SKB TECHNICAL REPORT 85-01

ANNUAL RESEARCH AND DEVELOPMENT REPORT 1984

Including Summaries of Technical Reports Issued during 1984.

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SVENSK KÄRNBRÄNSLEHANTERING AB

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PREFACE

The year 1984 was a very important year for the waste management program in Sweden. The fuel loading permits were granted by the Swedish government for the 11:th and 12:th reactors in the nuclear power program. This means that the main original goals of the KBS-project, later SKBF/division KBS, have been reached. In the meantime new assignments and duties have been placed on SKB by the nuclear power utilities. These assignments have their bases in the revised and partly new legislation in the nuclear field which was passed by the Swedish parliament early 1984. As a consequence of these events the organization of SKB has been restructed effective Jan. 1, 1985 (See Chapter 1). The research and development work that previously was started and carried on by the KBS-project and SKBF/division KBS will be continued by SKB division of Research and Development. The work will be reported in SKB Technical Reports which continues the previous KBS Technical Reports Series. This Annual Report 1984 is the first in the new series.

SWEDISH NUCLEAR FUEL AND WASTE MANAGEMENT CO.

Division of Research and Development

Per-Eric Ahlström

ABSTRACT

This is the annual report on the activities of the research and development division (formerly KBS-division) of the Swedish Nuclear Fuel and Waste Management Company, SKB. It contains background information on the Swedish nuclear waste management system. The research and development program of SKB is presented and a review is made of progress during 1984 in different areas of the program.

Lectures and publications during 1984 are listed in an appendix. The report also contains the summaries of all technical reports issued during 1984.

CONTENTS

1 BACKGROUND

1.1 THE SWEDISH NUCLEAR POWER PROGRAM

During 1984 10 reactors were in operation generating 41% of the total Swedish electric power production. The present Swedish nuclear power program is limited to 12 reactors in total according to a decision by the Swedish Parliament in 1980. The last two reactors in this program, Forsmark 3 and Oskarshamn 3, were granted fuelling permits in 1984 and they will be taken into commercial operation 1985.

1.2 THE SWEDISH NUCLEAR WASTE MANAGEMENT SYSTEM

Residues generated by operation of the reactors are spent nuclear fuel and different kinds of low- and medium level reactor wastes.

Furthermore, in the future, decommissioning waste will be generated when old reactors are decontaminated and dismantled.

The types and total quantities of various nuclear waste categories currently of relevance in Sweden are given in Table 1-1. The estimation of the waste quantities is based upon the assumption that no reactor in the program will be operated after the year 2010.

The quantity of spent fuel is estimated to about 7 500 metric tons. Approximately 12% of this amount or about 870 tonnes is covered by foreign reprocessing contracts. The present spent fuel management policy is, however, in favour of long-term intermediate storage with the aim of the direct disposal as the prefered option.

From the spent fuel which may be reprocessed, vitrified high-level waste and transuranic low- and medium level wastes will be

Swedish reactors

Reactor		Power MW_e	Commercial operation	Energy Availability in 1984 %
Oskarshamn 1	BWR	440	1972	82
Oskarshamn 2	BWR	580	1974	93
Oskarshamn 3	BWR	1050	1985	–
Barsebäck 1	BWR	580	1975	88
Barsebäck 2	BWR	580	1977	86
Ringhals 1	BWR	760	1976	80
Ringhals 2	PWR	820	1975	67
Ringhals 3	PWR	915	1980	72
Ringhals 4	PWR	915	1982	78
Forsmark 1	BWR	900	1980	92
Forsmark 2	BWR	900	1981	81
Forsmark 3	BWR	1050	1985	–

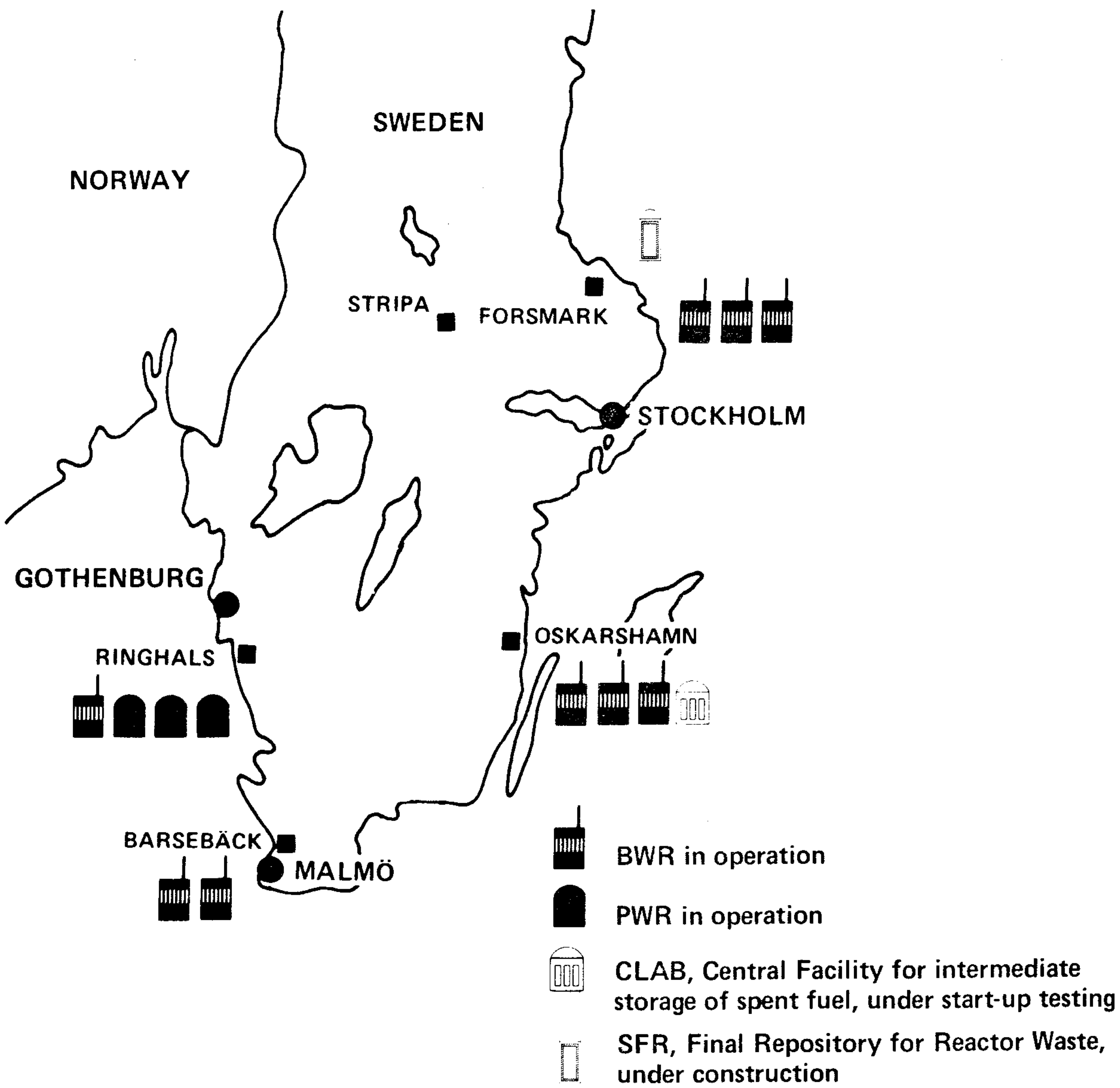

Figure 1-1. The Swedish nuclear power program.

Table 1-1. Waste categories

Waste category	Origin	Waste form	Properties	Quantity
1a Spent fuel	Operation of nuclear reactors	Fuel rods encapsulated in copper canisters	High heat flux and radiation at first. Contains long-lived nuclides	4 840 canisters
1b High-level waste	Residual products from reprocessing	Vitrified waste encapsulated in lead-titanium canisters	High heat flux and radiation at first. Contains long-lived nuclides	730 canisters
2 Transuranic-bearing waste	Waste from the reprocessing process	Solidified in concrete or bitumen	Low- to medium-level. Contains long-lived nuclides	4 500 m^3
3 Core components and internals	Scrap metal inside reactor tanks	Untreated or cast in concrete	Low- to medium-level. Contains certain long-lived nuclides	15 000 m^3
4 Reactor waste	Operating waste from nuclear power plants etc.	Solidified in concrete or bitumen. Compacted waste	Low- to medium-level. Limited life time	100 000 m^3
5 Dismantling waste	From dismantling of nuclear facilities	Untreated for the most part	Low- to medium-level. Limited life time	114 000 m^3

generated. These wastes may be sent back to Sweden for storage and disposal.

The main features of the planned system for nuclear waste management in Sweden are shown in Figure 1-2.

The following phases of the system are now being implemented.

* A central facility for intermediate storage of spent fuel from all Swedish reactors, CLAB, has been constructed and is scheduled to be operative in 1985. The facility consists of underground storage pools for 3 000 tonnes of fuel in a first phase which can be extended as needed.

* A central final repository for low- and medium level reactor waste from all Swedish reactors, SFR, is under construction and is planned to be operative in 1988. Radioactive wastes from industry, research and medicine will also be disposed of in SFR.

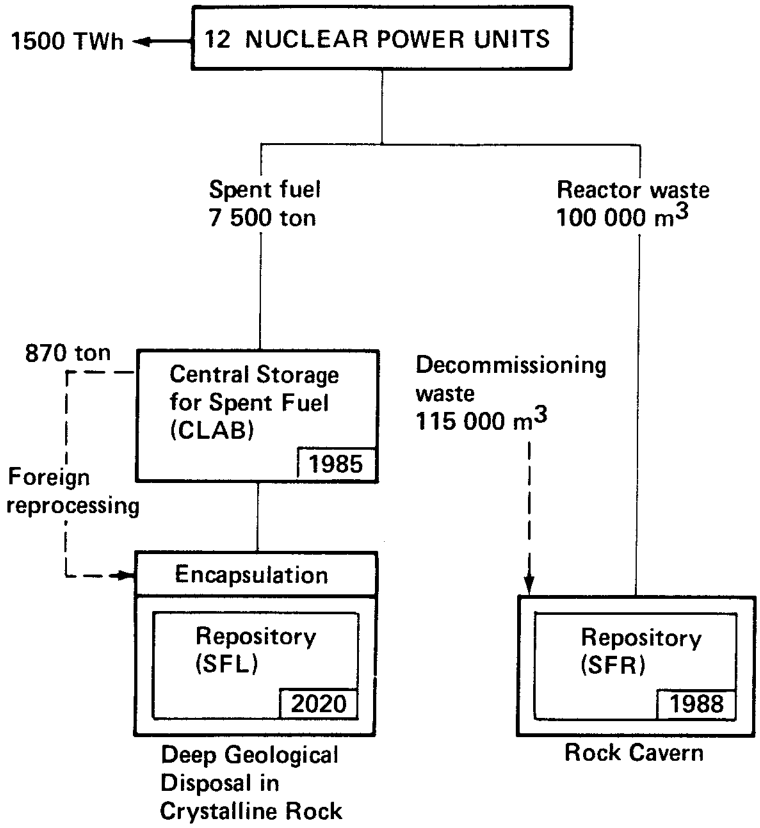

Figure 1-2. The Swedish nuclear waste management system.

* A sea transportation system for spent fuel of all kinds of nuclear wastes has been built. A special ship and transport casks for spent fuel were commissioned in 1982 and has been used for transportation of spent fuel to the reprocessing plant at la Hague in France. The ship named SIGYN will now be used for transportation along the Swedish coast of spent fuel from the reactors to the central storage facility, CLAB, and from 1988 on, of reactor wastes to the final repository for reactor waste, SFR.

* Research and development work on different aspects of final disposal of spent fuel and long-lived wastes is going on. These studies include the scientific basis as well as site investigations and system analysis.

The facility for intermediate storage of spent fuel (CLAB) constitutes a fundamental strategic function in the Swedish spent fuel management system. It will secure the uninterrupted nuclear power production and it will provide ample time for the R&D-work, the site selection and the system's design and optimization needed to achieve a suitable and safe complete system for spent fuel management. That includes also freedom of choice on the option to be finally implemented.

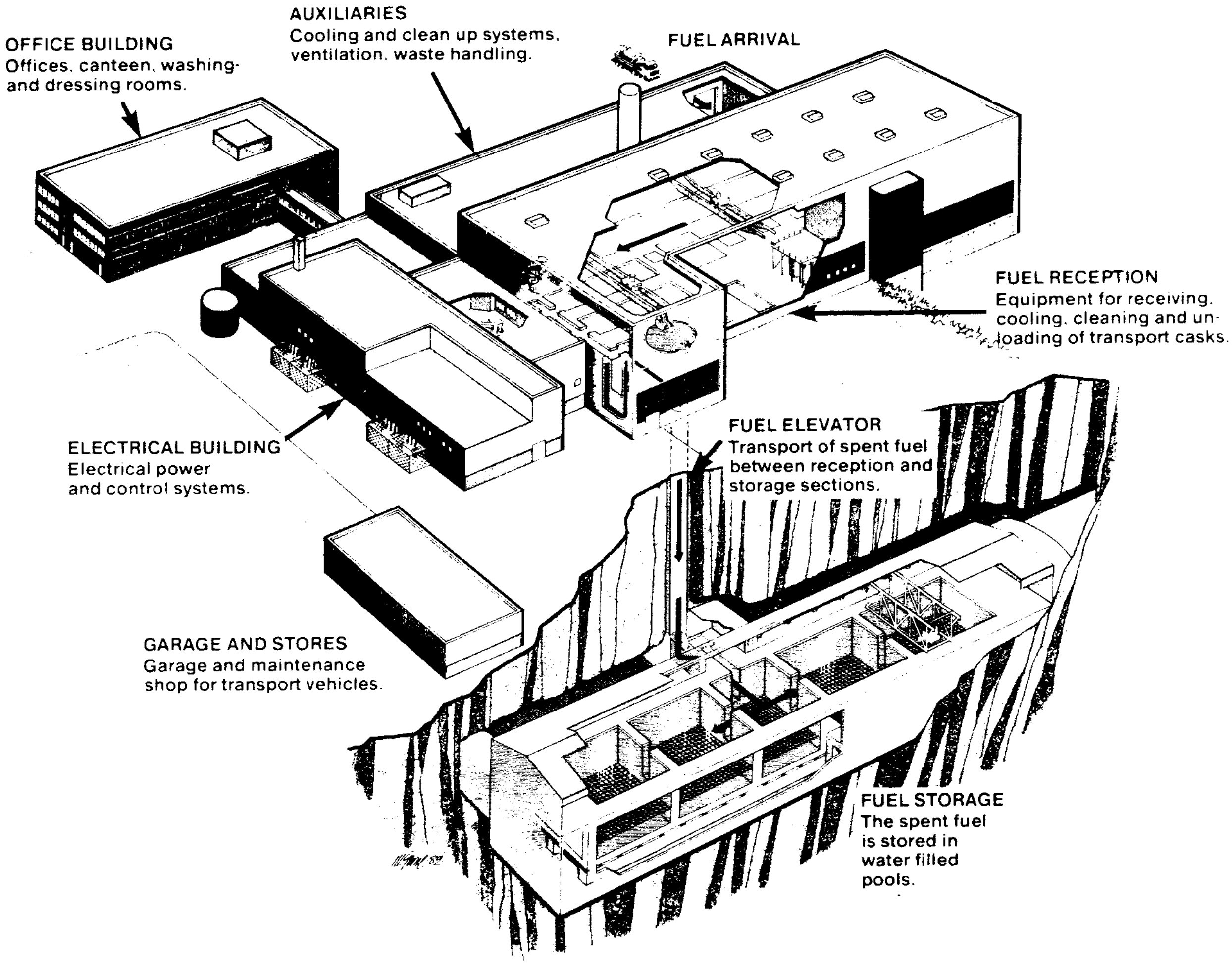

Figure 1-3. General layout of CLAB, Central Facility for Intermediate Storage of Spent Fuel, presently under start-up testing at Oskarshamn.

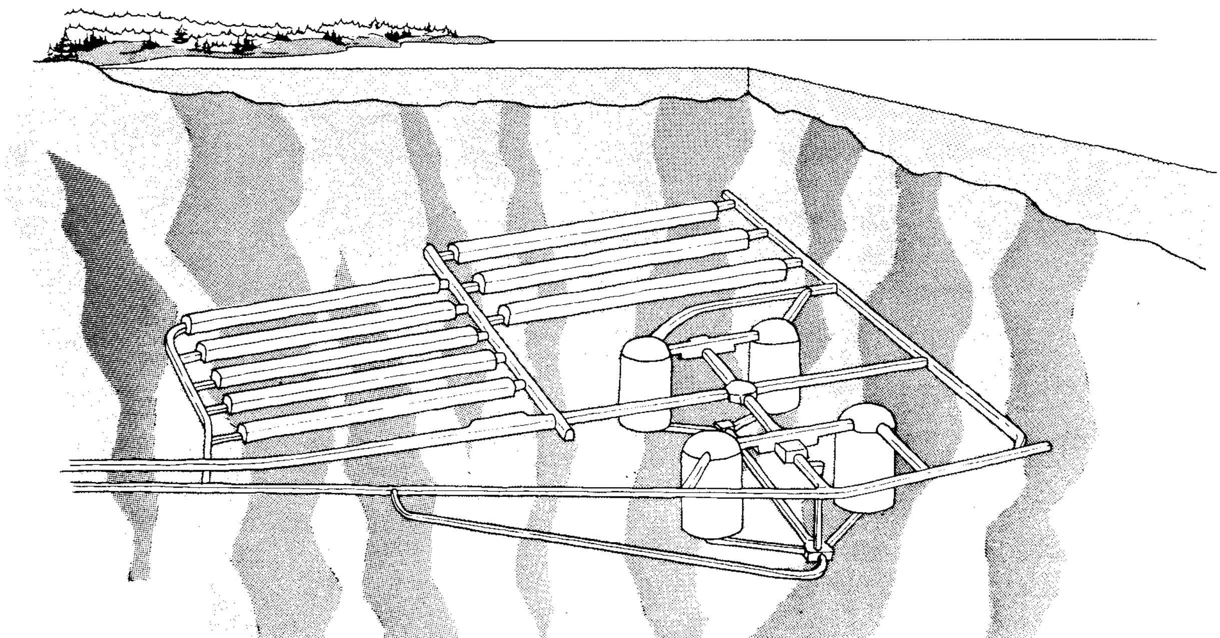

Figure 1-4. General Layout of SFR, Final Repository for Reactor Waste, presently under construction at Forsmark.

1.3 LEGAL AND ORGANIZATIONAL FRAMEWORK

A new Act on Nuclear Activities /1-1/ entered into force February 1, 1984. Together with the Act on Financing of Future Expenses ... and the Radiation Protection Act it gives a coherent and comprehensive legal system for the nuclear waste management activities in Sweden.

In the law it is stated that the producer of the waste is primarily responsible for its handling and final disposal. All costs for nuclear waste management including decommissioning and dismantling of nuclear installations shall be covered by special governmental funds built up through a fee on the nuclear energy produced. In 1984 this fee was set by the government at 0.019 SEK per kWh which is roughly equivalent to 2.1 US mills per kWh.

Thus the Swedish legislation explicitly puts the primary responsibility for the management of the radioactive waste on the nuclear power utilities. The role of the national authorities is mainly to supervise and control that the activities are sufficient, relevant and in accordance with legal requirements. The four utilities Vattenfall (The Swedish State Power Board), Sydkraft, OKG and Forsmarks Kraftgrupp (FKA) have delegated the execution of the activities in this field to the jointly owned Swedish Nuclear Fuel and Waste Management Co, SKB*. The responsibilities of SKB in the nuclear waste management field thus include

- all necessary research and development work,
- planning and cost calculations for the total nuclear waste management system (except handling and treatment at the reactor sites),
- design, construction and operation of all necessary facilities for storage and disposal of nuclear waste.
- all transport and handling of spent nuclear fuel outside the reactor sites.

SKB is a company, with approximately 40 employees, for management of the Swedish nuclear waste program. The bulk of the work is made by contractors. SKB has been reorganized during 1984. The new organization effective Jan. 1, 1985 is shown in Figure 1-5.

Mainly three national authorities deal with matters connected to radioactive waste management:

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* Before July 1, 1984 - Swedish Nuclear Fuel Supply Co, SKBF.

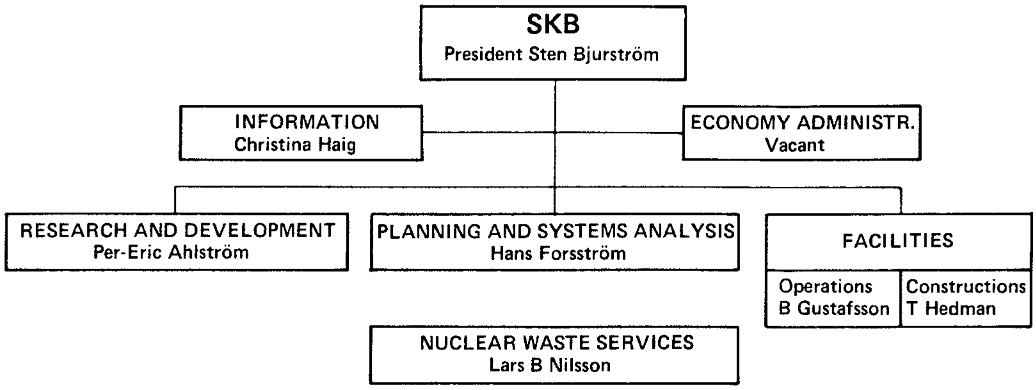


Figure 1-5. SKB organization as from Jan 1, 1985.

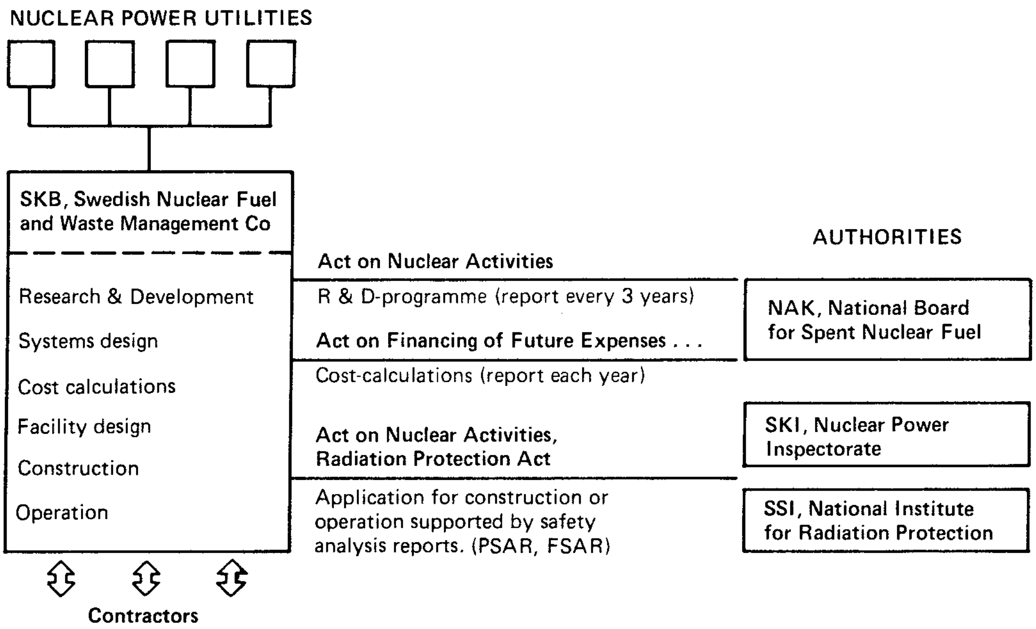


Figure 1-6. Legal framework for the activities of SKB.

SKI, The Swedish Nuclear Power Inspectorate, supervises and controls the safety at design, construction and operation of nuclear facilities.

SSI, The Swedish Radiation Protection Institute, supervises and controls that appropriate measures for radiation protection are taken by the utilities.

NAK, The National Board for Spent Fuel, supervises the planning and the research and development for the waste management program. NAK also administers the funds for future costs for radioactive waste management.

The legal responsibilities of SKB and the three national authorities are illustrated in Figure 1-6.

1.4 RESEARCH AND DEVELOPMENT PROGRAM

1.4.1 General

The management of the R&D work is made by a special division of SKB; Division of Research and Development (formerly division KBS). The scheme in Figure 1-7 illustrates the principle for the handling of assignments within this division. The main task of the staff is the initiation, planning and coordination of the work, the compilation and documentation of results and the practical application of the measures.

This is done in close cooperation with consultants and experts performing the investigations and experiments.

In order to get a thorough discussion of results and methods and a constructive feed-back to the program, the results are continuously published in the KBS technical report series and in appropriate scientific publications.

The start of the R&D-work goes back to 1976-1977 when a new law, the Stipulation Act (replaced in February 1984 by the Act on Nuclear Activities), specific to the final disposal of high-level radioactive waste was taken by Parliament. It stipulated that the reactor owner must, prior to the loading of a new reactor with fuel, show to the satisfaction of the government how and where high-level waste or spent fuel can be finally stored in a safe manner. In other words, this act required demonstration of the feasibility of a safe disposal but it did not ask for an optimized solution.

In 1979 and 1980, four reactors were granted fuel loading permits according to the Stipulation Act. The permits were based on a re-

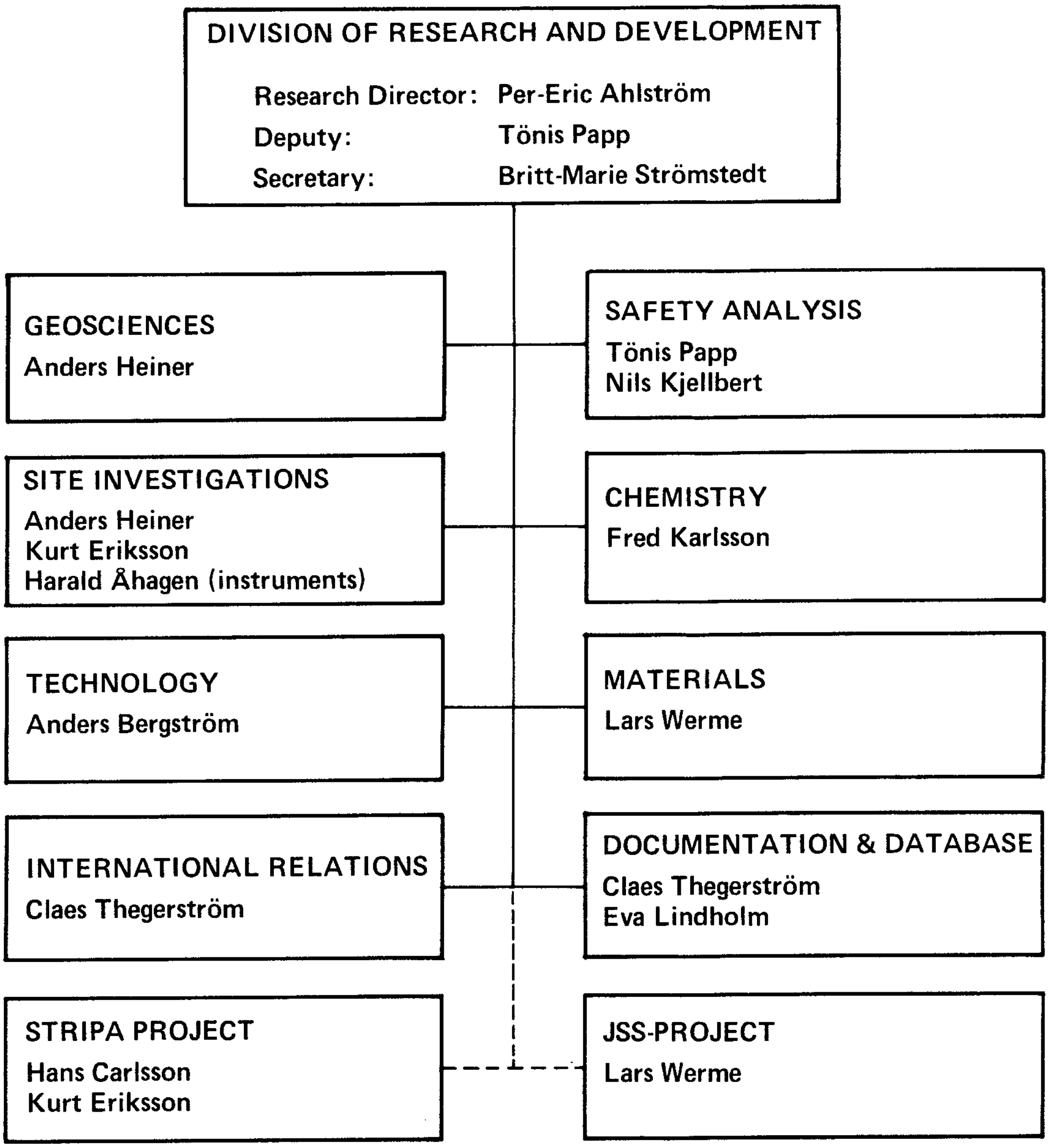


Figure 1-7. Organization of the R & D-division within SKB

Staff-Changes during 1984

1. Lars B Nilsson retired from the division KBS on May 1 and was replaced by Per-Eric Ahlström. The name of the KBS division has been changed into Division of Research & Development.
2. Britt-Marie Strömstedt succeeded Lilian Kumlin-Blomkvist on May 1.
3. Eva Lindholm started her work at SKB in June.
4. Anders Heiner started his work at SKB in October.
5. Kurt Eriksson started his work at SKB in November.
6. Åke Nässil finished working for SKB in December.
7. From May 1, Lars B Nilsson, Lilian Kumlin-Blomkvist, Harald Åhagen and Hans Forsström work for the new NWS group formed within SKB.
8. Hans Forsström was appointed general manager of the Div. of Planning and System from Jan. 1, 1985.

port (the KBS-1-report) describing how and where the vitrified high-level waste from reprocessed spent nuclear fuel could be finally disposed of, and on a contract for a limited amount of nuclear fuel to be reprocessed by COGEMA at la Hague in France.

In parallel with the studies underlying the KBS-1-report /1-2/ on final disposal of vitrified HLW, work also started on the final disposal of unreprocessed spent fuel. A first report (the KBS-2-report) on this option was presented in 1978. In May 1983 a substantially extended and updated report, the KBS-3-report /1-3/ was presented to the Swedish Government as a basis for an application to start up the two last reactors in the Swedish twelve-reactor program. The application was approved by the Swedish Government in June 1984. As a background for their decision the Government had asked for an extensive review by both Swedish and foreign bodies and Specialists /1-4/.

1.4.2 Long-range program

Even if the KBS-reports have shown that a safe final storage of long-lived radioactive waste can be effected with present-day technology and the geological conditions existing in Sweden, considerable work remains to be done to show, how these measures are to be realized in detail and in an optimal way.

The future work for realization and optimization of a safe system for final disposal of nuclear waste will comprise

* Continued research and development work in order to further deepen the scientific kowledge, that constitutes the base for the performance and safety assessment.

* Studies and evaluation of alternatives to the methods and concepts investigated so far.

* Optimization of systems as regards technology, economy and use of resources against the improved scientific knowledge.

* Investigations for site-selection.

A basis for the planning of the R&D-work is the overall timetable for realization of a final repository for spent fuel, see Figure 1-8. This timetable is based upon the present KBS-3-concept for management and final storage of spent fuel, which includes a 40-year´s intermediate storage of spent fuel before disposal. Construction of the final repository is planned to start at about 2010. The time available until then will be used for R&D-work, optimization and design, site-investigations and site-selection as indicated in Figure 1-8. An application for licensing of a repository is foreseen around the year 2000.

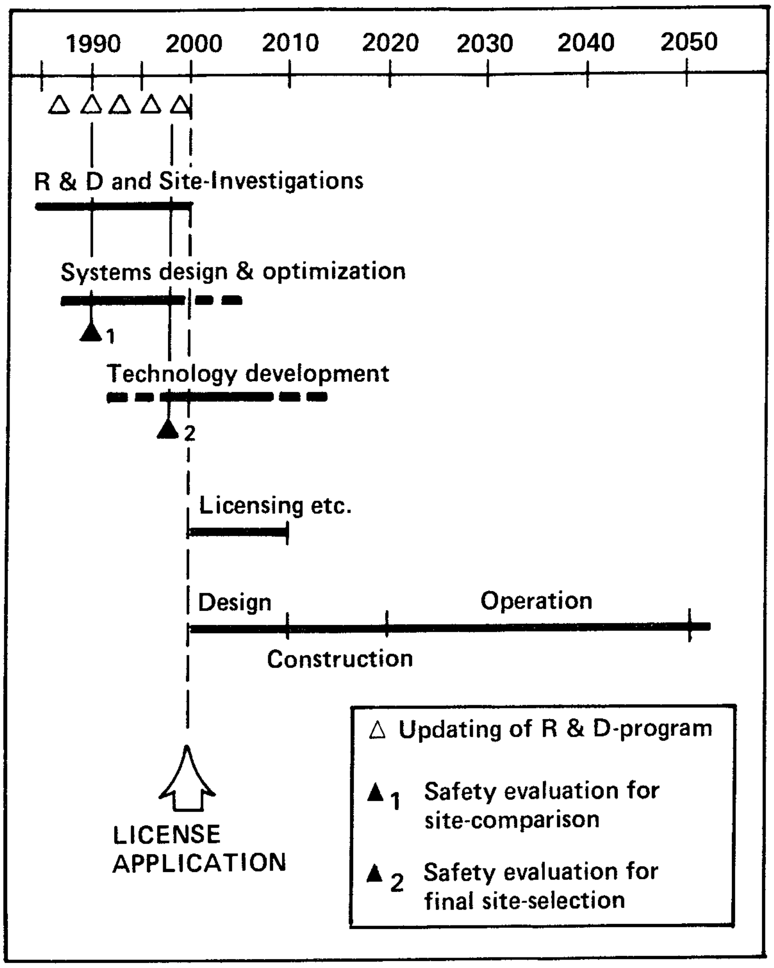


Figure 1-8. General time-table for realisation of a final repository for spent fuel.

Every third year the R&D-program, studies of alternative options included, will be updated and sent to the authorities for review as required by law. It is also planned that approximately every six year a comprehensive safety evaluation will be done. It is considered that such an evaluation is an important tool for making priorities in the R&D-program, for systems optimization and for the site-evaluation/site-selection process.

The choice of a site for a final repository for high-level waste can be of technical-economic importance and is also of great political and public interest. Extensive geological investigations are therefore made at many places in the country, so that a broad and reliable body of data will be available by the end of the century, see Chapter 5.

1.4.3 Short-range program

In 1984 a program for continued R&D was prepared /1-5/. It was based on the experiences from the feasibility study presented in KBS-3 and the comments received in the review of that report.

The main activities included in this program are summarized in Appendix 1.

During the next few years the program will be revised to cover also other possible designs for the final handling and storage of spent fuel. This revised program will be presented to the Government in September 1986.

During the 1980's the emphasis will be placed on the R&D necessary to provide a basis for the site comparison planned for the turn of the decade.

1.5 INTERNATIONAL COOPERATION

Cooperation and exchange of information with foreign and international organisations has continued during 1984. Among other things, staff members of SKB and experts engaged by SKB have participated in activities within the IAEA and the OECD/NEA.

International development in the field has been followed through participation in a number of conferences, where Swedish papers have also been presented, see Appendix 1.

The SKB-managed multinational OECD/NEA project at Stripa has progressed according to the schedule and a phase 2 of the project has been started. The following countries are now participating in the Stripa Project: Canada, Finland, France, Japan, Spain, Sweden, Switzerland, United Kingdom and the United States. See further Chapter 8.

An additional phase IV has been decided for the JSS project jointly sponsored by CRIEPI, Japan, NAGRA, Switzerland and SKB, Sweden, with SKB as the managing party. The project includes investigations an high-level radioactive glass obtained from France. See further Chapter 2.

SKB has bilateral information exchange agreements with USDOE, AECL-Canada, NAGRA-Switzerland and CEA-France. A similar agreement with Euratom is being discussed.

Information exchange without formal agreements has been made with organisations in the Federal Republic of Germany, Belgium, United Kingdom, Japan and the Nordic countries.

SKB is participating in the international HYDROCOIN-project which is coordinated by a secretariate set up by the Swedish Nuclear Power Inspectorate.

Together with US, Swiss and Brazilian organisations, SKB support studies of the large thorium mineralization at Morro do Ferro in

Brazil. Also other studies on natural analogues are made in international cooperation, see further Chapter 4. Together with USDOE SKB arranged a workshop on natural analogues in Chicago, October 1984.

2 MATERIALS

2.1 SPENT FUEL

Of the two candidate wasteforms, presently being considered by SKB, the main thrust during 1984 has been on spent fuel. As during previous years, the Swedish work has been made in close contact with groups in Canada and United States. A fourth workshop in the series of spent fuel workshops was organized in August 1984 in Canada.

The experiments carried out at Studsvik during 1984 have to one part been a continuation of the ongoing series started in 1982, where total contact times to up to over 900 days now have been achieved. In addition to these experiments, special studies of solubility constraints and the effects of alpha radiolysis have been started.

Throughout the experimental series, the procedure for sample handling and analyses previously described, has been adhered to /2-1, 2-2/.

As before, most of the experiments were carried out under oxidizing conditions, but for the alpha-radiolysis series, reducing conditions were also imposed, adopting the procedure previously described /2-3/. In addition to this method of achieving reducing conditions, tests using crushed granite as reductant have been made during 1984 and will be employed in forthcoming experiments in reducing environments.

2.1.1 Solubility constraints

The concentrations of U and Pu in the filtered solutions, very rapidly seem to attain an apparent solubility limit and stay constant irrespective of contact time (see Figures 2-1 and 2-2) /2-3/. Indeed, for Pu there is even some evidence for the concentrations going down with extended contact times. When subjecting the

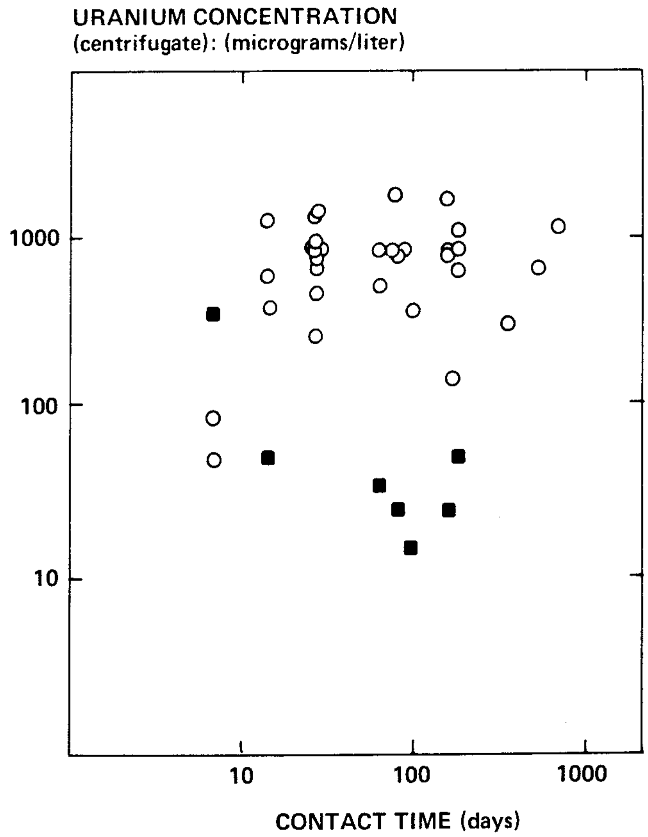


Figure 2-1. Measured concentrations of U at different contact times. Open rings: exposure in synthetic groundwater at pH 8.2 Solid squares: exposure in DI H_2O (pH 7.0).

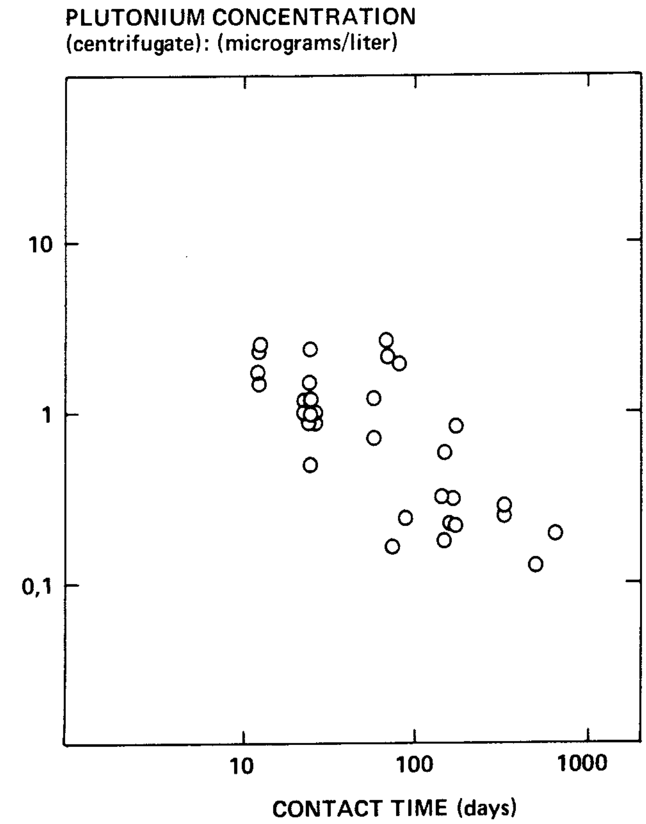


Figure 2-2. Measured concentrations of Pu at different contact times.

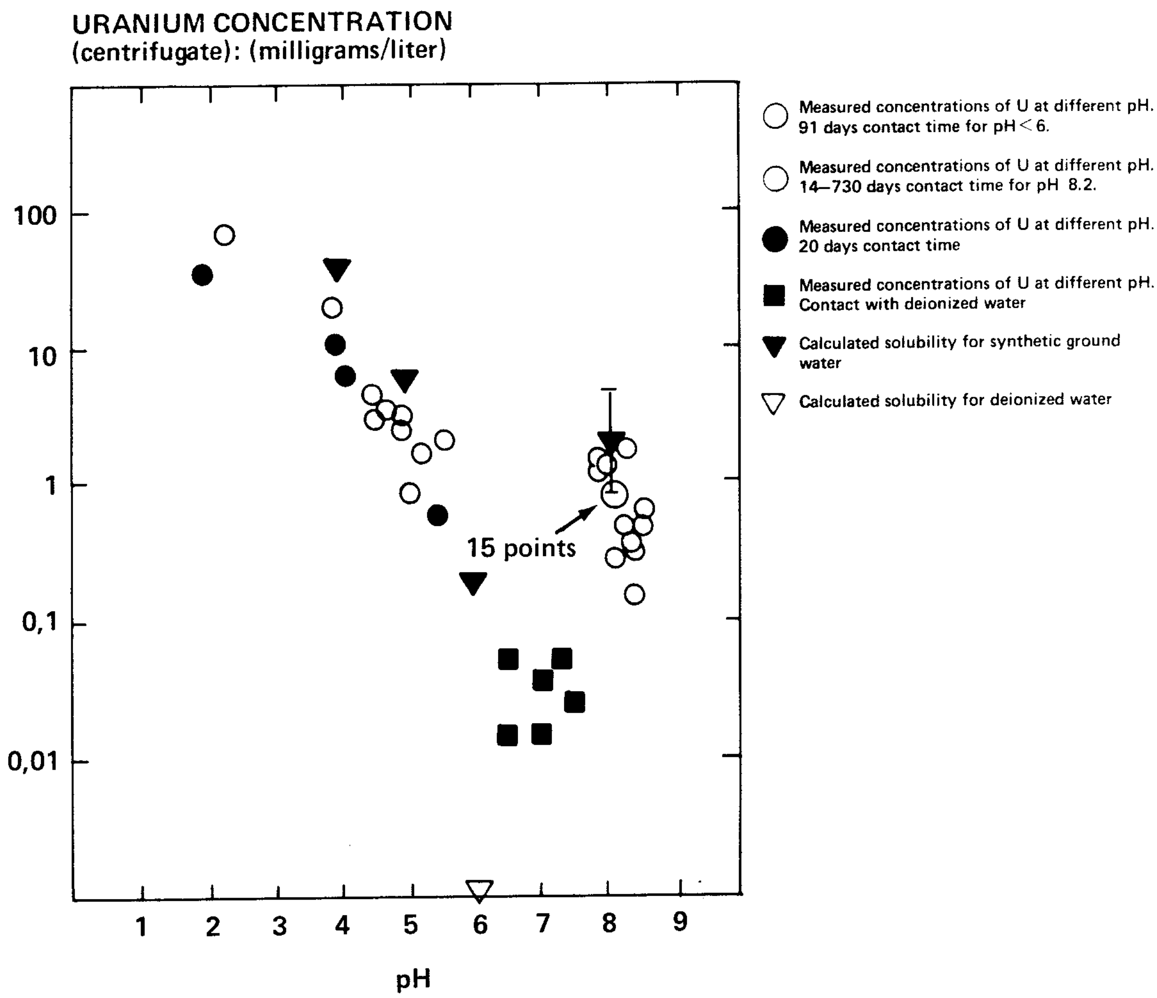


Figure 2-3. Measured concentrations of U at different pH. Open rings: In the acidic range, 91 days exposure. At pH 8.2 14 days to 2 years. Solid rings: 20 days exposure. Solid squares: Exposure in DI H_2O. Solid triangles: Calculated solubility in synthetic groundwater. Open triangle: Calculated solubility in DI H_2O.

specimens to leach solutions with lower pH values, the resulting concentrations plotted versus pH diagrams indicating solubility control of the uranium release.

Using the database for uranium developed at the Royal Institute of Technology, Stockholm, the experimentally determined equilibrium concentrations were compared with calculated solubilities. As can be seen in Figure 2-3, the agreement between the experiments and the calculations is very good. It is therefore reasonable to assume that the release of uranium from the spent fuel is indeed controlled by solubility constraints. Furthermore, the determined apparent solubilities, as well as the calculated ones are far lower than those conservatively used in the KBS-3 report.

For Pu, the agreement between the experiments and the calculations were poorer. However, much of the discrepancies observed may well be from uncertainties in the plutonium data base.

2.2 WASTE GLASS

2.2.1 Simulated inactive glass

Most of the ongoing program for studies of simulated inactive glass has been completed and reported. /2-4, 2-5/. At present, the largest part of these studies are conducted in support of the JSS-program.

The Stripa burial experiments using the glass compositions ABS 39 and 41 are continuing and exposure times of up to 3 years will be evaluated.

2.2.2 JSS-project

The JSS-project is a joint project between SKB, NAGRA (Switzerland) and CRIEPI (Japan).

The Phases I and II of the JSS-project have been completed and final reports of Phase I have been issued /2-6, 2-7, 2-8/.

In Phase I, it has been established that the radioactive glass behaves similarly to the non-radioactive ABS-118 simulated waste glass, i e the presence of radionuclides do not significantly affect the leach behaviour of the glass. Furthermore, solution analyses and SIMS profiles indicate matrix dissolution followed by precipitation to be the leaching mechanism.

2.3 CANISTER MATERIALS

2.3.1 Copper

Most of the studies on copper during 1984 have been directed along the lines suggested by the KBS-3 reviewers. The most important programs are studies on the creep properties of pure copper and inorganic reduction of sulphates to sulphide at elevated temperatures in the presence of copper.

These programs are by the nature of the phenomena studied running over several years and no data are available as yet.

2.3.2 Alternative canister materials

Studies on alternatives to copper as canister material are foreseen for the next several years. During 1984 some preliminary planning for these studies have started.

3 ENGINEERED BARRIERS, DESIGN AND TECHNOLOGY

3.1 CANISTERS

Two alternative designs of thick walled copper canisters have been studied; one with filling of lead and sealed by use of electron beam welding and the other with filling of copper powder which is solidified by use of hot isostatic pressing technique.

3.1.1 Welded copper canister

Earlier tests show that the open space in a canister can be filled up by lead to more than 98%, giving a final void of less than 2%.

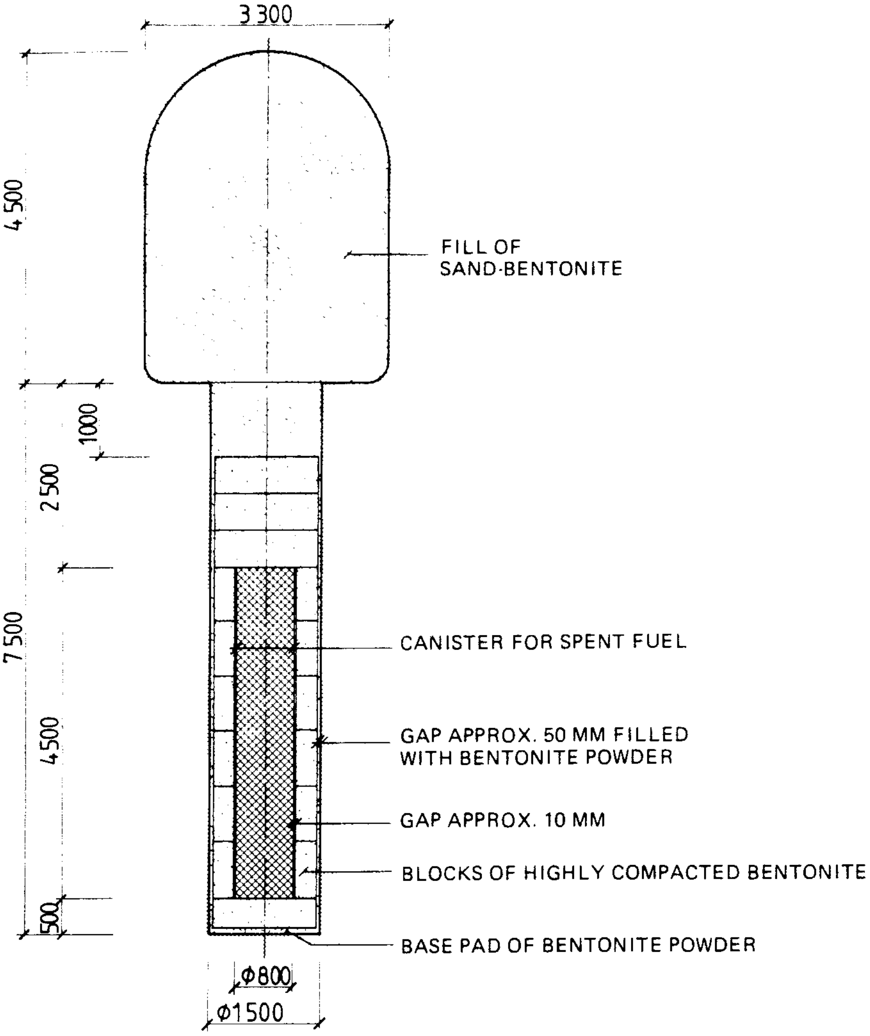
A theoretical analysis of the time dependent stresses and deformations of a copper canister when full outer pressure is applied has been made. It indicates that there will be no large elongation in the surface of the canister. The analysis is made with FEM-calculations on a simplified geometry and it is based on preliminary data on the copper rheology. Further analyses are planned to be performed with more detailed geometry and based on test data.

3.1.2 Pressed canister

Some results of mechanical tests on solidified copper powder indicate possible impurities may be important to the mechanical behavior. Material research has been initiated. See Chapter 2.

3.2 REPOSITORY DESIGN

The KBS-3 repository concept includes systems of horisontal tunnels with deposition holes drilled vertically down. See Figure 3-1.



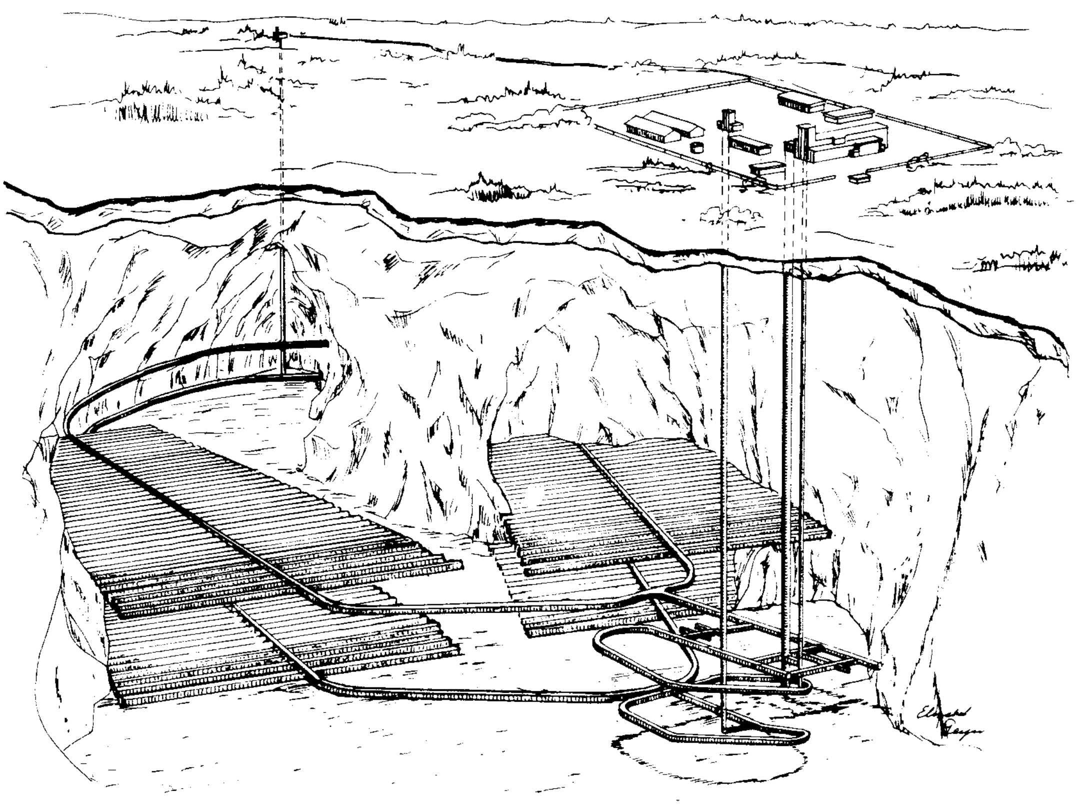


Figure 3-1. Deposition hole with canister and buffer material and with backfill in storage tunnel Tunnels and shafts in the final repository for spent fuel.

Modifications in the system layout of the tunnels should be made considering geology and temperature effects. Theoretically minimum length of tunnels will be achieved with extension in one direction. A study of a more cost effective design is running. An inventory of alternative designs is in progress.

An equipment normally used for raise boring was applied for testing downwards drilling one 1 m diameter hole without any pilot hole 15 m in hard rock. Test results indicated the need of modifications in drilling tool be made in order to gain among other things better precision and longer service life of cutters.

3.3 CLAY BARRIERS

3.3.1 General

In the concepts for final repositories for spent fuel and for reactor waste, SFR (see Chapter 9) bentonite based clay-mixtures are used as components in the multibarrier systems.

The main functional requirements of these buffer- and backfill materials are

* low hydraulic conductivity
* sufficient bearing capacity
* sufficient ductility to eliminate negative consequences of possible rock movements
* swelling potential to fill up voids
* sufficient heat conductivity
* chemical and physical stability

During 1984 the studies has focused on the physico-chemical properties of smectite clays and the mechanical interaction between the rock, the highly compacted bentonite and the canister.

Buffer and backfill materials have been studied in realistic environment in the Stripa project, see Chapter 8.

Hydraulic and gas conductivity of smectite-rich clay was tested, see Chapter 9.

3.3.2 Physico-chemical properties and chemical stability of smectite clays

In december 1983 a workshop on "Smectite Alterations" was held. The discussion centered on the mechanism for the smectite-illite conversion, time aspects and possible changes of the physical-chemical properties. During 1984 the proceedings were compiled and printed in KBS series of technical reports (KBS TR 84-11). Nuclear magnetic resonance studies on MX-80 bentonite/water/electrolyte are in progress. Preliminary results indicate that the pore-water chemistry is almost insignificant to the water mobility and consequently it is not crucial for the physical and mechanical properties of dense smectite-rich clay.

A theoretical model for stress/strain/time behavior of dense smectite clay will be tested in laboratory tests prepared during 1984. One test is a load test with simulated canister, other tests are planned to be made by use of a special equipment for shearing tests which is in preparation.

3.3.3 Sealing of boreholes, tunnels and shafts in rock

Tests of highly compacted bentonite as sealing material in boreholes shaft- and tunnel plugs are running in Stripa, see Chapter 8.

Suitable measures for complementory sealing of joints and fissures in adjacent rock are planned to be studied. In a first step the mechanisms and the substances of natural healing and some characterization of the conductivities of rock will be made. In another step the injection techniques known will be reviewed and some pilot testing will be done as well.

4 CHEMISTRY

4.1 GENERAL

The studies within the chemistry program are divided into four categories:

Groundwater chemistry investigations, including groundwater characterization and efforts to understand the variability due to natural circumstances or induced by the near-field reactions in the repository.

Radionuclide chemistry comprising the build-up of a firm chemical data base on the behaviour of radionuclides in the anticipated environment and correlated radiolytical effects.

Model development regarding canister degradation, the release of radionuclides to the rock-groundwater environment and the transport and dispersion of released radionuclides in that environment.

In-situ tests and natural analogue studies intended to bridge the gap between on one hand theoretical and short-term laboratory studies and on the other hand processes in a realistic environment and during a realistic, geological time frame.

4.2 GROUNDWATER CHEMISTRY

4.2.1 Groundwater analyses

A new mobile equipment has been developed and field tested. Included in this system is a zond with electrodes for in-situ measurements, and a fully equipped mobile water laboratory. The zond and the laboratory are connected with a multi-hose through which water is pumped to the surface, packers are inflated, the down-

Table 4-1. Field analyses of groundwater with the mobile water laboratory. The water conductive zone is isolated between packer sleeves.

ELECTRODES DOWNHOLE	
Eh(platinum)	pH
Eh(gold)	pS
Eh(glassy carbon)	
ELECTRODES ON SURFACE	
Eh(platinum)	pH
Eh(gold)	pS
Eh(glassy carbon)	pO_2
Conductivity	
ION CHROMATOGRAPH	
Sodium	Fluoride
Potassium	Chloride
Ammonium	Bromide
Nitrate	Sulphate
Nitrite	Phosphate
TITRATION	
Calcium	Alkalinity
Magnesium	
PHOTOMETER	
Iron(II)	Silicon(SiO_2)
Iron(total)	Sulphide(total)
Manganese	Phosphate
FLUORIMETER	
Uranine (drilling water tracer)	

hole pump is hydraulically operated and data from the zond transmitted.

The zond contains electrodes for in-situ measurements of Eh, pH and pS (sulphide) (see Table 4-1). The electrodes are isolated in the zone to be measured by the inflatable packers.

Table 4-2. Special analyses of selected groundwater samples. These samples are sent to specialized laboratories.

TRACE ELEMENTS	
Lithium	Radium
Strontium	Torium
Aluminium	Uranium
Iodine	
FILTER RESIDUES	
Calcium	Manganese
Aluminium	Sulphide
Iron	Silica
ORGANICS	
TOC (total organic content)	Fulvic and humic acids
ISOTOPES	
^{3}H and ^{2}H	^{14}C and ^{13}C
^{18}O in water	^{234}U
GASES	
Nitrogen	Helium
Oxygen	Hydrogen
Argon	Methane
Carbon dioxide	Radon

Water is pumped up through a tube in the multi-hose and further analysed in the mobile laboratory. For control purpose the same parameters are measured once again on the surface in a thermostated flow through system directly connected to the multi-hose line. Thus, in addition to Eh, pH and pS, electrodes for pO_2 (oxygen) and conductivity are included in the surface equipment (see Table 4-1).

The water is then directly analyzed for main components and sensitive trace constituents in the mobile laboratory. Further samples are sent to specialized laboratories for additional analyses of isotopes etc (see Table 4-2).

Table 4-3. Groundwater analyses of water from two boreholes in Fjällveden. The analyses done in 1984 are made with the new mobile equipment.

Hole, Zone	Fj2, 468 m		Fj7, 722 m	
Year	1983	1984	1983	1984
pH	7.0	6.9	9.1	8.7
Eh(V)	-0.25	-0.22	-0.33	-0.35
Na	42	34	53	325
K	2	1	3	2
Ca	22	27	11	38
Mg	3	3	2	1
HCO_3	180	180	150	17
Cl	4	4	2	550
F	0.9	0.5	-	7.6
SO_4	<0.5	1.9	<0.5	0.3
S(-II)	0.10	0.15	0.01	0.45
Mn(II)	0.27	0.25	0.05	<0.02
Fe(II)	0.2	0.680	<0.05	0.006
Fe tot	2.0	0.690	0.1	0.007

Astonishingly good results have been experienced during the testing period at Fjällveden, a previous investigation site. Markedly better iron analyses were obtained in the field. Iron(II) is an important but sensitive redox parameter (see Table 4-3). It is also important to note in Table 4-3 that a completely different water was found in the borehole Fj7 at 722 m depth as compared to previous measurements in this section. This is a result of the improved equipment and careful sampling procedure.

The flexibility of the new system has also been demonstrated in the new site investigations at Klipperås. For the first time uranine was used as drilling water tracer and the rapid on site fluorimetric analyses of drilling water residues made it possible this time to discard too contaminated water samples before they generated useless data. This increase in effectiveness makes it feasable to collect and analyze more useful water samples than before in one campaign.

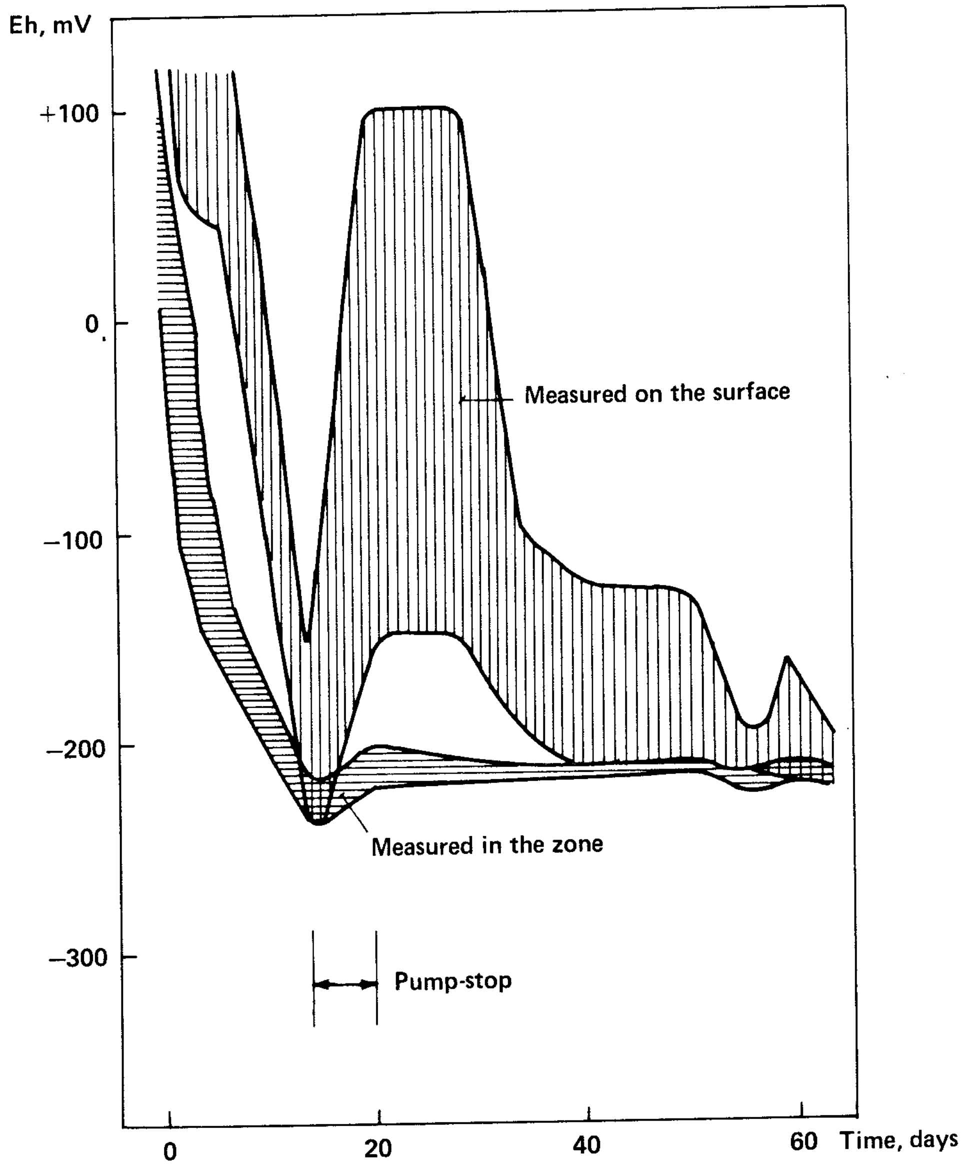


Figure 4-1. Redoxpotential measured on the surface and downhole in a 5 m packed off section at 468 m depth in Fj2. The intervalls cover the Eh-readings from three different electrodes (platinum, gold and glassy carbon).

In addition to the described program separate experiments can also be conducted in the mobile laboratory. An example of this is the collection of fulvic and humic acid samples from Fjällveden. Large amounts of groundwater were filtered through an ion exchanger and comparatively large amounts of organic material, specific for deep groundwater were obtained this way. These samples have now been purified and separated and are used in different radiochemical experiments.

4.2.2 Redox measurements

The primary aim of the redox investigations in field and laboratory is to characterize the reducing conditions at depth in gra-

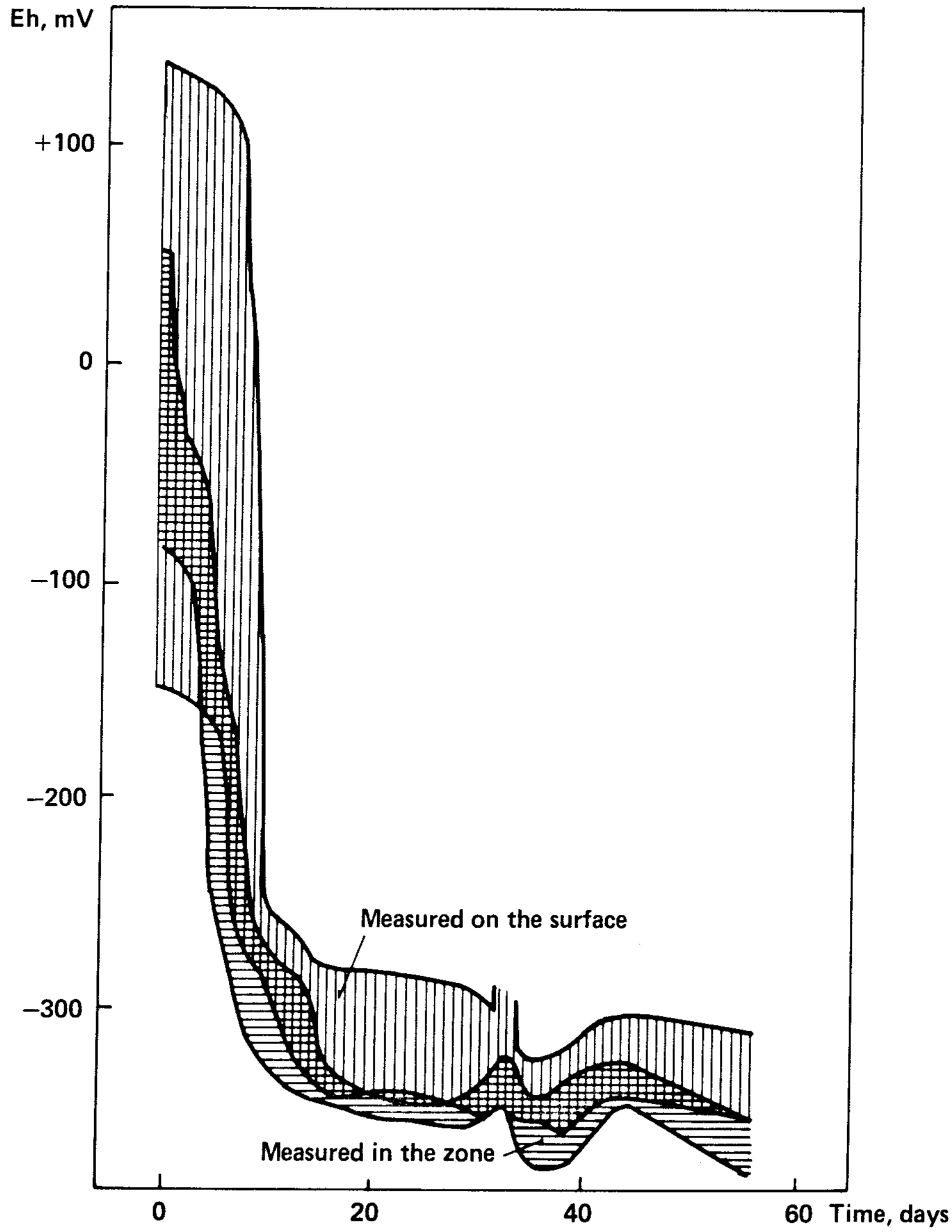


Figure 4-2. Redoxpotential measured on the surface and downhole in a 5 m packed off section at 722 m depth in Fj7. The intervalls cover the Eh-readings from three different electrodes (platinum, gold and glassy carbon).

nitic rock. To put it simple: we want to prove that oxygen is consumed and actinides like eg uranium are reduced to a less mobile state.

Electrode measurements are rapid and significant indicators of redox conditions and are therefore used throughout the program in addition to the complete analyses of redox sensitive components like iron(II), sulphide(-II) etc.

Redox electrodes are extremely sensitive to oxygen contaminations. Therefore, it is a considerable advantage to have electrodes placed in-situ between packer sleeves in the water conductive zone to be investigated. Electrodes on the surface contacted by the same water in a flow through cell isolated from the air

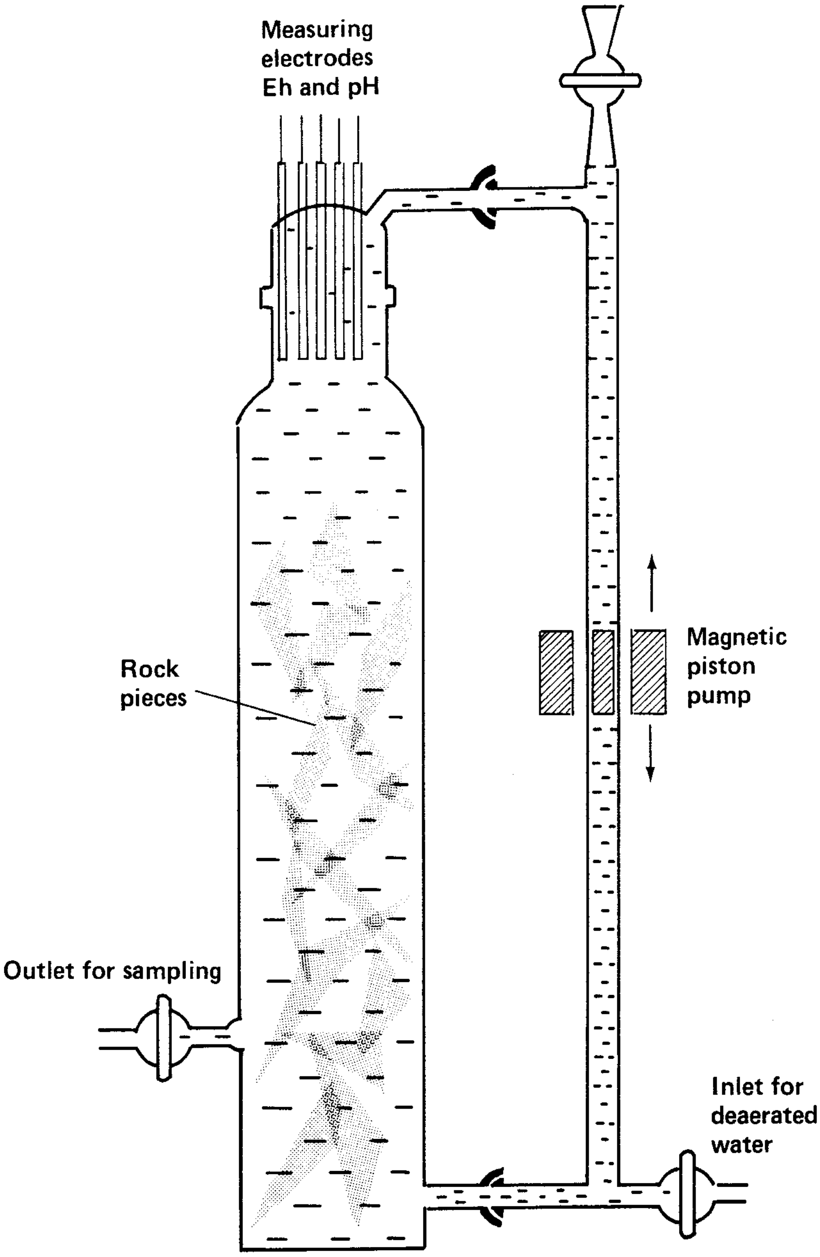


Figure 4-3. Equipment for simulating deep groundwater conditions in a closed system. Water is circulated throug a column filled with drillcore pieces. The parameters pH and Eh are measured continuously and water-samples are taken regulary.

show a similar but slower response indicating oxygen retention in this part of the system despite all efforts to keep it out. This can be studied in Figure 4-1 showing in-situ and surface electrodes response during measurements in Fjällveden at 468 m depth. The "hump" in the curves are caused by a pump-stop that typically affects the surface equipment more than the downhole.

A notable fast response down to low Eh-values can be observed in Figure 4-2, showing the same measurements carried out in Fjällveden at 722 m depth. This can be explained by the large sulphide content, giving this water a notable redox buffering capacity.

Laboratory experiments where groundwater is circulated through crushed rock in a closed system are carried out as a complement

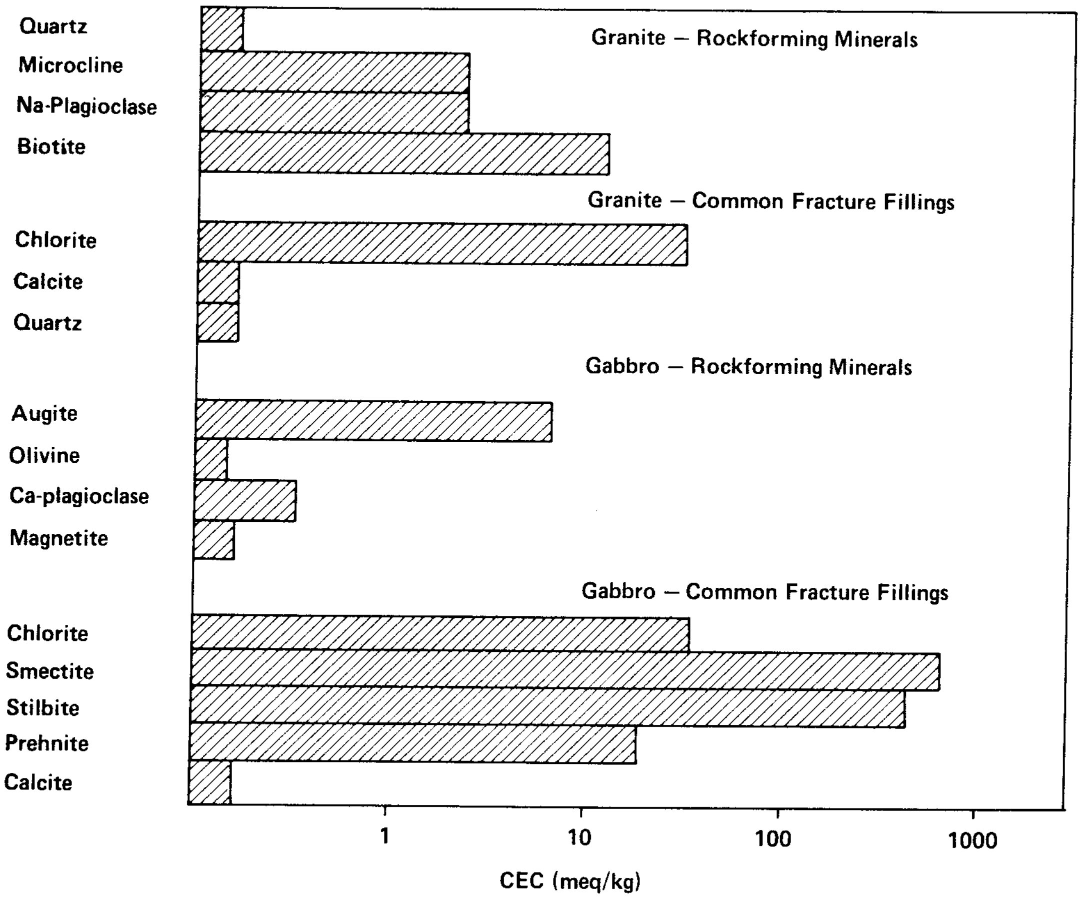


Figure 4-4. Cation exchange capacities (CEC) for common minerals in granite and gabbro.

to the field measurements. The same extreme sensitivity to oxygen contaminations are observed in the laboratory. Therefore, a new equipment has been constructed which should also work without the extra protection of an inert box (see Figure 4-3).

4.2.3 Fracture minerals and thermal effects

Fracture minerals, sampled by coredrilling in a gabbro massiv in Taavinnunanen, have been investigated (KBS TR 84-08). The recharge character of the area is reflected by the isotopic composition of the calcites. It was also shown that the cation exchange capacities of fracture filling minerals in gabbro are generally higher than those in granite, see Figure 4-4. However, it should be pointed out that this alone does not mean better radionuclide retaining properties in gabbro as compared to granite. Diffusion into the rock matrix and water flow characteristics must also be taken into account.

A repository for spent fuel or HLW will create a thermal pulse in the surrounding rock. Geochemical model calculations have been performed on the chemical impact of this pulse on the groundwater and minerals in the near-field of the waste canisters. Background parameters were selected from the Gideå site investigation. The result of this work is published in KBS TR 84-10. It was concluded that the general tendency is formation of weathering products with lower density than the bulk rock. So the expected net result is a dimished hydraulic conductivity in the nearfield.

4.3 RADIONUCLIDE CHEMISTRY

4.3.1 Thermodynamic properties

The measurements of basic thermodynamic constants, describing solubility, inorganic complex formation and redox properties of lanthanides and actinides in the carbonate containing groundwater system, have continued. Number 9 and 10 in a series of articles in this field has been published in international journals during 1984 /4-1 and 4-2/.

With few exceptions, the actinide experiments have so far been concentrated on uranium and the constants of other actinides have been evaluated by means of analogy to lanthanides and uranium in similar oxidation states or review of earlier published investigations /4-3/. The reason for that is of course the radioactivity of actinide isotopes which makes most of them difficult to handle when comparatively large amounts are needed.

However, within the frame of cooperative agreement between the French organization CEA and SKB, experiments of this kind have now been started with plutonium. Other actinides are also feasible. The expertise and equipment of the CEA actinide laboratory is an important factor in this context.

4.3.2 Colloides and organic complexes

The work on colloid transported radionuclides have been concentrated on pseudocolloides, ie radionuclides sorbed on inactive colloides. This kind of radio-colloides is the most likely colloidal species to occur under repository and ambient deep groundwater conditions. Therefore, the stability and mobility of these aggregate are being investigated.

Column experiments is one way to study pseudo-radiocolloide mobility. In Table 4-4 are shown column experiments with americium pseudo-colloides. The flow is evidently an important factor for the mobility.

Organic material from deep groundwaters have been isolated (see Section 4.2.1) and is now being further investigated. Fulvic acids, isolated from this material, have been sent to two diffe-

Table 4-4. Released fraction of americium-quartz and americium-montmorillonite colloides after an eluation of 180 column volumes. The columns are filled with alumina or granite (from Stripa). The column fill particles are 90-125 μm.

Column fill	Colloid	Flow ml/min	Release fraction %
Alumina	Am-quartz	0.05	0.5
"	"	0.5	6
"	Am-montm	0.05	0
"	"	0.5	6.2
Granite	Am-quartz	0.05	0
"	"	0.5	6
"	Am-montm	0.05	2.2
"	"	0.5	4.3

rent laboratories to be investigated with two different techniques: ion exchange and solvent extraction. The aim of these investigations is to determine the strength of complexes between radionuclides and natural occuring organic acids in deep groundwater. One of the two laboratories performing the studies has recently submitted a publication on the fundamental theories of metal ion complexation of humic and fulvic acids /4-4/.

4.3.3 Sorption and diffusion

The effects of pH and ion strength on sorption of radionuclides on minerals in rock are relatively well studied and understood (see eg KBS TR 83-61 and 63). However, despite its importance, the effect of natural reducing conditions on redox sensitive radionuclides like many actinides is comparatively little investigated. The reason for that is primarily to be found in the experimental difficulties. A simple method has therefore been developed. Reduction-sorption experiments using that method on technetium (VII), neptunium(V) and uranium(VI) mixed with rock forming minerals have been started in 1984 to be continued into 1985. Preliminary results indicate that reduction occurs but the process is slow, making long contact times necessary.

Diffusion of radionuclides into the micropores of the rock matrix is also being studied in laboratory. A series of penetration experiments with radionuclides and rock peices have been started during 1984. Numerous experiments have also been carried out with diffusion of non-sorbing species through rock pieces, resistivity and diffusivity measurements under pressure and penetration of inactive cesium and strontium.

Many experiments in the past have been devoted to the study of diffusion in the bentonite buffer. Both radionuclides and inactive species, colloids, organic substances, gases etc have been tested. The technical report KBS TR 84-05 summarizes the experimental techniques used and discusses the interpretation models.

4.3.4 Radiolysis

Theoretical calculations on the kinetics of water radiolysis is an important part in the evaluation of potential hydrogen gas and oxidant formation in a high level waste repository. The calculations used for the KBS-3 study have been supplemented with an evaluation of beta radiation influence on alfa radiolysis (KBS TR 84-03) and a conservative estimate of the formation of nitric and organic acids (KBS TR 84-12). Neither of these phenomena can be expected to cause any major problem.

It is our ambition to have the calculation models tested on realistic radiolysis experiments. Gamma and beta radiolysis of wet bentonite clay have been studied and reported (KBS TR 83-27). The same technique developed for the beta radiolysis experiments is now being used for alfa radiolysis of water in bentonite clay.

4.4 CHEMICAL TRANSPORT MODELS

The fact that crystalline rock is not a homogenous porous media has important implications for the dispersion of eventually released radionuclides in the rock. Two phenomena have been identified as being of special importance: diffusion of dissolved radionuclides into the micropores of the rock matrix and longitudinal dispersion of radionuclides due to the occurrence of preferred flow paths /4-5/. These two effects referred to as "rock matrix diffusion" and "channeling" are the targets of a number of investigations, theoretical and applied, in field and laboratory.

One way of studying these effects during radionuclide transport in single fissures is the laboratory investigations where natural fractures are overcored and used for flow and radionuclide retention experiments. A number of such experiments have been summarized and the data used for theoretical transport model evaluation (KBS TR 84-04). An example of calculated and measured strontium retention data from such an experiment are given in Figure 4-5.

The retention of radionuclides in relatively water conductive fracture zones has so far not been taken credit for in the safety

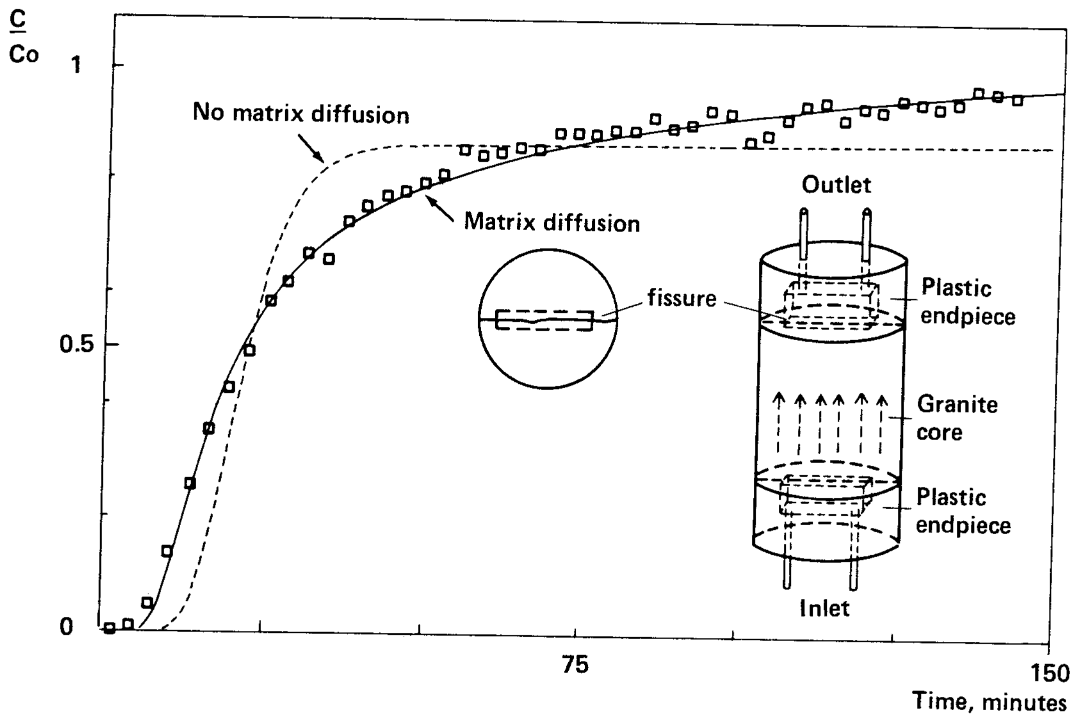


Figure 4-5. Break-through of Sr-85 in a laboratory single fracture experiment. The curves are fitted using the hydrodynamic dispersion model for sorbing tracer, with and without matrix diffusion. In both cases a Peclet number (Pe = 20) from non-sorbing tests were used.

analysis of a high level waste repository in crystalline rock. Once into such a zone, the radionuclides have been assumed to travel with the speed of water. However, sorption on and diffusion into the crushed material in such a zone should be important retention mechanisms. This is one reason for the field investigations of such a zone (see Chapter 5). A theoretical approach to the modelling of radionuclide migration in a fracture zone has been made /4-6/ and the fundamental theories of flow and diffusion in such a medium have also been formulated /4-7/.

4.5 IN-SITU TESTS AND NATURAL ANALOGUES

4.5.1 In-situ tests

In-situ experiments are of course important as tests and demonstrations of predicted radionuclide migration behaviour. It should be noted that the ability to simulate on a large scale then migration of strongly sorbing radionuclides are limited due to the extremely long retention times. Even moderately sorbing species like strontium and cesium can be difficult in this respect (see eg KBS TR 83-18).

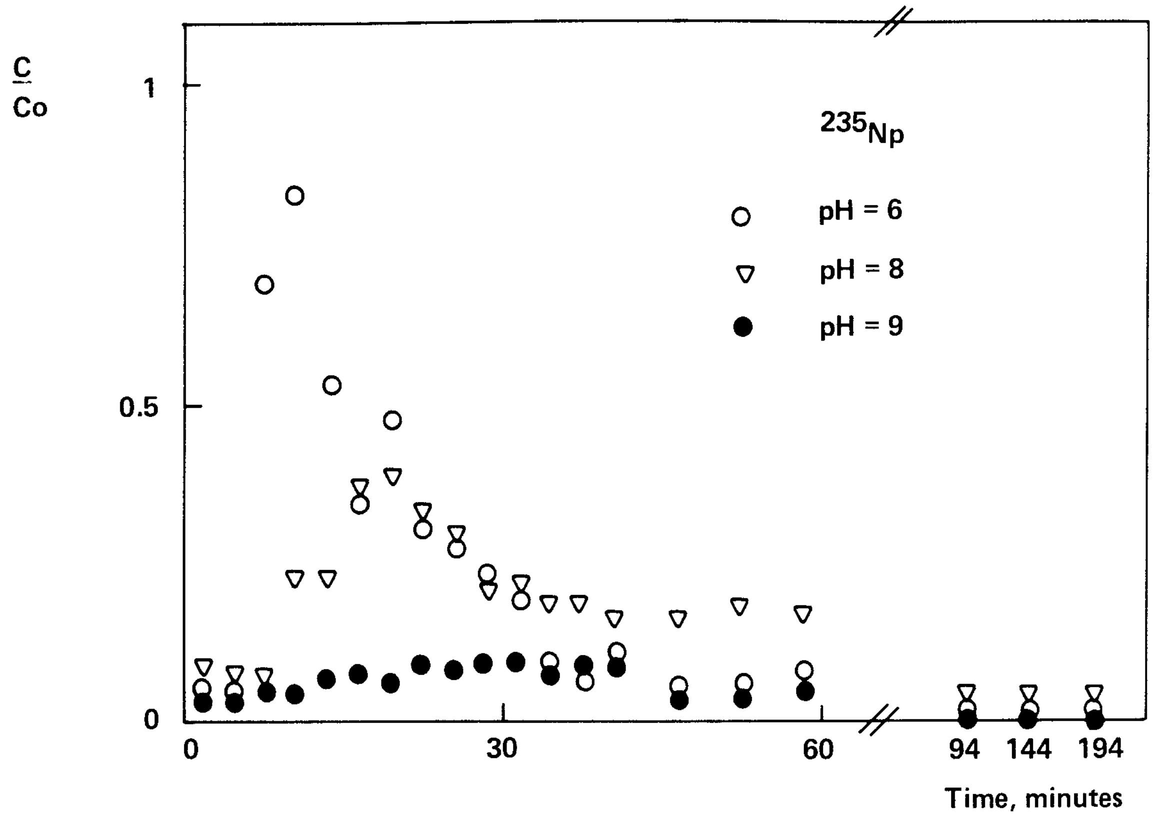


Figure 4-6. The effect of pH on the break-through curve from 15 minutes pulse of neptunium through an overcored natural fracture. The core diameter is 37 mm, core length 104 mm and the waterflow 0.12 cm^3/min.

An experiment in between true in-situ tests and the traditional laboratory batch and column experiments is to overcore natural fractures and use them for migration tests in the laboratory (see Figure 4-5). This technique has been used for migration experiments with americium, neptunium and technetium (KBS TR 84-01). The effect of pH on neptunium sorption can be studied in Figure 4-6. A scanning of the activity on the fracture surface after a flow-through experiment reveals that much of the remaining neptunium is left close to the inlet (see Figure 4-7).

In field, important information on water flow characteristics can be gained by tests with nonsorbing tracers. Sorbing tracers can be used when the experiment is designed accordingly.

The following areas are currently in use for field tests:

Finnsjön

The field experiments with strontium and cesium have been finished and reported (KBS TR 84-07). A fracture zone has now been selected in Finnsjön for further investigations within the fracture zone investigation program (see Chapter 5).

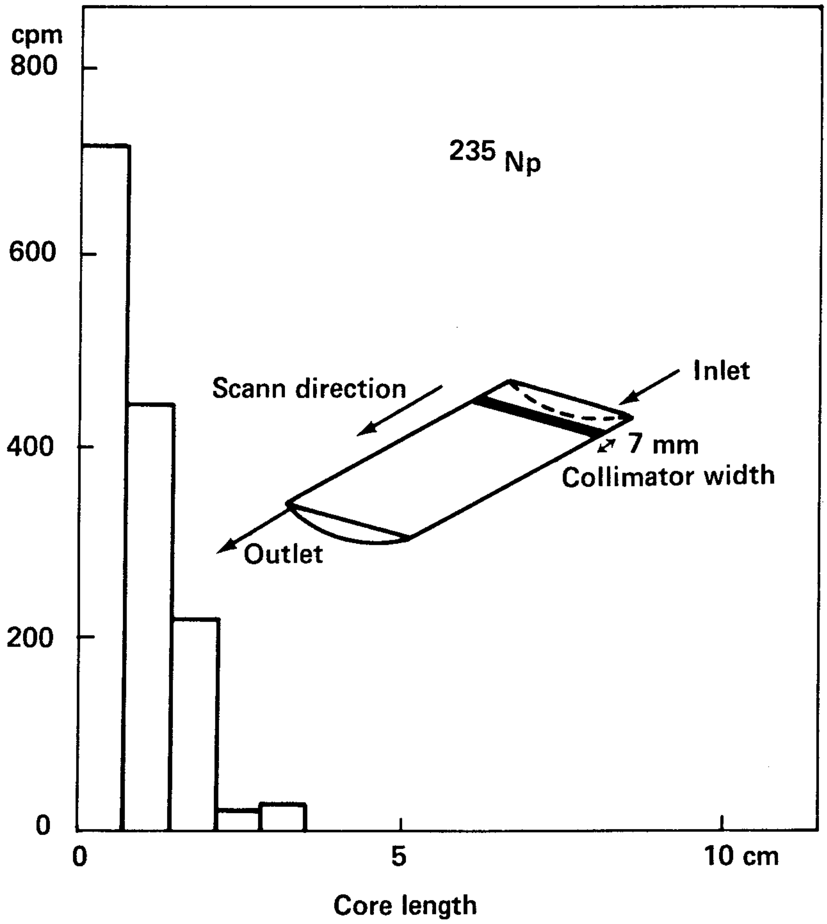


Figure 4-7. The distribution of remaining neptunium along the fracture surface 6.5 hours after the on set flow-through experiments with neptunium. Core diameter 37 mm and core length 104 mm.

Stripa

An extensive program of migration experiments is being carried out in the Stripa mine as a part of an international project. A large number of tracers, sorbing and non-sorbing, have been used (see Chapter 8).

4.5.2 Natural analogue studies

The very long time period that has to be considered in the performance assessment of a nuclear waste repository makes it attractive to supplement laboratory and field experiments with observations in nature, which reflect similar processes over geological time spans.

The potential of natural analogue studies in assessing systems for high level waste repositories have been discussed and evaluated in a recently reported study jointly sponsored by the Swiss organization NAGRA and SKB (KBS TR 84-16 and as a NAGRA report, NTB 84-41).

SKB has also, together with US DOE, arranged a workshop in Chicago on this theme. The proceedings from that workshop are presented in the report KBS TR 84-18.

The main conclusion from these exercises is that the natural analogue approach should be useful if the processes to be studied are carefully selected and evaluated in close cooperation between the investigating geoscientist on one hand and the modeller of these processes in waste repository systems on the other. The emphasis is on processes with well defined border conditions rather than complete geological repository systems, simply because of the low probability to find such an ideal case in nature where all aspects are represented.

During 1984, SKB has supported the following natural analogue investigations:

The Oklo reactor

The fossil nuclear reactor in Oklo in Gabon offers a unique opportunity to study the release and transport of nuclear waste products like eg technetium and plutonium from uranium oxide. No field work, covering this aspect, is going on at present, as far as we know. However, a bulk of information has been gathered in the past. Some of this data, with implications for the spent fuel disposal concept, is now being summarized and evaluated.

Morro do Ferro

Together with US, Swiss and Brazilian organizations, SKB support studies of the large thorium mineralization at Morro do Ferro in Brazil. A compilation of geologic information on Morro do Ferro has been made as a base for the planning of future programs /4-8/.

Uranium mobility

A study on local uranium mobility in granitic rock is being carried out in cooperation with the Swiss organization NAGRA. Samples have been selected from suitable drillcores taken from one granite location in Sweden (Kråkemåla) and two sites (granites and gneiss) in Switzerland (Grimsel and Böttstein). Results from the Kråkemåla samples show varying amounts of mobilization for uranium and its daughter products, depending on distance to a water conductive fracture crossing the drillcore.

Uranium solubility is a sensitive measure of redox conditions. The solubility is orders of magnitude lower under ambient reducing conditions in deep groundwaters as compared to surface conditions. Uranium in itself is an important nuclear waste product but its redox behaviour is also indicative for other radionuclides like eg neptunium.

An investigation of dissolved uranium and uranium containing fracture minerals from a borehole in an inflow area in Kamlunge, has revealed important features. Normally, the change from oxidizing to reducing conditions is completed within the first 100 m of granitic rock from the surface. However, the Kamlunge results indicate that weak oxidizing conditions can stretch down deeper than that in fracture zones with a marked inflow of surface water (KBS TR 84-06).

5 GEOSCIENCE

The activities within the field of geoscience are divided into two main parts:

- Site selection studies, ie investigation of the Swedish bedrock in order to find sites suitable for final disposal of spent nuclear fuel.

- Research and development.

5.1 SITE SELECTION STUDIES

The overall time-schedule for the site selection studies is shown in Figure 5-1. In total around sites will be investigated by surface and borehole observations following a defined standard program /5-1/. Two or three of those sites will be selected for detailed studies around year 1990 and the recommendation for a specific site is expected to be given around year 2000.

The major rock types studied so far are granite, gneiss and to some extent gabbro. Further studies of gabbroic rock will be initiated.

Figure 5-2 shows the location of sites investigated and below a short summary is given of results obtained during 1984.

5.1.1 Bjulebo

Only geological mapping and geophysical surface measurements have been carried out at site Bjulebo located in the south-eastern part of Sweden, see Figure 5-2. The further field activities were cancelled due to a proposal of a new law prohibiting in-

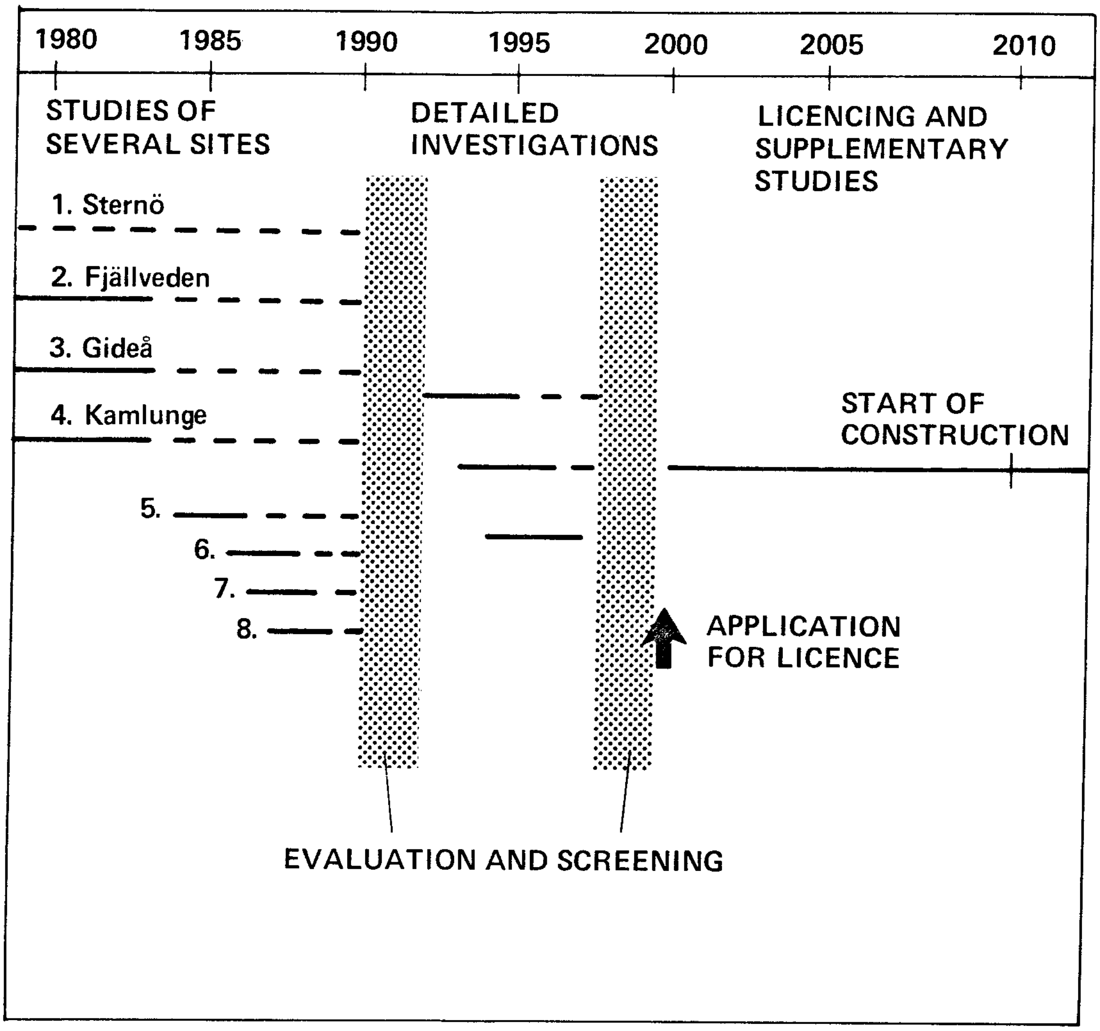


Figure 5-1. Time-schedule for site selection studies.

stallations related to nuclear power within the actual area along the coast of the Baltic.

However, the surface investigations show that the site is dominated by granite, but outcrops of gneissic rocks have also been found. The geophysical surface measurements resulted in distinct anomalies interpreted as fracture zones, with a frequency similar to that measured at previously investigated sites, eg site Gideå and Kamlunge. If the proposed law is not approved, further investigations will be considered.

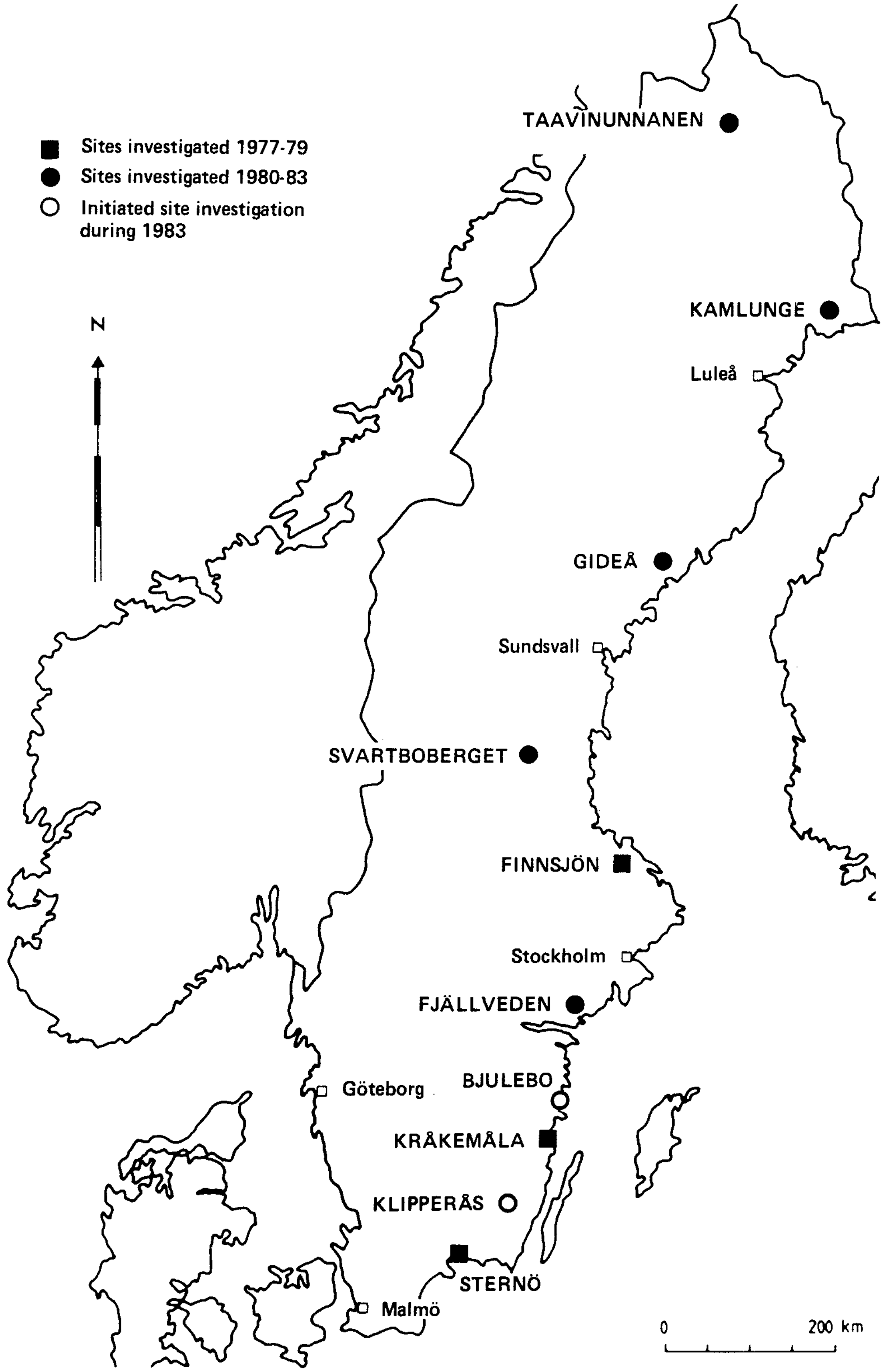


Figure 5-2. Location of study sites.

Figure 5-3. Slingram (horizontal loop EM) measurements are included in the geophysical surface measurements.

5.1.2 Klipperås

The site Klipperås is located in the south-eastern part of Sweden, see Figure 5-2. The site is characterized by a very flat topography and a widespread thin soil cover made the surface geological interpretation uncertain. The few existing outcrops were mapped as an homogeneous, red mediumgrained granite.

The geophysical surface measurements indicated several diabase dikes intersecting the site as well as and major fracture zones bordering portions of less disturbed rock.

A drilling program was initiated to verify the obtained geophysical anomalies. So far, 14 hammer boreholes and 11 cored boreholes have been drilled down to a maximum depth of 150 and 960 meters respectively. The total length of the cored boreholes is 4 280 meters.

Core logging, geophysical measurements, water sampling and hydraulic testing have been performed in the boreholes. The results obtained indicate that the site is dominated by granite, partially very tight but partially intersected by distinct fractures and fracture zones with a hydraulic conductivity of up to 10^{-6} m/s.

5.2 RESEARCH AND DEVELOPMENT

5.2.1 Geology

Fracture zone study

The groundwater flow and the transportation of radionuclides in existing fracture zones are of crucial importance in the safety assessment. In order to allow a better modelling of water movements and nuclide migration in fracture zones a 4-year program has been initiated where the following characteristics will be investigated

- geological characteristics, i a tectonic evolution, structure and age relationships,
- hydrogeological characteristics including migration properties,
- geochemical characteristics, i a hydrogeochemistry, fracture mineralogy, redox capacity and the interaction of water and rock,
- rock mechanical characteristics including deformation properties, rock stresses and the influence of rock stresses on the hydraulic properties.

A fracture zone, assumed to have similar characteristics as those intersecting previously investigated sites, was found in the reseach area of Finnsjön. Initial characterization by drilling and borehole investigations began during 1984.

Bedrock stability

The initial planning for a 3-year program on bedrock stability was also started in 1984. The purpose is to increase the knowledge on the mechanical behavior of the rock and to predict the effects of future stress changes by comprehensive geological, rock mechanical and seismological studies.

5.2.2 Geophysics

Crosshole radar measurements

The development of equipment and interpretation techniques for crosshole radar investigations carried out within the Stripa Project, see Chapter 8, has given promising results. As a consequen-

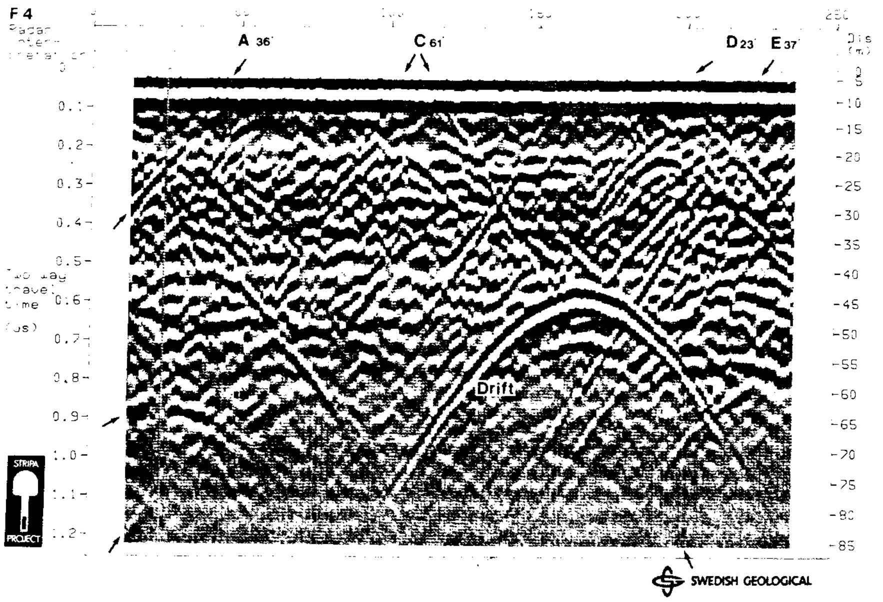


Figure 5-4. Examples of radar reflections obtained at Stripa.

sequence, SKB has initiated a development program for a radar system to be used within the site selection studies. The system will be designed for 56 mm holes and be capable of functioning down to a depth of 1 000 meters.

Scanning borehole TV-camera

A program for development of a scanning borehole TV-camera was initiated for the purpose of determining the true orientation of fractures intersecting boreholes. The results obtained will be similar to those obtained from the more familiar seismic televiewer. The scanning camera will be designed to fit boreholes down to a diameter of 56 mm.

Tube-wave

A tube-wave is initiated along a waterfilled borehole when a seismic wave is penetrating a waterfilled fracture intersecting the hole. The location and amplitudes of the initiated tube-waves can be measured by hydrophones in the hole. Tube-wave measurements may thus determine the location of waterbearing fractures intersecting the hole.

Tube-wave measurements have been carried out in several boreholes within the study sites and compared with previous measurements of the hydraulic conductivity. A good correlation has been obtained for sections with a hydraulic conductivity down to 10^{-9} m/s.

Borehole deviation measurements

Comparative measurements of borehole deviation was carried out at the Stripa mine using the Boremac and Fotobor instrumentation. The results indicated that the Fotobor gives a more reliable determination.

5.2.3 Hydrogeology

Measurement of groundwater flow

A probe for in-situ groundwater flow determinations has been developed and tested in the field. A tracer is injected in a sealed off section in a borehole and the groundwater flow through the section is determined by measuring the dilution of the tracer with time. So far, flow measurements have been carried out in sections with a hydraulic conductivity of 10^{-7} m/s resulting in a flow of around 10^{-9} m^3/s. Tests in sections with a lower conductivity is planned.

Tracer tests

A tracer test over short distances (1.5 m) has been carried out at the Stripa mine using Uranine. The purpose was to quantify the hydraulic parameters essential for the understanding of groundwater flow and material transport in low permeable rock. The investigation was carried out at a depth of 10-20 meters below the floor of a drift. Valuable information was gained from the experiment, e g the tracer conductive fractures were only 3% of the total number of fractures intersecting the boreholes.

Hydraulic modelling

SKB participated in a project where the purpose was to model the undisturbed stationary groundwater conditions of the area surrounding the Underground Research Laboratory (URL), Canada, presently under construction. The results indicated a homogeneous potential distribution and minor differences in hydraulic gradient above three existing subhorizontal fracture zones. Below these zones, however, the gradient was lowered rapidly. The flow rates in the rock mass at the URL laboratory level were in the order of 100 ml/m^2 yr.

Automatic groundwater registration

A device for automatic registration of groundwater level fluctuations was developed and tested during 1984. The equipment is con-

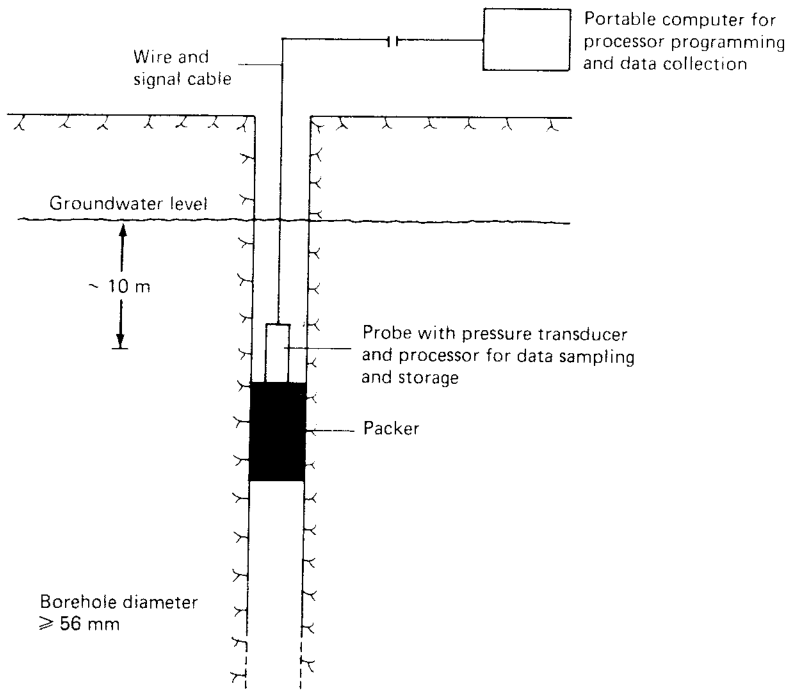


Figure 5-5. Schematic drawing of the equipment for automatic groundwater registration.

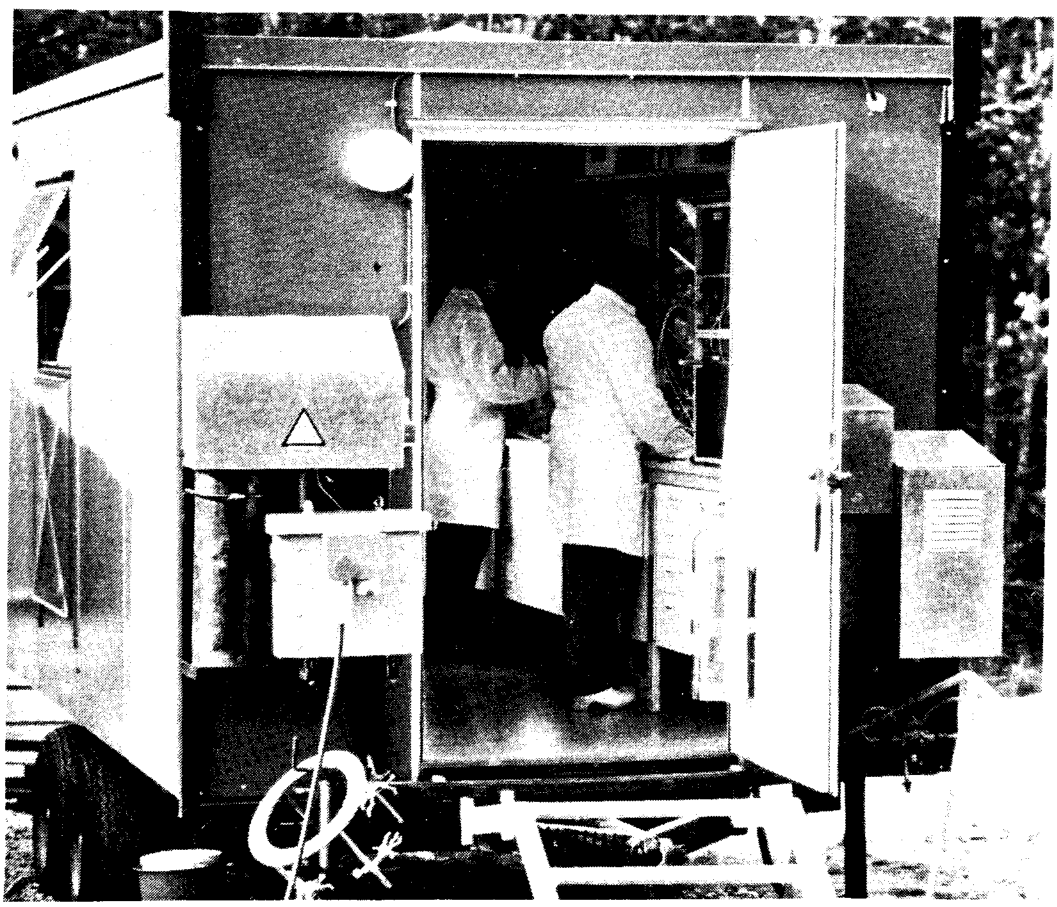


Figure 5-6. Mobile field laboratory for water analysis.

structed for use in 56 mm boreholes or larger. It consists of a packer which seals off the deep part of the borehole and a data control and registration unit mounted under the groundwater table. Measured data are registered and stored in a microprocessor in the probe. The storage capacity and the battery back up enables the equipment to operate for several weeks without supervision. Data are dumped to a portable microcomputer for further processing.

5.2.4 Hydrogeochemistry

Mobile field laboratory

A specially designed field laboratory for chemical analyses of groundwater was constructed and taken into operation during 1984. The complete equipment consists of a mobile laboratory and an umbilical hose with a downhole device for in-situ measurements of Eh, pH and pS^{2-} in the sealed off section. These parameters are remeasured on the surface where the pumped up water passes a flow through cell. In addition, the conductivity and dissolved oxygen are measured. The water is further analyzed with a photospectrometer and ionchromatograph where the major anions and cations are analyzed. See further Chapter 4.

6 THE BIOSPHERE

Groundwater is considered to be the main pathway for radioactive nuclides to reach biosphere and man. The characteristics of the groundwater recipient in the biosphere is therefore of major importance for the dispersion of radionuclides.

The natural ageing and change of the recipients in the biosphere occur often in a shorter time period than the duration of a possible release from the repository. Even with a constant release, this changing biosphere results in a variation in the radiological consequences with time.

The recipients are influenced by the various mechanisms for the change, for instance: by the eutrophication of lakes, by creation of bogs, by the use of a drained lake bottom for agriculture, or by the land rise causing change in the shore line. This will also affect the transfer factors for the various radionuclides from biosphere to man. This results in an unavoidable uncertainty in the assessment of the safety of the repository.

A quantification of this uncertainty would give a better understanding of the relevance of the dose consequence predictions, and would also indicate to what depths and details it would be meaningful to characterize the transfer of radionuclides in the many possible groundwater recipients.

To clarify this a study was launched early 1984. The goal of the first phase was to identify factors and parameters of importance for the radiological consequences in a normally evolving biosphere, and to recommend an area where to investigate these parameters by sampling and analysis. This work is reported in KBS TR 84-17.

Figure 6-1 shows the selected area close to the research laboratory in Studsvik at Nyköping and the various types of samples to be taken.

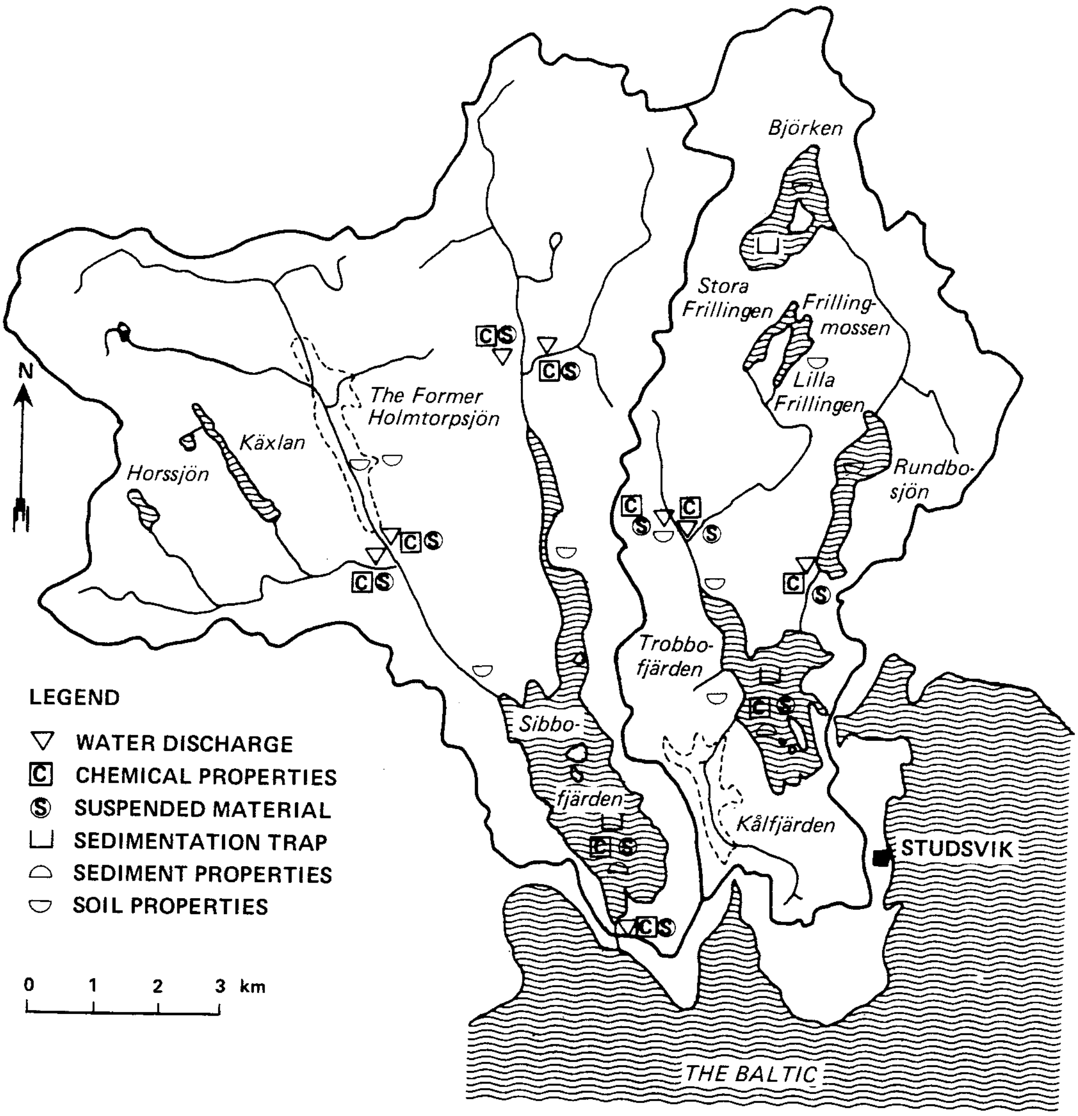


Figure 6-1. The catchment area of Trobbo- and Sibbofjärden. Sampling points are indicated by symbols.

The study (sampling and analysis) will be divided into three major categories.

1 Field studies

- Sampling of sediments along profiles from surrounding solid grounds to deep lake sediments in order to cover lake sediments of different age and character. Discharge measurements and water sampling of macro constituents and suspended materials for estimation of the material and water balance of the lake.

- Determination of the annual growth of sediments and its yearly cycle by using sedimentation traps at different sites and depths.

- Mapping of vegetation for the determination of biomass production.

2 Chemical studies comprising

- Determination of pore water composition in the sediments. Parameters of importance to the mobility of trace metals such as pH, Eh, pS^{2-}, ionic strength and contents of complex formers will be determined.

- Characterization of the solid sediment phase. Measurement of organic content cation exchange capacity and in some cases surface area and mineralogy.

- Trace metal distribution between pore water and solid to provide information on the mobility of natural occurring trace metals in sediments.

- The influence of organic complex formers on the mobility of trace metals. Fractionation according to molecular weight of trace metals complexes in the pore water gives information on the influence of organic complex formers on the mobility.

3 Transports study of regional nuclides in sediments

- In-situ studies to verify the transport of actinides and fission products in sediment. Sediment cores are collected in the recipient, the nuclides are inplanted at different depths in the core, thereafter these are transferred back to the recipient. On subsequent occasions the cores are recollected and distribution of the nuclides within the core is recorded. The migration is correlated to the physical and chemical sediment properties.

This phase will be finished in the middle of 1986, after which a third stage will follow, where the results will be utilized in the biosphere model development.

Beside this major study the general collection of biosphere and surface water data from the site investigation area have been going on. The two areas of Klipperåsen and Bjulebo will be reported in the middle of 1985.

7 SAFETY EVALUATIONS

The finalization of the KBS-3 study on the feasibility of safe final disposal of spent fuel represents a slight modification of the previous approach to performance and safety analysis. Going from more or less generic feasibility studies with conservative safety analyses to more realistic, site specific assessments, new tools will be needed for

- making R&D program priorities,
- design optimization,
- site screening and ranking
- safety assessment.

During 1984 existing methodologies have been assessed and the development work has been started on a computerized system that is intended to provide the features of

- fast calculations from source to dose with readily replaceable models for the near-field, geohydrology, geochemistry, biosphere, etc with their corresponding input data,

- evaluation of uncertainties in the result by allowing the inputs to be entered as probability distributions,

- sensitivity analysis to find the impact of the various parameters on the result and the uncertainty of the result.

Within the information exchange agreement with AECL a SKB employee was able to work with the SYVAC-development team in Whiteshell in April 1984. In the late summer of 1984 the SYVAC 1 code was implemented on a VAX-computer in Sweden. It will be evaluated and also used for comparison in the SKB development work.

An agreement on a close cooperation in the further development of SYVAC and SYVAC-like performance evaluation codes has been reached with the AECL. Similar discussions were held during 1984 with groups in other countries. An international cooperation within this area has also been recommended by both the IAEA and OECD/NEA.

It is the intention of SKB to support this strive for an international harmonization of methods, codes and praxis in performance evaluation of repositories for radioactive wastes.

8 STRIPA PROJECT

8.1 INTRODUCTION

The Stripa Project is an international cooperative project carried out in an abandoned iron mine located in central Sweden. The project concerns research in a realistic environment of different matters connected to disposal of highly radioactive wastes from nuclear power generation, deep underground in crystalline rock. The Stripa mine is only a test station and there is no intention to use or store any radioactive material in the mine.

The project is carried out as an autonomous project under the sponsorship of the OECD Nuclear Energy Agency (NEA) and is managed by the division Research and Development of SKB under the direction of representatives from each participating country.

The project was initiated in 1980. Some of the current activities will be completed in early 1985 whereas others are scheduled to be completed during 1986. However, negotiations regarding additional investigations resulting in a prolongation of the project, is underway. Participating countries are Canada, Finland, France, Japan, Spain, Sweden, Switzerland, United Kingdom and the United States.

The experiments are carried out mainly at the 360 m level in a massive, grey to light red, medium-grained granite.

The progress of the work is presented in a separate annual report. Below a short summary is given.

8.2 RESEARCH PROGRAM 1980 – 1986

The Stripa Project includes a number of subprojects with different objectives, budgets and time-schedules. Essentially, the research is concentrated to the following areas

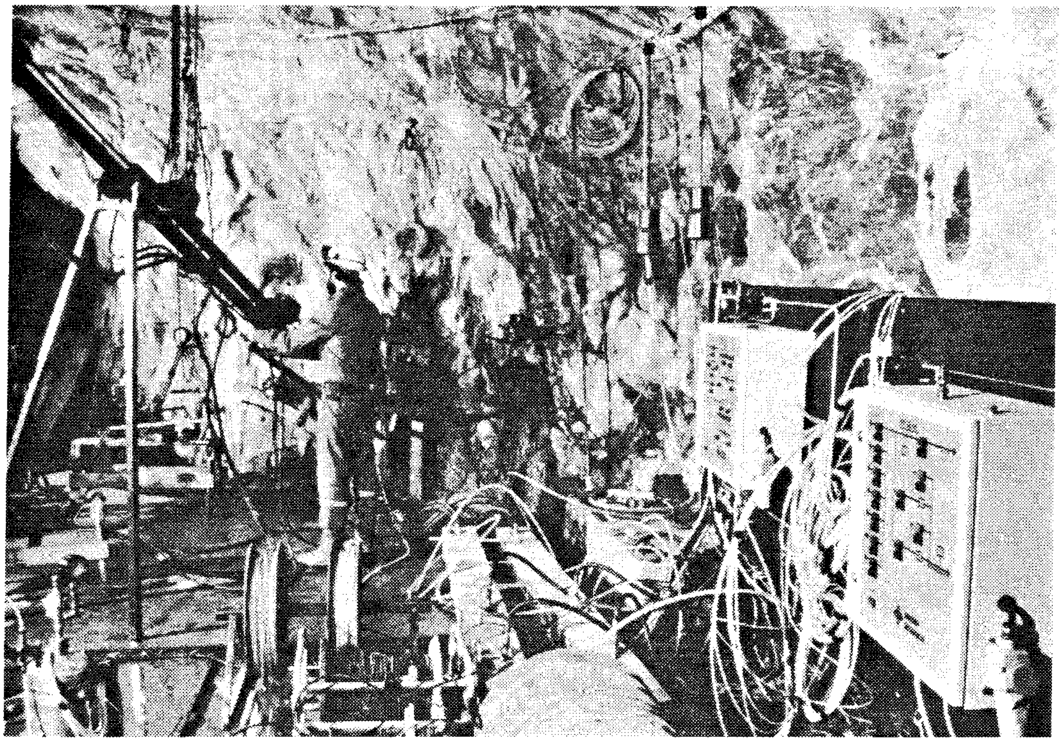


Figure 8-1. Equipment set-up for the hydraulic crosshole measurements.

- detection and mapping of fracture zones
- groundwater conditions and nuclide migration
- bentonite clay as backfilling and sealing material.

The results obtained are presented in quarterly, internal and technical reports.

An international symposium, where the experiments and obtained results will be presented, is scheduled for early June 1985.

8.2.1 Detection and mapping of fracture zones

The objective of the investigation is to develop and test geophysical and hydraulic methods and instruments for the purpose of detecting and mapping of fracture zones. Electromagnetic (radar), seismic and hydraulic (sinusoidal) techniques are developed and tested.

The investigations are carried out in both single and multiple boreholes, so called crosshole measurements. Different kinds of signals are transmitted through the rock. A disturbance like for instance a fracture zone will affect the transmission. The location and orientation of fracture zones may be determined by performing the tests from a large number of sections in the boreholes.

The major part of the investigations are carried out in an especially designed borehole configuration at the 360 meter level at Stripa. The drilling and characterization of the site by core logging and single hole geophysical and hydraulic measurements were completed during 1984.

A new borehole radar system has been constructed and tested at Stripa resulting in distinct reflections from the existing fracture zones as far as 100 meters away from the boreholes.

The construction of and the software development for the equipment used in the hydraulic program was completed and tests of the system were made. The equipment was found to meet the design criteria and worked very well during the initial tests.

A seismic crosshole survey has been carried out at Stripa and interpretation of the data is underway. However, seismic investigations may be carried out at distances up to 500 meters or more between the transmitter and the receiver, making the Stripa mine less suitable as test site. In addition to Stripa, the site Gideå, located in the northern part of Sweden is therefore used as a complementary test site. The geology at the Gideå site has been thoroughly investigated by SKB within the program on the selection of a potential repository site for spent nuclear fuel.

A large scale 3-D seismic test has been carried out at Gideå. Measurements were carried out in three boreholes simultanously and shots were also fired on the ground. Evaluation of the data were in progress by the end of the year using an improved tomographic analysis technique.

8.2.2 Groundwater conditions and nuclide migration

Part of the investigations at Stripa are aiming at determining the evolution of the Stripa groundwaters and to establish a general program for water sampling and analysis in crystalline rock. Water samples are taken from a maximum depth of 1 200 meters below the surface. One of the primary conclusions reached in these studies so far is that different dissolved constituents will provide different residence times because they have different origins and different evolutionary histories that may or may not be related to the overall evolution of the groundwater itself.

Nuclide migration in fissured crystalline rock is a complex phenomenon and has to be investigated thoroughly. The field work of the investigation entitled "Migration in a single fracture",

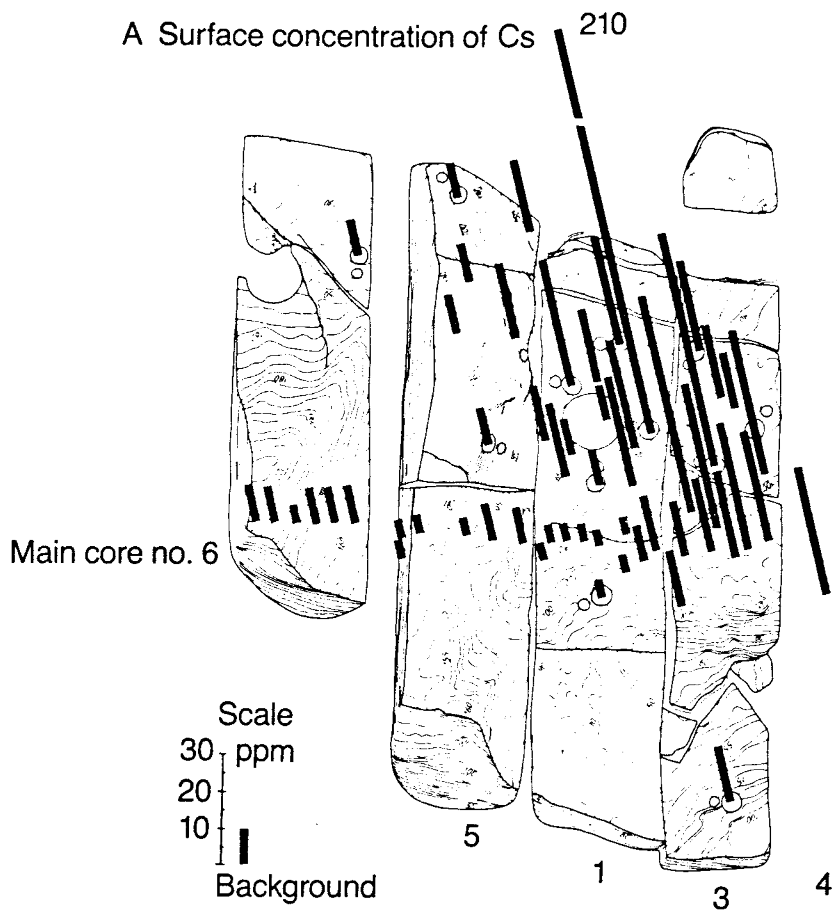


Figure 8-2. Distribution of Cesium around the injection point in the experiment »Migration in a single fracture».

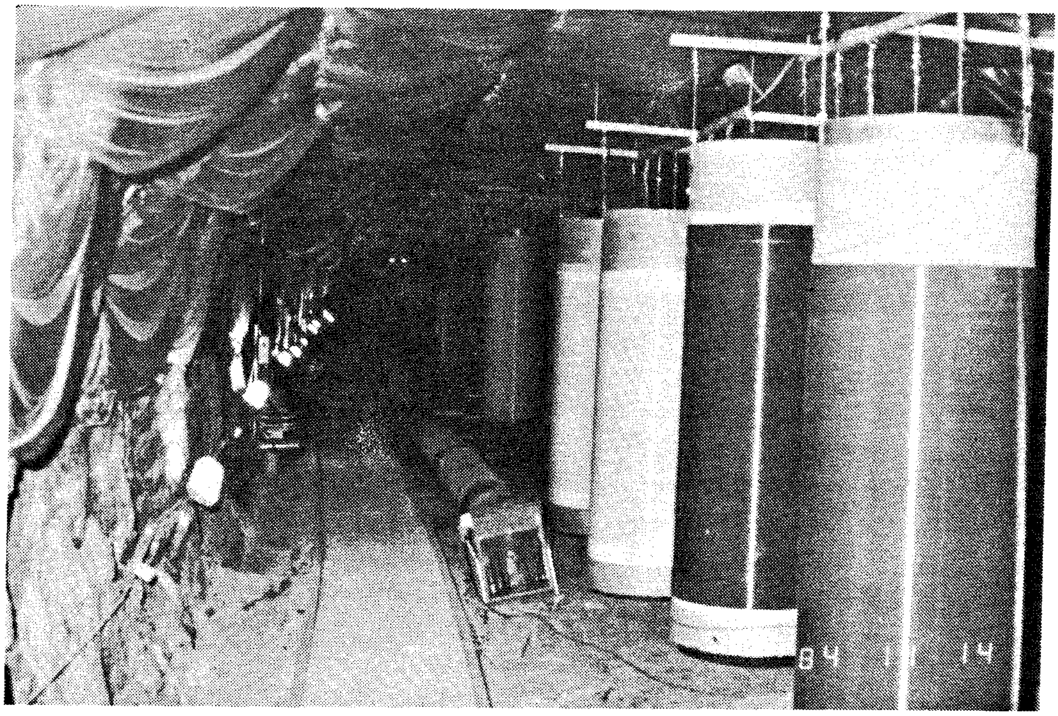


Figure 8-3. Part of the experimental drift used in the large scale migration test and covered with plastic sheets.

where both sorbing and nonsorbing tracers were used, was completed during 1984 and evaluation is in progress. The results show that water is flowing in channels within the fractures and with a very unevenly distribution of the flow indicating that the surface area available for sorption of nuclides is limited.

The preparations for the large scale migration experiment initiated in 1983 is completed and injection of tracers has started. The injection of nine different nonsorbing tracers will continue for about one year using a "constant" overpressure of about 10-15% above the natural pressure. The results obtained so far show that the water flow in to the drift is very unevenly distributed. About 90% of the total inflow was collected in only 5% of the enclosed area.

8.2.3 Bentonite clay as backfilling and sealing material

The so called buffer mass test was initiated in 1980 and will be completed in 1985. Six large boreholes with a diameter of 0.75 meters and a depth of about 3.5 meters, were used as deposition holes for simulated waste canisters surrounded by highly compacted bentonite and overlayed with a mixture of sand and bentonite. Part of the experimental drift was completely filled with the sand/bentonite mixture. The purpose of the test was to verify the suitability and predicted functions of bentonite-based buffer materials under realistic conditions. Thus, temperatures, water uptake, swelling and water pressures were measured. The heater power was set to 600 W but supplementary tests at elevated power, 1200 and 1800 W, were also carried out. The deposition holes were excavated after time periods ranging from 10 to 40 months and the bentonite mass was carefully examined. In the same way, the filled-in portion of the drift was excavated after 33 months of testing.

Most of the field work was completed during 1984. The experiment has been very successful and shows that the bentonite fills all voids after water saturation and swelling and thus verifies previously obtained laboratory data.

A number of tests were started in 1983 in order to investigate bentonite clay as sealing materials for boreholes, shafts and tunnels.

The purpose of the borehole sealing experiment is to test the application technique and the maturation rate of the bentonite as well as the adhesion of the clay material to the rock. Boreholes are equipped with cylindrical blocks of highly compacted bentonite surrounded by perforeated copper pipes or with pipes made of a steel net.

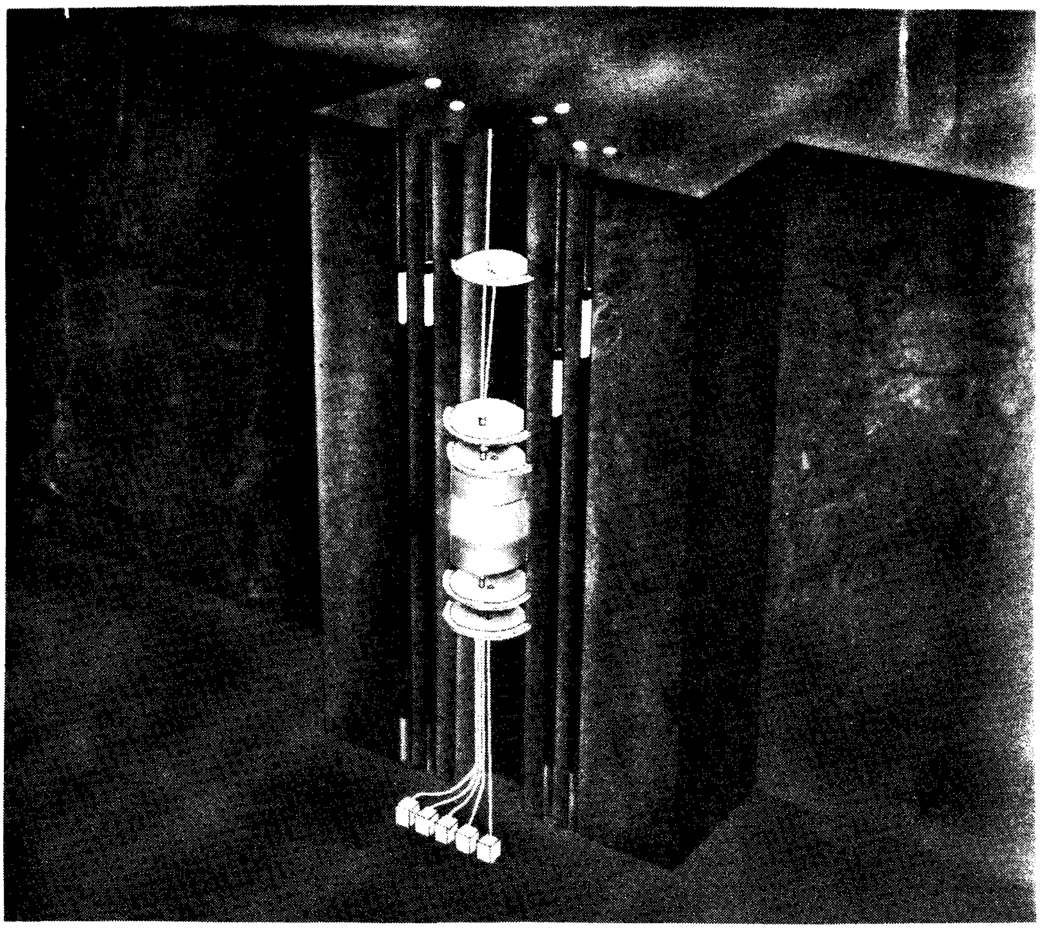


Figure 8-4. Schematic layout of the test where highly compacted bentonite is used for sealing a fractured waterbearing zone crossing a shaft.

Water bearing fractures or fracture zones intersecting shafts and tunnels may also be sealed or isolated by highly compacted bentonite. In the shaft sealing experiment two bentonite plugs are separated by a sandfilled injection chamber simulating a waterbearing fracture. The evaluation is made by comparing the injection flow with that of a preceding reference test with concrete instead of bentonite plugs. The reference test and the preparations for the main test using highly compacted bentonite was completed during 1984.

Highly compacted bentonite is also investigated for the purpose of sealing a highly fractured waterbearing zone crossing a drift. The waterbearing fracture zone is simulated by water pipes embedded in sand. All preparations are completed and the initial testing even at high water pressures (3MPa) shows that the sealing capacities of the bentonite is very good and in accordance with the predictions.

9 LOW AND MEDIUM LEVEL WASTE

9.1 GENERAL

The management outside of the power plant sites and the final disposal of low- and medium level wastes in Sweden are the responsibility of SKB. The low- and medium level wastes are produced in Sweden mainly at the nuclear power plants. These wastes that only contains short-lived radionuclides are called reactor waste. Similar waste is also generated at the Studsvik research facility and to a small extent in industry, health care and research.

Work on reactor waste has been given high priority during the last years with special emphasis on the design and construction of the Final Repository for Reactor Waste (SFR) and the safety assessment for this repository.

A second cathegory at low- and medium level waste that must be considered for final disposal in Sweden is reprocessing waste. The contracts between SKB and COGEMA concerning the reprocessing of spent nuclear fuel provides that all waste deriving from reprocessing may be returned to Sweden. This low- and medium level waste can contain substantial amounts of long-lived α-emitting radionuclides and are not intended to be disposed of in SFR. Similar waste is also generated in a small amount at Studsvik. During 1984 only a few studies concerning reprocessing waste have been performed.

9.2 REPOSITORY FOR REACTOR WASTE – SFR

9.2.1 General

SFR - The Final Repository for Reactor Waste - is being built close to the Forsmark nuclear power plant. The repository is

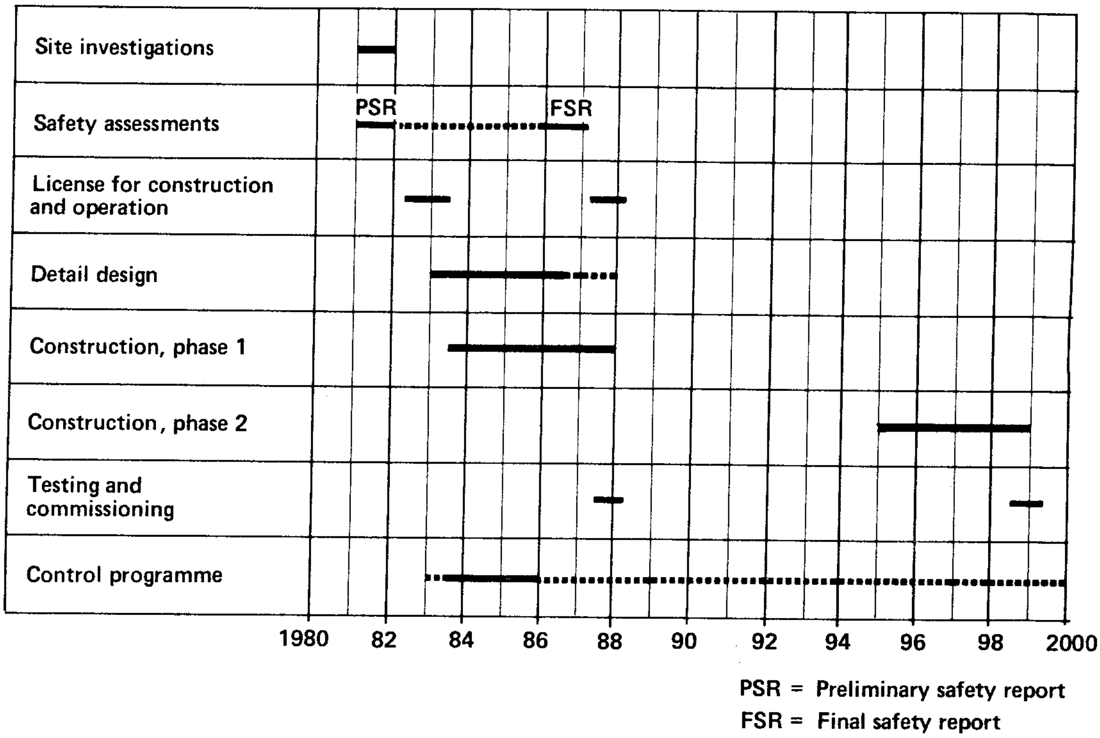


Figure 9-1. Time schedule for SFR.

constructed in crystalline rock at about 50 meters depth. It is placed under the sea about 1 km from the shore /9-1/.

Construction of SFR was started in July 1983. During 1984 the excavation of the two parallell tunnels from the ground to the repository level have been done. At the end of the year the tunnels had reached 1 000 m into the rock and the excavation of the storage chambers will commence early 1985.

Operation of SFR will commence 1988.

9.2.2 Special studies

In parallell with the design and construction work a test and verification program is performed. The purpose of this program is to verify that the safety-related functions of the barrier components in the real repository environment are at least as good as have been assumed in the preliminary safety analysis report.

During 1984 the focus has been on:

- the performance of the buffer material
- gas production and gas transport through the barriers

Figure 9-2. Acess tunnels to SFR.

- hydrological characteristics of the rock
- waste quality assurance (QA)

Two types of buffer material will be used in SFR. For top- and bottom buffers a mixture of 10% bentonite and 90% ballast material with a density of 2.1 t/m^3 (water-saturated) will be used, and for the wall buffer coarse granulated pure bentonite. Tests have been performed in the laboratory on the hydraulic and gas conductivity swelling pressure and rheological properties of these materials with the rather saline water found in the rock at Forsmark. Also field tests have been done to verify that the desired buffer densities can be obtained in practice.

Gas can be produced in SFR by radiolysis, microbial attack and corrosion. A study of microbial attack on bituminized waste is in progress. Experimentally aerobic degradation at a low rate has been shown. Also anaerobic experiments have been started. The rate of attack, if any, is so far below the detection limit.

Corrosion of steel and zircaloy in deaerated cement or cement pore water solutions is being measured by electrochemical methods.

For steel corrosion rates of a few µm/year are found, with hydrogen evolution being the only possible cathotic reaction. For zircaloy no active corrosion is found at the pH-range typical for cement. The passive corrosion rate was of the order of 10^{-2} µm/year (KBS TR 84-13).

These studies shows that a certain amount of gas may be produced in SFR. This gas must escape through the concrete, bentonite and rock barriers. Measurements of gas conductivity in concrete and bentonite have been performed.

In parallell with the construction work measurements and investigations on the hydraulic properties of the rock is being done. These measurements will be used in a detailed hydraulic model for the facility.

The first barrier in the repository is the waste itself. Before the waste will be disposed of in SFR, special permits will be needed for each waste type. To provide background data for these permits a program on the QA of the waste is running.

9.3 WASTE FROM REPROCESSING

In 1984 COGEMA has presented the first series of specifications for the low- and medium level waste from reprocessing at La Hague. These concern sludges and ion-exchange resins incorporated into bitumen and, fuel cladding hulls and other solid wastes incorporated into cement.

In France a large experimental program to verify the characteristics of these wastes is in progress. In the SKB program only a few complimentary studies are done.

Since 1980 an alternative method for the treatment of cladding hulls has been developed by ASEA, under contract with SKB. This method called HIPOW (Hot Isostatic Pressing of Waste) involves pressing of the hulls into a fully compact, pore-free zircaloy block at about 1000°C and 150 MPa. These tests have been reported in 1984 (KBS TR 84-14).

An experimental study of the behaviour of the actinides in concrete at a high pH is in progress. It comprises sorption and diffusion measurements for different concrete types. The sorption measurements were performed for Cs, I, Th, U, Np, Pu and Am on crushed concrete (KBS TR 84-15). Generally the sorption of Cs was lower than in a rock system. For the actinides the sorption in the concrete system was similar to (trivalent and tetravalent species) or higher (pentavalent and hexavalent species) than in must rocks. The differences between the various concretes were generally minor in terms of sorption.

The diffusion experiments are still running.

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MAIN ACTIVITIES IN SKB FUTURE R&D-PROGRAMME

SUBJECT AREA	GOAL	COMMENTS AND EXAMPLES OF PLANNED ACTIVITIES
FUEL AND FUEL DISSOLUTION	Basic research for description of dissolution processes.	The activities are being conducted in cooperation with Battelle PNL, USA and AECL, Canada. * Studies of solubility limits and dissolution kinetics. * Leach tests under realistic redox conditions, evaluation of influence of alpha-radiolysis. * Planning of integrated long-term tests under realistic conditions (start ~1986).
CANISTER AND CANISTER CORROSION	Collection of more detailed data to serve as a basis for canister design.	* Collection and evaluation of data for assessment of pitting in copper. * Experiments with corrosion of copper under influence of radiolysis. * Investigation of conditions for inorganic sulphate corrosion of copper. * Clarification of the risks of creep rupture in connection with very slow strain. * Work on remaining problems with joining and welding of copper canisters.
SPENT FUEL, FACILITIES AND EQUIPMENT	Collection of design basis data	Activities in the 1980s will be limited. * Studies of stress and strain states in copper canisters in different load cases. * Studies of techniques for canister component fabrication, joining and inspection. * Follow-up of technique for cautious blasting of rock caverns.
BUFFER AND BACKFILL MATERIAL	Collection of additional data on the long-term properties of bentonite and of the engineering application.	* Detailed characterization of the properties of bentonite in the short and long term. * Model calculation of stress and strain states in the rock/buffer/canister system under the influence of rock movements. * Development of methods for tight plugging, inspection.

SUBJECT AREA	GOAL	COMMENTS AND EXAMPLES OF PLANNED ACTIVITIES
CHEMISTRY OF THE GROUNDWATER AND THE FRACTURE SYSTEMS	Deepening and broadening of knowledge as a basis for predictions concerning the properties of the barriers.	* Development of methods and equipment for better and more extensive analysis directly in the field. * Further studies of the redox conditions in deep groundwaters. * Analysis of concentration and size distribution of particulate material in the groundwater. * Chemical model calculations of the composition of the water in relation to the minerals. * Development of method for dating of groundwater. * Studies of type, distribution and age of fracture-filling minerals.
RADIONUCLIDE CHEMISTRY AND NUCLIDE DISPERSAL	Continued research into fundamental phenonoma. More precise modelling of dispersal in the near field and the geosphere.	* Continued studies of solubility and complexation. * Theoretical and experimental studies of stability and transport properties of colloids and complexes. * Investigation of sorption on mineral surfaces and sorption mechanisms, continued tests with inward diffusion in the rock matrix. * Calculations and experiments concerning the effects of alpha-radiolysis. * Precipitation and coprecipitation experiments in the uranium-plutonium-water system. * Further development of model for nuclide migration in fracture zone. * In-situ tests with nuclide migration in rock, model verification. * Studies of natural analogies.

SUBJECT AREA	GOAL	COMMENTS AND EXAMPLES OF PLANNED ACTIVITIES
GEOLOGY	Establishment of seismic and tectonic premises for siting of final repository.	* Preparation of a regional tectonic lineament map over large parts of Sweden. * Drilling to great depths (>1 000 m). * Development of methods for registration of width, length and direction of deep fractures. * Studies of recent fault movements. * Model studies of rock-mechanics effects of stress redistributions. * Continuous registrations of earthquakes, detailed observations of after-shocks.
HYDROLOGY	Development of refined measurement methods and models for a more detailed description of the groundwater movements.	* Further development of methods for determination of hydraulic conductivity and groundwater flows. * Model development and verification of models, participation in HYDROCOIN. * Investigation concerning groundwater conditions in connection with an ice age. * Studies of the effects of the repository on regional hydrology. * Studies of the properties of fracture zones.
SITE INVESTIGATIONS	Site selection in stages. 1990: Selection of 2-3 sites for detailed studies. 2000: Final site selection.	* Continuous revision of the study programme. * Sinking of shafts for method and verification studies during the 1990s. * Establishment of central database system for site-specific data.

SUBJECT AREA	GOAL	COMMENTS AND EXAMPLES OF PLANNED ACTIVITIES
DISPERSAL IN THE BIOSPHERE, RADIATION DOSES AND SAFETY PRINCIPLES	Illumination of unavoidable uncertainties in biosphere and dose calculations. Improvement of data base and models.	* Studies for shedding light on the natural ageing of the biosphere recipients and how parameters of importance for the dose burden change. * Support to development of biosphere models and their adaptation for analysis of releases from waste repositories. * Ascertainment of natural radiological conditions on the study sites. * Model verification through studies of natural analogies.
SAFETY ANALYSIS	Development of an integrated, computerbased system for function and safety analyses.	Research activities concerning safety during operation will mainly be conducted after the year 2000. The system for function and safety analysis will contain routines for uncertainty and variation analyses. Model modules of varying complexity will be developed to simulate components and subsystems. Probablistic databases will be built up. A review of possible future scenarios will be performed including an assessment of their probabilities and consequences.

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December 1984

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KBS Technical Report No 84-01

RADIONUCLIDE TRANSPORT IN A SINGLE FISSURE
A LABORATORY STUDY OF Am, Np AND Tc

Trygve E Eriksen
Royal Institute of Technology, 1984-01-20

SUMMARY

Radionuclide migration has been studied in natural fissures running parallel to the axes of granitic drill cores. A short pulse of radionuclide solution was injected at one end of the fissure and the temporal change in radionuclide concentration of the eluate measured.

At the end of each experiment the fissure was opened and the radionuclide distribution on the fissure surfaces measured. The retardation of $^{241}Am(III)$ at pH 8.2 as well as the variation in $^{235}Np(V)$ retardation with pH are found to be in good agreement with K_d-values obtained in batch experiments.

The reduction of (TcO_4^-) to Tc(IV) leads as expected to increasing retardation.

KBS Technical Report No 84-02

RADIOLYSIS OF CONCRETE

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Erling Bjergbakke
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1984-03-16

SUMMARY

A computer based radiation chemical program has been used to simulate experiments with gamma and alpha radiolysis in concrete. The experiments have been performed at Savannah River by Ned Bibler and co-workers. The calculations showed that the gas yields were very sensitive to the pH of the water phase. At a pH of 12.3 fairly good agreement was obtained between measured and calculated gas yields, assuming that the gas production only took place in the free water phase of the concrete. The following main conclusions could be made from both measurements and calculations:

1) A steady state is obtained by gamma radiolysis of a NO_3 free concrete.

2) The yields are higher and a steady state is not obtained if NO_3 is present.

3) The yields are higher and a steady state is not obtained by alpha radiolysis.

Calculations were also carried out on radiolysis from cladding hull waste stored in a cement matrix assuming both alpha and beta radiation. In the presence of an aerated gas phase a steady state pressure of more than 0.21 MPa was obtained.

KBS Technical Report No 84-03

EFFECT OF ß-RADIOLYSIS ON THE PRODUCTS FROM α-RADIOLYSIS OF GROUND WATER

Hilding Christensen
Studsvik Energiteknik AB

Erling Bjergbakke
Risö National Laboratory
1984-07-10

ABSTRACT

In a previous study it has been shown that α-radiolysis of a thin water film on spent fuel stored in a copper canister may result in formation of substantial amounts of hydrogen and oxygen.

In the present investigation the effect of low LET radiation, such as γ- and ß-radiations, on these molecules when they diffuse out of the α-irradiated layer has been calculated using a computerized radiation chemical program. It has been found that low LET radiation under certain conditions may act as a recombiner for hydrogen and oxygen produced by α-radiolysis.

The effect of γ-radiation outside the canister is small - due to the low dose rates prevailing there. However, the relatively high ß-radiation field existing inside the fuel cladding recombines the major part of the hydrogen and oxygen produced by α-radiolysis of pure water, in particular if the gases have to diffuse out through bentonite.

KBS Technical Report No 84-04

ANALYSIS OF SOME LABORATORY TRACER RUNS IN NATURAL FISSURES

Luis Moreno
Ivars Neretnieks
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Department of Chemical Engineering

Trygve Eriksen
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Department of Nuclear Chemistry

SUMMARY

Tracer tests in natural fissures performed in the laboratory are analysed by means of fitting two different models. In the experiments, sorbing and non-sorbing tracers were injected into a natural fissure running parallel to the axis of a drill core. The models take into account advection, diffusion into the rock matrix, sorption onto the rock surface and dispersion. For the last mechanism, one of the models considers hydrodynamic dispersion while the other model assumes channeling dispersion. The models take into account time delays in the inlet and outlet channels.

The dispersion characteristics and water residence time were determined from the experiments with non-sorbing tracers. Surface and volume sorption coefficients and data on diffusion into the rock matrix were determined for the sorbing tracers. The results are compared with values independently determined in the laboratory. Good agreement was obtained using either model.

When these models are used for prediction of tracer transport over larger distances, the results will depend on the model. The model with channeling dispersion will show a greater dispersion than the model with hydrodynamic dispersion.

KBS Technical Report No 84-05

DIFFUSION IN CLAY - EXPERIMENTAL TECHNIQUES AND THEORETICAL MODELS

Trygve E Eriksen
Royal Institute of Technology

Arvid Jacobsson
University of Luleå
1984-06-28

SUMMARY

A large number of experiments have been carried out by this and adjacent research groups to assess the diffusivity of a wide variety of dissolved species such as cations anions, macromolecules and gases in watersaturated clay at differing compaction. The results have been reported in a series of KBS technical reports.

This report is a summary of the experiences gained by these experiments. Recommended experimental methods are described and a methodology to treat and interpret the experimental data is outlined. The mechanisms for diffusion in clay are also discussed in some detail - especially the influence of charge, molecular size and hydrolysis of the diffusing species.

KBS Technical Report No 84-06

URANIUM SERIES DISEQUILIBRIUM STUDIES OF DRILLCORE KM3 FROM THE KAMLUNGE TEST-SITE, NORTHERN SWEDEN

John A T Smellie
Swedish Geological, 1984-03-30

SUMMARY

Studies of the uranium decay series (^{238}U - ^{234}U - ^{230}Th) have been carried out on samples from unaltered bedrock and highly altered fracture/crush zones from drillcore Km3 (Kamlunge test-site). The fracture zones are characterised by abundant iron-oxide coatings (hemitite and hydroxy iron-oxides) resulting from the passage of hydrothermal solutions coeval with the Lina granite intrusion. Enrichments of uranium and thorium, thought to be due to coprecipitation (or preferential sorption) processes together with the iron-oxides, are also present.

The isotopic results show that out of a total of twelve rock samples measured, six indicate isotopic disequilibrium mostly due to unequal depletions of ^{234}U and ^{238}U; one near-surface sample indicated some minor assimilation of uranium. The major fracture zones generally indicate removal of total uranium. This has resulted from interaction with groundwaters which are still marginally oxidising, even at depths of 450 m.

Isotopic disequilibrium has occurred within recent geological times, i.e. during the last 0.5 Ma as imposed by the half-lives of ^{234}U and ^{230}Th. In terms of radioactive disposal considerations, the results are important in that (1) the investigated bedrock environment (100-600 m) is generally reducing, however (2) there is some evidence to indicate that rock/water interactions leading to the removal of total uranium have resulted from the presence of less reducing groundwaters within those large-scale fracture/crush zones which intersect the bedrock surface.

KBS Technical Report No 84-07

STUDY OF STRONTIUM AND CESIUM MIGRATION IN FRACTURED CRYSTALLINE ROCK

Erik Gustafsson
Carl-Erik Klockars
Swedish Geological Co
1984-09-28

SUMMARY

The purpose of this investigation has been to study the retardation and dilution of non-active strontium and cesium relative to a non-sorbing substance (iodide) in a well-defined fracture zone in the Finnsjön field research area. The investigation was carried out in a previously tracer-tested fracture zone, see Gustafsson, Klockars (1981).

The study has encompassed two separate test runs with prolonged injection of strontium and iodide (run 1) and of cesium and iodide (run 2). The tests have shown that:

- Strontium is not retarded, but rather sorbed to about 40% at equilibrium.

- At injection stop, 36.3% of the injected mass of strontium has been sorbed and there is no desorption.

- Cesium is retarded a factor of 2-3 and sorbed to about 30% at equilibrium.

- At injection stop, 39.4% of the injected mass of cesium has been sorbed. Cesium is desorbed after injection stop (400 h), and after 1300 hours only 22% of the injected mass of cesium is sorbed.

KBS Technical Report No 84-08

FRACTURE FILLINGS IN THE GABBRO MASSIF OF TAAVINUNNANEN, NORTHERN SWEDEN

Sven Åke Larson
Geological Survey of Sweden

Eva-Lena Tullborg
Swedish Geological Company
August 1984

SUMMARY

A characterization of fracture filling minerals as well as an account of the distribution of fractures and fracture fillings within the Taavinunnanen gabbro massif, northern Sweden, is reported. Sampling of a 700 metres long (vertical) drill core, of fracture filling minerals has been carried out from the gabbro but also from granite dikes intersecting the gabbro. Dominating fracture fillings in the gabbro are chlorite, smectite, calcite, prehnite and zeolite minerals. In the granite dikes calcite, chlorite and quartz are the dominating fracture fillings.

Generally fractures have steep or vertical dips all along the borehole. More or less horizontal fractures correspond to zones with more intense ingeous layering in the gabbro. Commonly fractures have been reactivated showing complex fracture fillings. The dominating strike of fractures, as measured on the surface, is NNE-NE.

The granite dikes have got more fractures than the gabbro and show higher hydraulic conductivities. Steep fractures are more common in the granite than within the gabbro.

The upper part of the borehole corresponds to high fracture frequency, high hydraulic conductivity and low frequency of calcite fracture fillings.

Several generations of both calcite and chlorite are present. One of the calcite generations is coprecipitated with prehnite. Zeolites have always been found to be younger than the calcite-prehnite generation. The oldest fracture fillings found are chlorite and smectite. Most fractures in the gabbro show alteration zones in contrast to what is found in the granite. However the granite dikes contain simple fillings of low-temperature minerals and show no reactivation or brecciation.

Analyses of $\delta^{18}O$ and $\delta^{13}C$ show a distinct group of calcites, some of which have been coprecipitated with prehnite. This group of calcites have got low $\delta^{18}O$(-20 o/oo to -25 o/oo) and relatively high $\delta^{13}C$(-3 o/oo to -5 o/oo). It is suggested that these calcites have been precipitated from hydrothermal solutions. The $\delta^{18}O$ and $\delta^{13}C$ of calcites sampled from open fissures with high hydraulic conductivity show influence of a recharge water.

It is known that Taavinunnanen is a typical recharge area where downward transport of surface water mainly through fractures within the granite dikes has occurred. Most of the fractures within the gabbro originated very early and have been reactivated. However some granite fractures are potentially young.

KBS Technical Report No 84-09

COMPARATIVE STUDY OF GEOLOGICAL, HYDROLOGICAL AND GEOPHYSICAL BOREHOLE INVESTIGATIONS

Kurt-Åke Magnusson
Oscar Duran
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September 1984

ABSTRACT

The understanding of the permeability of the bedrock can be improved by supplementing the results of the water injection tests with information from core mapping, TV-inspection and borehole geophysics. The comparison between different borehole investigations encompasses core mapping, TV-inspection and various geophysical borehole measurements. The study includes data from two different study areas, namely Kråkemåla and Finnsjön. In these two areas extensive geological, hydrological and geophysical investigations have been carried out.

The fractures and microfractures in crystalline rock constitute the main transport paths for both groundwater and electric currents. They will therefore govern both the permeability and the resistivity of the rock. In order to get a better understanding of the influence of fractures on permeability and resistivity, a detailed comparison has been made between the hydraulic conductivity and resistivity, respectively, and the character of fractures in the core and the borehole wall.

The fractures show very large variations in hydraulic conductivity. Microfractures and most of the thin fractures have no measurable hydraulic conductivity (in this case $<2 \times 10^{-9}$ m s^{-1}), while test sections which contain a single isolated fracture can have no measurable to rather high hydraulic conductivities ($>10^{-7}$ m s^{-1}). Wide fracture zones often have hydraulic conductivities which vary from very low (less than 2×10^{-9} m s^{-1}) to high values (10^{-5} m s^{-1}). This indicates that the hydraulic conductivity is governed by a few discrete fractures.

The resistivity shows a continuous variation in the range 1,000-100,000 ohm-m and a relatively poor correlation with hydraulic conductivities. The observed difference is considered to be the effect of restriction of water flow on a few channels, while electric surface condition, i.e. current transport through thin water films, makes current transport possible through fractures with very small aperatures.

KBS Technical Report No 84-10

GEOCHEMICAL SIMULATION OF THE EVOLUTION OF GRANITIC ROCKS AND CLAY MINERALS SUBMITTED TO A TEMPERATURE INCREASE IN THE VICINITY OF A REPOSITORY FOR SPENT NUCLEAR FUEL

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Institut de Géologie
July 1984

SUMMARY

The alteration of a granitic rock around a repository for spent nuclear fuel has been simulated considering the effect of an increase of temperature due to this kind of induced geothermal system. The results of the simulation have been interpreted in terms of mass transfer and volumic consequences. The alteration proceeds by dissolution of minerals (with an increase of the volumes of fissures and cracks) and precipitation of secondary minerals as calcite and clay minerals particularly (with a decrease of the porosity). The increase of the temperature from 10°C to about 100°C will favour the alteration of the granitic rock around the repository by the solution filling the porosity. The rock is characterized by a very low fissure porosity and a consequent very low water velocity. This too, favours intense water rock interactions and production of secondary clays and the total possible mass transfer will decrease the porosity. A combination of these thermodynamic mass balance calculations with a kinetic approach of mineral dissolutions gives a first attempt to calibrate the modelling in the time scale: the decrease of porosity can be roughly estimated between 2 and 20% for 100,000 years.

The particular problem of Na-bentonites behaviour in the proximate vicinity of the repository has been studied too. One must distinguish between two types of clay-water interactions:

- within the rock around the repository, Na-bentonites should evolute with illitization in slightly open system with low clay/water ratios,
- within the repository itself, the clay reacts in a closed system for a long time with high clay/water ratios and a self-buffering effect should maintain the bentonite type.

This chemical buffering effect is a positive point for the use of this clay as chemical barrier.

KBS Technical Report No 84-11

SMECTITE ALTERATION

Proceedings of a Workshop Convened at The Shoreham Hotel, Washington, D.C., December 8-9, 1983

Compiled by Duwayne M Anderson
Texas A&M University
November 1984

ABSTRACT

This report contains the proceedings of a second workshop in Washington DC December 8-9, 1983, on the alteration of smectites intended for use as buffer materials in the long-term containment of nuclear wastes. It includes extended summaries of all presentations and a transcript of the detailed scientific discussion.

The discussions centered on three main questions:

- What is the prerequisite for and what is the precise mechanism by which smectite clays may be altered to illite?

- What are likely sources of potassium with respect to the KBS project?

- Is it likely that the conversion of smectite to illite will be of importance in the 10^5 to the 10^6 year time frame?

The workshop was convened to review considerations and conclusions in connection to these questions and also to broaden the discussion to consider the use of smectite clays as buffer materials for similar applications in different geographical and geological settings.

SKBF/KBS technical report 83-03 contains the proceedings from the first workshop of these matters that was held at the State University of New York, Buffalo May 26-27, 1982.

KBS Technical Report No 84-12

FORMATION OF NITRIC AND ORGANIC ACIDS BY THE IRRADIATION OF GROUNDWATER IN A SPENT FUEL REPOSITORY

Hilbert Christensen
Studsvik Energiteknik AB
1984-07-13

ABSTRACT

The formation of nitric acid has been reported in the literature, e g in connection with studies of glass leaching during irradiation. These experiments were carried out in order to evaluate the behaviour of glass as a matrix for highly radioactive waste. The presence of a nitrogen containing gas phase has been found to be important for the yield of the reaction.

The formation of organic acids by radiolysis of carbonate solutions has been reported. The literature on the formation of organic and nitric acids has been surveyed and discussed. A conservative estimate of the formation of nitric and organic acids has been made based on dissolved nitrogen and carbonate in groundwater in a spent fuel repository.

Outside a 10 cm thick copper canister containing spent fuel (the Swedish KBS-3 concept) the conservatively estimated total production of nitric, formic and oxalic acids after 1 million years is 0.08 mol HNO_3, 0.06 mol HCOOH and 0.04 mol $(COOH)_2$.

For a thinner version of this canister with a wall thickness of only 1 cm, the conservatively estimated total production after 1 million years is 5 mol HNO_3, 4 mol HCOOH and 3 mol $(COOH)_2$.

KBS Technical Report No 84-13

THE CORROSION OF ZIRCALOY 2 IN ANAEROBIC SYNTHETIC CEMENT PORE SOLUTION

Carolyn M Hansson
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December 1984

SUMMARY

Measurements have been made of the corrosion rates of Zircaloy 2 tubes in anaerobic synthetic cement pore solution of pH 12.0-13.8. The samples were tested in the as-received condition by the polarization resistance technique using a Tafal constant of 52 mV/decade and, for all pH values, corrosion rates of ~3.10^{-5} A/m^2 (~0.03 µm/yr) were determined. These corrosion currents are at the lower limit of the experimental detection range of the technique used.

Some samples were then held at a low electrochemical potential, namely -1850 mV SCE, for several days but this treatment had only a minor effect on the behaviour of the Zircaloy: the value of corrosion rate was increased by a factor of ~3 and the free potential was temporarily lowered but drifted towards more positive values after the applied potential was removed.

Attempts were made to remove the passive film from the surface of the samples by electrochemical reduction. For practical, experimental reasons, this was not successful and, instead, the effect of removing the film by scratching the surface was investigated. At both the free potential and at applied cathodic potentials, an anodic current was detected immediately the surface was scratched but, in all cases, the scratched area repassivated within a few seconds and the anodic corrosion current fell accordingly.

Thus it may be concluded that active corrosion of Zircaloy 2 in anaerobic concrete will not occur and, by comparison with measurements on steel, it is likely that the passive corrosion rates will be even lower in concrete than those measured in the synthetic pore solution.

KBS Technical Report No 84-14

TREATMENT OF ZIRCALOY CLADDING HULLS BY HOT ISOSTATIC PRESSING

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Martin Burström
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1984 12 19

ABSTRACT

A method for the treatment of Zircaloy fuel hulls is proposed. It involves hot isostatic pressing (HIP) for making large, completely densified metallic bodies of the waste. The hulls are packed into a bellows-shaped container of steel. On packing the fuel hulls give a filling factor of only 14 %, which is too low for non-deformable compaction in a normal container, but by using a bellows-shaped container, a non-deformable compaction can be obtained without any pretreatment of the hulls. Fully dense and mechanically strong blocks of Zircaloy can be fabricated by holding them at temperatures of around 1000 °C for three hours.

It is also feasible to incorporate the other metallic parts of the fuel bundle, such as top and bottom tie plates and spacers, in the pressing.

The HIP-densified hulls provide an effective means of self-containment of radioactive waste due to the excellent corrosion resistance of Zircaloy. A waste loading factor of close to 100 % can be realized. Further, a volume reduction factor of 7 and a surface reduction factor of about 250 for a 1-ton canister can be achieved.

Equilibrium calculations have shown that tritium present in the hulls can quantitatively be contained in the HIPed block.

A study has been made of a possible process for industrial-scale use.

KBS Technical Report No 84-15

SORPTION OF Cs, I AND ACTINIDES IN CONCRETE SYSTEMS

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SUMMARY

Samples of 7 different concretes (Standard Portland cement of two types, sulphate resistant cement, blast furnace slag cement, high alumina cement, fly ash cement and silica cement) were prepared, and the pore waters were analyzed in the concretes, after storing the samples under water for a minimum of 14 months. Three samples from old concrete constructions (hydro power station dams) were collected and characterized with respect to the pore water composition.

Batch-wise distribution studies were performed for the elements Cs, I, Th, U, Np, Pu and Am at trace concentration levels, using crushed concrete (stored under water for a minimum of 12 months) as solid sorbents and with artificial pore waters as aqueous phases.

Generally the sorption of cesium was lower (by 1-2 orders of magnitude in terms of the distribution coefficients) than in a rock system (high salinity and low exchange capacity in the cement phase).

For iodine, a considerably higher sorption was observed on concrete than on most rock systems, possibly due to a partial oxidation to IO_3^- at high pH under oxic conditions.

The sorption of americium (trivalent) and thorium (tetravalent) was similar in concrete and rock systems.

For neptunium (pentavalent) and uranium (hexavalent) the distribution coefficients were much higher in concrete than in most rocks (by 1-2 orders of magnitude). This could illustrate the

role of carbonates in natural waters that normally dominate the solution chemistry of actinides in their higher oxidation states (forming anionic low-sorbing species).

Plutonium, which possibly could be oxidized to the hexavalent state at pore water pH, had a sorption similar to that of Th(IV), but also within the upper range of sorption data for U(VI).

The differences between the various concretes were generally minor in terms of their sorbing capacity. Most important parameters affecting the sorption were the time (increased sorption with time during several months in most systems) and pH (decreasing sorption with increasing pH in the penta- and hexavalent actinide systems). The addition of TBP or ethanol amine (2 weight % with respect to the amount of concrete) had no significant effect on the sorption of the actinides.

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THE POTENTIAL OF NATURAL ANALOGUES IN ASSESSING SYSTEMS FOR DEEP DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTE

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ABSTRACT

Many of the processes which will lead to the breakdown of engineered barriers and the mobilisation of radionuclides in a deep waste repository have analogies in natural geological systems. These "natural analogues" are seen as a particularly important means of validating predictive models, under the broad heading of radionuclide migration, which are used in long-term safety analyses. Their principal value is the opportunity they provide to examine processes occurring over geological timescales, hence allowing more confident extrapolation of short timescales experimental data.

This report begins by reviewing the processes leading to breakdown of containment in a high-level radioactive waste repository in crystalline bedrock and the subsequent migration mechanisms for radionuclides back to the biosphere. Nine specific processes are identified as being of the most significance in migration models, based on available sensitivity analyses. These processes are considered separately in detail, reviewing first the mechanisms involved and the most important unknowns then the types of natural analogue which could most usefully provide supporting evidence for the effects of the process. Existing studies are assessed and possibilities considered for additional analogues. Conclusions are drawn, for each process as to the extent to which analogues validate current predictions on scale and effect, longevity of function, etc. Where possible, quantitative evaluations are given, derived from analogue studies.

A summary is provided of the conclusions for each process, and the most important topics for further studies are listed. Specific examples of these requisite analogues are given.

The report emphasizes throughout the importance of linking analogues to well defined processes, concluding that analogues of complete disposal systems do not exist. While the report is aimed specifically at the Nagra (Switzerland) and KBS (Sweden) concepts for disposal of high-level wastes or spent fuel, the results are seen to be more widely applicable. A considerable amount of the information reviewed and presented could be used in the assessment of disposal of other waste types in other host rocks.

KBS Technical Report No 84-17

THE DYNAMICS OF LAKE, BOG & BAY - CONSEQUENCES OF EXPOSURE TO MAN RELATED TO FINAL STORAGE OF SPENT NUCLEAR FUEL

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ABSTRACT

The natural ageing of the environment, much shorter than a possible continuous release from a repository, results in uncertainties as regards to consequences of exposure to man related to final storage of spent fuel.

The Phase I project summarizes the identification of such factors and parameters of predominant importance for an assessment of possible radiological consequences in a normally evolving biosphere.

Three major areas are described: geomorphology and residence time such as phenomena associated with the formation of lakes in Sweden formed by the latest inland ice, lake development in terms of uptake of different elements and radionuclides as a function of the chemical composition of the water in a recipient and some aspects of sedimentation in lakes and coastal waters.

The chemical environment is also studied in the evolution of a lake or a Baltic bay into farming land or peat land. The most important parameters to the behaviour of trace metal ions seem to be pH, Eh, ionic strength and content of complex formers.

Finally, the human impact during the evolution of a lake or a bay is discussed.

The study also includes recommendation for a reference field test area and for laboratory studies of the chemical properties of radionuclides in sediments in a Phase 2 project.

KBS Technical Report No 84-18

NATURAL ANALOGUES TO THE CONDITIONS AROUND A FINAL REPOSITORY FOR HIGH-LEVEL RADIOACTIVE WASTE

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1984-12-21

SUMMARY

This report documents the proceedings resulting from a Workshop held at Lake Geneva, Wisconsin, USA, 1-3 October, 1984. The theme of the Workshop was entitled "Natural analogues to the conditions around a final repository for high-level radioactive waste", and was restricted to ultimate disposal in a crystalline bedrock environment. The Workshop provided an important first step in co-ordinating and focussing different national and individual interests and approaches towards natural analogue studies.

One of the points highlighted at the concluding forum of the meeting was the necessity to first define the geochemical processes which are assumed to occur after disposal of the radioactive waste, and then locate suitable analogue systems which can be used to test the mechanisms of one, or a simple combination of these geochemical processes. To be really useful, a natural analogue must be tightly constrained to a particular geochemical process, with the boundary conditions well defined. Even accepting that the choice of which geochemical process(es) to be selected for validation will obviously be sensitive to each national disposal strategy, far-field radionuclide retardation mechanisms in the geosphere were considered to be a central topic of importance, and should therefore be given high priority.

At this early stage in the development of natural analogue studies, it was not possible to cover all the important aspects. In retrospect: the role of the modeller should have received more attention, bridging the gap between geoscientists and the modellers was seen as being of prime importance.

KBS Technical Report No 84-19

GENERAL CORROSION OF Ti IN HOT WATER AND WATER SATURATED BENTONITE CLAY

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ABSTRACT

Titanium has been proposed as one of the candidates for canister materials for storing spent nuclear fuel in Swedish bed-rock. The deposition milieu was simulated on a laboratory scale by embedding titanium in compacted bentonite and the general corrosion rate was investigated. More fundamental studies were also performed where titanium was exposed to water in which special attention was paid to the NaCl content (0 or 1%) and oxygen content (saturated or free from oxygen). In reaction cells designed according to high vacuum principles it was possible to reduce the oxygen content to very low values. The exposure time ranged between 1 min. and 6 months. Analysis of the corrosion products was performed mainly with ESCA.

In water at 95°C the oxide growth follows a direct logarithmic law: $y = 8.7 + 3.65 \ln t$ (y is the oxide thickness (Å) and t is the exposure time (s)). Oxygen and salt do not influence the rate of the oxide growth significantly. The general corrosion rate is approximately the same as the oxide growth rate since the dissolution of Ti into the water-solution is very low. The oxide consists of an outer layer of TiO_2 and a few atomic layers of suboxide close to the oxide/metal interface. Transmission electron microscopy studies of the water-formed oxides indicate that these are amorphous.

The oxides formed on Ti exposed in bentonite is 70 - 100 Å thick for exposure times ranging between 4 months and 2 years. It is shown, that montmorrillonite - the main constituent in bentonite - is absorbed in the TiO_2 formed on these samples. If it is assumed that a logarithmic growth law is valid even for long-term exposure in bentonite, the growth law which will give the highest growth rate is $y = 5.5 \ln t$. An oxide thickness of 160 Å is obtained if this law is extrapolated to 100,000 years exposure.