UCRL-53154
Distribution Category UC-70

UCRL--53154

DE82 016139

Swedish Nuclear Waste Efforts

Jan Rydberg

Manuscript date: September 1981

LAWRENCE LIVERMORE LABORATORY
University of California • Livermore, California • 94550

Available from: National Technical Information Service • U.S. Department of Commerce
5285 Port Royal Road • Springfield, VA 22161 • $11.00 per copy • (Microfiche $3.50)

PREFACE

The 1976 Parliamentary election in Sweden resulted in a coalition government which imposed extremely stringent requirements for the waste produced by Swedish nuclear power plants. The industry responded with a crash study, the Nuclear Fuel Safety (KBS) project, with experts drawn from hundreds of universities and related scientific institutions. A year later, the industry presented "a complete scheme for absolutely safe storage of nuclear waste" in engineered facilities located at about 500 m depth in the Swedish granite bedrock (KBS-I). This was considered the first comprehensive and coherent scheme for "final" nuclear waste disposal. The KBS-I study was not only extensively examined and debated, it also brought down the coalition government.

This first study was later followed by others (the KBS-II scheme) and extensive research and development work in Sweden on all aspects of nuclear waste management. It is believed that in this area Sweden has the most complete program, attracting considerable international interest. Due to special circumstances I have been engaged on both the industry and government side of this project.

Early in 1980, while spending a sabbatical year from the Chalmers University of Technology in Gothenburg, Sweden, at the Lawrence Livermore National Laboratory, I presented a lecture on the KBS project at the Hoover Institution on War, Revolution and Peace. It met with great interest and--at the request of Dr. R. Williams of the Electric Power Research Institute--I have now written a brief summary of the Swedish nuclear waste activities using many of the actual illustrations from KBS reports. I hope that this synopsis can serve as a somewhat easier source of information than the 2-3 meters of reports so far presented by the KBS project.

I sincerely thank all of the organizations mentioned above and in the text for extensive help in presenting this report. I also want especially to thank Dr. Chris Gatrousis of Lawrence Livermore National Laboratory for his support.

CONTENTS

SWEDISH NUCLEAR WASTE EFFORTS

ABSTRACT

After the introduction of a law prohibiting the start-up of any new nuclear power plant until the utility had shown that the waste produced by the plant could be taken care of in an absolutely safe way, the Swedish nuclear utilities in December 1976 embarked on the Nuclear Fuel Safety Project, which in November 1977 presented a first report, "Handling of Spent Nuclear Fuel and Final Storage of Vitrified Waste" (KBS-I), and in November 1978 a second report, "Handling and Final Storage of Unreprocessed Spent Nuclear Fuel" (KBS II). These summary reports were supported by 120 technical reports prepared by 450 experts. The project engaged 70 private and governmental institutions at a total cost of US$15 million.

The KBS-I and KBS-II reports are summarized in this document, as are also continued waste research efforts carried out by KBS, SKBF, PRAV, ASEA and other Swedish organizations.

The KBS reports describe all steps (except reprocessing) in the handling chain from removal from a reactor of spent fuel elements until their radioactive waste products are finally disposed of, in canisters, in an underground granite depository. The KBS concept relies on engineered multibarrier systems in combination with final storage in thoroughly investigated stable geologic formations. The barrier systems are

- for reprocessed waste: waste bound in glass matrix and surrounded by layers of stainless steel, lead (100 mm) and titanium.
- for unreprocessed spent fuel elements: waste bound in UO_2 matrix surrounded by layers of zircaloy, lead and copper (200 mm).

The canisters are 0.6 and 0.8 m in diameter and 1.8 and 4.5 m long, respectively. The waste is allowed to cool altogether in interim storage for 40 years, until the canisters are emplaced in a bed of bentonite clay holes drilled in the floor of tunnels about 500 m deep in granite bedrock. The clay is highly impermeable to water and provides an elastic support for the canisters. Due to the low waste content of the canisters (each containing waste from about one tonne spent uranium fuel), the thick canister walls, and long cooling times, the radiation and heat load on the surroundings become

almost negligible: maximum canister surface temperature is 77°C and maximum rock temperature 60°C. These low temperatures reduce the corrosion rate of the canister, so its lifetime greatly exceeds 10 000 years. The reducing properties of the groundwater expand this lifetime to over a million years. The low temperatures and radiation fields also ensure that no unexpected chemical reactions will occur between waste, canister materials, clay, rock, and groundwater. Rock retention of the waste nuclides contributes to the KBS prediction that even under the worst possible conditions ("most pessimistic case") the radioactive nuclides, which may leak out, will be in such small amounts, that the doses to critical groups living near the repository will never exceed the natural radiation doses. The KBS thus contends that their waste handling concept leads to an absolutely safe storage of radioactive waste, in compliance with the law.

The KBS project has, in general, received strong support from the large number of Swedish and foreign reviewing agencies, e.g., the U.S. National Academy of Sciences. Since the KBS concept contains many of the conditions set forth in the Department of Energy proposal (DOE/NE 007) it upholds the Department of Energy's view that radioactive waste can be safely disposed of.

This report also briefly describes other activities carried out by the nuclear industry, namely, the construction of a central storage facility for spent fuel elements (to be in operation by 1985), a repository for reactor waste (to be in operation by 1988), and an intermediate storage facility for vitrified high-level waste (to be in operation by 1990). The R&D activities are updated to September 1981.

INTRODUCTION

In 1975 the Swedish Parliament decided on a program of 13 nuclear reactors to be in operation before 1985 and with a capacity of about 10 GW (the 1976 total electric capacity of Sweden was ~20GW, mainly water power). Of these, 6 were in operation at the end of 1976 with a capacity of 3.8 GW, providing about 20% of the Swedish electricity demand. Through the 1976 parliamentary elections a conservative coalition government came into power, strongly supported by antinuclear groups. In October 1976 the new government

declared that, because of the risks of nuclear power, it would allow no new nuclear power stations to be charged until the power industry had demonstrated how and where either the spent fuel elements or the high-active waste from reprocessing could be stored in an absolutely safe manner, in principle, for eternity. This declaration was later reworded in the Stipulation Law (Section 1 below).

In order to comply with the law, the Swedish nuclear power utilities organized the Nuclear Fuel Safety (KBS) Project. Through efficient planning and the cooperation of some 70 private and government institutions, a first report regarding the "Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste" (KBS-I) was ready in November 1977. The power utilities claimed that this report fulfilled the requirements set forth in the Stipulation Law and requested permission from the Government to start up two more reactors. As is customary in Sweden, the report was submitted for the review and comments of Swedish federal organizations, universities and research institutions (not previously involved in the KBS project), and by selected international organizations. Since the coalition partners of the Government could not agree on how to interpret the conclusions of the reviewers, most conclusions being favorable but some being critical, the coalition government resigned in October 1978. The successor conservative minority government in March 1979, approved the KBS-I concept after some further geologic studies, as providing an "absolutely safe waste storage," and issued permits to charge the Ringhals 3 and Forsmark 1 reactors.

In late 1978 the KBS Project issued its second report, concerning the "Handling and Final Storage of Unreprocessed Spent Nuclear Fuel" (KBS-II). KBS-II, like KBS-I, has been sent out for review and comments. These comments have now been received and collected in a report, awaiting further decisions of the nuclear power industry.

The KBS-I and KBS-II projects are described in Section 2 below. Appendix A contains a list of the technical reports so far published (about 150), and the organizations involved in the projects.

The accident at Three Mile Island in March 1979 led to a call for a referendum on nuclear energy in Sweden. In the meantime a "time-for-consideration-law" forbade charging any new reactors, including those which had received permits a few weeks before. After an exceptionally intense and rancorous public debate the referendum was held in March 1980. The voters

gave 58% support (39% against, 3% blank) for the "Yes" alternatives of a maximum of 12 nuclear power stations to be used during their economic and operational lifetime. At this time six reactors were in operation, and three others were ready to be charged. The KBS-II project is likely to be used to support the starting of the three remaining reactors; one will be ready in 1982, the two last ones in 1985.

Presently (Sept. 1981) 9 nuclear power reactors producing a total 6.5 GW are in operation, providing 35% of all Swedish electric power. In order to charge the three last reactors, it may be necessary for the Government to review the KBS-II proposal.

It may be added that in the referendum, about 2/3 of the voters supported the statement that nuclear power should be "a parenthesis," and that no more reactors should be built after the completion of the present 12-reactor program.

The KBS project has tried to address all practical aspects, from the transport of the spent reactor fuels away from the reactor to the permanent deposition of radioactive elements in an underground granite repository. Parallel to this work, resulting in the KBS-I and KBS-II reports, KBS or its parent organization SKBF has supported more long-range studies of geologic repositories (at Stripa) and canister material (at ASEA). The project has also been supported by funds from PRAV (see below). The KBS project has pointed to the need for further studies, particularly short-range pragmatic studies; some of these are described in Section 3.1.

The suitability of mined granite caverns for high-active waste storage has been studied in an abandoned mine at Stripa, Sweden, in a joint project between U.S. laboratories and KBS. Large-scale geophysical and hydrological parameters have been measured. This program has now expanded into an NEA (Nuclear Energy Agency) international project. These activities are summarized in Section 3-2 and in Appendix C.

The technique for encasing spent nuclear fuel elements in canisters of highly corrosion-resistant corundum, developed by ASEA (Swedish General Electric Company), is described in Section 3.3

In 1972 the Swedish Government appointed the AKA committee, with responsibility to look over the issue of radioactive waste and propose actions. In 1974 this committee was replaced by the National Council for Radioactive Waste (PRAV). The KBS project has benefited from its cooperation

with this council, whose main objective is to fund more long-range research and development on waste in the nuclear fuel cycle, as well as on other radioactive wastes. The council's activities are described in Section 3.4. PRAV ceased to exist on June 30, 1981. Recent changes are described in Section 2.9.

1. THE STIPULATION LAW

On April 21, 1977, the Swedish Government prohibited owners of nuclear reactors from loading nuclear fuel into new reactors without special permission from the Government. Permission could be granted only if the owner of the reactor either:

1. had presented a contract for reliably reprocessing used nuclear fuels, and in addition had shown how and where an absolutely safe ("helt säker") final storing of the high-active waste from the reprocessing could be achieved, or

2. had shown how and where an absolutely safe ("helt säker") final disposal of spent unreprocessed nuclear fuel could be accomplished.

The two alternatives are represented in Fig. 1. In the supplementary explanation of the law, it is stated that "absolutely safe" ("helt säker") shall not be interpreted in a "draconian" way. Although there is no official interpretation of "absolutely safe," it has been interpreted by Rydberg and Winchester (Sw. Dept. Energy, DsI 1978:17, p. II:22) to mean "that neither extreme happenings nor slow releases from the waste repository shall lead to radiation doses which now or in the future exceed the dose limits currently recommended by the International Commission on Radiation Protection (ICRP)."

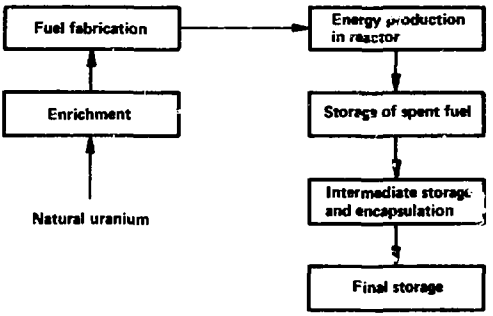


FIG. 1a. The direct deposition alternative (KBS-II). Flow scheme for the fuel cycle with direct storage of the spent fuel without reprocessing.

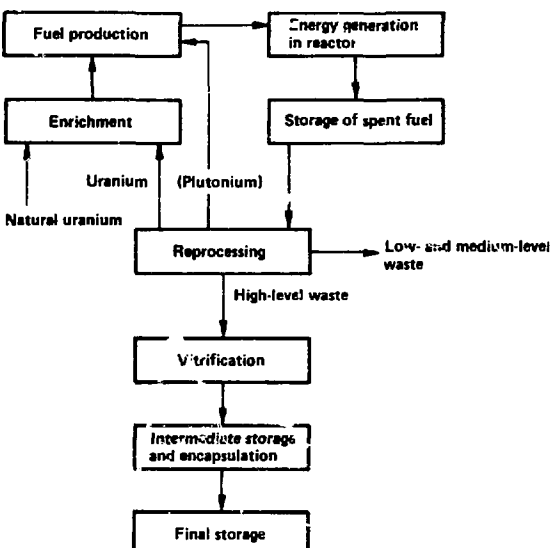


FIG. 1b. The reprocessing alternative (KBS-I). Flow scheme for the fuel cycle with reprocessing of spent fuel and vitrification of the high-level waste.

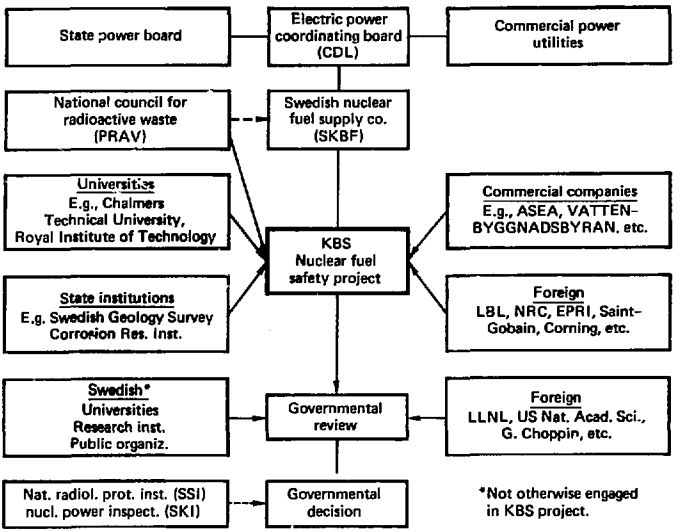
2. THE NUCLEAR FUEL SAFETY (KBS) PROJECT

The Swedish legislation puts full responsibility for investigations and demonstrations of the absolutely safe handling and storage of high-level waste on the owner of the reactor. It should be mentioned that use of the word "demonstration" does not imply that facilities shall be constructed and used to dispose of high-level waste or spent fuel. However, detailed and comprehensive information must be provided for a safety evaluation.

2.1 ORGANIZATION OF THE KBS PROJECT

As soon as the intentions of the new government became known, the Swedish nuclear power utilities, consisting of the State Power Board and private companies, organized a special joint project to meet the requirements of the Stipulaton Law, called the KBS project (Fig. 2). KBS stands for the Swedish word "KärnBränsleSäkerhet," which means nuclear fuel safety. The rapid organization was facilitated by the existing Electric Power Coordinating Board (Centrala driftledningen, CDL), an organization comprising all private and state electric power companies and their subsidiary, the Swedish Nuclear Fuel Supply Company (Svensk Kärnbränsleförsörjning AB, SKBF) in conjunction with the National Council for Radioactive Waste (Programrådet för radioaktivt avfall, PRAV). About 50 million Swedish crowns (~$10 million) were immediately appropriated to the KBS Project Manager, and the time limit for completing a first report on the handling of reprocessing waste ("KBS-I"), which would comply with the demand in point 1 of the Stipulation Law, was set at one year. The main reason for this haste was the fact that two nuclear power stations were standing idle, and every delay in starting up cost the power utilities $1-2 million per week. There was also a psychological reason: with every day the antinuclear forces seemed to grow stronger, supported by the major news media, focusing their criticism on the unsolved waste issue. It was felt that it was important to show that the waste problem could be solved before a majority of the public became convinced that it could never be solved.

The small Project Management Group (some 20 persons) set up an organization according to Fig. 2 and five special reference groups (for Geology, Safety Analysis, Corrosion, Buffer Materials, and Programmes). In addition,



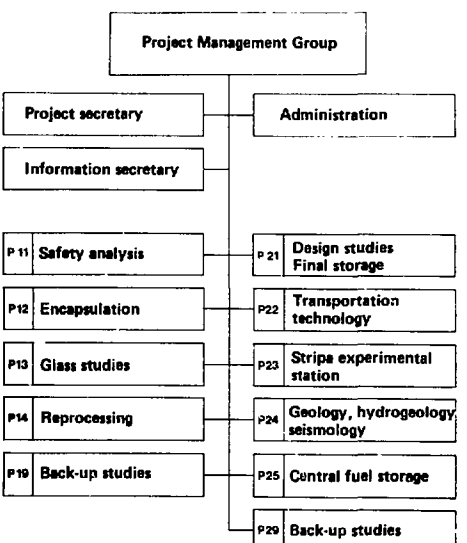


FIG. 2a. Organization plan (institution and scientific projects) for the Nuclear Fuel Safety Project (KBS).

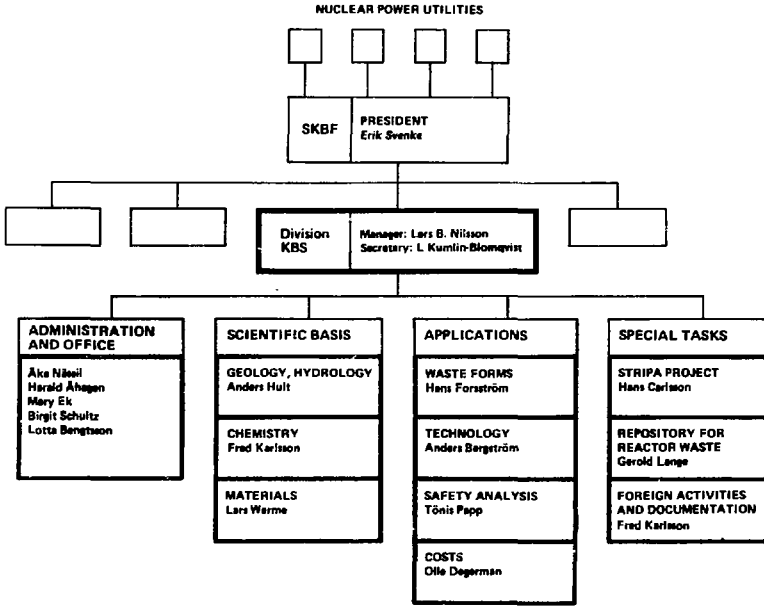


FIG. 2b. Organization chart of KBS management.

some 450 persons were engaged through the contracting of consultants from Swedish and foreign, private and state corporations and research institutions. The consultants are named in Appendix B.

In order for this scheme to work it was essential that the Management Group have a very free hand, in addition to dedication and high working capacity, much exceeding 50 hours per week for extended periods. (The average age of the Management Group was about 40.) It was also necessary to adopt the philosophy of relying on known techniques and known properties of materials, since there would be insufficient time for basic research. By engaging the greatest expertise in the country (and sometimes from abroad) the crucial difficulties were pinpointed early, which led to some immediate decisions among the waste management options (described in next section) as well as immediate laboratory and field studies.

The first KBS report (KBS-I) was finished 8 months after the Stipulation Law, or 13 months after the election of the new government. The main report

was supported by 60 technical reports. The second report (KBS-II) required an additional half year; it was supported by another 60 technical reports. All report titles are given in Appendix A. The expenses for both projects amounted to about $15 million.

2.2 SELECTION OF WASTE MANAGEMENT SCHEME

The KBS studies deal with the handling of reprocessing waste (KBS-I) and the handling of unreprocessed spent fuel (KBS-II). In the following, both alternatives will be considered simultaneously, where possible, because of the desirability of a design which can accommodate both options, leaving final choice to political institutions at a later time. The KBS handling scheme, and particularly the various facilities designed for the handling, comes close to this flexibility, as indicated in Fig. 3. The spent fuel elements are transported from the reactor to a Central Storage Facility (Centrallager för aktivt bränsle, CLAB), where they are allowed to cool for some time. In the reprocessing case, the spent fuel elements are transported abroad to a

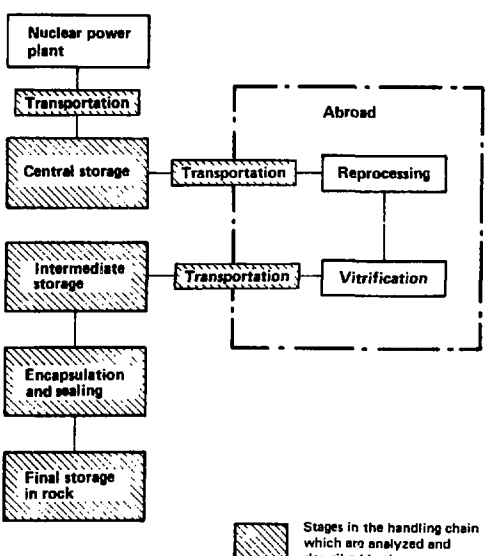


FIG. 3a. Handling chain for spent fuel and vitrified waste.

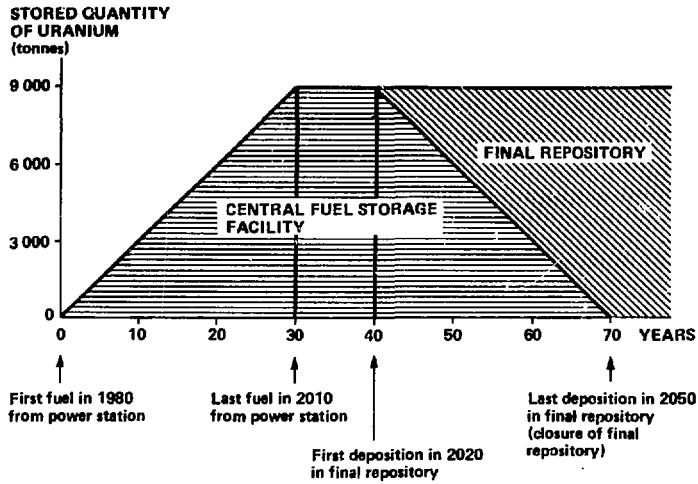


FIG. 3b. Diagram illustrating the capacity requirements from 1980 to 2050 for the central fuel storage facility and the final repository for KBS-II.

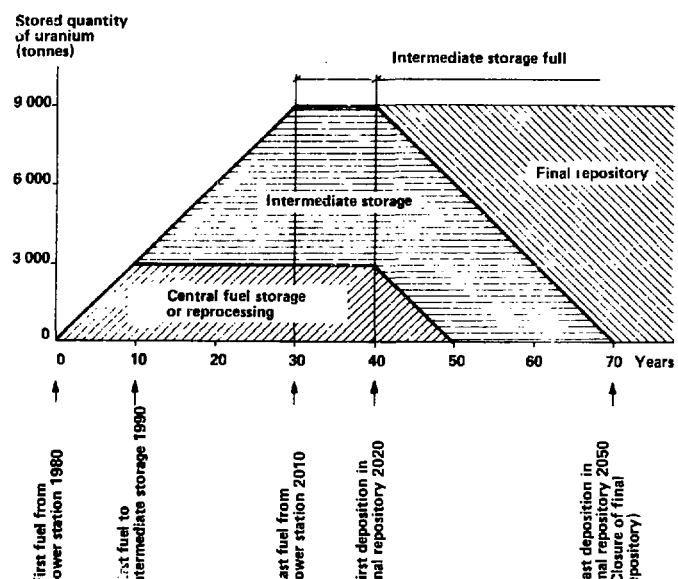


FIG. 3c. Diagram showing capacity requirements from 1980 to 2050 for the central fuel storage facility and the final repository for KBS-I.

reprocessing contractor, from which the vitrified reprocessed waste is returned in steel canisters some time later. These canisters are stored in an Intermediate Storage Facility before they are recapped, sealed, and emplaced in the Final Depository. In the no-reprocessing case, the spent fuel elements are kept in the CLAB until it is time for recapping and deposition. In this latter case, the Intermediate Storage Facility is not needed, but all other facilities are common, including the transportation system.

The KBS design is based on a nuclear power program of 12 reactors (mostly boiling-water reactors), which during their lifetimes are assumed to produce 9000 tonnes of spent fuel elements. (A 1 GW nuclear power station discharges annually 20 to 25 tonnes of spent uranium fuel (SUF).) Since each tonne will produce approximately one canister of waste, either solidified reprocessing waste or spent fuel elements, the total program amounts to about 9000 canisters. The first shipments of spent fuel to the Central Storage Facility will begin around 1985, and the maximum handling amount (about 260 tonnes of spent fuel a year) will be reached around 1987. Extra space will temporarily be rented out to Finland.

The waste-handling scheme covers the route from the input material (spent fuel elements) to the final deposition of the waste. Table 1 summarizes the

TABLE 1. KBS selected conditions for final storage of high-active waste.

Waste conditions	Reprocessed high-active waste in glass (KBS-I).
	Unreprocessed spent fuel elements (KBS-II).
Barrier system	Engineered (multilayers of metals).
	Bentonite clay (a natural material).
	Nature (granite bedrock at 500 m depth).
Location	Earthquake-free, low-fault zone.
	Valueless minerals.
	Low groundwater flow.
Physical conditions	Site percolated by groundwater.
	Maximum temperature outside canisters <100^{o}C.
	Very low radiation field.

selected final conditions. These conditions developed during the KBS project and were not prescribed, or later approved, by any authority. The only design goal has been the stipulation by the law that storage be "absolutely safe" (as interpreted above).

It was obvious that the safest storage place in Sweden would be in the granite rock which underlies most of the country. This 600-million-year-old rock body is considered among the most stable in the world. However, the rock is permeated by groundwater even at considerable depth. The design had therefore to take into account corrosion by groundwater, and the possibility that dissolved waste may be transported from the depository to reservoirs of water for human use. The effect of sorption of dissolved waste from groundwater onto rock surfaces was not known at the time a waste-handling scheme had to be decided on. It turns out that sorption reduces the velocity of waste transport, but since this effect could not be taken into account, the chosen design was ultra-conservative:

(i) In order to eliminate radiation effects between waste, canister, groundwater, and rock, which could lead to high corrosion rate, the waste was to be surrounded by a radiation shield (lead or copper).

(ii) In order to slow down the rate of dissolution and other reactions with the environment, the maximum canister surface temperature was not allowed to exceed 100°C. This could be achieved by reducing the waste amount of each canister to about 1 tonne SUF and by having a rather long cooling period to allow radioactive decay to reduce the heat output. Storage for unreprocessed spent fuel would be 40 years in water-filled pools. The reprocessing plan called for 10 years of pool storage, and 30 more years of storage--after reprocessing--in an "intermediate-storage" facility.

(iii) A multilayer barrier system was selected, each barrier having an independent high corrosion resistance, so that if one barrier failed there would be a high probability that the others would not. To meet this purpose the metals lead, titanium and copper were chosen. As a backup, corundum canisters were developed.

(iv) In order to reduce the risk of mechanical damage by geologic faulting, the canisters were to be supported by an elastic "backfill" material (quartz plus bentonite clay).

In later studies, some of these materials have been found even more advantageous than when originally contemplated. In addition, retention of the radionuclides by the rock has been found to be another powerful barrier.

The canister composition, waste content, and dimensions are summarized in Table 2 for the two concepts. Although the amount of radioactive waste is about the same in the two concepts, the reprocessed spent fuel canisters are much lighter, due to the selection of copper instead of lead as radiation shielding. Copper was chosen for the spent uranium fuel because the corrosion resistance was considered higher.

TABLE 2. Some data for the two concepts for storage of high-level waste and spent uranium fuel.

Property	Reprocessed vitrified high-level waste	Unreprocessed spent uranium fuel
Equivalent waste amount[a]	From 1 tonne SUF[b]	1.3 tonne SUF
Central part composition	420 kg borosilicate glass with 9% fission products[c]	550 fuel pins + lead in voids
Central part weight	450 kg, incl. 3 mm stainl. steel cladding	2 tonne fuel + 2.5 tonne lead
Outer canning	100 mm lead + 6 mm titanium	200 mm copper
Canister:		
Dimensions	0.6 × 1.8 m	0.8 × 4.7 m
Weight	3.9 tonne	20 tonne
Surface temp max[d]	65°C	77°C
Surface temp after 1000 yr	35°C	50°C
Bentonite backfill (outer barrier)	0.2 m	0.4 m
Rock hole dimensions	1.0 × 5.0 m	1.5 × 7.7 m

[a] At 33 000 MWd/tonne U burn-up.
[b] 38 kg fission products (FP), <3kg U, <0.05 kg Pu, <1 kg other transuranium elements.
[c] Glass of this composition will be returned to Sweden after reprocessing at the French site at La Hague.
[d] Maximum temperature reached about 15 years after discharge of fuel from the reactor.

The handling timetables are presented in Table 3 and in Fig. 3. All Swedish power reactor waste should be deposited and the depository closed by year 2050, according to these schemes.

2.3 SPENT FUEL STORAGE FACILITY

Regardless of whether the spent fuel is to be reprocessed or disposed of without reprocessing, additional storage capacity away from the reactors is required for the spent fuel. KBS proposes a storage time of 10 years before eventual reprocessing, and of 40 years if the spent fuel is not to be reprocessed. Storage will be in water-filled basins underground, see Fig. 4. The corrosion experience from zircaloy-clad fuel is so good that no additional canning is considered. Storage will be in racks, so that $k_{eff} < 0.95$ even for new unused fuel; thus criticality risks are eliminated. The water will be cooled by sea water through heat exchangers. Pool temperature will rise to 40°C if one heat exchanger breaks down, and to about 85°C if all break down. The facility has an automatic water level control system connected to a surface lake.

The Government has licensed SKBF to construct the Central Fuel Storage Facility (CLAB), to be located close to the Oskarshamn nuclear power plant on the Baltic Sea. The facility is expected to be ready in 1985 with a capacity

TABLE 3. The KBS waste-handling schemes.

Time (yr)	High-level wastes (KBS-I)	Spent fuel elements (KBS-II)
0	Removal from reactor	
1-2	Cooling in reactor pond	
2-10	Cooling in central pool facility	
10	Transport to reprocessing plant Return of high-active waste in glass Storage of cans of high-active waste in glass in air-vented underground vaults	
40	Additional canning in lead and titanium	Canning in lead-filled copper canisters
40	Final deposition at 500 m depth in clay-filled holes in granite bedrock	

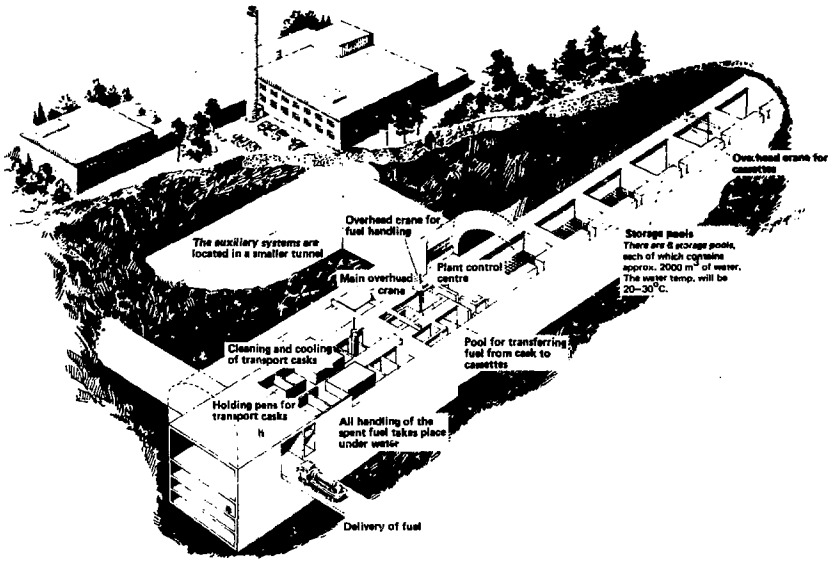


FIG. 4. Perspective drawing of the central storage facility for spent fuel.

of 3000 tonnes of SUF. The facility will then be successively expanded as need arises. See also Sections 2.9 and 3.1.

2.4 SELECTION OF FINAL STORAGE SITE (DEPOSITORY)

Granite rocks belonging to the 600-million-year old Fennoscandian shield underlie most of Sweden. Volcanism has not occurred in the last 125 million years, and strong earthquakes are historically unknown. The maximum magnitude ever recorded is around 5 (Richter scale). Some rather large areas bordering the Baltic sea seem to be completely free from earthquakes (greater than force 2) and are considered to belong to the world's most stable geologic areas.

The final depository, it is suggested, should be located in such an area, and at a depth of $\geq$500 m. It has been shown that dozens of ice ages have had no other effect at this depth than minor increased fracturing. Extensive drilling has been carried out in 4 major areas, and geophysical and hydrological data collected; test drillings, are presently being done at 12

other scattered locations (see Section 3.1). In general, the rock is fractured, with several fractures per drilled meter, but the fractures are small (mm-size and smaller) and filled with chlorite or other clayish material. The distance between crush zones surrounding such fractured granite rock bodies is of the order of 100 meters. Particularly intensive studies have been carried out at Stripa, see Section 3.2.

Groundwater flows through the fracture system. Because the groundwater table falls very slowly towards the sea level along the Swedish coast, very small hydraulic gradients (water fall per distance) are encountered. This contributes to a slow velocity of the groundwater flow. Groundwater has been analyzed with regard to pH, Eh (electrochemical, or redox potential), chemical composition (over 20 constituents; see Table 4), age, flow rate (Table 5), etc. Thus, a very detailed knowledge of groundwater conditions has been acquired. This knowledge is necessary for determining the two essential parameters: corrosion, which may release radioactivity into the groundwater, and water flow, which may transport the radionuclides to water used by man.

Although the hydraulic conductivity, which determines flow rate (together with the hydraulic gradient and rock porosity) varies considerably along investigated bore holes, the value of $\sim 10^{-9}$ m/s was selected by the KBS as typical and acceptable for a granite rock depository. Large scale measurements indicate even lower values, and consequently lower water flow rates. With "typical values" for the hydraulic parameters, the groundwater velocity becomes 0.1 m/yr, and groundwater flow 0.2 liter$\cdot$yr$^{-1}\cdot$m^{-2}. These values are used in the safety analysis, Section 2.7.

The chemical composition of Swedish groundwater (Table 4) does not vary very much and is usually of drinking quality. It does not contain any particularly corrosive ions. The pH is close to neutral. However, there is one particular property which merits special attention, and that is the water redox potential. Due to the very small amount of oxygen (O_2), and considerably larger amounts of divalent iron (Fe^{2+}), the redox potential turns out to be negative, see Table 6. The effect of this is twofold: (1) The water will not be able to oxidize metals (used as canister material), and (2) most dissolved waste products will be in a low valency state. This dramatically increases canister lifetime and delays waste product migration to repositories of water used by man. This is further discussed in Section 2.7.

As described in Sections 2.4 and 3.2, an extensive exploration is going on for finding and selecting suitable sites for the depository. Many places

TABLE 4. Probable composition of groundwater in crystalline rock at great depth, according to Rennerfelt and Jacks.

Analysis	Probable range	Minimum[a] value	Maximum[a] value
Conductivity	400-600 µS/cm		1100
pH	7.2-8.5		9.0
$KMnO_4$ consumption	20-40 mg/liter		50
COD_{Mn}	5-10 mg/liter		12.5
Ca^{2+}	25-50 mg/liter	10	60
Mg^{2+}	5-20 mg/liter		30
Na^{+}	10-100 mg/liter		100
K^{+}	1-5 mg/liter		10
Fe-tot	1-20 mg/liter		30
Fe^{2+}	0.5-15 mg/liter		30
Mn^{2+}	0.1-0.5 mg/liter		3
HCO_3^{-}	60-400 mg/liter		500
CO_2	0-25 mg/liter		35
Cl^{-}	5-50 mg/liter		100
SO_4^{2-}	1-15 mg/liter		50
NO_3^{-}	0.1-0.5 mg/liter		2
PO_4^{3-}	0.01-0.1 mg/liter		0.5
F^{-}	0.5-2 mg/liter		8
SiO_2	5-30 mg/liter		40
HS^{-}	<0.1-1 mg/liter		5
NH_4	0.1-0.4 mg/liter		2
NO_2	<0.01-0.1 mg/liter		0.5
O_2	<0.01-0.07 mg/liter		0.1

[a]The estimated probability that a value will fall between the minimum value and maximum value is 95%. Higher values occur locally.

TABLE 5. Hydrological conditions at 500 m depth in Swedish granite bedrock.

Average age of groundwater	3500 ± 1000 yr
Average velocity of groundwater	0.1 m/yr
Groundwater flow:	
Average	0.2 liter·yr^{-1}·m^{-2}
Karlshamn	0.002 liter·yr^{-1}·m^{-2}
Hydraulic conductivity:	
Common range	10^{-7} -10^{-11} m/s
Average	10^{-9} m/s
Karlshamn	2×10^{-12} m/s
Stripa mine	2×10^{-11} m/s

TABLE 6. The redox potential in deep groundwater from Precambrian Swedish bedrock, according to Grenthe.

Sample No.	Site	Eh (mV)
7	Stripa	-152
8	Stripa	-31
18	Stripa	-173
19	Finnsjö area	-191
21	Stripa	-210
14	Stripa	-26
20	Finnsjö area	-157
24	Stripa	-140

will be investigated, to provide a number of site options, because it is believed that only few places will not be met by public opposition.

2.5 DISPOSAL OF UNREPROCESSED SPENT FUEL ELEMENTS

The various steps of the disposition of unreprocessed spent fuel elements are shown diagrammatically in Fig 5.

After 40 years in the Central Fuel Storage Facility (CLAB), the fuel is transported to an encapsulation station. The method of transportation is identical to that used to transfer the fuel from the power station to the CLAB.

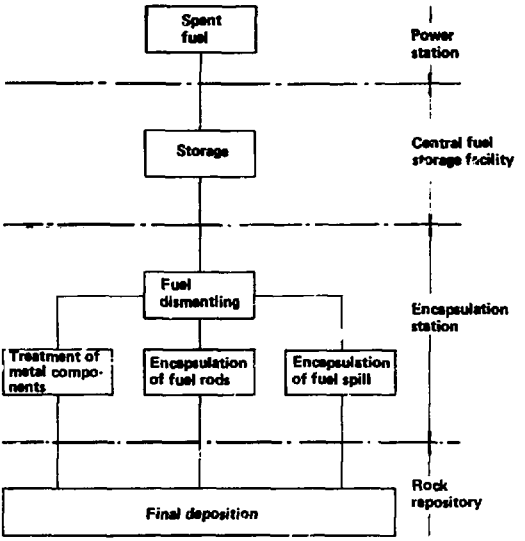


FIG. 5. Flow scheme illustrating the handling chain for the spent fuel from the power station to the final repository.

The encapsulation station, which is shown in successively increasing detail in Figs. 6, 7, and 8, is situated above ground, connected with the final depository. In the encapsulation station the fuel is dismantled, separating the fuel rods from metal components of the assemblies. The fuel rods are enclosed in copper canisters with walls 200 mm thick. (See Figs. 9 and 10.) Each container weighs about 16 tonnes and can hold 500-640 rods, depending on the type of fuel. The space between the fuel rods is filled with lead. Fuel spill from handling of the fuel assemblies is encapsulated in a similar manner.

The copper canisters are transferred to a final depository approximately 500 m down in the bedrock. It is proposed to design the depository as a system of tunnels 25 m apart (see Fig. 11), with storage holes drilled in the floor. The holes will be 6 m apart, 7.7 m deep, and 1.5 m in diameter, (see Figs. 11-15). The canisters will be placed on and surrounded by precompacted bentonite. This material will swell during water uptake, filling fractures in the surrounding rock. To contain the swelling pressures the

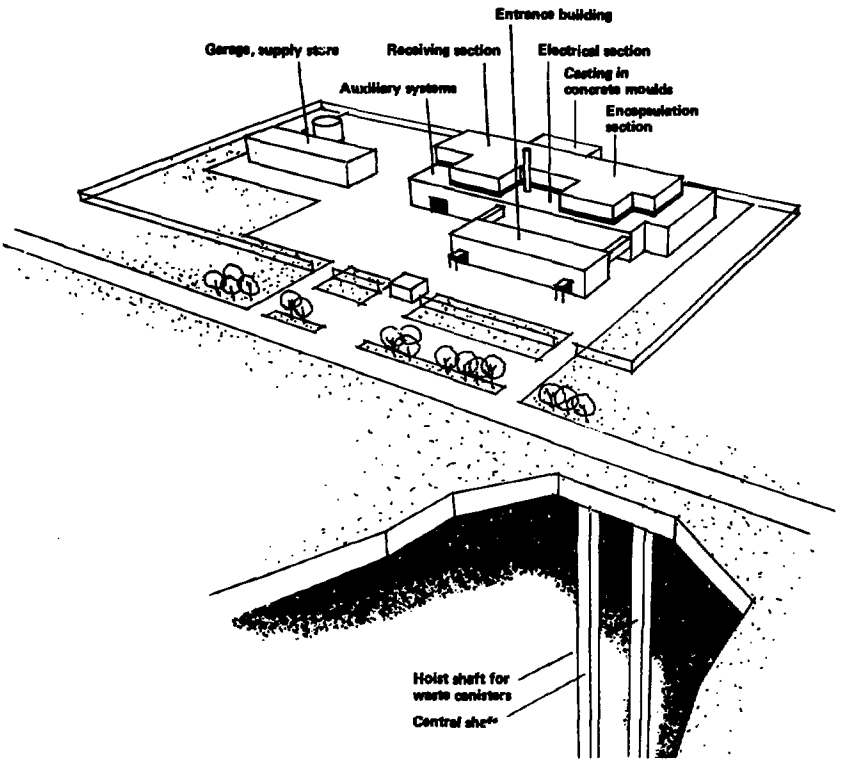


FIG. 6. Encapsulation station for spent nuclear fuel. The facility is located at ground level above the final repository.

holes will be plugged at the top with a concrete bar. The bentonite also acts as an elastic support, allowing considerable movement of the surrounding rock without damaging the canister. The bentonite also has a very low water permeability, 10^{-12}–10^{-14} $m \cdot s^{-1}$. Further, the bentonite has ion-exchange properties, leading to a retention of ions eventually migrating through the bentonite. After the canisters have been deposited, and the holes filled with bentonite, the tunnels will be filled with a bentonite-quartz mixture (Figs. 14 and 15). Since the dose rate of the canister surface at the time of deposition is $\lesssim$1 rem/h (Fig. 16), the deposition operation can be carried out without excessive remote-handling equipment.

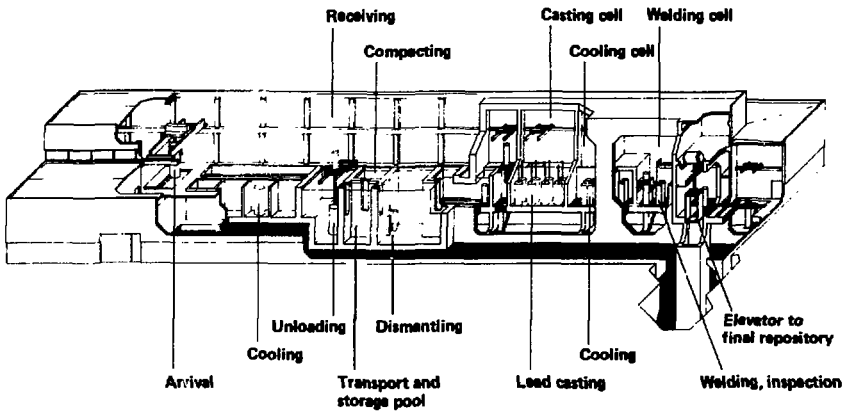


FIG. 7. Perspective drawing of encapsulation station's process building.

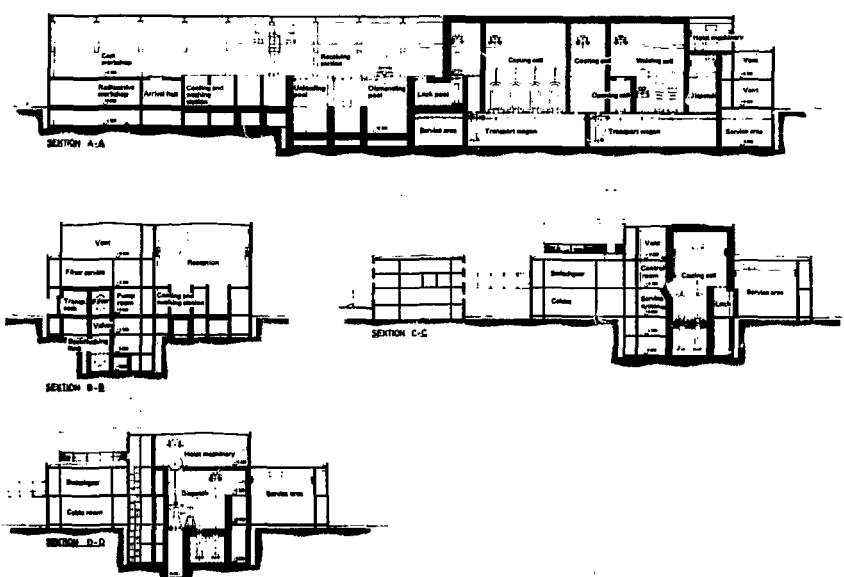


FIG. 8. Building layout of encapsulation station.

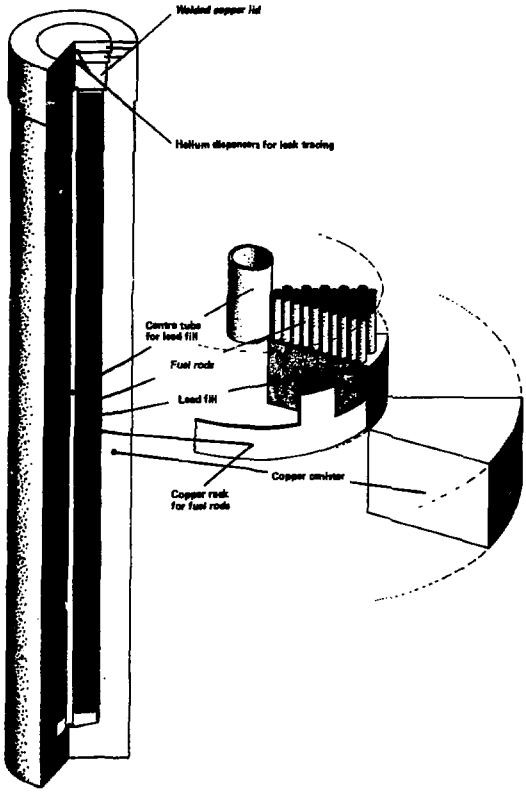


FIG. 9. Cutaway drawing of copper canister filled with fuel rods. After dismantling of the fuel assemblies, the fuel rods are placed in a cylindrical rack whose perforated copper shell encloses the fuel rods. The filled rack is covered with a perforated lid and lowered into the canister. The center tube is used for lifting and lead casting.

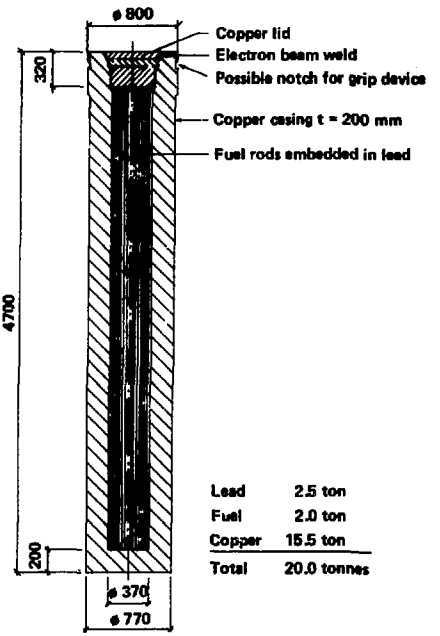


FIG. 10. Longitudinal section of copper canister with fuel rods embedded in lead (dimensions in mm).

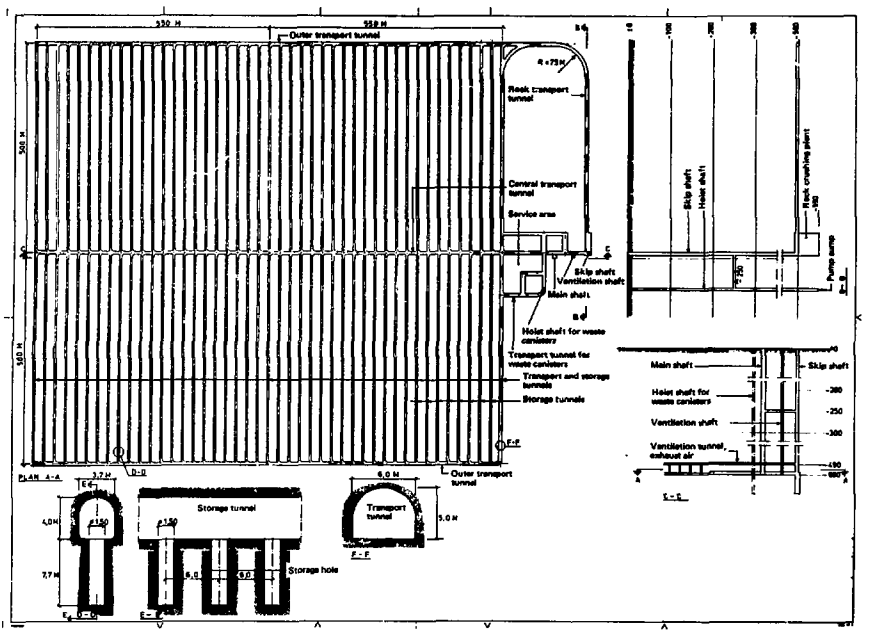


FIG. 11. Layout of final repository.

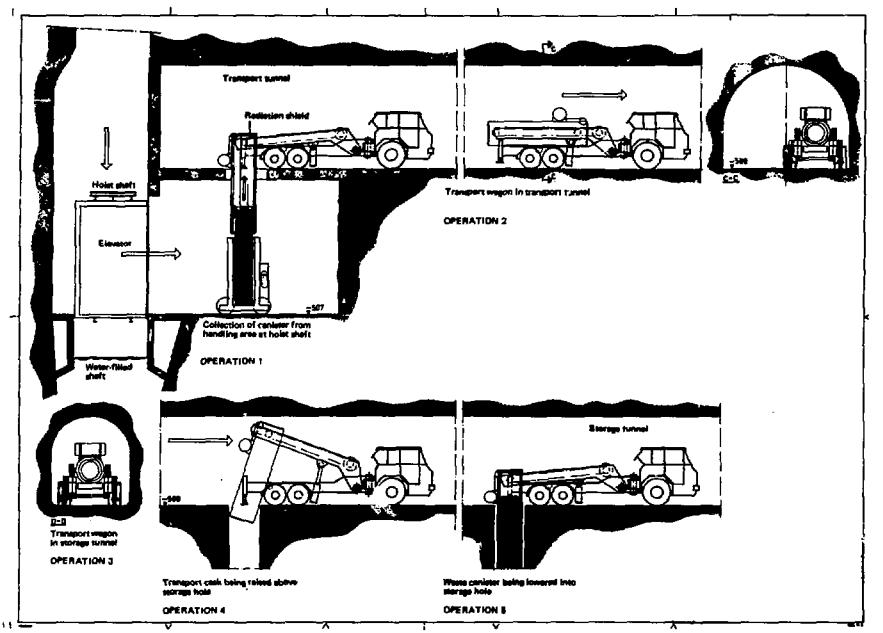


FIG. 12. Transport and deposition of waste canister in final repository.

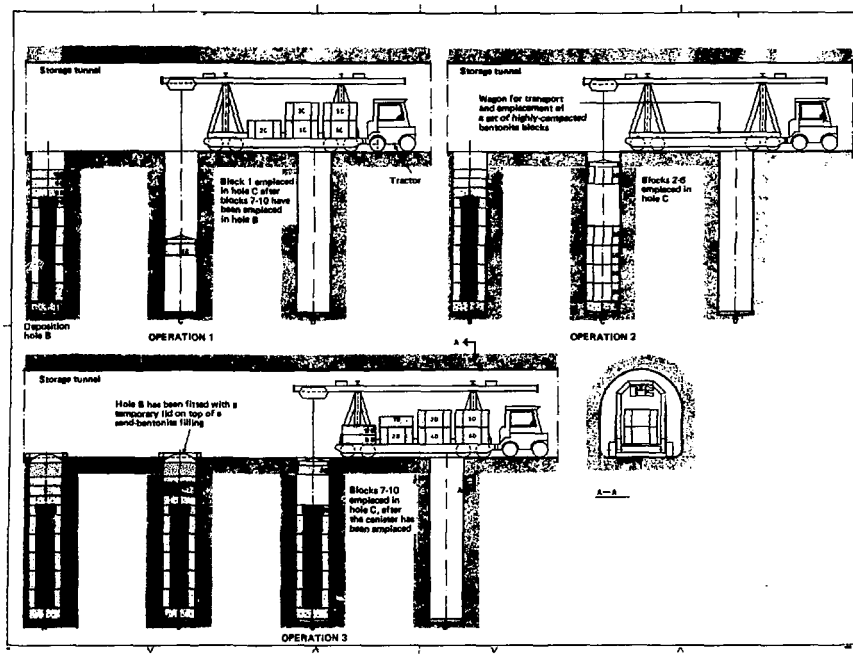


FIG. 13. Emplacement of bentonite blocks in storage holes.

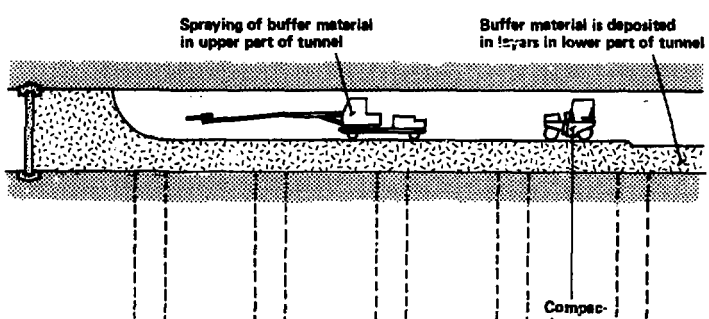


FIG. 14. When the final repository is sealed, the tunnels are filled with a mixture of quartz-sand and bentonite. The lower part of the fill is deposited by tractors and vibrorolled. The upper part of the tunnel is filled by spraying.

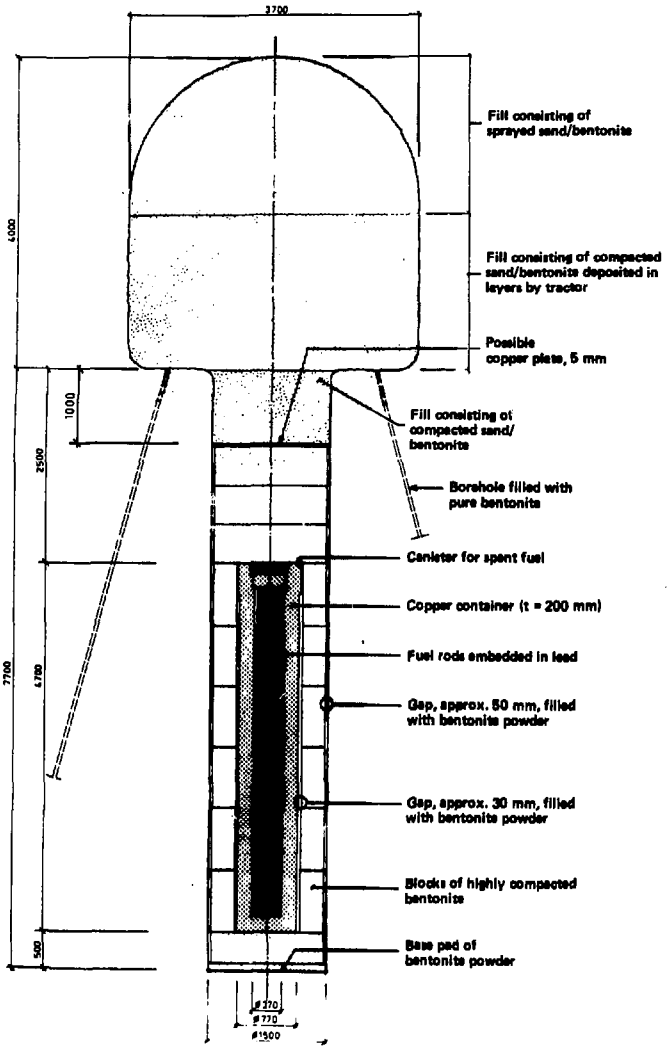


FIG. 15. The sealed final repository. The canister is surrounded in the storage hole by highly compacted bentonite. The gaps are filled with bentonite powder. The tunnel is filled with a mixture of quartz-sand and bentonite. A copper plate can, if desired, be placed on top of the bentonite block to serve as a diffusion barrier. Dimensions are in mm.

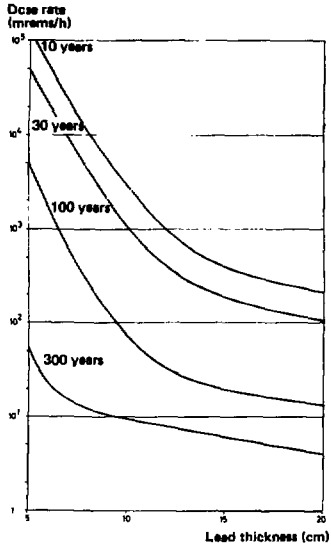


FIG. 16a. Total dose rate from gamma radiation and neutrons on the outside of encapsulated vitrified waste from pressurized waste reactors at different lead thicknesses.

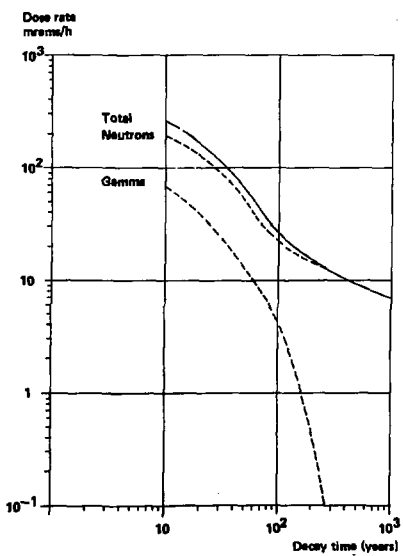


FIG 16b. Radiation dose rates on surface of copper canister after different decay times.

2.6 HANDLING OF REPROCESSING WASTE

The various steps of the reprocessing cycle are shown in Fig. 3a. After about 10 years of storage in the CLAB, the spent fuel elements are transported to a reprocessing plant abroad (the Swedish Fuel Supply Company has contracts with the French Company Cogema and British Nuclear Fuels Ltd.), where uranium and plutonium are removed and the high-active waste is vitrified and returned to Sweden. Each canister of vitrified waste will contain about 9% fission products and about 1% of actinides in 420 kg borosilicate glass of the French AVM type in a stainless steel (3 mm) canister. Each canister will contain the high-active waste from reprocessing 1 tonne spent fuel (Table 2).

The canisters will be returned to Sweden and stored in an Intermediate Storage Facility (ISF) located underground directly connected with the final depository. See Figs. 17 and 18. Because of French experience, these canisters will not be water-cooled but air-cooled. Cooling redundancy is achieved by having four cooling fans, two at the storage vaults, one in reserve nearby, and one in reserve at the ground level. The motors are powered from an electricity grid and have emergency diesel generators as well. Figures 19 and 20 show various sections of the ISF and its ventilation system. If all fans fail, natural convection is sufficient to prevent destruction of the canisters. The heat power of a canister is about 1 kW at 10 yr, and about 0.5 kW at 40 yr.

About 40 years after reactor discharge, the canisters with the vitrified waste are transferred to the encapsulation station (Fig. 18) and surrounded by 10 cm of lead and an outer container of 6 mm titanium. Lead was chosen because of its high radiation-shielding capacity and titanium because of its corrosion resistance, even in salt waters. Figure 21 shows the recapped vitrified waste canister in the final depository. Its weight is about 4 tonne (Table 2) and the radiation level on its surface is slightly below 1 rem/hr (Fig. 16a).

The recapped canisters are transferred to the final depository about 500 m below the ISF in the granite bedrock. The final depository is designed as for the spent fuel elements, though the holes drilled in the tunnel floors are smaller (diameter 1 m, depth 5 m) and closer (4 m apart). The deposition and filling technique is illustrated in Fig. 22. The vitrified waste canisters are surrounded by a bentonite buffer.

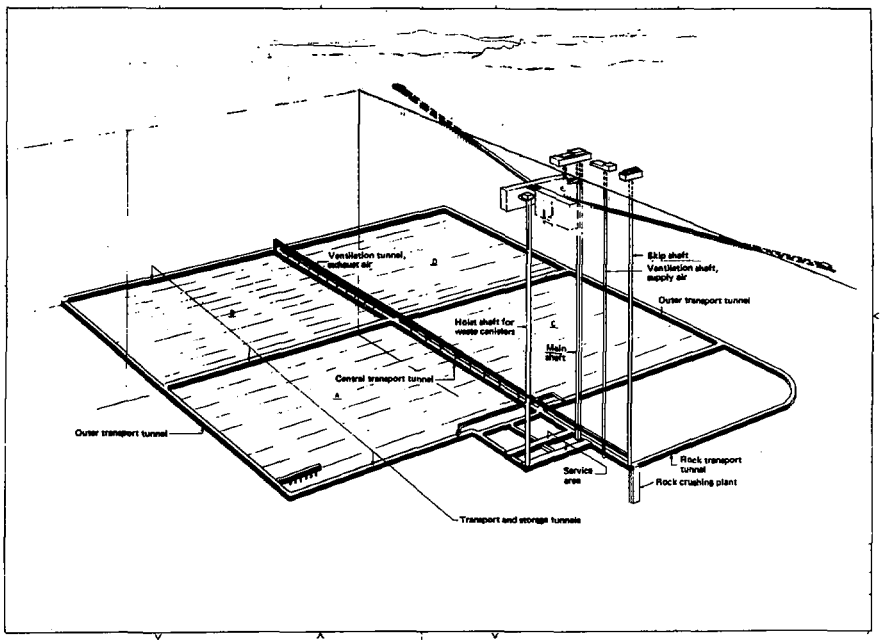


FIG. 17. Perspective drawing of final repository.

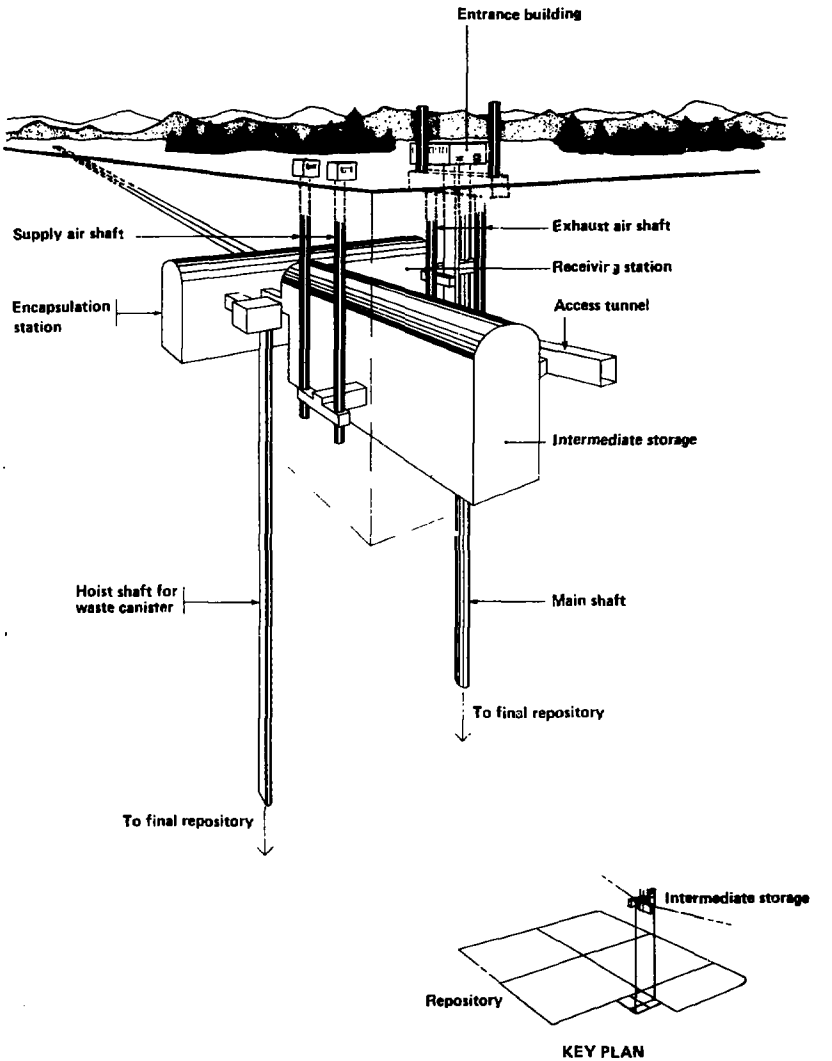


FIG. 18. Perspective drawing of plant for intermediate storage and encapsulation. It is located underground with a rock cover approximately 30 m thick. The plant is located above the final repository.

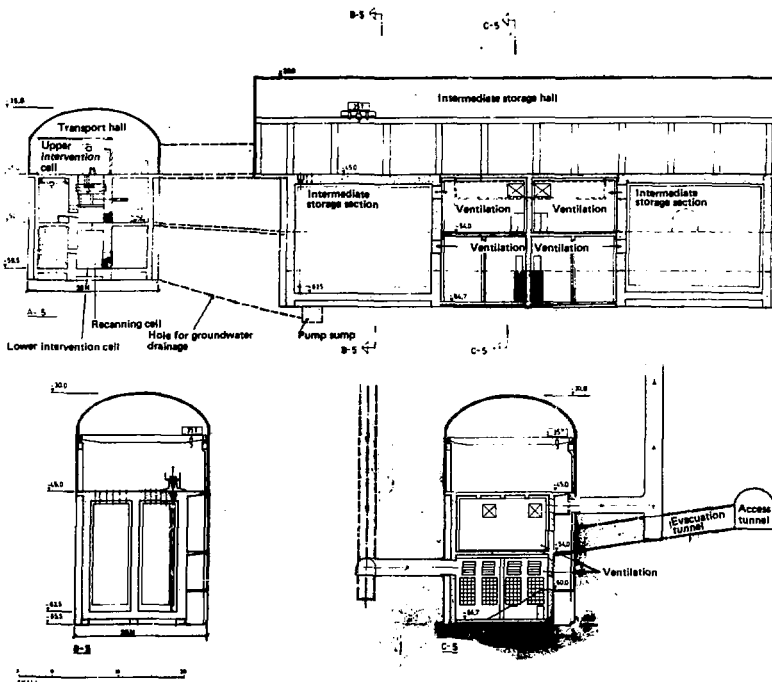


FIG. 19. Longitudinal section and cross sections of intermediate storage and encapsulation plant (cf. Fig. 18).

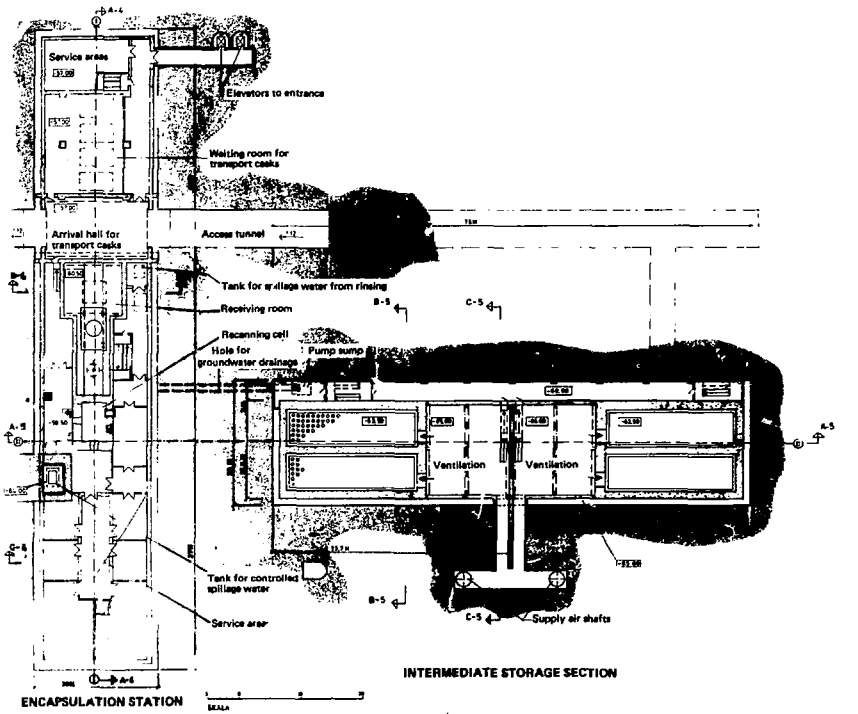


FIG. 20. Intermediate storage and encapsulation plant.

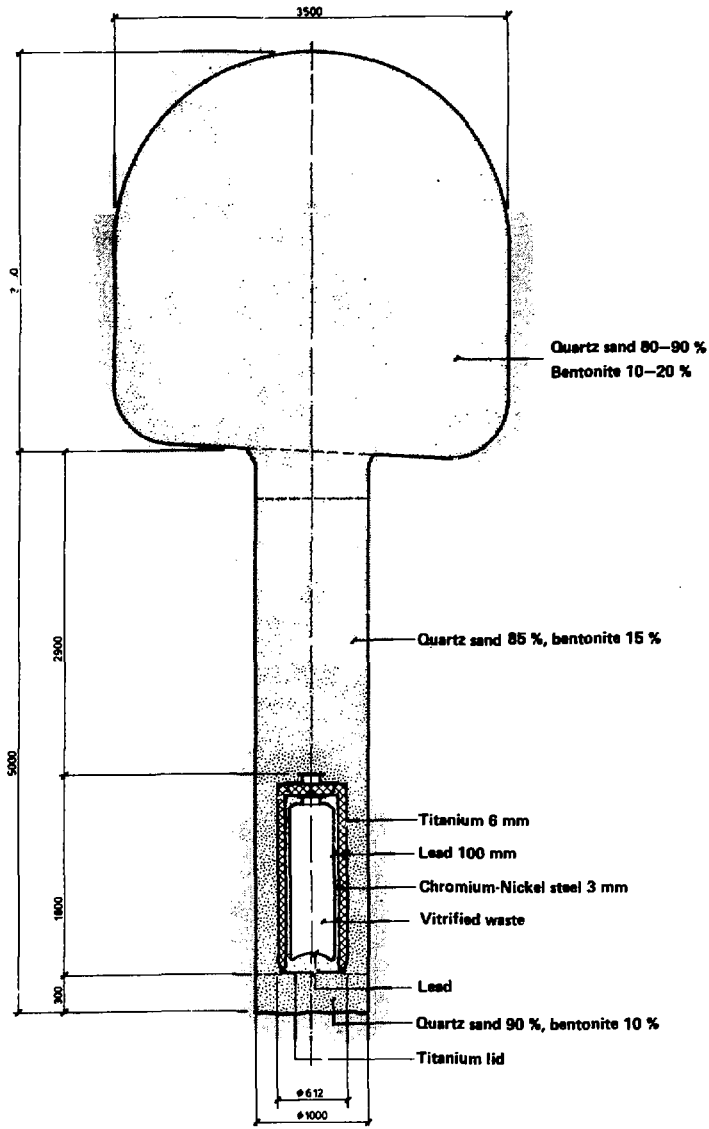


FIG. 21. Sealed final repository. Dimensions are in mm.

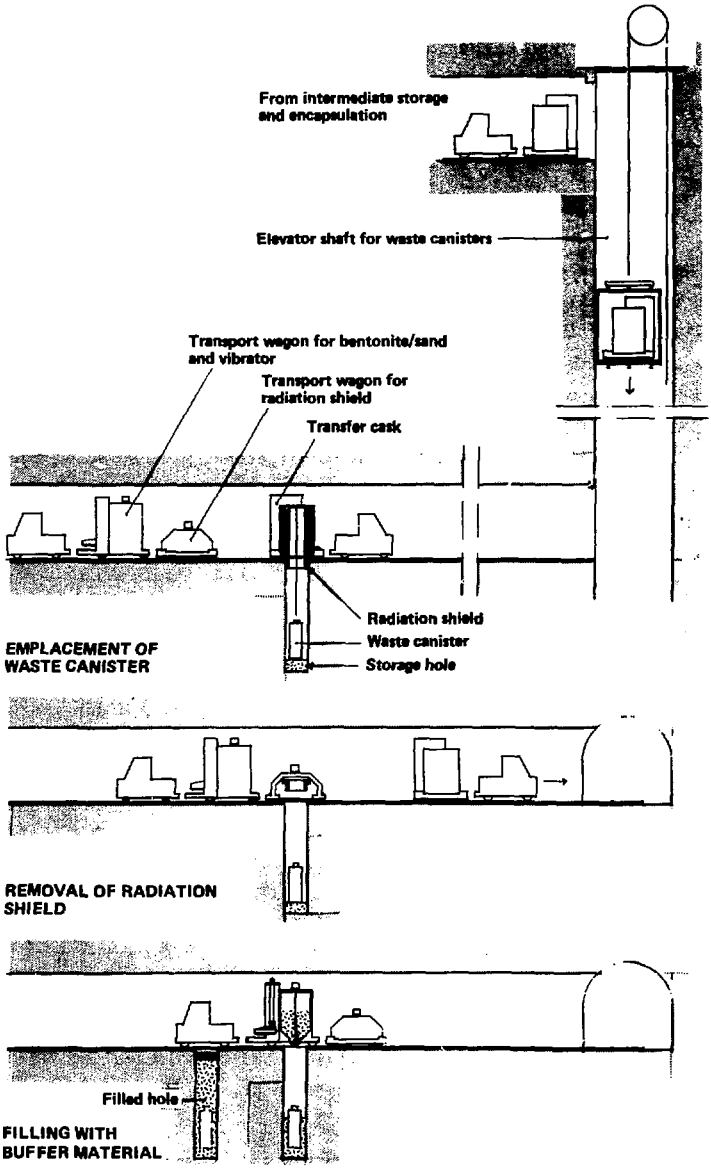


FIG. 22. Handling of waste canisters in final repository.

Safety in the handling, transportation, and intermediate storage has been investigated through failure analysis. The data are too detailed to be reviewed here, but as a general conclusion one can say that no area of excessive risk has been identified.

2.7 SAFETY ANALYSIS OF THE FINAL DEPOSITORY

Although a great number of risks are encountered in the handling chain from the removal of the spent fuel elements from the reactor until the waste is disposed of in the final depository, these risks are not very different from those which man encounters in the daily work with radioactive materials. Also the risks are encountered over a short time, and normally can be rapidly corrected for, covering altogether a human "lifetime" of 50-70 years. The risk from the final depository is quite different: it "appears" only after the depository is closed, when the possibilities for discovery are small and for correction almost nil,* and it will last for "millenia." Therefore, the risk from the waste in the closed depository must be made much smaller than the risks in the preceding handling chain.

In order to meet this requirement, the KBS relies on a number of barriers, of which three are man-made and one is natural, each being in principle independent of the others:

1. A poorly dissolvable matrix (UO_2 or glass) containing the waste.
2. A number of corrosion-resistant metallic barriers (lead, copper, titanium).
3. An elastic clay buffer, which also acts as a water and radionuclide barrier.
4. Rock retention, slowing down the radionuclide movement.

The long-term stability and reliability of each of these barriers have been analyzed by KBS, and are summarized below.

2.7.1 Geology and Hydrology

The KBS concludes that the geological and hydrological conditions are stable and will remain stable for a Swedish granite waste repository for

*However, keeping the waste depository tunnels open and using groundwater radiation detectors, would allow for corrective measures if needed.

millenia (Section 2.4). Most of the critics of the KBS project have addressed this point. Though a majority of the geologists reviewing the KBS project support the KBS conclusion, some geologists have commented that "geology is a descriptive science, not a predictive." While the long-term stability of the depository area is a matter of belief in extrapolations from past stability data, the continuing hydrogeologic studies indicate that groundwater moves more slowly than is assumed for KBS main case (hydraulic conductivities close to 10^{-11} m/s rather than the value 10^{-9} m/s assumed by KBS). This adds an extra safety margin to the hydrogeological assumptions. However, as some critics have pointed out, determination of groundwater age is extremely complicated, and the results are therefore subject to doubt. Ongoing research in this field is of utmost importance for the safety analysis, and for public trust.

2.7.2 Radiation Effects

Experiments with radiation doses higher than will be encountered in reality show that the canister materials will be unaffected by the radiation from the radioactive waste nuclides. The heavy radiation shielding (copper and lead) reduces the radiation doses at the canister surface to such low levels that radiolysis of water and radiation effects on the clay become negligible. Experiments and careful calculations show that no radiolysis gases (H_2 and O_2) will appear.

2.7.3 Heat Effect

The small amount of waste in each canister, the long cooling times, the large canister surface areas, and the space distribution of canisters in the depository (producing <5 W/m^2 floor area) all contribute to the low canister surface temperature (maximum 75°C) and mean rock temperature (maximum 60°C) shown in Fig. 23. The moderate surface temperatures guarantee the stability of the clay buffer and chemical interactions between groundwater and canister materials are not greatly accelerated at these temperatures. Stripa experiments show that the temperature increase does not lead to excessive fracturing but lowers the hydraulic conductivity.

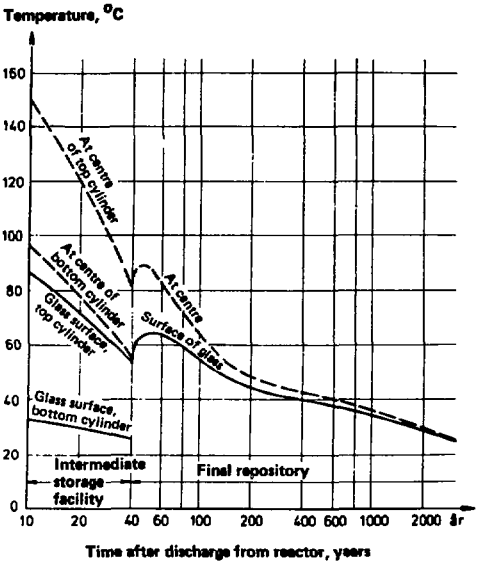


FIG. 23a. Temperature of the vitrified waste in the intermediate storage facility and the final repository.

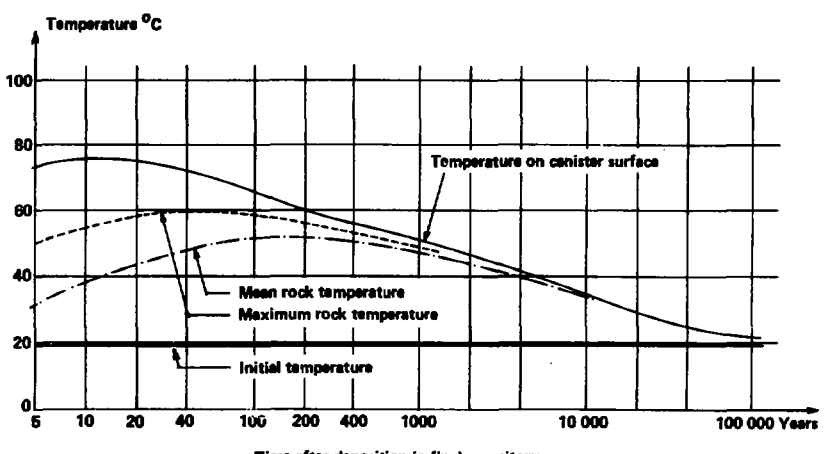


FIG. 23b. Temperatures on canister surface and in rock at different points in time after deposition in final repository.

2.7.4 Dissolution Rates

Table 7 summarizes the dissolution times calculated by KBS for the canisters from the two waste chains. The calculations are based on experimental corrosion rates at relevant temperatures, for water which is as similar as possible to typical groundwater. The dissolution time for the glass is a particularly controversial issue, because the rate of dissolution strongly depends on glass composition, groundwater composition, and temperature. Also, dissolution of most glasses becomes much slower as time passes. In an experiment, the leaching rate of strontium (^{90}Sr) from a Canadian glass went from $\sim 10^{-7}$ down to 10^{-10} $g \cdot cm^{-2} \cdot d^{-1}$ in about 10 years. (The first measurement was in the laboratory and the second was in a field test.) The KBS uses a value of 2×10^{-7} $g \cdot cm^{-2} \cdot d^{-1}$, which would dissolve the glass in 3000 years. Experience indicates that this is a highly conservative value.

In order to dissolve a metal, there must be an oxidant as well as a sufficient amount of water; also, complexations can increase solubility. All these factors have been taken into account: it is observed both that the amount of oxidants (mainly O_2) in deep groundwater is small (see also next section) and that the amount of groundwater itself is small (about 2.3 m^3/yr for the copper canister hole), leading to very long dissolution times for the metals copper and lead. (For groundwater containing 0.1 ppm O_2 (see

TABLE 7. Calculated canister dissolution times (yr).

KBS-I:		
Titanium (6 mm)	$>10^4$	pinhole >500
Lead (100 mm)	$>10^6$	
Borosilicate glass	$3 \times 10^4 - 3 \times 10^6$	
KBS-II:		
Copper (200 mm)	$>10^6$	
Lead (interstitial)	not accounted for	
Zircaloy (~1 mm)	not accounted for	
Uranium dioxide	$\sim 10^6$ (carbonate complex)	
Alumina (corundum)	$>10^4$	

Table 4), and because 1 g O_2 can oxidize only 4 g Cu, and the fact that a canister contains 15 tonnes Cu, the dissolution time would be $15 \times 10^9/(0.1 \times 4 \times 2{,}3 \times 10^3)$ = 16 million years, if each oxidation leads to dissolution.)

A criticism of the KBS dissolution times is that the chemistry is more complex in practice than KBS assumes from simple idealized experiments. For example, special types of corrosion can lead to earlier penetration of barriers. On the other hand, studies of old archeological objects and volcanic and meteoritic glasses indicate that corrosion rates could be slower than KBS assumes. Since dissolution rates depend on groundwater composition and flow rate, they will be highly site-specific.

2.7.5 Radionuclide Transport Through the Ground

Once dissolved, radionuclides will move with the groundwater velocity, if they do not sorb on the rock fractures carrying the groundwater. Experiments initiated by KBS and carried out elsewhere show that most radionuclides sorb on granite rocks. The sorption depends on the chemistry of the nuclide and of the groundwater. The effect of the sorption is a retardation of the radionuclide, expressed through the retention factor. For a retention factor of 1, the radionuclide moves with the same velocity as the groundwater; for a retention factor of 1000, the radionuclide moves 1000 times more slowly than the groundwater.

Many of the waste products can exist in different valency states in water. Different valency states have different retention factors. The redox potential of the groundwater determines the valency state of the dissolved waste product. To illustrate this effect, KBS quotes a Russian study (see Fig. 24). It shows groundwater percolating through a sandstone layer. Close to the surface, the groundwater carries uranium in its oxidized state (hexavalent uranium) in the form of a highly soluble carbonate complex with very poor retention. In Zone II, containing larger amounts of organic carbon, uranium is reduced to its tetravalent state, which strongly sorbs in this layer. Beyond Zone II, the groundwater, still having a reducing (negative) redox potential (Eh), is quite depleted of uranium. Many uranium ores have been formed in this way.

Deep rock groundwater in Sweden is commonly reducing (Table 6). This recently discovered condition was not included in the KBS-I report, but was considered in the later KBS-II report. Table 8 gives retention values, based

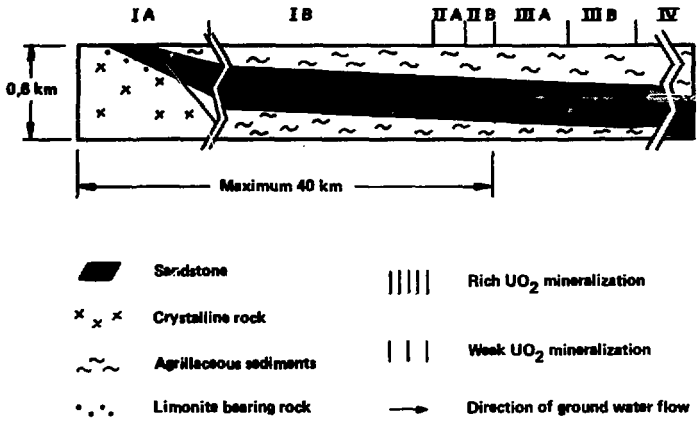


FIG. 24. Schematic illustration of the geological situation and the different zones around a uranium ore exposed to groundwater.

TABLE 8. Best estimate of retention factors of various elements for reducing environment and slow groundwater transport.

Element	Retention factor	Element	Retention factor
Strontium, Sr	1 500	Radium, Ra	48 000
Technetium, Tc	950	Uranium, U	23 000
Iodine, I	1	Neptunium, Np	23 000
Cesium, Cs	4 000	Plutonium, Pu	5 700
Europium, Eu	200 000	Americium, Am	610 000

on laboratory measurements, for reducing groundwater. As is seen, only iodine moves with the groundwater velocity. Had the groundwater not been reducing, the retention factor of technetium (Tc) would also have been 1, and the factors for uranium and neptunium would have been much smaller.

The importance of the retention factor may be illustrated by an example: If the water transport time is 500 years (from waste depository to water reservoir used by man), it will take plutonium 500 x 5700 years = 2.8 million years to reach man. However, since the half-life of the longest lived

plutonium isotope is 24 000 years, 2.8 million years correspond to about 100 plutonium half-lives. Therefore, while plutonium slowly moves through the ground, it will decay radioactively, and in practice no plutonium will ever reach the water reservoir. The retention factors in Table 8 are used for the final calculation of the amounts of radionuclides reaching man.

Retention factors have also been calculated for the clay barrier. It turns out that the two most hazardous fission products, cesium and strontium, are so strongly held by the clay that they decay completely before they can penetrate the barrier.

For similar minerals and groundwaters, the retention factors observed by the KBS investigators are fairly close to those which other investigators have found *in situ* and through laboratory experiments.

2.7.6 *Dose Calculations*

From the amount of groundwater seeping through the depository one can calculate the amount of radionuclides dissolved at a certain time under equilibrium conditions. Knowing groundwater movement and retention factors, the amount of radionuclides reaching a water reservoir can be estimated. The various reservoirs of the ecosystem are described by Fig. 25. For each reservoir, radionuclide uptakes by plants and animals have been calculated, using known radioecological "transfer factors." Finally, the radionuclide intake by man from food and water has been estimated. Three main cases have been studied: (1) The radioactive water flows into a valley in which there is a well near a lake (the "critical group" lives here). (2) The radioactive water flow is distributed equally between a nearby lake and a lake system downstream. (3) The radioactive water flows into the Baltic Sea. Knowing the radionuclide intake, the dose received by exposed persons can be calculated from known "dose factors."

The calculation scheme is summarized in Fig. 26. Three different computer programs have been used. KBS claims that their "main case" is very conservative, i.e., the values chosen are all on the higher-risk side compared to the most probable values. Some essential parameters for the main case and most pessimistic case for the spent fuel alternatives are given below:

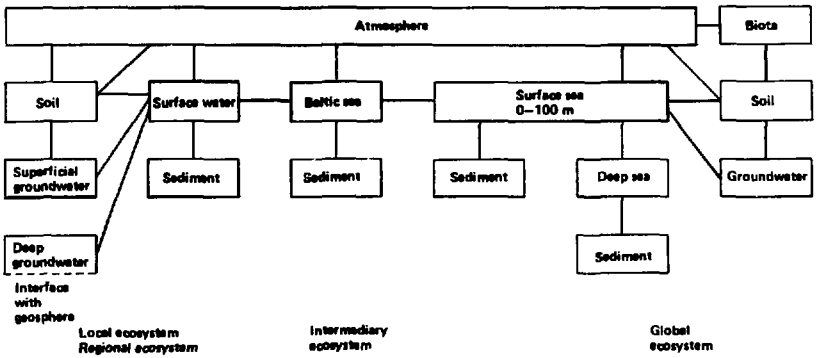


FIG. 25a. Reservoirs for the various ecosystems.

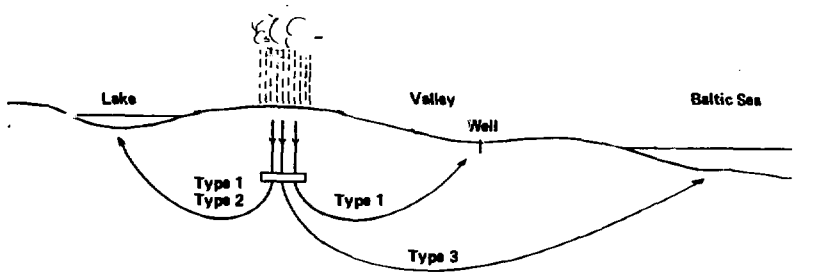


FIG. 25b. The three main paths of transport of radioactive substances to the biosphere.

Case	Canister penetration time	Water transit time	Retention type
Main	100 000 years	3 000 years	Reducing
Pessimistic	100 000 years	400 years	Oxidizing

Some reviewers of the KBS project have criticized the canister penetration (or dissolution) time, which largely relies on the low oxidizing power of the groundwater.

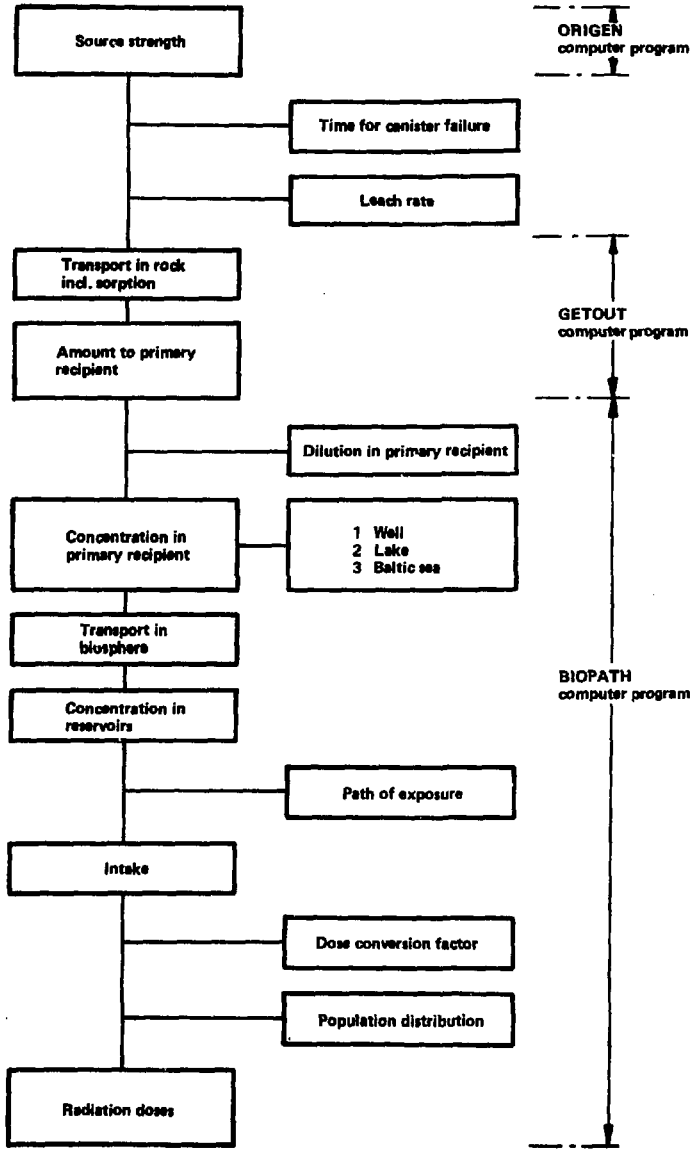


FIG. 26. Scheme of calculation of radiation doses from the radioactive elements which might be released from a final repository for spent fuel.

The doses received by critical groups are presented in Fig. 27a for the main and the pessimistic case. The first peak, arriving after about 100 000 years, is due to long-lived ^{129}I (half-life 1.6×10^7 yr), the second peak is due to long-lived ^{135}Cs (half-life 3×10^6 yr), and the third peak at 100 million years is due to uranium and daughter products. Figure 27b shows the radionuclide sources of the calculated doses received according to Fig. 27a.

Similar calculations have been made for the dissolution, transport, and uptake by man of radionuclides released from a storage of vitrified high-active waste from reprocessing. Figure 28 gives such a calculation based on the following data: the canister is penetrated after 1000 years, glass dissolves in 30 000 years, water transit time is 400 years in rock, but including clay buffer it is 1500 years, and retention factors for oxidizing groundwater conditions. Figure 28a gives the maximum individual doses for this case and Fig. 28b shows the radionuclide sources of the doses shown in Fig. 28a.

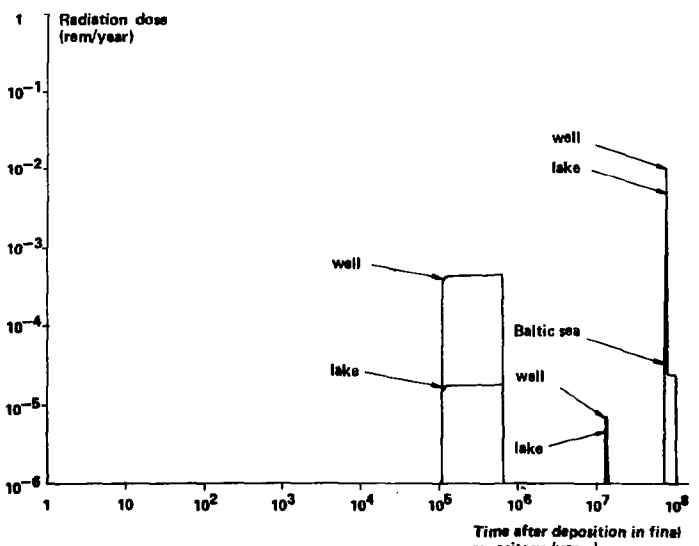


FIG. 27a. Calculated individual doses to critical groups (nearby residents) for different primary recipients. Main case.

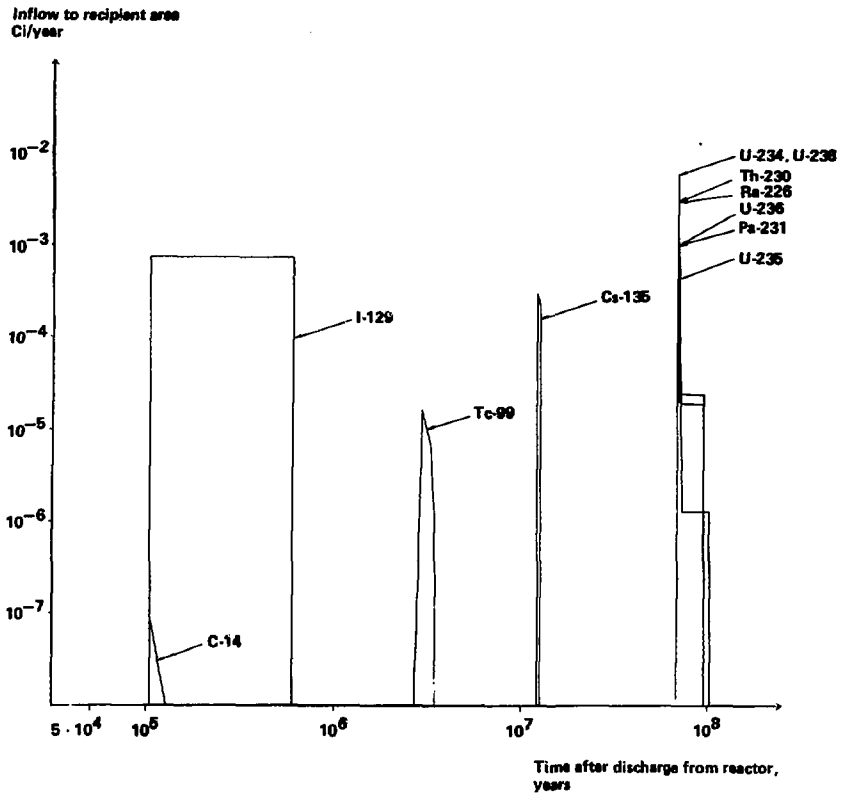


FIG. 27b. Results of GETOUT calculations. Inflow of radioactive elements to the recipient area in the main case.

Figures 29 and 30 summarize the doses expected to be received in the main and the most pessimistic case from released waste radioactivities for storage of spent fuel elements and of vitrified reprocessing waste. In all cases the doses are below that of natural radiation and the ICRP recommended limits.

2.8 REVIEWS OF THE KBS PROJECT

Both KBS projects have been reviewed by experts. KBS-I was reviewed by many national and international bodies, and KBS-II by 22 national and 12

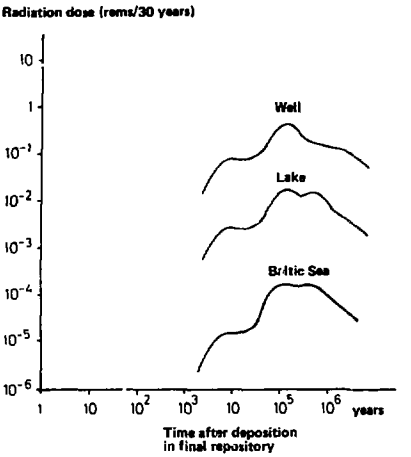


FIG. 28a. Maximum calculated individual doses to critical group (nearby residents) for different primary recipients. The calculations assume a glass leaching period of 30 000 yr.

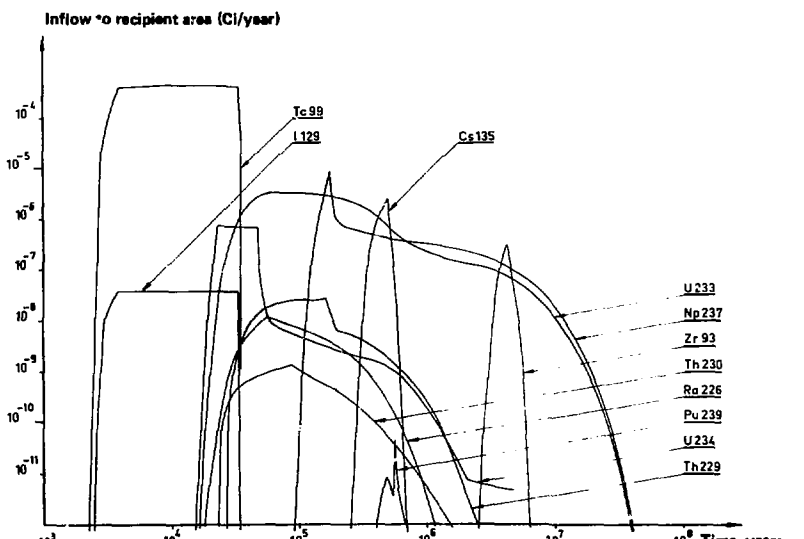


FIG. 28b. Example of calculation of inflow to recipient area at various points in time, carried out using GETOUT computer program. The dissolution time for the glass is 30 000 yr.

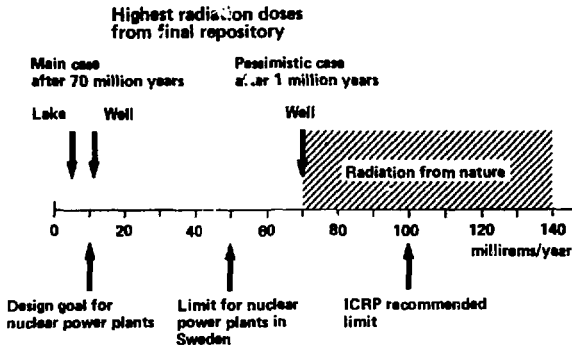


FIG. 29. The calculated maximum radiation doses from the final storage of spent fuel to an individual compared with some limit values. ICRLP is the International Commission on Radiological Protection. Radiation from nature varies from one place to another in Sweden and lies within the hatched area.

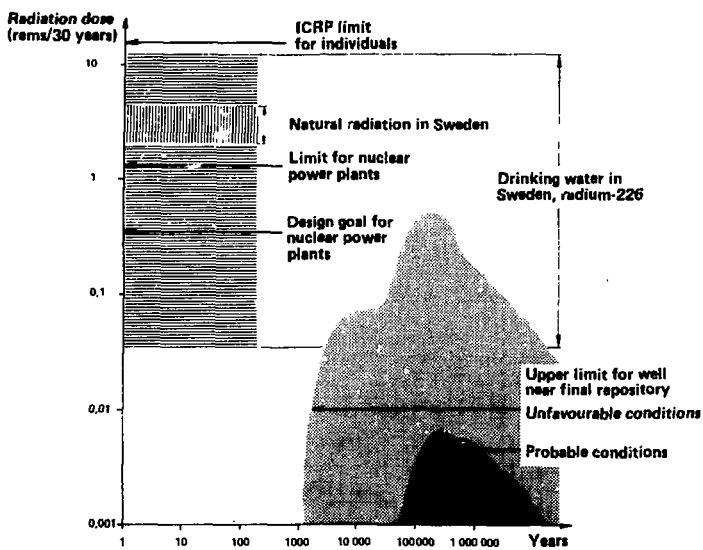


FIG. 30. Calculated upper limit for radiation doses to people who live near the final repository (critical group). The calculations pertain to the slow decomposition of the canister with a well as the primary recipient. For purposes of comparison, the dose load from several natural radiation sources as well as a number of established dose limits have also been plotted in the diagram.

international bodies, comprising scientific academies and institutions, individual experts, and public-opinion groups. The international reviews of KBS-I have been published in a Swedish Government report (DsI 1978: 99), while the reviews of KBS-II (and power industry comments) presently are collected in an unpublished report. The KBS has been allowed to respond to the reviewer's viewpoints. The Government's decision is made on the reviews and rebuttals obtained.

Some of the criticisms of the KBS projects have been touched upon above. A survey of the critical viewpoints will be presented below, but presentation of all reviewer's viewpoints and KBS rebuttals would require too much space.

2.8.1 KBS-I (Disposal of Reprocessing Waste)

For an initial, early evaluation of the KBS-I project, the Swedish Government requested Professors J. Rydberg (Chalmers Technical University, Gothenburg, Sweden) and J. Winchester (Florida State University, Tallahassee) to reach an agreement on scientific certainties and uncertainties of the KBS proposal. In their over-200-page report (DsI 1978:17) these evaluators commend the general multibarrier concept and storage in granite rock, but raise questions about a number of issues, such as (1) the proper interpretation of the Stipulation Law, (2) the long intermediate storage time before final disposal, (3) the availability of a sufficiently large rock body with sufficiently low permeability, (4) the effect of high temperature on corrosion of metal barriers and on the effect of clay, and (5) the reliability of the water ages presented. They also questioned the wisdom of basing a safety analysis on a series of "pessimistic assumptions" rather than the most probable case (with statistical spread to "optimistic" and "pessimistic" cases). In their overall conclusions Rydberg and Winchester could not agree: Rydberg concluded that the KBS concept as a whole guaranteed that no more radioactivity would leach out to nearby wells of drinking water than these wells presently contain because of natural radioactivity of the granite, while Winchester concluded that the number of uncertainties showed that the concept did not fulfill the Stipulation Law.

Most reviewers have accepted the KBS statements on glass and canister stability (or dissolution times), though several pointed to the lack of extensive knowledge about the dissolution processes. In general, the safety analysis scheme (Fig. 26) was also accepted. Criticism focused on questions

of geologic stability and the availability of a sufficiently large rock body with low groundwater permeability ($\lesssim 10^{-9}$ m/s). Also, some geologists were not satisfied that groundwater flow would be unaffected by future ice ages. Several reviewers contended that the KBS concept was not "absolutely safe," *because of limited knowledge of geochemical reactions between depository* components (canisters, backfill, and rock), especially retention factors. The long time scale bothered some reviewers, considering the instability of present society and possibilities for sabotage. Many of the nonchemist reviewers (physicists, geologists, etc.) refused to consider retention as contributing to the safety of the depository.

2.8.2 KBS-II (Disposal of Unreprocessed Spent Fuel Elements)

The international reviewers include National Academy of Sciences, Environmental Protection Agency, United States Geological Survey, and Lawrence Livermore National Laboratory in the United States and CEA (France), IAEA (Vienna), NEA (Brussels), and NRPB (UK).

The National Academy of Sciences states:

"Both experiment and theory have demonstrated that the canisters will have sufficient mechanical strength and corrosion resistance to survive in the designed repository environment for hundreds of thousands and probably more than a million years ..."

"Extensive research on bentonite clay has shown fairly convincingly that this material can protect the canisters against mechanical disturbance and corrosive attack by groundwater ..."

"The existence of at least one site in Swedish bedrock that meets the minimum criteria of dimension and low groundwater movement, though not conclusively demonstrated, is reasonably assured, and it can be inferred from available geologic data that other equally good or better sites exist in Sweden ..."

"Despite doubts by a very few Swedish geologists, there is substantial evidence that over large parts of Sweden the bedrock is tectonically stable so that a well chosen repository site is in little danger of damage from either slow rock movement or rapid dislocations accompanying earthquakes. Observations of the effects of past glaciation in Scandinavia indicate that possible renewed glaciation will cause no damage to a well-constructed repository or damage too slight to pose a threat to the post-glacial biosphere ..."

"Temperatures will be low enough so that their effect on corrosion of the canisters and on the properties of the bentonite will be negligible. The rise in temperature in the rock around a repository is expected to be below the level that might cause damage either by setting up convection cells in groundwater or by changing the fracture hydrology of the rock ..."

"It has been demonstrated fairly convincingly that the planned bentonite seals and backfill for shafts, tunnels, and boreholes after a repository is filled will be adequate to prevent channeling of groundwater. Nevertheless, the Subcommittee thinks that this is the weakest part of the KBS-II Plan ..."

"If unexpectedly rapid corrosion or a flaw in a canister should permit groundwater to come in contact with spent fuel rods, escape of dissolved nuclides will be greatly retarded by the insolubility of the uranium oxide pellets and by sorption and ion exchange on bentonite and on mineral surfaces in the rock through which the groundwater moves. The retardation, plus effects of dilution and dispersion, is expected to ensure that concentrations in moving groundwater will not reach unacceptable levels ..."

"The Subcommittee agrees that the available technical data are adequate to support the conclusion in the KBS-II Plan that radionuclides will not escape at unacceptable rates from a repository built as specified in the KBS-II report, provided that construction is well engineered and a proper site is used."

The United States Geological Survey is concerned (as in their review of KBS-I) about groundwater flow, composition, and chemical reactions as well as effects of thermal stress on the rock. Although they agree that the proposal is sufficient to fulfill the Stipulation Law, they fear that the economic value of copper, lead, and uranium in the repository makes future human intrusion probable.

The United States Environmental Protection Agency contends that "solubility considerations drastically limit the amount of the nuclides that can be transported to people" and credit the geohydrochemical description of the report. The Agency fears future disturbances of the repository because of its metal values and suggests a more thorough study of the consequences of such an intrusion.

The Energy Technology Center (at Chalmers Technical University, Sweden) concludes that storage of spent fuel elements is inherently inferior to the storage of reprocessing waste, considering handling technique and risks, economy, space, plutonium control, resource preservation, environmental burden

on society, and long-range risks. The Energy Technology Center is critical of KBS safety analysis, especially of calculations of nuclide transport in the biosphere and the resulting doses to man, and it points out a number of technical deficiencies of the KBS-II report. Still, the Center finds the concept as a whole sufficiently trustworthy for a realization.

The Lawrence Livermore National Laboratory makes a technically detailed review, which is too extensive to summarize here. They conclude that in a socially unstable world it would be better to dispose of the spent fuel elements as fast as possible. Though the radioactive risk after 1000 years is comparable to other natural risks, the dissolution of the canister may lead to lead poisoning, which is a larger risk. They recommend using an alumina canister and reprocessing to reduce the above risks. In summary, they conclude that the KBS concept is "exceedingly safe."

In accordance with Swedish rules, KBS has been given the opportunity to respond to criticism from the reviewers. Discussions have been conducted with KBS and consultants, and the final KBS reply is now ready. KBS is now considering whether to publish its final conclusion--which will lead to additional public debate--or to wait until early in 1983. That will be the latest possible time, because in May 1983 SKBF (parent organization of KBS) has to request permission from the government to load the reactors Forsmark III and Oskarshamn III. Such loading will be granted only when the nuclear industry has proven that spent reactor fuel can be safely stored (KBS-II concept). KBS reasons that it can be advantageous to wait until early 1983 for more confirming results for their KBS-II proposal.

2.9 RECENT CHANGES

A new law was introduced in Sweden on July 1, 1981. According to this law (see Appendix E for full text) the nuclear power producer is responsible for all nuclear fuel back-end management. This law makes the power industry responsible for designing and operating all handling equipment, treatment, and storage facilities for spent fuel elements and all kinds of radioactive waste (low, medium, and high) for all time. Part of this responsibility can be met on a contracting basis, as in the example of reprocessing, for which SKBF (the joint organization of the nuclear power industry) has a contract with Cogema in France. The government has created three organizations to oversee this activity:

1. Statens stralskyddsinstitut (SSI). The National Radiation Protection Institute is responsible for radiation levels and can stop all operations which exceed permitted levels. The present head is Professor B. Lindell.

2. Statens kärnkraftsinspektion (SKI). The National Nuclear Power Inspectorate sets standards for reliable and safe operation of all nuclear facilities and can order the shutdown of any facility not meeting these standards. The present head is Professor L. Nordstrom.

3. Nämnden för hantering av använt bränsle (NÄHAB).* The "State Board for Spent Fuel Management" was created on July 1, 1981. It has to periodically review the nuclear power industry activities in dealing with waste, as further described below. The present chairman is S. Romanus, and the director is Gerhard Rundquist.

Organizationally, SSI is part of the Department of Agriculture, and SKI and NAHAB are parts of the Department of Industry.

According to the new law, the power companies have to set aside funds for future expenses for waste management. At present, 1 öre/kWh is set aside (electricity now costs about 12 öre/kWh). The funds are controlled by NÄHAB, but used for all steps of the backend, such as:

(i) Construction of transport vessels and ships; to be delivered in 1982.

(ii) Construction of central spent fuel storage facility (CLAB); will be ready in 1985.

(iii) Construction of storage for medium and low-level reactor waste (SFR); construction permit will be requested in February 1982, building complete in 1988.

(iv) Construction of intermediate storage for high level waste; building will be ready to accept HLW-glass in 1990.

(v) Test drillings and design of final storage for spent fuel elements and HLW-glass.

*Preliminary abbreviation.

(vi) Research and design for all above activities; $10 million will be set aside annually for problems related to the safety of the final waste storage (glass dissolution, geochemistry, radionuclide migration, etc.); probably one third of this will be spent on chemical problems.

On all these questions KBS will let the contracts and draw money from the fund. Control by NÄHAB will probably be exercised only after a project has been running for some years.

3. PRESENT RESEARCH EFFORTS

The Swedish research efforts described below concern mainly high-level waste, though work on medium- and low-level waste is also studied by the research organizations mentioned. (See also Fig. 31.) Most of the efforts have bee stimulated by passage of the Stipulation Law, though PRAV conducted some waste activities even before them. In addition to the applied research described below, the National Science Research Council supports some basic research of importance to the waste studies—for example, studies of actinide complexes in aqueous media.

3.1 KBS RESEARCH AND DEVELOPMENT EFFORTS

There has been a close collaboration between the research and development efforts of the present KBS project, with an annual budget of 20-25 Mkr ($5-6 million), and the PRAV projects (Section 3.4), with an annual budget of about 15 Mkr (about $3.5 million). Originally, PRAV was to support more long-range and semi-basic projects and KBS was to meet the more immediate demands of the nuclear industry. However, the activities have been overlapping in this respect, though not duplicating. The good collaboration between PRAV, KBS, and SKBF has been facilitated by their adjacent administrative organizations in the same building. From July 1, 1981, PRAV is replaced by NÄHAB (see Section 2.9). Some of the KBS efforts are summarized below.

The comments of the reviewers on the KBS-I and KBS-II reports have made it clear that, in general, they considered the isolating properties of the geologic barrier not to be as well verified as the containment function of the engineered barriers. Accordingly, the major emphasis in the ongoing research and development program is on further studies of the properties of the geologic barrier.

3.1.1 Identification and Selection of Rock Bodies for Waste Disposal

It is considered necessary, mainly for political reasons, to make a complete survey of the country for rock bodies suitable as depositories. This survey proceeds in steps. It starts with a general survey of (1) portions of the country with the aid of air photos, physiographic and geologic maps and

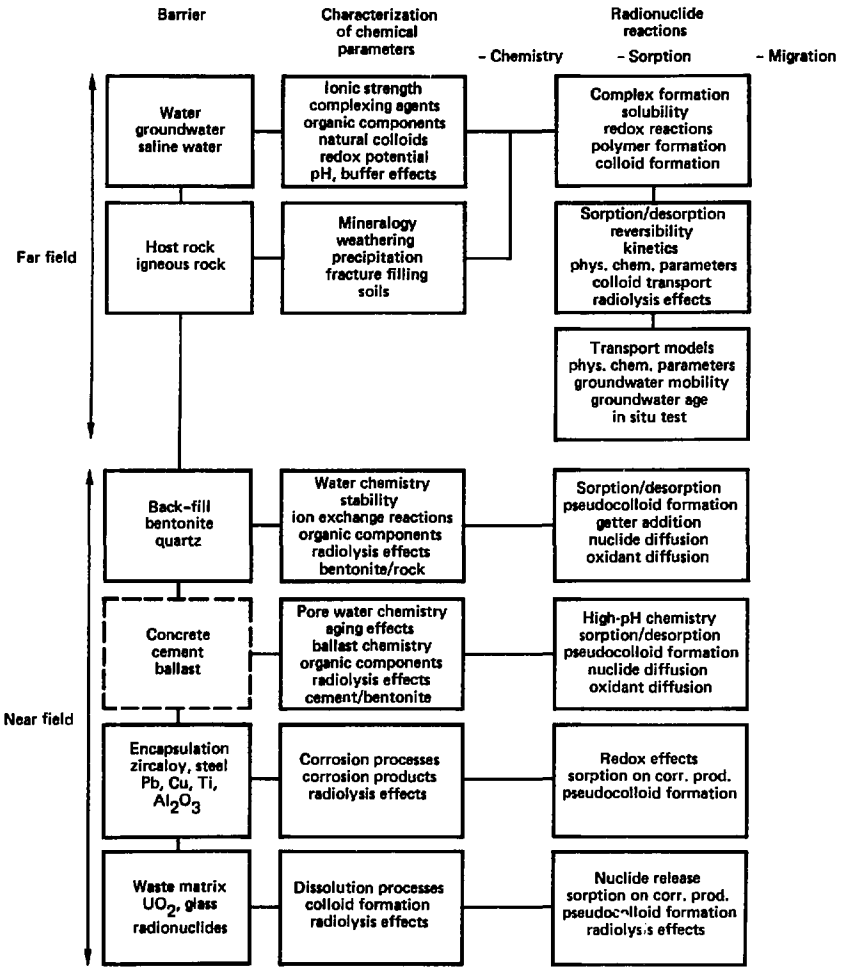


FIG. 31. Swedish fields of research on nuclear waste safety.

(2) ...e archives on well drillings and rock work at the Geologic Survey of Sweden. It continues with (3) ground reconnaissance of the promising areas. The two or three areas that are best suited for geologic investigation and are judged to have the desired quality are (4) selected for geophysical measurements, drillings, and borehole loggings. This program will over a five-year period give data from 10 to 12 sites which should be representative of all types of suitable bedrock in Sweden.

A second five-year period will be used for further studies at a few of the sites that represent different rock types. During this period KBS should be able to include investigations suggested by experience from the hydrological and geochemical work.

3.1.2 Seismicity and Neotectonics

Programs for these studies are now being developed. There is reason to hope that the knowledge about the existence and precise extension of active fault zones will increase considerably during the next 10-year period. A few fault lines have been identified where vertical displacements have taken place since the last glaciation. Studies of one or two of these are being planned.

3.1.3 Hydrogeology

The bedrock at Finnsjön was used as model site in the safety assessment of the KBS report. The hydrogeologic program at that site has been greatly expanded. Studies will be made of groundwater balances, infiltration, water table variations, groundwater runoff, hydraulic connections in the fracture system, groundwater age at various depths, and age differences along identified groundwater flow lines. Field applications of the geochemical research program could also be made here with advantage, as tracer flow tests (Br^-, etc.). Improved well-logging instruments are also tested.

3.1.4 Geochemistry

This ambitious program, which has been carried out with support from PRAV, will cover the spectrum, from research on chemical equilibria in representative groundwater types to *in situ* measurements of radioactive nuclide migration in fractures in crystalline rock. Some of the programs are

- Chemical equilibria between released radionuclides and species in natural groundwater.
- Formation of radiocolloids.
- Interaction between rock minerals and groundwater.
- Sorption of noncolloid radionuclides on rock surfaces and on pure minerals (batch and column).
- Radiocolloid migration (column).
- Chemistry and transport properties of radionuclides in artificial barriers (clay, concrete, etc,).
- Transport models considering pore water effects.
- *In situ* radionuclide migration in fractures.

There are also ongoing investigations of fracture mineralizations in core samples and boreholes of various rock types. Since medium- and low-level waste may be stored close to the surface, underlying peat and clay minerals are studied for permeability, nuclide retention, etc.

3.1.5 Stripa Experiments

Stripa is an abandoned iron mine in central Sweden. The iron ore body was located in a larger granite formation, which now is crisscrossed with corridors down to about 400 m below the surface. The Stripa project, started by KBS in 1977 in collaboration with the US Department of Energy, intended to study the potential of granite rock formations to isolate radioactivity over very long periods of time, and to perform appropriate tests. In 1981 the "International Stripa Project" was launched by the Nuclear Energy Agency (NEA) with seven contracting countries. The project is further described in Appendix C.

3.1.6 Safety Analysis

This includes updating of present programs, improved modeling for water flow in rock fissures, introducing retention data for UO_2, etc. Uptake of uranium and radon by plants is studied at Studsvik and Ultuna.

3.1.7 Glass Research

Stability and leach rate of various host glasses for high-active waste are studied through the Glass Research Institute in collaboration with the Department of Energy and NAGRA (Switzerland). *In situ* glass leaching has started at Stripa.

3.1.8 Bentonite

The colloid properties of bentonite have been carefully studied. Full-scale injection tests of bentonite in rock fractures are going on. Bentonite properties (swelling, permeability, diffusion, pressure, etc.) at different temperatures are studied *in situ* at Stripa. Project leader is Professor R. Pusch at the Luleå Technical University.

3.1.9 Waste Handling

Handling of medium- and low-level waste containing α-emitters is studied on the basis of the KBS concept. Enclosure by hot isostatic pressing into Al_2O_3 is tested; the technique is being evaluated by Cogema. Also, ASEA is developing encapsulation in copper by hot isostatic pressing.

3.1.10 CLAB

SKBF is managing a construction project for an "away-from-reactor" wet storage for spent fuel (CLAB) with a design capacity of 9000 tonnes. This project is now in the construction stage. It is located on the Oskarshamn power plant site. The first one third (i.e., 3000 tonnes) will be in operation in 1985.

3.1.11 SFR

A facility for final storage of reactor waste (SFR, previously named ALMA) is being designed. It is planned to locate it at the Forsmark power plant. A construction permit will be requested in 1982; the facility will be

ready in 1988. It will consist of 4 large concrete silos, 50 m deep and 30 m in diameter, at 40 m depth in bedrock covered by ~10 m of water (Baltic Sea). Bitumen barrels, concrete blocks, etc. (about 100 000 m^3) will be stored in the silos, surrounded by a bentonite packing.

3.1.12 Transportation

Transport containers for spent fuel elements from boiling-water reactors and pressurized-water reactors have been designed in collaboration with Cogema, as well as a 2000-tonne transport ship, which is now being built in France.

3.1.13 Time Schedule

There has been no authorization in Sweden of a master time schedule for the development of a geologic disposal facility. The research and development program is therefore running on an ad-hoc time schedule based on the KBS proposal that no high-level waste should be disposed until around 2010. This leads to a very coarse overall division of the available time.

3.2 THE STRIPA PROJECT

In July 1977 the Swedish State Power Board and the U.S. Department of Energy, through their contractors, the Swedish Nuclear Fuel Supply Company (SKBF) and the Lawrence Berkeley Laboratory (LBL), began to study the feasibility of storing radioactive wastes in mined granite caverns, at Stripa, about 200 km northwest of Stockholm. During the last three years a series of tests has been carried out addressing the problem of predicting the thermomechanical behavior of a heterogeneous and discontinuous granite mass and predicting the movement of groundwater through such granite.

The KBS work program includes the following:

- Studies of mining techniques conducted by Ställbersbolagen, the mining company which owns and operates the Stripa mine.
- Subsurface geologic mapping of fractures in the underground openings, by the Swedish Geologic Survey.
- Effect of pressure and temperature gradients on rock permeability. Hagconsult is measuring water flow rates between vertical holes in the rock as a function of pressure and temperature.

- Grouting studies.
- Thermal stress studies.
- Hydrologic studies in a deep borehole (410-900 m).
- Determination of *in situ* stress (University of Luleå).
- Laboratory determination of rock properties.

The U.S.-Sweden cooperative program has covered the following tasks:

- Task 1. Full-Scale Heater Experiment, to investigate the effects of temperature increases in crystalline rocks. Two heaters the size of waste canisters, with an output of up to 5 kW, have been operated for about two years.
- Task 2. Time-Scaled Heater Experiment, to determine the long-term effect of waste heat in a fractured rock mass.
- Task 3. Assessment of Fracture Hydrology, to define the surface and subsurface hydrological conditions of the fractured granite rock mass as a function of time and temperature, using various borehole tests.
- Task 4. Geophysical Assessment of Fractured Rock Masses, to determine the applicability of differert surface, subsurface, and borehole techniques to locate the fracture system in granite rock masses.
- Task 5. Laboratory Measurement of Material Properties, to measure the properties of rock samples from the Stripa mine.
- Task 6. Mass Transfer of Water to the Ventilation System, to determine the gross seepage rate in the low-permeability granite rock at Stripa.
- Task 7. Measurement of *in Situ* Stresses by Hydraulic Fracturing, to determine the virgin state of stress in the Stripa rock mass.

Figure 32 summarizes the hydrogeological theoretical approach of the Stripa studies.

In the Stripa mine iron ore has been produced since 1485. It was closed in 1976 by its latest owner, Ställbergsbolagen. The mine drew its ore from hematite-rich zones in leptite, a quartzofeldspathic metamorphic rock of the granulite facies. This leptite was intruded by a medium-grained granite in which a considerable part of the mine accessways are located. It is from such drifts at the 360-m level of the mine that drifts were excavated for the *in situ* heater tests and other experiments at the 338-m level.

The essential result is that the heat load tested has relatively small effect on the overall hydraulic conductivity, which is 10^{-11} m/s in large-scale tests.

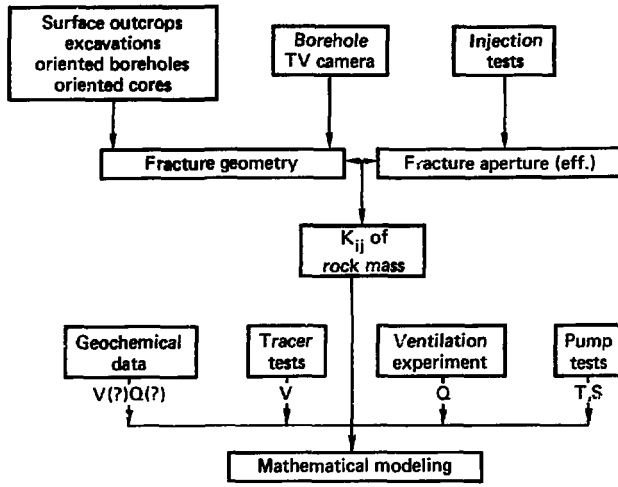


FIG. 32. Block diagram of steps in calculating directional permeabilities from fracture data.

Groundwater seems to be divided into two regions, one originating from the surface and penetrating down to at least the 400-m level, and one region consisting of fossil seawater and subglacial melting water below about 600 m. This water seems to be very old, as is indicated in Fig. 33.

The findings at Stripa are being published in a series of joint reports by SKBF And LBL. Appendix C is a summary of the hydrological findings up to March 1980.

In 1981 OECD/NEA entered into the "International Stripa Project" with Finland, Japan, Sweden, Switzerland, and the United States as full participants, and Canada and France as associate members. The work program of the project, already under way, concentrates on the following main areas:

- Hydrogeological and geochemical investigations in boreholes excavated from the mine "bottom" at about 350 m below the surface down to 1400 m.
- Ion migration tests in single fissures to study radionuclide transport mechanisms through the geologic medium.
- Investigation of backfill material used to buffer the waste from the rock and to plug access and boreholes that offer potential pathways to the biosphere.

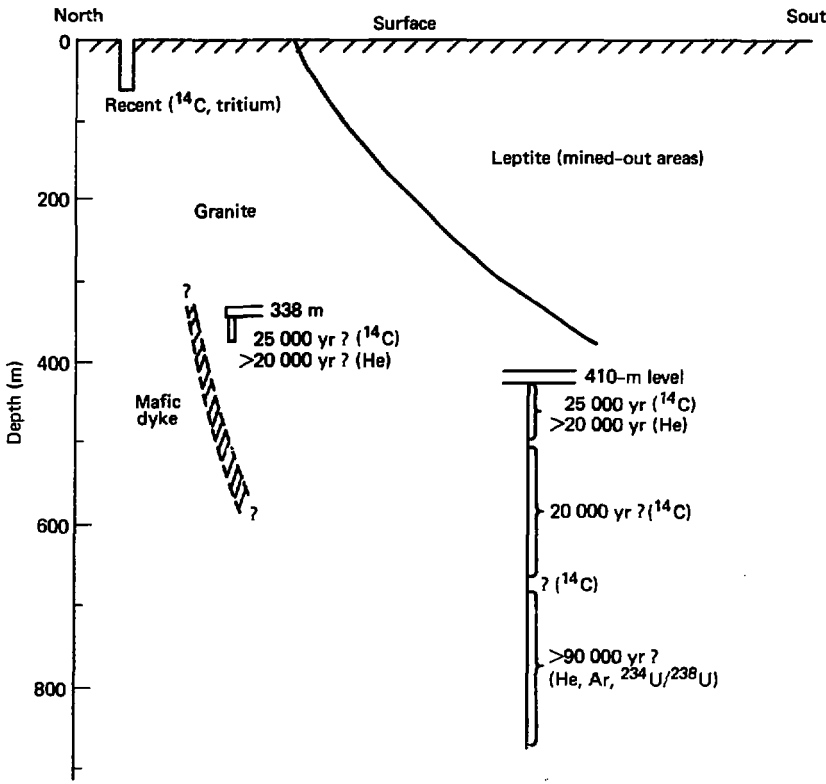


FIG. 33. Preliminary results of isotope age dating of groundwater samples.

- Leaching of glass in true geologic environment.

This project will be continued until 1984 at a cost of about $10 million.

3.3 ASEA CERAMIC CANISTER

The ASEA company is the largest electric supply company in Sweden; ASEA-Atom, a subsidiary company, builds boiling-water reactors, and has so far delivered eight such stations. ASEA also manufactures industrial diamonds, tool steel, and cemented carbide of extreme hardness. These products are

fabricated through the use of hot isostatic pressing, at temperatures up to about 2000°C, and pressures up to several thousand bars (100 MPa). The pressure is applied evenly (isostatic) through a gas system. Pieces with dimensions up to several meters can be fabricated in this fashion.

Using this technique, ASEA has developed a container for spent reactor fuel elements at its laboratory at Robertsfors. The container is α-aluminum oxide (Al_2O_3) of high purity, (99.8%) sintered at 1400°C and 100-300 MPa. The process transforms the Al_2O_3 into corundum; the density of the sintered material is 99.5% of the theoretical. Corundum is, next to diamond, the hardest material in nature. It appears naturally as ruby, sapphire, and emery.

Since the material is found in nature in geological formations which are a thousand million years old, it is obvious that the corundum has an extremely high resistance to natural destructive processes. Thus, in water of 80°C and pH 8.5--conditions expected in a radioactive waste repository--the dissolution rate is equivalent to less than 0.07 mm in 1000 years. Stated otherwise, a 10-cm (4-in.) thick corundum vessel will not dissolve completely in one million years. In fact, experiments indicate that the granite rock will dissolve sooner than corundum canisters placed in the rock.

The process for canning spent fuel elements in full-scale corundum canisters has been tested. Figure 34 shows a full-size canister. The spent fuel elements (two at a time) are placed in a stainless steel tube, air is removed an the tube is sealed, the tube is then rolled flat, coiled in a specially designed mill, and placed inside the preformed canister. When the canister is filled to one half its volume with rolled coils, the upper internal volume is filled with Al_2O_3 fibers and MgO as heat insulators, before the top is sintered onto the body of the canister. In this way a seamless seal is produced. The technique for enclosing unrolled spent fuel elements is presently being developed.

Figure 35 shows the production chain for the aluminum oxide canisters, and Fig. 36 illustrates the encapsulation procedure for the spent fuel elements. This is also illustrated in Fig. 37. Figure 38 shows the closing of the container, followed by hot isostatic pressing, which produces a completely seamless canister.

Table 9 summarizes the characteristics of the container material. It can be seen that the dissolution rate is less by a factor of about 10 than the presently assumed rate for the dissolution of vitrified waste. Dissolution studies in true groundwater have, in effect, shown that no dissolution at all

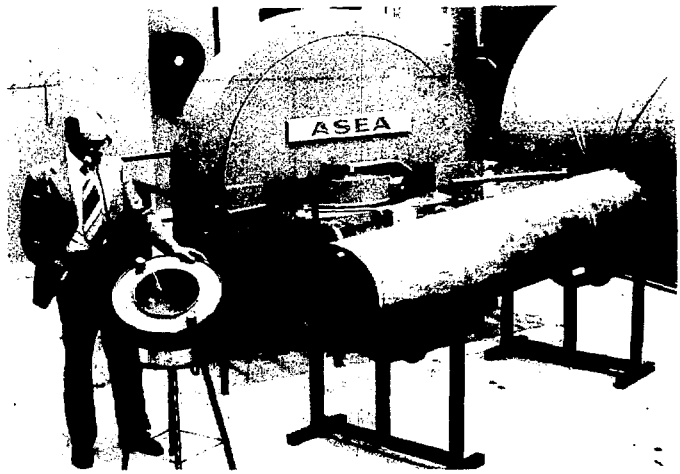


FIG. 34. Full-size canister of aluminum oxide for direct disposal of spent nuclear fuel. In the background, the QUINTUS press for hot-isostatic pressing.

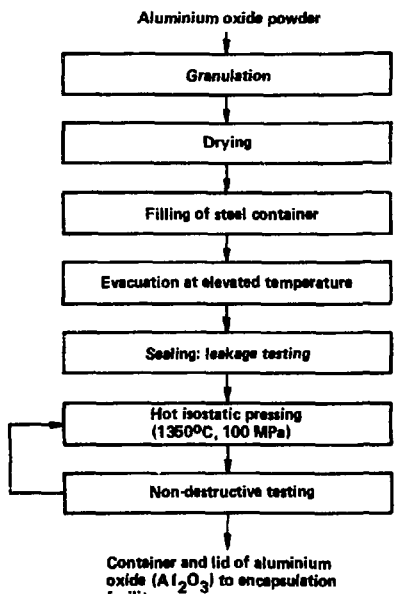


FIG. 35. Schematic diagram of production chain for aluminum oxide canister for spent nuclear fuel.

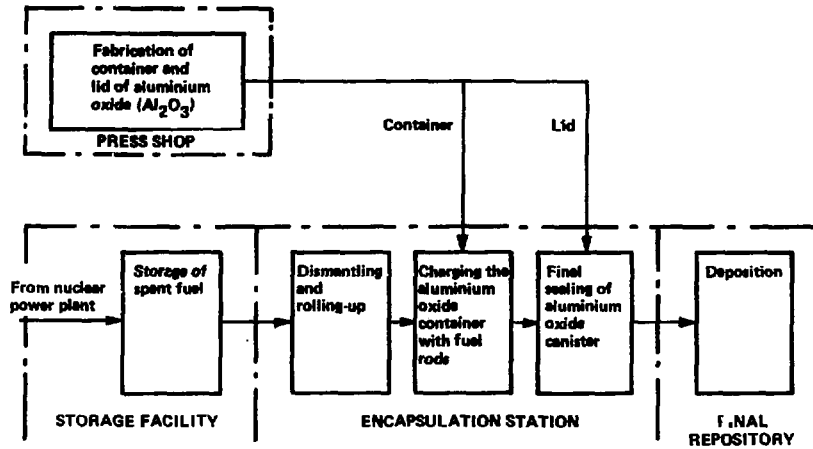


FIG. 36. Block diagram illustrating encapsulation of spent nuclear fuel in aluminum oxide canister.

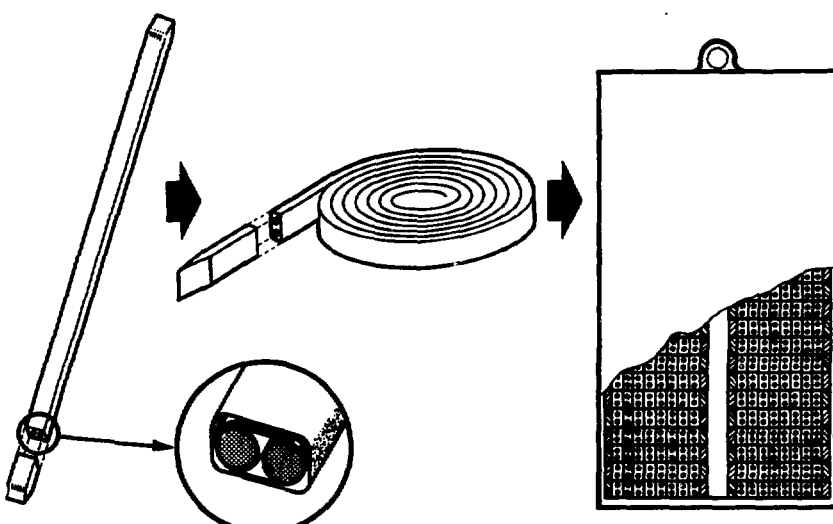


FIG. 37. The fuel rods are encapsulated in pairs in steel sheaths which are then rolled up. The resultant coils are packed in a cylinder of stainless steel.

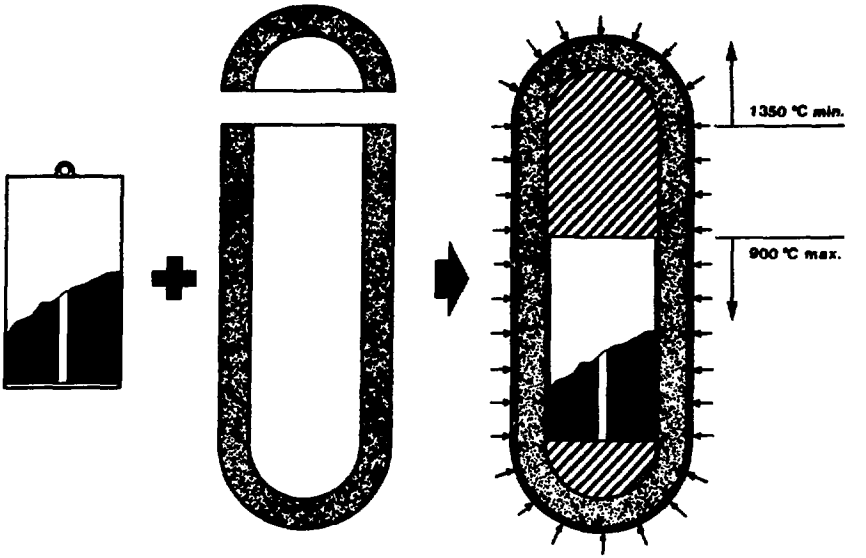


FIG. 38. The stainless steel cylinder with fuel rods is placed in the container. The lid and the container are sealed by means of hot isostatic pressing to produce a completely seamless canister.

takes place. Instead, silica and metal ions in the groundwater react with the alumina to form alumino-silicates of rock type, which leads to some increase in the wall thickness.

Although the technique has been developed for unreprocessed spent fuel elements, there is no reason why it should not be applicable to high-level vitrified waste. It should also be noted that the canister is not irretrievable. It can be cut open with a diamond tool, a technique which is well established.

Resources of bauxite and other aluminium-bearing raw materials are virtually inexhaustible. The cost of high-purity alpha-aluminium oxide powder is currently around SKr 5/kg (US $0.5/lb).

3.4 NATIONAL COUNCIL FOR RADIOACTIVE WASTE

The council was formed late in 1975 in order to support the exploratory and developmental work necessary for the nuclear power program which the

TABLE 9. Characteristic properties of >99.8%-pure alpha-aluminium oxide (Al_2O_3).

Property	Significance
High chemical resistance:	
Extremely high corrosion resistance:	Groundwater affects canister's integrity very slowly
Lower solubility in water at pH values in final repository than most other minerals	
Dissolution extremely slow at temperatures in final repository	Less than 8×10^{-8} $g \cdot cm^{-2} \cdot d^{-1}$ or less than 0.07 $\mu m \cdot yr^{-1}$
Difficult to reduce	Not affected by hydrogen or carbon
Aluminium in its highest oxidation state	Not affected by oxygen
High stability:	
Thermodynamically stable from room temperature up to melting point (>2000°C)	Spontaneous transformation to other structure is impossible without action of other substances
A single-phase material	Mechanical and chemical properties are well-defined
High strength and hardness:	
Flexural strength approximately 500 $MN \cdot m^{-2}$	
Hardness near that of diamond	

Parliament had decided on earlier the same year. The expenses for the council are paid by the nuclear power utilities, but administered by the Government Department of Energy. The present annual budget approaches 13 Mkr (about $3.5 million).

When the KBS program began in late 1976, it became necessary to coordinate the activities of KBS and PRAV. Because the KBS project was supposed to be intensive but short term, the PRAV program was concentrated on more long-range aims. However, both programs have supported short- as well as long-term projects in rather close cooperation. The approximately 30 research and development projects are running on ad-hoc time schedules and are contracted in consecutive steps as appropriate in each case.

PRAV's present operating plan, carried out by researchers from Studsvik Energiteknik AB, the Swedish Geological Survey, and a number of universities, pertains to the following areas of interest:

- Handling of low-level and moderately radioactive reactor wastes. The goals are volume reduction of the bulky waste and separation of long-lived radionuclides for separate handling.
- Transport and central storage of used reactor fuel and worn-out reactor components. The goal is to have a central facility for spent fuel storage and a supporting transport system in use by about 1985.
- Recycling of used reactor fuel and conversion of hot waste to solid form.
- Final storage of waste in rock formations. Initial efforts are directed to locating rock formations with few cracks and low groundwater tables, to developing techniques for measuring the age and mobility of groundwater at depths from 100 to 1000 m, and to determining materials and methods which are most suitable for packaging waste materials for geologic storage.
- Removal of actinides from high-level waste from spent fuel reprocessing (actinide fractionation). This program ended in June 1981.
- Hazard evaluation models for the above aspects of waste management.

The major research and development projects are further specified below:

Reactor waste handling, transport, and disposal

- Conceptual design of a medium level waste disposal facility in a rock cavern.

- Sea transport system for medium- and low-level waste from nuclear plants to disposal facility.
- Pathway analysis for nuclide migration from waste bodies to the biosphere.
- Hazard evaluation for sea transport of waste.

Treatment methods of medium- and low-level waste

- Development of inorganic ion-exchange resins (zeolites, titanates).
- Pilot tests on transfer of nuclides from contaminated waste (e.g., organic filter resins) to inorganic ion exchangers.
- Sintering of inorganic ion-exchange resins to ceramic bodies.
- Fluidized bed incineration of organic ion-exchange resins.
- Investigations on long-term storage properties of concrete as matrix and encapsulation for waste.

Final disposal of high-level waste

- Surveying of suitable rock formations.
- Evaluation of geophysical logging techniques.
- Chemical equilibria and radionuclide sorption on bedrock material in interaction with groundwater.
- Sorption mechanisms in rock fissures.
- *In situ* studies of radionuclide migration in fissured bedrock in Studsvik.
- Investigations of fissure mineralogy.

On July 1, 1981, PRAV ceased to exist. Ongoing research is being taken over by SKBF/KBS, under supervision of the new organization, NÄHAB. The expected budget for the program is about $10 million.

APPENDIX A

LIST OF KBS TECHNICAL REPORTS

01 Källstyrkor i utbränt bränsle och högaktivt avfall från en PWR beräknade med ORIGEN
("Emission rates in spent fuel and high-level waste from a PWR, calculated using ORIGEN")
Nils Kjellbert
AB Atomenergi, 1977-04-05

02 PM angående värmeledningstal hos jordmaterial
("Memorandum concerning the thermal conductivity of soil")
Sven Knutsson
Roland Pusch
Luleå Institute of Technology, 1977-04-15

03 Deponering av högaktivt avfall i borrhål med buffertsubstans
("Deposition of high-level waste in boreholes containing buffer material")
Arvid Jacobsson
Roland Pusch
Luleå Institute of Technology, 1977-05-27

04 Deponering av högaktivt avfall i tunnlar med buffertsubstans
("Deposition of high-level waste in tunnels containing buffer material")
Arvid Jacobsson
Roland Pusch
Luleå Institute of Technology, 1977-06-01

05 Orienterande temperaturberäkningar för slutförvaring i berg av radioaktivt avfall, Rapport 1
("Preliminary temperature calculations for the final storage of radioactive waste in rock, Report 1")
Roland Blomqvist
AB Atomenergi, 1977-03-17

06 Groundwater movements around a repository, Phase 1, State of the art and detailed study plan
Ulf Lindblom
Hagconsult AB, 1977-02-28

07 Resteffekt studier för KBS ("Decay power studies for KBS")
Del 1 Litteraturgenomgång ("Part 1 Review of the literature")
Del 2 Beräkningar ("Part 2 Calculations")
Kim Ekberg
Nils Kjellbert
Göran Olsson
AB Atomenergi, 1977-04-19

08 Utlakning av franskt, engelskt och kanadensiskt glas med högaktivt avfall
("Leaching of French, English and Canadian glass containing high-level waste")
Göran Blomqvist
AB Atomenergi, 1977-05-20

09 Diffusion of soluble materials in a fluid filling a porous medium
Hans Häggblom
AB Atomenergi, 1977-03-24

10 Translation and development of the BNWL-Geosphere Model
Bertil Grundfelt
Kemakta Konsult AB, 1977-02-05

11 Utredning rörande titans lämplighet som korrosionshärdig kapsling för kärnbränsleavfall
("Study of suitability of titanium as corrosion-resistant cladding for nuclear fuel waste")
Sture Henriksson
AB Atomenergi, 1977-08-24

12 Bedömning av egenskaper och funktion hos betong i samband med slutlig förvaring av kärnbränsleavfall i berg
("Evaluation of properties and function of concrete in connection with final storage of nuclear fuel waste in rock")
Sven G Bergström
Göran Fagerlund
Lars Rombén
The Swedish Cement and Concrete Research Institute, 1977-06-22

13 Urlakning av använt kärnbränsle (bestrålad uranoxid) vid direktdeponering
("Leaching of spent nuclear fuel (irradiated uranium oxide) following direct deposition")
Ragnar Gelin
AB Atomenergi, 1977-06-08

14 Influence of cementation on the deformation properties of bentonite/quartz buffer substance
Roland Pusch
Luleå Institute of Technology, 1977-06-20

15 Orienterande temperaturberäkningar för slutförvaring i berg av radioaktivt avfall, Rapport 2
("Preliminary temperature calculations for the final storage of radioactive waste in rock, Report 2")
Roland Blomqvist
AB Atomenergi, 1977-05-17

16 Översikt av utländska riskanalyser samt planer och projekt rörande slutförvaring
("Review of foreign risk analyses and plans and projects concerning final storage")
Åke Hultgren
AB Atomenergi, August 1977

17 The gravity field in Fennoscandia and postglacial crustal movements
Arne Bjerhammar
Stockholm, August 1977

18 Rörelser och instabiliteter i den svenska berggrunden
("Movements and instability in the Swedish bedrock")
Nils-Axel Mörner
University of Stockholm, August 1977

19 Studier av neotektonisk aktivitet i mellersta och norra Sverige, flygbildsgenomgång och geofysisk tolkning av recenta förkastningar
("Studies of neotectonic activities in central and northern Sweden, review of aerial photographs and geophysical interpretation of recent faults")
Robert Lagerbäck
Herbert Henkel
Geological Survey of Sweden, September 1977

20 Tektonisk analys av södra Sverige, Vättern - Norra Skåne
("Tectonic analysis of southern Sweden, Lake Vättern - Northern Skåne")
Kennert Röshoff
Erik Lagerlund
University of Lund and Luleå Institute of Technology, September 1977

21 Earthquakes of Sweden 1891 - 1957, 1963 - 1972
Ota Kulhánek
Rutger Wahlström
University of Uppsala, September 1977

22 The influence of rock movement on the stress/strain situation in tunnels or boreholes with radioactive canisters embedded in a bentonite/quartz buffer mass
Roland Pusch
Luleå Institute of Technology, 1977-08-22

23 Water uptake in a bentonite buffer mass
A model study
Roland Pusch
Luleå Institute of Technology, 1977-08-22

24 **Beräkning av utlakning av vissa fissionsprodukter och aktinider från en cylinder av franskt glas**
("Calculation of leaching of certain fission products and actinides from a cylinder made of French glass")
Göran Blomqvist
AB Atomenergi, 1977-07-27

25 Blekinge kustgnejs. Geologi och hydrogeologi
("The Blekinge coastal gneiss, Geology and hydrogeology")
Ingemar Larsson Royal Institute of Technology
Tom Lundgren Swedish Geotechnical Institute
Ulf Wiklander Geological Survey of Sweden
Stockholm, August 1977

26 Bedömning av risken för fördröjt brott i titan
("Evaluation of risk of delayed fracture of titanium")
Kjell Pettersson
AB Atomenergi, 1977-08-25

27 A short review of the formation, stability and cementing properties of natural zeolites
Arvid Jacobsson
Luleå Institute of Technology, 1977-10-03

28 Värmeledningsförsök på buffertsubstans av bentonit/pitesilt
("Thermoconductivity experiments with buffer material of bentonite/pitesilt")
Sven Knutsson
Luleå Institute of Technology, 1977-09-20

29 Deformationer i sprickigt berg
("Deformations in fissured rock")
Ove Stephansson
Luleå Institute of Technology, 1977-09-28

30 Retardation of escaping nuclides from a final depository
Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1977-09-14

31 Bedömning av korrosionsbeständigheten hos material avsedda för kapsling av kärnbränsleavfall. Lägesrapport 1977-09-27 samt kompletterande yttranden
("Evaluation of corrosion resistance of material intended for encapsulation of nuclear fuel waste. Status report, 1977-09-27, and supplementary statements")
Swedish Corrosion Research Institute and its reference group

32 Egenskaper hos bentonitbaserat buffertmaterial
("Properties of bentonite-based buffer material")
Roland Pusch
Arvid Jacobsson
Luleå Institute of Technology, 1978-06-10

33 Required physical and mechanical properties of buffer masses
Roland Pusch
Luleå Institute of Technology, 1977-10-19

34 Tillverkning av bly-titan kapsel
("Fabrication of lead-titanium canister")
Folke Sandelin AB
VBB
ASEA-Kabel
Swedish Institute for Metals Research
Stockholm, November 1977

35 Project for the handling and storage of vitrified high-level waste
Saint Gobain Techniques Nouvelles, October 1977

36 Sammansättning av grundvatten på större djup i granitisk berggrund
("Composition of groundwater deep down in granitic bedrock")
Jan Rennerfelt
Orrje & Co, Stockholm, 1977-11-07

37 Hantering av buffertmaterial av bentonit och kvarts
("Handling of buffer material of bentonite and quartz")
Hans Fagerström, VBB
Björn Lundahl, Stabilator
Stockholm, October 1977

38 Utformning av bergrumsanläggningar
("Design of rock cavern facilities")
Arne Finné, KBS
Alf Engelbrektson, VBB
Stockholm, December 1977

39 Konstruktionsstudier, direktdeponering
("Design studies, direct deposition")
Bengt Lönnerberg
ASEA-ATOM, Västerås, September 1978

40 Ekologisk transport och stråldoser från grundvattenburna radioaktiva ämnen
("Ecological transport and radiation doses from groundwater-borne radioactive substances")
Ronny Bergman
Ulla Bergström
Sverker Evans
AB Atomenergi, 1977-12-20

41 Säkerhet och strålskydd inom kärnkraftområdet.
Lagar, normer och bedömningsgrunder
("Safety and radiation protection in the field of nuclear power. Laws, standards and grounds for evaluation")
Christina Gyllander
Siegfried F Johnson
Stig Rolandson
AB Atomenergi and ASEA-ATOM, 1977-10-13

42 Säkerhet vid hantering, lagring och transport av använt kärnbränsle och förglasat högaktivt avfall
("Safety in the handling, storage and transportation of spent nuclear fuel and vitrified high-level waste")
Ann Margret Ericsson
Kemakta, November 1977

43 Transport av radioaktiva ämnen med grundvatten från ett bergförvar
("Transport of radioacitve elements in groundwater from a rock repository")
Bertil Grundfelt
Kemakta, 1977-12-13

44 Beständighet hos borsilikatglas
("Durability of borosilicate glass")
Tibor Lakatos
Glasteknisk Utveckling AB, Växjö, December 1977

45 Beräkning av temperaturer i ett envånings slutförvar i berg för förglasat radioaktivt avfall
("Calculation of temperatures in a single-level final repository in rock for vitrified radioactive waste")
Report 3
Roland Blomquist
AB Atomenergi, 1977-10-19

46 Temperaturberäkningar för använt bränsle
("Temperature calculations for spent fuel")
Taivo Tarandi
VBB, June 1978

47 Investigations of groundwater flow in rock around repositories for nuclear waste
John Stokes
Roger Thunvik
Department of agricultural hydrotechnics, Royal Institute of Technology, 1978-02-28

48 The mechanical properties of the rocks in Stripa, Kråkemåla, Finnsjön and Blekinge
Graham Swan
Luleå Institute of Technology, 1977-08-29

49 Bergspänningsmätningar i Stripa gruva
("Measurements of rock stresses in the Stripa mine")
Hans Carlsson
Luleå Institute of Technology, 1977-08-29

50 Lakningsförsök med högaktivt franskt glas i Studsvik
("Leaching trials with high-level French glass at Studsvik")
Göran Blomqvist
AB Atomenergi, November 1977

51 Seismotectonic risk modelling for nuclear waste disposal in the Swedish bedrock
F Ringdal
H Gjöystdal
E S Husebye
Royal Norwegian Council for scientific and industrial research, October 1977

52 Calculations of nuclide migration in rock and porous media penetrated by water
H Häggblom
AB Atomenergi, 1977-09-14

53 Mätning av diffusionshastighet för silver i lera-sand-blandning
("Measurement of rate of diffusion of silver in clay-sand-mix")
Bert Allard
Heino Kipatsi
Chalmers University of Technology, 1977-10-15

54 Groundwater movements around a repository

54:01 Geological and geotechnical conditions
Håkan Stille
Anthony Burgess
Ulf E Lindblom
Hagconsult AB, September 1977

54:02 Thermal analyses
Part 1 Conduction heat transfer
Part 2 Advective heat transfer
Joe L Ratigan
Hagconsult AB, September 1977

54:03 Regional groundwater flow analyses
Part 1 Initial conditions
Part 2 Long term residual conditions
Anthony Burgess
Hagconsult AB, October 1977

54:04 Rock mechanics analyses
Joe L Ratigan
Hagconsult AB, September 1977

54:05 Repository domain groundwater flow analyses
Part 1 Permeability perturbations
Part 2 Inflow to repository
Part 3 Thermally induced flow
Joe L Ratigan
Anthony Burgess
Edward L Skiba
Robin Charlwood
Hagconsult AB, September 1977

54:06 Final report
Ulf Lindblom et al
Hagconsult AB, October 1977

55 Sorption av långlivade radionuklider i lera och berg
("Sorption of long-lived radionuclides in clay and rock")
Part 1
Bert Allard
Heino Kipatsi
Jan Rydberg
Chalmers University of Technology, 1977-10-10

56 Radiolys av utfyllnadsmaterial
("Radiolysis of filler material")
Bert Allard
Heino Kipatsi
Jan Rydberg
Chalmers University of Technology, 1977-10-15

57 Stråldoser vid haveri under sjötransport av kärnbränsle
("Radiation doses in the event of a failure during the transport of nuclear fuel by sea")
Anders Appelgren
Ulla Bergström
Lennart Devell
AB Atomenergi, 1978-01-09

58 Strålrisker och högsta tillåtliga stråldoser för människan
("Radiation hazards and maximum permissible radiation doses for human beings")
Gunnar Walinder
FOA, Stockholm, 1977-11-04

59 Tectonic Lineaments in the Baltic from Gävle to Simrishamn
Tom Flodén
University of Stockholm, 1977-12-15

60 Förarbeten till platsval, berggrundsundersökningar
("Preliminary studies for site choice, bedrock studies")
Sören Scherman

Berggrundvattenförhållanden i Finnsjöområdets nordöstra del
("Groundwater conditions in the northeastern sector of the Finnsjö district")
Carl-Erik Klockars
Ove Persson
Geological Survey of Sweden, January 1978

61 Permeabilitetsbestämningar
("Permeability determinations")
Anders Hult
Gunnar Gidlund
Ulf Thoregren

Geofysisk borrhålsmätning
("Geophysical borehole survey")
Kurt-Åke Magnusson
Oscar Duran
Geological Survey of Sweden, January 1978

62 Analyser och åldersbestämningar av grundvatten på stora djup
("Analyses and age determinations of groundwater at great depths")
Gunnar Gidlund
Geological Survey of Sweden, 1978-02-14

63 Geologisk och hydrogeologisk grunddokumentation av Stripa försöksstation
("Geological and hydrogeological documentation at the Stripa research station")
Andrei Olkiewicz
Kenth Hansson
Karl-Erik Almén
Gunnar Gidlund
Geological Survey of Sweden, February 1978

64 Spänningsmätningar i Skandinavisk berggrund - förutsättningar, resultat och tolkning
("Stress measurements in Scandinavian bedrock - premises, results and interpretation")
Sten G A Bergman
Stockholm, November 1977

65 Säkerhetsanalys av inkapslingsprocesser
("Safety analysis of encapsulation processes")
Göran Carleson
AB Atomenergi, 1978-01-27

66 Några synpunkter på mekanisk säkerhet hos kapsel för kärnbränsleavfall
(Viewpoints on the mechanical reliability of a canister for nuclear waste")
Fred Nilsson
Royal Institute of Technology, Stockholm, February 1978

67 Mätning av galvanisk korrosion mellan titan och bly samt mätning av titans korrosionspotential under γ-bestrålning
("Measurement of galvanic corrosion between titanium and lead and measurement of corrosion potential of titanium under gamma radiation")
3 technical memorandums
Sture Henriksson
Stefan Potura
Maths Åsberg
Derek Lewis
AB Atomenergi, January-February 1978

68 Degraderingsmekanismer vid bassänglagring och hantering av utbränt kraftreaktorbränsle
("Degradation mechanisms in connection with pool storage and handling of spent nuclear reactor fuel")
Gunnar Vesterlund
Torsten Olsson
ASEA-ATOM, 1978-01-18

69 A three-dimensional method for calculating the hydraulic gradient in porous and cracked media
Hans Häggblom
AB Atomenergi, 1978-01-26

70 Lakning av bestrålat UO_2-bränsle
("Leaching of irradiated UO_2 fuel")
Ulla-Britt Eklund
Ronald Forsyth
AB Atomenergi, 1978-02-24

71 Bergspricktätning med bentonit
("Rock fissure sealing with bentonite")
Roland Pusch
Luleå Institute of Technology, 1977-11-16

72 Värmeledningsförsök på buffertsubstans av kompakterad bentonit
("Thermal conductivity tests on buffer material of compacted bentonite")
Sven Knutsson
Luleå Institute of Technology, 1977-11-18

73 Self-injection of highly compacted bentonite into rock joints
Roland Pusch
Luleå Institute of Technology, 1978-02-25

74 Highly compacted Na bentonite as buffer substance
Roland Pusch
Luleå Institute of Technology, 1978-02-25

75 Small-scale bentonite injection test on rock
Roland Pusch
Luleå Institute of Technology, 1978-03-02

76 Experimental determination of the stress/strain situation in a sheared tunnel model with canister
Roland Pusch
Luleå Institute of Technology, 1978-03-02

77 Nuklidvandring från ett bergförvar för utbränt bränsle
("Nuclide migration from a rock repository for spent fuel")
Bertil Grundfelt
Kemakta konsult AB, Stockholm, 1978-08-31

78 Bedömning av radiolys i grundvatten
("Evaluation of radiolysis in groundwater")
Hilbert Christenssen
AB Atomenergi, 1978-02-17

79 Transport of oxidants and radionuclides through a clay barrier
Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1978-02-20

80 Utdiffusion av svårlösliga nuklider ur kapsel efter kapselgenombrott
("Diffusion of poorly soluble nuclides from a canister following canister penetration")
Karin Andersson
Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1978-03-07

81 Tillverkning av kopparkapsel för slutförvaring av använt bränsle
("Fabrication of copper canisters for final storage of spent nuclear fuel")
Jan Bergström
Lennart Gillander
Kåre Hannerz
Liberth Karlsson
Bengt Lönnerberg
Gunnar Nilsson
Sven Olsson
Stefan Sehlstedt
ASEA, ASEA-ATOM, June 1978

82 Hantering och slutförvaring av aktiva metalldelar
("Handling and final storage of radioactive metal components")
Bengt Lönnerberg
Alf Engelbrektsson
Ivars Neretnieks
ASEA-ATOM, VBB (The Swedish Hydraulic Engineering Co., Ltd.),
Royal Institute of Technology, June 1978

83 Hantering av kapslar med använt bränsle i slutförvaret
("Handling of canisters for spent fuel in the final repository")
Alf Engelbrektsson
VBB (The Swedish Hydraulic Engineering Co., Ltd.),
April 1978

84 Tillverkning och hantering av bentonitblock
("Fabrication and handling of bentonite blocks")
Alf Engelbrektsson et al
VBB, ASEA, ASEA-ATOM, Gränges Mineralprocesser, June 1978

85 Beräkning av kryphastigheten hos ett blyhölje innehållande en glaskropp under inverkan av tyngdkraften
("Calculation of the creep rate of a lead jacket containing a glass body under the influence of gravity")
Anders Samuelsson

Förändring av krypegenskaperna hos ett blyhölje som följd av en mekanisk skada
("Alteration of creep properties of a lead jacket as a result of mechanical damage")
Göran Eklund
Institute of Metals Research, September 1977 - April 1978

86 Diffusivitetsmätningar av metan och väte i våt lera
("Diffusivity measurements of methane and hydrogen in wet clay")
Ivars Neretnieks
Christina Skagius
Royal Institute of Technology, Stockholm, 1978-01-09

87 Diffusivitetsmätningar i våt lera Na-lignosulfonat, Sr^{2+}, Cs^{+}
("Diffusivity measurements in wet clay, Na-lignosulphonate Sr^{2+}, Cs^{+}")
Ivars Neretnieks
Christina Skagius
Royal Institute of Technology, Stockholm, 1978-03-16

88 Groundwater chemistry at depth in granites and gneisses
Gunnar Jacks
Royal Institute of Technology, Stockholm, April 1978

89 Inverkan av glaciation på deponeringsanläggning belägen i urberg 500 m under markytan
("Influence of glaciation on a waste repository situated in primary bedrock 500 m below the surface of the ground")
Roland Pusch
Luleå Institute of Technology, 1978-03-16

90 Copper as an encapsulation material for unreprocessed nuclear waste - evaluation from the viewpoint of corrosion
Final report, 1978-03-31
The Swedish Corrosion Research Institute and its reference group

91 Korttidsvariationer i grundvattnets trycknivå
("Short-term variations in the pressure level of the groundwater")
Lars Y Nilsson
Royal Institute of Technology, Stockholm, September 1977

92 Termisk utvidgning hos granitoida bergarter
("Thermal expansion of granitoid rocks")
Ove Stephansson
Luleå Institute of Technology, April 1978

93 Preliminary corrosion studies of glass ceramic code 9617 and a sealing frit for nuclear waste canisters
I D Sundquist
Corning Glass Works, 1978-03-14

94 Avfallsströmmar i upparbetningsprocessen
("Waste flows in reprocessing")
Birgitta Andersson
Ann-Margret Ericsson
Kemakta, March 1978

95 Separering av C-14 vid upparbetningsprocessen
("Separation of C-14 in reprocessing")
Sven Brandberg
Ann-Margret Ericsson
Kemakta, March 1978

96 Korrosionsprovning av olegerat titan i simulerade deponeringsmiljöer för upparbetat kärnbränsleavfall
("Corrosion testing of unalloyed titanium in simulated deposition environments for reprocessed nuclear fuel waste")
Sture Henriksson
Marian de Pourbaix
AB Atomenergi 1978-04-24

97 Colloid chemical aspects of the "confined bentonite concept"
Jean C Le Bell
Institute of Surface Chemistry, 1978-03-07

98 Sorption av långlivade radionuklider i lera och berg Del 2
("Absorption of long-lived radio nuclides in clay and rock Part 2")
Bert Allard
Heiro Kipatsi
Börje Torstenfelt
Chalmers University of Technology, 1978-04-20

99 Lakning av högaktivt franskt glas
("Leaching of high-level radioactive French glass")
Status report 1978-06-01
Göran Blomqvist
AB Atomenergi, 1978-06-19

100 Dos och dosinteckning från grundvattenburna radioaktiva ämnen vid slutförvaring av använt kärnbränsle
("Dose and dose committment from groundwater-borne radioactive elements in the final storage of spent nuclear fuel")
Ronny Bergman
Ulla Bergström
Sverker Evans
AB Atomenergi, 1978-10-06

101 Utläckning av Ni-59 från ett bergförvar
("Leakage of Ni-59 from a rock repository")
Ivars Neretnieks
Karin Andersson
Lennart Henstam
Royal Institute of Technology, Stockholm, 1978-04-24

102 Metod att bocka bestrålade bränslestavar
("Method for bending irradiated fuel rods")
Torsten Olsson
ASEA-ATOM, 1978-03-29

103 Some aspects on colloids as a means for transporting nuclides
Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1978-08-08

104 Finit elementanalys av bentonitfyllt bergförvar
("Finite element analysis of bentonite-filled rock repository")
Ove Stephansson
Kenneth Mäki
Tommy Groth
Per Johansson
Luleå Institute of Technology, July 1978

105 Neutroninducerad aktivitet i bränsleelementdetaljer
("Neutron-induced radioacitvity in fuel assembly components")
Nils A Kjellberg
AB Atomenergi, 1978-03-31

106 Strålningsnivå och till vatten deponerad strålningsenergi utanför kapslar i slutförvaret
("Radiation level and radiant energy imparted to water outside of canisters in the final repository")
Klas Lundgren
ASEA-ATOM, 1978-05-29

107 Blyinfodrad titankapsel för upparbetat och glasat kärnbränsleavfall - Bedömning ur korrosionssynpunkt
("Lead-lined titanium canister for reprocessed and vitrified nuclear fuel waste - Evaluation from the viewpoint of corrosion")
The Swedish Corrosion Institute and its reference group
Final report
1978-05-25

108 Criticality in a spent fuel repository in wet crystalline rock
Peter Behrenz
Kåre Hannerz
ASEA-ATOM, 1978-05-30

109 Lakningsbar spaltaktivitet
("Leachable gap activity")
Lennart Devell
Rolf Hesböl
AB Atomenergi, October 1978

110 In situ experiments on nuclide migration in fractured crystalline rocks
Ove Landström
Carl-Erik Klockars
Karl-Erik Holmberg
Stefan Westerberg
Studsvik Energiteknik and
The Geological Survey of Sweden
July 1978

111 Nuklidhalter i använt LWR-bränsle och i högaktivt avfall från återcykling av plutonium i PWR
("Nuclide levels in spent LWR fuel and in high level waste from the recycling of plutonium in PWR")
Nils Kjellberg
AB Atomenergi, 1978-07-26

112 Säkerhetsanalys av hanteringsförfarandet vid inkapsling av utbränt bränsle i kopparkapsel
("Safety analysis of the handlir edure in the encapsulation of spent fuel in cc . canisters")
Erik Nordesjö
ASEA-ATOM, 1978-03-20

113 Studier av keramiska material för inkapsling av högaktivt avfall
("Studies of ceramic material for encapsulation of high-level waste")
Lennart Hydén et al
ASEA-ATOM, September 1978

114 γ-radiolysis of organic compounds and α-radiolysis of water
Hilbert Christensen
Studsvik Energiteknik AB, 1978-09-07

115 Accelererad utlösning av uran från α-aktivt UO_2
("Accelerated dissolution of uranium from α-active UO_2")
Gösta Nilsson
Studsvik Energiteknik AB, 1978-04-27

116 Lakning av Al_2O_3 under simulerande deponeringsbetingelser
("Leaching of Al_2O_3 under simulated repository conditions")
Britt-Marie Svensson
Lennart Dahl
Studsvik Energiteknik AB, 1978-06-02

117 Lakning av Al_2O_3 i dubbeldestillerat vatten
("Leaching of Al_2O_3 in twice-distilled water")
Britt-Marie Svensson
Göran Blomqvist
Studsvik Energiteknik AB, 1978-05-29

118 Slutrapport Al_2O_3 kapsel
("Final report on Al_2O_3 canister")
The Swedish Corrosion Institute and its reference group

119 Slutförvaring av aktiverade ståldetaljer i betong
("Final storage of activated steel components in concrete")
Lars Rombén
Kyösti Tuutti
The Swedish Cement and Concrete Research Institute,
1978-07-14

120 Some notes in connection with the KBS studies of final disposal of spent fuel
Ivars Neretnieks
Royal Institute of Technology, September 1978

1977-78

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

1979

79-01 Clay particle redistribution and piping phenoma in bentonite/quartz buffer material due to high hydraulic gradients
Roland Pusch
University of Lulea 1979-01-10

79-02 Försöksområdet vid Finnsjön
Beskrivning till berggrunds- och jordartskartor
Karl-Erik Almén
Lennart Ekman
Andrzej Olkiewicz
Sveriges Geologiska Undersökning November 1978

79-03 Bergmekanisk bedömning av temperaturbelastning vid slutförvaring av radioaktivt avfall i berg
Ove Stephansson
Bengt Leijon
Högskolan i Luleå 1979-01-10

79-04 Temperatur- och spänningsberäkning för slutförvar
Taivo Tarandi
VBB Vattenbyggnadsbyrån, Stockholm februari 1979

79-05 Kompletterande berggrundsundersökningar inom Finnsjö- och Karlshamnsområdena
Andrzej Olkiewicz
Soren Scherman
Karl-Axel Kornfält
Sveriges Geologiska Undersökning 1979-02-02

79-06 Kompletterande permeabilitetsmatningar i Karlshamnsområdet
Gunnar Gidlund
Kenth Hansson
Ulf Thoregren
Sveriges Geologiska Undersökning februari 1979

79-07 Kemi hos berggrundvatten i Blekinge
Gunnar Jacks
Institutionen för Kulturteknik, KTH, februari 1979

79-08 Beräkningar av grundvattenrörelser inom Sternoområdet i Blekinge
John Stokes
Institutionen för Kulturteknik, KTH, februari 1979
Preliminar utgåva

79-09 Utvärdering av de hydrogeologiska och berggrundsgeologiska förhållandena på Sternö
Kaj Ahlbom
Leif Carlsson
Gunnar Gidlund
C-E Klockars
Sören Scherman
Ulf Thoren
Sveriges Geologiska Undersökning, Berggrundsbyrån, februari 1979

79-10 Model calculations of groundwater condition on Sternö peninsula
Carl-Lennart Axelsson
Leif Carlsson
Geological Survey of Sweden september 1979

79-11 Tolkning av permeabilitet i en befintling berganläggning
Ulf Lindblom
Alf Norlén
Jesús Granero
Kent Adolfsson
Hagconsult AB februari 1979

79-12 Geofysisk borrhalsmätning i 2 st borrhål på Stenö
Kurt-Åke Magnusson
Oscar Duran
Sveriges Geologiska Undersökning februari 1979

79-13 Bildning av fritt väte vid radiolys i lerbädd
Trygve Eriksen
Johan Lind
Institutet för Kärnkemi KTH 1979-03-28

79-14 Korrosionsprovning av olegerat titan i simulerade deponeringsmiljöer för uppar betat kärnbränsleavfall
Slutrapport
Sture Henrikson
Marian de Pourbaix
Studsvik Energiteknik AB 1979-05-07

79-15 Kostnader för hantering och slutförvaring av högaktivt avfall och använt kärnbränsle
Arne W Finné
Åke Larson Byggare, april 1979

79-16 Beräkning av permeabilitet i stor skala vid bergrum i Karlshamns hamn
Ulf Lindblom
J J Granero

79-17 Water percolation effects on clay-poor bentonite/quartz buffer material at high hydraulic gradients
R Pusch
Div. Soil Mechanics, University of Lulea, 1979-05-31

79-18 Sammanställning och utvärdering av genomförda GETOUT- och BIOPATH-körningar
M Elert
B Grundfelt
C Stenquist
Kemakta AB, Studsvik Energiteknik AB, 1979-08-13

79-19 Diffusion in the rock matrix - An important factor in radionuclide retardation?
Ivars Neretnieks
Royal Institute of Technology May 1979

79-20 Hydraulisk konuktivitet bestämd i stor skala i ytliga partier av Blekinge kustgnejs
Ulf Lindblom, Hagconsult AB, Göteborg
Torbjörn Hahn, Fortifikationsförvaltningen, Stockholm
Göteborg juni 1979

79-21 Teknik och kostnad för rivning av svenska kärnkraftverk
Utarbetad av en särskild arbetsgrupp inom SKBF/KBS,
oktober 1979

79-22 Technology and costs for dismantling a Swedish nuclear power plant.
Prepared by a special working group within SKBF/KBS,
October 1979

79-23 Release of ^{129}I to the coolant
Hilbert Christensen
Studsvik Energiteknik AB 1979-07-20

79-24 Development of glass compositions with 9% waste content for the vitrification of high-level waste from LWR nuclear reactors
Tibor Lakatos
Swedish Glass Research Institute, Växjö October 1979

79-25 Svenska torv- och lerlager - Egenskaper av betydelse för slutlagring av låg- och medelaktivt avfall
Ove Landstrom
Hans Tollbäck
Studsvik Energiteknik AB, November 1979

Mäktiga lerformationer i Sydsverige - En litteraturstudie
Henrik Norlander
Kvartärgeologiska Institutionen
Stockholms Universitet, September 1979

79-26 Leakage of Niobium-94 from an Underground Rock Repository
K Andersson
B Torstenfelt
J Rydberg
Department of Nuclear Chemistry
Chalmers University of Technology, 1979-11-05

LIST OF KBS TECHNICAL REPORTS 1980

Microfiche copies of reports are available from

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

Report No.	List of the Technical Reports
1977-78	
TR 121	KBS Technical Reports 1 - 120. Summaries. Stockholm, May 1979.
1979	
TR 79-28	The KBS Annual Report 1979. KBS Technical Reports 79-01--79-27. Summaries. Stockholm, March 1980.
1980	
*) TR 80-01	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

*)= In Swedish

Report No.	List of the Technical Reports
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
*) 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

*) = In Swedish

Report No.	List of the Technical Reports
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
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

*)= In Swedish

Report No.	List of the Technical Reports

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

APPENDIX B

CONTRACTED AND CONSULTING COMPANIES, INSTITUTIONS AND EXPERTS

INDUSTRIAL AND CONSULTING COMPANIES

AB Atomenergi (=Atomic Energy Company of Sweden)
Ahlsell AB
ASEA
ASEA-ATOM AB
ASEA-Hafo AB
ASEA-Kabel AB
Avesta Jernverks AB
AB Exportspråk
Forsgren Produktion AB
The Mining Industries' Work Study and Consultancy Programme
Gränges Mining Mineral Process Laboratory
Hagby Bruk AB
H Folke Sandelin AB
Hagconsult AB
Hydroconsult AB
IFÖ Electric Högspänning AB
IPA-Konsult
Kemakta Konsult AB
Orrje & Co AB, Ingenjörsfirman
Salénrederierna AB
Stabilator AB
The National Swedish Institute for Materials Testing
Ställbergs Grufveaktiebolag
The Swedish Academy of Engineering Sciences
Teleplan AB
AB Vattenbyggnadsbyrån
WP-system AB

INSTITUTIONS OF HIGHER LEARNING

University of Lund

Division of quarternary geology	E Lagerlund, Ph.D.

University of Gothenburg

Division of marine microbiology	Professor K Gundersen

University of Stockholm

Arrenius Laboratory	T Wadstēn, Ph.L.
Department of geology	Dr. N-A Mörner
	T Flodēn, Ph.L.
Department of microbial geochemistry	Dr. R Hallberg

University of Uppsala

Department of Seismology	Dr. O Kulhānek

Swedish University of Agriculture

Department of chemistry	Professor I Lindqvist

Lund Institute of Technology

Department of inorganic chemistry	Dr. Sten Andersson

Chalmers University of Technology

Department of geology	L Bergström, Ph.D.
Department of nuclear chemistry	Professor J Rydberg
	B Allard, D.Eng.
Department of engineering metals	Dr. I Olefjord
Department of inorganic chemistry	Professor N-G Vannerberg

Royal Institute of Technology

Department of mining	Professor I Janelid
Department of physical chemistry	Dr. C Leygraf
Department of geodesy	Professor A Bjerhammar
Department of chemical engineering	Professor I Neretnieks
Department of land improvement	Professor Y Gustafsson
and drainage	Professor I Larsson
Department of nuclear chemistry	Dr. T E Eriksen
Department of inorganic chemistry	Professor I Grenthe
Department of applied electro-	Professor G Wranglēn
chemistry and corrosion science	

University of Luleå

Division of rock mechanics	Professor O Stephansson
	Dr. K Röshoff
Division of geotechnics	Professor R Pusch
	A Jacobsson, Ph.D.

OTHER INSTITUTIONS

Cement and Concrete Research Institute, Stockholm	Professor S G Bergström
National Defence Research Institute, Stockholm	G Walinder L H Andersson
Glass Research Institute, Växjö	B Simmingsköld, Lic.Eng. T Lakatos, Ph.D.
Institute for Metals Research	G Eklund, D.Eng.
Swedish Corrosion Institute, Stockholm	Professor E Mattsson L Ekbom, Lic.Eng.
Swedish Silicate Institute, Gothenburg	R Carlsson, Lic.Eng.
Tandem accelerator laboratory in Uppsala	Professor A Johansson
Geological Survey of Sweden	Dr. O Brotzen U Thoregren, Ph.L.
Institute of Surface Chemistry	Dr. P Stenius J le Bell, Lic.Eng.

EXPERTS AND RESEARCH WORKERS

Dr. Sten G A Bergman
Claes Helgesson, D.Eng.
Dr. Peter Fritz

FOREIGN ORGANIZATIONS AND COMPANIES

The Risö Research Station
ROSKILDE, Denmark

Royal Norwegian Council for Scientific and Industrial Research
KJELLER, Norway

Frauenhofer-Gesellschaft
Institut für Silicatforschung
WüRZBURG, West Germany

Lehrstuhl für Glas und Keramik
Institut für Steine und Erder
CALUSTAHL-ZELLEFELD, West Germany

Saint Gobain Techniques Nouvelles
PARIS, France

COGEMA
PARIS, France

University of Western Ontario
Professor W Fyfe
LONDON, Ontario, Canada

Corning Glass Works
CORNING, New York, USA

Electric Power Research Institute
PALO ALTO, California, USA

Department of Energy
WASHINGTON D.C., USA

Dr. Ralph E Grim
URBANA, Illinois, USA

Lawrence Berkeley Laboratory
University of California
BERKELEY, California, USA

Nuclear Regulatory Commission
WASHINGTON D.C., USA

Office of Waste Isolation
Union Carbide Corp.
OAK RIDGE, Tennessee

Personnel from the Swedish State Power Board, Oskarshamnsverkets Kraftgrupp AB (OKG), the Swedish Nuclear Fuel Supply Company (SKBF), the Swedish Central Power Supply Board (CDL) and the National Council for Radioactive Waste Management (PRAV) have also contributed their services to KBS. The project has organized reference and working groups within the following areas:

GEOGROUP

Dr. Sten G A Bergman	Stocksund
Dr. Otto Brotzen	Geological Survey of Sweden
Professor Y Gustafsson	Royal Institute of Technology
Dr. Rudolf Hiltscher	Saltsjö-Boo
Professor Ingvar Janelid	Royal Institute of Technology
Dr. U Lindblom	Hagconsult
Professor Roland Pusch	Luleå Institute of Technology
Jan Rennerfelt, Lic.Eng.	Orrje & Co. AB
Professor Ove Stephansson	Luleå Institute of Technology

REFERENCE GROUP FOR CORROSION

Roger Carlsson, Lic.Eng.	Swedish Silicate Research Institute
Chief Engineer Thomas Eckered	Nuclear Power Inspectorate

Lars Ekbom, Lic.Eng.	Swedish Corrosion Institute
Göran Eklund, D.Eng.	Institute for Metals Research
Professor Ingemar Grenthe	Royal Institute of Technology
Dr. Rolf Hallberg	University of Stockholm
Sture Henriksson, M.A.	AB Atomenergi
Professor Einar Mattsson	Corrosion Research Institute
Professor Nils-Gösta Vannerberg	Chalmers University of Technology
Professor Gösta Wranglén	Royal Institute of Technology

CHEMISTRY GROUP

Bert Allard, D.Eng.	Chalmers University of Technology
Dr. Otto Brotzen	Geological Survey of Sweden
Professor Ingemar Grenthe	Royal Institute of Technology
Kåre Hannerz, Lic.Eng.	ASEA-ATOM
Arvid Jacobsson, Ph.D.	Luleå Institute of Technology
Professor Ivars Neretnieks	Royal Institute of Technology
Professor Roland Pusch	Luleå Institute of Technology

GROUP FOR DESIGN STUDIES

Alf Engelbrektsson, M.Eng.	Swedish Hydraulic Engineering Company, Ltd.
Göran Fröman, M.Eng.	Swedish Hydraulic Engineering Company, Ltd.
Bo Gustafsson	SKBF
Bertil Mandahl	OKG

GROUP FOR LEACHING STUDIES

Dr. Hilbert Christenssen	AB Atomenergi
Lennart Devell, M.Eng.	AB Atomenergi
Roy Forsyth, Ph.D.	AB Atomenergi
Ragnar Gehlin, M.Eng.	AB Atomenergi
Kåre Hannerz, Lic.Eng.	ASEA-ATOM

GROUP FOR SAFETY ANALYSIS

Peter Behrenz, graduate physicist	ASEA-ATOM
Ronny Bergman, Ph.D.	AB Atomenergi
Lennart Devell, M.Eng.	AB Atomenergi
Bertil Grundfelt, M.Eng.	Kemakta Konsult AB
Kåre Hannerz, Lic.Eng.	ASEA-ATOM
Professor Ivars Neretnieks	Royal Institute of Technology
Curt Stenqvist	AB Atomenergi

APPENDIX C: HYDROLOGICAL STUDIES OF A POTENTIAL WASTE REPOSITORY IN GRANITE: A SURVEY OF THE STRIPA PROJECT

APPENDIX C

HYDROLOGICAL STUDIES OF A POTENTIAL WASTE REPOSITORY IN GRANITE: A SURVEY OF THE STRIPA PROJECT

ABSTRACT

In July 1977 the Swedish State Power Board and the U.S. Department of Energy, through their contractors, the Swedish Nuclear Fuel Supply Company (SKBF) and the Lawrence Berkeley Laboratory (LBL), joined in a feasibility study of storage of radioactive wastes in mined granite caverns. A program was developed for determining physical parameters and hydrology of rock at an abandoned iron ore mine near Stripa, about 200 km northwest of Stockholm. The mine's tunnels form a labyrinthine maze covering approximately 250 km at 15 levels, part of these through a massive body of granite at about 340 m below the surface. During the last three years a series of tests has been carried out addressing the problem of predicting the thermomechanical behavior of a heterogeneous and discontinuous granite mass and predicting the movement of groundwater through such granite. The findings are being published by SKBF and LBL in a series of joint reports. They are referred to by the designation SAC-01, SAC-02, etc., and are listed at the end of this appendix. These reports provide the basis of this summary of the hydrological findings up to March 1980. General conclusions drawn by the present author are summarized in Section 10.

C-1. BASIC DATA REQUIREMENTS

Storage in mined underground repositories is presently considered as the least objectionable way of disposing of radioactive waste. Crystalline rocks—such as granite—are believed to have properties suitable for the extremely long storage times required. Radionuclides from the stored waste may dissolve in the groundwater and migrate with it through the fracture system penetrating the rock mass. In order to estimate the time for such

radionuclides to reach water bodies used by man, it is essential to know the amount, velocity, and other properties of the groundwater. The amount and velocity depend on the fracture system and its behavior during the thermal cycle associated with the radioactive decay.

At the Stripa mine the suitability of the granite rock for storing radioactive waste is studied by four different methods:

1. Mechanical characterization, including monitoring the responses to thermal loading of jointed rock *in situ*, and mechanical tests on cores from 25 mm to 1 m in diameter.

2. Geological characterization, including detailed surface mapping, subsurface mapping, and core mapping.

3. Geophysical characterization, using a variety of borehole techniques (TV-scanning, gamma logging, neutron logging, sonic tests, measurement of electrical resistivity, etc.).

4. Hydrologic characterization, through injection tests, pump tests, water pressure measurements, and controlled inflow tests to tunnels.

In this report we cover only hydrologic characterization. First, though, we describe the Stripa mine test site.

C-2. STRIPA MINE TEST SITE

Iron ore has been produced from the Stripa mine since 1485. It was closed in 1976 by its latest owner, Stallbergsbolagen. The mine, which is about 200 km northwest of Stockholm, Sweden (Fig. C-1), drew its ore from hematite-rich zones in leptite, a quartzofeldspathic metamorphic rock of the granulite facies. This leptite was intruded by a medium-grained granite in which a considerable part of the mine accessways are located (Figs. C-2 and C-3). It is from such drifts at the 360-m level of the mine that drifts were excavated for the *in situ* heater tests and other experiments at the 338-m level (Fig. C-4).

C-3. HYDROLOGIC DATA AND RADIONUCLIDE TRANSPORT

When dissolved nuclides are transported by groundwater they tend to sorb on rock surfaces along the fractures. This sorption can be measured and is given by the distribution factor D (amount of radionuclide per kilogram of

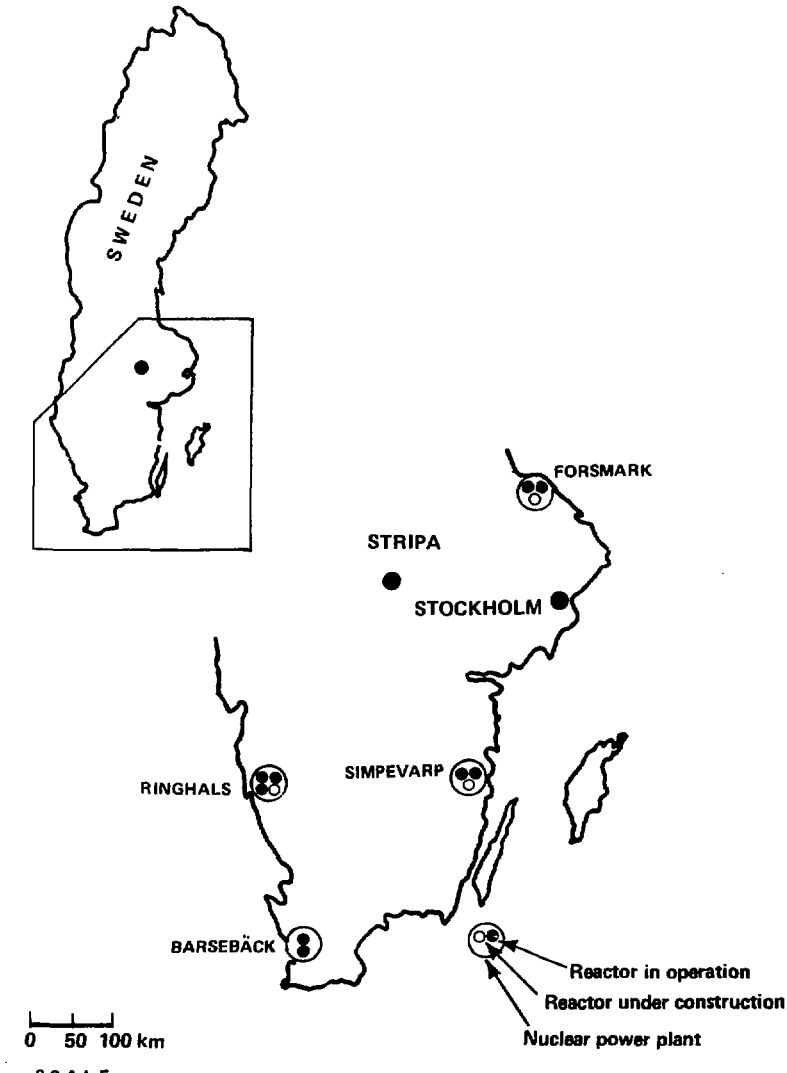


FIG. C-1. Location of Stripa Mine.

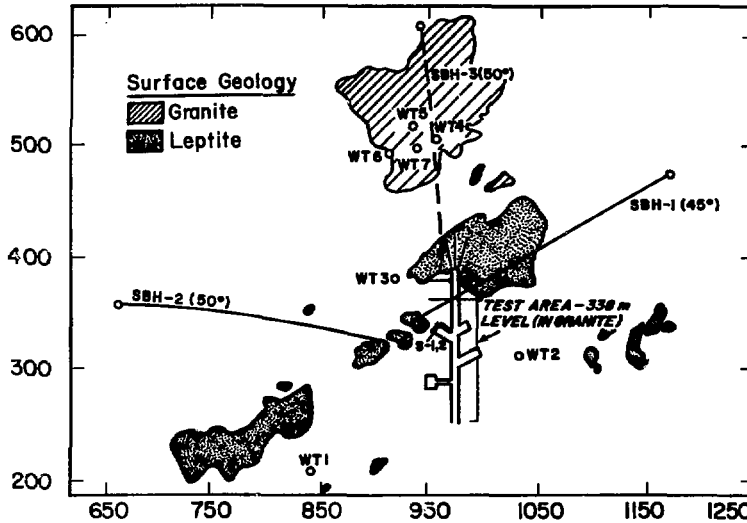


FIG. C-2. Relative locations of bedrock outcrops, surface boreholes (SBH), and test excavations (WT).

rock divided by amounts of radionuclide per cubic meter of water). D depends on the chemical properties of the radionuclide, of the rock, and of the groundwater. One effect of the sorption is that the radionuclide usually travels with a lower velocity, v_n, than the velocity of the groundwater, v_{aq}, according to

$$v_n = v_{aq} \left(1 + D \cdot \frac{\rho}{\epsilon}\right)^{-1} ,$$

where ρ is the rock density (kg/m^3) and ϵ is the rock bulk porosity (volume of water in rock divided by total volume of rock).

The specific discharge of water, v_d, is defined by

$$v_d = Q/A ,$$

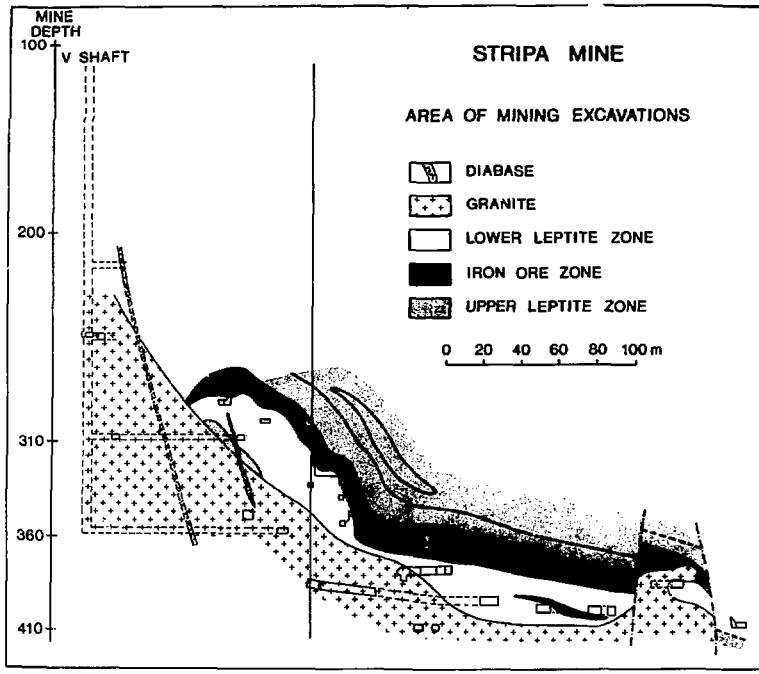


FIG. C-3. Vertical profile showing the subsurface geology in the area of the test excavations along cross section A-B in Fig. C-1.

where Q is the water flow (m^3/s) through the area A(m^2). The water flow through a specific site can be calculated from Darcy's law,

$$Q = K \cdot A \cdot i \quad ,$$

for the hydraulic conductivity K(m/s)* and hydraulic gradient i (or "head",

*K is sometimes referred to as permeability. The (specific or intrinsic) permeability k, however, is defined as $k = K \cdot \mu/(\rho \cdot g)$, where μ is the kinetic viscosity, ρ is the density of the fluid (water), and g is the gravitational constant.

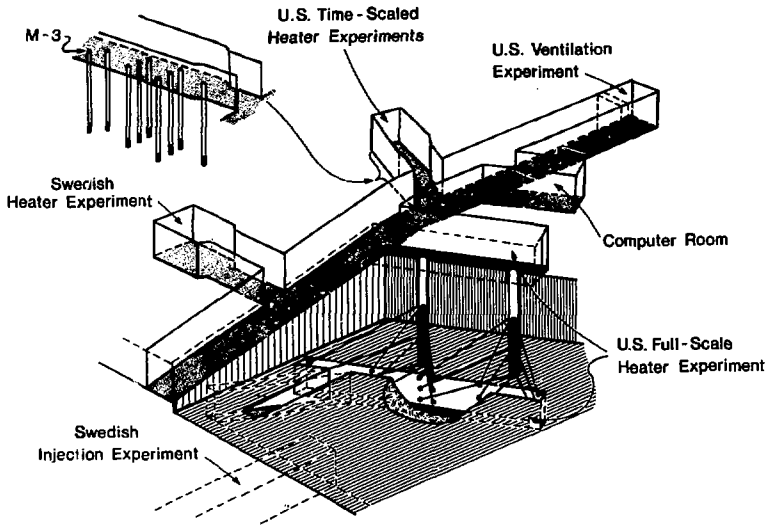


FIG. C-4. Underground excavations near the 330-m level, showing the location of boreholes, including M-3 (SAC-01, SAC-27).

i.e., drop in water pressure, [m], per horizontal distance, [m]). The linear water velocity, in the rock v_{aq} (m/s), is

$$v_{aq} = Q/(\varepsilon \cdot A) \quad ,$$

because water moves only in the rock fractures.

It is obvious that in order to predict the radionuclide transport rate it is essential to know the intrinsic parameters D, K, and ε. The Stripa project has focused on the determination of representative values for K and ε.

C-4. SINGLE BOREHOLE TESTS AND REPRESENTATIVE ELEMENTARY VOLUME

Darcy's equation is valid only for a homogeneous isotropic medium. However, a granite repository is crisscrossed by fractures of different sizes and in different directions, the larger of the fractures being filled with

other minerals, so that each fracture or rock segment is being characterized by its own values of K and ε. The main problem, therefore, becomes how to assign representative values for K and ε assuming that a properly selected repository is free of larger crush-zones, where water flows freely. This problem may be illustrated by the following case.

The hydraulic conductivity has been measured in single holes (10-m deep, 76-mm in diameter) drilled in the floor at the 360-m level of the Stripa mine, see Figs. C-4 and C-5 (SAC-02). Visual inspection of the holes shows 5 to 10 fractures per meter. Some fractures are mm-wide, filled with chlorite (a clay-like weathering product of granite), while fractures <0.2 mm could not be observed (TV Scanning). Double-packer tests* over fractures show that not all fractures are water-bearing.

The results of measurements of hydraulic conductivity (Table C-1) lead to some general conclusions, also verified in other experiments:

- The hydraulic conductivity varies strongly from hole to hole, even when the holes are close together (Fig. C-5).
- The average value of K obtained by the double-packer test does not agree with the (integrated) value of the single-packer test over the same section. In general, the double-packer tests give a higher value of K (up to seven times!).

The hydraulic conductivity varies considerably along a borehole, covering several powers of 10 over distances as short as 10 meters (Fig. C-6). For borehole location, see Fig. C-2. Consequently, when a single measurement of K (or ε) is made in an arbitrary volume of rock, there is no way of knowing, *a priori*, whether or not the measured value is representative of a larger rock body contemplated as a waste repository. As the volume of the fractured rock sample increases from zero, the average value will oscillate as either fractured rock or solid rock is added to the sample (Fig. C-7). When the volume of rock becomes so large that K or ε is no longer sensitive to the effects of individual fractures, the oscillations will subside. An average value can then be assigned to that volume of rock, which is called the representative elementary volume (REV) (Fig. C-7, V3). Theoretically, volumes

*In single-packer tests, the lower part of a borehole is sealed off by a rubber packing at the upper end, and the hydraulic properties (pressure and water flow as a function of time) are then measured below this packer. In the double-packer test the hydraulic properties are measured between two packers at a certain distance between each other.

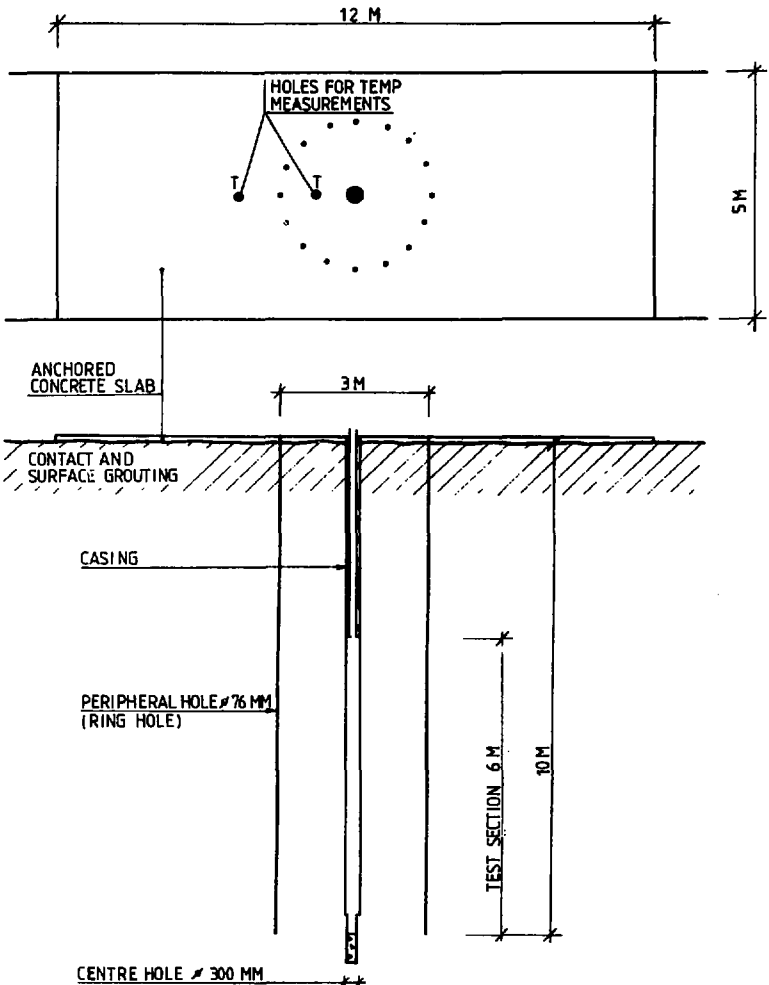


FIG. C-5. Layout of each test place.

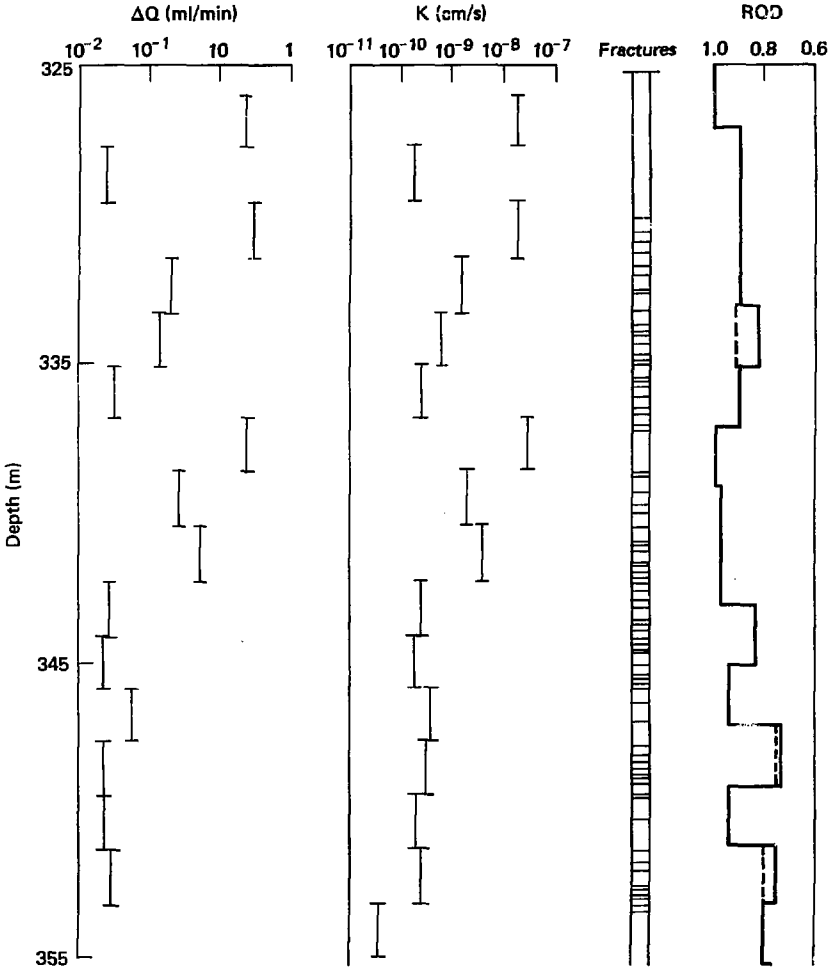


FIG. C-6. Injection test results and fracture data for the interval 325 to 355 m in SBH-1.

TABLE C-1. Hydraulic conductivity, K, in the lower 6-m section of some peripheral boreholes (10^{-9} m/s).

Borehole number	Double packer test	Single packer test
1	3.1	2.0
4	1.3	0.20
8	0.87	0.34
14	2.3	2.6

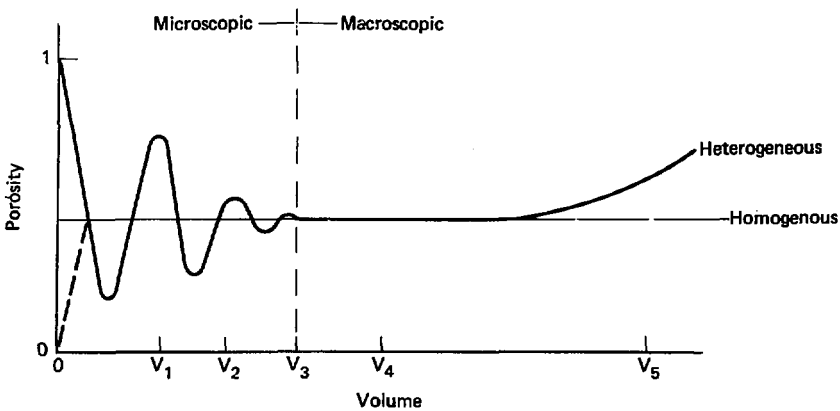


FIG. C-7. Microscopic and macroscopic domains and the representative elementary volume V_3 (from Freeze and Cherry, 1979).

of rock the size of the REV can be treated as porous media for regional analyses of groundwater flow.

In fractured rocks, where the discontinuities themselves may occupy areas on the order of 10^2 m^2, it is reasonable to expect REVs, if they exist, to be on the order of 10^4 or 10^5 m^3. The large-scale permeability experiment at Stripa will permit a measurement of the average permeability of 10^5 to 10^6 m^3 of rock. There is no assurance that this volume will be as large as the REV; however, the experiment, taken along with other

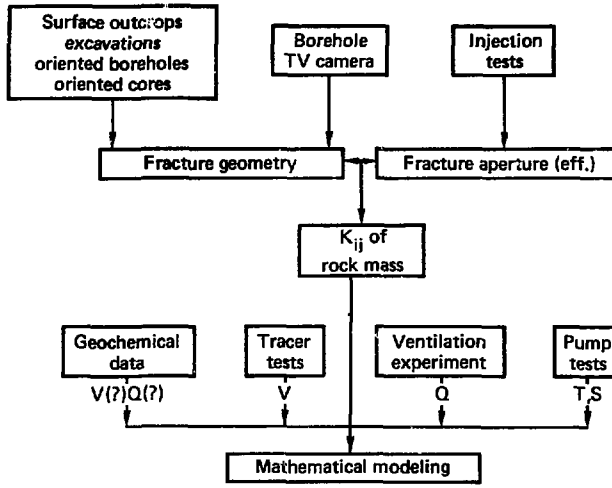


FIG. C-8. Block diagram of steps in calculating directional permeabilities from fracture data.

small-scale tests performed at the same site, should provide an indication of the existence and size of the REV.

The overall hydrological studies at Stripa are summarized in Fig. C-8.

C-5. INTERMEDIATE-SCALE HYDRAULIC CONDUCTIVITY TESTS (SAC-02)

In order to obtain reasonable flows in the experimental setup of Fig. C-5, pressure had to be applied to the peripheral holes. The effect of pressure on the hydraulic conductivity was therefore tested in a separate experiment. Values of K (in 10^{-9} m/s) obtained were 0.88 (100 kPa overpressure) and 0.96 (300 kPa overpressure) at one test hole, and 0.51 (300 kPa overpressure) and 0.55 (900 kPa overpressure) at another test hole. Thus, a threefold pressure increase obviously has little influence on the value of K.

In the main test, the inflow of water into the central hole after pressure injection of water into the peripheral holes was converted to hydraulic conductivity, with the special geometry being taken into account.

The rock was heated by circulating warm water in a closed system. The constant overpressures at the peripheral holes were 100 and 500 kPa. For 10°C, K was calculated at 4×10^{-11} m/s, while for 36°C the value dropped to 2×10^{-11} m/s. These values of K are lower by a factor of about 100 than the values obtained in single-testing the peripheral holes (Section 4).

The porosity of the rock was 1.3×10^{-4} (0.013%) calculated from measurements of flow of a fluorescent dye from the peripheral holes into the central hole.

C-6. DIRECTIONAL PERMEABILITIES

If the fractures are assumed to consist of two-dimensional infinite planes with preferential orientation in the rock, it should, in principle, be possible to calculate the hydraulic conductivity and porosity from borehole measurements of fracture widths, spacings, and orientation. The program to assess the directional permeability addresses this question (Fig. C-8).

Fracture orientations, spacings, and continuity have been obtained by mapping the fractures in surface outcrops and in the walls and the floors of the subsurface excavations. Another source of data is a group of three slanted boreholes drilled from the surface (Fig. C-2) to the 340-m test level. These holes are logged with TV cameras and double-packer tests. Data are obtained every 2 m (every 0.3 m for interesting fractures). Figures C-6 and C-9 present some experimental data. The irregular hydrostatic pressure below the 200-m level shows the effect of long-term drainage from the mined caverns. The RQD (Rock Quality Designation) indicates the degree of fracturing.

It will probably be a long time before final results from these studies can be presented.

C-7. LARGE-SCALE HYDRAULIC CONDUCTIVITY MEASUREMENTS

A 33-m section of a tunnel with a 30-m^2 cross section has been sealed off, so that the volume, humidity, and temperature of air entering and leaving the tunnel can be monitored. Also, 15 holes, 30-40 m long, have been driven

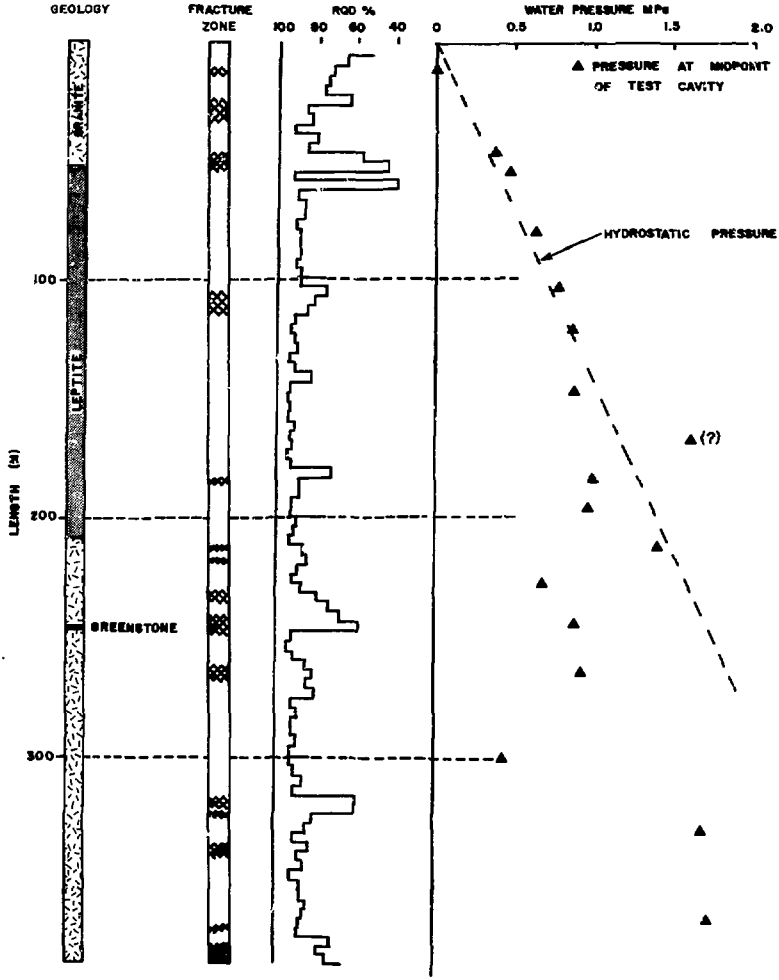


FIG. C-9. Results of fracture and pressure measurements in SBH-1.

into the tunnel walls. See Fig. C-10. By measuring the amount of water vapor being removed by the ventilation system, a direct measure of the volume of water seeping in can be obtained. Furthermore, knowledge of the groundwater pressure gradient near the tunnel walls allows calculation of the overall hydraulic conductivity.

The pressure gradient in the rock walls is being measured in the radial holes. Figure C-11 shows that a pressure of about 1 MPa is reached about 30 m from the wall. These unusually low pressures are a result of the drainage that has gone on for some years in the adjacent mine workings. The figure also illustrates the existence of connections between the fractures, because

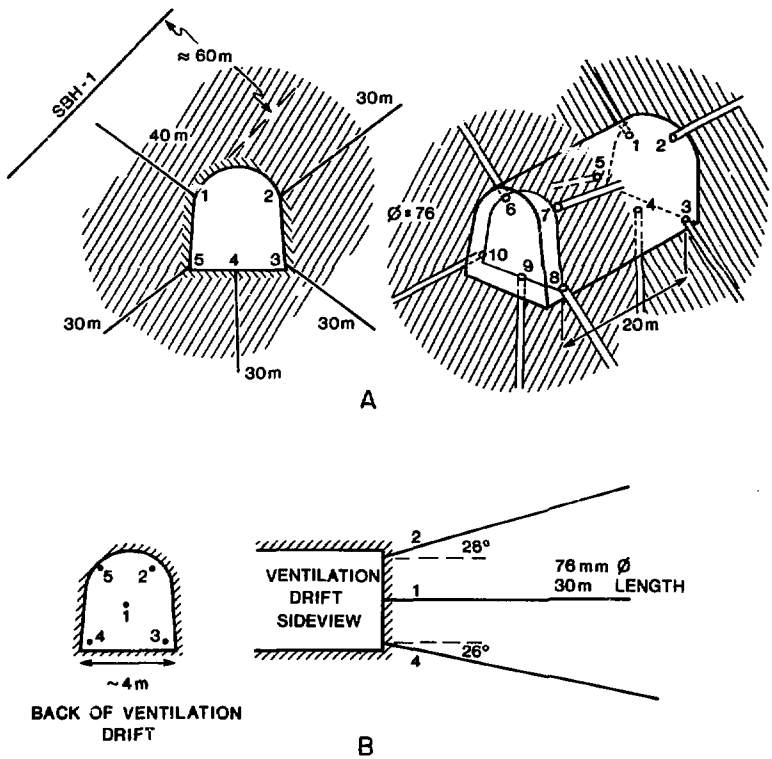


FIG. C-10. Arrangement of subsurface boreholes in the tunnel section used for ventilation experiment at Stripa.

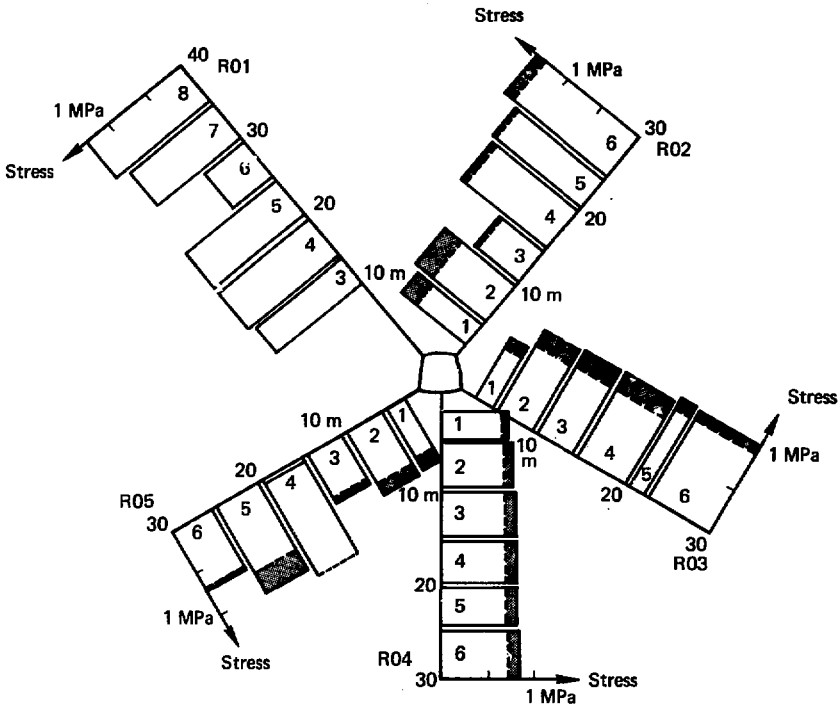


FIG. C-11. Pressure measurements in radial boreholes of ventilation drift at Stripa. Stippled area shows pressure increases eight days after packing off R01. 1 MPa = 145 psi.

the water pressures in all holes were found to increase where one hole (R01) was packed off. After all holes had been packed off, a marked increase in drips and wet spots was observed in the tunnel.

To produce the current seepage rate of 50 ml/min under the observed pressure gradients, the average hydraulic conductivity of the rock is about 10^{-11} m/s. This value refers to a rock mass of 10^5 to 10^6 m^3, and compares favorably with the value presented in the previous paragraph for an "intermediate-sized" rock mass of the order of 10^2 m^3.

C-8. HYDRAULIC PROPERTIES OF FRACTURES (SAC-17)

The hydraulic properties of a single fracture can be studied in the laboratory. The advantage of laboratory study is that the experimental conditions can be carefully controlled, allowing testing of theories for the permeability of fractures. Figure C-12 shows schematically a test sample with a horizontal fracture. The water flow through this fracture as a function of applied load has been measured for test samples of various sizes (diameter 0.1 to 1.0 m) and rock compositions (mainly granite), and with natural or artificially made fractures. Increasing the load (pressure from 0.1 to 30 MPa) tightens the fracture and reduces water flow, and thus hydraulic conductivity. The general conclusion of the tests is that the small samples which are generally used for laboratory tests (about 0.15 m in diameter), yield permeabilities that are too low as compared to permeabilities obtained *in situ*. It is therefore necessary to determine optimum specimen size before reliable results on flow in fractures under stress can be obtained.

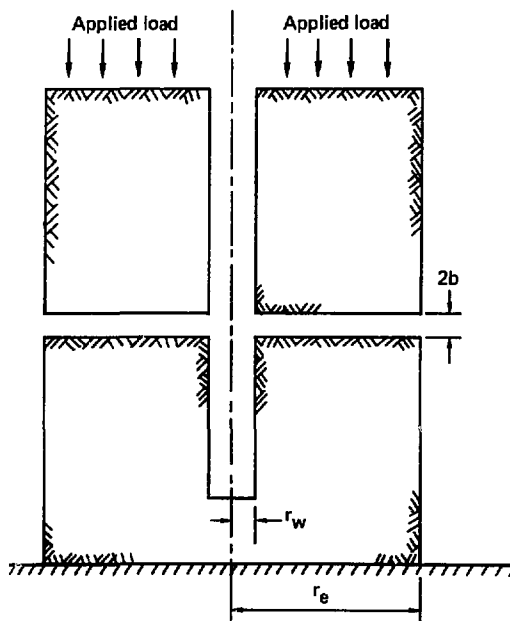


FIG. C-12. Radial flow model for laboratory experiments on a horizontal fracture.

C-9.1 SOURCES AND ANALYSIS

Groundwaters have been extensively analyzed at Stripa in order to provide an independent approach to the problem of overall hydraulic conductivity of the rock. An assessment of fracture hydrology requires information about composition, origin, and age of groundwaters within the rock. The groundwater age is defined as the average transport time of water from the surface, where it is in equilibrium with surface waters, to the measuring point.

Water samples were obtained from 4 private wells and 12 boreholes entering at the surface. (See Figs. C-3 and C-13; some boreholes, e.g., SBH1-3, are rather deep, going down to the 290-m level.) Samples were also obtained from two boreholes entering at the 340-m level and one rather deep borehole entering at the 410-m level and going down to 890 m. Samples were taken and analyzed in the laboratory for

a. conductivity

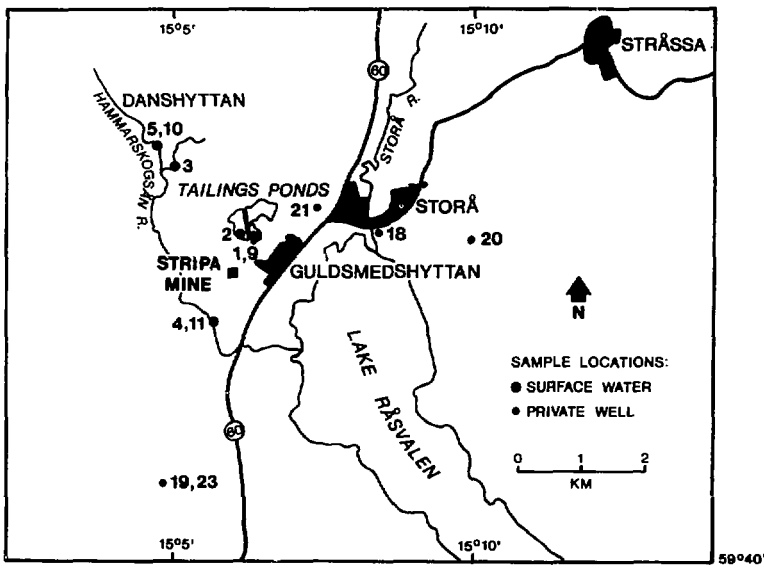


FIG. C-13. Locations of samples (surface water and private wells) in the vicinity of the Stripa mine. The samples were subjected to chemical and isotopic analyses (see Fig. C-14).

b. temperature

c. pH

d. dissolved O_2

e. alkalinity

f. chemical composition (Ca, Mg, Na, K, Cl, SO_4, HCO_3, SiO_2, Fe_{tot}, PO_4, NO_3 and organic C).

For the age determinations, analyses were also made for

g. dissolved noble gases (He, Ne, Ar, Kr, Xe)

h. stable isotope ratios ($^2H/^1H$, $^{18}O/^{16}O$, $^{13}C/^{12}C$)

i. radioactive isotopes (^{234}U, ^{238}U and decay products).

The water sources themselves were monitored at site for properties (a) through (e). Seven laboratories participated in the large analytical task. Great care was taken to avoid contaminations and to obtain high sensitivity. The analytical procedures are briefly described in SAC-12, where the results of about 150 samples are also collected.

The analytical data have provided important information on the geochemical evolution, origin, and age of Stripa waters. However, due to difficulties of interpretation, the results summarized below are still only tentative.

C-9.2 THE CHEMICAL CHARACTERISTICS OF THE GROUNDWATERS

The surface waters sampled in the Stripa area are very dilute, with total dissolved solids (TDS) of less than 35 mg/liter. On a molal basis, the dominant species are $Ca(\simeq HCO_3) > Cl \simeq Na \simeq SO_4 > SiO_2$ ($> HCO_3$) $> K > Fe$. The pH is 6.5 to 6.8, nitrate is about 0.5 ppm, dissolved organic carbon is high (5 ppm), and the waters are unsaturated with respect to calcite.

Shallow groundwaters (from less than 100 m) obtained from private wells contain TDS of 120 to 325 mg/liter. These groundwaters are dominantly calcium carbonate water with

$$HCO_3^- > CA > Na \simeq Mg \simeq SiO_2 > Fe \simeq K.$$

The pH is 6.8 to 7.9, nitrate is very variable (0 to 2.5 ppm), and organic carbon concentrations are much lower than in surface waters--generally about 0.6 to 0.8 ppm carbon. These shallow groundwaters are, with few exceptions, unsaturated with respect to calcite.

Deeper groundwaters from the 330-m level, and from the upper part of the 410-m-level borehole, are sodium chloride-bicarbonate waters with TDS of about 200 to 230 mg/liter, and molal concentrations decreasing in the order

$$Na > Cl > HCO_3 > Ca > SiO_2 \simeq Mg > K \simeq Fe.$$

The pH is 8.8 to 9.05, nitrate is 0 to 1.2 ppm, and organic carbon is low (0.6 ppm carbon). Calcite saturation is slightly exceeded.

The deepest samples from below 700 m (from the borehole at the 410-m level) have the highest TDS (375 to more than 510 mg/liter) and are sodium-calcium chloride groundwaters with

$$Cl > Na > Ca > HCO_3 > SO_4 \simeq S_iO_4 > K > Fe.$$

These waters have high pH (9.5 to 9.8), low nitrate (<0.3 ppm), and low organic carbon content (<0.8 ppm). Calcite saturation is exceeded in all samples.

Geochemical variations in groundwaters from Stripa are illustrated in Fig. C-14, where the change in some geochemical parameters with depth can be seen. Depths of samples are measured from the surface and are only approximate in most cases.

Table C-2 presents a comparison of the concentration ranges in chemical species found in groundwaters from the Stripa mine (excluding private wells), from granites and gneisses in Sweden, and from granite of the Bohemian Massif (SAC-12).

The differences in Table C-2 are important: much higher concentrations of Na, Cl, and HCO_3, much higher pH values, but lower concentrations of Mg, SO_4, and Fe were encountered at Stripa and other Swedish sites than in the Bohemian Massif. The differences might be explained on the basis of

- Different atmospheric input; different geochemical conditions affecting groundwater in the unsaturated zone or in other rock-types before the groundwater enters granite.
- Mineralogical variation in granites.
- Control of geochemistry by fracture mineralogy which would be different for different granites.

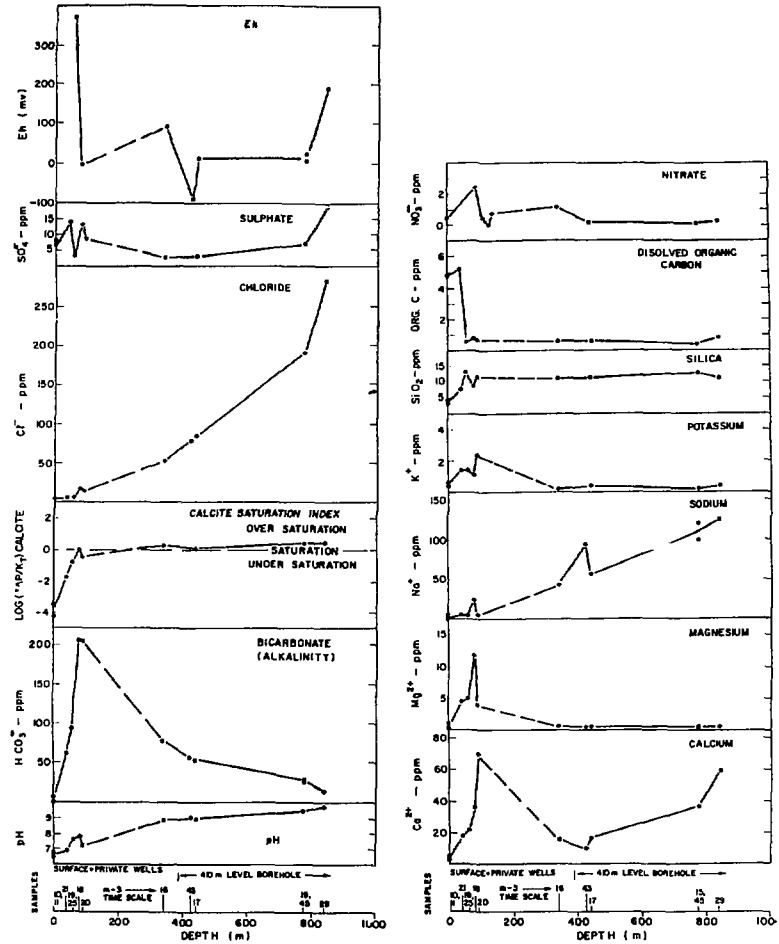


FIG. C-14. Geochemical variation of groundwaters with depth. See Fig. C-13 for location of sample numbers.

TABLE C-2. A comparison of the concentration ranges in chemical species in groundwaters from Stripa, Swedish granites, and gneisses from the Bohemian Massif (mg/liter).

	Source of groundwater		
	Stripa	Sweden	Bohemian Massif
Species:			
Ca	10 to 59	~10 to >40	8.0 to 62.5
Mg	0.5	~5 to 15	trace to 20.7
Na	43 to 125	~10 to >40	2.5 to 10.5
K	0.2 to 5.4	~3 to 5	trace to 4.5
Cl	52 to 283	~3 to >5000[a]	1.8 to 7.8
SO_4	2.7 to 19	~3 to 50	18.5 to 235
HCO_3	15.4 to 78.7	~60 to 300	3.7 to 18.3
SiO_2	11.0 to 12.8	~10 to 60[b]	5.5 to 18.0
Fe	0.02 to 0.24		trace to 1.9
pH:	8.85 to 9.75	~6 to 9	4.4 to 5.96

[a]Fossil seawater contamination assumed.
[b]No actual analyses given but general quote of literature data.

- Different degrees of water-rock interaction due to different groundwater temperatures, different durations of water-rock contact, different physical fracture characteristics, etc.
- Presence of brines or seawater in the Stripa rocks.

C-9.3 GEOCHEMICAL PROCESSES AND THE GEOCHEMICAL EVOLUTION OF GROUNDWATER

Plots of groundwater composition or mineral stability diagrams strongly indicate that all groundwaters sampled have followed similar evolutionary trends. Therefore, the Stripa groundwaters are considered to evolve from original recharged waters to the presently observed groundwaters. Water-rock interaction is considered to be the major process causing geochemical change as waters flow in the subsurface. In view of the assumed increasing age of groundwaters with depth (see next paragraphs) deeper samples are thought to have undergone more geochemical evolution than shallow samples. The

assumption is also made that modern shallow samples are similar to deeper samples as they existed at an earlier stage of evolution, or, to put it differently, that deeper samples used to be like modern shallow samples.

The increased concentration of Ca^{2+} and Na^{+}, and decreased concentration of HCO_3^- and Mg^{2+} at the deepest level (Fig. C-14) is explained as being due to mineralogic transformations and the mixing of normal groundwater with brine, perhaps residual seawater. The increasing pH is believed to be due to incongruent dissolution of silicates.

C-9.4 DISSOLVED GASES

The concentrations of dissolved gases give different kinds of information. Nitrogen is added to the original water from air, but may also be formed through microbiological denitrification (reduction of dissolved nitrate). Oxygen originates from the air but is consumed by oxidation processes, possibly also microbiologically (producing CO_2). CH_4 is produced through fermentation processes under anaerobic conditions. In principle, it should be possible to calculate the redox potential of the groundwater from the O_2 and CH_4 concentrations; such calculations have failed, however.

The He and Ar are of radiogenic origin and can be used for groundwater dating (see Section 9.7).

This leaves Ne, Kr, and Xe as "undisturbed" gases. The solubility of these gases depends on the water temperature. For example, Xe dissolves to 18 ppb (volume gas/volume water) at 1°C, and to 7 ppb at 30°C. The concentrations of these gases in Stripa groundwaters indicate a recharge temperature of 5-9°C at the 330-m level, and 0-5°C for the bottom of the deep hole from the 410-m level. This result supports the assumption that the recharge occurred during the glacial period.

C-9.5 GROUNDWATER REDOX POTENTIAL

Eh* was measured with a combination glass-calomel and platinum-disc electrode and values obtained were calibrated against a "Zobell solution" of

*Eh is the electrical potential of groundwater in relation to a hydrogen electrode ("water redox potential").

known Eh. Measurements were carried out in a flow cell at the sample site. Readings were taken when stable values were obtained; these values are believed to be correct to ±5 mV. The following values were recorded: +169 mV around the 800-m level, +122 mV below 460-m, and +92 mV at the 360-m level. Attempts to use these emf values to calculate pO_2 and pCH_4 failed completely. The failure may be attributable to incorrectness, for unknown reasons, of the measured values of Eh. It is possible, for example, that samples were contaminated with air.

The previous paragraph is paraphrased from Ref. SAC-12. There are reasons to question these results. Measurements by Prof. I. Grenthe (Royal Technical University, Stockholm) for very carefully collected Stripa samples of pH 8.2 yield Eh values from -31 mV to -210 mV. These low redox potentials can be ascribed to the presence of Fe(II)-minerals in the rock and they agree well with theoretical calculations as well as foreign observations (B3).

C-9.6 STABLE ISOTOPE RATIOS

When molecules take part in physicochemical reactions (evaporation, precipitation, etc.), the relative concentrations of their isotopes may change slightly. Water contains the hydrogen isotopes ^1H (almost 100%) and ^2H (0.015%), and the oxygen isotopes ^{16}O (99.76%), ^{17}O (0.04%), and ^{18}O (0.204%), leading to the existence of molecules like $^1H^1H^{16}O$ (~10^6 ppm), $^1H^1H^{18}O$ (~2000 ppm), $^1H^2H^{16}O$ (~320 ppm), etc. If the ratio of the hydrogen isotopes ($^2H/^1H$) is referred to as δ_x and δ_s in a given water sample and a standard, respectively, then the deviation of the sample from the standard is expressed as

$$\delta(^2H) = 1000(\delta_x - \delta_s)/\delta_s \ (^o/oo).$$

$\delta(^{18}O)$ is defined in a similar way. The common standard is SMOW (Standard Mean Oceanic Water).

As a rule of thumb, the following values of deviation have been found (Dansgaard, 1964):

$$\delta(^2H) = 5.6t_a - 100 \ (^o/oo)$$
$$\delta(^{18}O) = 0.7t_a - 13.6 \ (^o/oo) \ ,$$

for water precipitated by rain or snow at an annual average temperature t_a (^oC). Thus from measurement of isotopic ratios, water precipitation temperatures, t_a, may be estimated. Because water of the same origin must have the same t_a, there is a common relation:

$$\delta(^2H) = 8\delta(^{18}O) + 10(^o/oo) \quad .$$

A plot of $\delta(^2H)$ against $\delta(^{18}O)$ yields a straight-line referred to as the global Meterological (or Meteoric) Water Line (MWL). In a given region (like central Sweden), the precipitation follows such a line—the slope and intercepts vary slightly between different regions. Deviations from this line indicate secondary processes, such as rock-water interactions at elevated temperature.

In most hydrogeologic situations the ^{18}O and ^2H contents of a groundwater reflect the recharge environment. For example, groundwaters formed immediately following the retreat of the last glaciation are lower in heavy isotopes than groundwater formed there today.

Figure C-15 presents the data from Stripa. As a general rule, lower concentrations of heavy isotopes signify lower average annual temperatures at the recharge area. Therefore, the deep "saline" groundwaters, which have the lowest ^{18}O and ^2H contents, must have recharged at lower average annual temperatures than the shallower groundwaters. This is confirmed by the rare gas analyses. One must therefore conclude that the deep groundwaters have an origin different from that of the shallower ones.

This conclusion is further substantiated by comparing ^{18}O with the chloride concentrations as shown in Fig. C-16. Here, it is apparent that the deep groundwaters, especially those at the bottom of the 410-m hole, are distinctly different from the shallow groundwaters. In other words, the different fracture systems in the granite at Stripa carry different types of water because the systems are isolated from each other.

One could argue that the lower ^{18}O isotope content in the deeper groundwaters is an indication of subglacial recharge. This conclusion is not supported by the ^{13}C analyses; all waters from the mine levels have $\delta^{13}C$ levels close to or below $-15^o/oo$. These levels indicate that biogenic carbon is present in the dissolved organic carbon. The presence of biogenic carbon would signify that these groundwaters infiltrated through soil horizons, that is, were generated during an interglacial period.

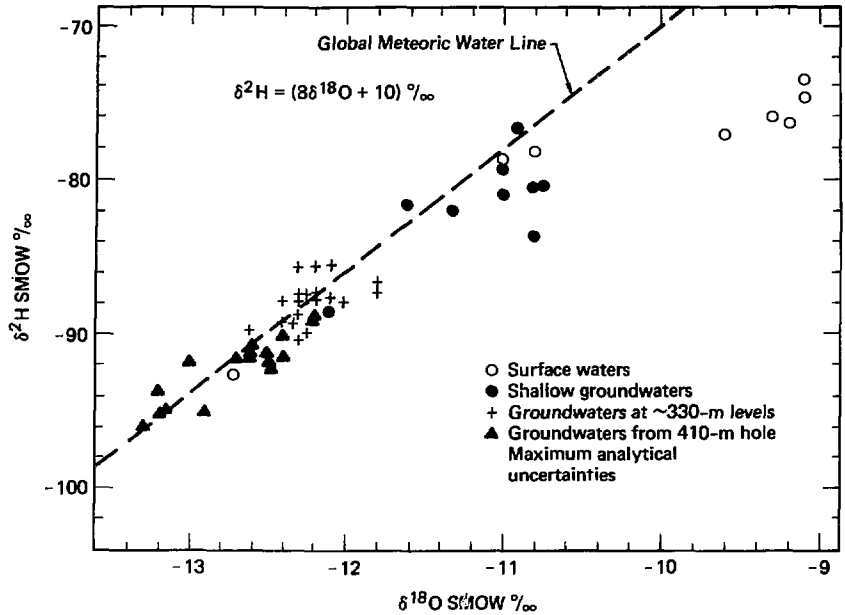


FIG. C-15. Comparison of $\delta^{18}O$ and δ^2H values for Stripa groundwaters. The analyses are reported as δ‰ values with reference to SMOW. A $\delta^{18}O$ of -10‰ signifies that the sample has 10‰ (per mil) less ^{18}O than the reference standard, which closely reflects average seawater.

C-9.7 RADIOACTIVE DATING

The most difficult and inconclusive part of the geochemical investigation was the attempt to date the groundwaters from the different mine levels. Tritium level approaching 100 TU* were found in all shallow groundwater (<100 m) and, interestingly enough, even in the mine waters of the old workings. However, tritium was not encountered (<0.5 TU) in any of the deep groundwaters from the granite, despite the above-mentioned drainage that has decreased water pressures below hydrostatic (see Fig. C-9). This lack of

*1 TU is equal to one tritium atom (^3H) per 10^{18} atoms of ^1H.

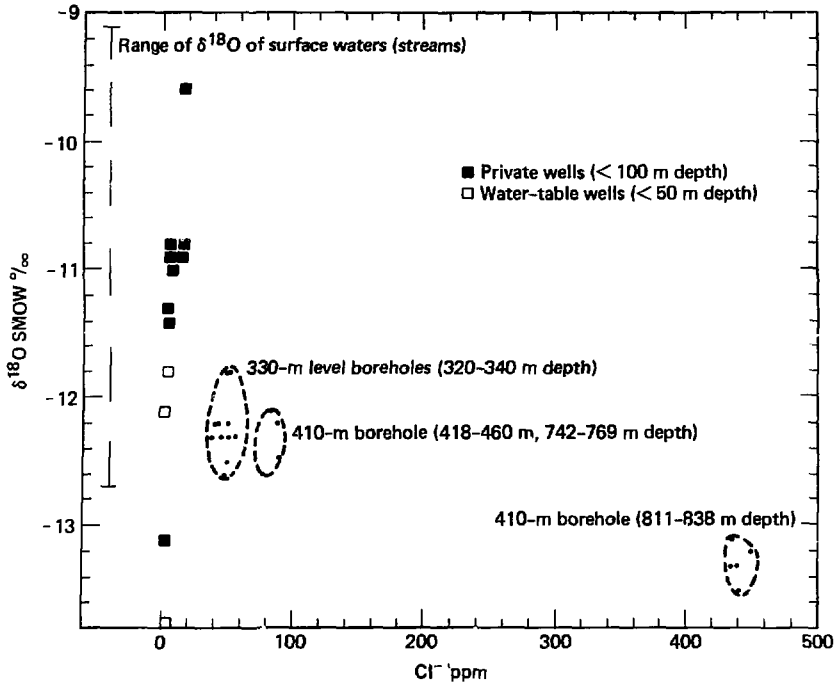


FIG. C-16. Comparison of chloride with $\delta^{18}O$ values shows that the different fracture systems in the Stripa granite carry different types of water.

tritium indicates that the deep waters do not contain any surface water component originating more than 30-40 years ago, i.e., when testing of hydrogen weapons began (producing large amounts of tritium).

Major problems were encountered in attempting ^{14}C age dating because of the very low content of dissolved inorganic carbon. It was necessary to treat 2000 to 3000 liters of water to obtain sufficient carbon for analysis. The results indicate that waters at the 330-m level, and probably also from the 410-m borehole, exceed 20 000 years in age. Contamination of water samples from the 410-m borehole has prevented a more precise result.

Three different approaches to age dating based on the uranium decay series were also investigated: (a) uranium activity ratios, (b) helium contents, and (c) radium-radon relationships. The $^{234}U/^{238}U$ activity

ratio in the groundwaters decreases from 10.4 at the 330-m level to about 6 at the top of the 410-m borehole and to almost 4 in the high "saline" waters at the bottom of this hole. This decay in activity ratio can be used to date waters (SAC-12). Although the method is still under development and subject to some uncertainties, ages exceeding 100 000 years are obtained for the groundwaters from the 410-m borehole.

Somewhat lower ages were determined from the helium concentrations. The atmospheric concentration (volume helium/volume water) at 5°C is 4.9×10^{-8}, whereas the concentrations in the groundwaters at Stripa are five orders of magnitude higher, ranging from 0.3×10^{-3} at the 330-m level to 1.4×10^{-3} in the 410-m borehole. From these data one can compute ages that range from tens of thousands to hundreds of thousands of years.

If ^{222}Ra accumulates as a recoil product and is in equilibrium with ^{226}Ra, then ^{234}U and ^{222}Ra activities will eventually reach equilibrium. Ages calculated by this model are 10 000 to 35 000 years for the different mine waters. An extension of this approach considers the ^{226}Ra concentrations in the rock minerals and the ^{226}Ra in the water. If equilibrium between the two exists, that is, if the recoil rate from the rock equals decay in the solution, then the waters must be at least 8000 years old (five half-lives of ^{226}Ra). There is evidence that this is the case, again supporting the earlier results that the waters presently found in the deep granite rock mass at Stripa are indeed many thousands of years old.

The preliminary results of the radioactive dating are summarized in Fig. C-17. It is apparent that a careful investigation of the geochemistry and isotope hydrology provides an independent and powerful approach to the critical problem of elucidating the degree of isolation that has developed in the Stripa groundwater systems.

C-10. *SOME GENERAL CONCLUSIONS*

Studies at Stripa have demonstrated that for large granite rock bodies (10^2-10^6 m^3) consistent and low hydraulic conductivities are found (of the order of 10^{-11} m/s), even when the rock body is highly fractured with several fractures per metre, and double-packer tests over the fractures give conductivities varying between $<10^{-11}$ and 10^{-8} m/s. Data from these large-body studies compare favorably with measurements on granite tunnels ($>>10^6$ m^3)

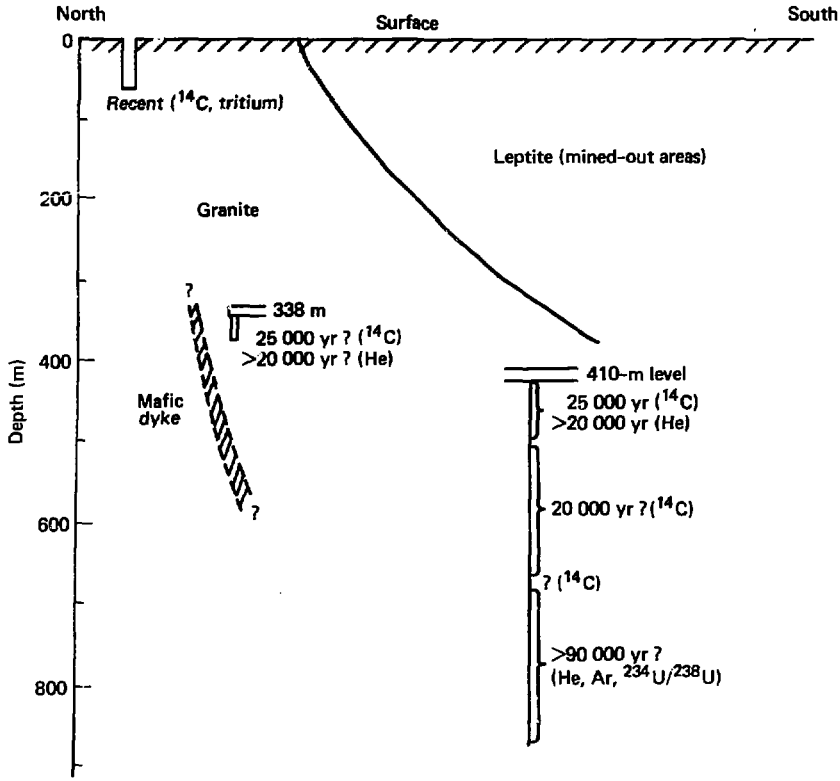


FIG. C-17. Preliminary results of isotope age dating of groundwater samples. The method of dating is given in parentheses after the estimate. Question marks indicate uncertainty.

excavated in Sweden for military purposes, where in many cases overall conductivities below 10^{-11} m/s have been observed.

Introducing measured hydrological data (K, i, ε) into Darcy's law yields volumes and velocities of water flow. An alternative technique for determination of flow velocities relies on measurement of the time it has taken for the water to move from the surface down into the rock (the "water age"). Water age can be measured using "radioactive clocks." A number of such clocks

(U/He, K/Ar, ^{14}C, etc.) have been tested at Stripa. Although *all* indicated that the water is old, thousands of years or more, *none* is conclusive. Obviously, this is an extremely difficult technique. However, its potential value as an independent technique for establishing water flow rates is so high that it is worthwhile trying to improve it. It should be noted, as an additional complication, that water in different fractures may be of different age; age determinations usually require large water samples and give therefore only average ages for the sampling volume.

Unlike the chemical analyses, stable isotope ($^{2}H/^{1}H$, $^{18}O/^{16}O$) and gas measurements do not yield water ages directly, but relate water at different levels to each other. Therefore, these methods can serve only to support hypotheses about age and flow of water.

Although the Stripa mine project is only a geophysical and hydrological experiment, it is of some theoretical interest to consider the mine's suitability for final storage of highly radioactive waste.

There seems to be good geochemical evidence for the existence of two groundwater regions. In the lower region (below 600 m, say) the water's composition indicates that it contains fossil seawater mixed with glacial-melt water. About 10 000 years ago this area was below the present sea level. It is reasonable to assume that the ice melting and land uplift at that time caused water to penetrate down to the lower region at Stripa. The gas analysis and stable isotope analysis indicate that this water was cold. The radioactive dating supports the inference that the low-level water is very old, perhaps more than 10 000 years. The ^{13}C data do not necessarily contradict this inference.

The stagnant water would guarantee that radioactive waste stored at this level would not be transported to surface repositories (including shallow wells) before the end of a next ice age (assumed to begin 10-15 thousand years from now). At that future time the waste hazard would come from a few actinide isotopes (^{237}Np, ^{239}Pu, ^{243}Am), which could be removed from the waste before storage, thus eliminating the hazard almost completely. However, the existence of corrosive saline deep water might require the selection of container materials other than those now suggested. As a consequence of these findings, the possibility and consequences of establishing an "extra-deep" depository wetted by mainly stagnant saline groundwater should be considered as an alternative to present plans for a depository at a depth of 500 m.

C-11. REFERENCES

A. SAC reports

These reports are the results of the Swedish-American cooperative research program. The program has been sponsored by the Swedish Nuclear Power Utilities through the Swedish Nuclear Fuel Supply Company (SKBF), and the U.S. Department of Energy (DOE) through the Lawrence Berkeley Laboratory (LBL).

The principal investigators are L. B. Nilsson and O. Degerman for SKBF, and N. G. W. Cook, P. A. Witherspoon, and J. E. Gale for LBL. Other participants appear as authors of the individual reports.

Technical reports in this series are listed below. Not all of the listed reports are specifically referred to in the text.

SAC-01. P. A. Witherspoon and O. Degerman, *Swedish-American Cooperative Program on Radioactive Waste Storage in Mined Caverns*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7049, 1978.

SAC-02. Lars Lundstrom and Haken Stille, *Large Scale Permeability Test of the Granite in the Stripa Mine and Thermal Conductivity Test*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7052, 1978.

SAC-03. Graham Swan, *The Mechanical Properties of the Stripa Granite*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7074, 1978.

SAC-04. H. Carlsson, *Stress Measurements in the Stripa Granite*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7078, 1978.

SAC-05. Pavel J. Kurfurst, T. Hugo-Persson, and G. Rudolph, *Borehole Drilling and Related Activities at the Stripa Mine*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7080, 1978.

SAC-06. Hans Carlsson, *A Pilot Heater Test in the Stripa Granite*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7086, 1978.

SAC-07. Dennis B. Jamison and Neville G. W. Cook, *An Analysis of Measured Values for the State of Stress in the Earth's Crust*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7071, 1978.

SAC-08. B. Andersson and P. A. Halen, *Mining Methods Used in the Underground Tunnels and Test Rooms at Stripa*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7081, 1978.

SAC-09. Tin Chan, Neville G. W. Cook, and Chin-Fu Tsang, *Theoretical Temperature Fields for the Stripa Heater Project*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7082, 1978.

SAC-10. Neville G. W. Cook, *Mechanical and Thermal Design Considerations for Radioactive Waste Repositories in Hard Rock. Part I: An Appraisal of Hard Rock for Potential Underground Repositories of Radioactive Wastes;* Neville G. W. Cook and Paul A. Witherspoon, *Part II: In Situ Heating Experiments in Hard Rock: Their Objectives and Design*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7073, 1978.

SAC-11. Neville G. W. Cook and Michael Hood, *Full-Scale and Time-Scale Heating Experiments at Stripa: Preliminary Results*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7072, 1978.

SAC-12. P. Fritz, J. F. Barker, and J. E. Gale, *Geochemistry and Isotope Hydrology of Groundwaters in the Stripa Granite: Results and Preliminary Interpretation*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8285, 1979.

SAC-13. R. H. Burleigh, E. P. Binnall, A. O. DuBois, D. U. Norgren, and A. R. Oritz, *Electrical Heaters for Thermo-mechanical Tests at the Stripa Mine*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7063, 1979.

SAC-14. Maurice B. McEvoy, *Data Acquisition, Handling, and Display for the Heater Experiments at Stripa*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7062, 1979.

SAC-15. J. E. Gale and P. A. Witherspoon, *An Approach to the Fracture Hydrology at Stripa: Preliminary Results*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7079, 1979.

SAC-16. P. Nelson, B. Paulsson, R. Rachiele, L. Andersson, T. Schrauf, W. Hustrulid, O. Duran, and K. A. Magnusson, *Preliminary Report on Geophysical and Mechanical Borehole Measurements at Stripa*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8280, 1979.

SAC-17. P. A. Witherspoon, C. H. Amick, J. E. Gale, and K. Iwai, *Observations of a Potential Size-Effect in Experimental Determination of the Hydraulic Properties of Fractures*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8571, 1979.

SAC-18. P. A. Witherspoon, P. Nelson, T. Doe, R. Thorpe, B. Paulsson, J. Gale, and C. Forster, *Rock Mass Characterization for Storage of*

Nuclear Waste in Granite, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8570, 1979.

SAC-19. K. H. Waters, S. P. Palmer, and W. E. Farrell, *Fracture Detection in Crystalline Rock Using Shear Waves*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7051, 1978.

SAC-20. R. Thorpe, *Characterization of Discontinuities in the Stripa Granite--Time Scale Heater Experiment*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7083, 1979.

SAC-21. A. Olkiewicz, J. E. Gale, R. Thorpe, and B. Paulsson, *Geology and Fracture System at Stripa*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8907, 1979.

SAC-22. T. Chan and N. G. W. Cook, *Calculated Thermally Induced Displacements and Stresses for Heater Experiments at Stripa*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-7061, 1979.

SAC-23. P. A. Witherspoon, J. Wang, K. Iwai, and J. E. Gale, *Validity of Cubic Law for Fluid Flow in a Deformable Rock Fracture*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-9557, 1979.

SAC-24. J. Jeffry, T. Chan, N. G. W. Cook and P. A. Witherspoon, *Determination of In-Situ Thermal Properties of Stripa Granite from Temperature Measurements in the Full-Scale Heater Experiments: Methods and Primary Results*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8424, 1979

SAC-25. T. Schrauf, H. Pratt, E. Simonson, W. Hustrulid, P. Nelson, A. DuBois, E. Binnall, and R. Haught, *Instrumentation Evaluation, Calibration, and Installation for Heater Tests Simulating Nuclear Waste in Crystalline Rock, Sweden*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8313, 1979.

SAC-26. M. Hood, *Part I: Some Results from a Field Investigation of Thermo-Mechanical Loading of a Rock Mass When Heater Canisters are Emplaced in the Rock;* M. Hood, H. Carlsson, and P. H. Nelson, *Part II: The Application of Field Data from Heater Experiments Conducted at Stripa, Sweden for Respository Design*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-9392, 1979.

SAC-27. P. A. Witherspoon, N. G. W. Cook, and J. E. Gale, *Progress with Field Investigations at Stripa*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-10559, 1980.

SAC-28. C. B. Forster and J. E. Gale, *A Laboratory Assessment of the Use of Borehole Pressure Transients to Measure the Permeability of Fractured Rock Masses*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-8674, 1980.

SAC-29. T. Chan, E. Binnall, P. Nelson, O. Wan, C. Weaver, K. Ang, J. Braley, and M. McEvoy, *Thermal and Thermomechanical Data for In Situ Heater Experiments at Stripa, Sweden*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-11477, 1980.

SAC-30. P. H. Nelson, R. Rachiele, and A. Smith, *The Effect of Radon Transport in Groundwater Upon Gamma Ray Borehole Logs*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-11180, 1980.

SAC-31. R. Thorpe, D. J. Watkins, W. E. Ralph, R. Hsu, and S. Flexser, *Strength and Permeability Tests on Ultra-Large Stripa Granite Core*, Lawrence Berkeley Laboratory, Berkeley, CA, LBL-11203, 1980.

B. Other references

1. P. A. Witherspoon, N. G. W. Cook, and J. E. Gale, Waste Management Conf., Tucson, Arizona, 1980.
2. R. A. Freeze and J. A. Cherry, *Groundwater* (Prentice-Hall, 1979).
3. KBS-II Report, Volume II, p. 116.
4. W. Dansgaard, Stable isotopes in precipitation. *Tellus*, 16, 436 (1964).

APPENDIX D

EQUATIONS FOR FRACTURE CONDUCTIVITY (SAC-17)

Equations describing fluid flow between two parallel plates have been derived by Polubarinova-Kochina (1962), Snow (1965), Louis (1969), Noorishad, Witherspoon, and Brekke (1971), Bear (1972), Gale (1975) and several other workers in the field of fracture flow. These authors have shown that the intrinsic permeability of a fracture with an aperture of 2b is given by

$$k_f = \frac{(2b)^2}{12} \quad , \tag{D1}$$

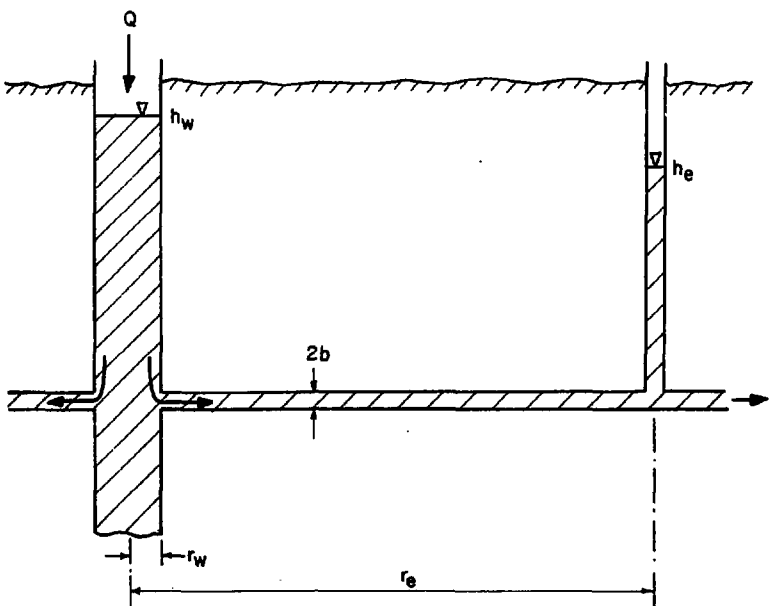


FIG. D-1. Radial flow from a well into a horizontal fracture.

and this may be converted to the hydraulic conductivity of a fracture by

$$K_f = \frac{k_f \rho g}{\mu} = \frac{(2b)^2 \rho g}{12\mu} . \qquad (D2)$$

If one has steady-state radial flow from a borehole into a horizontal fracture (as shown in Fig. D-1) and the hydraulic head drops from h_w at the wellbore radius r_w to h_e at the outer radius r_e, then the injection rate, Q, may be determined from Darcy's law and is given by

$$Q = \frac{2\pi(2b)^3 \rho g (h_w - h_e)}{12\mu \ln r_e/r_w} . \qquad (D3)$$

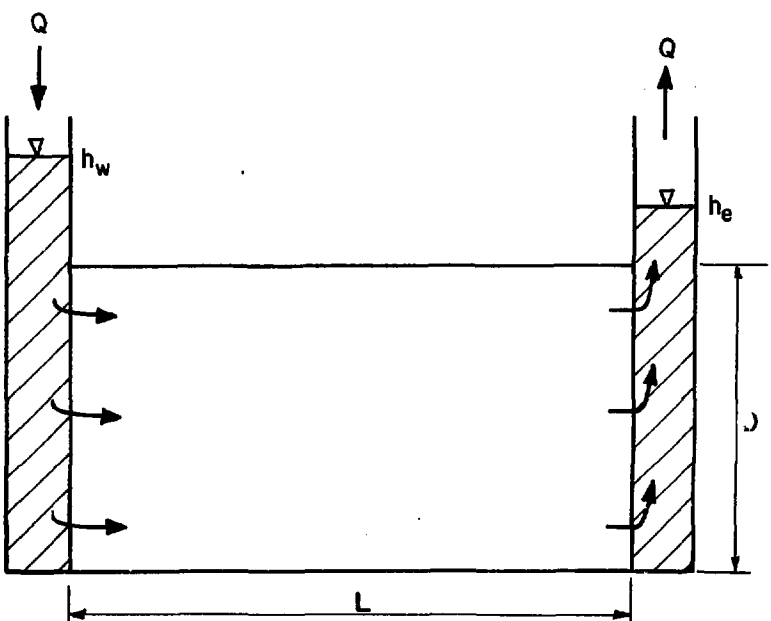


FIG. D-2. Linear flow in a vertical fracture.

In laboratory investigations where the aperture is unknown, Eq. (D3) can be used to solve for 2b directly, since all other parameters are measurable. The hydraulic conductivity can then be determined using Eq. (D2).

If one has steady-state linear flow in a vertical fracture of extent D (as shown in Fig. D-2) and the hydraulic head drops from h_w to h_e over length L, then the flow rate, Q, may be determined from

$$Q = \frac{(2b)^3 \rho g D(h_w - h_e)}{12\mu L} \tag{D4}$$

In experimental investigations, Eq. (D4) can be used to solve for 2b, and Eq. (D2) to solve for k_f.

REFERENCES

1. J. Bear, *Dynamics of Fluids in Porous Media.* (Amer. Elsevier Publishing Company, New York, 1972).
2. J. E. Gale, *A Numerical Field and Laboratory Study of Flow in Rocks with Deformable Fractures.* Ph.D. Thesis, University of California, Berkeley, CA (1975).
3. C. Louis, *A Study of Groundwater Flow in Jointed Rock and its Influence on the Stability of Rock Masses.* Imperial College Rock Mechanics Research Report No. 10, Imperial College, London (1969).
4. J. Noorishad, P. A. Witherspoon, and T. L., Brekke, *A Method for Coupled Stress and Flow Analysis of Fractured Rock Masses.* Geotechnical Engineering Report No. 71-6, University of California, Berkeley, CA (1971).
5. P. Ya. Polubarinova-Kochina, *Theory of Groundwater Movement.* Translated by J. M. R. DeWiest. (Princeton University Press, Princeton, NJ, 1962).
6. D. T. Snow, *A Parallel Plate Model of Fractured Permeable Media.* Unpublished Ph.D. Dissertation, University of California, Berkeley, CA.

APPENDIX E

THE LAW ON NUCLEAR WASTE, JULY 1, 1981 (TRANSLATION).

An Act Concerning the Financing of Future Expenditure for Spent Nuclear Fuel, Etc.

Issued as of 18 June, 1981

It is hereby promulgated as follows.

Section 1

The holder of a license pursuant to the Nuclear Energy Act (1956:306) to own or operate a nuclear reactor for power generation (the reactor owner) shall be responsible for ensuring that

1. nuclear fuel irradiated in the reactor and radioactive waste deriving therefrom is handled and disposed of in a safe manner,

2. the reactor facility will be shut down and dismantled in a safe manner, and

3. such research and development activities are conducted and other measures taken as are required to ensure compliance with the provisions under 1 and 2 above.

Section 2

In addition to the costs resulting from its obligations pursuant to Section 1, the reactor owner shall also be liable in respect of costs incurred by the State for

1. research and development activities supplementary to those referred to in Section 1, item 3,

2. review of the matters referred to in the second paragraph of Section 5 and in Section 10, and

3. surveillance and inspection of final repositories.

Section 3

In consultation with other reactor owners, the reactor owner shall establish a program for such research and development activities and other measures as are referred to in Section 1. The program shall contain an outline of all measures which may become necessary and shall specify more detail the measures intended to be implemented within a period of at least the next five years. The program shall be annually updated.

The program shall be submitted annually to the Government or to the authority designated by it.

Section 4

The authority designated by the Government shall establish a program of the activities referred to in Section 2, items 1 and 3.

Section 5

In order to ensure the availability of funds to cover the costs resulting from the reactor owner's obligations pursuant to Section 1 and the costs referred to in Section 2, an annual fee shall be payable by the reactor owner to the State for such time as the reactor is in operation. As far as the costs for the handling and final disposal of spent nuclear fuel are concerned, the obligation to pay the annual fee shall relate to such costs attributable to the fuel after its removal from the reactor facility. The fee shall be proportionate to the energy delivered by the reactor facility.

The amount of the fee shall be established annually in respect of the succeeding calendar year by the Government or the authority designated by it.

The annual fee shall be established such that the aggregate amount of fees paid during the operating life of the reactor shall cover the costs referred to in the first paragraph of this Section 5.

Section 6

The annual fee shall be established on the basis of the programs referred to in Sections 3 and 4 and the estimated expenditure in respect of the

measures referred to in Sections 1 and 2. With regard to measures referred to in Section 1, consideration shall be given to measures previously adopted for the disposal of spent nuclear fuel and radioactive waste deriving therefrom, to the characteristics of the reactor, and to other circumstances which may influence the calculation of the fee.

If the basis for the calculation of the fee is lacking or inadequate, the fee shall be established at a reasonable amount.

Section 7

The fee shall be paid to the authority designated by the Government. Collected fees shall be deposited by it in an interest-bearing account with the National Bank of Sweden Accrued interest shall be capitalized.

Section 8

Loans may be granted to a reactor owner out of the fees paid in by it against the provision of collateral.

Such loans shall carry interest at a rate no less than that given by the National Bank of Sweden on the collected fees deposited with it. Interest paid shall be deposited in the account referred to in Section 7 and shall be capitalized.

Loans shall otherwise be subject to such terms and conditions as are required to ensure the availability of funds for the purposes referred to in Section 9.

Section 9

Fees paid by a reactor owner may be used in reimbursement of the costs incurred by the reactor owner in respect of measures referred to in Section 1 subject to the limitation resulting from the provision in the second sentence of the first paragraph of Section 5 and further to defray the costs incurred by the State in respect of measures referred to in Sections 1 and 2 which are attributable to the reactor or to fuel utilized in the reactor.

Section 10

Matters relative to supplementary research and development activities, surveillance and inspection of final repositories, loans to reactor owners out of remitted fees and the terms and conditions of such loans, as well as the application of collected fees shall be reviewed by the Government or the authority designated by it.

Section 11

Upon request, the reactor owner shall provide such information and such documents as are necessary for the Government or the authority designated by it to establish the fee in accordance with the second paragraph of Section 8 and to review the matters referred to in Section 10. To the extent required for the aforementioned purposes, the reactor owner upon request shall further grant access to facilities or sites where spent nuclear fuel or radioactive waste deriving therefrom is stored or handled.

Section 12

Anyone who, intentionally or through gross negligence, provides incorrect information or otherwise disregards his obligations pursuant to Section 11 shall be liable to a fine, unless the deed is punishable under the Criminal Code.

This Act shall come into effect on 1 July, 1981, provided that the provisions of the first paragraph of Section 5 shall apply as from calendar year 1982.

DPH/ca

U.S. Government Printing Office: 1982/6-587-002/4066