SKBF KBS TEKNISK RAPPORT

80-26

KBS ANNUAL REPORT 1980

Including Summaries of Technical Reports Issued during 1980

KBS Stockholm, March 1981

SVENSK KÄRNBRÄNSLEFÖRSÖRJNING AB / PROJEKT KÄRNBRÄNSLESÄKERHET

POSTADRESS: Kärnbränslesäkerhet, Box 5864, 102 48 Stockholm, Telefon 08-67 95 40

KBS ANNUAL REPORT 1980

Including Summaries of Technical Reports Issued during 1980

KBS Stockholm, March 1981

TABLE OF CONTENTS

ABSTRACT

The KBS Project was organized in late 1976 by the Swedish nuclear power utilities within the framework of the jointly owned Swedish Nuclear Fuel Supply Co, SKBF. The original purpose of KBS was to perform the studies and investigations necessary to fulfill the requirements of the Swedish "Stipulations Act" of 1977, which says that the owner of a new nuclear reactor has to demonstrate how and where spent nuclear fuel or high-level radioactive waste from reprocessing can be stored in an absolute safe manner before the Government can grant him permission to charge the reactor with fuel.

Subsequently, KBS has been assigned to carry out R&D work concerning the treatment and final disposal of all kinds of radioactive wastes from nuclear power production as well as the decommissioning of nuclear facilities. KBS has therefore been transformed from a temporary project into a permanent division of SKBF.

This annual report provides a summary of KBS activities during 1980.

The studies concerning the long-term behavior and safety of repositories for radioactive wastes have been organized as shown in the attached diagram.

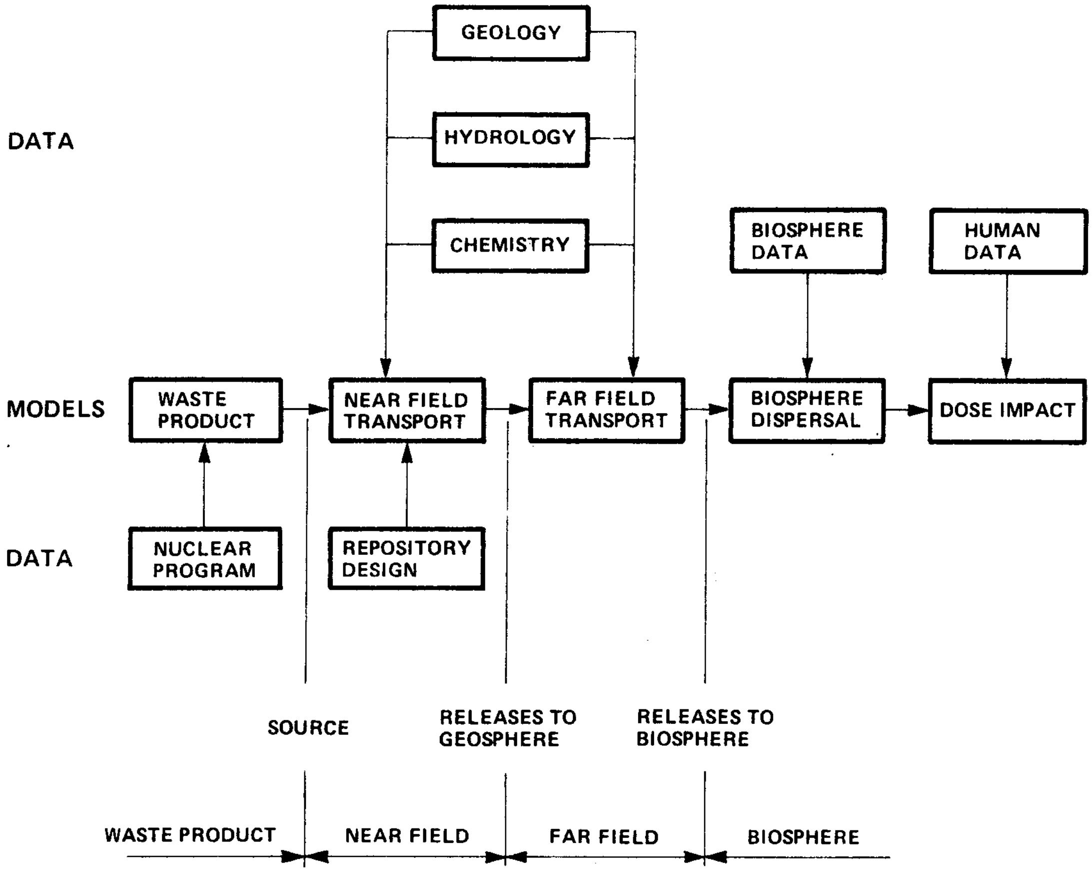
The main efforts in different fields during 1980 have been the following:

Reviews: Much time has been spent on studies and commenting on review statements received concerning KBS-2 (2).

Materials: Studies of waste glasses and canister materials.

Engineered barriers, technology: Continued studies of properties and behavior of bentonite clays.

Chemistry: Studies of chemistry in the "near-field".



Geology and hydrology: Improvement of hydrological models, further development of hydrological instruments and methods, gathering of field data, primarily from the Finnsjö area.

Safety analysis: Improved modelling of nuclide migration.

Low- and medium-level wastes: Design studies and field investigations for a final repository for reactor wastes, characterization of wastes from operation of reactors and from reprocessing.

Stripa Project: An autonomous OECD/NEA international project managed by SKBF/KBS, started officially May 1st, 1980. Preparations for in situ tests have been made.

Foreign contacts: Formal agreements on information exchange have been signed with US DOE, AECL Canada and NAGRA, Switzerland.

The international situation and developments in other countries have been followed through an exchange of reports and participation by KBS representatives in conferences, symposia and workshops, where papers have been presented by KBS and its contractors.

As in 1979, the work on radioactive waste management has been divided and coordinated between the utility-owned KBS and the government organization Prav (Programrådet för Radioaktivt Avfall = The National Council for Radioactive Waste). In order to provide a full picture of the R&D work on radioactive waste management that is being conducted in Sweden, short summaries of Prav's activities during 1980 are also included in this report. More detailed information on Prav's work will be furnished in Prav's Annual Report later this year.

The first part of this report, Chapter 1-7, deals with work related to high-level waste, even if some of the findings could be applied also on low- and medium-level waste. In Chapter 8 an overview of the program and the progress of the international Stripa Project is given. Studies regarding treatment and storage methods including safety analyses for low- and medium-level waste are described in Chapter 9.

1 GENERAL BACKGROUND

1.1 THE SWEDISH NUCLEAR POWER PROGRAM

The following decisions of the state authorities have determined the present Swedish nuclear power program and thereby also the basis for the planning of radioactive waste management in Sweden.

1975 Parliament sets the target for the Swedish nuclear power program at 13 reactors for the foreseeable future

1977 "The Stipulations Act" requires owners of new reactors to demonstrate how and where the high-level radioactive waste from reprocessing or spent fuel can be finally disposed of in a safe way.

No new reactor is allowed to be charged with fuel unless this demonstration of disposal capability has been approved as satisfactory by the Government.

1979 The Government approves the applications for charging four reactors (Ringhals 3 and 4, Forsmark 1 and 2) with fuel. The applications were based on the KBS-1 report and a reprocessing agreement with the French Company Cogema.

1979 It is agreed between the political parties that a referendum on the Swedish nuclear program shall be held.

1980 A referendum is held on the future Swedish nuclear program. About 60% of the voters were in favor of a 12 reactor program and about 40% were for decommissioning of existing nuclear power plants as soon as realistically possible.

1980 Parliament adopts the outcome of the referendum and sets a program of 12 reactors, of which none should be operated beyond the year 2010.

Table 1-1. Swedish nuclear power reactors

Name	Type	Power/MW	In operation
Oskarshamn 1	BWR	440	1972
Oskarshamn 2	BWR	580	1974
Barsebäck 1	BWR	580	1975
Ringhals 2	PWR	820	1975
Ringhals 1	BWR	760	1976
Barsebäck 2	BWR	580	1977
Forsmark 1	BWR	900	1980
Ringhals 3	PWR	915	1980
Forsmark 2	BWR	900	(1981)
Ringhals 4	PWR	915	(1982)
Oskarshamn 3	BWR	1050	(1985)
Forsmark 3	BWR	1050	(1985)

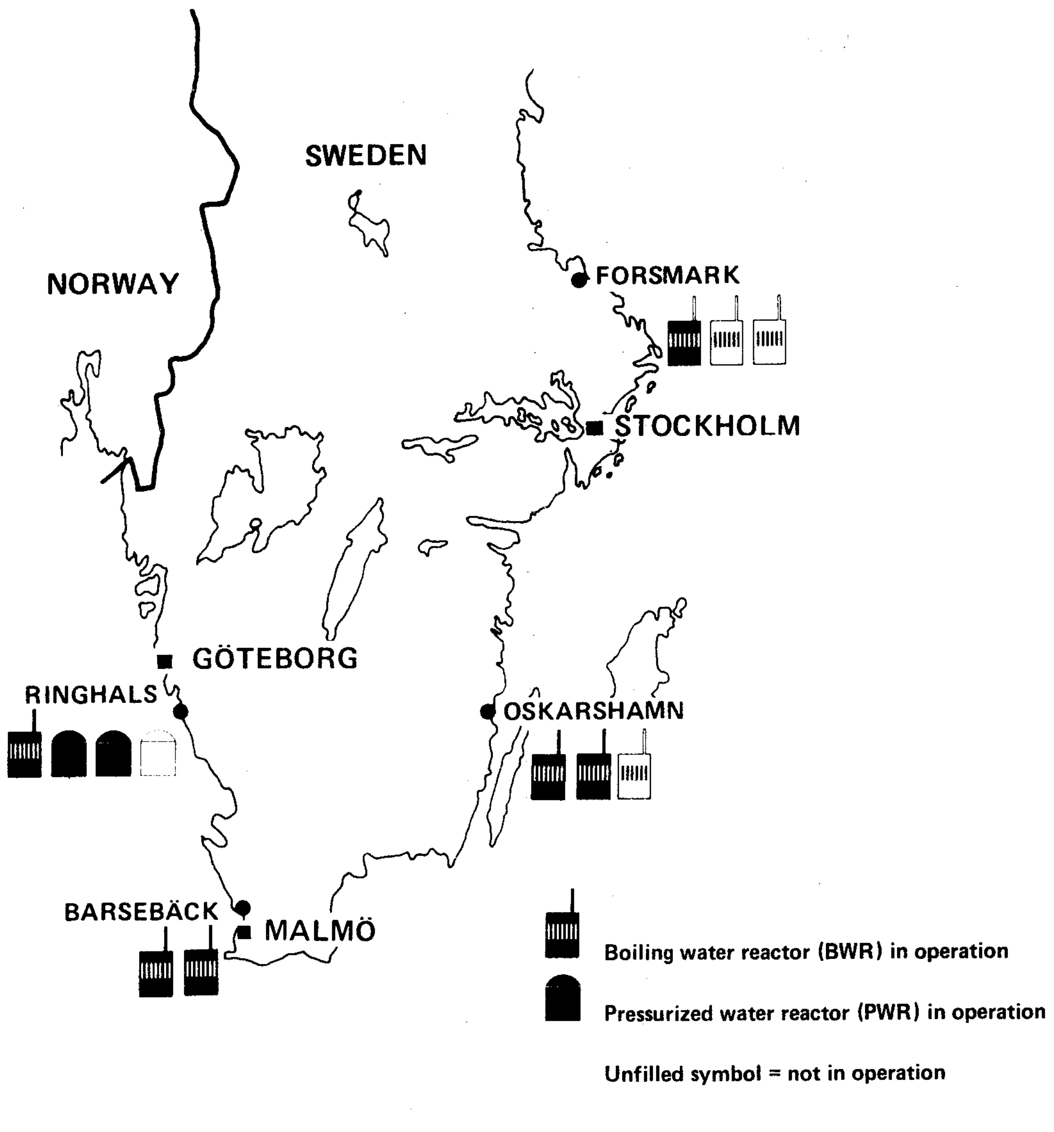


Fig.1–1. The Swedish nuclear power program in accordance with the 1980 parliamentary resolution

Of these twelve reactors, eight are in operation, two under commissioning and two under construction and scheduled to be operative in 1985. The reactors are situated as shown in Figure 1-1, where their capacity is also given. Nuclear energy now supplies about 25% of the electricity produced in Sweden and this share is expected to rise to 45% in the late eighties.

Current planning of the waste management program is based on the above-mentioned decision of Parliament in 1980.

One very urgent need is to provide additional storage capacity for spent fuel within the next few years, as the storage pools at some Swedish reactors will soon be fully utilized. In May 1980 - when all necessary licenses and permits had been granted - the construction work on a central spent fuel storage facility called CLAB (in US an AFR facility) was started at the Oskarshamnsverket site: see Figure 1-2. CLAB is scheduled to be operational in early 1985.

A transportation system for spent fuel and radioactive wastes is also at an early stage of realization. A special ship and fuel transport casks have been ordered for delivery in 1982. During the first few years, the transportation system will be used for the transportation of spent fuel from Sweden to France for reprocessing.

A special group established by SKBF and its owners is responsible for the management of the CLAB and transportation system projects.

1.2 THE KBS ORGANIZATION

KBS was organized at the end of 1976 as a special project within the legal framework of SKBF (AB Svensk Kärnbränsleförsörjning = Swedish Nuclear Fuel Supply Co), which is jointly owned by the Swedish nuclear power utilities. The first task for KBS was to perform the investigations and calculations needed to demonstrate how and where the high-level waste and spent fuel could be finally disposed of. This first task was accomplished in 1978 and the results are published in two main reports, one on "Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste" (KBS-1, November 1977) and the other on "Handling and Final Storage of Unreprocessed Spent Nuclear Fuel" (KBS-2, September 1978).

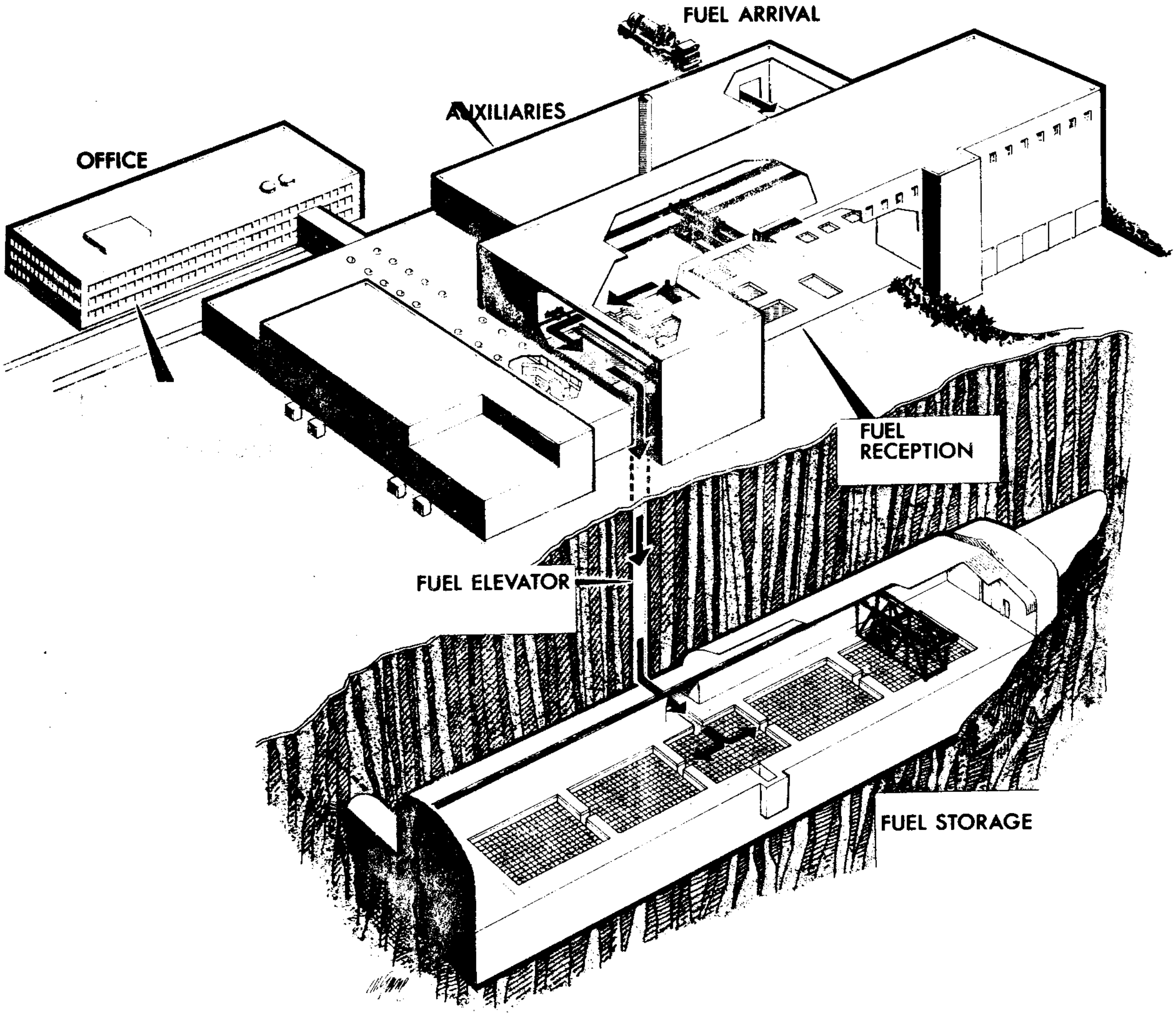


Fig.1–2. The central fuel storage facility CLAB

In 1979 the nuclear power utilities (owners of SKBF) decided that KBS should continue its work with more detailed studies regarding all kinds of radioactive wastes from nuclear power production and reprocessing. Decommissioning of nuclear facilities should also be included in the KBS studies.

This means that, organizationally, KBS has changed from being a temporary project to being a permanent division of SKBF. This new situation occasioned a change in the meaning of the abbreviation KBS from "Kärnbränslesäkerhet" (Nuclear Fuel Safety) to "Kärnkraftavfallets Behandling och Slutförvaring" (Handling and Final Disposal of Nuclear Power Waste).

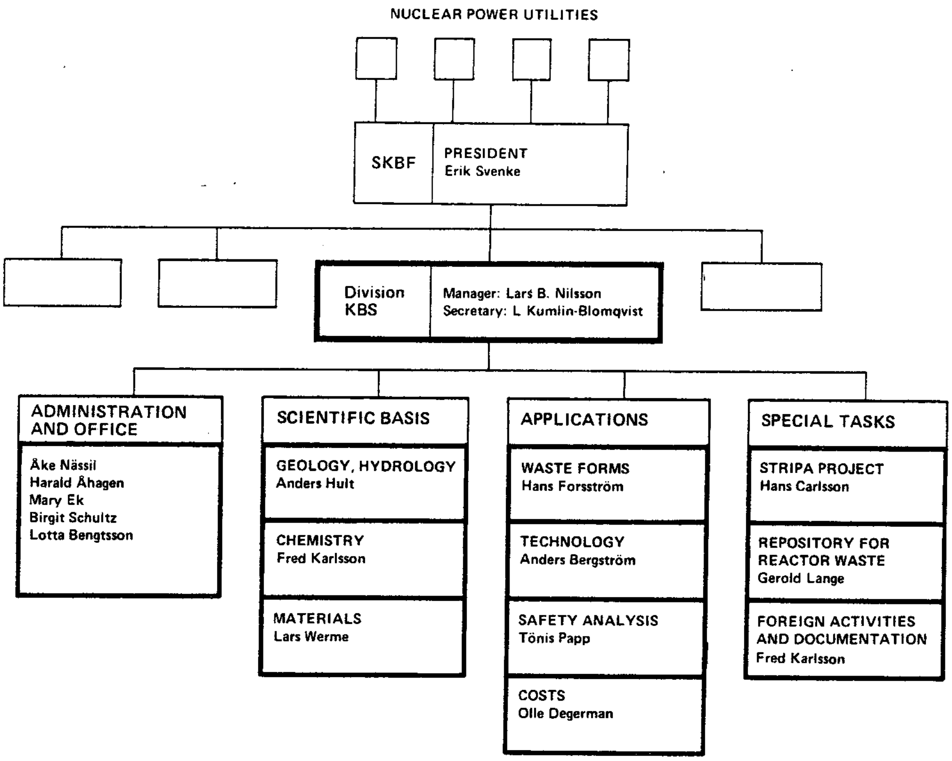


Fig. 1–3. The present organization of KBS

The present organization of KBS is shown in Figure 1-3. KBS is a management group which contracts and coordinates investigations performed by consultants and experts. The contractors engaged in 1980 are listed in Appendix 6.

1.3 KBS CONCEPTS FOR AN HLW REPOSITORY

Later in this report several references are made to the concepts for handling and final disposal of high-level waste known as KBS-1 (1) and KBS-2 (2). Short descriptions of their main features are therefore given below.

KBS-1 deals with vitrified high-level radioactive waste from reprocessing. It is proposed that the waste glass be stored in an aircooled facility for 40 years from the time when the spent fuel is taken out of the reactor. During this intermediate storage period the radioactivity and heat generation will decay to about half the initial values, which considerably simplifies subsequent handling and final storage. Prior to final storage, the glass cylinders are encapsulated in canisters of 100 cm of lead encased in containers of 6 mm of titanium: see Figure 1-5. The encapsulated waste is deposited at a depth of 500 m in a selected body of crystalline rock with low hydraulic conductivity. The canisters

Fig.1–4. Standing from the left: Lars Werme, Fred Karlsson, Gerold Lange, Hans Forsström and Anders Bergström. Sitting from the left: Anders Hult, Harald Åhagen, Hans Carlsson, Lars B Nilsson, Lilian Kumlin-Blomqvist, Birgit Schultz, Olle Degerman, Åke Nässil and Tönis Papp. Missing: Mary Ek and Lotta Bengtsson

are lowered into boreholes drilled from the bottom of small tunnels and then surrounded by a backfill of a sand-bentonite mixture. The tunnels and shafts are finally also filled with a mixture of sand and bentonite.

KBS-2 deals with spent nuclear fuel that has not been reprocessed. In this alternative as well, it is assumed that deposition takes place 40 years after the fuel is taken out of the reactor. As in the case of vitrified high-level waste, intermediate storage - in this case in water pools, CLAB, see Figure 1-2 - will simplify the following steps in the handling chain. Due to the much higher content of long-lived radionuclides in the spent fuel, a copper canister with 200 mm thick walls: see Figure 1-5 is proposed instead of the lead-titanium canister

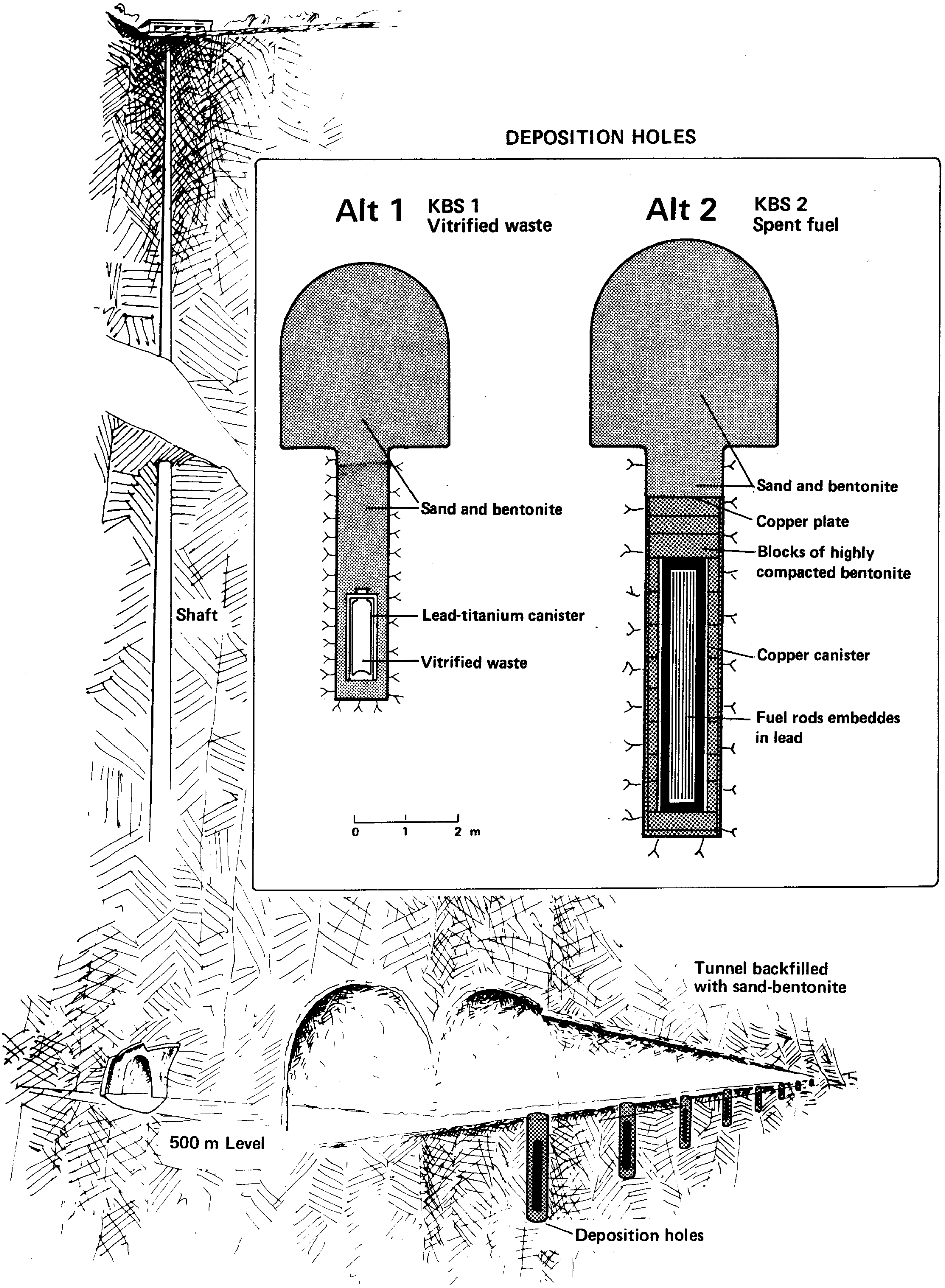


Figure 1-5. Final repository for high level radioactive waste according to the KBS-1 and KBS -2 concept

for vitrified waste. The proposed final repository is in principle very similar to the repository for vitrified waste. The differences are that the deposition holes are drilled at greater distances and that the backfill in the holes consists of highly compacted pure bentonite instead of a mixture of sand and bentonite.

1.4 PROPOSED CHANGES IN LEGISLATION

In 1979 a special investigator was appointed by the Government to study and submit a proposal on how to organize and finance the management of the nuclear wastes in Sweden. His suggestions were published at the end of May 1980 (SOU 1980:14).

The main principle in the proposal is to charge the utilities, through a jointly owned company, with the responsibility for all the handling, treatment and disposal of the waste, including the necessary R&D work. The long-term goals, as well as the annual planning and budget, would have to be approved by a supervisory government agency. This agency would also have the funds to conduct supplementary investigations and research of its own.

Since some of the costs for disposing of the waste will arise long after the reactors have been taken out of operation, the investigator also sketches the outlines of how funds should be reserved for these costs.

The intention is that the utilities should reserve funds according to a plan approved by the above-mentioned supervisory agency, which would also approve methods for maintaining the real value of the funds and an adequate cash flow.

The supervisory agency shall be independent of the agency responsible for the licensing and inspection of nuclear facilities, The Swedish Nuclear Power Inspectorate, as well as from the agency responsible for the licensing of radiological work, The National Institute of Radiation Protection.

Prav, the National Council for Radioactive Waste, is proposed to be dissolved.

The Government has circulated the investigator's proposals for comment to those concerned and is expected to submit a proposed bill along these main lines to Parliament in the spring of 1981.

1.5 BUDGET

The KBS budget for 1980 was SEK 26.5 million (1 US$ = 4.3 SEK).

The budget for 1981 is SEK 38 million.

The budget for 1981 is based on the assumption that the new organization mentioned in Section 1.3, will be put into effect on July 1st and that KBS will then assume responsibility for most of Prav's present activities.

2 THE SCIENTIFIC REVIEW OF KBS-2

The Stipulations Act gives two alternative bases for the Government to give a permission to fuel a new nuclear reactor. The first was that the reactor owner had presented an acceptable agreement for reprocessing and also a plan for completely safe final storage of the high-level radioactive waste from the reprocessing KBS-1. According to this alternative four reactors have been granted permits to fuel.

The other alternative in the law is that the reactor owner shows that a completely safe final storage of spent unreprocessed nuclear fuel is possible to effect. The KBS-2 report "Final Storage of Spent Nuclear Fuel" is compiled to fulfil the requirements in the law regarding spent fuel. Until now no application for permission to fuel has been made based on this alternative.

Due to the present limitations on reprocessing capacity, the spent fuel alternative is the only option for fuelling permits during the next few years. This does, however, not imply that the spent fuel must be disposed of as waste. The law requires the feasibility of one available disposal method to be shown before fuelling is permitted.

In 1979 the Department of Industry sent the KBS-2 report for a scientific review to experts both in Sweden and abroad. In May 1980 the last review report was received, and a total number of 19 Swedish and 11 foreign reviewers have now given their views on the document. KBS was given the opportunity to comment on the material. The KBS comments will be submitted to the Department of Industry during 1981.

Some main issues dealt with in the review are summarized below.

- The degree of safety demanded by the expression complete (or absolute) safety has been regarded as unscientific by many reviewers. Such a level of safety is not possible to achieve in any human activity.

- The multibarrier concept has been considered favorably, as well as the freedom to choose within wide limits the dimensions of engineered and natural barriers.

- In some cases the KBS conceptual design has been seen as a description of a repository to be constructed. Consequently, the safety margins have been considered to be far beyond what would be required by normal safety standards. It should be stressed that the aim of the report is to prove the feasibility to achieve an extremely high degree of safety for a final repository.

- Some doubts have been expressed on the long-term ability of the hostrock to confine the buffer material as well as the methods to measure the hydraulic conductivity of the bedrock.

- The service life of the copper canister is generally considered to have been chosen too short and/or the canister walls unnecessary thick.

- There is no disagreement on the choice of dose limits that could be accepted. Many reviewers, however, have been of the opinion that dose calculations for periods millions of years in the future are meaningless.

- Many and valuable suggestions have been given on modifications of methods and concept. They will be considered in the future work of KBS.

To sum up the review gives the general impression that there is scientific agreement that the presented barrier system composed of both natural and engineered barriers can provide an extremely safe final storage and that such a repository can be constructed with present-day technology.

3 MATERIALS

3.1 WASTE FORMS (HLW)

KBS activities during 1980 have been concentrated on vitrified high-level waste. The glass composition from the process to be used by the French Cogema Company will be decided within the next few years. It is therefore important to coordinate KBS efforts with foreign activities pertaining to the La Hague process.

The corrosion resistance of spent fuel is much more difficult to assess than that of glasses. Furthermore, the technique of using simulated waste loads is not readily applicable to this waste form. However, it has been concluded that it is unlikely that enough new information on the durability of UO_2 will be gained at a pace allowing a revision of this part of the safety analysis in KBS-2 before 1983.

3.2 VITRIFIED HIGH-LEVEL WASTE

A three-year research program for HLW glass was prepared in 1980. The objective of this program is to gain deeper insight into the mechanisms governing the corrosion of HLW glass under the envisaged repository conditions in order to provide a better basis for assessment of the durability of the glass. Glass composition development activities have been cut back and future efforts will be concentrated on the evaluation of two different borosilicate glasses. The composition of the glass from the La Hague plant has not been finally determined, and the two simulations (Table 3-1) have been chosen to cover a span within which the final HLW glass composition will most probably be found.

In order to ensure reasonably complete coverage of the field of glass durability problems in a comprehensive way, an international glass reference

Table 3-1 Simulated waste glass composition (% by weight)

	SiO_2	Al_2O_3	Na_2O	B_2O_3	UO_2	Fe_2O_3	ZnO	Li_2O	FP[x)]
ABS 41	52.0	2.5	9.4	15.9	1.66	3.6	3.0	3.0	9.0
ABS 39	48.5	3.1	12.9	19.1	1.66	5.7			9.0

x) Fission product oxides

panel to KBS has been formed and had its first meeting in September, 1980. The participants and their affiliations are listed below:

R Carlsson, The Swedish Silicate Research Inst
L L Hench, The University of Florida
H P Hermansson, Studsvik Energiteknik AB
J O Isard, The University of Sheffield
N R Jacquet-Francillon, CEA/Marcoule
T Lakatos, The Swedish Glass Research Inst
I Neretnieks, The Royal Inst of Technology
L Werme, KBS

It is intended that this panel should meet at regular intervals to discuss the results obtained in the studies and to propose a course of action for subsequent work.

3.2.1 Glass Corrosion

An investigation of the durability of one silicate-alumino-calcium (SAC) glass in both vitreous and ceramic states and four borosilicate (ABS) glasses has been carried out for KBS at the University of Florida. Of the borosilicates two were soda-borosilicates with high and low Fe_2O_3content, respectively, and two were zinc-borosilicates, also with both high and low Fe_2O_3 contents. The experimental method used was static corrosion, similar to the MCC-1 (Battelle PNL, Materials Characterization Center) test. Infrared reflection spectroscopy was used to investigate corrosion attacks on the glass surfaces. The findings of the study were: The zinc borosilicates are superior to the soda borosilicates and have 5-10 times lower leach rates. Increasing the Fe_2O_3 content in either glass improves leach resistance by a factor of 3-5. The silicate-alumino-calcium (SAC) glass has better leach resistance than soda borosilicate glasses, but when it forms a glass ceramic, its leach resistance is greatly reduced.

It has been concluded that the SAC type glasses offer no substantial advantage over borosilicate glasses, and no further studies of durability will be conducted.

A study to determine the effects of Stripa granite on the surface reactions of the glasses has also been carried out. In this experiment, glass specimens were corroded in the presence of granite with different ratios of specimen surface area to solution volume. Granite specimens alone were also corroded. The large variation in micro-structure of the Stripa granite makes it difficult to understand its corrosion behavior. This variaton also appears to lead to differences in the corrosion behavior of glasses exposed to the granite. Some of the phases present in the granite show leach rates in the same range as the better zinc borosilicate glasses. A more detailed surface study of corrosion-induced changes in the granite micro-structure is clearly needed in order to understand the corrosion of granite and its effect on the corrosion of glasses. Judging from the results of the present study, however, the better zinc borosilicate glasses seem to be relatively insensitive to the presence of granite. A final report on this study will be published in 1981.

The effect of bentonite on the corrosion of glasses is also under investigation and results will be published in 1981.

A program for the investigation of the effects of ionic species in the groundwater on glass durability has been drawn up and will be executed at Studsvik Energiteknik AB during 1981.

3.2.2 Surface Analysis

Surface analysis is an important part of a glass corrosion study and provides essential information, in addition to that gained from solution analysis. Based on a screening of available and practical surface analysis methods, a choice has been made to use infrared reflection spectroscopy as the routine method. When more elaborate methods are called for, ESCA, Auger and Secondary Ion Mass Spectrometry (SIMS), will be used at laboratories where these services are available on a commercial basis.

3.2.3 Phase Separation

The separation of a molybdate-rich phase in the waste glass has been discussed. Other forms of phase separation in the borosilicate glass may also occur during the slow cooling of the molten glass.

Investigations of these phenomena were initiated in 1980 and are being conducted at The University of Uppsala and at The Technical University of Denmark.

Some findings were reported at the Annual Meeting of the Materials Research Society in Boston in 1980: see Appendix 1. Final reports will be published in 1981.

3.2.4 In Situ Experiments at Stripa

In situ tests, where specimens of simulated waste glasses will be buried in boreholes at the Stripa Mine, will be performed with professor Larry L Hench of The University of Florida as scientific advisor. The objective of the experiment is to evaluate the validity of laboratory experiments and to model in the most realistic way the actual interactions in the glass-canister-bentonite rock system. Simultaneous laboratory experiments modelling the interactions will also be performed and are in some cases already in progress. The intention is to extend the in situ tests to a 24 month period or more. An experimental set-up, using miniature canisters with bentonite buffers, is shown in Figure 3-1.

3.3 SPENT FUEL

No activities in 1980.

3.4 CANISTER MATERIALS

3.4.1 Titanium/Lead Canister

Corrosion testing of unalloyed titanium has previously (KBS TR 79-14) been carried out by Studsvik Energiteknik. The tests, using gravimetric techniques, showed very low oxidation rates (0.01-0.1 μm/year). To complete the corrosion studies, surface physical methods have now been employed to investigate the build-up of the protective TiO_2-layer in water with a high oxygen content and water with a low oxygen content. Surface chemistry will also be monitored in a bentonite environment over a three-year period. This work is being done at Chalmers Technical University, Gothenburg.

3.4.2 Alumina Canister

Extensive corrosion tests have been carried out on the alumina (corundum) canister to provide a basis for corrosion evaluation by the Swedish Corrosion Institute (KBS TR 80-15). Several methods have been used, such as atomic absorption and spectrofluorometric measurements of dissolved alumina, gravimetry and ^{15}N-hydrogen profiling. ESCA was also employed

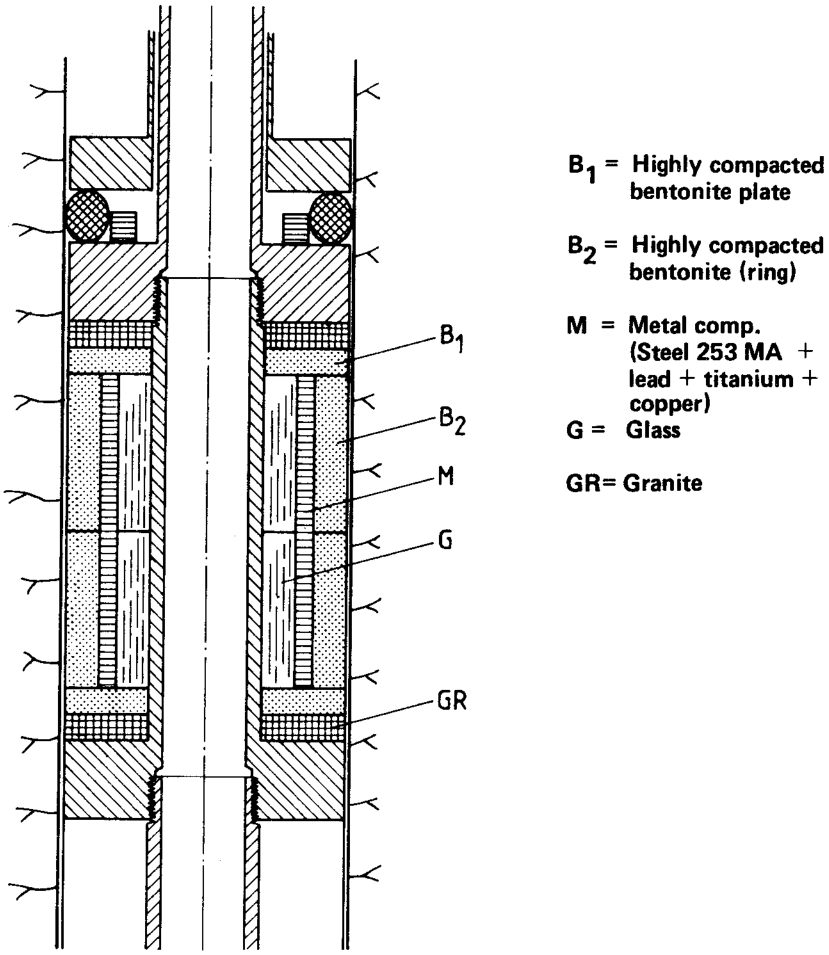


Fig.3–1. The experimental setup for investigation of glass corrosion under realistic conditions in situ.

in the study of corrosion layers on alumina exposed to synthetic groundwater.

Groundwater with dissolved SiO_2 and multivalent metal ions is expected to dissolve less Al_2O_3 than pure water does, since poorly soluble compounds may precipitate on the Al_2O_3 surface. In agreement with this, measurements made by Studsvik Energiteknik revealed that a first stage with weight loss is followed by a second stage with weight gain, i e deposition of minerals on the specimen surface. The ESCA study showed a surface formation of chiefly $Mg(OH)_2$. These layers were up to 7500 Å thick (specimen exposed for 268 days). However, the surface products were not evenly distributed, occurring as more or less interconnected modules.

The nuclear reaction:

$$^{15}N + {}^{1}H \longrightarrow \alpha + {}^{12}C + \gamma \ (4.43\ MeV)$$

has a very narrow resonance at 6385 keV (laboratory energy). Since the energy of the ^{15}N ions decreases with depth of penetration in a well-known way, this reaction can be used to measure the depth of hydration. For simplicity, the depth of hydration is defined as the depth where the content of hydrogen has decreased by 50%.

The corrosion rates found are given in Table 3-2. On the basis of these results and results obtained by The University of Western Ontario, the Swedish Corrosion Institute has concluded that the corrosion rate will be less than 0.1 μm/year and probably one or several orders of magnitude less.

Table 3-2 Corrosion rate for alumina in water

Method	Temp	Corrosion rate nm/year at pH 7	pH 8.5	pH 9.3
Al in the water, atomic absorption	90^{o}	10-12	-	10-12
Gravimetry	80^{o}	-	68	-
Spectrofluorometry	80^{o}	-	16	-
^{15}N-hydrogen profile x)	90^{o}	34	-	22

x) See text for definition of depth of hydration

Another factor of great concern for the lifetime of a ceramic canister is the validity of extrapolations of crack growth and strength data. The prevention of delayed failure in the alumina canister requires consideration of two potential fracture sources: direct extension of existing defects and extension of cracks formed by corrosion reactions.

It has been concluded that the former can be avoided by subjecting the canister to proof testing at a stress level of 150 MPa and using acoustic emission technique to detect crack growth. The latter can be avoided by ultrasonic testing, where canisters with inclusions larger than 100 μm within a 100 μm deep surface zone are rejected (KBS TR 80-15).

4 ENGINEERED BARRIERS, TECHNOLOGY

4.1 CANISTERS

4.1.1 Copper Canister

In the KBS-2 concept, copper is used as an engineered barrier in a repository for unreprocessed spent nuclear fuel: see Figure 1-5. Design studies during 1980 have been limited to welding technology using existing electron beam equipment. Blocks of copper 100 mm thick are welded together. The welded material will be tested with reference to homogeneity, mechanical and chemical behavior.

An alternative copper canister design is based on encapsulation of spent fuel in copper by a hot isostatic pressing process without lead filling and welding of lids, but with copper surrounding each rod of spent fuel. The copper between the fuel rods is charged in the form of a powder that is transformed into a solid mass during the hot isostatic pressing. The feasibility of using such a process is being discussed with ASEA's High-Pressure Laboratory.

4.1.2 Alumina Canister

In KBS-2, Appendix 1, a status report was presented on the use of an alumina canister for the final storage of spent nuclear fuel.

The Swedish Corrosion Research Institute has evaluated the service life of a canister fabricated by means of hot isostatic pressing. In the final report (KBS TR 80-15), is concluded that the canister's corrosion rate will be less than 0.1μm/year, probably one or two powers of ten less, and it would be possible to eliminate the risk of delayed fracture by only accepting canisters that pass production inspection including test loading and acoustic wave examination. A canister made of high-purity Al_2O_3 with a wall thickness of 100 mm that has passed such production inspection is estimated to have a service life of hundreds of thousands of years and probably considerably longer.

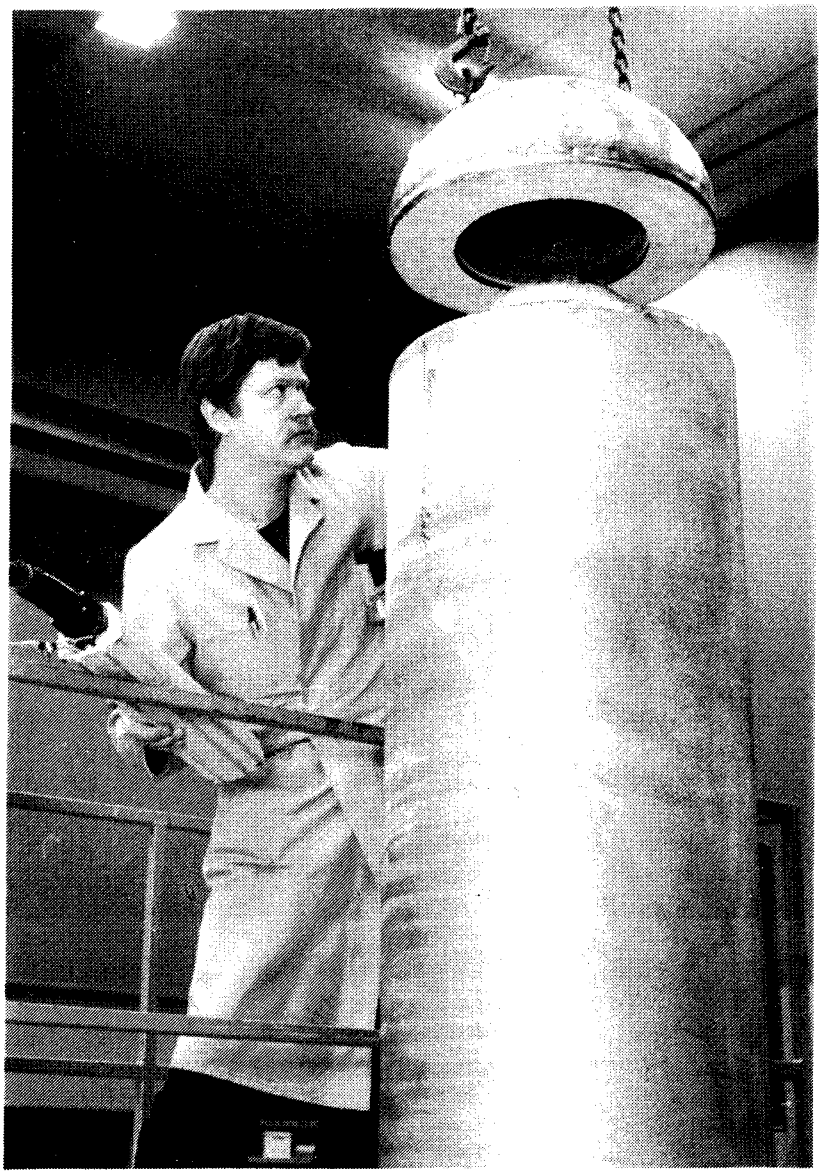


Fig.4–1. Lid being fitted to full-scale, OD 0.5 m by 3 m long synthetic corundum canister.

In a development program supported by KBS, the ASEA High-Pressure Laboratory at Robertsfors produced an alumina canister with an outside diameter of 0.5 m and a length of 3 m. Figure 4-1. During 1980, ASEA-ATOM and ASEA have jointly carried out a preliminary study of the technical feasibility of encapsulating spent nuclear fuel in alumina by means of hot isostatic pressing (HIP).

The study indicates that the spent nuclear fuel can be encapsulated in an alumina canister with a length of 5.6 m and a diameter of 0.6 m, without the lengths of the fuel rods having to be reduced: see Figure 4-2.

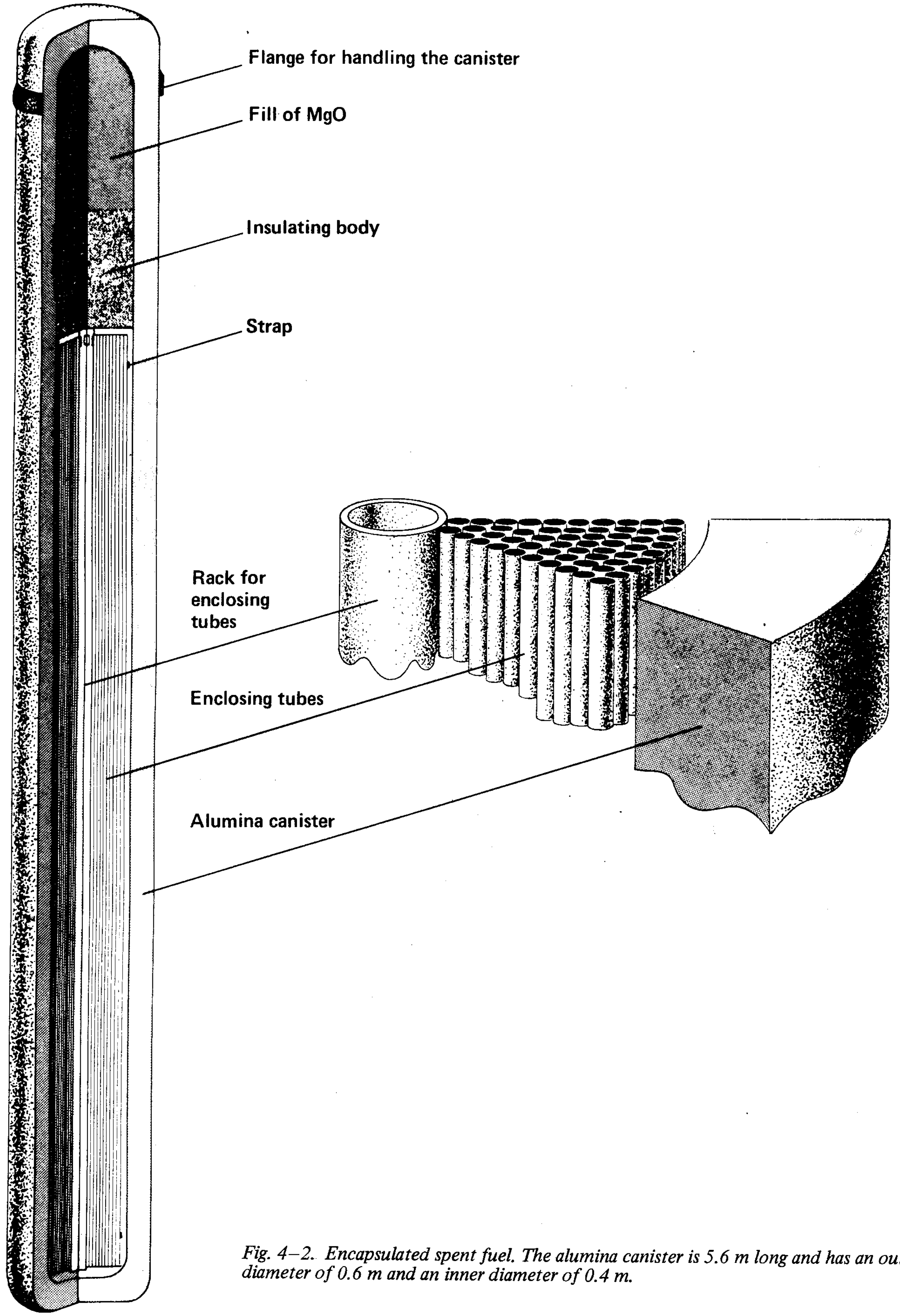


Fig. 4–2. Encapsulated spent fuel. The alumina canister is 5.6 m long and has an outer diameter of 0.6 m and an inner diameter of 0.4 m.

4.2 BUFFER AND BACKFILL

4.2.1 General

In KBS-1, bentonite/quartz mixtures were used to fill the annulus around the canisters in the deposition holes, and in KBS-2, highly compacted bentonite was used: see Figure 1-5.

During 1980, studies have been conducted at the Department of Soil Mechanics, University of Luleå, in order to gain a deeper understanding of swelling properties, permeability and water uptake associated with the hydraulic and thermal gradients in bentonite-based buffer substances. Wyoming and Bavarian bentonites are being tested. Postglacial illitic clay is being studied as well.

The chemical stability of bentonite and illitic clay is being studied in cooperation with the State University of New York and the University of Western Ontario.

4.2.2 Swelling Pressure

Using rigid oedometers, the swelling pressure in sodium and calcium bentonites has been measured at temperatures in the range of 20 to 90°C under salt- and fresh-water conditions. A long series of swelling pressure determinations has been reported (KBS TR 80-13). The results indicate that pore water chemistry is not a determinant of the swelling pressure when bulk density is very high.

Further measurements on bentonites of lower density are being performed.

4.2.3 Hydraulic Conductivity

The flow rate of water through highly compacted bentonite confined in oedometers is being measured using high hydraulic gradients and bulk densities in the interval 1.8 to 2.2 ton/m^3 and at temperatures of 20°C and 70°C.

The results indicate that the hydraulic conductivity of Na and Ca bentonite is in the range of 10^{-14} to 10^{-12} m/s when density is high (> 1.8 t/m^3) /KBS TR 80-11/. The Ca bentonite was 2 to 5 times more permeable than the Na bentonite. At around 70°C, hydraulic conductivity is approximately 5-10 times higher than at 20°C for both clays. Tests are presently being conducted on Na- and Ca-bentonites of lower densities.

Crude bentonites are being leached in order to get water samples for determination of leachable organic content.

4.2.4 Water Uptake and Swelling Capacity

Water uptake and migration and swelling characteristics of unsaturated and saturated highly compacted bentonite have been studied on a laboratory scale. The test results and experience from excavation (KBS TR 80-11) in rock containing seams of montmorillonite clay clearly demonstrate the swelling capacity of bentonite of high density when it is exposed to water, and its property of self-healing to a homogeneous state under confined conditions.

4.3 BOREHOLE AND SHAFT PLUGGING, ROCK JOINT SEALING

Boreholes in hard rock can be sealed by filling them with highly compacted blocks of bentonite. A technique using perforated metal pipes to fill up core-drilled holes of small diameter has been proposed and tested on a laboratory scale: see Figure 4-3 (KBS TR 80-11). Further testing will be conducted on a larger scale.

The isolating capacity of the clay/rock system surrounding the canisters will be largely improved if water-bearing joints in the rock can be sealed effectively. This could be achieved by filling the joints with clay to a depth of a few meters from the deposition hole. Even a clay gel of moderate density would reduce average diffusivity and conductivity to such low values that there will be practically no water transport to or from the deposition hole when the buffer material in the hole has been fully saturated. The problem is to seal joints in the range of 0.01 - 0.1 mm. One proposed technique based on electrophoresis has been tested successfully in the laboratory (KBS TR 75). The method is being tested on a large scale in a tunnel at Forsmark where a verticalhole 1 m in diameter and 5 m in depth in hard rock has been used. Bentonite slurry under the influence of an electrical potential field or under pressure was used. It was found that in this rock, with a measured hydraulic conductivity of 10^{-9} - 10^{-8} m/s, bentonite migrated several meters into joints, without lowering the initial conductivity because of the fissure widening produced by pressure injection.

Fig.4–3. Blocks of highly compacted bentonite in a perforated metal pipe to be used for plugging of boreholes.

4.4 ROCK MECHANICS AND SEISMOTECTONICS

The thermomechanical response of the rock mass, stress conditions in the native rock and the swelling pressure of the compacted bentonite in the deposition hole could affect the widths of joints and thereby the conductivity of the rock adjacent to the deposition hole. Geotechnical two-dimensional numerical modelling and analysis of rock mass deformation have been conducted (KBS TR 80-02) using Barton and Choubey's joint model and parameters chosen to correspond to typical granitic rock, particularly for the Stripa Mine. The results indicate small displacements and no stresses exceeding strength.

Seismicity is low in Sweden and there have been few opportunities to study earthquake mechanisms. On December 23, 1979, a seismic event took place in a zone of very low seismicity in eastern central Sweden (Bergshamra). For Swedish earthquakes, the epicenter was located at unusually shallow depth. Using available data from seismograms and field survey in the epicentral area, the dynamic source parameters were evaluated and they indicated a focal depth of about 2 km, fault length of 0.8 km and an average displacement of 0.5 cm. (KBS TR 80-09).

More data are needed in order to provide explanations in seismotectonic terms of the origin of earthquakes in Sweden and deformations and stress conditions in the adjacent rock.

4.5 CONSTRUCTION METHODS

Technical and economic aspects of drilling of vertical holes are addressed in a state-of-the-art report (KBS TR 80-12). It is concluded that, at the present time, equipment is not available for drilling deposition holes 1 m or 1.5 m in diameter and 8 m in depth. On the other hand, it is claimed that existing equipment and technique will after some modifications, have the capability to drill such holes at a reasonable cost.

5 CHEMISTRY

It is possible to distinguish three main barriers that will retard the transport of oxidants to the canisters and the escape of radionuclides to the environment from a high-level waste repository. The first barrier is the metal canister containing the waste. The bentonite buffer surrounding the canister forms the second barrier and the rock the third.

Groundwater, originally in equilibrium with undisturbed host rock, will enter the repository, equilibrate with the buffer mass and canisters and become the transporting medium for dissolved species. Knowledge of the groundwater chemistry is therefore essential for the investigations of radionuclide sorption, canister corrosion and long-term behavior of the bentonite buffer.

Since all waste canisters will be placed in carefully selected deposition holes, and since recent investigations have revealed the importance of the first few meters of good host rock around the canisters for the retention of oxidants and radionuclides it is convenient to treat the transport phenomena here and in the buffer mass separatly from in the entire rock mass. The transport of oxidants, which determine canister life, is only of interest in this near field, and the transport of radionuclides out of the near field will serve as a source for the calculations of far field migration.

Responsibility for the studies of the transport of radionuclides in the repository and the geosphere is divided between KBS and Prav, with KBS primarily responsible for the near field and Prav for the far field. Since the transport of radionuclides in the far field is very closly related to the KBS work, Prav's work is also briefly described here.

5.1 GROUNDWATER CHEMISTRY

Water samples have been collected at various depths in sections of deep boreholes in the Finnsjö area. Some samples have also been collected at Sternö in southern Sweden: see Figure 6-2. Altogether some 100 samples have been taken from about 25 levels. The samples from the Finnsjö area show large chemical variation. The groundwater in the southern part of this area is normal, but in the northern part it has high salt content at great depth. Samples with an extreme value of more than 5000 ppm of Cl^- have been found. This is a larger concentration than in the Baltic Sea today (ca 3500 ppm at Utö). The sodium, calcium and sulfate concentrations are also high. The content of organic material is being studied at Battelle Columbus Laboratories in the USA. Preliminary results will be published in the summer of 1981 and additional sampling will be performed in two deep holes in the Finnsjö area.

Prav has initiated in situ measurement of temperature, spontaneous potential, pH, redox potential and water conductivity in open boreholes. The results are often in good agreement with analyses of samples taken during pumping from sections of the holes. Temperature measurement can be used as an indicator of groundwater movement in the open holes (3).

A computing model for groundwater dating with C^{14} has been published together with a brief review of methods for dating of groundwater (KBS TR 80-08). A sensitivity analysis shows that some parameters have a great influence on the calculated age.

Dating of groundwater from the Finnsjö area shows C^{14}-ages of about 5000-6000 years in the southern part and up to 12 000 years in the high salinity samples from the northern part.

Dating is an aid in evaluating flow times. For this, however, it is necessary to have knowledge of dispersion and the possible influence of chemical mechanisms, for example of diffusive exchange of water in micropores (4) or sorption on calcite in fractures (5). This sorption may be considerable and is caused by ion exchange with carbonate minerals and thus gives rise too high C^{14}-ages.

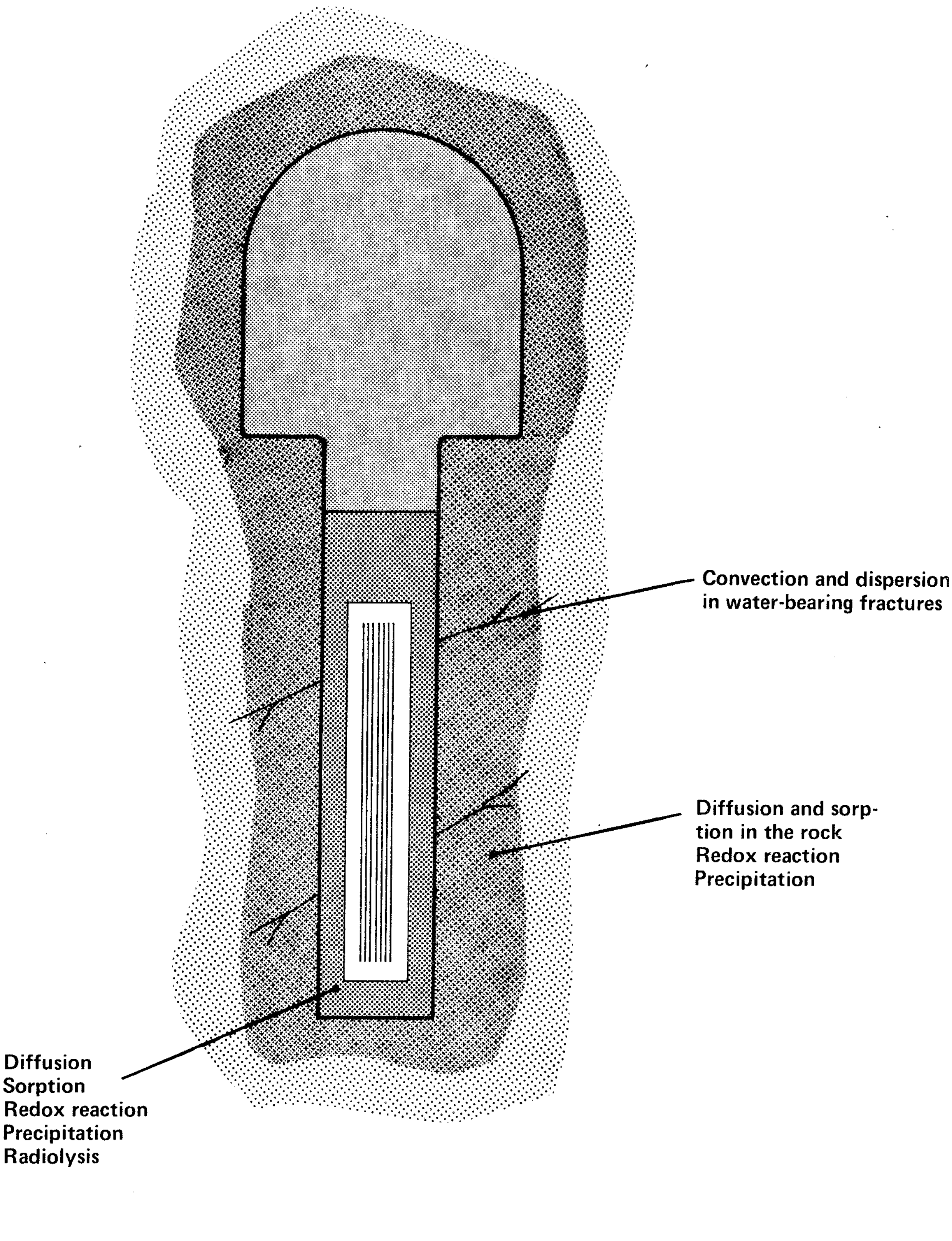


Fig.5–1. Mechanisms of importance for the transport of radionuclides and oxidants in the near-field part of the deposition hole in the spent fuel concept. The copper canister is embedded in highly compacted bentonite and the tunnels filled with a bentonite-sand mixture.

5.2 NEAR FIELD TRANSPORT OF DISSOLVED SPECIES

5.2.1 A Migration Model for the Near-Field

This problem has already been treated in a simplified manner (KBS TR-79). However, a new three-dimensional model is now being developed which incorporates the mechanisms which are most important for the transport of the oxidants and radionuclides, without being unwieldy. The mechanisms not previously included are the following

- precipitation
- influence of chemical reactions
- influence of radiolysis
- diffusion and sorption in the rock

The method of finite element analysis will be used to solve the model equations.

Further investigations have been initiated in order to supply this new model with realistic input data.

5.2.2 Ion Migration in Compacted Bentonite

Ion diffusion experiments on highly compacted bentonite have revealed the need for refined experiments. Measurements of Cs-134, Sr-85 and I-125 diffusion through highly compacted (2.1 t/m^3) Na-bentonite with the original oedometers have been concluded and a new type of euipment is under construction for more comprehensive diffusion experiments to be conducted in 1981.

5.2.3 Organic Content of Bentonite

Leachates from compacted samples of bentonite are being analyzed in cooperation with Battelle Columbus Laboratories.

5.3 FAR FIELD TRANSPORT OF RADIONUCLIDES

5.3.1 Diffusion into the Rock Matrix

Studies initiated by KBS show that the diffusion of radionuclide into the rock mass might be a dominating factor for the retention of radionuclides (KBS TR 79-19).

Sorption and desorption of Sr, Cs and some non sorbing compounds on crushed rocks and pieces of rocks are being studied in laboratory in order to substantiate the diffusion model x). Preparations are being made for an in situ experiment with non sorbing agents in the Stripa mine. x) Investigations of diffusion through weathered rocks or fracture minerals are considered. A literature study of geological evidence for diffusion has been begun.

5.3.2 Equlibrium Reactions between Groundwater and Radionuclides

Equilibrium reactions for U(IV) and U(VI) with CO_3^{2-} in groundwater are being studied. Redox diagrams for U, Np and Pu in an artificial groundwater have been calculated. All actinides will most probably have a valence of +4 in groundwater. The investigations suggest that earlier published equilibrium constants for U(IV)-hydroxide complexes have to be revised.

5.3.3 K_d-Values

Sorption of Am, Np and Pu on various minerals has been studied. Sorption of Cs, Sr and Am(III) on freshly sawn, polished and artificially weathered rock surfaces is being studied. Sorption of Am(III) on non reactive or nearly non reactive surfaces of TiO_2 and Al_2O_3 is being performed. x)

The behavior of THO (tritiated water), sodium-lignosulfonate, SCN^-, Sr^{2+}, Cs^+ and Eu^{3+} in natural fractures (200 x 300 mm) is being studied by means of pulse tests and continuous injection. x)

5.3.4 Redox Potential

The influence of some common minerals on the redox potential of groundwater and the redox buffer capacity of the hard rock is being studied. x) This work is being supported by the National Swedish Research Council, NFR.

5.3.5 In Situ Migration Experiments

Experiments on the retention of radionuclides are being conducted in the laboratory on crushed material and rock pieces, on natural fractures in drill cores and in a single fracture in the Stripa mine: see Chapter 8.

---

x) This work is being handled by Prav

The results of the different experiments will be compared with the results of in situ experiments in a larger volume of rock at Studsvik. Preparatory studies have been made on site hydrogeology and chemistry and background radioactivity of the bedrock and groundwater (6,7). Injection methods have been developed and tracer experiments have been started.[x)] Methods and instruments are being developed in connection with tracer experiments in the Finnsjö area: see Section 6.2.1.

---

x) This work is being handled by Prav

6 GEOLOGY AND HYDROLOGY

6.1 GENERAL

Investigations of groundwater transport in fractured crystalline rock aim at a better understanding of the following parameters

- Velocity
- Direction
- Dispersion
- Quantity
- Long-term changes in the geologic-hydraulic system

Figure 6-1 illustrates the investigations being performed.

The efforts initiated in 1979 have been continued during 1980, with an emphasis on the following topics

- Development of measuring methods and instruments
- Collection of field data
- Development of analytical models

Most of the investigations are being conducted in the Finnsjö area: see Figure 6-2. This area is well suited for these studies, as it is well known from investigations connected with KBS-1 and KBS-2. Figure 6-3 shows the Finnsjö area with seven deep boreholes and shallow holes for piezometric measurements. The area has been chosen as a model area for fundamental studies in the field of hydrogeology. Studies are also conducted at Stripa (see Chapter 8) and at Studsvik. The latter investigations which were initiated by Prav, are being conducted in preparation for planned in situ experiments concerning the transport of radionuclides.

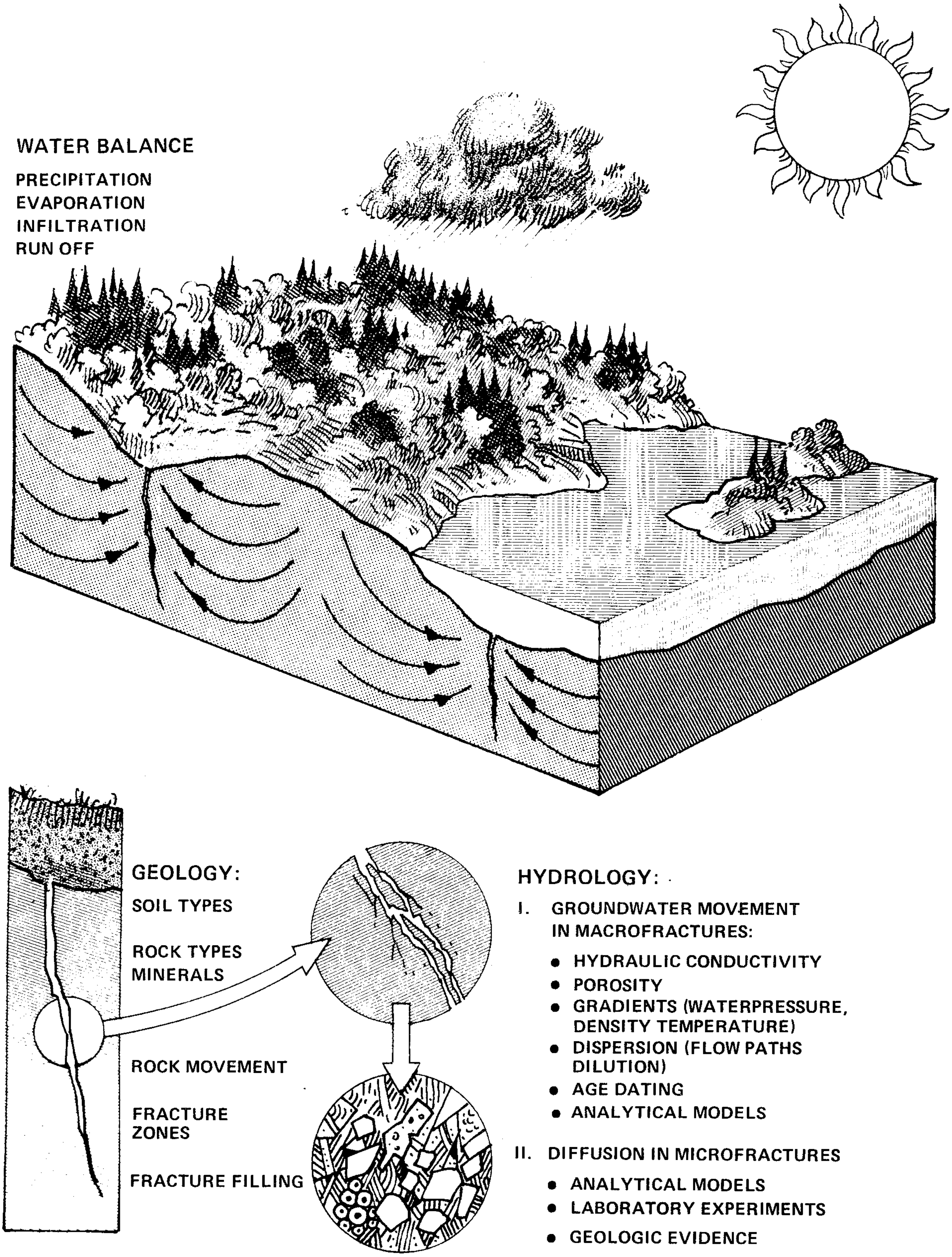


Fig.6–1. Hydrogelogical investigations.

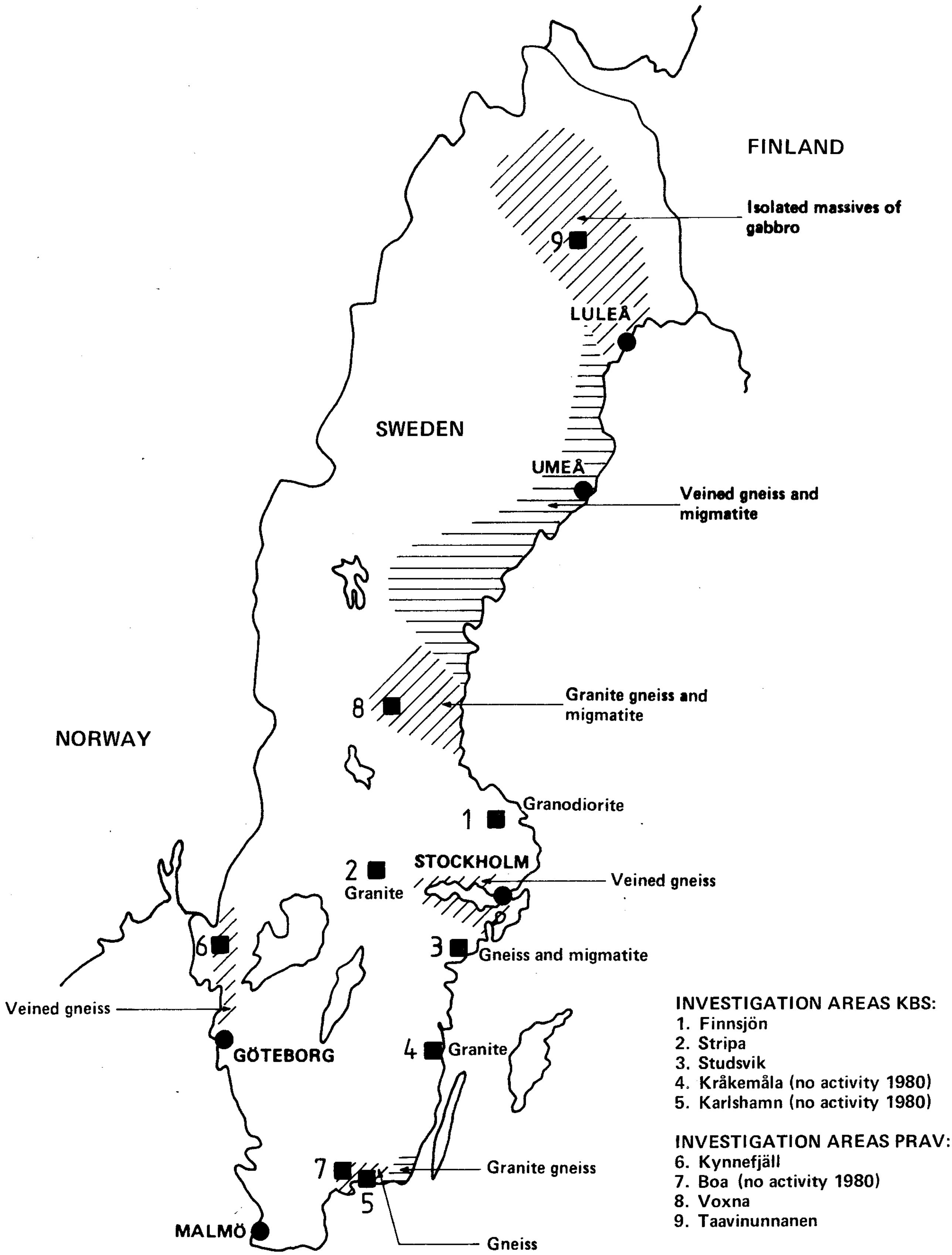


Fig.6–2. Investigation areas

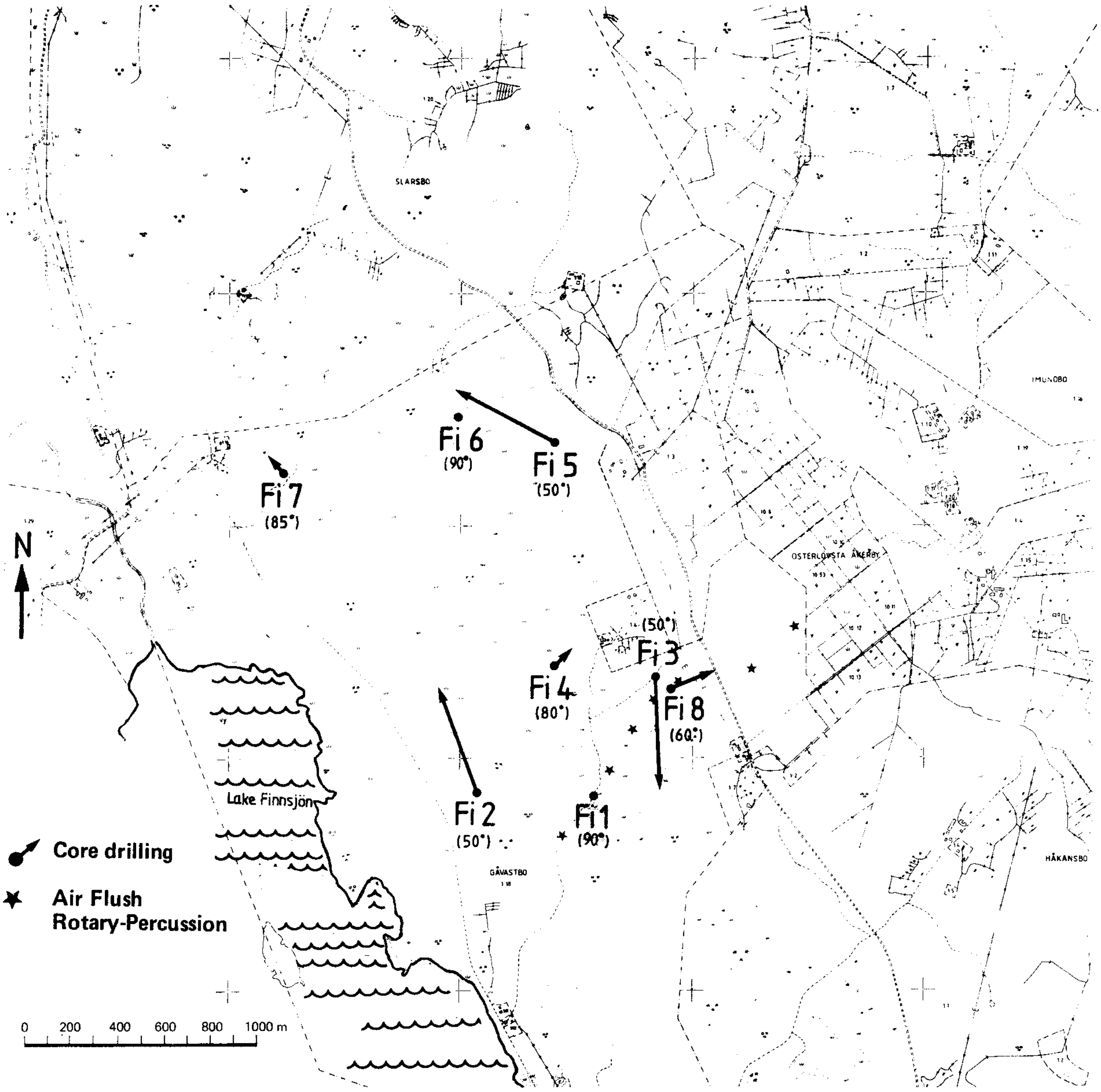


Fig.6–3. The Finnsjö area.

6.1.1 Measuring Methods

A literature survey on the influence of well storage and skin factors in transient tests has been performed.

The literature research will continue concerning the evaluation of pulse and slug tests.

Botanical investigations are being conducted in the Finnsjö area to study the possibility of using variations in the type of vegetation, as revealed by aerial photos, for mapping inflow and outflow areas of groundwater.

Surface water and groundwater have different conductivities. Studies of the variation of conductivity of surface water at different places in the Finnsjö area are being conducted to investigate the possibility of using this as a method for mapping outflow areas.

Prav has initiated studies to develop methods for mapping fracture zones between boreholes and between boreholes and the groundsurface (cross hole technique). Seismic, electromagnetic and electrical methods are being studied.

Various geophysical methods are being used for borehole investigations. An evaluation of the availability of a combination of different methods has been carried out (8). This survey is being continued with an evaluation of geophysical methods in combination with geological and hydrological methods for investigating boreholes.x)

6.1.2 Instrumentation

At present, there are separate instruments for water pressure tests and pumping tests. A pump to replace the separate instruments and to improve handling for use in 56 mm boreholes down to a depth of 800 m is being developed. The capacity of the pump will be about 2 l/min at a differential pressure of 2 atm. It will be used between packers with a variable section length. The system is lowered by a multitube which contains all hydrological and electrical connections between the pump and the surface. Measured data can be recorded directly on tape. The equipment will be tested in the field in the beginning of 1981.

A groundwater sampler is being developed and will be tested in the summer of 1981. The sampler can be directed by time or changes in the pH, Eh, S^{2-} content, O_2 content or conductivity of the groundwater. These parameters are automatically measured and recorded. Gases dissolved in the groundwater are also sampled. A report on the function of the pump and groundwater sampler will be submitted in 1981.

The methods used so far in the groundwater experiments do not provide direct information on the natural groundwater movement in fractures. An instrument is therefore being developed for measurement of the natural groundwater flow in a section between packers in a borehole: A dye is portioned out and mixed with the water between the packers and a colorimeter measures the decrease of color intensity caused by the natural groundwater flow. Testing will start in 1981.

---

x) This work is being handled by Prav

Instruments for measuring changes of pressure with depth and time in boreholes have been developed: see Section 6.2.2.

6.2 COLLECTION OF FIELD DATA

Field data collected from seven deep boreholes in the Finnsjö area and from five deep holes at Sternö up to 1980 have been published (KBS TR 80-10 and 80-01). An evaluation of these data and their compatibility with theoretical models will be completed by the summer of 1981.

6.2.1 Tracer experiments are being performed in a part of the Finnsjö area with various tracers and with pulse and continuous injections. The experiments are related to water pressure tests and pumping tests. In this manner groundwater movement in two dimensions as well as dispersion and porosity in certain fractures, are being investigated. The results will be published in early 1981.

6.2.2 For calculations of groundwater flow times,the groundwater table has conservatively been considered to coincide with the ground surface (maximum gradients).

A profile of boreholes has been drilled in the Finnsjö area for the measuring of changes in groundwater pressure with depth and time. Measurements will be performed in sections of the holes. The profile covers expected inflow and outflow areas: see Figure 6-3. Three of the holes are 400-500 m deep and seven holes are about 100 m deep. Tests will start in 1981. The results will form a basis for modelling variations of flow direction, mixing with various waters and groundwater velocity with time.

6.2.3 Data on precipitation, evaporation, surface flow etc are being collected for studies of the water balance in the area.

6.3 DEVELOPMENT OF ANALYTICAL MODELS

6.3.1 The work on analytical models has continued with one report on the description of the hydraulic properties of fractured rock using the concept of a porous medium (KBS TR 80-65).

A method for measuring the hydraulic properties is described and the properties of radial flow when the flow pattern is changed are discussed.

The various requirements on the procedure used for determining the rock parameters are discussed and the formal apparatus needed for the determination of anisotropy, the time constant, hydraulic conductivity, porosity, rock compressibility and variation in space is developed. The effects of varying fluid properties on the hydraulic properties are shown. The report will serve as a basis for planned 3-dimensional investigations where anisotropy will also be studied.

6.3.2 A mathematical model for the flow of groundwater and heat in fractured rock has been developed (KBS TR 80-19). The model has been developed to study the effect of heat released from a radioactive waste repository on groundwater flowtimes. The model consists of a set of coupled non-linear partial differential equations for heat and groundwater flow. Fluid density and viscosity as functions of pressure and temperature are considered. Changes in fracture widths caused by temperature changes are not considered in this work. The model is based on the continuum approach and the rock is treated either as two overlapping continua, in which one represents the network of fractures and the other the solid blocks, or as a single equivalent medium.

There are numerical solutions for the hydrothermal conditions. The 2-dimensional solutions illustrate the effect of heat under various conditions with regard to topography, permeability, porosity, adjacent fracture zones and temperature decreases with time. Later, a 3-dimensional analysis and application to the Finnsjö area will be performed.

Depending on the natural direction and velocity of the groundwater, the effect of heat can prolong or shorten the flow times. In the theoretical analysis, these changes can be of interest locally. Depending on the decrease of heat with time, the flow direction of an imagined particle can change considerably with time.

6.4 TECTONICS

The investigations described above reveal the hydrogeology of a certain situation. Studies are also being conducted in an attempt to predict future changes.

The reason for the present uplift in almost all of Sweden has been studied with grants from Prav. The results seem to give a conclusive proof of the isostatic origin of most of the uplift, which means that the uplift is caused by the disappearance of the ice cap that formerly covered northern Europe (9).

Studies of seismic events are described in Section 4.4.

6.5 FRACTURE MINERALS

Prav has initiated a study to develop methods for the dating of fractures. A literature survey has been published (10). A study of the formation of fracture minerals shows that calcite in fractures in Stripa and the Finnsjö area has been formed at a temperature higher than 100°C and/or in water with a different chemistry than at present. The natural temperature gradient in the hard rock was higher than today's during the formation. This can be interpreted as indicating that these fractures are very old and were formed before the last ice age. The study also shows that there is a difference in dip between open and healed fractures in one investigated borehole (11).

6.6 SITE INVESTIGATIONS

Prav is responsible for an inventory of various rock types to serve as a basis for the choice of suitable rock types for a future waste repository.

The rock types of main interest are at present gneissic rocks and gabbro.

The initial choice of areas is based on existing geological maps and aerial measurements of the magnetic field. Selected areas are then investigated in detail in the field, after which possible drilling sites are chosen. A survey of sites for investigation has been published by Prav (12). The results of geological, tectonic and geophysical studies at Kynnefjäll have also been published by Prav (13).

So far, three areas have been chosen for more detailed investigations. The areas are Kynnefjäll, Voxna and Taavinunnanen: see Figure 6-2. Planned drillings at Kynnefjäll have been delayed. At Voxna, drillings started just before Christmas, and at Taavinunnanen, drillings will start 1981.

7 SAFETY ANALYSIS

7.1 GENERAL

Assessments of the safety of the disposal system are included in the safety analysis program.

Hitherto the assessments have been highly generic and intended to show that an acceptable level of safety can be achieved with the barrier systems proposed and within the limits of present-day technology. In the decades to come, the assessments will serve as a basis for a general optimization of the barrier system and as a guidance for the selection of acceptable geological sites for a repository for radioactive waste.

In a final stage, the safety assessments performed within the program will provide part of the data necessary to decide whether the disposal scheme is acceptable to society or not.

So far most of the work within the program has been aimed at establishing a broader data base concerning the behavior of naturally occurring radioisotopes in the biosphere.

The emphasis in mathematical models today is to provide an improved model for nuclide transport with groundwater flow in fractured rock. Both diffusion into the solid rock matrix and dispersion along the flow-path are expected to have a great influence on the safety assessment.

A mathematical model for the influence of thermal gradients on groundwater flow has been tested during 1980 and the effect of temperature-dependent permeabilities will be investigated.

An effort to devise models describing mass transport in the near-field around the waste cylinders has been started. The aim is to be able to take into account in the safety assessment, e g:

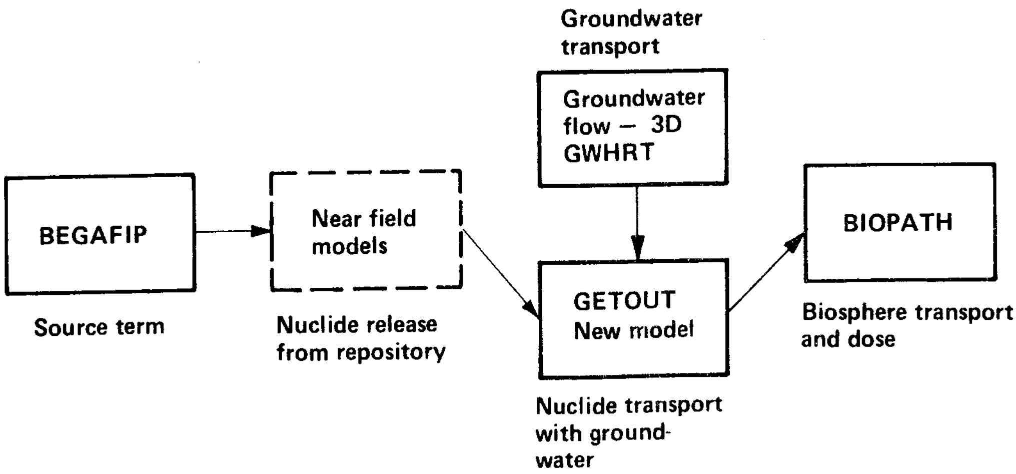


Fig. 7–1. Mathematical models and their interrelationship in safety assessment.

- the low solubility of the corrosion products of the canister
- the ion-exchange mechanisms for the waste species within the buffer zone and
- the precipitation mechanisms related to the change of oxidation state upon diffusion from the oxidizing environment caused by radiation close to the radioactive waste to the reducing environment dominated by the Fe^{2+}-rich normal groundwater.

The work is expected to take many years and is aimed at producing a number of successively more refined models.

7.2 MATHEMATICAL MODELS

The general areas where mathematical models are used as a standard within the KBS are indicated in Figure 7-1, as are the names of the codes and their interrelationship in the safety assessments.

BEGAFIP - a computer code to calculate the amounts of radioactive substances in the repository. The updating of the data library that was started during 1979, has been completed along with the addition of subroutines to calculate neutron sources. Benchmark calculations have been performed to compare BEGAFIP with CASMO. Good results were obtained, even better than with the ORIGEN code. The work is reported in KBS TR 80-20.

NEAR-FIELD TRANSPORT - model for mass transport in the repository is under development: see Section 5.2.1.

GWHRT - a groundwater transport code including effects of decay heat in the repository: see Section 6.3.

GETOUT - a model for the transport of elements with the groundwater flow. Updating performed during 1979 has been reported in KBS TR 80-03.

FAR-FIELD TRANSPORT - a model for transport of elements with groundwater in fractured rock. The program has been modified to be able to handle radioactive decay-chains. Test calculations have been carried out and the results show good agreement with analytical solutions and results from runs with the GETOUT code.

Problems have been encountered with the combination of a decay-chain and diffusion into the rock matrix. The time-step for a stable solution is so short that the calculations tend to become very expensive.

Alternative methods for solving the equations in the computer code have been tested. A direct solution by "a sparse matrix solver" gave good results for the back-slope of the break-through curve. However, the method requires a large memory capacity and will only be practical for one-dimensional cases.

The following reports have been published:
KBS TR 80-23, 80-24 and 80-25.

BIOPATH - a biosphere transport model. The addition of routines for three-nuclide decay-chains and time-variable reservoir sizes for nuclides in the biosphere has been completed and reported in KBS TR 80-17 and 80-18. Further variation analyses have been performed in order to identify the need for better data for the transfer rates between the compartments in the model.

7.3 ENVIRONMENTAL DATA

Two limited studies on naturally occurring radioisotopes are almost completed.

One is on exposure from the Th-229 in beach sediments. Experiments indicate a rather homogeneous distribution of thorium in beach material at the level of around 20-40 Bq/kg. Dose calculations have been performed.

The other study comprised a study of equilibrium levels of U and Ra in soil, groundwater, lake water and lake sediment in Finnsjö Lake in Sweden in order to establish transfer factors for the elements between these compartments. The results give transfer factors between soil and lake water that are around 100 times lower than those used previously. This has revealed the need for more data on background levels of naturally occurring radioisotopes in various environments.

Consequently, a number of studies have been started during 1980 in order to extend the data base on U and Ra.

Data collected by the Swedish Geological Survey on uranium levels in water and sediments (mostly creeks) will be compiled and analyzed statistically with regard to different geological areas. The data has been collected over the past 10 years in connection with uranium exploration in Sweden. The material will be published during 1981.

A number of wells in a national network of groundwater wells are being sampled and analyzed for U, Ra and Rn. The network has been sampled for 10 years with regard to flow and water chemistry to provide background standards on natural groundwater and to study naturally occurring variations. The work is expected to be completed during 1981.

Three springs with well defined run-off beds and high levels of radioactivity have been identified for water analysis. Around 300 samples will be taken at each to define how the Ra and U will be depleted from the water and how they will accumulate in the biota and the sediments downstream of the spring. The samples are planned to be analyzed and the results reported in 1982.

8 STRIPA TEST STATION

8.1 COOPERATIVE PROGRAM WITH US DEPARTMENT OF ENERGY

The investigations that were performed in Stripa in cooperation with the US Department of Energy (DOE) and under the supervision of Lawrence Berkeley Laboratory (LBL) were more or less completed during 1980. Only limited and mainly supplementary studies and measurements are planned to be carried out during 1981.

Some of the data produced by the investigations that were carried out by LBL are still under evaluation. Nevertheless, results are published continuously in a special series of reports. At the end of the year, some 30 reports had been published in this series. A summary of activities during 1980 is given below.

8.1.1 Full-Scale Heater Test

Recording of the cooling of the rock mass surrounding the heaters in the full-scale heater test continued until the end of February 1980, when the experiments were interrupted. The two simulated waste canisters, 3.6 and 5.0 kW respectively, had to that date produced heat for some 400 days and the cooling effects in the rock were recorded for about 200 days. Very good agreement was achieved between predicted and measured temperatures: see Figure 8-1.

The extensometer and stress gauge readings have yielded some puzzling results which reveal the complex problems involved in attempting to predict the thermo-mechanical behavior of a discontinuous rock mass. Further laboratory tests of the thermo-mechanical behavior of the Stripa granite have shown a greater temperature dependence than expected with regard to the elastic modulus E, Poissons ratio ν and the coefficient of thermal expansion α.

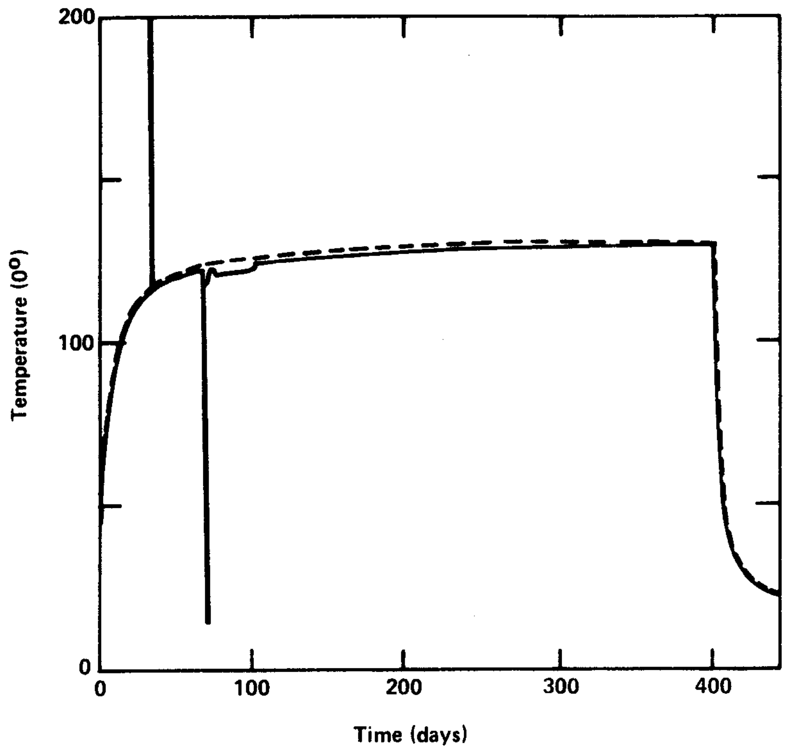


Fig.8–1. Predicted (dashed) and measured (solid) temperatures plotted as a function of time at a radius of 0.4 m from the 3.6 kW heater along the heater midplane. Variations in measured signals at early time was caused by corrosion of stainless steel thermocouple sheath. (From Witherspoon et al).

8.1.2 Time-Scaled Heater Test

A time-scaled experiment was conducted at Stripa in which the times were compressed by a ratio of 1:10, that is, each year of data from the time-scaled experiment is equivalent to 10 years of data from the full-scale set up.

The time-scaled experiment was started in 1978 and completed in late 1979. The data from the temperature and deformation gauges were evaluated in 1980, and again, as in the full-scale experiment, remarkably good agreement between measured and predicted rock temperature was found.

8.1.3 Hydrological Investigations

The fracture hydrology program at Stripa is essentially devoted to an understanding of the hydrological behavior of a complex network of fractures. Three of the most important aspects of this program are:

- assessing directional permeabilities
- a macro-permeability experiment
- geochemistry and isotope hydrology

Basic data on fracture orientation, spacing and continuity have been obtained by careful mapping on the surface, in boreholes and in subsurface openings. A borehole injection test program has also been developed to provide information on the distribution of effective fracture apertures. The field work was completed during 1980 and an analysis of these data is now underway in an effort to assess directional permeabilities for the granite mass.

The macropermeability experiment is an attempt to measure the permeability of a large volume of low-permeability, fractured rock. The experiment was conducted in a 5 m x 5 m x 33 m drift called the ventilation drift at the 355 m level of the Stripa mine. The water pressures in the rock were continuously measured ant the water inflow into the drift was measured as the net moisture pickup of a ventilation system inside a sealed portion of the drift. The experiment began in November, 1979, and ended in September, 1980.

Tests were run for three different room temperatures of about 20°C, 30°C and 45°C, followed by a cooldown test back to 20°C.

Moisture pickup for the 20° test averaged about 50 ml/min. Assuming that 80% of the observed 50 ml/min occurred as radial flow, the average hydraulic conductivity of the monitored rock mass is about 1.0 x 10^{-11} m/s. Preliminary data for both the 45°C and the 20°C cooldown tests are currently being analyzed.

A comprehensive program of investigations of the geochemistry and isotope hydrology of the Stripa groundwater has been carried out using samples collected from the surface, shallow private wells and underground boreholes. The work is now completed and the geochemical analyses show that total dissolved solids increases steadily with depth down to 840 m, which was the deepest point reached in one borehole. The results of C-14 dating indicate that water at the 330 m level, and probably also in the deep borehole that extends down to 840 m, is more than 20 000 years old.

8.1.4 Stress Measurements

Stress measurements were made during 1980 using both hydraulic fracturing and overcoring in a borehole drilled from the surface to a depth of 381 meters. The stress measurements were performed to provide data for analyzing the heater experiments. Preliminary analysis of the data indicates that the horizontal components of the overcoring data are in close agreement with the hydraulic fracture results.

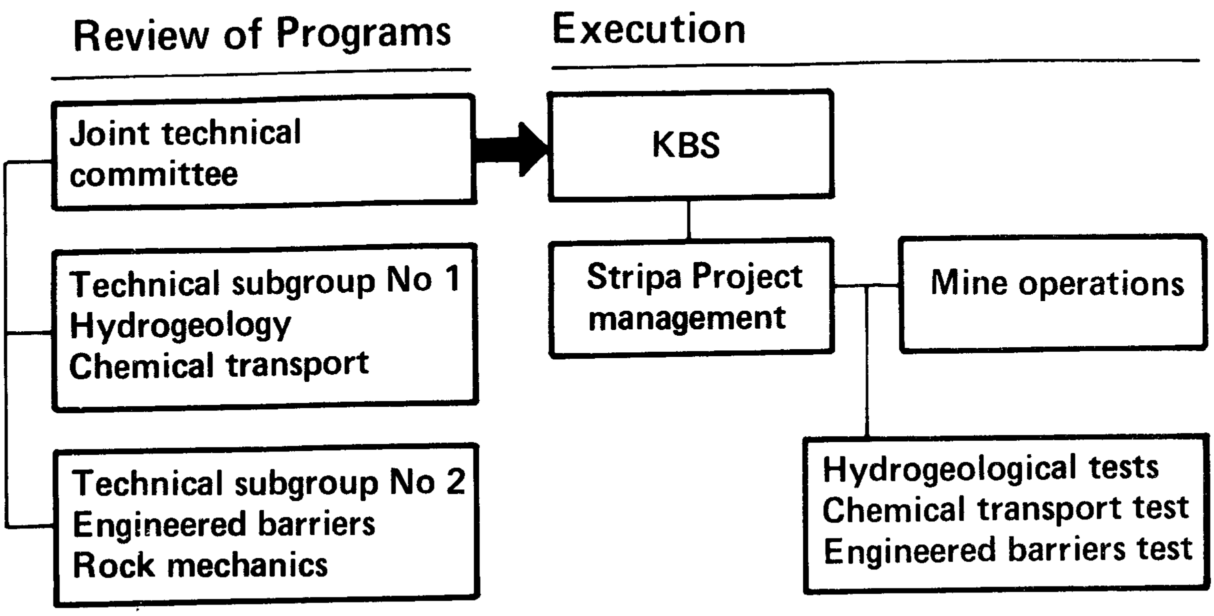


Fig.8–2. Organization plan for the Stripa Project.

8.2 THE INTERNATIONAL STRIPA PROJECT

A new joint research program - the Stripa Project - in the field of radioactive waste storage, organized as an autonomous OECD/NEA project, will now be conducted in the Stripa Mine during 1980 - 1984. Participating countries so far are Finland, Japan, Sweden, Switzerland and the United States.

8.2.1 Organization

The organization plan for the Stripa Project is shown in Figure 8-2. The Swedish SKBF/KBS organization, in charge of the operation of the Stripa Mine, is acting as the host organization for the project. Experimental programs are recommended by the participating countries and reviewed by technical subgroups (TSGs). The technical subgroups cover the fields of hydrology and chemical transport, rock mechanics and engineered barriers. The TSGs consist of members from the participating countries, and each task is assigned to a principal investigator.

Supervision of the project is entrusted to a Joint Technical Committee (JTC) composed of representatives of participating countries that provide financial support. Full membership in the Joint Technical Committee is contingent on a commitment to at least SEK 7.2 million for the four year period, unless otherwise agreed by the committee.

8.2.2 Objectives

The experimental work that is being performed in the Stripa Mine should increase our knowledge of

- Logging systems for boreholes measurements
- Flow of water through fractures in the rock
- Geochemistry of groundwater at great depth
- Migration rate of different substances in rock fractures
- Behavior and function of backfill material in a real geological environment

8.2.3 Technical Programs

Three defined investigations are currently in progress in Stripa. These are

- Hydrogeological investigations in boreholes. The principal investigators are Drs L Carlsson and T Olsson, Geological Survey of Sweden (SGU)
- Migration in a single fracture. The principal investigator is Professor I Neretnieks, Royal Institute of Technology (KTH), Sweden
- Buffer Mass Test. The principal investigator is Professor R Pusch, University of Luleå (LuH), Sweden.

The details of the investigations are described by Carlsson et al (15).

8.2.4 Reporting

The results from the project will be published in a special series of reports. A separate report will be published annually, describing the progress of the Stripa Project.

9 LOW- AND MEDIUM LEVEL WASTE

9.1 GENERAL

Low- and medium-level radioactive waste arises in all stages of the nuclear fuel cycle. Of particular interest to the Swedish utilities are the waste from reactor operation, fuel storage and reprocessing. The purpose of the KBS work in this field is primarily to develop safe final disposal methods for the different waste forms. Besides studies of final disposal, the work also includes studies of treatment methods and product characterization.

Work on the different waste forms is in different stages of development. As regards the waste from reactor operation and fuel storage facilities, strong efforts are being made to get a final repository operational as soon as possible. This work is at present in a pre-project phase. The final disposal of low- and medium level wastes from reprocessing, on the other hand, will most probably be realized,after a period of intermediate storage, in connection with the final disposal of high-level waste, i e after 2010. The present work within this field is therefore of a more basic R&D nature.

Decommissioning of nuclear power plants and handling of the waste from decommissioning is also being studied in this program, but at present with low priority.

9.2 REACTOR WASTE TREATMENT AND HANDLING

9.2.1 Waste Forms and Current Treatment Methods

Table 9-1 gives a list of the most important waste forms from reactor operation. The treatment methods currently being used at the Swedish nuclear power plants are also indicated. The waste that will be produced at the central fuel storage facility, CLAB, will be similar to the reactor waste and will be treated in a similar manner.

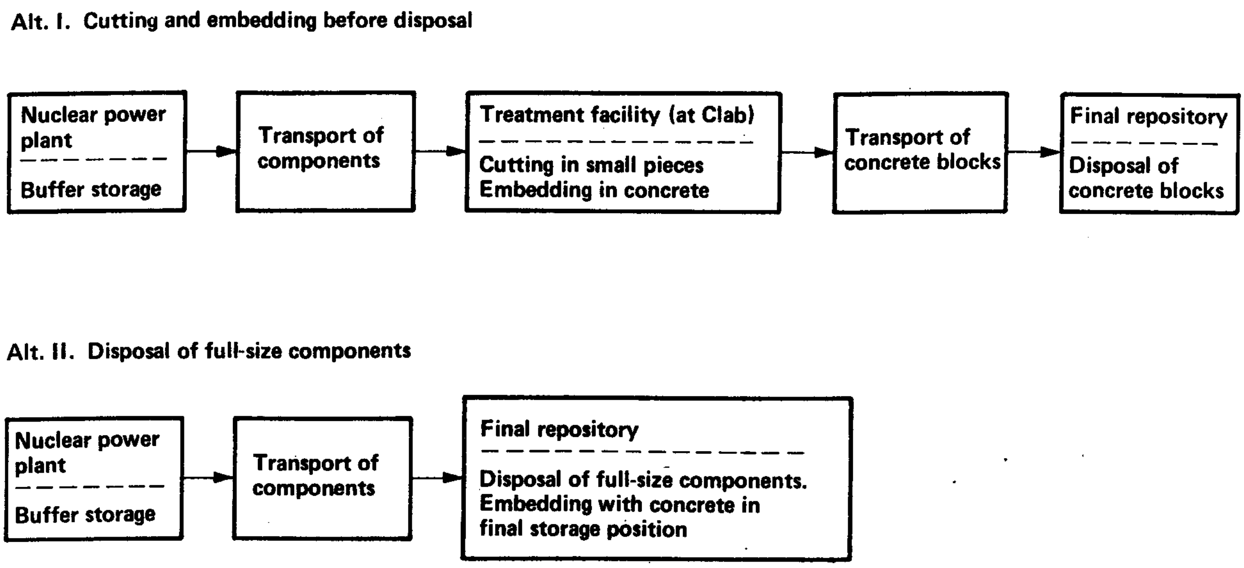


Fig.9–1. Alternatives for the treatment and final disposal of core components.

Table 9-1. Waste from a nuclear power plant and treatment methods currently used in Sweden

Waste form	Treatment method
Core components	No treatment
Ion exchange resins and sludges	Solidification with cement or bitumen. Dewatering
Metal scrap	Encapsulation in cement or no treatment
Low level solid waste	Compaction or incineration

At present, the treated waste is stored at each reactor site. The existing stores will be full between 1984 and 1989, depending on the site. Extensions on site or a final repository will then be needed. The possibility of having a final repository ready to receive waste by 1988 is investigated.

9.2.2 Treatment of Core Components for Final Disposal

At present, removed core components are stored untreated in the fuel storage pools at the power plants. Their high dose rate necessitates heavy shielding during handling and transport. The components also contain substantial amounts of long-lived nuclides (Ni-59).

Two alternative handling schemes are being conside-red for the treatment and disposal of the core components. They are illustrated schematically in Figure 9-1.

In alternative I, the components are cut into small pieces before being embedded in concrete in a shielding concrete container. The container is then transported to the final repository and disposed of in the same way as concrete blocks with other waste as described in Section 9.3.2. Due to the high dose rate of the core components, the cutting and embed-ding operation will have to be performed in a hot cell. A preliminary design study of such a treat-ment facility located close to CLAB has been made.

In alternative II no mechanical treatment of the components will be made, apart from what is necessa-ry to be able to remove them from the power plants. The components will be transported directly to the repository. This transport must be done in a type B container. An ordinary fuel transport cask could be used. Since this is an expensive solution a study of a special simple transport cask for these components has been made.

In the repository, the components will be placed in precast holes in a concrete structure. The holes are then filled with cement mortar, so that the components are totally embedded. A special transfer machine will be used for the transfer of the com-ponents from the transport cask to the hole. Figure 9-2 shows schematically how disposal can be done with this machine.

A preliminary study has shown that alternative II will give the lowest cost for the entire handling chain, even though the repository volume is larger. Doses to the personnel and the production of second-ary waste will also be lower in this alternative. Long-term safety for the alternatives is equivalent since the components will, in both cases, be in-corporated in a solid concrete structure, the high pH of which will give a low corrosion rate as well as a low solubility of Ni ions.

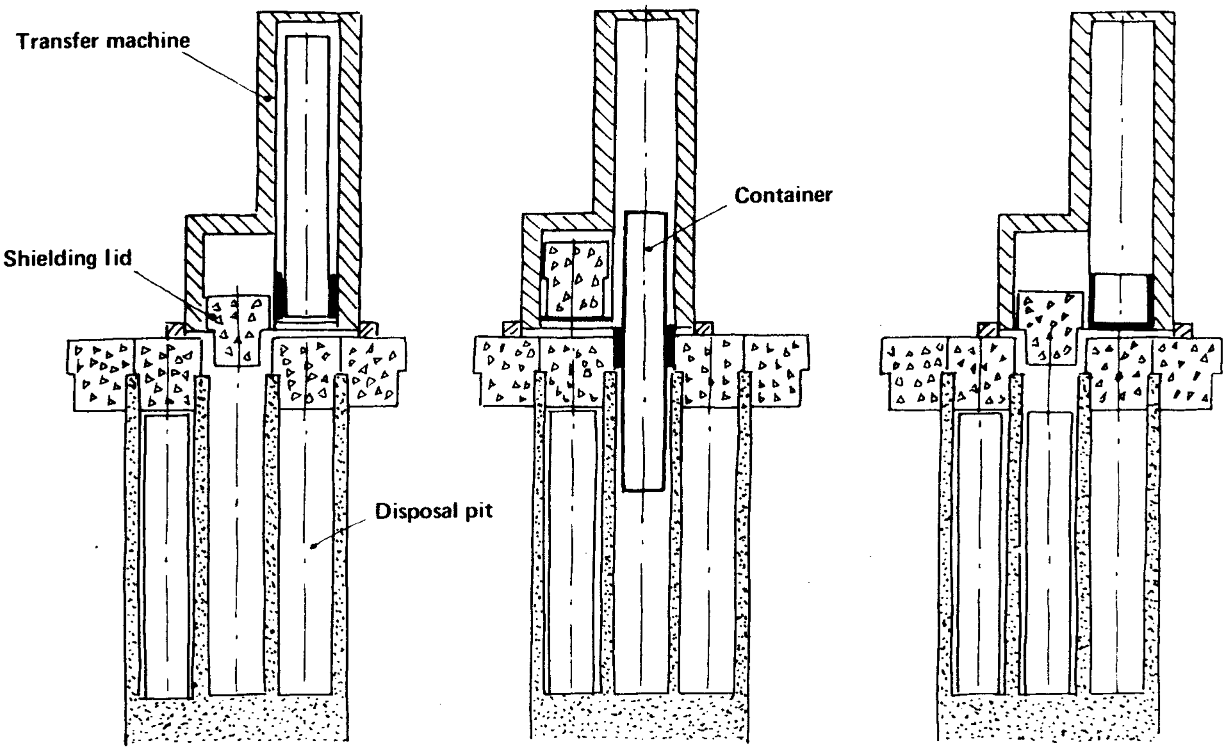


Fig.9–2. Sequence of disposal operations for a core component container.

The control rods in an ASEA-ATOM BWR contain boron carbide powder as neutron absorber. During irradiation, some of the boron will be transformed into tritium. The tritium could be released when the rods are cut in pieces, which is necessary for their transportation from the reactor station. In order to investigate the magnitude of this problem, a test involving cutting a used control rod was carried out at the Oskarshamn power plant. The result of the test shows that the tritium is strongly bound in the boron carbide powder. Only about 15 μCi of the calculated 100 Ci of tritium were released. The test also showed that the cutting method used gave a very clean operation with low chip formation.

9.2.3 Treatment of Ion-Exchange Resins and Sludges

Waste Solidified with Cement

The properties of ion-exchange resins solidified with cement have been studied by Prav (16). A study of solidification in cement of decontamination sludge has been initiated.

Waste Solidified with Bitumen

A report on the properties and long-term behavior of bitumen and radioactive waste-bitumen mixtures has been published (KBS TR 80-14). Special emphasis was put on the behavior of the bitumenized waste in the environment foreseen in the repository for reactor waste in crystalline rock. In the report, physical and chemical effects on stability, such as radiation effects, thermal effects, mechanical stability, chemical stability, interaction with surrounding media, gas generation and microbial stability, are considered.

It is concluded that properly prepared and composed radioactive-waste-bitumen mixtures form products with adequate properties and a long-term stability that ensures a satisfactory containment of radionuclides with a sufficient degree of reliability for the required storage period (<1000 years).

One point where further studies have been deemed necessary is, however, the behavior of dried ion-exchange resins in bitumen when they will come in contact with water in the final repository. A study of the swelling that will result is in progress.

Another investigation that is in progress is a measurement of the radioactivity released during a fire in bitumenized waste.

Other Treatment Methods

A review of alternative treatment methods for wet wastes has been reported (KBS TR 80-06). It concludes that the development of modern polymer matrices can give products of better quality, from a leaching point of view, than cemented or bitumenized waste. However, the cost of the treatment will be higher. Also, very little is known about the long-term stability of the products.

An interesting new method for chemical treatment of ion-exchange resins is being studied by Prav (17). The radio-activity is transferred to an inorganic ion-exchange resin of small volume with very good leach resistance. These studies will be continued so that an economic evaluation can be made in early 1982.

9.2.4 Handling of Low-Level Solid Wastes

Low-level solid wastes constitute the largest volume of waste produced by reactor operation. They consist of paper, clothing, insulation material, scrap metal etc. Their activity is normally very low.

The alternatives considered for treatment and disposal of these wastes are:

1 Shallow land burial at the reactor sites

2 Incineration of combustible waste and compaction of incombustible waste with subsequent final storage in the repository for reactor waste

3 Compaction of all wastes and subsequent final storage in the repository for reactor waste

An analysis of the potential radiological impact of the alternatives has been made. It can be concluded that radiation exposure of the public will be very low in all cases, due to the very low level of activity in this waste ($\sim$ 1 Ci/year and reactor unit). For the shallow-land burial alternative, exposures have been calculated both for the disposal phase and for the phase after institutional control has ceased.

Exposure of personnel is lower in alternatives 1 and 3 than in the case of incineration. However, the doses are very low in all cases.

The cost for shallow-land burial on site is by far the lowest. The cost for incineration is such that even the third alternative, compaction and subsequent disposal in a bedrock repository, may be competitive.

9.3 REPOSITORY FOR REACTOR WASTE

9.3.1 General

As mentioned earlier, the local intermediate stores for reactor wastes on the reactor sites will be full by the second half of the eighties. It is therefore highly desirable to take a repository for these wastes into operation as soon as possible.

A conceptual study of a central repository for Sweden was performed during 1977 - 1980 by Prav. Due to the geological conditions in Sweden no other host media than the bedrock was considered. This work included descriptions of a transportation system for the waste and a preliminary layout of the repository, with its barriers. An assessment of the long-term safety of the repository as well as of safety during transport was made (18, 19).

In mid-1980 the work was taken over by KBS and is now in an initial design phase. In parallel with these studies, a survey of possible sites was conducted. The survey was restricted to the vicinity of the existing nuclear facilities, since it is considered appropriate to locate the repository at such a site. In order to improve the knowledge geological site investigations were commenced during the autumn at Forsmark and Studsvik.

The schedule for the continued work includes preparation of siting applications to be submitted to the authorities in 1982. Construction can start in 1984, and then the repository can be ready to receive waste in 1988.

9.3.2 Description of the Facility

Waste Types and Packages

The repository is primarily intended to receive waste from reactor operation and similar waste from the operation of CLAB and the Studsvik research facility. In a later phase waste from decommissioning of nuclear facilities will also be disposed of here.

Most of the reactor waste packages will be either in the form of cubical concrete blocks (1.2 x 1.2 x x 1.2 m) containing cemented waste or 200 l drums (Ø 0.6, h 0.9 m) containing bitumenized or compacted waste. The repository will therefore be designed specially for these waste packages. Provision will also be made to receive other types of waste packages, however.

Activity Content

The total activity of the wastes to be disposed of in this repository is of the same order of magnitude as that of the high-level waste corresponding to one metric ton of reprocessed spent fuel. The decay time of the activity is governed primarily by Cs-137

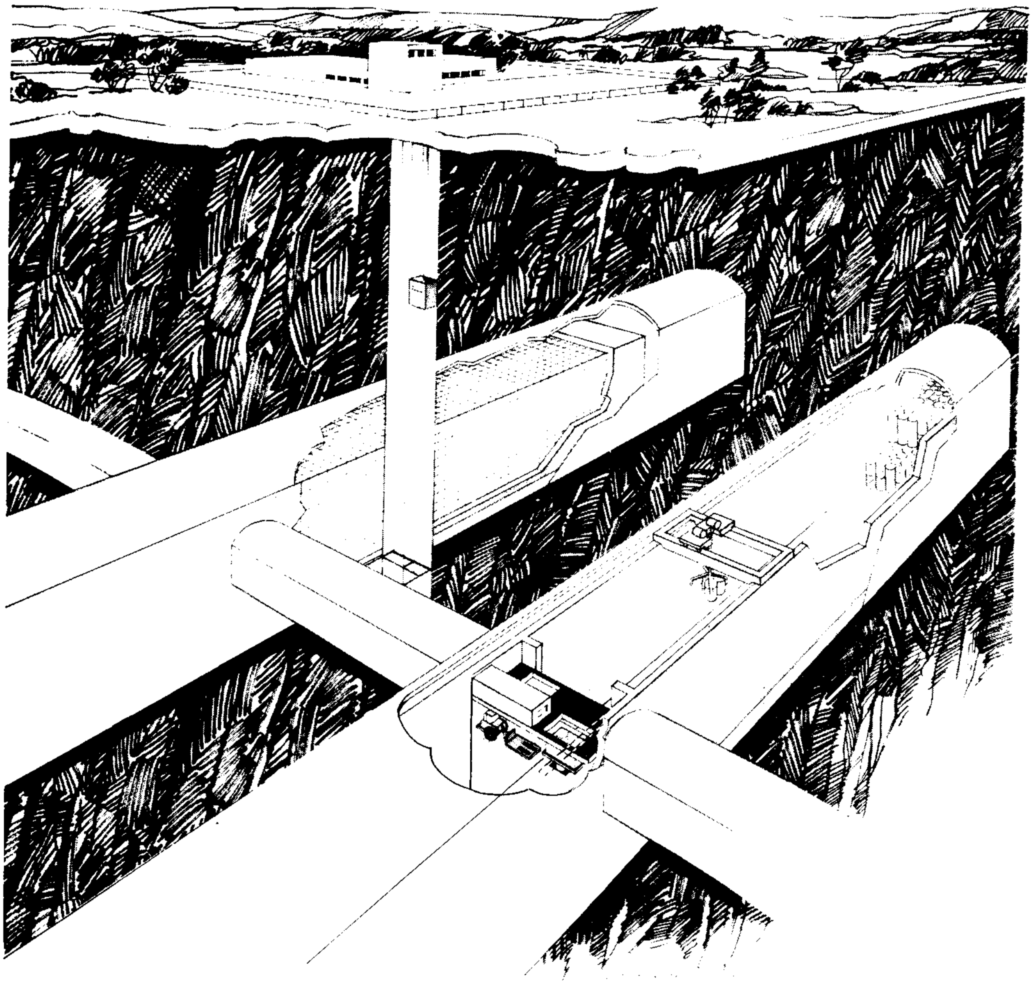


Fig.9–3. Repository for reactor waste. Alternative rockvaults.

and Sr-90 and it will have decreased on an innocuous value in about 500 years. The content of long-lived α-activity is very low.

Design of the facility

The following description is based on the present preliminary concept. Changes in the design are likely to be made. The repository will be located in bedrock at a moderate depth (∿ 50 m). At an earlier stage, repositories in clay or peat were also investigated, but the ideas were abandoned for cost and availability reasons.

Two alternatives for the layout of the storage chambers have been investigated, horizontal rock

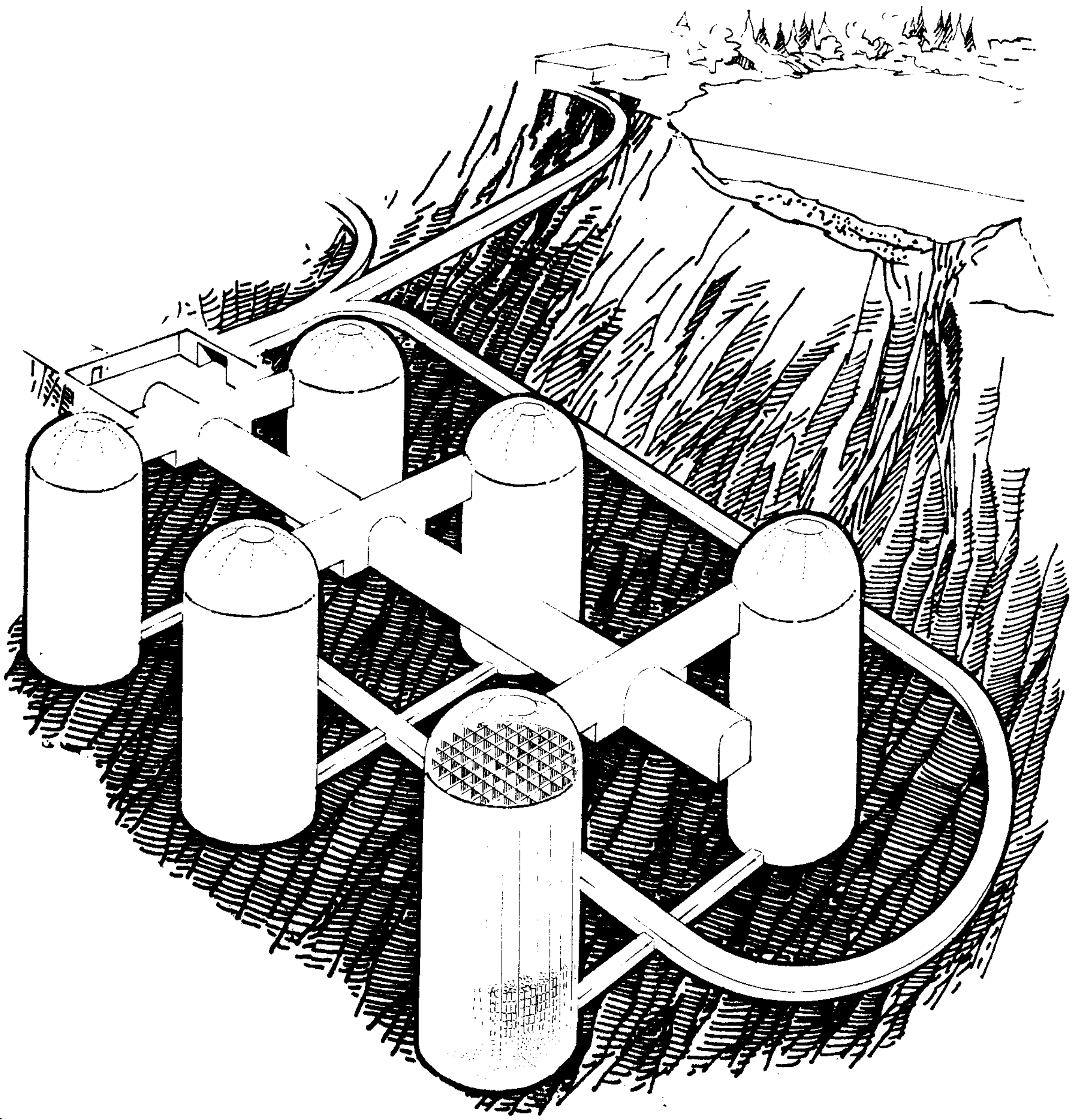


Fig.9–4. Repository for reactor waste. Alternative Silos.

vaults or vertical cylinders (silos). They are shown in Figures 9-3 and 9-4. The rock vaults have a cross-section of about 25 x 30 m. For all the reactor waste that will be produced by the Swedish nuclear program, i e about 125.000 m^3, two rock vaults about 200 m in length will be needed. In the silo alternative, six silos 60 m in height and 30 m in diameter will be needed.

In the storage chambers, a large storage pit with concrete walls and floor and an internal lattice structure will be built. The waste packages will be deposited in layers in the lattice. All handling will be remotely operated. After a few layers have been filled, the gaps between the packages will be filled with cement mortar.

When the storage chamber has been completely filled, a concrete slab will be poured on top and the space between the concrete structure and the rock will be filled with a mixture of sand and bentonite. The bentonite content will be chosen so that a self-healing function is guaranteed.

Barriers

In the described design, the following barriers will exist between the waste and the environment:

- The waste package itself
- The concrete between the waste packages and the concrete wall around the pit
- The sand/bentonite layer
- The rock

The concrete wall and the sand/bentonite layer are considered to be the main barriers.

Release Analysis

A preliminary release analysis has been performed for the design described above. The results show that the release rate of activity will be very low. Only if a well for drinking water is drilled directly through the repository will exposure of any significance occur. This risk can be avoided by locating the repository in rock under the bottom of the sea.

9.3.3 Transportation System

Since all the Swedish nuclear facilities are located on the coast, a sea transportation system is considered the best for economic and safety reasons. A special ship designed for transport of spent nuclear fuel is currently under construction. The same ship can also be used for reactor waste.

Normally, the activity content of the solidified waste is such that they can be classified as low-level solid waste and can be transported in a "strong industrial package". Shielding concrete transport containers will be used for this kind of waste. The main components of the transportation system are shown in Figure 9-5.

An analysis of the safety of the sea transportation scheme has been made. It shows that even in the case of a severe ship collision where part of the cargo is lost, the consequences, as far as radiation is concerned, will be very limited.

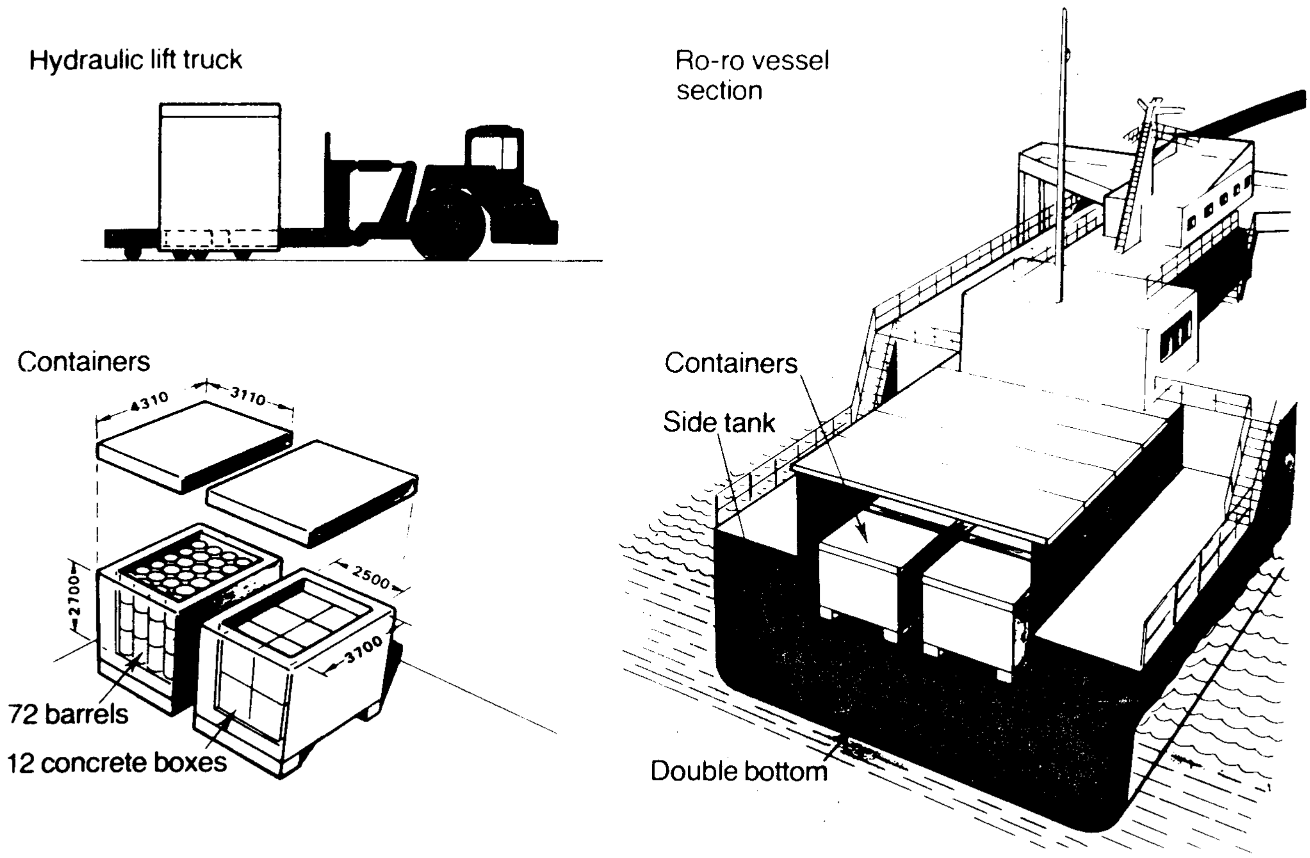


Fig.9–5. Transportation system for reactor waste.

9.4 LOW- AND MEDIUM LEVEL WASTE FROM REPROCESSING

9.4.1 General

Under the terms of the reprocessing agreement between SKBF and COGEMA, it is anticipated that, in addition to the vitrified high-level waste, low- and medium-level waste arising from reprocessing may also be returned to Sweden. The waste forms will be defined by COGEMA in consultation with its customers. SKBF is being represented by KBS in these discussions.

9.4.2 Repository for Low- and Medium-Level Reprocessing Waste

Some of the reprocessing waste, e g cladding hulls and some low-level solid waste, will have an actinide content of the same magnitude as the high-level vitrified waste (per tonne of reprocessed fuel). Demands on long-term isolation will thus also be the same.

In order to establish the level of confinement needed in a repository for reprocessing waste, preliminary calculations of the release of radionuclides from an idealized repository have been carried out. The results indicate the need for further knowledge in the following fields.

- Potential content of complexing agents in the reprocessing waste
- Radiolysis gas formation
- Chemistry of actinides at a high pH
- Long-term stability and changes in the chemistry of concrete
- Interaction between concrete and bentonite
- Influence of the high pH on the retention capacity of the surrounding rock

Most of the points are due to the fact that much of the waste will be solidified with concrete.

9.4.3 Hot Isostatic Pressing of Reprocessing Waste

The method currently proposed by the reprocessor for the treatment of most low- and medium-level reprocessing wastes is embedding in concrete. An alternative method for some of the wastes, e g fuel cladding hulls, has been investigated by ASEA (20). The hulls are compacted by Hot Isostatic Pressing into large, fully dense diffusion bonded and pore-free blocks of zircalloy. This is achieved at about 1000^{o}C and 150 MPa.

The process will give products of minimum volume. The long-lived radionuclides which were accessible on the surfaces of the hulls prior to pressing will be uniformly distributed within the block after the treatment. This means that the leach rate of the products will be governed by the corrosion rate of zircalloy, which is extremely low.

Inactive tests were performed during 1980 in order to find the best operating parameters, such as filling density, gas pressure and temperature, holding time, container material etc. A study of a possible process on an industrial scale was also conducted.

One problem involved in the hot treatment of hulls is the release of tritium. However, due to the small gas volume in the press and the high solubility of hydrogen in zircalloy contamination of the press gas has been calculated to be very low. This will be verified by experiments.

The HIP process can also be used for other α-bearing wastes, such as fines and ashes.

9.5 DECOMMISSIONING OF NUCLEAR FACILITIES

A study of the cost and time required to dismantle a Swedish nuclear power plant was made in 1979 (KBS TR 79-22). The Oskarshamn II unit, a 590 MWe BWR of ASEA-ATOM design, was chosen as a reference plant.

The method used in this study was applied during 1980 for a calculation of the cost of dismantling the other Swedish nuclear power plants. For a BWR, the cost is about 800 SEK/kW (at 1979 price level). The specific cost is not very size-dependent, which can be explained by the fact that the newer power plants, due to more severe safety requirements, have a higher specific volume and mass per kW.

REFERENCES

1 Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste.
KBS Final Report, Volume I-IV
Stockholm, 1977

2 Handling and Final Storage of Unreprocessed Spent Nuclear Fuel. KBS Final Report,
Volume I and II
Stockholm, 1978

3 DURAN O and MAGNUSSON K-Å
Measurements of Temperature, Self Potential pH, Eh and Water Conductivity in open Boreholes
(in Swedish)
Prav Report 4.15
Stockholm, May 1980

4 NERETNIEKS I
Age Dating of Groundwater in Fissured Rock:
The Influence of Water Volume in Micropores Water Resources Research (in print)

5 TORSTENFELT B, ANDERSSON K and ALLARD B
Sorption Studies of $H^{14}CO_3^-$ on some Geologic Media and Concrete
The International Symposium on the Scientific Basis for Nuclear Waste Management
Boston, USA, November 17-20, 1980 (in print)

6 KLOCKARS C-E, PERSSON O, CARLSSON L, DURAN O, LINDSTRÖM D, MAGNUSSON K-Å and SCHERMAN S
Preparatory Hydrogeologic Investigations for in Situ Migration Experiments in Studsvik
Prav Report 4.17
Stockholm, December 1980

7 LANDSTRÖM O
Preparatory Work on in Situ Migration Experiments in Studsvik
Prav Report 4.18
Stockholm, Decemner 1980

8 BROTZEN O, DURAN O and MAGNUSSON K-Å
Evaluation of Geophysical Borehole Studies
Prav Report 4.14
Stockholm, February 1980

9 BJERHAMMAR A, SVENSSON L and STOCKI S
The Uplift Process in Fennoscandia and the Corresponding Geopotential Field from Satellites
Prav Report 4.19
Stockholm, April 1980

10 LARSON S Å
A Literature Survey of Fracture Mineralogy
(in Swedish)
Prav Report 4.13
Stockholm, January 1980

11 LARSSON S Å, TULLBORG E L and LINDBLOM S
Investigations on Properties of Fracture Mineralizations in Rock Samples from Finnsjön, Sternö and Stripa
Prav Report 4.20
Stockholm, March 1981

12 AHLBOM K, OLKIEWICZ A and SCHERMAN S
Selection of Sites for Further Investigations of Rock Formations in Southern Sweden
(in Swedish)
Prav Report 4.11
Stockholm, May 1979

13 AHLBOM K, AHLIN S, ERIKSSON L and SAMUELSSON L
Bed-Rock Geological, Tectonical and Geophysical Studies at Kynnefjäll
(in Swedish)
Prav Report 4.16
Stockholm, May 1980

14 WITHERSPOON P A, COOLE N G W and GALE J E
Progress with Field Investigations at Stripa
Lawrence Berkeley Laboratory Report
LBL-10559, SAC-27
Berkeley, California, USA 1980

15 CARLSSON L, OLSSON T, NERETNIEKS, I and PUSCH R
Summary of Defined Programs
SKBF/KBS Stripa Project Technical Report 81-01
Stockholm 1981

16 BEIJER O
Casting Granular Ion Exchange Resins with Medium-Active Waste in Cement. Summary
Prav Report 1.36
Stockholm 1980

17 HULTGREN Å, FÄLTH L, FORBERG S, THEGERSTRÖM C and LILJENZIN J O
Research Programme on the Conditioning of Nuclear Power Waste. Progress report. July 1980
Prav Report 3.23
Stockholm 1980

18 KJELLBERT N, HÄGGBLOM H, CEDERSTRÖM M and LUNDGREN T
Central Repository for Low- and Intermediate-Level Waste (ALMA).
Conceptual Design, Siting and Safety Study
Prav Report 1.29
Stockholm 1980

19 DEVELL L, EDLUND O, KJELLBERT N, GRUNDFELT B and MILCHERT T
Safety Analysis of Sea Transportation of Solidified Reactor Wastes
Prav Report 1.31
Stockholm, June 1980

20 LARKER H and TEGMAN R
Treatment of Cladding Hulls by the HIPOW Process.
Paper presented at the Scientific Basis for Nuclear Waste Management Conference in Boston, November 1980

Appendix 1

LECTURES AND PUBLICATIONS 1980

Safe Handling and Storage of High-Level Radioactive Waste
P-E Ahlström (KBS), S Löfveberg (Prav), L B Nilsson and T Papp (KBS)
Radioactive Waste Management 1, 1 (1980) 57

Feasibility of Safe Terminal Disposal of Spent Nuclear Fuel
L B Nilsson and T Papp (KBS)
IAEA Bulletin 22, 3/4 (1980) 63

Ceramic and Pure-Metal Canisters in Buffer Material for High-Level Radioactive Waste
P-E Ahlström (KBS)
Nuclear and Chemical Waste Management 1, 1 (1980) 77

Towards Nuclear Waste Storage
B Allard (Chalmers University of Technology)
CHEMTECH, April (1980)

Highly Compacted Bentonite for Borehole and Shaft Plugging
R Pusch (University of Luleå) and A Bergström (KBS)
OECD/NEA Workshop on Borehole and Shaft Plugging
Columbus, Ohio, May 1980

Radioactive Waste Management in Sweden
T Papp (KBS)
ANS Annual Meeting, Las Vegas, June 1980

The Back End of the Nuclear Fuel Cycle Technical Solutions and Safety Analysis in the KBS Concept
P-E Ahlström, L B Nilsson (KBS)
11th World Energy Conf., Munich, Sept. 1980

The Swedish Nuclear Waste Management Programme
T Papp (in Swedish) (KBS)
NKA Seminar 1980, Ebeltoft, Denmark, 29 Sept.-1 Oct.

The Nuclear Fuel Safety Project (KBS)
F Karlsson (KBS)
IAEA Seminar, Gothenburg, 18-22 Aug. 1980

Final Disposal of Unreprocessed Spent Fuel (in German)
F Karlsson (KBS)
Symposium at Haus der Technik, Essen
Fed. Rep. of Germany, Nov. 1980

The Investigations for Final Disposal of High Level Waste in Sweden (in German)
A Hult (KBS)
Symposium at Schweizerische Vereinigung für Atomenergie, Okt. 1980

The Influence of Microfissures in Crystalline Rock on Radionuclide Migration
I Neretnieks (The Royal Institute of Technology, Stockholm)
26th International Geological Congress, Paris, July 1980

Treatment of Cladding Hulls by the HIPOW Process
H Larker and R Tegman (ASEA Robertsfors)
MRS Seminar 1980, Boston, 16-21 Nov. 1980

Some Difficulties in Interpreting In-Situ Tracer Tests
I Neretnieks (Royal Institute of Technology, Stockholm)
MRS Seminar 1980, Boston, 16-21 Nov. 1980

Highly Compacted Bentonite - A Self-Healing Substance for Nuclear Waste Isolation
R Pusch (University of Luleå)
A Bergström (KBS)
MRS Seminar 1980, Boston, 16-21 Nov. 1980

An ESCA Investigaiton of Molybdenum Containing Silicate Glasses
R Nyholm (University of Uppsala)
L Werme (KBS)
MRS Seminar 1980, Boston, 16-21 Nov. 1980

Modeling of Rock Mass Deformation for Radioactive Waste Repositories in Hard Rock
O Stephansson, P Johansson
(Universtiy of Luleå) and T Groth (The Royal Institute of Technology, Stockholm)
MRS Seminar 1980, Boston, 16-21 Nov. 1980

Hard Rock and the High-Level Waste (in Swedish)
KBS Information 09, February 1980

Diffusion in the Rock Matrix:
An Important Factor in Radionuclide Retardation?
I Neretnieks (The Royal Institute of Technology, Stockholm)
Journal of Geophysical Research 85, B8 (1980) 4379

Exact Solution of a Model for Diffusion in Particles and Longitudinal Dispersion in Packed Beds
A Rasmuson and I Neretnieks (The Royal Institute of Technology, Stockholm)
American Journal of Chemical Engineering 26 (1980) 686

Progress with Field Investigations at Stripa
P A Witherspoon (Lawrence Berkeley Laboratory)
NGW Cook (University of California) and J E Gale (University of Waterloo)
Swedish-American Cooperative Project
Technical Reports SAC-27, LBL-10559, UC-70

LIST OF KBS TECHNICAL REPORTS 1980 Appendix 2

In order to have the full reports available to all interested persons, they have been filed as microfish at

INIS CLEARINGHOUSE
International Atomic Energy Agency
P O Box 590
A-1011 VIENNA, AUSTRIA

<u>Report No List of the Technical Reports</u>

<u>1977-78</u>

TR 121 KBS Technical Reports 1 - 120.
Summaries. Stockholm, May 1979.

<u>1979</u>

TR 79-28 The KBS Annual Report 1979.
KBS Technical Reports 79-01--79-27.
Summaries. Stockholm, March 1980.

<u>1980</u>

*) TR 80-01 Komplettering och sammanfattning av geohydrologiska undersökningar inom sternöområdet, Karlshamn ("Supplementary and concluding geohydrological investigations within the Sternö area in Karlshamn")
Lennart Ekman
Bengt Gentzschein
Sveriges geologiska undersökning, mars 1980

*)= In Swedish

Report No List of the Technical Reports

TR 80-02 Modelling of rock mass deformation for radioactive waste repositories in hard rock
Ove Stephansson
Per Jonasson
Department of Rock Mechanics
University of Luleå

Tommy Groth
Department of Soil and Rock Mechanics
Royal Institute of Technology, Stockholm
1980-01-29

TR 80-03 GETOUT - a one-dimensional model for groundwater transport of radionuclide decay chains
Bertil Grundfelt
Mark Elert
Kemakta konsult AB, January 1980

TR 80-04 Helium retention
Summary of reports and memoranda
Gunnar Berggren
Studsvik Energiteknik AB, 1980-02-14

TR 80-05 On the description of the properties of fractured rock using the concept of a porous medium
John Stokes
Royal Institute of Technology, Stockholm
1980-05-09

*) TR 80-06 Alternativa ingjutningstekniker för radioaktiva jonbytarmassor och avfallslösningar
("Alternative methods for solidification of ion exchange resins and waste solutions")
Claes Thegerström
Studsvik Energiteknik AB, 1980-01-29

TR 80-07 A calculation of the radioactivity induced in PWR cluster control rods with the origen and casmo codes
Kim Ekberg
Studsvik Energiteknik AB, 1980-03-12

TR 80-08 Groundwater dating by means of isotopes
A brief review of methods for dating old groundwater by means of isotopes
A computing model for carbon - 14 ages in groundwater
Barbro Johansson
Naturgeografiska Institutionen
Uppsala Universitet, August 1980

TR 80-09 The Bergshamra earthquake sequence of December 23, 1979
Ota Kulhánek, Norris John, Klaus Meyer, Torild van Eck and Rutger Wahlström
Seismological Section, Uppsala University
Uppsala, Sweden, August 1980

*) = In Swedish

Report No List of the Technical Reports

*)	TR 80-10	Kompletterande permeabilitetsmätningar i finnsjö-området ("Supplementary geohydrological investigations in the Finnsjö area") Leif Carlsson, Bengt Gentzschein, Gunnar Gidlund, Kenth Hansson, Torbjörn Svenson, Ulf Thoregren Sveriges geologiska undersökning, Uppsala, maj 1980
	TR 80-11	Water uptake, migration and swelling characteristics of unsaturated and saturated, highly compacted bentonite Roland Pusch Luleå 1980-09-20 Division Soil Mechanics, University of Luleå
	TR 80-12	Drilling holes in rock for final storage of spent nuclear fuel Gunnar Nord Swedish Detonic Research Foundation
	TR 80-13	Swelling pressure of highly compacted bentonite Roland Pusch Division Soil Mechanics, University of Luleå Luleå 1980-08-20
	TR-80-14	Properties and long-term behaviour of bitumen and radioactive waste-bitumen mixtures Hubert Eschrich Eurochemic, Mol, October 1980
	TR 80-15	Aluminium oxide as an encapsulation material for unreprocessed nuclear fuel waste - evaluation from the viewpoint of corrosion Final Report 1980-03-19 Swedish Corrosion Institute and its reference group
	TR 80-16	Permeability of highly compacted bentonite Roland Pusch Division Soil Mechanics, University of Luleå 1980-12-23
	TR 80-17	Input description for BIOPATH Jan-Erik Marklund Ulla Bergström Ove Edlund Studsvik Energiteknik AB, 1980-01-21
*)	TR 80-18	Införande av tidsberoende koefficientmatriser i BIOPATH ("Introduction of time independent coefficient matrices in BIOPATH") Jan-Erik Marklund Studsvik Energiteknik AB, januari 1980

*) = In Swedish

Report Nr List of the Technical Reports

TR 80-19	Hydrothermal conditions around a radioactive waste repository Part 1 A mathematical model for the flow of groundwater and heat in fractured rock Part 2 Numerical solutions Roger Thunvik Royal Institute of Technology, Stockholm, Sweden Carol Braester Israel Institute of Technology, Haifa, Israel December 1980
*) TR 80-20	BEGAFIP. Programvård, utveckling och benchmarkberäkningar ("BEGAFIP. Program updating, development and bench mark calculations") Göran Olsson Peter Hägglöf Stanley Svensson Studsvik Energiteknik AB, 1980-12-14
TR 80-21	Report on techniques and methods for surface characterization of glasses and ceramics Bengt Kasemo Mellerud, August 1980
TR 80-22	Evaluation of five glasses and a glass-ceramic for solidification of Swedish nuclear waste Larry L Hench Ladawan Urwongse Ceramics Division Department of Materials Science and Engineering University of Florida, Gainesville, Florida 1980-08-16
TR 80-23	Exact solution of a model for diffusion in particles and longitudinal dispersion in packed beds Anders Rasmuson Ivars Neretnieks Royal Institute of Technology, August 1979
TR 80-24	Migration of radionuclides in fissured rock - The influence of micropore diffusion and longitudinal dispersion Anders Rasmuson Ivars Neretnieks Royal Institute of Technology, December 1979
TR 80-25	Diffusion and sorption in particles and two-dimensional dispersion in a porous media Anders Rasmuson Royal Institute of Technology, January 1980

*) = In Swedish

AUTHORS OF KBS TECHNICAL REPORTS 1980 IN ALPHABETICAL ORDER

Appendix 3

AUTHOR	TECHNICAL REPORT NO 80-
Berggren, Gunnar Studsvik Energiteknik AB	80-04
Bergström, Ulla Studsvik Energiteknik AB	80-17 x)
Braester, Carol Israel Institute of Technology	80-19 x)
Carlsson, Leif Geological Survey of Sweden	80-10 x)
van Eck, Torild University of Uppsala	80-09 x)
Edlund, Ove Studsvik Energiteknik AB	80-17 x)
Ekberg, Kim Studsvik Energiteknik AB	80-07
Ekman, Lennart Geological Survey of Sweden	80-01 x)
Elert, Mark Kemakta AB	80-03 x)
Eschrich, Hubert Eurochemic	80-14

x) The report is written by more than one person

AUTHOR	TECHNICAL REPORT NO 80
Gentzschein, Bengt Geological Survey of Sweden	80-01 x), 80-10 x)
Gidlund, Gunnar Geological Survey of Sweden	80-10 x)
Groth, Tommy Royal Institute of Technology	80-02 x)
Grundfelt, Bertil Kemakta AB	80-03 x)
Hansson, Kenth Geological Survey of Sweden	80-10 x)
Hench, Larry L University of Florida	80-22 x)
Hägglöf, Peter Studsvik Energiteknik AB	80-20 x)
John, Norris University of Uppsala	80-09 x)
Johansson, Barbro University of Uppsala	80-08
Jonasson, Per University of Luleå	80-02 x)
Kasemo, Bengt Mellerud	80-21
Kulhánek, Ota University of Uppsala	80-09 x)
Marklund, Jan-Erik Studsvik Energiteknik AB	80-17 x), 80-18
Meyer, Klaus University of Uppsala	80-09 x)

x) The report is written by more than one person

AUTHOR	TECHNICAL REPORT NO 80
Neretnieks, Ivars Royal Institute of Technology	80-23 x), 80-24 x)
Nord, Gunnar Swedish Detonic Research Foundation	80-12
Olsson, Göran Studsvik Energiteknik AB	80-20 x)
Pusch, Roland University of Luleå	80-11, 80-13, 80-16
Rasmuson, Anders Royal Institute of Technology	80-23 x), 80-24 x), 80-25
Stephansson, Ove University of Luleå	80-02 x)
Stokes, John Royal Institute of Technology	80-05
Svenson, Torbjörn Geological Survey of Sweden	80-10 x)
Svensson, Stanley Studsvik Energiteknik AB	80-20 x)
Thegerström, Claes Studsvik Energiteknik AB	80-06
Thoregren, Ulf Geological Survey of Sweden	80-10 x)
Thunvik, Roger Royal Institute of Technology	80-19 x)
Urwongse, Ladawan University of Florida	80-22 x)
Wahlström, Rutger University of Uppsala	80-09 x)

x) The report is written by more than one person

KEY WORD REGISTER TO TECHNICAL REPORTS Appendix 4

KEY WORDS	TECHNICAL REPORT NO 80-
Activity	07
Alumina	15
BEGAFIP	20
Bentonite	11, 13, 16
BIOPATH	17, 18
Bitumen	14
Buffer material	11, 13, 16
Canisters	15
Ceramics	21, 22
Construction	12
Corrosion	15
Diffusion	23, 24, 25
Dispersion	23, 24, 25
Earthquakes	09
Field investigations	
Finnsjön	10
Sternö	01
Geology	01
GETOUT	03
Glass	21, 22
Groundwater dating	08
Helium	04
Hydrogeological modeling	05, 19
Hydrogeology	01, 05, 10, 19
Ion exchangers	06
Low- and medium-level waste	06, 14

KEY WORDS	TECHNICAL REPORT NO 80-
Materials characterization	14, 21, 22
Migration geosphere	03, 23, 24, 25
Migration biosphere	17, 18
Permeability	16
Residual power	20
Rock mechanics	02
Safety analysis	03, 17, 18
Sorption (retention)	23, 24, 25
Stress and strain	02
Thermal convection	19
Waste form	06, 14, 21, 22

SUMMARIES OF TECHNICAL REPORTS Appendix 5

KBS Technical Report No 80-01

SUPPLEMENTARY AND CONCLUDING GEOHYDROLOGICAL INVESTIGATIONS WITHIN THE STERNÖ AREA IN KARLSHAMN

Lennart Ekman
Bengt Gentzschein
Geological Survey of Sweden, March 1980

SUMMARY

The Geological Survey of Sweden (SGU) has on contract with Project Kärnbränslesäkerhet, KBS (The Swedish Nuclear Fuel Safety Project) conducted some supplementary geohydrological investigations of five core boreholes, Ka 1 - Ka 5, which were drilled on the peninsula of Sternö outside Karlshamn in south-eastern Sweden. The boreholes are between 577.3 and 802.6 m long and have a diameter of 56 mm. Their inclination varies between 50° and 80°. They were thoroughly flushed with compressed air before the borehole-investigations.

The field work included water injection tests with double and single packer equipment in those parts of the boreholes that were not previously tested. Some measurements of groundwater levels and simple pressure measurements were also made. Previously published results of water injection tests are presented again in order to give a complete picture of the permeability in all boreholes. Permeability, k, was evaluated using a formula assuming a stationary state.

In the double packer test water was injected under excess pressure between packers. In the single packer test water was injected between the lower-most packer and the bottom of the borehole. 2 m double packer sections were used in Ka 1 and 3 m sections in all the other boreholes. The packers were pressed against the borehole wall in a hydraulic-mechanical way. The groundwater pressure and the injection

pressure, in the tested section were monitored by a pressure transducer placed close to the upper packer with hydraulic connection with the section. In most cases the excess pressure of 0.2 or 0.4 MPa was used for the double packer measurements, but other excess pressures were also used for the measurements with the single packer. The water was injected through a steel pipe and the flow measurements by flow gauges of the rotameter type. They could measure flows of between 0.00085 l/min and 65 l/min.

The sources of error associated with this method can be divided into practical-technical and theoretical ones. The practical errors tend to overestimate the hydraulic conductivity of the rock mass, partly due to leakages and short-circuitflows around the packers.

The results of water injection tests with double packers showed that high permeabilities were measured down to about 350 m length (about 440 m in Ka 4). This corresponds to between 270 m and 425 m vertical depths in the different boreholes. In these upper fractured zones many permeabilities > 1.0×10^{-8} m/s were recorded including values as high as between 1.0×10^{-6} m/s and 5.0×10^{-6} m/s. Below about 350 m (440 m in Ka 4), permeability diminished considerably. With the exception of Ka 5, where an instrument leakage was discovered, the values were below 5.0×10^{-9} m/s, most of them even lower than 1.0×10^{-9} m/s. Ka 1 showed low permeabilities from top to bottom ($k_{max} = 1.1 \times 10^{-8}$ m/s). However, the upper 400 m generally exhibited lower permeability values than the section below. Only a few were higher than 5.0×10^{-10} m/s, above 400 m. under that length all values were below 2.9×10^{-9} m/s.

With the single packer equipment, permeabilities from 1.3×10^{-10} m/s (in Ka 1) to 2.9×10^{-8} m/s (in Ka 4) were found in the section between highest packer position used and the bottom of the borehole.

Comparison between double and single packer measurements in 50 m long sections (sometimes other lengths) showed that the agreement between the methods usually was good. However, the calculations performed on data from single packer measurements gave in some cases negative values because of, among other things, the accuracy in the measurements being substracted from each other. In the low-permeable parts towards the bottom of many boreholes, the double packer measurements gave higher permeability than those with single packer equipment. There are several reasons for this. First, the measuring limit is higher with the double packer method than with the single one. Second, the risk of instrument leakage or packer leakage is greater with the double packers. Furthermore, the effects of a leakage with the single packer were evened out over a longer section.

The double packer measurements with 0.2 MPa and 0.4 MPa showed that the lower excess pressure used gave higher permeability values within the interval 1.0 x 10^{-9} m/s - 1.8 x 10^{-6} m/s in most of the sections. When the permeability was below 1.0 x 10^{-9} m/s, which usually was the case in the lower parts of the boreholes, 0.4 MPa induced higher permeability in about 50 % of the sections, except in Ka 5, where the corresponding value was 90 %. The differences were, however, small in most cases.

The single packer measurements at different pressures did not give a uniform picture of the permeability differences, when all five boreholes were considered. However, high pressures usually gave lower permeability. In all cases, the difference was smaller than a factor of 8.9, even when the pressure differences were great.

Repeated measurements with double packers were carried out in 85 sections. In 11 cases, large discrepancies were noted (larger than a factor of 10). These were caused by an instrument leakage in eight of the eleven cases. Corresponding repeated measurements with a single packer showed good reproducibility.

The variation of the groundwater level was studied during the spring and autumn of 1979 in the deep boreholes and during all of 1979 except for December in eight shallow boreholes within the area. Some trends might be correlated between the deep and the shallow holes. Many of the latter seemed to be influenced by the adjacent oil storage caverns, situated several metres below sea level.

KBS Technical Report No 80-02

MODELLING OF ROCK MASS DEFORMATION FOR RADIOACTIVE WASTE REPOSITORIES IN HARD ROCK

Ove Stephansson
Per Jonasson
Department of Rock Mechanics, University of Luleå

Tommy Groth
Department of Soil and Rock Mechanics
Royal Institute of Technology, Stockholm
1980-01-29

ABSTRACT

Heating of the rock mass at a repository for high level radioactive waste will result in thermal expansion. The thermomechanical response of the rock mass, together with virgin stresses and swelling pressure of compacted bentonite in the deposition hole, could alter the joints and thereby the permeability. Geotechnical numerical models capable of predicting the response of the rock mass to various loadings are vital for the success of a radioactive waste disposal program. This paper reviews the Swedish concept for storage of radioactive waste and presents results of a finite element thermal mechanical analysis of rock mass deformation around a deposition hole in hard rocks. On the basis of experimental data on joint properties and swelling pressure of bentonite, the various sequences in mining, loading and sealing a repository are modelled for various virgin stresses. It is concluded that the tunnel and deposition hole are overall structurally stable for the applied virgin stresses, and that the maximum compressive and tensile stresses in the solid blocks of the rock mass never exceed the strength of a granitic rock.

KBS Technical Report No 80-03

GETOUT - A ONE-DIMENSIONAL MODEL FOR GROUNDWATER TRANSPORT OF RADIONUCLIDE DECAY CHAINS

Bertil Grundfelt
Mark Elert
Kemakta konsult AB, January 1980

Summary

The GETOUT-code, originally developed at Batelle Pacific Northwest Laboratories (PNL), was used in the KBS-project to calculate the radionuclide discharges from the repository. The version used in KBS was a translation of the PNL BASIC-language version as by december 1976. In this report a new version , mathematically compatible whith the PNL FORTRAN version as by 1979-08-15, is documented. Details are given on the differences between this new version and the version used in the KBS project up to now.

KBS Technical Report No 80-04

HELIUM RETENTION

Summary of reports and memoranda

Gunnar Berggren
Studsvik Energiteknik AB, 1980-02-14

SUMMARY

In order to elucidate the formation and retention of helium in spent fuel investigations have been performed. These have during 1978 and 1979 been presented as internal reports and memoranda listed below. The essentials of this work are presented in this summary.

KBS memos of 78-03-08 and 78-03-28 by L Devell concerning helium problems.

KBS-TR-111 (Report) of 78-07-26 by N Kjellbert -Radionuclide inventories.

Asea-Atom memo RB 78-99 of 78-04-18 by G Vesterlund concerning stress in fuel canning.

KBS memo of 78-04-12 by L Devell concerning diffusion of helium (with PM by U Engman and C Nilsson)

Studsvik Report K1/4-79/35 by J Chyssler and P Kresten -He-, Kr- and Xe Retention in Uraninite Minerals.

Studsvik Report K1/4-79/72 by G Berggren -Helium Retention in UO_2-fuel.

All the reports and memoranda are in Swedish.

KBS Technical Report No 80-05

ON THE DESCRIPTION OF THE PROPERTIES OF FRACTURED ROCK USING THE CONCEPT OF A POROUS MEDIUM

John Stokes
Royal Institute of Technology, Stockholm
1980-05-09

Summary and conclusions

In order to describe the flow of groundwater through fractured rock, water is either assumed to flow through a pervious continuum (the porous medium approach mostly founded on Darcy´s law) or through descrete fractures between impervious blocks of rock. The latter approach being the one demanding more information on the rock, problems on groundwater flow are usually discussed using the porous medium approach. The description is then devided into two steps. The first is to measure the properties of the porous medium in the field, the second to use these properties and given boundary- and initial-conditions to describe the flow.

It is often a question of debate wether the continuum approach is applicable to the fractured rock under consideration. The concept of a porous medium is connected to the properties of the fractured rock by the aid of mean values. If the statistical variation of the properties of the porous medium is required to be small over small distances, the number of fractures involved in the flow must be large. It is only when this requirement of continuity is fulfilled that the porous medium approach is appropriate. Therefore, it is essential that after assuming that a certain flow region acts as a porous medium, we use a procedure for measuring the properties that at the same time gives a test of this assumption.

When giving a description of groundwater flow, the goal is often a presentation of pathlines and flowtimes between points of interest and the ground surface. Using a porous medium approach, this means that hydraulic conductivity and porosity must be known through the medium. In order to cope with transient flow, we must also know the time constant governing the development of the flow. The pathlines depend to a great extent on the variation of conductivity through space. A conductivity decreasing with depth will force the pathlines to the surface giving local flow. If instead the conductivity is constant, the flow is regional. It is therefore important to know the gradient of hydraulic conductivity. Finally, as we know that the flow takes place through a geological structure, the anisotropic behaviour of the rock must be known in order to describe the flow.

In this report a procedure to measure the properties listed above is developed. It is subject to the following requirements:

1) Parameters should represent prescribed regions. Only then is it possible to exclude the possibility that the obtained values are determined by boundary effects.
2) Measurements should be performed under the same conditions as the parameters describing the porous medium are to be used. This demand for in-situ measurements is due to scale-effects and also to the fact that rock properties depend strongly on the measuring conditions.
3) The procedure should give a test on the assumption that the region of flow acts as a porous medium. At the same time a measure of quality of the performed measurement should be presented.
4) When determining anisotropy, the dependence of pressure development on direction should be used. This method is faster and also better defined than tracer methods.
5) If possible those field quantities which are difficult to relate to theory should be avoided. This excludes the measurement of filtration velocities. Particle flowtimes obtained from tracer tests ought really also be avoided. However, the only method known today to determine porosity makes these measurements necessary.

6) The theoretical relation used to interprete the field data should be stable from a numerical point of view. This excludes the simultaneous use of source dimensions and source potential.

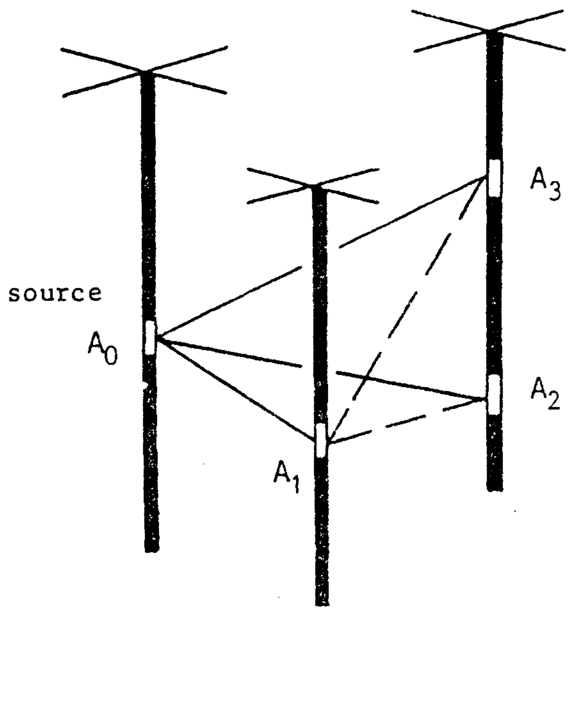
Before looking at the other sections the reader should go through section 2 defining the systems of reference. Sections 4 and 5 are then dedicated to the equation of continuity and Darcy´s law. In these sections the various parameters describing the properties to be measured are introduced. Section 6 then gives the potential generated by a point source when the parameters describing the porous medium are constant through the region to be investigated. This potential is used in section 7 to discuss the properties of radial flow such as which times and corresponding distances are involved when an introduced source changes the old flow pattern to a new one. It is found that the change in flow conditions can be described by a velocity which is inversely proportional to the distance to the source and this independently of the flow being 1-, 2- or 3-dimensional.

The various requirements on the procedure used for determining the rock parameters are discussed in section 8. In section 9 the relation between tracer concentrations and "particle flowtimes" is discussed ending up with a list over the entities which are to be measured in the field. The formal apparatus needed for the determination of anisotropy, the time constant, hydraulic conductivity, porosity and rock compressibility is then developed in sections 10 - 13, still under the assumption that the parameters describing the flow are constant through the region of flow.

After this simplified approach, the formal apparatus is completed in sections 14 - 15 so as to be applicable to cases where the hydraulic conductivity varies in space. The simplest assumption is that the variation is exponential in some direction, this giving a constant coefficient in the first-order term in the equation of flow. An example is a flow region where the conductivity decreases to for instance half it´s value for every 100 metres of depth. It is shown that both the magnitude and the direction of the conductivity gradient can be found with only a small change in the procedure presented in sections 10 - 13. Section 16 ends the discussion on rock properties with an investigation on random variations in porosity.

Finally, sections 17 - 21 are dedicated to the effects of varying fluid properties on the hydraulic conductivity. Changing temperature, pressure and solute concentration affects viscosity and density, both of which are involved in the conductivity. It is shown that the dominating effect is that of temperature on viscosity. However, when gradients of solute concentration become considerably larger than one ppm / metre, these will dominate when no temperature effects besides the geothermal gradient are present. Over all it is shown that effects of varying fluid properties can often be neglected when the effective region of flow is limited to a hundred metre scale.

The procedure used to measure the quantities necessary for determining the parameters of the porous medium is the following: Three wells should be used in order to determine anisotropy. Chosing a point source A_0 and three measuringpoints A_1, A_2 and A_3 in the wells according to the figure, all three wells are



cased along their entire length. Two reference times $t = 0$ and $t = t_o$ are now chosen. At $t = 0$ water is injected at the source A_0, keeping the waterloss constant until the source is shut down. At $t = t_o$ a tracer is introduced in the same source keeping the concentration constant until the source is shut down.

The pressurehead and solute concentration is now measured at points A_1, A_2 and A_3 as functions of time. The measurements are continued until it is seen that the pressurehead and concentration curve begin to stabilize. The source is then shut down and the whole procedure repeated with each point A_1, A_2 and A_3 in turn used as a source (source circulation).

The data collected from the complete measurement is

$(x,y,z)_m$	location of the four points A_m, $m = 0,1,2,3$
Q	magnitude of constant source flow for $t > 0$
c	concentration of tracer introduced in the source when $t > t_o$
$h_1,h_2,\ldots,h_N$	pressurehead at times $t_1,t_2,\ldots,t_N$ collected at each measuring point. This gives in all 12 time-developments when the source is circulated
$c_1,c_2,\ldots,c_N$	tracer concentration at times $t_1,t_2,\ldots,t_N$ collected at each measuring point. This gives in all 12 particle flowtimes when the source is circulated.

The numerical procedure necessary to determine the properties of the porous medium from these data consists of an adjustment of a theoretical potential, the anisotropy relations and Darcy's law to the data. This adjustment, founded on the method of least squares, is most easily performed on an electronic computer. The results are obtained in three steps. First the magnitude and principle axis of anisotropy are determined. At the same time we obtain the time constant and the magnitude of the gradient of hydraulic conductivity. In the second step the conductivity it self is determined together with the direction of it's gradient. Finally, in the third step porosity is determined by utilizing all the previously determined parameters.

When the least-squares solution is performed using standard subroutines, we not only obtain the parameters listed above but also residuals together with variances and confidence-intervals of the parameters. Using these, the criterion for the porous medium approach to be adequate is

a) The confidence intervals should not contain the point zero.
b) The residuals should not contain any significant trends.

The last statement means that if the residuals are approximated by a sum of Legendre-functions, none of the terms should be significant. With this criterion it is seen that the test on instrumentation- and measurement quality can be performed only in those cases where the flow region has been accepted as a porous medium. A measure of quality is then given by the largest quotient sqrt(variance) / (adjoined parameter). A small value ($\cong 0$) indicates good quality while a large value ($\cong 1$) indicates bad quality.

The measurement procedure uses point sources to generate the necessary transients. As a consequence of the requirement of continuity mentioned above, a source is a point source as soon as it is smaller than some critical length specific to the porous medium. In the case of fractured rock, this critical length is so large that a borehole used as a source is either a point source or a line source. As infinite line sources are bad approximations for wells in a medium with conductivity gradients, the point source approach is chosen due to mathematical simplicity.

In the field we will face two problems: The boreholes must be cased along their entire length exluding source and measuringpoints. We must also have some procedure to test if the source and the measuringpoints are connected to the porous medium to be investigated. The first problem is one of engineering and will not be discussed here. The second problem can be solved in the following way: A borehole is used as a finite line source by introducing two packers with a distance between them that is large compared to the critical length. For this line source we determine the water loss per unit length of the source. If this procedure is repeated for a point source, we will generally not obtain the same value due to the length of the point source being to short (below the critical length). The criterion for the point source being connected is then that the water loss per unit length is larger than that determined for the line source. The same of course goes for the measuringpoints.

The critical length will sometimes be so large that it must also be considered when deciding for a suitable distance between the boreholes used in the measurement. These distances forced to be large, the time used for measurement will also be large. Using values for the time constant from available literature, this constant is found to span some 8 orders of magnitude. The time needed for a specific measurement can therefore only be found by trial. However, it is felt that the necessary time is such that automatic devices should be used to keep the water loss and tracer concentration constant in the source. In most cases it will also be practical to use automatic devices to register the pressure heads and tracer concentrations in the measuringpoints.

As the procedure for measurement uses several sources in turn (source circulation), pressure heads in a measuringpoint will generally be caused by several sources, effects of a previous source not yet having died away. If the measurements are performed in such a way that this is suspected, the potentials from the previous sources should be added to the potential from the last source in the theoretical expression used to determine the parameters. Also, if different tracers are used for different sources, a new source can be introduced without waiting for the concentration from the old source to die away.

KBS Technical Report No 80-06

ALTERNATIVE METHODS FOR SOLIDIFICATION OF ION EXCHANGE RESINS AND WASTE SOLUTIONS

Claes Thegerström
Studsvik Energiteknik AB, 1980-01-29

Summary

Methods, that are used or are under development for solidification of radioactive ion exchange resins or liquid concentrates, utilize normally cement, bitumen or some polymere as matrix material.

This report contains a review and a description of these solidification processes and their products, especially of relatively new techniques that are under development in different countries.

The following methods are discussed:

- Incorporation of liquid waste in urea-formaldehyd. A method that has been used at power plant in the United States. Drawbacks of the method are formation of free acidic water at solidification and a low product quality.

- Solidification of dewatered ion exchange resins in polystyrene/divinylbenzen. A mobile system, FAMA, developed by GNS/STEAG in the Federal Republic of Germany and used for campaign solidification of medium-level resins at power plants in FRG and Belgium.

- Solidification in polyethylene with extruder A method used at some power stations in FRG and Netherlands. It seems however that in future bitumen will be used due to the high viscosity and low radiation stability of polyethylene.

- Incorporation of liquid waste and resins in polyesteremulsion. A method developed by Dow Chemicals. The liquid wastes are micro-encapsulated in the plastic matrix by forming an emulsion of the waste and the polymere. Full scale campaigns have been done at power stations in the United States. A similar method is studied at Washington State University, Pullman.

- Solidification of dried concentrates and dewatered resins in polyester. This method is used since 1975 at the Nuclear Research Center in Grenoble. A full-scale plant has been built by ECOPOL and EDF at the power station SENA-CHOOZ. It is also possible to use epoxi as matrix material.

- Evaporation of waste concentrates and slurries in hot silicone oil and solidification in epoxi resin. A system (Inert Carrier Radwaste Process, ICRP) under development by United Technologies and General Electric.

- Solidification in bitumen. A wellknown method. The caracteristics of different bitumen processes are shortly reviewed in the report.

- Solidification in concrete. There are now also mobile systems for solidification in concrete. Methods for polymere impregnation of concrete are under development in the U.S. and in Italy.

- Other methods. To be mentioned for instance methods for solidification in polymere-cement composites.

Thus many different solidification techniques are under development. They all have advantages and disadvantages and no method could be judged as a radical technical-economical improvement. Utilities in different countries are waiting to get the experience from industrial pilot-scale application.

It is possible that solidification in thermosetting resins will be more used in the future, especially when product quality requirements are high (for instance when solidifying medium level resins) or when special waste categories has to be solidified. However it is not probable that thermosetting resins will be extensively used in a broad application as matrix material. In that case the methods are to complicated and expensive compared to, for instance, solidification in concrete.

Systems for incorporation in polyesteremulsions (Dow-process) have a potential as they are quite simple and can accept a large variation of liquid wastes.

Some methods in an early stage of development (for instance Inert Carrier Radwaste Process) will have to be tested in active application before they can be further evaluated.

Information about the properties of waste products is often missing or incomplete. Especially it would be very valuable if the different waste products were tested and compared at independent laboratories.

KBS Technical Report No 80-07

A CALCULATION OF THE RADIOACTIVITY INDUCED IN PWR CLUSTER CONTROL RODS WITH THE ORIGEN AND CASMO CODES

Kim Ekberg
Studsvik Energiteknik AB, 1980-03-12

SUMMARY

The radioactivity induced in PWR cluster control rods during reactor operation has been calculated using the computer programme ORIGEN. Neutron fluxes and spectrum conditions as well as the strongly shielded cross sections for the absorber materials Ag, In and Cd have been obtained by running the cell and assembly code CASMO for a couple of typical cases.

The results show that Ag-110m, Fe-55 and Co-60 give the largest activity contributions in the interval 1-10 years after the end of irradiation, and Ni-63 and Cd-113m in a longer time perspective.

KBS Technical Report No 80-08

GROUNDWATER DATING BY MEANS OF ISOTOPES

A brief review of methods for dating old groundwater by means of isotopes

A computing model for carbon - 14 ages in groundwater

Barbro Johansson
Naturgeografiska Institutionen
Uppsala Universitet, August 1980

SUMMARY

Isotopes useful for dating

Carbon-14 is the isotope most used today for dating old groundwater. The problems of the method are rather well known. In the future, it maybe possible to use other isotopes. The one which seems to be the most "promising" for very old water is chlorine-36 Both chlorine-36 and carbon-14 are radioactive isotopes produced in the atmosphere. Their concentration in the atmosphere is known, and the age of a watersample can be computed from the laws of radioactive decay.

The ratio uranium-234/uranium-238 in a watersample can be used for dating, along with the accumulation of helium-4. These methods seems to be rather complicated to use, but are deemed to be worth studying.

The short half-lives of the radioactive isotopes argon-39 and silicon-32 make them unsuitable for dating old water, but they can still provide important information.

Oxygen-18 and deuterium are valuable mainly for checking ages determined with other methods.

Carbon-14 in groundwater dating

Carbon dioxide, released by root respiration of plants, and by decomposition of organic matter in the soil, is dissolved in soil water. The production of carbon dioxide in the root zone is high and the $^{14}C/^{12}C$ ratio is therefore assumed to be the same as in the plants. The residence time of water in the groundwater zone (a closed system) may then be computed, according to the laws of radioactive decay.

$$t = -\frac{5730}{\ln 2} \ln \frac{^{14}C}{^{14}C_0'}$$

^{14}C = the activity of the water

$^{14}C_0'$ = the activity of the plants

t = the residence time of water in the groundwater zone

However, this formula cannot always be used. Calcite weathering in the groundwater zone brings carbon of low activity into the water, which results in a too-high computed age if the formula above is used. The assumption that the $^{14}C/^{12}C$ ratio in the soil void air is the same as in the plants is valid only when the amount of carbon dioxide produced biologically considerably exceeds the amount of carbon brought into the water through weathering. This is not always the case if calcite dissolution takes place well below the root zone but above the groundwater table.

A more general formula for computing the residence time of water in the groundwater zone would be

$$t = -\frac{5730}{\ln 2} \ln \left(\frac{^{14}C}{^{14}C_0} \cdot \frac{C}{C_0} \right)$$

^{14}C = the activity of the water

$^{14}C_0$ = the activity of the water when it reached the groundwater zone

C_0 = the amount of carbon in the water when it reached the groundwater zone

$C = C_0$ + the amount of carbon released in the groundwater zone

No attempts have been made to compute the changes in the carbon-14 content of the soil air due to weathering. To do this, one would need to consider diffusion of gaseous carbon dioxide into and out of the soil. The amount of carbon entering the water through weathering in the groundwater zone may be computed if certain assumptions are made. To know if these assumptions are valid for the water of a special area, detailed knowledge about the area (including its past) is required. This knowledge is normally not available, hence the computed age is still uncertain. A method where the amount of calcite weathered in the groundwater zone is computed is used in the computer program ISOTOP, developed by Reardon and Fritz (1978).

Another method is described in this report. In this method, an attempt is made to follow the changes in the composition of the water as the water moves through the ground. The differentiated equilibrium equations of the carbon dioxide system and the ionic balance are used for the calculations. It is assumed that when calcite is present in the ground, weathering of other minerals may be neglected. At the ground surface, it is assumed that $\{HCO_3^-\} = \{H^+\}$ A value of the ratio between the amount of calcite weathered and the amount of other minerals weathered has to be assumed.

The effects in the calculations of two variables on the computed age are tested.

These variables are a) the pH when the water becomes saturated with respect to calcite, b) the pH when calcite weathering starts (calcite is not necessarily present in all parts of the ground). The calculations are performed in a FORTRANprogram.

In order to test its usefulness, the method has been tried on groundwater from a borehole in Kråkemåla (Kl, 493m). The results are very much dependent on the values of some of the parameters used in the calculations. The $\delta^{13}C$ values (of the soil air, the water sample and the weathered calcite) especially have a great influence on the calculated age. Very small changes in the measured carbon-14 activity of the water sample also lead to great changes in the computed carbon-14age. As long as additional information on conditions at different depths remains unavailable, it seems impossible to determine the age of water with any accuracy Only a range, which sometimes embraces several thousand years, can be given. A good aid to a better estimate of the age would be obtained if samples of water along a flow path were available. One way to get such samples would possibly be to drill close to the groundwater divide.

KBS Technical Report 80-09

THE BERGSHAMRA EARTHQUAKE SEQUENCE OF DECEMBER 23, 1979

Ota Kulhánek, Norris John, Klaus Meyer,
Torild van Eck and Rutger Wahlström
Seismological Section, Uppsala University
Uppsala, Sweden, August 1980

SUMMARY

On December 23, 1979 an earthquake sequence occurred near Bergshamra-Roslagen, Sweden, about 50 km northeast of Stockholm. The main shock, which has been assigned a magnitude $M_L = 3.2$, has been followed, with a 3 minute delay, by a shock of magnitude $M_L = 2.6$ and, with additional 21-minute delay, by a third shock of magnitude $M_L = 2.0$. Whereas the main shock was recorded by almost all Finnish, Norwegian and Swedish permanent stations, the whole sequence has been observed only at UPP ($\Delta = 68$ km). A six-week field survey in the epicentral area revealed a number of small aftershocks located close to the main shock. The first, largest shock of December 23 was generally felt over an area of about 150 km^2 with maximum intensity V+ (MM) on western Löparö, indicating a very shallow hypocentre with focal depth of about 2 km. Pg- and Pn-readings suggest for the main shock reverse faulting with a strike of N36^{o}E and a dip of 55^{o} to the south-east. Spectral analysis applied to available seismograms from Finnish, Norwegian and Swedish stations gave for the dynamic source parameters a fault length of 0.8 km, a seismic moment estimate of 6.6×10^{20} dyne cm and a stress drop of 6 bars.

The Bergshamra sequence took place in a zone of very low seismicity in eastern central Sweden and for Swedish earthquakes at unusual shallow depth. Since the epicentre lies less than 100 km from a nuclear power plant in Forsmark, the sequence received publicity which was not in proportion to the size of the shock. At this occasion, some rather strange explanations of the shock emerged.

KBS Technical Report No 80-10

SUPPLEMENTARY GEOHYDROLOGICAL INVESTIGATIONS IN THE FINNSJÖ AREA

Leif Carlsson,
Bengt Gentzschein,
Gunnar Gidlund,
Kenth Hansson,
Torbjörn Svenson,
Ulf Thoregren
Geological Survey of Sweden, Uppsala, May 1980

SUMMARY

The Geological Survey of Sweden (SGU) has conducted some supplementary geohydrological investigations in the Finnsjö area in northeastern Uppland on contract with Project Kärnbränslesäkerhet, KBS, (The Swedish Nuclear Fuel Safety Project). Four new cored boreholes, Fi 4 - Fi 7, were drilled during the winter 1978 - 79. The boreholes are between 552.7 and 750.5 m long and have a diameter of 56 mm. Their inclination varies between 51^{o} and 90^{o}. They were all thoroughly flushed with compressed air before the borehole-investigations.

The field work included water injection tests with double packer equipment in all four boreholes and with single packer equipment in two of the holes, Fi 4 and Fi 6. Hydraulic conductivity, k, was evaluated using a formula assuming a stationary state.

In the double packer tests, water was injected under excess pressure in a 3 m long section between the packers. In the single packer tests, water was injected between the lowermost packer and the bottom of the borehole. The packers were pressed against the borehole wall in hydraulic-mechanical way. The groundwater head and the injection pressure in the tested section, were monitored by a pressure transducer in hydraulic connection with the section but placed close to the upper packer. In most cases excess pres-

sure of 0.2 or 0.4 MPa was used for the double packer measurements, but other excess pressures were also used for the measurements with the single packer. The water was injected through a steel pipe and flow measured by flow gauges of the rotameter type. They could measure flows of between 0.00085 l/min and 65 l/min.

The sources of error associated with this method can be divided into practical-technical and theoretical ones. The practical errors tend to overestimate the hydraulic conductivity of the rock mass, partly due to leakage and short-circuit flows around the packers. Also elasticity in the equipment used, might influence the values of measurements.

The results of the waterinjection tests with double packers showed great variations. In the borehole Fi 4 three sections with high conductivity values were observed, of 87 m, 51 m and 50 m lengths. In these sections the values varied between 5.0×10^{-10} and 5.0×10^{-5} m/s. Below 392 m vertical depth the conductivity was between 1.9×10^{-10} (measuring limit) to 1.2×10^{-8} m/s, and a general decrease of hydraulic conductivity with depth was noticed.

In the borehole Fi 5, which was drilled through a topographically indicated tectonic zone, a highly permeable section of about 75 m was registered(164 - 239 m below groundsurface).The conductivity varied between 1.3×10^{-7} m/s up to more than 1.2×10^{-4} m/s. Also in the rest of the borehole comparatively high values were obtained. The measuring limit was reached only in one 3 m section. Below 386 m vertical depth, however, all values were below 5.6×10^{-8} m/s.

The most low permeable zone in the borehole Fi 6 was found above the 200 m level. A highly permeable section (250 - 271 m) exhibited conductivity values of between 1.0×10^{-7} and 2.2×10^{-5} m/s. Below that, a 132 m section with a conductivity variation between 2.0×10^{-9} m/s and 8.1×10^{-7} m/s was registered,except for two lower values. The lowermost 275 m of the borehole had hydraulic conductivities between 3.7×10^{-10} m/s and 4.5×10^{-7} m/s.

In Fi 7, which is the least permeable borehole, no value higher than 6.7×10^{-9} m/s could be found below 350 m vertical depth. Above this level, however, higher hydraulic conductivities up to 8.8×10^{-6} m/s were obtained.

With the single packer equipment hydraulic conductivities from 6.2×10^{-8} m/s (in Fi 4) to 6.8×10^{-8} (in Fi 6) were found in the section between the highest packer position used and the bottom of the borehole.

Comparison between double and single packer measurements in 50 m long sections (sometimes other lengths) showed, that the agreement between the methods usually was good. In the low permeable parts towards the bottom of many boreholes the double packer measurements gave higher permeability than those with single packer equipment.

The double packer measurements with 0.2 MPa and 0.4 MPa showed that the higher excess pressure used gave lower hydraulic conductivity in a majority of the sections with values above 1.0×10^{-9} m/s. When the hydraulic conductivity was below 1.0×10^{-9} m/s, 0.4 MPa induced higher conductivity in most cases. The differences were, however, small in most cases.

The single packer measurements at different pressures gave no uniform picture of the conductivity differences. However, high pressures usually gave lower hydraulic conductivity.

Repeated measurements with single and double packers were made in totally 14 sections. At no occasion discrepancies larger than a factor of 10 were noticed.

KBS Technical Report No 80-11

WATER UPTAKE, MIGRATION AND SWELLING CHARACTERISTICS OF UNSATURATED AND SATURATED, HIGHLY COMPACTED BENTONITE

Roland Pusch
Division Soil Mechanics, University of Luleå
Luleå 1980-09-20

SUMMARY

The experimental part of the study was conducted by means of swelling pressure oedometers. Samples of Na and Ca bentonite were compacted in such oedometers at high pressures and were then water saturated at constant, as well as increased volume conditions. The time required for the development of a constant swelling pressure formed the basis of the formulation of a mathematical model to describe the rate of water uptake. The simple diffusion equation was found to be applicable for this purpose and later tests on a larger scale gave further support of this model.

The parameters which govern the rate of swelling of saturated clay are the "suction" potential (negative pore pressure) and the permeability, if the access of water is not a limiting factor. In practice, this is probably very often the case and the choice of a host rock with very few joints may lead to considerable delay in the uptake of water.

A very important conclusion was drawn from field observations with respect to the required property of bentonite to swell spontaneously if it gets an opportunity to occupy larger space. The swelling potential of very old, smectite-rich clays in nature indeed offer excellent evidence of this ability.

KBS Technical Report No 80-12

DRILLING HOLES IN ROCK FOR FINAL STORAGE OF SPENT NUCLEAR FUEL

Gunnar Nord
Swedish Detonic Research Foundation

SUMMARY

This report deals with the technical and economic aspects of the drilling of vertical holes with diameters of 1.5 metres and 1 metre in the Swedish bedrock. The holes will be 7.7 metres in depth and located on a level approximately 500 metres below the ground surface.

At the present time, there is no directly applicable technique for the construction of the above-mentioned holes from a small tunnel. On the other hand, it is possible to develop the equipment to construct the holes by making slight changes in existing techniques. The data presented in this report are based to a great extent on information supplied by the manufacturers of drilling equipment, and by underground construction contractors.

Three different techniques for drilling the holes have been dealt with in the report: shaft boring, stitch drilling (three alternative methods) and core drilling. In order to produce the required 233 holes per year, the following sets of equipment must be purchased, and personnel engaged. A cost estimate has also been prepared, and the margin of error in this calculation is considered to be ± 15 %.

Hole diameter 1.5 m, depth 7.7 m.

	Shaft boring	Stitch drilling method A	Stitch drilling method B	Stitch drilling method C	Core drilling
Sets of equipment	3	7	5	3	7
No. of employees	12	27	23	15	29
Cost per hole (Skr)	24,200	31,200	28,800	25,500	55,300

Hole diameter 1.0 m, depth 7.7 m.

	Shaft boring	Stitch drilling method A	Stitch drilling method B	Stitch drilling method C	Core drilling
Sets of equipment	3	5	4	3	6
No. of employees	10	22	18	12	25
Cost per hole (Skr)	19,700	25,800	23,200	21,200	43,900

Interest costs for equipment acquired should rightly be included in a calculation of the costs, but have been listed separately in this report. These interest costs have have been shown to have very little effect on the relative costs of the different methods. Interest costs account for 10% - 15 % of the above-mentioned costs. Our aim has been to calculate the costs for the different methods on as similar a basis as possible, but a margin of error of ± 15 % entails an overlapping of the cost span for most of the methods considered.

KBS Technical Report No 80-13

SWELLING PRESSURE OF HIGHLY COMPACTED BENTONITE

Roland Pusch
Division Soil Mechanics, University of Luleå
Luleå 1980-08-20

SUMMARY

The experimental study required the development of a sufficiently rigid and dependable swelling pressure oedometer. Samples of Na and Ca bentonite were compacted in such oedometers at high pressures. During water saturation under constant volume conditions the exerted swelling pressure was found to increase stepwise, which can be explained by various phases and microstructural reactions in course of the water uptake.

It was concluded that the pore water chemistry is hardly a determinant of the swelling pressure in the bulk density interval of primary interest ($\rho > 2\ t/m^3$).

The explanation of this is probably that electrical double-layers are not fully formed at high densities. The practical implication would be that future ground water changes will not alter, appreciably, the swelling pressure. It also seemed to be of similar magnitude for the investigated Na and Ca bentonites.

The influence of temperature is not very obvious but some reduction is expected at higher temperatures although it will be very moderate (30-50% at maximum) if an increase only from 20 to 70 ^{o}C is considered.

KBS Technical Report No 80-14

PROPERTIES AND LONG-TERM BEHAVIOUR OF BITUMEN AND RADIOACTIVE WASTE-BITUMEN MIXTURES

Hubert Eschrich
Eurochemic, Mol, October 1980

ABSTRACT

This report consists of two main parts.

Part I represents a survey of the properties and the long-term behaviour of pure bitumens and mixtures of bitumens with radioactive reactor and reprocessing wastes.

This survey includes information on the origin, amounts, and composition of the various wastes considered for bituminization and the different waste bituminization techniques used. The influence of various factors on the quality of waste-bitumen products and on the radiological safety during transport, short- and long-term storage of the final products is described.

Special consideration is given to the most important safety relevant factors associated to the use of bitumen as matrix material for radioactive wastes, such as leach-resistance, radiolysis, chemical and mechanical stability, combustibility, and microbial attack.

Part II consists of a comprehensive bibliography on the bituminization of radioactive wastes, giving about 300 references to literature published from the beginning of the use of bitumen in radioactive waste management in 1960 until the beginning of 1979.

The bibliography serves at the same time as source for the literature referred to in the text of Part I of this report.

Methods for the quality control of bituminous materials and some useful data are given in an annex.

KBS Technical Report No 80-15

ALUMINIUM OXIDE AS AN ENCAPSULATION MATERIAL FOR UNREPROCESSED NUCLEAR FUEL WASTE - EVALUATION FROM THE VIEWPOINT OF CORROSION

Final Report 1980-03-19

Swedish Corrosion Institute and its reference group

SUMMARY

To fulfil the requirements of the so-called "Stipulation Law" the Nuclear Fuel Safety Project (KBS) has proposed that spent unreprocessed nuclear fuel shall be disposed of by encapsulation in canisters of high-purity alumina sintered under isostatic pressure. The canisters will have a wall thickness of 100 mm and are to be placed in vertical boreholes extending from horizontal tunnels 500 m below ground in igneous rock. In each borehole one canister is deposited embedded in a quartz sand/ bentonite buffer.

The Swedish Corrosion Institute has been assigned the task of evaluating this proposal from the viewpoint of corrosion and of estimating the life of the canisters under the given conditions. To do this work, the Swedish Corrosion Institute has appointed an expert group of 10 Swedish specialists, mainly from the fields of corrosion and materials technology. The expert group arrived at the following conclusions. Witn one exception (G. Wranglên) the group was unanimous in its evaluation.

The alumina is not thermodynamically stable in water. In pure water hydration will occur, below 100^{o} C leading to the formation of either $Al(OH)_3$ in the amorphous state or crystalline gibbsite ($Al_2O_3 \cdot 3H_2O$). Corrosion may take place by slow dissolution or flaking off of a surface layer. Various immersion tests showed that the corrosion rate will be less than 0.1 μm/year, probably one or two powers of ten lower.

If the alumina canister in the storage has sufficiently large surface defects and is under sufficiently high mechanical tension the defects may grow slowly into propagating cracks, ultimately leading to fracture, so-called delayed fracture. On the basis of results from fracture mechanical studies and after introduction of safety factors with respect to possible unknown features of the delayed fracture it was judged possible to eliminate the risk of delayed fracture if the canisters pass the following production control:

- Proof testing at 150 MN/m^2, using acoustic emission technique to ensure that crack growth does not occur during the unstressing cycle.
- Surface acoustic wave examination with respect to surface inclusions, canisters with inclusions larger than 100 µm within a 100 µm deep surface zone being rejected.

Canisters which pass the production control mentioned are estimated to have a life of hundreds of thousands of years, or probably considerably more, under the conditions specified.

It may be questioned whether the very hard production control prescribed is reasonable. As the knowledge of delayed fracture in ceramics will increase, however, it will certainly be possible to relax the conservative requirements.

KBS Technical Report No 80-16

PERMEABILITY OF HIGHLY COMPACTED BENTONITE

Roland Pusch
Division Soil Mechanics, University of Luleå
1980-12-23

SUMMARY

Samples of Na and Ca bentonite were compacted in oedometers at high pressures. After water saturation under constant volume conditions, percolation experiments were conducted to measure the permeability. It was found that high densities reduce the permeability to very low values ($k \sim 2 \cdot 10^{-14} - 5 \cdot 10^{-14}$ m/s when the bulk density is 2.1 t/m^3).

The experiments showed an obvious gradient-dependence of the permeability. Also, it was found to be sensitive to temperature changes. Thus, at 70^{o}C the permeability is roughly 5 - 10 times higher than at 20^{o}C. As concerns the influence of the adsorbed cation type it was concluded that Ca bentonite is 2 - 5 times as permeable as Na bentonite at higher densities than 1.8 t/m^3.

The gradient-dependence indicates that the water migration is largely dependent on physico/chemical and microstructural features. It can be concluded that while water flow through clays of low density can be considered as the motion of a fluid with a definite and constant viscosity, it should rather be regarded as a shear-induced displacement of a "structured" medium in the case of dense bentonites.

KBS Technical Report No 80-17

INPUT DESCRIPTION FOR BIOPATH

Jan-Erik Marklund
Ulla Bergström
Ove Edlund
Studsvik Energiteknik AB, 1980-01-21

SUMMARY

Two versions of BIOPATH, BIOPATH-1 and BIOPATH-2 have been defined. The main difference is that BIOPATH-2 can handle transfer matrices that vary with time. A short summary of the basic theory for BIOPATH is given in chapter 2.

The control cards needed to run the versions are discussed in chapter 3.

In chapter 4 the input data for both versions are presented. BIOPATH-2 needs some additional input data compared to BIOPATH-1, in order to specify the integration method to be used and the time dependence of the transfer coefficients.

An example of a possible set of input data is presented in chapter 5.

KBS Technical Report No 80-18

INTRODUCTION OF TIME INDEPENDENT COEFFICIENT MATRICES IN BIOPATH

Jan-Erik Marklund
Studsvik Energiteknik AB, January 1980

SUMMARY

The computer program BIOPATH /1,15/ computes the dispersion of materia (normally radioactive nuclides) within a set of "compartments" (e.g. different eco-systems) by solving the differential equation.

(1) $y' = Ay + g$

The program then computes activity and dose burden by application of suitable factors /1/.

The present report gives an account of some amendments and changes that have been made to BIOPATH in the winter 1979/80. It also contains a short description of the main programming features, as well as an account of some testcases that have been run.

The main objective of the work has been to make it possible to treat such problems, where the coefficients of the dispersion matrix depend on time. This has been accomplished by adding a number of subroutines defining and controlling the variation of the coefficients as well as some new integration routines. Another goal has been to improve the accuracy for problems containing nuclides with very slow decay. This has been

achieved by the introduction of integration routines suited for stiff differential equations /4-8, 11/. The new version thus created is called BIOPATH-2 and the old version BIOPATH-1. For such problems, that are possible to treat with either version, the two versions should give identical results.

The input data for both versions are described in a separate report /15/.

A number of testcases have been run in order to check the new integration methods and the models for the variation of the coefficients. The tests have confirmed the aptness of the new models and that the new integration methods in some cases are considerably more accurate than the old ones. In most cases, however, the new integration method and the old one give results that are practically identical.

KBS Technical Report No 80-19

HYDROTHERMAL CONDITIONS AROUND A RADIOACTIVE WASTE REPOSITORY

Part 1 A mathematical model for the flow of groundwater and heat in fractured rock
Part 2 Numerical solutions

Roger Thunvik
Royal Institute of Technology, Stockholm, Sweden

Carol Braester
Israel Institute of Technology, Haifa, Israel
December 1980

ABSTRACT PART 1

A mathematical model describing the hydrothermal conditions around a hard rock repository for disposal of nuclear fuel waste is presented. The model was developed to study the effect of heat released from a radioactive waste repository on the flow times from the repository to the ground surface. The model consists of a set of coupled non-linear partial differential equations for heat and ground water flow. In addition there are two equations of state relating fluid density and viscosity to pressure and temperature. The system of equations is solved numerically using the finite element method in either two or three dimensions. The model is based on the continuum approach. The fractured rock is treated either as two overlapping continua in which the one represents the network of fractures and the other the solid blocks or as a single equivalent medium. The first approach assumes quasi-steady state heat transfer from the rock to the fluid, allowing a linear heat transfer function to be used. The second approach assumes instantaneous equilibrium between the fluid and the rock. This report presents the theoretical background of the model. Numerical solutions for the problems are contained in a separate report entitled "HYDROTHERMAL CONDITIONS AROUND A RADIOACTIVE WASTE REPOSITORY, Part 2 - Numerical Solutions".

ABSTRACT PART 2

Numerical solutions for the hydrothermal conditions around a hard rock repository for nuclear fuel waste are presented. The presented solutions illustrate the effect of heat released from a hypothetical radioactive waste repository under various conditions with regard to topography, the location of the repository, permeability, porosity and adjacent fracture zones. Major interest in the analysis is directed towards flow patterns and travel times for water particles from the repository to the ground surface or alternatively to a major fracture zone in the vicinity of the repository. The solutions were obtained using a mathematical model for the flow of groundwater and heat through a fractured rock mass. The model consists of a set of coupled non-linear partial differential equations for heat and groundwater flow. The set of equations is solved numerically using the finite element method in two or three dimensions. The presented results show that the heat emitted by the decaying radioactive waste may have a significant influence on the initial flow patterns and subsequently also on the travel times from the repository under certain conditions.

KBS Technical Report No 80-20

BEGAFIP. PROGRAM UPDATING, DEVELOPMENT AND BENCH MARK CALCULATIONS

Göran Olsson
Peter Hägglöf
Stanley Svensson
Studsvik Energiteknik AB, 1980-12-14

ABSTRACT

This report summarizes improvements to BEGAFIP (the Swedish equivalent to the Oak Ridge computer code ORIGEN). The improvements are

- addition of a subroutine making it possible to calculate neutron sources
- exchange of fission yields and branching ratios in the data library to those published by Meek & Rider in 1978.

In addition, BENCHMARK-calculations have been made with BEGAFIP as well as with ORIGEN regarding the build-up of actinides for a fuel burnup of 33 MWd/kg U. The results were compared to those arrived upon from the more sophisticated code CASMO.

The work performed was made under contract with SKBF (Swedish Nuclear Fuel Supply Co).

KBS Technical Report No 80-21

REPORT ON TECHNIQUES AND METHODS FOR SURFACE CHARACTERIZATION OF GLASSES AND CERAMICS

Bengt Kasemo
Mellerud, August 1980

Summary

This report is intended to serve as a basis for experimental studies of corrosion phenomena on surfaces of glasses and ceramics. An introductory orientation is given about the problems connected with quantitative surface analysis and about the related surface physics questions. The main part of the report is a description and evaluation of the existing surface sensitive methods. Conclusions are drawn about suitable combinations of methods for characterization of surfaces of glasses/ceramics. Alternative solutions with respect to capability and cast are given. A rather detailed list of references is supplied covering the relevant literature about the mentioned methods and phenomena. Finally, the report contains names of laboratories where surface analytical investigations are performed (adress, contact persons, methods used and the cost per hour or day).

KBS Technical Report No 80-22

EVALUATION OF FIVE GLASSES AND A GLASS-CERAMIC FOR SOLIDIFICATION OF SWEDISH NUCLEAR WASTE

Larry L Hench
Ladawan Urwongse
Ceramics Division
Department of Materials Science and Engineering
University of Florida, Gainesville, Florida
1980-08-16

SUMMARY

A study of the relative leaching resistance four borosilicate glasses, one alkali-alkaline earth alumino-silicate glass and one crystallized glass or glass ceramic has been performed. All six materials had a simulated waste loading of $9^{w}/o$ and a low melting temperature, $< 1150^{o}$ C, required for the French AVM process. The findings were: Leach rates of zinc borosilicate glasses in 90^{o}C deionized water are in the range $5 \times 10^{-7} - 5 \times 10^{-6}$ g/cm^2 day for Na^+ and Si^{4+}. These values are better than for soda-borosilicate glasses ($1\text{-}5 \times 10^{-5}$ g/cm^2 day). Increasing the Fe_2O_3 content in either sodaborosilicate glasses or zinc borosilicate glasses improves the leach resistance by a factor of 3-5. Fe_2O_3 additions appear to concentrate within the surface films as leaching proceeds.

The alkali-alkaline earth-alumino-silicate glass had a leach resistance nearly equivalent to a high Fe_2O_3 zinc borosilicate glass. However, a crystallization of that glass seriously degraded its leach resistance, apparently due to attack of a residual vitreous phase in the grain boundaries which concentrates alkali ions.

The leach resistance of alkali borosilicate glasses, zinc borosilicate glasses and alkali-alkaline earth-alumino silicate glasses was not sensitive to surface area to solution volume (SA/V) ratios 0.01 cm^{-1} to 100 cm^{-1}. This was apparently because loss of amphoteric ions into solution prevented the solution pH from becoming alkaline and attacking the glass network more rapidly at high SA/V ratios.

KBS Technical Report 80-23

EXACT SOLUTION OF A MODEL FOR DIFFUSION IN PARTICLES AND LONGITUDINAL DISPERSION IN PACKED BEDS

Anders Rasmuson
Ivars Neretnieks
Royal Institute of Technology, August 1979

SUMMARY

An analytical solution of a model for diffusion in particles and longitudinal dispersion in porous media is derived. The solution is obtained by the method of Laplace transform. The result is expressed as an infinite integral of five dimensionless quantities. The extension for a decaying species is given.

KBS Technical Report No 80-24

MIGRATION OF RADIONUCLIDES IN FISSURED ROCK - THE INFLUENCE OF MICROPORE DIFFUSION AND LONGITUDINAL DISPERSION

Anders Rasmuson
Ivars Neretnieks
Royal Institute of Technology, December 1979

SUMMARY

The migration of radionuclides in the fissures in the bedrock surrounding a repository is discussed. A one-dimensional transport model is presented. It includes diffusion of the nuclides into the microfissures of the rock, and linear sorption and longitudinal dispersion in the bedrock. An analytical solution to the model is given in terms of an infinite integral. The integrand is a sometimes highly oscillatory function of the system parameters. A special integration method is developed to evaluate the infinite integral. The method utilizes the oscillatory behavior of the integrand.

The assessment of input parameters is discussed in some detail. Dimensionless breakthrough curves are given for the approximate range of variation of the input parameters. Calculations are made for a repository of spent fuel surrounded by fissured but fairly good rock ($K_p = 10^{-9}$ m/s and fissure spacing S = 50 m). Longitudinal dispersion may significantly affect the amount of radioactive material reaching the biosphere.

Radionuclides, which would decay completely without longitudinal dispersion, may arrive in non-negligible concentrations. Dispersion effects of the magnitude considered in this study can significantly diminish the retardation effects of matrix diffusion.

KBS Technical Report No 80-25

DIFFUSION AND SORPTION IN PARTICLES AND TWO-DIMENSIONAL DISPERSION IN A POROUS MEDIA

Anders Rasmuson
Royal Institute of Technology, January 1980

SUMMARY

A solution of the two-dimensional differential equation of dispersion from a disk source, coupled with a differential equation of diffusion and sorption in particles, is developed. The solution is obtained by the successive use of the Laplace and the Hankel transforms and is given in the form of an infinite double-integral. If the lateral dispersion is negligible, the solution is shown to simplify to a solution presented earlier. Dimensionless quantities are introduced. A steady-state condition is obtained after long times. This is investigated in some detail. An expression is derived for the highest concentration which may be expected at a point in space. An important relation is obtained when longitudinal dispersion is neglected. The solution for any value of the lateral dispersion coefficient and radial distance from the source is then obtained by simple multiplication of a solution for no lateral dispersion with the steady-state value. A method for integrating the infinite double integral is given. Some typical examples are shown.

CONTRACTORS TO SKBF/KBS DURING 1980

INDUSTRIAL AND CONSULTING COMPANIES

ABV
ASEA AB
ASEA-ATOM
Geological Survey of Sweden
Hagby Bruk AB
Hagconsult
IFÖ Electric Högspänning AB
IPA-Konsult
Jacobson & Widmark
Kemakta Konsult AB
Orrje & Co AB, Ingenjörsfirman
Oskarshamnsverkets Kraftgrupp AB
Rejlers Ingenjörsbyrå AB
Stabilator AB
Statskonsult
Swedish State Power Board
Studsvik Energiteknik AB
Ställbergs Grufveaktiebolag
Sydkraft AB
TEFA-Konsult
Teleplan AB
VBB
VIAK AB

UNIVERSITIES

University of Uppsala

Dep of seismology
Dep of hydrology

Royal Institute of Technology

Dep of agricultural hydrotechnics
Dep of chemical engineering design
Dep of inorganic chemistry
Dep of nuclear chemistry
Dep of soil and rock mechanics

University of Umeå

Dep of inorganic chemistry
Dep of rock mechanics

University of Luleå

Dep of soil mechanics

Chalmers University of Technology

Dep of engineering metals
Dep of nuclear chemistry
Dep of inorganic chemistry

OTHER INSTITUTIONS

Cement and Concrete Research Institute,
Gothenburg
Glass Research Institute,
Växjö
Swedish Corrosion Instiute,
Stockholm
Swedish Detonic Research Foundation,
Stockholm
Swedish Silicate Institute
Gothenburg

FOREIGN ORGANIZATIONS AND COMPANIES

BATTELLE,
USA

Saint Gobain Techniques Nouvelles,
Paris, France

Transnuklear,
West Germany

EXPERTS AND RESEARCHERS

Bergman, Sten G A, Stockholm, Sweden
Braester, Carol, Haifa, Israel
Brotzen, Otto, Stockholm, Sweden
Eschrich, Hubert, Mol, Belgien
Fyfe, William S, London, Canada
Hench, Larry L, Gainesville, USA