemistry in a repository for spent fuel

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November 1992

ABSTRACT

Predictions of near field geochemistry are made using a thermodynamic model for bentonite/groundwater interaction. This model is a refinement and extension of the model developed by the senior author. It is based on recent experiments performed at high solid/water ratio and adapted to the Swedish type of HLW repository design. Thus, from the obtained experimental results on solution composition, the model includes chemical reactions resulting from both the impurities and the main clay fraction within the bentonite. Ion exchange reactions are treated both with and without the contribution of edge sites. Due to its thermodynamic basis, the model exhibits prediction capability over a wide range of conditions in terms of solid/water ratio.

The modelling of repository conditions implies, due to the lack of experimental information, simplifications with regard to thermodynamic properties of the bentonite. This mainly involves the non-consideration of the temperature effect and of the acid/base properties of the solid. Nevertheless, our results yield insight into important processes affecting porewater chemistry. Thus, the model suggests that proton exchange reactions may exert a strong control on calcite dissolution within highly compacted bentonite. Estimations of chemical changes over time in the bentonite were done on the basis of a mixing tank model. These results indicate transformation of Na-bentonite to Ca-bentonite over time. The extent of this process, however, critically depends on the amount of carbonate present in the bentonite.

SKB Technical Report 92-38

Climatic changes and uplift patterns – past, present and future

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November 1992

ABSTRACT

Our knowledge about the Pleistocene (= the last 2.5 million years) climatic changes and their global environmental effects on the Earth system, e.g. the glacial- interglacial cycles, the sea level changes, and the significant crustal movements in glaciated regions, has increased greatly during the last decades. This report outlines the historical background and the present state-of-the-arts on these matters. Because the driving mechanisms and feed-back effects behind these changes have been more and more discussed in earth-science literature, analysed, and probably also better and better understood, it has become possible to present theoretical models for future climates (not including man's influence on the earth system). The report presents and discusses one such climate model (short of predicting man's future behaviour and its consequent effect on climate) and its likely implications on future climatic and glacial conditions, and bedrock movements, with focus on the Stockholm region. Possibilities for Quaternary geologists to establish and map postglacial fault zones, related to irregular bedrock movements, are also briefly outlined in the report.

SKB Technical Report 92-39

Characterization of crystalline rocks in deep boreholes. The Kola, Krivoy Rog and Tyrnauz boreholes

NEDRA

December 1992

ABSTRACT

SKB studies, as one alternative, the feasibility of disposing of spent nuclear fuel in very deep boreholes. As a part of this work NEDRA has compiled geoscientific data from three superdeep boreholes within the former Soviet Union. The holes considered were: the Kola borehole, 12261 m deep and located on the Kola Peninsula, the Krivoy Rog borehole, 5000 m deep and located in Ukraine, and the Tyrnauz borehole, 4001 m deep and located between the Black Sea and the Caspian Sea. These boreholes all penetrate crystalline formations, but major differences are found when their tectonic environments are compared. Excluding the uppermost horizon affected by surface phenomena, data do not indicate any general correlation between depth and the state of rock fracturing, which is instead governed by site specific, lithological and tectonical factors. This applies also to fracture zones, which are found at similar frequencies at all depths. As opposed to the structural data, the hydrogeological and hydrochemical information reveals a vertical zonation, with clear similarities between the three boreholes. An upper zone with active circulation and fresh or slightly mineralized groundwaters reaches down 1000-2000 m.

The interval from 1000-2000 m down to 4000-5000 m can be characterized as a transition zone with lower circulation rates and gradually increasing mineralization. Below 4000-5000 m, strongly mineralized, stagnant, juvenile or metamorphogenic waters are found. Geothermal data verify the existence of this zonation.

SKB Technical Report 92-40

PASS-Project on Alternative Systems Study. Performance assessment of bentonite clay barrier in three repository concepts: VDH, KBS-3 and VLH

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December 1992

ABSTRACT

The three repository concepts VDH, KBS-3 and VLH have been investigated with respect to their functions in short- and long-term perspectives. The study shows that while KBS-3 does not require development of new techniques for excavation and application of buffers and canisters, such development is needed for VLH and VDH. The various physical processes in the deployment part of VDH are more critical and less understood than those in KBS-3 and VLH, but the sealing effect of the plugged "low-temperature" part is sufficiently good to make the concept qualify as a candidate. VLH has the highest and KBS-3 the lowest temperature and the latter has the highest potential for good long-term function.

SKB Technical Report 92-41

Buoyancy flow in fractured rock with a salt gradient in the groundwater. A second study of coupled salt and thermal buoyancy

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November 1992

ABSTRACT

An underground nuclear waste repository produces heat that will induce a buoyancy flow of groundwater in fractures and other permeable regions in the surrounding rock. The radioactive material may then, in case of penetrated canisters, possibly reach the biosphere. Measurements of ground water in crystalline rock show an increasing salt content with depth. The resulting increase of water density counteracts the thermal buoyancy, and it may create a natural barrier for the groundwater flow between the repository and the biosphere.

The aim of the study is to analyse this barrier effect and to assess the maximum upward displacement of water starting from the vicinity of the repository. The coupled flow process for groundwater, salt and heat with buoyancy due to temperature and salt concentration differences is studied. The equations have been analysed in great detail, and a numerical model has been developed for the case of groundwater flow in a fracture plane.

The largest upward displacement from the repository has been determined with the model for any heat release. Approximate formulas, which are shown to be sufficiently accurate for assessments, have been derived. The main formula concerns the case, when the canisters are stacked on top of each other in a very deep borehole. There are no restrictions on the position of the fracture plane. The borehole may even lie directly in the fracture plane.

We find a strong barrier effect. In a reference case with a salt concentration increase of 2% per km downwards and with 300 canisters placed over a length of 2000 m in the borehole (the total amount of released heat is 0.32 TWh), the largest upward displacement from the top of canisters becomes, according to the formula, 60 m. The case, when the fractured rock is considered as a homogeneous porous medium, is also dealt with. The groundwater flow is then three-dimensional. The largest upward displacement now becomes 67 m for the reference case.

The main formula shows that the barrier effect is remarkably insensitive to variations of the involved parameters. A change of salt gradient, or total amount of released heat, by a factor 10 causes a change by 10 = 3.2 of the upward displacement.

The parameters that do not enter into the formulas are noteworthy. The hydraulic conductivity of the flow plane, which is the most uncertain of all parameters, does not matter in the balance between thermal buoyancy and counteracting salt buoyancy.

SKB Technical Report 92-42

Project on Alternative Systems Study – PASS. Comparison of technology of KBS-3, MLH VLH and VDH concepts by using an expert group

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September 1992

ABSTRACT

This report constitutes a technical comparison and ranking of four repository concepts for final disposal of spent

nuclear fuel, that have been studied by SKB: KBS-3, Medium Long Holes (MLH), Very Long Holes (VLH) and Very Deep Holes (VDH). The technical comparison is part of the project "Project on Alternative Systems Study, PASS", which was initiated by SKB /with the objective of presenting a ranking of the four concepts. Besides this comparison of Technology the ranking is separately made for Long-term Performance and Safety, and Costs before the merging into one verdict.

The ranking regarding Technology was carried out in accordance with the method Analytical Hierarchy Process, AHP, and by the aid of expert judgement in the form of a group consisting of six experts. The AHP method implies that the criteria for comparison are ordered in a hierarchy and that the ranking is carried out by pairwise comparison of the criteria. In the evaluation process a measure of the relative importance of each criterion is obtained.

The result of the expert judgement exercise was that each expert individually ranked the four concepts in the following order with the top ranked alternative first: KBS-3, MLH, VLH and VDH. The common opinion among the experts was that the top ranking of KBS-3 is significant and that the major criteria used in the study could change substantially without changing the top ranking of KBS-3.

SKB Technical Report 92-43

Project on Alternative Systems Study – PASS. Analysis of performance and long-term safety of repository concepts

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ABSTRACT

This study is a part of the Project on Alternative Systems study, PASS, with the overall aim to perform a technical/economical ranking of alternative repository concepts and canisters for the final storage of spent nuclear fuel. The comparison should in the first stage separately assess Technology in construction and operation, long-term Performance and Safety, and Costs.

Three of the repository concepts are assumed to be located at a depth of approximately 500 m in the host rock, KBS-3, Very Long Holes (VLH) and Medium Long Holes (MLH). In the KBS-3 concept the canisters are deposited in vertical deposition holes in a system of parallel storage tunnels. In the VLH concept larger canisters are deposited in long horizontal tunnels. The MLH concept, is an evolution of the two other concepts, with KBS-3 type canisters deposited in horizontal tunnels. Smaller canisters are to be deposited in deep bore holes at a depth between 2000 to 4000 m in the Very Deep Holes (VDH) concept. In all concepts the canisters will be surrounded by a bentonite buffer.

The aim of the present study is to analyze and compare the performance and long-term safety of the repository concepts. Only a qualitative comparison of the concepts is made as no calculations of radionuclide releases or dose to man have been performed. The ranking of the repository concepts is based on differences in the performance of various mechanisms for isolation or retention as well as on the performance of the integrated repository barrier system. To evaluate and compare the performance of the repository systems a set of comparison criteria has been used. The comparison considers uncertainties associated with the estimates of doses, the importance of individual mechanisms and processes for the overall performance, the possibilities to validate and demonstrate the longterm performance and the sensitivity to rare events.

The ranking of the repository concepts was carried out by comparing the VDH, VLH and MLH concept with the KBS-3 concept. The performance and long-term safety of the repositories located at 500 m level will be based on a multiple barrier system and the predictions for the concepts will involve similar uncertainties. The differences that have been identified in the performance of the barriers are not distinguishing any of the concepts as more favourable than the others with respect to overall safety. The conclusion is also that all the 500 m level alternatives can be expected to comply with strict requirements concerning the long-term safety. The performance and long-term safety of the VDH concept is determined by the function of only one individual barrier and does not fulfil the multiple barrier principle in the same way as the other concepts.

SKB Technical Report 92-44

Project on Alternative Systems Study – PASS. Cost comparison of repository systems

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ABSTRACT

The development of alternative repository systems and different canister alternatives is being studied by SKB within the frame of the "Project on Alternative Systems Study, PASS", with the objective of presenting a ranking of systems being currently studied. The ranking is primarily made for three different headings: technology, long-term performance and safety, and costs. The rankings for each of these headings are eventually to be merged into one ranking. The present report presents the basis for the ranking regarding costs. The following four systems have been studied and are presented in order of cost (the less

expensive first): Medium Long Holes (MLH), KBS-3 (modified Plan 92 system), Very Long Holes (VLH), and Very Deep Holes (VDH). A significant outcome of the study was the clear difference in cost between the very expensive system VDH and the other three.

SKB Technical Report 92-45

Mechanical integrity of canisters

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December 1992

ABSTRACT

This document constitutes the final report from "SKB's Reference Group for Mechanical Integrity of Canisters for Spent Nuclear Fuel". A complete list of all reports initiated by the Reference Group can be found in the Summary Report in this document. The main task of the Reference Group has been to advice SKB regarding the choice (ranking of alternatives) of canister type for different types of storage. The choice should be based on requirements of impermeability for a given time period and identification of possible limiting mechanisms.

The main conclusions from the work were:

From mechanical point of view, low phosphorous oxygen free copper (Cu-OFP) is a preferred canisters material. It exhibits satisfactory ductility both during tensile and creep testing. The residual stresses in the canisters are of such a magnitude that the estimated time to creep rupture with the data obtained for the Cu-OFP material is essentially infinite. Based on the present knowledge of stress corrosion cracking of copper there appears to be a small risk for such to occur in the projected environment. This risk needs some further study. Rock shear movements of the size of 10 cm should pose no direct threat to the integrity of the canisters. Considering mechanical integrity, the composite copper/steel canister is an advantageous alternative.

The recommendations for further research included continued studies of the creep properties of copper and of stress corrosion cracking. However, the studies should focus more directly on the design and fabrication aspects of the canister.